PSNH PUBLIC SERVICE
Company of New Hampshire

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September 21, 1982

SBN - 329
T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. Janis B. Kerrigan, Acting Chief
Licensing Branch 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC, Letter, dated April 26, 1982, "Request for Additional Information - Seabrook Station, Units 1 & 2", F. J. Miraglia to W. C. Tallman

Subject: Response to 440 Series RAIs; (Reactor Systems Branch)

Dear Ms. Kerrigan:

We have enclosed responses to the following Requests for Additional Informat· n (RAIs) which you forwarded in Reference (b):

440.105, 440.107, 440.108, 440.110, 440.111, 440.112, 440.113, 440.114, 440.117, 440.121, 440.123, 440.124, 440.125, 440.127, 440.128, 440.129, 440.131, and 440.135.

The outstanding RAIs are:

440.115, 440.118, 440.132, 440.133, and 440.134.

Responses to the outstanding RAIs will be submitted in the near future.

Please note that responses to the enclosed RAIs had been previously fc.warded (informally) to the reviewer on July 22, 1982.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis

John DeVincentis
Project Manager

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440.105 Per the requirements of BTP RSB 5-2, events that are excluded from the analysis, the controls to prevent these events should be included in the Tech. Specs. Address compliance with this requirement.

RESPONSE: Each Technical Specification pertinent to protection against cold overpressurization events is discussed below and attached to this response for easy reference. Changes or additions to the current specifications are identified by vertical bars in the right-hand margin.

Technical Specification 3.1.2.2 requires only one operable boron injection flow path whenever the temperature of one or more of the RCS cold legs is $\leq 305^{o}F$ in mode 4. This permits compliance with Technical Specification 3.1.2.4 which requires a maximum of one operable centrifugal charging pump under this condition.

Technical Specification 3.1.2.3 requires a maximum of one operable charging pump in mode 5. This provides assurance that a mass addition transient can be relieved by the operation of a single PORV.

Technical Specification 3.1.2.4 requires a maximum of one operable centrifugal charging pump whenever the temperature of one or more of the RCS cold legs is $\leq 305^{o}F$ in mode 4. This provides assurance that a mass addition transient can be relieved by the operation of a single PORV.

Technical Specification 3.4.1.3 limits the startup of a Reactor Coolant Pump (RCP) when the RCS temperature is $\leq 305^{o}F$ in mode 4. Technical Specification 3.4.1.4 limits the startup of an RCP in mode 5. This provides assurance that a heat addition transient can be relieved by the operation of a single PORV.

Technical Specification 3.4.10.3 requires two operable PORV's when the RCS temperature is $\leq 305^{o}F$. Alternatively, the RCS may be depressurized and vented through an equivalent opening. The required PORV setpoint ensures that the RCS is protected from pressure transients which could exceed the limits of Appendix G to 10CFR50. Either PORV has sufficient capacity when the transient is limited to either (1) the start of an idle RCP constrained by Technical Specifications 3.4.1.3 and 3.4.1.4 or (2) the start of an SI or charging pump when the RCS is water solid.

Technical Specification 3.5.1.2 requires isolation of each accumulator when the RCS temperature is below $305^{o}F$. This assures that an overpressure transient will not be caused by mass addition from an accumulator.

Technical Specification 3.5.3 requires a maximum of one operable centrifugal charging pump when the temperature of the RCS is $\leq 305^{o}F$. This limits the number of operable ECCS pumps to assure that a mass addition transient can be relieved by the operation of a single PORV.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

c. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

Only one boron injection flow path is required to be OPERABLE whenever the
305 temperature of one or more of the RCS cold legs is less than or equal to 275°F. If this path is inoperable, the above ACTION statement shall apply except that only one flow path need be restored to OPERABLE status within the stated time limits.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification (3.1.2.1) shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to (later) psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours, except when the reactor vessel head is removed, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

Note: See also Section 3.5.3 ACTION a.

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to (later) psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to
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275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 275°F. If this pump is inoperable, the above ACTION statement shall apply except that only one charging pump need be restored to OPERABLE status within the stated time limits.

