ENCLOSURE

SER Section	Title
5.2.2	Overpressure Protection
5.4.7	Residual Heat Removal System
6.3	Emergency Core Cooling System
15	Accident Analysis

MILLSTONE NUCLEAR POWER STATION UNIT 3

SAFETY EVALUATION REPORT

REACTOR SYSTEMS BRANCH

5.2.2 Overpressure Protection

Overpressure Protection for Millstone Unit 3, has been reviewed in accordance with Section 5.2.2 of the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800. A review of each of the areas listed in the "Areas of Review" portion of the SRP was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for Overpressure Protection is acceptable.

Overpressure protection for the Reactor Coolant Pressure Boundary (RCPB) is provided by means of three safety and two power operated relief valves in combination with the reactor protection system, and operating procedures. The combination of these features provides overpressurization protection as required by the General Design Criterion 15, the ASME Boiler and Pressure Vessel Code, Section III, and the 10 CFR 50 Appendix G. The above requirements assure RCPB overpressure protection for both power operation, and low temperature operation (start up and shutdown). The following is a discussion of both modes of overpressure protection.

5.2.2.1 Overpressure Protection During Power Operation

For this mode, the pressurizer power operated relief valves are sized to limit system pressure to a value not exceeding the safety valve setpoint (2485 psig) to minimize challenges to the safety valves. The pressurizer spray system is designed to maintain the reactor coolant system pressure below the power operated relief valve setpoint of 2335 psig during a step reduction in power level of up to 10%. The power operated relief valves limit the pressurizer pressure to a value below the high pressure reactor trip setpoint of 2385 psig for all design anticipated transients up to and including the design basis 50% step load reduction with steam dump to the condensers. Credit is taken only for safety valves in analyzing operational transients and faulted conditions.

Each of the pressurizer safety valve is spring-loaded and has a relieving capacity of 420,000 pounds per hour of saturated steam at 2485 psig. The combined capacity of two of these three safety valves is adequate to prevent the pressurizer pressure from exceeding the ASME Boiler and Pressure Vessel Code, Section III limit of 110 percent design pressure following the worst reactor coolant systems pressure transient, identified to be a 100 percent load rejection resulting from a turbine trip with concurrent loss of main feedwater. This event was analyzed with no credit taken for operation of reactor coolant system power operated relief valves, main steam line atmospheric steam dump valves, condenser steam dump system, pressurizer level control system, and pressurizer spray system.

The evaluation is supported by a generic sensitivity study of required safety valve flow rate versus trip parameter presented in WCAP-7769, Revision 1. The study indicates that the safety valves are sized sufficiently to protect RCS overpressurization assuming that the reactor is tripped from any one of the following four safety-grade trips: pressurizer high pressure, overtemperature Δ T, low main feedwater flow, and low-low steam generator water level.

The above analyses were performed using the LOFTRAN Code, a digital simulation which includes point neutron kinetics, Reactor Coolant System including the reactor vessel, hot leg, primary side of the steam generator, cold leg, pressurizer, and pressurizer surge line. This code has been reviewed and approved by the staff.

The safety valves are designed in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III and periodic testing and inspection are performed in accordance with Section XI. In Chapter 14 of the FSAR the applicant has described his preoperational test program, which includes testing of the pressure relieving devices discussed in this SER section, and has indicated that these tests would be conducted in full compliance with the intent of Regulatory Guide 1.68. Additionally, Items II.D.1 and II.D.3 of NUREG-0737 require performance testing of the relief and safety valves, and position indication of the valves. Conformance of these items are addressed in Sections * and * of this SER. We conclude that the overpressure protection provided for Millstone at power operating conditions will comply with the guidelines of Standard Review Plan 5.2.2 and the requirements of General Design Criterion 15.

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* LPM to provide section numbers.

5.2.2.2 Overpressure Protection During Low Temperature Operation

The SRP Sections 5.2.2 requires that the overpressure protection system during low temperature operation of the plant shall be designed in accordance with the requirements of Branch Technical Position RSB 5-2.

