

ENCLOSURE 1

NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
REQUEST TO AMEND INSERVICE INSPECTION
TECHNICAL SPECIFICATIONS

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3.4 STANDBY LIQUID CONTROL SYSTEMApplicability

The Limiting Conditions for Operation apply to the operating status of the Standby Liquid Control System.

Objective

The objective of the Limiting Conditions for Operation is to assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

SpecificationsA. Normal System Availability

During periods when fuel is in the reactor and prior to startup from the Cold Shutdown Condition the standby liquid control system shall be operable except:

1. When performing control rod drive maintenance, at which time Specification 3.10.E shall be met,

or,
2. When operating with an inoperable component, at which time Specification 3.4.B shall be met,

or,
3. When the reactor is in the Cold Shutdown Condition and all control rods capable of normal insertion are inserted and the requirements of Specification 3.3.A are met.

4.4 STANDBY LIQUID CONTROL SYSTEMApplicability

The Surveillance Requirements apply to the periodic test and examination of the Standby Liquid Control System.

Objective

The objective of the Surveillance Requirements is to verify the operability of the Standby Liquid Control System.

SpecificationsA. Normal Operational Tests

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. Monthly
Verify the continuity of the explosive charge in each loop.
2. As required by Specification 4.6.K
Each pump loop shall be locally started and functionally tested by recirculating demineralized water to the test tank.
3. Each Operating Cycle
At least once during each operating cycle:
 - a. Check that the setting of the system relief valve is 1325 ± 75 psig.
 - b. Verify that each pump will deliver 43 gpm against a system head of at least 1190 psig.
 - c. Initiate one of the Standby Liquid Control System loops from the control room after arranging suction from the test tank and pump demineralized water into the reactor

4.4.A.3 Each Operating Cycle (Continued)

- c. vessel. This test checks the explosive charge, proper operation of the associated valves and selected pump operability. The replacement charge to be installed will be selected from a manufactured batch which has been tested.
- d. Both loops including both explosive valves should be tested in the course of two operating cycles.

3.4.B Operating with Inoperable Components

If one Standby Liquid Control redundant component is inoperable the reactor may remain in operation for a period not to exceed seven (7) days provided the redundant component is operable.

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. Volume

The volume of the liquid control solution in the liquid control tank shall be maintained as required in Figure 3.4-1.

2. Concentration

The concentration of the liquid control tank shall be maintained as required in Figure 3.4-1.

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the liquid control solution:

1. Volume

Check the standby liquid control tank volume at least once per day.

2. Concentration

Check the concentration of the liquid in the standby liquid control tank by chemical analysis:

3.4.B Operation with Inoperable Components (Continued)

redundant component upstream of the explosive valves may be out of operation should be consistent with the very small probability of failure of both the control rod shutdown capability and the alternate component in the system, together with the fact that nuclear system cooldown takes several hours while liquid control solution injection takes about two hours. This indicates the considerable time available for testing and restoring the Standby Liquid Control System to an operable condition after testing, while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by the surveillance testing required by Specifications 4.4.A and 4.6.K.

Each positive displacement pump is sized to inject the solution into the reactor in 50 to 125 minutes, independent of the amount of solution in the tank. The slower rate assures that the boron gets into the reactor considerably quicker than the cooldown rate. The faster injection rate limit assures that there is sufficient mixing so that the boron does not recirculate through the core in uneven concentrations which could possibly cause the nuclear power to rise and fall cyclically.

The maximum solution volume in Figure 3.4-1 is determined by the tank size. A minimum required pump flow rate of 41.2 gpm has been determined by the rate required using one pump to inject the maximum volume of control solution (5150 gallons) within the maximum allowed time of 125 minutes. Using the minimum pump rate of 41.2 gpm and the fastest injection time of 50 minutes, a minimum quantity of 2060 gallons of solution having a 20.2 percent sodium pentaborate concentration is required to meet the shutdown requirement. For the maximum expected pump capacity of 43 gpm a minimum volume of 2150 gallons is that volume which could be injected in the minimum allowed time of 50 minutes.

C. Sodium Pentaborate Solution

Limiting Conditions for Operation:

The liquid control solution is acceptable if the combination of volume and concentration of the solution is maintained in the region required as shown in Figure 3.4-1 and the solution temperature is maintained 10°F above the corresponding saturation temperature (Figure 3.4-2) to guard against boron precipitation.

Surveillance Requirements:

Level indication and an alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The combination of volume and concentration required of the solution is such that should evaporation occur from any point within the acceptable region, a low level alarm will annunciate before the combination of volume and concentration requirements are unacceptable. The test interval has been established in consideration of these factors. The solution temperature and volume are checked at a high enough frequency to assure a high reliability of acceptability of the solution should it ever be required. Temperature and liquid level alarms for the system are annunciated in the control room.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 CORE AND CONTAINMENT COOLING SYSTEMSApplicability

The Limiting Conditions for Operation apply to the operational status of the core and containment cooling systems.

Objective

The objective of the Limiting Conditions for Operation is to assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

SpecificationsA. Core Spray (CS) System1. Normal System Availabilitya. The CS System shall be operable:

- (1) Prior to reactor startup from a cold condition, or
- (2) When irradiated fuel is in the the reactor vessel and the reactor pressure is greater than atmospheric pressure, except as stated in Specification 3.5.A.2.

4.5 CORE AND CONTAINMENT COOLING SYSTEMSApplicability

The Surveillance Requirements apply to the core and containment cooling systems when the corresponding limiting conditions for operation are in effect.

Objective

The objective of the Surveillance Requirements is to verify the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

SpecificationsA. Core Spray (CS) System1. Normal System Availability

CS system testing shall be performed as follows:

- | <u>Item</u> | <u>Frequency</u> |
|---|----------------------|
| a. Simulated Automatic Actuation Test | Once/Operating Cycle |
| b. System flow rate:
Each loop shall deliver at least 4625 gpm against a system head corresponding to a reactor pressure of at least 113 psig. | Once/3 months |

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.A.2. Operation with Inoperable Components

If one CS system loop is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days providing all active components in the other CS system loop, the RHR system LPCI mode and the diesel generators are operable.

3. Shutdown Requirements

If Specification 3.5.A.1.a or 3.5.A.2 cannot be met the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)1. Normal System Availability

a. The RHR System shall be operable:

- (1) Prior to reactor startup from a cold condition, or
- (2) When irradiated fuel is in the reactor vessel and the reactor pressure is greater than atmospheric except as stated in Specification 3.5.B.2.

4.5.A.2. Surveillance with Inoperable Components

When it is determined that one core spray loop is inoperable at a time when operability is required, the diesel generators associated with the remaining operable core spray loop shall be demonstrated to be operable immediately.

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)1. Normal Operational Tests

RHR system testing shall be performed as follows:

<u>Item</u>	<u>Frequency</u>
a. Air test on drywell headers and nozzles and air or water test on torus headers and nozzles.	Once/10 years

3.5.B.1 Normal System Availability (Cont.) 4.5.B.1. Normal Operational Tests (Cont.)

b. One RHR loop with two pumps or two loops with one pump per loop shall be operable in the shutdown cooling mode when irradiated fuel is in the reactor vessel and the reactor pressure is atmospheric except prior to a reactor startup as stated in Specification 3.5.B.1.a.

c. The reactor shall not be started up with the RHR system supplying cooling to the fuel pool.

d. During reactor power operation, the LPCI system discharge cross-tie valve, Ell-F010, shall be in the closed position and the associated valve motor starter circuit breaker shall be locked in the off position. In addition, an annunciator which indicates that the cross-tie valve is not in the fully closed position shall be available in the control room.

e. Both recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

2. Operation with Inoperable Components

a. One LPCI Pump Inoperable

If one LPCI pump is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days provided that the remaining LPCI pumps, both LPCI subsystem flow paths, the Core Spray system, and the associated diesel generators are operable.

b. One LPCI Subsystem Inoperable

A LPCI subsystem is considered to be inoperable if (1) both of the LPCI pumps within that system are inoperable or (2) the active valves in the subsystem flow path are inoperable.

<u>Item</u>	<u>Frequency</u>
b. Simulated Automatic Actuation Test	Once/Operating Cycle

c. System flow rate: Each RHR pump shall deliver at least 7700 gpm against a system head corresponding to a reactor pressure of at least 20 psig.	Once/3 months
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d. Both recirculation pump discharge valves shall be tested for operability during any outage exceeding 48 hours, if operability tests have not been performed during the preceding month.

2. Surveillance with Inoperable Components

a. One LPCI Pump Inoperable

When one LPCI is inoperable, the diesel generators associated with the remaining LPCI pumps shall be demonstrated to be operable immediately and daily thereafter, until the inoperable LPCI pump is restored to normal service.

b. One LPCI Subsystem Inoperable

When one LPCI subsystem is inoperable, the diesel generators associated with the remaining LPCI subsystem shall be demonstrated to be operable, immediately

3.5.B.3 Shutdown Requirements

If Specification 3.5.B.1.a or 3.5.B.2 cannot be met, the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

C. RHR Service Water System1. Normal System Availability

The RHR service water system shall be operable:

- a. Prior to reactor startup from a Cold Condition, or
- b. when irradiated fuel is in the reactor vessel and the reactor vessel pressure is greater than atmospheric pressure except as stated in Specification 3.5.C.2., or
- c. when irradiated fuel is in the reactor vessel and the reactor is depressurized at least one RHR service water loop shall be operable.