REACTIVITY CONTROL SYSTEMS

BASES

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

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The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 275°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either ~~2000~~ later gallons of 7000 ppm borated water from the boric acid storage tanks or ~~9600~~ later gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH ban~~k~~d minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that 1) acceptable power distribution limits are maintained, 2) the minimum SHUTDOWN MARGIN is maintained, and 3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop (D) and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal Loop (A),
6. Residual Heat Removal Loop (B),

b. At least one of the above coolant loops shall be in operation.**

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 275°F unless 1) the pressurizer water volume is less than 1600 cubic feet or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiative corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The residual heat removal loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

Four filled and intact reactor coolant loops may be substituted for one residual heat removal loop. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 275°F unless 1) the pressurizer water volume is less than 1600 cubic feet or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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* The normal or emergency power source may be inoperable.

** The residual heat removal pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

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The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

. Two power operated relief valves (PORVs) with a lift setting of less than or equal to ~~450 psig, or~~ the maximum setpoint defined by Figure 3.4-4, or

b. A reactor coolant system vent of greater than or equal to(later) square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 275 305°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a(later)square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a(later)square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any correction action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

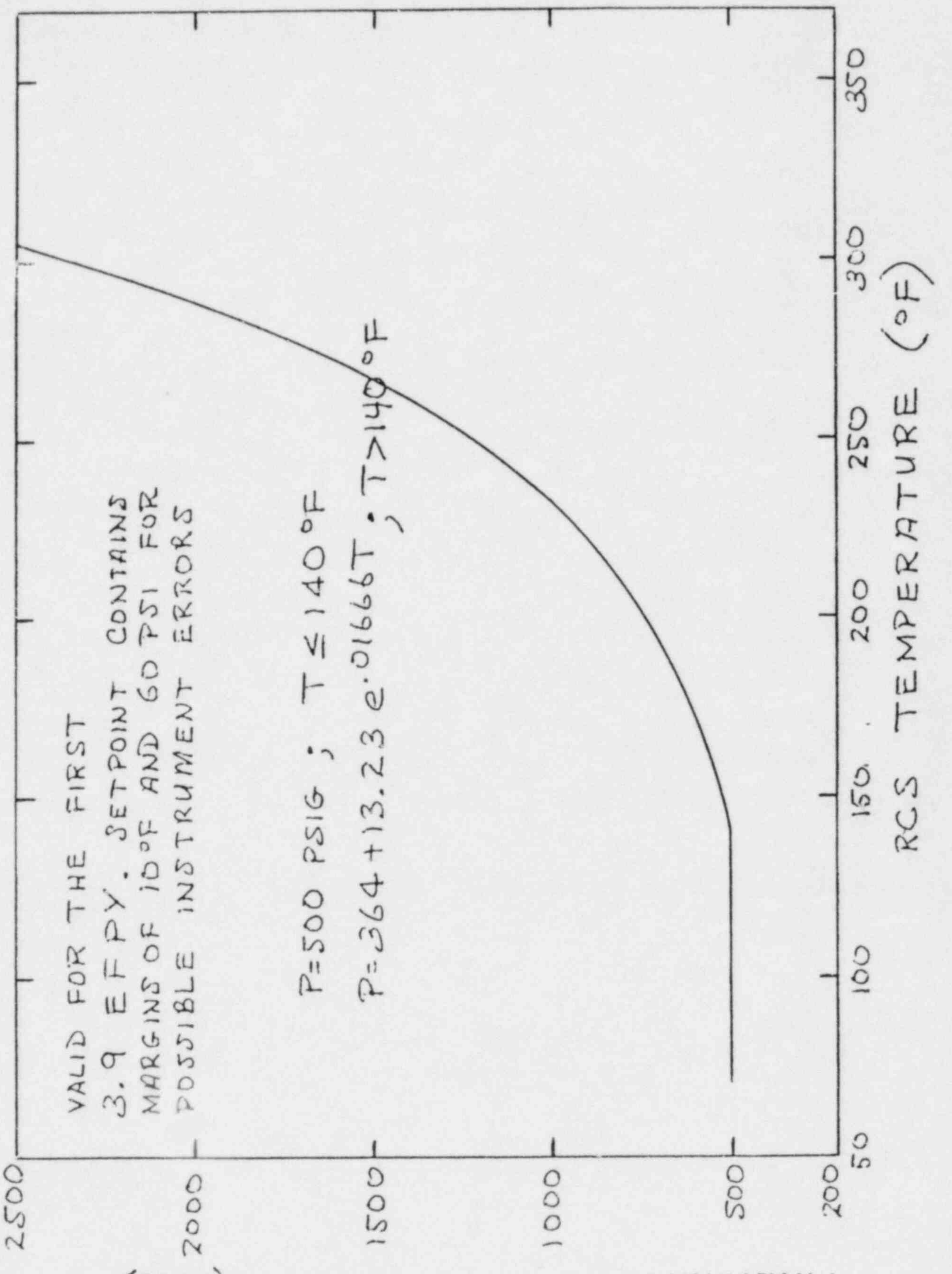

FIGURE 3.4-4

RCS COLD OVERPRESSURE PROTECTION SETPOINTS

REACTOR COOLANT SYSTEM

BASES

later

The OPERABILITY of two PORVs or an RCS vent opening of greater than ( ) square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to (275)°F. 305 Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a HPSI or charging pump and its injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, and Addenda through Summer 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between (later) and (later) gallons,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between (later) and (later) psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within one hour and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

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* Pressurizer pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to (1% of tank volume) by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig, by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:

1. When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) setpoint,

2. Upon receipt of a safety injection test signal.

4.5.1.1.2
~~4.5.1.2~~ Each accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of a CHANNEL FUNCTIONAL TEST.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each reactor coolant system accumulator isolation valve shall be shut with power removed from the valve operator.

APPLICABILITY: MODES 4* and 5

ACTION:

With one or more accumulator isolation valve(s) open and/or power available to the valve operator(s) , immediately close the accumulator isolation valve and/or remove power from the valve operator(s).

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each accumulator isolation valve will be verified shut with power removed from the valve operator at least once per 31 days.