The applicant states that administrative procedures are available to assist the operator in controlling RCS pressure during low temperature operation. However, to provide a backup to the operator and to minimize the frequency of RCS overpressurization, an automatic system is provided to mitigate any inadvertent pressure excursions.

Protection against such overpressurization event is provided through the use of two pressurizer power operated relief valves (PORV). The applicant states that during startup and cooldown operation, the RCS is always "water solid" and the mitigation system is required during these low temperature operations.

The low temperature protection is primarily provided by the pressurized power operated relief valves with automatically adjusted opening setpoints which vary as a function of reactor coolant temperatures. The PORVs are each supplied with actuation logic to ensure that an automatic and independent RCS pressures control feature is available. The reactor coolant temperature measurements are auctioneered to obtain the lowest value. This temperature is translated into a PORV setpoint curve which is below the maximum allowable system pressure set forth by 10 CFR 50, Appendix G. If the measured reactor coolant pressure approaches the PORV setpoint curve within a certain limit, an alarm is sounded in the control room indicating that a pressure transient is occurring. On further increases in the measured pressure up to the PORV systems pressure, the PORVs are opened to relief system pressure. We have requested the applicant to address failures in the temperature auctioneer circuitry as both PORVs could be rendered inoperable by the failure of a single auctioneering circuit. We will report the resolution of this issue in a supplement to this SER. However, the consequences of the failure of the vital DC bus which causes the RHRS to isolate as well as defeating the PORV has not been addressed. Also, it is not clear whether there is a Tech Spec requirement for the RHR system to be operable whenever low temperature overpressure protection is required.***

As a backup to the low-temperature overpressure protection system the Residual Heat Removal System (RHRS) has two relief valves located at the RHR pump section line with a capacity of 900 gpm each at a setpoint pressure of 450 psig. The relieving capacity of each valve is more than adequate to relieve the combined flow of the two centrifugal charging pump. The relief valves at the RHR pump suction lines provide additional LTOP relieving capacity only when RHR suction isolation valves are open.

In response to the staff's concerns with regard to a vital DC bus failure which causes normal letdown to isolate and also results in the loss of one of the two PORVs plus a postulated single failure (closed) of the other PORV would fail the mitigative system for this event. The applicant, in a letter dated August 29, 1983, states that whenever the RCS is in a condition

***The staff is still pursuing these unresolved items and will report the resolution of these issues in a supplement to the SER.

in which the low temperature overpressure protection systems is required to be operable, all but one charging pump are required to be made inoperable and the operator is instructed to remove power to the safety injection pumps. This requirement assures that only one charging pump would be operating at the initiation of the event. Also one RHR loop is required to be in operation and the other RHR loop is required to be operable. This requirement ensures that at least one RHR suction line relief valve is available for overpressure protection.

Although the applicant indicates in a letter dated August 29, 1984, that the combined flow rate of two charging pump is below the relieving capacity of a single RHR relief valve. However we require that the safety injection pump flow capacity should be used for the analysis which demonstrates the adequacy of the low-temperature overpressure protection system. We will report the resolution of this issue in a supplement to this SER.***

We have reviewed the overpressure protection system for both normal and low temperature operations and conclude (with the exception of the unresolved issues indicated above) that the system is acceptable and meets the relevant requirements of GDC 15 and 31 and Appendix G to 10 CFR Part 50. This conclusion is based on the following:

The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients.

***The applicant did not use the safety injection pump flow capacity for the analysis.

Overpressurization protection is provided by three safety valves. These valves discharge to the pressurizer quench tank through a common header from the pressurizer. The safety and power operated relief valves in the primary system, in conjunction with the steam generator safety and power operated relief valves in the secondary system, and the reactor protection system, will protect the primary system against overpressure in the event of a complete loss of heat sink.

The peak primary system pressure following the worst pressure transient is limited to the ASME Code allowable value (110% of the design pressure) with no credit taken for nonsafety-grade relief system . The Millstone plant was assumed to be operating at design conditions (102% of rated power) and the reactor is shutdown by a high pressurizer pressure trip signal. The calculated pressure is less than 110% of the design pressure.