2. One Pump Inoperable

If one RHR service water pump is inoperable the reactor may remain in operation for a period not to exceed thirty (30) days provided all other active components of both subsystems are operable.

4.5.C. RHR Service Water System1. Normal Operational Tests

RHR service water system testing shall be performed as follows:

<u>Item</u>	<u>Frequency</u>
Pump Capacity Test: Each RHR service water pump shall deliver at least 4000 gpm at a system head of at least 847 feet.	Once/3 months

3.5.C.3 Two Pumps Inoperable

If two RHR service water pumps are inoperable, the reactor may remain in operation for a period not to exceed seven (7) days provided all redundant active components in both of the RHR service water subsystems are operable.

4.5.C.3. Two Pumps Inoperable

When two RHR service water pumps are inoperable, the diesel generators associated with the remaining operable RHR service water subsystems shall be demonstrated to be operable immediately and daily thereafter for seven (7) days or until the inoperable components are returned to normal operation.

4. Shutdown Requirements

If Specifications 3.5.C cannot be met, the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

D. High Pressure Coolant Injection (HPCI) System1. Normal System Availability

a. The HPCI System shall be operable:

- (1) Prior to reactor startup from a cold condition, or
- (2) When irradiated fuel is in the reactor vessel and the reactor pressure is greater than 150 psig, except as stated in Specification 3.5.D.2.

D. High Pressure Coolant Injection (HPCI) System1. Normal Operational Tests

HPCI system testing shall be performed as follows:

<u>Item</u>	<u>Frequency</u>
a. Simulated Automatic Actuation Test	Once/Operating Cycle
b. Flow rate at normal reactor vessel operating pressure and Flow rate at 150 psig reactor pressure	Once/3 months Once/Operating Cycle

4.5.D.1.b Normal Operational Tests (Continued)

The HPCI pumps shall deliver at least 4250 gpm during each flow rate test.

3.5.D.2 Operation with Inoperable Components

If the HPCI system is inoperable, the reactor may remain in operation for a period not to exceed fourteen (14) days provided the ADS, CS system, RHR system LPCI mode, and RCIC system are operable.

With the surveillance requirements of Specification 4.5.D.1 not performed at the required frequencies due to low reactor steam pressure, reactor startup is permitted and the appropriate surveillance will be performed within 12 hours after reactor steam pressure is adequate to perform the tests.

3. Shutdown Requirements

If Specification 3.5.D.1. or 3.5.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor vessel pressure shall be reduced to 150 psig or less within 24 hours.

E. Reactor Core Isolation Cooling (RCIC) System1. Normal System Availability

- a. The RCIC system shall be operable with an operable flow path capable of (automatically) taking suction from the suppression pool and transferring the water to the reactor pressure vessel:

- (1) Prior to reactor startup from a cold condition, or

2. Surveillance with Inoperable Components

When the HPCI system is inoperable, the ADS actuation logic shall be demonstrated to be operable immediately. The ADS logic shall be demonstrated to be operable daily thereafter until the HPCI system is returned to normal operation.

E. Reactor Core Isolation Cooling (RCIC) System1. Normal Operational Tests

RCIC system testing shall be performed as follows:

<u>Item</u>	<u>Frequency</u>
a. Simulated Automated Actuation (and restart*) Test	Once/Operating Cycle

*Automatic Restart on a Low Water Level Which is Subsequent to a High Level Trip.

LIMITING CONDITIONS FOR OPERATIONSURVEILLANCE REQUIREMENTS3.5.E.1. Normal System Availability (Cont.)

- a.(2) when there is irradiated fuel in the reactor vessel and the reactor pressure is above 150 psig, except as stated in Specification 3.5.E.2.

4.5.E.1 Normal Operational Tests (Cont.)

<u>Item</u>	<u>Frequency</u>
b. Verifying that suction for the RCIC system is automatically transferred from the CST to the suppression pool on a simulated low CST level or high suppression pool level signal.	Once/Operating Cycle
<u>Item</u>	<u>Frequency</u>
c. Flow rate at normal reactor vessel operating pressure and Flow rate at 150 psig reactor pressure	Once/3 months Once/Operating Cycle

The RCIC pump shall deliver at least 400 gpm during each flow test.

2. Operation with Inoperable Components

If the RCIC system is inoperable, the reactor may remain in operation for a period not to exceed seven (7) days if the HPCI system is operable during such time.

3. Shutdown Requirements

If Specification 3.5.E.1 or 3.5.E.2 is not met, an orderly shutdown shall be initiated and the reactor shall be depressurized to less than 150 psig within 24 hours.

3.5.G Minimum Core and Containment Cooling Systems Availability

During any period when one of the standby diesel generators is inoperable, continued reactor operation is limited to seven (7) days unless operability of the diesel generator is restored within this period. During such seven (7) days all of the components in the RHR system LPCI mode and containment cooling mode shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours. Specification 3.9 provides further guidance on electrical system availability.

Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems and the LPCI and containment cooling subsystems of the RHR system may be inoperable provided that the shutdown cooling subsystem of the RHR system is operable in accordance with Specification 3.5.B.1.b and that no work is being done which has the potential for draining the reactor vessel.

H. Maintenance of Filled Discharge Pipes

Whenever the core spray system, LPCI, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The suction of the HPCI pumps shall be aligned to the condensate storage tank.

4.5.G Surveillance of Core and Containment Cooling Systems

When it is determined that one of the standby diesel generators is inoperable, the remaining diesels shall be demonstrated to be operable immediately and daily thereafter.

H. Maintenance of Filled Discharge Pipes

The following surveillance requirements shall be performed to assure that the discharge piping of the core spray system, LPCI, HPCI, and RCIC are filled when required:

1. Every month prior to the testing of the LPCI and core spray systems, the discharge piping of these systems shall be vented

3.5.J Plant Service Water System1. Normal Availability

The reactor shall not be made critical from the cold shut-down condition unless the Plant Service Water System (including 3 plant service water pumps and the standby service water pump) is operable.

2. Inoperable Components

- a. The standby service water pump may be inoperable for a period not to exceed 60 days provided that an alternate Unit 1 plant service water cooling source to the 1B diesel generator is OPERABLE.
- b. One PSW pump may be inoperable for a period not to exceed 30 days provided all diesel generators associated with the operable PSW components are operable.
- c. One PSW pump and the standby service water pump may be inoperable for a period not to exceed 30 days provided all diesel generators associated with the operable PSW components are operable.
- d. Two PSW pumps or one PSW division may be inoperable for a period not to exceed 7 days provided the diesel generators associated with the operable PSW components are operable.

4.5.J Plant Service Water Systems

1. The automatic pump start functions and automatic isolation functions shall be tested once per operating cycle.

2. Inoperable Components

- a. With the standby service water subsystem inoperable for up to 60 days, provide Unit 1 service water cooling to the 1B diesel generator by verifying OPERABILITY of an alternate Unit 1 service water cooling source within 8 hours. Otherwise, declare the 1B diesel generator inoperable and take the action required by Specification 3.9.B.2.
- b. When one PSW pump is made or found to be inoperable, all diesel generators associated with the operable PSW components shall be demonstrated to be operable immediately and weekly thereafter.
- c. When one PSW pump and the standby service water pump are made or found to be inoperable, all diesel generators associated with the operable PSW components shall be demonstrated to be operable immediately and weekly thereafter.
- d. When two PSW pumps or one PSW division are made or found to be inoperable, the diesel generators associated with the operable PSW components shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.J Plant Service Water System4.5.J Plant Service Water Systems2. Inoperable Components (Cont'd)2. Inoperable Components (Cont'd)

- e. Two PSW pumps or one PSW division, and the standby service water pump may be inoperable for a period not to exceed 7 days provided the diesel generators associated with the operable PSW components are operable.

- e. When two PSW pumps or one PSW division, and the standby service water pump are made or found to be inoperable, the diesel generators associated with the operable PSW components shall be demonstrated to be operable immediately and daily thereafter.

For each condition above in which the standby service water pump is inoperable, cooling water to diesel generator 1B shall be intertied with the PSW divisional piping supply.

When cooling water to diesel generator 1B is intertied with the PSW divisional piping supply, operability of the divisional interlock valves shall be demonstrated.

3. Shutdown Requirements

If the requirements of Specifications 3.5.J.1 and 3.5.J.2 cannot be met the reactor shall be placed in the cold shutdown condition within 24 hours.

3.5.K Equipment Area Coolers4.5.K Equipment Area Coolers

- 1. The equipment area coolers serving the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray or Residual Heat Removal (RHR) pumps must be operable at all times when the pump or pumps served by that specific cooler is considered to be operable.
- 2. When an equipment area cooler is not operable, the pump(s) served by that cooler must be considered inoperable for Technical Specification purposes.