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* When one or more of the RCS cold legs is less than or equal to 305° F.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank ~~upon~~ and capable of being manually realigned ~~and transferring~~ to transfer the suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump ~~and one safety injection pump~~ shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to (275)°F.
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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 275°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

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EMERGENCY CORE COOLING SYSTEMS

BASES

The limitation for a maximum of one centrifugal charging pump ~~and one safety injection pump~~ to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable below 275°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.
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The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 22,500 ppm boron.

440.107 Provide a discussion for the case where the initiating event is assumed to be a DC bus failure which results in isolation of the letdown flow path. If a charging pump is operating, this isolation of letdown could initiate an overpressure event. Could this DC bus failure also disable PORV's?

RESPONSE: The Seabrook low temperature overpressure mitigation system utilizes redundant safety grade pressurizer PORV's to mitigate an LTOP event. These PORV's are powered by separate dc power sources, therefore, a single failure resulting in the loss of one dc bus would not disable both PORV's. Only one PORV is required for relief protection to mitigate the overpressure event resulting from letdown isolation and an operating charging pump.

The control circuitry for the LTOP system is also designed against single failure of both PORV's by loss of a dc power source. Additional design changes are presently being incorporated into the LTOP circuitry to resolve a problem of single failure within the auctioneering circuits. Under the new design separate auctioneering circuits will be provided for both the arming and actuating signals for each train.

Therefore, with the new design, no single failure in the PORV power supply, the LTOP circuitry power supply, or the LTOP circuitry itself will disable the automatic opening of both PORV's.

440.108 Your response to Q440.7 does not address the point of assuming reactor trip comes from the <u>second</u> safety grade trip signal. Address this issue.

RESPONSE: See the response providing supplemental information to RAI 440.5.

440.110 Your response to Q440.13 is not sufficient. Provide a response which demonstrates that:

1. You have the capability to achieve cold shutdown utilizing <u>only</u> safety grade equipment.

2. Cold shutdown (normal) can be achieved from the control room.

3. Adequate leak detection and isolation capabilities are provided.

RESPONSE: The response to this RAI is provided in the response to RAI 440.133 where each requirement of BTP RSB 5-1 is addressed.

440.111 Your response to Q440.23 is not sufficient. The staff position is that all actions to achieve cold shutdown must be performed from the control room. Your response did not address the accumulator isolation valves or charging pump circuit breakers.

RESPONSE: When RCS pressure has decreased below 1000 psig during plant cooldown, the SI accumulator isolation valves are closed to prevent injection of the accumulators' volume into the RCS as RCS pressure is reduced. This action involves energizing the MCC's powering the accumulators' MOV and then closing the valves. These actions are all performed in the control room.

Should a single failure disable the power supply to one or more of the SI accumulator isolation valves, solenoid operated vents will be provided on each SI accumulator to allow relieving of the nitrogen overpressure gas to the containment. These solenoids will be Class 1E, powered by the emergency electrical train opposite that powering the SI accumulator isolation valve, and will be operable from the control room.