Overpressure protection during low-temperature operation of the plant is provided by two PORVs in conjunction with administrative controls. As a backup to the PORVs, the RHR suction line relief valves provide additional relief capacity.

The applicant has met GDC 15 and 31 and Appendix G since they have implemented the guidelines of BTP RSB 5-2. In addition, the applicant has incorporated in their design the recommendations of Task Action Plan Items II.G.I, II.D.1 and II.D.3 of NUREG-0737.

5.4.7 Residual Heat Removal System

The Residual heat Removal System (PHRS) for Millstone Unit 3, has been reviewed in accordance with Section 5.4.7 of the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800. A review of each of the areas listed in the "Areas of Review" portion of the SRP was performing according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for Residual Heat Removal is acceptable.

The residual heat removal system (RHRS) is designed to remove heat from the reactor coolant system after the system temperature and pressure have been reduced to approximately 350°F and 425 psig, respectively. The RHRS is capable of reducing the reactor coolant temperature to the cold shutdown condition and maintain this temperature until the plant is started up again.

The RHRS operates in the following modes:

(1) Emergency Core Cooling System (ECCS), Injection Mode
Functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the Refueling Water Storage Tank (RWST) into the RCS cold legs during the injection phase following a Loss of Coolant Accident (LOCA).

(2) Refueling
Both RHR pumps may be used during refueling to pump borated water from the refueling water storage tank to the refueling cavity. Following refueling, the RHR pumps are used to drain

the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the refueling water storage tank.

(3) Cold Shutdown

Removes RCS decay heat and maintains cold shutdown conditions. The relief valve on the RHRS suction line may be used for low temperature overpressure protection backup.

(4) Startup

Connected to the chemical and volume control system (CVCS) via the low pressure letdown line to control reactor coolant pressure. The relief valve on the RHRS suction line may be used for low temperature overpressure protection backup.

Design parameters for the RHRS are as follows:

(1)	Design pressure (psig)	600**
(2)	Design temperature (°F)	400**
(3)	Pump capacity (gpm)	4000
(4)	Number of independent trains	2

** Applicable to the tube side of the residual heat exchanger.

With only one RHR pump and heat exchanger in service and the heat exchanger supplied with component cooling water at a design flow and temperature of 3.3×10^6 lb/hr and 92.2 °F respectively, the RHRS is capable of reducing the reactor coolant temperature from 350°F to 200°F within 30 hours.

The two RHR trains are independent in action and powered by separate essential power supplies to provide redundancy.

5.4.7.1 Functional Requirements

As required by SRP Section 5.4.7, the RHRS for Millstone must meet General Design criteria Items 1 through 5. Items 1 through 4 regarding Quality Standards and Records, Design Bases for Protection Against Natural Phenomena, Fire Protection, and Environmental and Missile Design Bases are covered in Section *, *, *, and * of this report respectively, General Design Criteria 5, sharing of structures, systems and components, is met for the Millstone RHRS since components are not shared between unit.

During normal plant shutdown when nonsafety related equipment and offsite power are available, the decay heat removal function is performed by using the main feedwater system, the condenser steam dump system and service water system. During plant emergency shutdown, assuming offsite power and nonsafety related equipment are not available, the heat is transferred from the core by natural circulation with the steam generator as the heat sink.

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* LPM to provide section numbers

To achieve this, the safety related steam generator safety valves and power operated atmospheric relief valves are used to vent vaporized secondary coolant. Only two out of four atmospheric relief valves need to be operable for plant cooldown. Secondary coolant makeup is provided vi the Auxiliary Feedwater System (AFWS) from the seismic Category I tornado missile protected demineralizer water storage tank (DWST) with a capacity of 340,000 gals. The applicant states that the amount of water provided is sufficient to maintain the plant at hot standby condition for up to 10 hours then cooling the reactor to 350°F hot leg temperature within 6 hours, at which time the RHRS will be initiated. However, there is no indication regarding the minimum quantity of water required and allocated exclusively to the AFWS to perform this function.*** A single failure of any active component would not render all steam generators ineffective as a heat sink. Any one of the three auxiliary feedwater pumps has sufficient capacity to provide for all steam generator makeup requirements.