- 1. Each equipment area cooler is operated in conjunction with the equipment served by that particular cooler; therefore, the equipment area coolers are tested at the same frequency as the pumps which they serve.

3.5 CORE AND CONTAINMENT COOLING SYSTEMSA. Core Spray (CS) System1. Normal System Availability

Analyses presented in Section 6 of the FSAR and Appendix I of the HNP-2 PSAR demonstrated that the core spray system provides adequate cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to below 2,300°F which assures that core geometry remains intact and to limit any clad metal-water reaction to less than one percent. Core spray distribution has been shown in tests of systems similar in design to HNP-1 to exceed the minimum requirements. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel.

The intent of the CS system specifications is to prevent operation above atmospheric pressure without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgement based on experiences and supported by availability analysis. Assurance of the availability of the remaining systems is increased by demonstrating operability immediately and by requiring selected testing during the outage period.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core. Requiring two operable RHR pumps and one CS pump provides redundancy to ensure makeup water availability.

2. Operation with Inoperable Components

Should one core spray loop become inoperable, the diesel generators associated with the remaining operable core spray loop are demonstrated to be operable to ensure their availability should the need for core cooling arise. The surveillance testing required by Specifications 4.5.A, 4.5.H, and 4.6.K ensures the availability of the remaining core spray loop. The surveillance testing required by Specifications 4.5.B, 4.5.C, 4.5.H, and 4.6.K ensures the availability of the RHR system. These provide extensive margin over the operable equipment needed for adequate core cooling. With due regard for this margin, the allowable repair time of 7 days was chosen.

B. Residual Heat Removal (RHR) System (LPCI and Containment Cooling Mode)1. Normal System Availability

The RHR system LPCI mode is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the core spray system; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI mode of the RHR system and the core spray system provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high-pressure emergency core cooling systems.

3.5.B.1. Normal System Availability (Continued)

Observation of the stated requirements for the containment cooling mode assures that the suppression pool and the drywell will be sufficiently cooled, following a loss-of-coolant accident, to prevent primary containment overpressurization. The containment cooling function of the RHR system is permitted only after the core has been reflooded to the two-thirds core height level. This prevents inadvertently diverting water needed for core flooding to the less urgent task of containment cooling. The two-thirds core height level interlock may be manually bypassed by a keylock switch.

The intent of the RHR system specifications is to prevent operation above atmospheric pressure without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. The allowable repair times have been selected using engineering judgement based on experiences and supported by availability analysis. Assurance of the availability of the remaining systems is increased by demonstrating operability immediately and by requiring selected testing during the outage period.

When the reactor vessel pressure is atmospheric, the limiting conditions for operation are less restrictive. At atmospheric pressure, the minimum requirement is for one supply of makeup water to the core.

2. Operation with Inoperable Components

With one LPCI pump inoperable or one LPCI subsystem inoperable, adequate core flooding is assured. The surveillance testing required by Specifications 4.5.B, 4.5.C, 4.5.H, and 4.6.K ensures the availability of the redundant LPCI pumps and LPCI subsystems. The surveillance testing required by Specifications 4.5.A, 4.5.H, and 4.6.K ensures the availability of the Core Spray system. In addition, the associated diesel generators are demonstrated to be operable.

3.5.D.2 Operation with Inoperable Components

The HPCI system serves as a backup to the RCIC system as a source of feedwater makeup during primary system isolation conditions. The ADS serves as a backup to the HPCI system for reactor depressurization for postulated transients and accidents. The ADS is checked for operability if the HPCI system is determined to be inoperable. In addition, the surveillance testing required by Specifications 4.5.E, 4.5.H, and 4.6.K ensures the operability of the RCIC system. Considering the redundant systems, an allowable repair time of seven (7) days was selected.

E. Reactor Core Isolation Cooling (RCIC) System

1. Normal System Availability

The various conditions under which the RCIC system plays an essential role in providing makeup water to the reactor vessel have been identified by evaluating the various plant events over the full range of planned operations. The specifications ensure that the function for which the RCIC system was designed will be available when needed.

Because the low-pressure cooling systems (LPCI and core spray) are capable of providing all the cooling required for any plant event when nuclear system pressure is below 150 psig, the RCIC system is not required below this pressure. RCIC system design flow (400 gpm) is sufficient to maintain water level above the top of the active fuel for a complete loss of feedwater flow at the design power.

Two sources of water are available to the RCIC system. Suction is initially taken from the condensate storage tank and is automatically transferred to the suppression pool upon low CST level or high suppression pool level.

2. Operation With Inoperable Components

Consideration of the availability of the RCIC system reveals that the average risk associated with failure of the RCIC system to cool the core when required is not increased if the RCIC system is inoperable for no longer than seven (7) days, provided that the HPCI system is operable during this period. The surveillance testing required by Specifications 4.5.D, 4.5.H, and 4.6.K ensures the operability of the HPCI system.

F. Automatic Depressurization System (ADS)

1. Normal System Availability

This specification ensures the operability of the ADS under all conditions for which the depressurization of the nuclear system is an essential response to Unit abnormalities.

The nuclear system pressure relief system provides automatic nuclear system depressurization for small breaks in the nuclear system so that the low-pressure coolant injection (LPCI) and the core spray system can operate to protect the fission product barrier. Note that this Specification applies only to the automatic feature of the pressure relief system.

3.5.F.1. Normal System Availability (continued)

Specification 3.6 states the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures yet be fully capable of performing their pressure relief function.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the Core Standby Cooling Systems.

2. Operation with Inoperable Components

With one ADS valve known to be incapable of automatic operation six valves remain operable to perform their ADS function. However, since the ECCS Loss of Coolant Accident analysis for small line breaks assumed that all seven ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for seven (7) days provided that the actuation logic for the (remaining) six ADS valves is demonstrated to be operable. The surveillance testing required by Specifications 4.5.D, 4.5.H, and 4.6.K ensures the availability of the HPCI system.

3. Minimum Core and Containment Cooling Systems Availability

The purpose of this Specification is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray loop were out of service and the diesel which powered the opposite core spray were out of service, only 2 RHR pumps would be available. Specification 3.9 must also be consulted to determine other requirements for the diesel generators. In addition, refer to definition 1.0.00 for Cumulative Downtime requirements.

This specification establishes conditions for the performance of major maintenance, such as draining of the suppression pool. The availability of the shutdown cooling subsystem of the RHR system and the RHR service water system ensure adequate supplies of reactor cooling and emergency makeup water when the reactor is in the Cold Shutdown condition. In addition, this specification provides that, should major maintenance be performed, no work will be performed which could lead to draining the water from the reactor vessel.

3.5.J/4.5.J Plant Service Water System

The Plant Service Water (PSW) system consists of two subsystems (divisions) of two pumps each and a separate standby service water pump system for diesel generator 1B. During normal full power operation the two subsystems function as a 3 out of 4 pump cross connected system supplying cooling water to the turbine and reactor building cooling systems. In the event of an accident signal, non-safety-related cooling loads are isolated and the PSW pumps in the two subsystems supply cooling water to diesel generators 1A and 1C, the reactor building cooling system and the control room air conditioners, while the standby service water pump is available to automatically supply cooling water to diesel generator 1B should it be needed. Additionally, diesel 1B has a manual backup water supply available from the Unit 1 Division 1 or Division 2 PSW subsystems so that during maintenance on the standby diesel service water pump, either division of the PSW system can manually be aligned to supply cooling water to the 1B diesel. The two subsystems and the standby service water pump system are split in the accident mode for greater reliability with one pump in each of the two subsystems automatically starting while a start signal from diesel generator 1B initiates standby service water pump operation. Only one of the Division 1 PSW pumps and one of the Division 2 PSW pumps are required for cooling diesel generators 1A and 1C, respectively, while the standby service water pump provides adequate cooling water to diesel generator 1B. In the event that the standby service water pump is inoperable, the HNP-1 Division 1-Division 2 intertie supply piping can be aligned to cool the 1B diesel. In this condition, one PSW pump is capable of supplying the cooling requirements for the reactor building cooling system, the control room air conditioners, and the 1A, 1B, and 1C diesel generators.

The PSW system can supply all power generation systems at full load and the diesel generators with redundancy if one PSW pump and/or the standby service water pump are inoperable. Hence, a 60-day outage time is justified if the standby service water pump is inoperable since all four PSW pumps are available (divisional intertie to 1B diesel required). In addition, a 30-day outage is justified if one PSW pump is inoperable, or if one PSW pump and the standby service water pump are inoperable (divisional intertie to 1B diesel required). Should two PSW pumps (or one subsystem) become inoperable, or should two PSW pumps (or one subsystem) and the standby service water pump become inoperable (division intertie to 1B diesel required) plant operation will probably only continue at less than full power. However, safety-related loads are still adequately powered for these conditions. Therefore, a 7 day outage time is justified for such events.

The surveillance testing required by Specification 4.6.K will provide adequate assurance that the PSW system will be operable when required.

K. Engineering Safety Features Equipment Area Coolers

The equipment area cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguard compartments without adequate ventilation flow or cooling is such that continued operation of the safeguard equipment or associated auxiliary equipment cannot be assured.