Additionally, during plant cooldown, one or more of the centrifugal charging pumps and both SI pumps will be made inoperable to preclude overpressurization events at low temperatures. This action can also be performed in the control room.

440.112 Concerning your response to Q440.25, provide additional information describing construction of the RWST enclosure (is the enclosure heated?, insulated?). Discuss steam heater capacity and basis and vent line location.

RESPONSE: FSAR Subsection 6.2.2.3 will be revised to provide additional information demonstrating that freezing temperatures will not occur. See the attached sheets for the revised wording.

RESPONSE TO RAI #440.112

to maintain the maximum containment pressure during an accident as low as practical while keeping the setpoint as high as practical to minimize the probability of spray actuation following a small high energy line break. The engineered safety features actuation system is further described in Section 7.3.

o. Equipment Qualification

Components in the CBS system which are required to function during the accident are qualified (vendor certification) to verify the ability of the components to perform their intended functions under the conditions specified in the purchase documents and/or by test. Environmental qualification of safety-related equipment is discussed in Section 3.11. Tests and inspections are discussed in Subsection 6.2.2.4. Seismic qualification is addressed in Section 3.10. Pump and valve operability assurance is discussed in Subsection 3.9.3.2.

p. Containment Spray System Response Time

Containment spray system response time is discussed in Subsection 6.2.1.1.c.

6.2.2.3 Design Evaluation

The analyses of the post-accident containment pressure transients are discussed in Subsection 6.2.1. The double train concept insures that sufficient heat removal capacity will exist, even with a single active failure. Containment design pressure is not exceeded and containment pressure reduction reduces containment leakage to 50% of the design leak rate within 24 hours after the DBA.

~~The refueling water storage tank and the spray additive tank are located in the yard. During cold weather conditions, external steam heating coils protect against freezing. The maximum temperature of the water in the tanks is calculated to be 86°F. The heat of reaction upon the mixing of the boric acid and the sodium hydroxide would raise the temperature of the tank contents approximately 2°F thereby raising the RWST maximum supply temperature to 88°F. Neither tank is protected against tornado missiles and a tornado and accident are not considered simultaneous events. In the event of tornado damage to either tank, the affected unit would be shut down.~~

REPLACE WITH THE REVISED WORDING ON THE FOLLOWING PAGE.

The SAT is connected to the RWST by two parallel lines each with an automatic motor operated valve. The valves are actuated and powered from separate sources to insure that the NaOH solution can be added to the containment spray even in the event of a single active failure.

The method of addition of 20% NaOH solution in required concentrations to the borated water drawn from the RWST immediately following a LOCA is primarily dependent on passive components, such as tanks, pipes and a

RAI 440.112 - REVISED WORDING FOR FSAR PAGE 6.2-48

The refueling water storage tanks and the spray additive tanks for Units 1 and 2 are located within enclosure buildings in the yard area. For Unit 1, the RWST and SAT are fully enclosed with insulated siding and roof, as well as by two heated buildings (PAB and WPB). Included within these enclosures is the associated piping, vent lines, and instrument tubing.

During cold weather conditions, both the SAT and RWST are heated by steam heating panels mounted on the exterior surface of the tanks. Calculations for Unit 1 demonstrate that the RWST heating panel can maintain a minimum water tank temperature of 50°F, and concurrently provide sufficient heat into the enclosure area to maintain an enclosure temperature of 39°F. No credit was taken for heat contributions from the SAT heaters. The site environmental condition for this design evaluation assumed -17°F and 30 mph winds, and are more conservative than the minimum outdoor conditions listed in FSAR Figure 3.11(B)-1.

For the above environmental conditions, the heat loss from the enclosure building, including infiltration losses, is 158,000 BTU/hr, as compared to an RWST heating panel capacity of 674,000 BTU/hr. Accordingly, freeze protection is provided for all equipment. In addition, both tank low temperature and enclosure temperature alarms are provided in the main control room.

For Unit 2, the building geometry is different, and a separate analysis is being performed to verify freeze protection.

The maximum temperature of the water in the tanks is calculated to be 86°F, using plant specific meteorological data, assuming maximum solar heat gain and failure of the ventilation fans. The heat of reaction upon the mixing of the boric acid and the sodium hydroxide would raise the temperature of the tank contents approximately 2°F thereby raising the RWST maximum supply temperature to 88°F. Neither tank is protected against tornado missiles, and a tornado and accident are not considered simultaneous events. In the event of tornado damage to either tank, the affected unit would be shut down.