The reactor coolant system (RCS) depressurization is accomplished by the combination of RCS contraction due to the cooldown or opening of one of the two safety related pressurizer power operated relief valves. The discharge is directed to the pressurizer relief tank where it is condensed and cooled.

The depressurization process is integrated with the cooldown process to maintain the RCS within normal pressure-temperature limits. Just before initiating RHR cooling at 350°F, the RCS is depressurized to less than 425 psig.

***We are still pursuing unresolved issue and will report the resolution of this issue in a supplement to this SER.

The second stage of the cooldown is from 350°F to cold shutdown. During this stage, the RHRS is brought into operation. Circulation of the reactor coolant is provided by the RHR pumps, and the heat exchangers in the RHRS serve as the means of heat removal from the RCS. In the RHR heat exchangers, the residual heat is transferred to the component cooling water system which ultimately transfers the heat to the service water system and the ultimate heat sink.

The RHRS is a fully redundant system. Each RHR subsystem includes one RHR pump and one RHR heat exchanger. Each RHR pump is powered from a different emergency bus and each RHR heat exchanger is served from a different component cooling water system loop. Portions of the component cooling water system and the service water system associated with the RHR system are designed and constructed to safety related standards. All systems are capable of being operated from the control room with either only on-site or only off-site power.

If any component in one of the RHR subsystems were rendered inoperable as the result of a single failure, cooldown of the plant could still be achieved by using the remaining operable subsystem of the system.

We have requested the applicant to address situations when the reactor coolant system has been partially drained, improper reactor coolant inventory, or operating the RHRS at an inadequate NPSH has resulted in air binding of the RHR pumps with a subsequent loss of shutdown cooling. In a letter dated August 29, 1983, the applicant stated that a response to the staff's concern would be provided in a later date. We will report the resolution of this issue in a supplement to this SER.

Core reactivity is controlled during the cooldown by adding borated water to the RCS in conjunction with the cooldown. Boration is accomplished using safety related portions of the chemical and volume control systems. During the cooldown one of the three centrifugal charging pumps would take suction from one of the two boric acid tanks (BAT) and inject borated water into the RCS. The capacity of one BAT is sufficient to make up for reactor coolant contraction as a result of RCS cooldown from normal operating temperatures to the temperature when RHR initiation can commence. The two BATs, three centrifugal charging pumps and the associated piping and valves are designed to safety related standards. The backup borated water sources is provided from the RWST.

All systems are capable of being operated from the control room. Only in the event of a most limiting single failure, i.e., the failure of a RHR suction isolation valve interlock circuitry or emergency generator failure in conjunction with loss of off-site power that limited operator action outside the control room is required to open the suction isolation valve. The water supply provided to the auxiliary feedwater system to enable the plant to facilitate a safe shutdown condition is sufficient to hold the plant at hot standby for up to 10 hours and to provide a cooldown period of 6 hours to 350°F hot leg temperature at which time the RHRS can be initiated.*** If operator action is required to open the RHR isolation valve, which is just outside the containment, the operator would have ample time to perform the task. The staff consider this justification acceptable.***

*** The amount of AFW devoted for this purpose has not yet been specified by the applicant.

Redundancy in the RHRS is provided by two independent trains for each unit. Leak detection for the RHRS is discussed in Section * of this SER. Isolation valve and power supply redundancy are discussed under their separate topics in this section. The staff has reviewed the description of the residual heat removal system and the piping and instrumentation diagrams to verify that the system can be operated with or without offsite power and assuming a single failure. The two residual heat removal pumps are connected to separate buses which can be powered by separate emergency diesel generator in the event of loss of offsite power.

SRP Section 5.4.7 requires that the RHRS must be operable from the control room in accordance with GDC 19. Limited manual actions are permitted outside the control room after a single failure, if justified.

To assure Emergency Core Cooling System readiness and to protect RHR pumps, valve positions and pump running status indications are provided in the control room.