The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. The testing is adequate to assure the operability of the equipment area coolers.

L. References

1. FSAR Section 6, Core Standby Cooling System.
2. HNP-2 PSAR Appendix I, Conformance to NRC Interim Acceptance Criteria for Emergency Core Cooling Systems.

- b. With the relief valve function and/or the low low set function of more than one of the above required reactor coolant system relief/safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3.6.I Jet Pumps

Whenever the reactor is in the Start & Hot Standby or Run Mode with both recirculating pumps operating, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

4.6.I Jet Pumps

Whenever both recirculating pumps are operating with the reactor in the Start & Hot Standby or Run Mode, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously.

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
3. The diffuser to lower plenum differential pressure reading on an individual jet pump vary from the mean of all jet pump differential pressures by more than 10%.

3.6.J Recirculation Pumps Speeds

1. Core thermal power shall not exceed 1% of rated thermal power without forced recirculation.
2. Operation with a single recirculation pump is permitted for 24 hours unless the recirculation pump is sooner made operable. With one recirculation pump not in operation, initiate action within 15 minutes or continue action to reduce reactor power to or below the limit specified in Figure 3.6-5 within 2 hours. If the pump cannot be made operable or the limit of Figure 3.6-5 cannot be met within the required time, the reactor shall be in cold shutdown within 24 hours.

4.6.J Recirculation Pump Speeds

Recirculation pump speeds shall be recorded at least once per day.

3.6.J Recirculation Pump Speeds (continued)

3. Following one pump operation the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

3.6.K STRUCTURAL INTEGRITY1. Normal Condition

The structural integrity of ASME Code Class 1, 2, and 3 (equivalent) components shall be maintained in accordance with the Surveillance Requirements of Specification 4.6.K.

2. Off-Normal Conditions

- a. With the structural integrity of any ASME Code Class 1 component not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.
- b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 212°F.
- c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

4.6.K STRUCTURAL INTEGRITY

Surveillance Requirements for in-service inspection and testing of ASME Code Class 1, 2, and 3 (equivalent) components shall be applicable as follows:

1. In-service inspection of ASME Code Class 1, 2, and 3 (equivalent) components and in-service testing of ASME Code Class 1, 2, and 3 (equivalent) pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10CFR50, Section 50.55a(g) (6) (i).
2. Performance of the above in-service inspection and testing activities shall be in addition to other specified Surveillance Requirements.
3. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

These pages have been left blank.

3.6.K STRUCTURAL INTEGRITY

In-service inspection of ASME Code Class 1, 2, and 3 (equivalent) components and in-service testing of ASME Code Class 1, 2, and 3 (equivalent) pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10CFR50.55a(g). This objective will maintain the structural integrity of safety-related components, pumps, and valves which are necessary to safely shut down the plant or mitigate the consequences of an accident.

These pages have been left blank.

4.7.C.1. Surveillance While Integrity Maintained (Cont'd)

- b. Secondary containment capability to maintain a minimum 1/4-inch of water vacuum under calm wind (≤ 5 mph) conditions with each filter train flow rate not more than 4000 cfm shall be demonstrated at each refueling outage, prior to refueling.

3.7.C.2 Violation of Secondary Containment Integrity

- a. Without Hatch-Unit 1 secondary containment integrity, restore Hatch - Unit 1 secondary containment integrity within 4 hours, or perform the following (as applicable):
 - (1) Suspend irradiated fuel and/or fuel cask handling in the Hatch-Unit 1 secondary containment.
 - (2) Be in at least Hot Shutdown within the next 12 hours and meet the Conditions of 3.7.C.1.a within the next 24 hours.
- b. Without Hatch-Unit 1 secondary containment, refer to the following Hatch-Unit 2 Technical Specifications, for LCO's to be followed for Hatch-Unit 2:
 - (1) Section 3.6.5.1.
 - (2) Section 3.9.5.1.

2. Surveillance After Integrity Violated

After a secondary containment violation is determined the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment. The ability to maintain the remainder of the secondary containment at 1/4-inch of water vacuum pressure under calm (≤ 5 mph) wind conditions shall be confirmed.

D. Primary Containment Isolation Valves1. Valves Required to be Operable

During reactor power operation, all primary containment isolation valves listed in Table 3.7-1 and all reactor coolant system instrument line excess flow check valves shall be operable except as stated in Specification 3.7.D.2.

D. Primary Containment Isolation Valves1. Surveillance of Operable Valves

Surveillance of the primary containment isolation valves shall be performed as follows:

- a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and the closure times specified in Table 3.7-1.

ENCLOSURE 2

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In addition to the obvious changes made as a result of amendments issued to some of the affected Technical Specification pages since the October 3, 1978 submittal, the following are significant changes made to "modernize" the aforementioned submittal:

<u>Page</u>	<u>Tech Spec</u>	<u>Change</u>	<u>Reason for Change</u>
iv	Table of Contents	Page numbers for 3.6/4.6.I and 3.6/4.6.J change from pg. 3.6-9 and 3.6-9a, respectively to 3.6-9b for both 3.6/4.6.I and 3.6/4.6.J.	Editorial
3.4-5	3.5.C	Add degree sign to LCO for sodium pentaborate solution relative to solution temperature maintenance.	Editorial
3.5-1	4.5.A.1.b	Add the words "corresponding to a reactor pressure" relative to system head.	Proposed wording addition consistent with Hatch 2 Tech Specs.
3.5-2	4.5.B.1.a	Test frequency changed from "Once/5 years" to "Once/10 years".	Proposed testing frequency change consistent with requirements of ASME Section XI Code.
3.5-3	4.5.B.1.c	Add the words "corresponding to a reactor pressure" relative to system head.	Proposed wording addition consistent with Hatch 2 Tech Specs.
3.5-3	4.5.B.2.a	Add the words "associated with the remaining LPCI pumps" relative to diesel generators with one LPCI pump inoperable.	Editorial in nature to be consistent with proposed wording in other system Tech Spec sections, e.g., RHR SW.
3.5-3	4.5.B.2.b	Add the words "associated with the remaining LPCI subsystem" relative to diesel generators with one LPCI subsystem inoperable.	Editorial. Proposed wording similar to that proposed in other system Tech Spec sections, RHR SW.
3.5-5	3.5.C.1.b	Word "or" added	Editorial

ENCLOSURE 2 (Continued)

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<u>Page</u>	<u>Tech Spec</u>	<u>Change</u>	<u>Reason for Change</u>
3.5-5	4.5.C.1	Deleted the words "After pump maintenance and" under "Frequency".	Pump test required by ASME Section XI Code and by Hatch plant procedures after pump maintenance. Tech Spec redundancy not necessary due to Code requirements and plant procedural requirements.
3.5-13	3.5.K.2	Existing wording concerning equipment area cooler(s) inoperability relative to pump(s) served by cooler(s) retained.	Plant desires retention of existing Tech Spec.
3.6-9b	4.6.I, 3.6.J, and 4.6.J	4.6.I.2 and 3, 3.6.J.1 and 2, and 4.6.J relocated from pg. 3.6-10.	Editorial in nature to accommodate new wording for 3.6.K/4.6.K on pg. 3.6-10.
3.6-9c	3.6.J	3.6.J.3 relocated from pg. 3.6-10.	Editorial in nature to accommodate new wording for 3.6.K/4.6.K on pg. 3.6-10.
3.6-10	4.6.K.1	Word "written" deleted relative to specific relief.	Should emergency Code relief ever be necessary, verbal relief could suffice to support inspections, etc. pending receipt of written relief from NRC.
3.6-10	4.6.K	Deleted 4.6.K.4 which concerned reporting of inservice inspection results to NRC after 5 years.	Submittal of report to NRC in 1980 (i.e., 5 years after start of commercial operation) completed this Tech Spec requirement; thus, 4.6.K.4 proposed previously is no longer required.

ENCLOSURE 2 (Continued)

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<u>Page</u>	<u>Tech Spec</u>	<u>Change</u>	<u>Reason for Change</u>
3.6-13	4.7.D.1	Deleted references to "Specification 4.6.K" previously shown in 4.7.D.1.b and 4.7.D.1.d on pgs. 3.6-13 and 3.6-14, respectively, in October 3, 1978 submittal.	Awaiting NRC review and approval of proposed Appendix J Tech Specs (as discussed in GPC June 25, 1985 letter NED-85-483). Possible addition of reference to Specification 4.6.K in affected sections at later date after NRC review and approval of previously proposed Appendix J Tech Spec changes.

ENCLOSURE 3

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Pursuant to 10 CFR 50.59, the Plant Review Board has reviewed the attached proposed amendment to the Plant Hatch Unit 1 Technical Specifications and has determined that implementation of the proposed amendment does not constitute an unreviewed safety question.