440.113 Your response to Q440.26 is not sufficient, provide justification and supporting analysis for the statement that adequate cooling capability can be maintained via the charging pumps and RHR pumps without taking credit of safety injection pumps or confirm that valve No. SIV93 will be normally locked open with position indications and alarms provided in control room.

RESPONSE: SI-V93, the combined recirculation isolation valve from both safety injection pumps, is a normally open, motor-operated valve. This valve is closed by the operators, from the control room, during the switchover to the recirculation mode of safety injection. This valve will be locked in the open position locally at the valve to prevent manual mis-positioning.

Red/green valve position indication and valve full-closed monitor light is provided on the main control board. Additionally, any time SI-V93 leaves the full open position, an annunciator alarms for both the "SI Train A Inoperative" and the "SI Train B Inoperative" status monitoring alarms.

To prevent spurious operation or operator error, the control circuit for the motor operator is equipped with a dual contactor arrangement (see FSAR Figure 8.3-45). This circuit requires two separate operator actions, involving the normal valve control switch plus a separate key-operated switch, to re-position the valve.

440.114 Your response to Q440.29 indicates that 32 gpm corresponds to 4.3 SCFM. Utilizing this correlation the 1" relief valve would have a capacity in excess of 11,000 gpm. Provide the estimated water relief capacity of the accumulator relief valves and justify the assumed fluid temperature for this calculation.

RESPONSE: RAI 440.29 requested information relative to the adequacy of the SI accumulator relief valves to prevent overpressurization due to postulated RCS back leakage or during level adjustments. As stated in the response to RAI 440.29, the maximum fill rate for the accumulator at the relief valve setpoint pressure is 32 gpm. However, this was erroneously equated to 4.3 scfm. Assuming isothermal conditions, an input of 32 gpm into the SI accumulator, at an accumulator pressure of 700 psig, results in a gas flow rate of approximately 209 scfm through the relief valve. This is considered negligible compared to the relief valve capacity of 1500 scfm. Since the design transient is the case of maximum nitrogen make-up to the accumulator, a coincident water fill operation has a very small effect on the relief valve capacity.

Water relief is not a design basis for the accumulator relief valves. Accumulator relief valves are procured to specifications requiring certain gas relieving capacities at certain temperatures. Those temperatures are 60° - 120°F. If these valves had to relieve water, it is expected that they would do so at rates of from 50 to 150 gpm at temperatures of 60° to 120°F.

440.117 Your response to Q440.48 is not sufficient. Provide justification for the statement that "loss of the safety injection pump would not degrade ECCS operation to an unacceptable level" or confirm that the common header of the miniflow lines for two safety injection pumps will be designed to maintain its integrity after a postulated SSE) or confirm that a single failure in the combined miniflow line (failing closed of the single isolation valve as well as integrity after a postulated SSE) will not result in damage to the two safety injection pumps.

RESPONSE: The statement that the loss of SI pumps would not degrade ECCS operation to an unacceptable level is essentially correct in the context of the response to Q440.48, part b. The SI pumps would not be available if miniflow isolation were necessary when the RCS pressure is above the SI pumps discharge head. Small break LOCA's with breaks less than or equal to 3/8" equivalent diameter will not result in RCS depressurization below the SI pump shutoff head, and the SI pumps would have to be stopped in the event that the non-seismic miniflow header failed. However, small LOCA's with an equivalent diameter of 3/8" or less are considered leaks and the charging flow from the volume control system is capable of replacing the lost inventory. Small LOCA's with equivalent break diameters larger than 3/8" will result in RCS depressurization below the SI pump shutoff head and avert the need to stop the SI pumps. Hence, ECCS operation would not be impaired for LOCA's.

440.121 Provide justification for your assumed core flow during a major steam line break. This is a Standard Review Plan (SRP 15.1.5) Acceptance Criteria requirement (6g).

RESPONSE: Two cases are analyzed in the FSAR; one with full flow, another with a flow coastdown. Variation in initial flow rate is addressed in WCAP-9226 (Steam Line Break Topical) Section 3.1.1.8.

QUESTION 440.123

Your response to Q440.57 is not sufficient. Provide a table listing components required, response time, capacity and test provisions for each event analyzed in Chapter 15 of the FSAR.