In accordance to Branch Technical Position RSB 5-1, the System(s) shall be capable of bringing the reactor to a cold shutdown condition with only offsite or onsite power available within a reasonable period of time following shutdown, assuming the most limiting single failure. A reasonable period of time is considered to be 36 hours. The staff has requested the applicant to identify the most limiting single failure and provide an analysis to show that the reactor can be brought to the RHR entry condition within 36 hours. We will report the resolution of this issue in a supplement to the SER.

* LPM to provide section numbers

5.4.7.2 RHR System Isolation Requirements

The RHRS valving arrangement is designed to provide adequate protection to the residual heat removal system from overpressurization when the reactor coolant system is at high pressure operation.

The RHRS is isolated from the RCS on the suction side by three normally closed, motor-operated valves in series on each suction line. They are closed during normal operation and are opened only for residual heat removal during a plant cooldown after the RCS pressure is reduced to 425 psig or lower and RCS temperature is reduced to approximately 350°F. Two of the motor operated valves in each inlet lines are provided with both "prevent open" and "auto closure" interlocks which are designed to prevent possible exposure. The "prevent open" interlock will prevent the valves to open if the RCS pressure is greater than 425 psig. The "auto closure" will close the valves automatically if the RCS pressure exceeds 750 psig.

The use of two independently powered motor-operated valves in each of the two inlet lines, along with two independent pressure interlock signals for each function, assures a design which meets applicable single failure criteria. The RHRS inlet isolation valves are provided with red-green position indicator lights on the main control board and the auxiliary shutdown pannel.

Isolation on the discharge portion of the RHRS, from the high pressure RCS is provided by a normally open motor-operated valve and three check valves in series. These check valves are located in the ECCS and their testing is described in Section *. We find the RHRS isolation design acceptable.

* LPM to provide section numbers.

5.4.7.3 RHR Pressure Relief Requirements

Overpressure protection of the residual heat removal system is provided by four relief valves, one on each of the suction and discharge lines. Each suction line relief valve has a capacity of 900 gallons per minute (gpm) at 450 psig which is sufficient to discharge the flow from both charging pumps at the relief valve setpoint. Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. The fluid discharge through the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged through the discharge side relief valves is collected in the primary drain tank of the equipment and floor drain system. These relief valves are adequate to protect the residual heat removal system from overpressurization. We conclude that this design is acceptable.

5.4.7.4 RHR Pump Protection

Each of the two RHR pumps has a mini-flow bypass line to prevent overheating in a loss of adequate discharge flow, and to prevent pump deadheading. A valve located in each mini-flow line is regulated by signal from the flow transmitters located in each pump discharge header. The control valves open when the RHR pump discharge flow is less than 500 gpm and close when the flow exceeds 1000 gpm, flow indicators are provided in the control room. A pressure sensor in each pump discharge header provide a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor. It is noted however, that the applicant has not provided adequate means to prevent RHR pump damage as a result of low flow or low suction pressure. We request the applicant to provide either low flow alarms which will alert the operator to take corrective action or some other appropriate *** means to protect the RHR pumps. We will report the resolution of this issue in a supplement to the SER.

5.4.7.5 Test, Operational Procedures, and Support Systems

The plant preoperational and startup test program provides for demonstrating the operation of the residual heat removal system in conformance with Regulatory Guide 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants" as specified in SRP Section 5.4.7.III.12.

Verification of adequate mixing of borated water added to the RCS under natural circulation condition, and conformation of natural circulation cool-down ability will be accomplished either by reference to the results of the tests from a plant of similar design or actual testing to be conducted at Millstone. We will require the applicant to provide a report justifying the applicability of the results of the boron mixing and natural circulation tests to be conducted at Diablo Canyon to the Millstone design. If the Diablo Canyon tests are not completed or do not provide satisfactory results to support the Millstone design, we request the applicant to perform such tests at Millstone during startup after the first refueling.

The staff has reviewed the component cooling water system to assure that sufficient cooling capability is available to the RHRS heat exchangers. The acceptability of this cooling capacity and its conformance to General Design Criteria 44, 45, and 46 are discussed in Section *.