PROPOSED CHANGES

The proposed Technical Specification changes relative to inservice inspection/inservice testing provide for the following:

1. Deletion of individual surveillance specifications for pumps and valves by adding new wording to Technical Specification sections 3.6.k and 4.6.k which requires inservice testing of ASME Code Class 1, 2, and 3 (equivalent) pumps and valves per the ASME Section XI Code in accordance with 10 CFR 50.55a;
2. Deletion of Technical Specification Table 4.6-1 since Technical Specifications are being changed to require inservice inspection/inservice testing per an external document (i.e., ASME Section XI Code);
3. Deletion of requirement to demonstrate operability of safety-related components (e.g., ECCS, service water, etc.) when a redundant or associated safety-related component is declared inoperable. This change is consistent with that of Standard Technical Specifications as discussed in the transmittal letter for this Enclosure. The only surveillance requirements which are applicable are those normally performed in accordance with ASME Section XI pursuant to 10 CFR 50.55a;
4. Clarification of Technical Specifications 4.5.A.1.b and 4.5.B.1.c for Core Spray and RHR, respectively, to indicate that flow rate testing will be conducted at a system pressure corresponding to a reactor pressure as given in the aforementioned sections. This will eliminate any confusion that may arise over point of pressure measurement and is consistent with Hatch Unit 2 Technical Specifications wording;
5. Change of frequency of RHR drywell and torus spray headers and nozzles air/water test from once/5 years to once/10 years. Proposed testing frequency is consistent with ASME Section XI Code requirements;
6. Deletion of the requirement in existing Technical Specification 4.5.C.1.b for a pump capacity test following pump maintenance. Post-maintenance testing of the affected pump(s) is required by the ASME Section XI Code and existing plant surveillance procedures; thus, redundancy in Technical Specifications is not necessary;

ENCLOSURE 3 (Continued)

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7. Revision of Technical Specification "Bases" sections to reflect inservice inspection/inservice testing per ASME Section XI Code requirements by Technical Specification section reference (i.e., 4.6.K) and other Technical Specification references, as appropriate; and,
8. Editorial changes in "Table of Contents", etc., as appropriate, to support inservice inspection/inservice testing-related Technical Specification changes.

BASIS

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety are not increased above those analyzed in the Final Safety Analysis Report (FSAR) due to these changes because the changes do not change equipment operation but only testing requirements. The possibility of an accident or malfunction of a different type than analyzed in the FSAR does not result from these changes because the changes have no effect on equipment operation, thus, no new modes of failure are created. The margin of safety as defined in Technical Specifications is not reduced as a result of the changes because equipment operability is adequately assured by inservice inspection/inservice testing in accordance with ASME Boiler and Pressure Vessel Code requirements pursuant to 10 CFR 50.55a.

ENCLOSURE 4

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Pursuant to 10 CFR 50.92, Georgia Power Company has evaluated the attached proposed amendment for Plant Hatch Unit 1 and has determined that its adoption would not involve a significant hazard. The basis for this determination is as follows:

PROPOSED CHANGES

On page iv of the Table of Contents, change the page numbers for items 3.6.I/4.6.I and 3.6.J/4.6.J and omit the reference to "Primary Pressure Boundary" under items 3.6.K/4.6.K.

BASIS

The changes are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register. The page numbers for items 3.6.I/4.6.I and 3.6.J/4.6.J are revised due to text processing to accommodate new proposed wording for 3.6.K/4.6.K on page 3.6-10 of the Technical Specifications. The deletion of the reference to "Primary Pressure Boundary" is necessary since the proposed inservice inspection/inservice testing-related changes to the Technical Specifications covers a broader scope than the primary pressure boundary. As a result, the existing title for item 3.6.K/4.6.K is modified to reflect this.

PROPOSED CHANGE

Delete all references to Table 4.6-1 in the List of Tables on page viii of the Technical Specifications.

BASIS

This change is consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register. The change is administrative in nature in that it removes reference to the old Inservice Inspection Program (Table 4.6-1) since the old program was superceded by a new inspection program as required by 10 CFR 50.55a. In addition, the subject table is deleted since it does not reflect current inservice inspection/inservice testing requirements.

ENCLOSURE 4 (Continued)

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PROPOSED CHANGES

Revise Technical Specification 4.4.A to reflect the following for the Standby Liquid Control System:

1. Add new Technical Specification 4.4.A.1 for monthly verification of explosive charge in each loop.
2. Install 10 CFR 50.55a testing requirements (i.e., ASME Section XI Code) in new Technical Specification 4.4.A.2 for pump testing.
3. Re-number existing Technical Specification 4.4.A.2 to read 4.4.A.3 to accommodate new Technical Specification 4.4.A.1.

BASIS

The following changes as identified above are listed by Technical Specification section number (proposed revision). These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Proposed Technical Specification 4.4.A.1: The change is consistent with Item (ii) of the "Examples" because it adds the requirement of additional control of monthly continuity checks of the explosive charges to the existing surveillance.
2. Proposed Technical Specification 4.4.A.2: This change is consistent with Item (vi) of the "Examples" and installs the 10 CFR 50.55a testing requirements for the pumps and are of a lesser frequency than the existing Technical Specifications, 3 months vice 1 month. As a result, this change may appear to slightly reduce an existing safety margin. However this change still lies within the criteria of Standard Review Plan (SRP) section 9.3.5 for periodic testing of components.
3. Proposed Technical Specification 4.4.A.3: The re-numbering of existing Technical Specification 4.4.A.2 to 4.4.A.3 is consistent with Item (i) of the "Examples" since it is an administrative change necessary to accommodate new Technical Specification 4.4.A.1.

ENCLOSURE 4 (Continued)

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PROPOSED CHANGE

Delete Technical Specification 4.4.B concerning surveillance with Standby Liquid Control System inoperable components.

BASIS

This change is consistent with Item (vi) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register. This change may possibly reduce a safety margin if the redundant portion of the Standby Liquid Control System fails in conjunction with a drastic reduction of control rod shutdown capability. Due to the low probability of such an occurrence, and the consistency of system testing and inspection required by 10 CFR 50.55a (as referenced by proposed wording in Technical Specification 4.6.K in the attached amendment), the results of this change still lie within criteria defined in SRP section 9.3.5. One portion of 4.4.B concerned itself with explosive charge continuity testing. Removal of that portion of Technical Specification 4.4.B is consistent with Item (i) since the continuity testing requirements is now covered by proposed Technical Specification 4.4.A.1.

PROPOSED CHANGES

Change the "Bases" section of Technical Specification to reflect the following:

1. In Section 3.4.B, add references to inservice testing criteria; and,
2. In Section 3.4.C, correct typographical error relative to sodium pentaborate solution maintenance temperature.

BASIS

The above changes are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register. The change to section 3.4.B incorporates the stipulations of 10 CFR 50.55a, Inservice Testing criteria (as referenced by new wording proposed for Technical Specification 4.6.K). As a result, the change to 3.4.B is consistent with the proposed changes to Technical Specification 4.4.A. The change to section 3.4.C provides for a correction of a typographical error (10F to 10⁰F). As a result, it is administrative in nature.

ENCLOSURE 4 (Continued)

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PROPOSED CHANGES

Revise Technical Specification 4.5.A to reflect the following for the Core Spray System:

1. Add clarification to the term "system head" in Technical Specification 4.5.A.1.b.
2. Delete existing Technical Specification 4.5.A.1.c concerning pump operability testing and frequency thereof.
3. Delete existing Technical Specification 4.5.A.1.d concerning motor operated valve operability testing and frequency thereof.
4. Revise Technical Specification 4.5.A.2 to remove the requirement to demonstrate the operability of the remaining Core Spray loop and the RHR LPCI mode when one Core Spray loop is inoperable. In addition, remove from section 4.5.A.2 the requirement to test the operable Core Spray loop immediately (with one loop inoperable) and daily thereafter until both loops are operable.

BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 4.5.A.1.b: The change is administrative in nature and is consistent with Item (i) of the "Examples" because it provides clarification of the term "system head". The proposed wording is similar to that which exists in the Hatch Unit 2 Technical Specifications.
2. Technical Specification 4.5.A.1.c: The deletion of the monthly pump operability testing from Technical Specifications is deemed to be consistent with Item (vi) of the "Examples" since it may appear to reduce slightly an existing safety margin. Pump operability will be conducted under the auspices of the ASME Section XI Code as referenced in 10 CFR 50.55a and at the Code-specified frequency. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of inservice inspection/in-service testing per 10 CFR 50.55a. The change lies within the criteria of SRP 6.2.2 for periodic testing of components.