RESPONSE:

The valves presented in Table 440.57-1 (Table attached) are those valves which were used in all accident analyses, providing credit for the valves is taken. The use (non-use) of these valves is discussed for each accident in its respective section.

TABLE 440.57-1

COMPONENT	RESPONSE TIME	CAPACITY	TEST PROVISIONS
Main Steamline Isolation Valves	2 sec. logic and delay 5 sec. closure	- -	See Table 14.2-3 item 13
Main Feedwater Isolation Valves	2 sec. logic and delay 5 sec. closure	- -	See Table 14.2-3 item 14
Pressurizer Power Operated Relief Valves	-	2 Valves @ 210000 lbm/hr	See Table 14.2-3 item 2
Pressurizer Safety Valves	-	3 Valves @ 420000 lbm/hr	See Table 14.2-3 item 40
Steam Generator Safety Valves	-	120% of rated full power steam flow (rated flow = 15.14×10^6 lbm/hr)	See Table 14.2-3 item 40
* Emergency Feedwater	60 second delay with or without offsite power	Feedline rupture - 470 gpm to two intact steam generators Loss of feedwater w/ AC 650 gpm to all steam generators Loss of feedwater w/o AC 470 gpm to all steam generators	See Table 14.2-3 item 14

* For Steamline Rupture, see response to Question 440.70

440.124 Concerning your response to Q440.70, there has been no response to Item 6, "alarms and indications provided to assist the operator in determining the correct course of action". Provide the required response.

RESPONSE: As stated in the response to Question 440.70, only one operator action is assumed in the steam line rupture analysis. The alarms and indications available to the operator include:

a) high steam pressure rate;

b) low steam generator level;

c) low steam pressure;

d) low pressurizer pressure;

e) high containment pressure, and;

f) low pressurizer level.

See also the Westinghouse Owners Group emergency operating procedures.

Question 440.125

The response to Q440.64 is inadequate as you still have not supplied the assumed single failure for feedwater temperature reduction, excessive steam flow, misaligned RCCA or small break containing RCS water outside of containment events. Provide justification that the assumed single failure are the worst possible failures.

Response:

As stated in the FSAR, the feedwater temperature reduction analysis is bounded by the analysis for excessive steam flow (Section 15.1.3). For these two accidents, no protective action by the protection system is required. Therefore, assume there is no limiting single failure in the protection system and any failure which could occur would have no impact. For RCCA misalignment and small line breaks outside containment, no transient analysis is performed. Again, the protection system is not required to mitigate the consequences of the accidents and there is no limiting failure which would impact the results.

440.127 Your response to Q440.89 did not address the assumed worst single failure, % fuel failure assumed, and resulting off-site release. Due to the fact that this is the limiting transient for this category a complete analysis should be supplied in the FSAR.

RESPONSE: The worst single failure for the locked rotor/shaft break accident is the loss of one protection train. This failure has no impact since there are two trains.

There is no fuel failure predicted for the locked rotor/shaft break accident. Although some rods may enter DNB for a short period of time, fuel failure for the locked rotor/shaft break incident is treated on a mechanistic basis, thus eliminating the need to automatically equate DNB with fuel rod failure.* In this method, the transient time-temperature history of the fuel rod is compared with oxidized clad failure as given in Reference 1. In addition, the evaluation procedure considers other potential fuel rod failure mechanisms such as: fuel pellet melting, clad collapse or ballooning/bursting.

Thus, the radiological consequences of the locked rotor accident are bounded by the results of the conservative analysis shown in FSAR Section 15.3.3.3. Therefore, the off-site doses are well within 10CFR100 limits. The complete analysis for this transient (excluding the above information) containing the assumptions, method of analysis, results and figures is provided in the response to Q440.89.

* The development and use of such mechanistic methods for quantifying fuel failure is specifically allowed by Section 4.2, of the NRC's Standard Review Plan (NUREG-0800).

Reference 1

R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout", NUREG-0562, Office of Nuclear Regulatory Research, USNRC, Washington, D. C. 20555 (June 1979)

440.128 The response to 440.93 did not address each of the valves as required. Provide the following information for each of the valves listed. Type, function, method of lockout, and effects and consequences of single failures or operator errors. The BTRS as the only source of dilution flow was not justified. (See your response to 440.94.) Provide this justification. Your response also indicated several additional valves which are not listed in the FSAR. Address this inconsistency.