* LPM to provide section numbers.
*** The applicant has not provided adequate means for RHR pump protection.

The applicant states that the RHRS is housed within a structure that is designed to withstand tornadoes, floods, and seismic phenomena. This area is addressed further in Section *.

The residual heat removal system capability to withstand pipe whip inside containment as required by General Design Criterion 4 and Regulatory Guide 1.46 is discussed in Section *. Protection against piping failures outside of containment in accordance with General Design Criterion 4 is discussed in Section *.

5.4.7.6 Conclusions

The residual heat removal function is accomplished in two phases: the initial cooldown phase and the residual heat removal system operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system (AFWS) and the atmospheric dump valves. The AFWS in conjunction with the SGs and PORVs are used to reduce the reactor coolant system temperature and pressure to the condition permitting operation of the RHRS. The RHRS removes core decay heat and provides long-term core cooling following the initial phase of reactor cooldown. The scope of review of the RHRS for the Millstone plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHRS and his analysis of the adequacy of those criteria and bases and the conformance of the design to these criteria and bases.

* LPM to provide section numbers.

With the exception of the unresolved issues indicated above, the staff concludes that the design of the Residual Heat Removal System is acceptable and meets the requirements of General Design Criteria 2, 5, 19, and 34. This conclusion is based on the following:

(1) The applicant has met the General Design Criterion 2 with respect to position C-2 of Regulatory Guide 1.29 concerning the seismic design of systems, structures and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

(2) The applicant has met the requirements of General Design Criterion 5 with respect to sharing of structure, systems and components by demonstrating that such sharing does not significantly impair the ability of the Residual Heat Removal System to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(3) The applicant has met General Design Criterion 19 with respect to the main control room requirements for normal operations and shutdown and General Design Criterion 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in Branch Technical Position RSB 5-1.

The staff reviews of the following Task Action Plan Items are addressed in Section * of this report.

(1) Task Action Plan Item II.E.3.2 of NUREG-0660 as it relates to systems capability and reliability of shutdown heat removal systems under various transients.

(2) Task Action Plan Item II.E.3.3 of NUREG-0660 as it relates to a coordinated study of shutdown heat removal requirements.

(3) Task Action Plan Item III.D.1.1 of NUREG-0737 as they relate to primary coolant sources outside of containment.

6.3 Emergency Core Cooling System

The Emergency Core Cooling System for Millstone Unit 3, has been reviewed in accordance with Section 6.3 of the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800. A review of each of the areas listed in the "Areas of Review" portion of the SRP was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for Emergency Core Cooling is acceptable.

* LPM to provide section numbers.

As specified in SRP Section 6.3.I.2 the design of the ECCS was reviewed to determine that it is capable of performing all of the functions required by the design bases. The emergency core cooling system (ECCS) is designed to provide core cooling as well as additional shutdown capability for accidents that result in significant depressurization of the reactor coolant system (RCS). These accidents include mechanical failure of the reactor coolant system piping up to and including the double-ended break of the largest pipe, rupture of a control rod drive, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the main steam piping.

The principal bases for the staff's acceptance of this system are conformance to 10 CFR 50.46 and Appendix K to 10 CFR 50, and GDC 2, 4, 5, 17, 27, 35, 36, and 37.

The applicant states that the requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.1 System Design

As specified in SRP Section 6.3.I.2 the design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS design is based on the availability of a minimum of three accumulators, one charging pump, one safety injection pump, one residual heat removal (RHR) pump, and one containment recirculation pump together with associated valves and piping. Following a postulated LOCA, passive (accumulators) and active (injection pumps and associated valves) systems will operate. After the water inventory in the refueling water storage tank (RWST) has been depleted, long-term recirculation will be provided by taking suction from the containment sump and discharging to the RCS cold and/or hot legs. The low-pressure passive accumulator system consists of four pressure vessels partially filled with borated water and pressurized with nitrogen gas to approximately 640 psia. Fluid level, boron concentration, and nitrogen pressure can be remotely monitored and adjusted in each tank. When RCS pressure is lower than accumulator tank pressure, borated water is injected through the RCS cold legs.