ENCLOSURE 4 (Continued)

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BASIS (Continued)

3. Technical Specification 4.5.A.1.d: The deletion of the monthly motor operated valve operability testing from Technical Specifications is deemed to be consistent with Item (vi) of the "Examples" since the testing will be conducted under the auspices of the ASME Section XI Code as referenced in 10 CFR 50.55a. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of inservice inspection/in-service testing per 10 CFR 50.55a. The test frequency remains unchanged due to Code requirements even though the test frequency is deleted in Technical Specifications.
4. Technical Specification 4.5.A.2: Removal of the requirements to demonstrate the operability of remaining Core Spray loop and the RHR LPCI mode when one Core Spray loop is inoperable and the removal of the requirement to test the operable Core Spray loop immediately (with one loop inoperable) and daily thereafter until both loops are operable are consistent with Item (vi) of the "Examples." The reasons for this determination are as follows:
 - a. The removal of the requirement to demonstrate the operability of the remaining Core Spray loop and the RHR LPCI mode when one Core Spray loop is inoperable may reduce slightly an existing safety margin if a loss of RHR LPCI occurs in conjunction with the loss of one Core Spray loop. The chances of such an event are small however, and even with such an occurrence, the redundant Core Spray loop is still available. Based on these considerations and the testing requirements imposed as a result of 10 CFR 50.55a, the change still lies within the criteria of SRP sections 6.3 and 5.4.7.
 - b. The removal of the requirement to test the remaining operable Core Spray loop immediately and daily thereafter until both Core Spray loops are operable may reduce slightly an existing margin of safety by limiting the assurance of operability. However, this change still lies within the criteria for Core Spray systems as found in SRP Section 6.2.2 and its proposed revision. This is because the testing requirements of General Design Criteria (GDC)-40 (10 CFR 50 - Appendix A) are adhered to. In addition, system functionality is assured by the criteria of 10 CFR 50.55a which follows ASME Section XI

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Code guidance for inservice inspection/inservice testing. As a result, necessary assurance of the functionality of the redundant Core Spray loop is provided by consistent testing and monitoring of any degradation within the subject system.

PROPOSED CHANGES

Revise Technical Specification 4.5.B to reflect the following for the Residual Heat Removal System:

1. Change the frequency of the air/water tests for the drywell and torus spray headers and nozzles in Technical Specification 4.5.B.1.a from once/5 years to once/10 years.
2. Add clarification to the term "system head" in Technical Specification 4.5.B.1.c.
3. Delete existing Technical Specification 4.5.B.1.d concerning pump operability testing and frequency thereof.
4. Delete existing Technical Specification 4.5.B.1.e concerning motor operated valve operability and frequency thereof.
5. Re-number existing Technical Specification 4.5.B.1.f to read 4.5.B.1.d as a result of the proposed deletion of Technical Specifications 4.5.B.1.c and 4.5.B.1.d.
6. Remove the requirement to verify operability of Core Spray system and the remaining LPCI pumps and associated flow paths in Technical Specification 4.5.B.2.a in the event of loss of one LPCI pump.
7. Remove the requirement to verify operability of the active components of the remaining LPCI subsystem and the Core Spray system in Technical Specification 4.5.B.2.b in the event of loss of one LPCI subsystem.

BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

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1. Technical Specification 4.5.B.1.a: Changing of the frequency of the drywell and torus spray header and nozzle air/water testing from once/5 years to once/10 years is consistent with item (vii) of the "Examples" because the change is being made in order to maintain consistency with the requirements of the ASME Section XI Code as referenced by 10 CFR 50.55a. Technical Specification 4.6.K is being revised by the attached amendment to reference the subject Code and 10 CFR 50.55a.
2. Technical Specification 4.5.B.1.c: The change is administrative in nature and is consistent with Item (i) of the "Examples" because it provides clarification of the term "system head". The proposed wording is similar to that which exists in the Hatch Unit 2 Technical Specifications.
3. Technical Specification 4.5.B.1.d: The deletion of the monthly pump operability testing from Technical Specifications is deemed to be consistent with Item (vi) of the "Examples" since it may appear to reduce slightly an existing safety margin. Pump operability will be conducted under the auspices of the ASME Section XI Code as referenced in 10 CFR 50.55a and at the Code-specified frequency. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of the inservice inspection/in-service testing per 10 CFR 50.55a. The change lies within the criteria of SRP section 5.4.7.
4. Technical Specification 4.5.B.1.e: The deletion of the monthly motor operated valve operability testing from Technical Specifications is deemed to be consistent with Item (vi) of the "Examples" since the testing will be conducted under the auspices of the ASME Section XI Code as referenced in 10 CFR 50.55a. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of inservice inspection/in-service testing per 10 CFR 50.55a. The test frequency remains unchanged due to Code requirements even though the test frequency is deleted in Technical Specifications.
5. Technical Specification 4.5.B.1.f: The renumbering of Technical Specification 4.5.B.1.f to 4.5.B.1.d is consistent with Item (i) of the "Examples" since it is an administrative change necessitated by the deletion of Technical Specifications 4.5.B.1.d and 4.5.B.1.e.

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6. Technical Specification 4.5.B.2.a: Removal of the requirement from Technical Specification to verify the operability of the remaining LPCI pumps and associated flow paths and Core Spray system with one LPCI pump inoperable is deemed a change consistent with Item (vi) of the "Examples". The change may reduce an existing margin of safety if a loss of the Core Spray system occurs in conjunction with a loss of one LPCI pump. However, this is an unlikely occurrence, and even in such a case, the redundant LPCI system is available. Based on these considerations, and the testing requirements imposed as a result of 10 CFR 50.55a (as referenced in Technical Specification 4.6.K in the attached amendment), this change still lies within the criteria of SRP sections 5.4.7 and 6.3.
7. Technical Specification 4.5.B.2.b: Removal of the requirement from Technical Specifications to verify operability of the active components of the remaining LPCI subsystem and the Core Spray system immediately and daily thereafter with one LPCI subsystem inoperable is deemed a change consistent with Item (vi) of the "Examples". The change may slightly reduce an existing safety margin. However, this change still lies within the criteria of SRP sections 5.4.7 and 6.3. In addition, testing requirements invoked by 10 CFR 50.55a (as referenced in Technical Specification 4.6.K in the attached amendment) add additional assurance of system operability.

PROPOSED CHANGE

Add the word "or" to Technical Specification 3.5.C.1.b which specifies a condition when the RHR service water system shall be operable.

BASIS

This change is consistent with Item (i) of the "Examples of Amendments that are Considered Not likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register. The wording change results from a change in nomenclature to distinguish the two cases considered (i.e., Technical Specification 3.5.C.b - reactor vessel pressurized, Technical Specification 3.5.C.c - reactor vessel depressurized).

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PROPOSED CHANGE

Revise Technical Specification 4.5.C to reflect the following for the RHR Service Water System:

1. Delete existing Technical Specification 4.5.C.1.a concerning pump and valve operability testing and frequency thereof.
2. Revise Technical Specification 4.5.C.1.b "nomenclature" to accommodate deletion of 4.5.C.1.a.
3. Delete requirement in Technical Specification 4.5.C.1.b to perform testing after pump maintenance.
4. Delete Technical Specification 4.5.C.2 concerning surveillance with one pump inoperable.
5. Delete the requirement in Technical Specification 4.5.C.3 to test the remaining operable RHR service water subsystems immediately and daily thereafter with two RHR service water pumps inoperable.

BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 4.5.C.1.a: The deletion of the quarterly pump and valve operability testing (4.5.C.1.a) from Technical Specifications is deemed to be a change consistent with Item (vi) of the "Examples" since it may appear to reduce slightly an existing safety margin. Pump and valve operability will be conducted under the auspices of the ASME Section XI code as referenced in 10 CFR 50.55a and at the Code-specified test frequency. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of the inservice inspection/in-service testing per 10 CFR 50.55a.
2. Technical Specification 4.5.C.1.b: As a result of the deletion of Technical Specification 4.5.C.1.a, it is no longer necessary to have an item "b" suffix for Technical Specification 4.5.C.1. Deletion of the letter "b" is consistent with Item (i) of the "Examples" since it is an administrative change.

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BASIS (Continued)

3. Technical Specification 4.5.C.1.b: Deletion of the pump capacity test from the Technical Specifications after pump maintenance is performed is a change consistent with Item (vi) of the "Examples" since it may appear that an existing safety margin is being reduced. Pump testing will be conducted under the auspices of the ASME Section XI code as referenced in 10 CFR 50.55a. Technical Specification 4.6.K is being revised by the attached amendment to reference the conduct of the inservice inspection/in-service testing per 10 CFR 50.55a. The Code specifically requires a pump test after the performance of maintenance on the subject pump(s). In addition, the plant surveillance procedures also require that a pump test be conducted following pump maintenance.
4. Technical Specification 4.5.C.2: The deletion of the requirement to verify operation of the remaining RHR service water pumps/subsystems upon failure of one RHR service water pump is a change consistent with Item (vi) of the "Examples" since it may appear to be a reduction in an existing margin of safety only if the failure of the other pumps, etc. is considered. Since there are four pumps total, sufficient redundancy is provided to make such a consideration a highly unlikely event. This change does not alter the consistency of the RHR Service Water System design, operability, inspection, and testing with that of SRP Section 5.4.7.
5. Technical Specification 4.5.C.3: The removal of the Technical Specification requirements to verify immediately and daily thereafter, proper operation of the remaining two functioning RHR service water subsystems upon loss of two RHR service water pumps is considered a change consistent with Item (vi) of the "Examples" since there may be a slight reduction in an existing safety margin. However, this change does not alter the consistency of the RHR Service Water System design, operability, inspection, and testing with that of SRP section 5.4.7. This is because the testing requirements not only remain in accordance with SRP 5.4.7, but, are covered by the details of 10 CFR 50.55a as well. Technical Specification 4.6.K is being modified by the attached amendment to reference the conduct of the inservice inspection/in-service testing per 10 CFR 50.55a.