RESPONSE: The following describes the valves listed in FSAR Section 15.4.6.2:

Valve #	Type	Function
FCV-110B	2" normally closed, air-operated diaphragm valve	Flow control outlet valve from the Boric Acid Blender to the Charging Pumps Suction (FSAR Figure 9.3-16).
FCV-111B	2" normally closed, air-operated diaphragm valve	Flow control outlet valve from the Boric Acid Blender to the Volume Control Tank (FSAR Figure 9.3-16).
RMW-V36	2" normally closed, manually-operated globe valve	Reactor Makeup Water Pump discharge to Charging Pumps Suction (FSAR Figure 9.3-16).
RMW-V39	3/4" normally closed, manually-operated diaphragm valve	Reactor Makeup Water system supply to the CVCS Chemical Mix Tank, CS-TK-2 (FSAR Figure 9.3-14).
CS-V452	1" normally closed, manually-operated globe valve	Boric Acid Transfer Pump discharge to Charging Pumps Suction (FSAR Figure 9.3-16).

These valves, when closed, all prevent flow of non-borated water from the Reactor Makeup Water System into the Reactor Coolant System.

However, in the preparation of the response to RAI 440.93, it was recognized that the use of these specific valves for isolation purposes might not be the most practical. For instance: it is more difficult to physically lock closed an air-operated valve than a manual valve with a handwheel. Additionally, it was recognized that other manual valves, which provide the same isolation function, might be more appropriate to lock closed from an operational standpoint.

The valves indicated in the response to RAI 440.93 perform similar isolation functions as those listed in FSAR Section 15.4.6.2, but are considered more appropriate for administrative controls. The FSAR will be revised to reflect this change.

The following describes the additional valves presented in the response to RAI 440.93, but not identified in FSAR Section 15.4.6.2:

Valve #	Type	Function
RMW-V34	2" normally open, manually-operated diaphragm valve	Reactor Makeup Water system supply to the Boric Acid Blender (FSAR Figure 9.3-16).
CS-V438	3" normally open, manually-operated diaphragm valve	Boric Acid Batch Tank, CS-TK-5, outlet to the Boric Acid Tanks, CS-TK-4A&B (FSAR Figure 9.3-16).

The method of lockout for all these valves involves the use of physical locking devices to prevent inadvertent operation along with administrative tagging controls.

The effects and consequences of single failure or operator error associated with these valves, could possibly result in the addition of non-borated water to the Reactor Coolant System. However, due to administrative controls during refueling which lockout one of the Reactor Makeup Water Pumps, the resulting dilution rate would be less than that assumed in the FSAR analysis.

During refueling, administrative controls such as valve and pump lockout and administrative tagging prevent an uncontrolled dilution of the Reactor Coolant System. The possible dilution sources were identified in the response to RAI 440.94. The most limiting source of dilution water is from the BTRS. For conservatism in considering the consequences of a single failure or operator error which would initiate a boron dilution event, the most limiting source (the BTRS) was assumed in the analysis.

440.129 Provide clarification for the definition of a small break LOCA. There is a discrepancy in the size of a small break LOCA between Section 6.3.3 and Section 15.6.5 of the FSAR.

RESPONSE: A small break LOCA is defined as a rupture of the reactor coolant pressure boundary with a total cross sectional area less than 1.0 ft^2. The area associated with a large break is equal to or greater than 1.0 ft^2. Page 6.3-21 of the FSAR will be revised per the attachment. Both small and large break sizes will be revised. The Chapter 15 references to break sizes are correct.

The small break analyses deal with breaks of up to ~~0.5~~ 1.0 ft^2 in area, where the safety injection pumps play an important role in the initial core recovery because of the slower depressurization of the RCS.

The RCS depressurization and water level transients show that for a break of approximately 3.0 inch equivalent diameter, the transient is turned around and the core is recovering prior to accumulator injection. For a 3.5 inch equivalent diameter break, the core remains uncovered with a decreasing level until accumulator action. Thus, the maximum break size showing core recovery prior to accumulator injection will be approximately 3.0 inch equivalent diameter. Accumulator injection commences when pressure reaches 600 psig, i.e., approximately 1200 seconds for the 3.0 inch break size.

The analysis of this break has shown that the high head portion of the ECCS, together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperature below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the ECCS in the event of a small break LOCA.

6.3.3.3 Large Break LOCA

A major LOCA is defined as a rupture ~~0.5~~ 1.0 ft^2 or larger of the RCS piping including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system. The boundary considered for LOCA as related to connecting piping is defined in Section 3.6.