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PROPOSED CHANGES

Revise Technical Specification 4.5.D to reflect the following for the High Pressure Coolant Injection (HPCI) System:

1. Add the word "continued" to the title line for the remainder of Technical Specification 4.5.D.1.b located on Technical Specification page 3.5-7.
2. Delete existing Technical Specification 4.5.D.1.d concerning pump operability testing and frequency thereof.
3. Delete existing Technical Specification 4.5.D.1.e concerning motor operated valve operability testing and frequency thereof.
4. Delete the requirements in Technical Specification 4.5.D.2 to test the RCIC System, the RHR system LPCI mode, and the Core Spray system immediately with HPCI inoperable. In addition, in the aforementioned specification, delete the requirement to test the RCIC system daily until the HPCI system is returned to normal.

BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 4.5.D.1.b: The change is consistent with Item (i) of the "Examples" since it provides for syntax correction by adding a "continued" statement to the continuation of Technical Specification 4.5.D.1.b on page 3.5-7 of the Technical Specification.
2. Technical Specification 4.5.D.1.d: Deletion of the pump operability test requirements from the Technical Specification is a change consistent with Item (vi) of the "Examples" since it may appear that an existing margin of safety is being reduced slightly. However, the testing requirements will be covered by the requirements of the ASME Section XI code per 10 CFR 50.55a and at the Code-specified test frequency. Testing in this manner is still in keeping with the testing and inspection criteria of SRP section 9.6.3.

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3. Technical Specification 4.5.D.1.e: Deletion of the motor operated valve operability test requirements from the Technical Specifications is a change consistent with Item (vi) of the "Examples" since it may appear that an existing margin of safety is reduced slightly. However, the testing requirements will be covered by the requirements of the ASME Section XI Code per 10 CFR 50.55a and at the Code-specified test frequency. Testing in this manner is still in keeping with the testing and inspection criteria of SRP section 9.6.3.
4. Technical Specification 4.5.D.2: The removal of the requirement to verify immediate operability of the Core Spray, the RHR LPCI mode, and the RCIC system with HPCI inoperable is a change consistent with Item (vi) of the "Examples" since it may reduce a possible safety margin. Similarly, deletion of the requirement to test the RCIC system daily until the HPCI system is returned to normal if HPCI system components are inoperable may reduce a possible safety margin. A reduction in safety margin would occur if RCIC became unavailable in conjunction with the loss of HPCI, as well as, the low pressure ECCS capability of the redundantly configured Core Spray and LPCI systems. The redundant capabilities of Core Spray and LPCI covers postulated losses of low pressure cooling. The specific testing and inspection requirements introduced through 10 CFR 50.55a (which references the ASME Section XI Code) will help assure system operability. The change still remains within the criteria of SRP section 6.3.

PROPOSED CHANGES

Revise Technical Specification 4.5.E to reflect the following for the Reactor Core Isolation Cooling (RCIC) System:

1. Delete existing Technical Specification 4.5.E.1.d concerning pump operability testing and frequency thereof.
2. Delete existing Technical Specification 4.5.E.1.e concerning motor operated valve operability testing and frequency thereof.
3. Delete existing Technical Specification 4.5.E.2 concerning surveillance with inoperable components.

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BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 4.5.E.1.d: Deletion of the pump operability test requirements from the Technical Specifications is a change consistent with Item (vi) of the "Examples" since it may appear that an existing margin of safety is being reduced. However, the testing requirements will be covered by the requirements of the ASME Section XI Code per 10 CFR 50.55a and at the Code-specified test frequency. The change is still in keeping with the testing and inspection criteria of SRP section 5.4.6.
2. Technical Specification 4.5.E.1.e: Deletion of the motor operated valve operability testing requirements from Technical Specifications is a change consistent with Item (vi) of the "Examples" since it may appear that an existing margin of safety is being reduced. However, the testing requirements will be covered by the requirements of the ASME Section XI Code per 10 CFR 50.55a and at the Code-specified frequency. The change is still in keeping with the testing and inspection criteria of SRP section 5.4.6.
3. Technical Specification 4.5.E.2: The removal of the requirement to verify immediately and daily thereafter the proper operation of the HPCI system when RCIC is inoperable is a change consistent with Item (vi) of the "Examples" since it may slightly reduce a safety margin. However, this change is still in keeping with SRP section 5.4.6. HPCI, as an alternate or backup system to RCIC, is subjected to similar testing criteria as RCIC. Testing criteria are now based on 10 CFR 50.55a (as referenced in Technical Specification 4.6.K in the attached amendment) and provides adequate assurance of system availability in the event of RCIC inoperability.

PROPOSED CHANGE

In Technical Specification 4.5.G, remove the reference to testing immediately and daily thereafter the components of the RHR LPCI mode and containment cooling mode connected to the operable diesel generators, upon loss of one of the diesel generators.

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BASIS

The proposed change has been determined not to involve a significant hazard because:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident because:
 - a. The components under consideration are redundant and have separate sources of electrical power. Therefore, the operable diesels would be able to power at least one of the redundant systems in the event power is shifted to the diesel generators;
 - b. Actual availability of the components under consideration is verified through other Technical Specification surveillance requirements and/or through the requirements of 10 CFR 50.55a;
 - c. Single component failure in redundant safety systems has been evaluated; and,
 - d. The proposed change will still include the requirement for testing all remaining operable diesels daily until the inoperable unit is placed back in service.
2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the change does not modify the original intent of the surveillance requirement. The intent is to ensure adequate and reliable core and containment cooling capacity in the event that one source of backup power (i.e., one diesel generator) is lost. The change merely simplifies this intent by focusing attention on the problem source which is an inoperable diesel generator and taking the originally intended corrective measure of increased diesel testing frequency. Verification of operability of the systems omitted in this change is accomplished in other Technical Specification surveillance requirements and/or the ASME Section XI Code requirements which are referenced in 10 CFR 50.55a.
3. The proposed change does not involve a significant reduction in the margin of safety because of the adequate operability assurance provided for by other surveillance requirements and/or 10 CFR 50.55a for the omitted systems (i.e., RHR LPCI and Containment Cooling).

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PROPOSED CHANGES

Revise Technical Specification 3.5.J to reflect the following for the Plant Service Water System:

1. In Technical Specification 3.5.J.1, change the number of plant service water pumps from 4 to 3.
2. In Technical Specification 3.5.J.2.b, add wording such that the diesel generators are those associated with the operable Plant Service Water (PSW) components.
3. In Technical Specification 3.5.J.2.c, add wording such that the diesel generators are those associated with the operable PSW components.

BASIS

The following changes as identified above are listed by Technical Specification section number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 3.5.J.1: The change is consistent with item (i) of the "Examples" since it corrects the number of plant service water pumps.
2. Technical Specification 3.5.J.2.b: The change is consistent with Item (i) of the "Examples" since it was made to provide consistency with other sections of the Technical Specifications which are concerned with equipment that can be diesel powered. This section references operability of only those particular diesels which can power the equipment in question.
3. Technical Specification 3.5.J.2.c: The change is consistent with Item (i) of the "Examples" since it was made to provide consistency with other sections of the Technical Specifications which are concerned with equipment that can be diesel powered. This section references operability of only those particular diesels which can power the equipment in question.

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PROPOSED CHANGES

Revise Technical Specification 4.5.J to reflect the following for the Plant Service Water System:

1. In Technical Specification 4.5.J.2.b, delete the requirements to demonstrate operability of the standby service water pump, the remaining PSW pumps, and both PSW divisions when one PSW pump is made or found to be inoperable.
2. In Technical Specification 4.5.J.2.b, add wording such that the diesel generators are those associated with the operable PSW components.
3. In Technical Specification 4.5.J.2.c, delete the requirement to demonstrate operability of the remaining PSW pumps and both PSW divisions when one PSW pump and the standby service water pump are made or found to be inoperable.
4. In Technical Specification 4.5.J.2.c, add wording such that the diesel generators are those associated with the operable PSW components.
5. In Technical Specifications 4.5.J.2.d, delete the requirement to demonstrate operability of the standby service water pump and all active components of the operable PSW division(s) when two PSW pumps or one PSW division are made or found to be inoperable.
6. In Technical Specification 4.5.J.2.e, delete the requirements to demonstrate operability of all active components of the operable PSW division(s) when two PSW pumps or one PSW division and the standby service water pump are made or found to be inoperable.

BASIS

The following changes as identified above are listed by Technical Specification number. These changes are consistent with the appropriate Item number of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Consideration" as listed on page 14870 of the April 6, 1983 issue of the Federal Register for the reasons given below:

1. Technical Specification 4.5.J.2.b: Deletion of the requirement to demonstrate operability of the standby service water pump, the remaining PSW pumps, and both PSW divisions immediately and weekly thereafter when one PSW pump is made or found to be inoperable is a change consistent with Item (vi) of the "Examples" since it may

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BASIS (Continued)

possibly reduce an existing safety margin if failure of the remaining components is considered. Considering the redundancy involved, significant failure of the remaining components is a very small possibility. In addition, testing and inspection requirements resulting from 10 CFR 50.55a adds an extra assurance of component reliability. Based on these considerations and the fact that the change is still within SRP criteria (i.e., SRP section 9.2.1), a significant hazard is not applicable.