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. Reactor trip occurs and the safety injection system is actuated when the pressurizer low pressure trip setpoint is reached. Reactor trip and safety injection system actuation are also provided by a high containment pressure signal. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection provide additional negative reactivity insertion to supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

When the pressure falls below approximately 600 psig the accumulators begin to inject borated water. The conservative assumption is made that injected accumulator water bypasses the core and goes out through the break until the termination of the blowdown phase. This conservatism is again consistent with the Final Acceptance Criteria.

The pressure transient in the reactor containment during a LOCA affects ECCS performance in the following ways. The time at which end of blowdown occurs is determined by zero break flow which is a result of achieving pressure equilibrium between the RCS and the containment. In this way, the amount of accumulator water bypass is also affected by the containment pressure, since

QUESTION 440.131:

Westinghouse has recently indicated that for some plant designs, full flow ECC assumed in the LOCA analysis results in a higher peak clad temperature. Provide an assessment for the LOCA analysis, assuming NO failures in the ECCS system.

RESPONSE:

On December 17, 1981, members of the Westinghouse Nuclear Safety Department staff met with representatives of the U. S. Nuclear Regulatory Commission at the Westinghouse Licensing Office in Bethesda to discuss an error in the ECCS analysis of four Westinghouse designed plants which are currently undergoing operating license review and one Westinghouse designed plant that is currently operating. The error discussed concerned the worst single failure of the emergency safeguards equipment assumed in the analyses. Correction of the error does not result in the violation of the ECCS Acceptance Criteria embodied in 10CFR50.46 because of margin to the limits as well as offsetting unused benefits in the input data. Westinghouse notified the NRC formally of the intent to modify the approved Westinghouse ECCS evaluation model which will permit the accurate modeling of the worst single failure as noted in Reference 1. Westinghouse also notified the owners of the affected plants. The Seabrook plant (NAH) did not have an error in the worst single failure assumption.

Background Information

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded pump performance, and the loss of one RHR pump as the most limiting single failure. This

is the limiting single failure assumption when offsite power is unavailable for most Westinghouse plants. However, for some Westinghouse four loop, non-UHI, non-burst node limited plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assumes minimum injection line resistances, enhanced ECCS pump performance and no single failure, results in the highest amount of flow delivered to the RCS.

The maximum safety injection flow results in competing effects which impact each plant differently. The higher safety injection flow results in a faster lower plenum filling and earlier bottom of core recovery which are beneficial to predicted core cooling. The downcomer will also fill faster and more completely on those plants which do not fill the downcomer before the accumulators empty. Again, this effect is beneficial to predicted core cooling. The higher amounts of safety injection flow under maximum safeguards will result in enhanced downcomer overfilling which will increase the maximum possible downcomer water level and result in a greater spillage from the reactor vessel cold leg stub to the containment. The former effect is beneficial to predicted core cooling while the later effect is deleterious to predicted core cooling. The higher safety injection flow also results in higher spillage from the broken loop. The increased spillage reduces the containment pressure which is deleterious to predicted core cooling. The higher safety injection flow is sufficient to condense all of the intact loop steam flow which reduces the intact loop cold leg pressure drop and reactor vessel broken loop cold leg stub pressure drop. The net effect of the condensation of the intact loop steam is a penalty in core reflood rate. All of these effects combine to impact the flooding rate which affects the peak cladding temperature.

Westinghouse has evaluated the impact of assuming no single failure by calculating the response to maximum safeguards for a number of different types of plants. The calculations used the maximum safeguards safety injection flow as well as a conservative modeling of the increase in the downcomer liquid height. These sensitivity studies indicate that the

440.135 Your response to Item II.K.3.17 is not adequate. Confirm that you will submit a report in accordance with the review schedule for licensing. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken.

RESPONSE: A procedure for collecting and submitting information concerning ECC system outages will be developed and implemented three months prior to fuel load. This procedure will delineate the methods to be used to compile a 5-year report which will contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC system or components involved in the outage; and (4) corrective action taken. This report will be submitted in accordance with the requirements of Item II.K.3.17 of NUREG-0737.

assumption of maximum safeguards would be a benefit in terms of peak cladding temperature for the Seabrook plant.

Reference

1. Letter from E. P. Rahe of Westinghouse Electric Corp. to R. L. Tedesco and T. P. Speis of the U.S. NRC dated December 22, 1981. NS-EPR-2538.