2. Technical Specification 4.5.J.2.b: The addition of wording such that the diesel generators tested are those associated with the operable PSW components is a change which is consistent with Item (i) of the "Examples" since it was made to provide consistency with other sections of the Technical Specifications which are concerned with equipment that can be diesel powered. This section references operability of only those particular diesels which can power the equipment in question.
3. Technical Specification 4.5.J.2.c: Deletion of the requirement to demonstrate operability of the remaining PSW pumps and both PSW divisions immediately and weekly thereafter when one PSW pump and the standby service water pump are made or found to be inoperable is a change consistent with Item (vi) of the "Examples" since it may possibly reduce an existing safety margin if failure of the redundant components is considered. Considering the redundancy involved, significant failure of the remaining components is a very small possibility. In addition, testing and inspection requirements resulting from 10 CFR 50.55a adds an extra assurance of component reliability. Based on these considerations and the fact that the change is still within SRP criteria (i.e., SRP section 9.2.1), a significant hazard is not applicable.
4. Technical Specification 4.5.J.2.c: The addition of wording such that the diesel generators tested are those associated with the operable PSW components is a change which is consistent with Item (i) of the "Examples" since it was made to provide consistency with other sections of the Technical Specifications which are concerned with equipment that can be diesel powered. This section references operability of only those particular diesels which can power the equipment in question.

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5. Technical Specification 4.5.J.2.d: Deletion of the requirements to demonstrate operability of the standby service water pump and all active components in the operable PSW division(s) immediately and daily thereafter when two PSW pumps or one PSW division are made or found to be inoperable is a change consistent with Item (vi) of the "Examples" since it may possibly reduce an existing safety margin if failure of the remaining components is considered. Considering the redundancy involved, significant failure of the remaining components is a very small possibility. In addition, testing and inspection requirements resulting from 10 CFR 50.55a adds an extra assurance of component reliability. Based on these considerations and the fact that the change is still within SRP criteria (i.e., SRP section 9.2.1), a significant hazard is not applicable.
6. Technical Specification 4.5.J.2.e: Deletion of the requirements demonstrate operability of all active components of the operable PSW division(s) immediately and daily thereafter when two PSW pumps or one PSW division and the standby service water pump are made or found to be inoperable is a change consistent with Item (vi) of the "Examples" since it may possibly reduce an existing safety margin if failure of the remaining components is considered. Considering the redundancy involved, significant failure of the remaining components is a very small possibility. In addition, testing and inspection requirements resulting from 10 CFR 50.55a adds an extra assurance of component reliability. Based on these considerations and the fact that the change is still within SRP criteria (i.e., SRP section 9.2.1), a significant hazard is not applicable.

PROPOSED CHANGE

Change the "Bases" section of Technical Specifications to reflect the following:

1. In Section 3.5.A.2 relative to the Core Spray System, delete the wording concerning testing of the remaining Core Spray loop and the RHR system should one Core Spray loop become inoperable;
2. In Section 3.5.A.2, add wording such that the diesel generators tested are those associated with the remaining operable Core Spray loop should one Core Spray loop become inoperable; and,
3. In Section 3.5.A.2, add references to inservice inspection/in-service testing criteria.

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BASIS

The above changes to the "Bases" section of Technical Specifications for the Core Spray System are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since the changes are either administrative in nature or are provided for consistency with other Technical Specifications, existing and/or proposed by the attached amendment.

PROPOSED CHANGES

Change the "Bases" section of Technical Specifications to reflect the following:

1. In Section 3.5.B.2 relative to the RHR System, revise core flooding assurance statement to delete statements concerning demonstrated operability of the redundant LPCI pumps and subsystems and the Core Spray system. In addition, remove statement concerning out-of-service period; and,
2. In section 3.5.B.2, add references to inservice inspection/in-service testing criteria.

BASIS

The above changes to the "Bases" section of Technical Specifications for the RHR System (LPCI and Containment Cooling Mode) are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since the changes are either administrative in nature or are provided for consistency with other Technical Specifications, existing and/or proposed by the attached amendment.

PROPOSED CHANGE

Change Technical Specification "Bases" section 3.5.D.2 relative to the HPCI System to reference inservice inspection/in-service testing criteria for the backup RCIC System.

BASIS

The change to the subject "Bases" section is a change consistent with Item (ii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since it provides an additional control not presently in the Technical Specifications.

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PROPOSED CHANGE

Change Technical Specification "Bases" section 3.5.E.2 relative to the RCIC System to reference inservice inspection/inservice testing criteria for the HPCI System.

BASIS

The change to the subject "Bases" section is a change consistent with Item (ii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since it provides an additional control not presently in the Technical Specifications.

PROPOSED CHANGES

Change Technical Specification "Bases" section 3.5.F.1 relative to ADS to reflect the following:

1. Change the title entry "F.1" to read "3.5.F.1"; and
2. Correct the spelling of the word "failures".

BASIS

The above changes are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since the changes are either administrative in nature or correct a typographical error in the Technical Specifications.

PROPOSED CHANGES

Change Technical Specifications "Bases" section 3.5.F.2 relative to ADS to reference inservice inspection/inservice testing criteria for the HPCI System.

BASIS

The change is consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since it provides consistency with other Technical Specifications changes concerning HPCI requested in the attached amendment.

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PROPOSED CHANGE

Change Technical Specification "Bases" section 3.5.J/4.5.J relative to the Plant Service Water System to reference inservice inspection/inservice testing criteria to provide increased assurance of PSW System operability.

BASIS

The change to the subject "Bases" section is a change consistent with Item (ii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since it inserts additional criteria to provide increased assurance of PSW System operability. Therefore, it constitutes an additional control not presently included in the Technical Specifications.

PROPOSED CHANGES

Relocate existing Technical Specifications 4.6.I.2, 4.6.I.3, 3.6.J.1, 3.6.J.2, and 4.6.J from Technical Specifications page 3.6-10 to page 3.6-9b and relocate Technical Specification 3.6.J.3 from Technical Specifications page 3.6-10 to new page 3.6-9c.

BASIS

The changes are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register since they are administrative in nature to accommodate new proposed wording for Technical Specifications 3.6.K/4.6.k on page 3.6-10 of the Technical Specifications.

PROPOSED CHANGES

Delete in their entirety the existing Technical Specifications 3.6.K/4.6.K and substitute new proposed wording for the subject Technical Specifications similar to that found in Standard Technical Specifications.

BASIS

This change is consistent with Item (vii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register due to changes in inservice inspection/inservice testing requirements resulting from 10 CFR 50.55a. The new proposed wording is

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BASIS (Continued)

similar to that in Standard Technical Specifications and supercedes the existing outdated Technical Specification requirements. The existing Technical Specifications surveillance requirements for inservice inspection were for a program written to the 1971 Edition of the ASME Section XI Code with Addenda through Summer 1972. Pursuant to the requirements of 10 CFR 50.55a, the licensee is required to periodically update the inspection program to a later edition of the Code and the requirements it sets forth. The existing Technical Specifications do not reflect the current inspection requirements. Further, deletion of the existing Technical Specifications is justified since some of the requirements (e.g, reporting of inspection results after 5 years presumably from the commercial service date upon which the inservice inspection is based) have already been met and, thus, are antiquated. The new proposed Technical Specifications are written such that they will not require revision each time the program is revised per 10 CFR 50.55a since no specific edition/addenda of the Code is referenced in the proposed Technical Specifications.

PROPOSED CHANGE

Delete in its entirety Technical Specification Table 4.6-1, "Inservice Inspection Program".

BASIS

The change is consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register since deletion of Table 4.6-1 provides consistency with the proposed deletion of outdated Technical Specifications existing in Technical Specifications 3.6.K/4.6.K. New proposed wording for Technical Specifications 3.6.K/4.6.K are provided in the attached amendment. The subject table does not reflect the current inspection program as provided for by the requirements of 10 CFR 50.55a which requires program update periodically.

PROPOSED CHANGE

Revise Technical Specifications "Bases" section 3.6.K to reflect the current inservice inspection/in-service testing requirements per 10 CFR 50.55a and to delete any outdated material currently existing.

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BASIS

The change is consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register to provide consistency with the new wording proposed for Technical Specifications 3.6.K/4.6.K as shown in the attached amendment. The changes are as a result of applying criteria set forth by the current requirements of 10 CFR 50.55a.

PROPOSED CHANGES

Add "less than" symbols relative to calm wind speed to Technical Specifications 4.7.C.1.b and 4.7.C.2.

BASIS

The changes are consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" as listed on page 14870 of the April 6, 1983 issue of the Federal Register since the changes correct omission of the "less than" symbols from the last Technical Specification amendment (i.e., Amendment 100) to page 3.7-13 of the Technical Specifications.