The high head injection system consists of two centrifugal charging pumps which provide high pressure injection of boric acid solution into the RCS. In addition to the high head charging pump system, two intermediate head safety injection pumps deliver fluid to the RCS. Both high and intermediate head pumps are aligned to take suction from the RWST for the injection phase of their operation. Low head injection is accomplished by two RHR pump subsystems taking suction from the RWST during the short-term ECCS injection phase. For long-term recirculation, the containment recirculation pumps will take suction from the containment sump.

The RWST minimum water inventory is 1,162,800 gallons of 2000-ppm borated water. The refueling water storage tanks is a vertical seismic Category I tank mounted on and secured to a reinforced concrete foundation. The borated water in the RWST is maintained at a maximum temperature of 50°F by circulating the RWST water through the refueling water cooler, which used chilled water from the seismic Category 1, tornado and missile protected chilled water system. The RWST is insulated to limit the temperature rise of the water to

1/2°F or less per 24 hour period whenever the chilled water system is inoperable. However, the applicant did not specify how the minimum temperature of the RWST content could be maintained to prevent boron precipitation. We will report the resolution of this issue in a revision to the SER.***

Water temperature in the RWST is indicated in the control room. Four water level indicator channels, which indicate in the control room, are provided. The High and Low level alarm are provided to initiate and stop makeup to assure that a sufficient volume of water is always available in the RWST. The Low-Low level alarm stops the RHR pumps and alerts the operator to realign the ECCS from the injection to the recirculating mode following an accident. We have requested the applicant to provide and justify the minimum time available to the operator to complete the switchover to the recirculation mode. The applicant states that a response to our question will be provided at a later date. We will review the response when it becomes available and report our findings later.

As specified in SRP Section 6.3.II the ECCS system is initiated either manually or automatically on (1) low pressurizer pressure, (2) high containment pressure, or (3) low pressure in any main steam line. This meets the requirements of GDC 20

The ECCS may also be manually actuated, monitored, and controlled from the control room as required by GDC 19. The ECCS is supplemented by instrumentation that will enable the operator to monitor and control the ECCS equipment

***Applicant did not discuss the minimum temperature of the RWST content and how to prevent following a LOCA so that adequate core cooling may be maintained. The acceptability of the proposed ECCS instrumentation and controls is addressed further in Section *.

As specified in SRP Section 6.3.III.3, the available net positive suction head for all the pumps in the ECCS (the safety injection, centrifugal charging and RHR pumps) should be shown to provide adequate margin by calculations performed to meet the safety intent of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." In response to the staff request for additional information, the applicant in Amendment 3 to the FSAR provided data of the required NPSH for each type of ECCS pump. We have reviewed the information and find it acceptable.

As required in SRP Section 6.3.III.11 the valve arrangement on the ECCS discharge lines have been reviewed with respect to adequate isolation between the RCS and the low pressure ECCS.

Isolation of the low pressure portions of the ECCS at the interface with the high pressure RCS is provided by three check valves in series. This arrangement is acceptable.

* LPM to provide section numbers.

Test lines are provided for periodic leakage checks of reactor coolant system pressure boundaries. This is discussed further in Section *. Containment isolation features for all ECCS lines, including instrument lines, the requirements of GDC 56 and Regulatory 1.11, "Instrument Lines Penetrating Primary Reactor Containment," are discussed in Section *.

In response to the staff concern with regard to the effects of water hammer that may occur in the ECCS lines, the applicant in a letter dated August 29, 1983, indicated that proper initial fill and venting of the ECCS ensures that the water hammer will not occur. In addition, the head of water provided by the RWST further assures that the lines will remain full. High point vents in the ECCS lines are provided to ensure means for proper venting of lines and pumps. Also the effects of water hammer have been considered in the design of the ECCS components.

In response to the staff concern with regard to the containment sump design and its effect on long term cooling following a loss of coolant accident, the applicant in Amendment 2 to the FSAR indicated that the containment sump vortex control was verified by means of a 1:3.25 scale model test. A wide range of possible approach flow distributions, bar rack and screen blockages, water levels, and pump operation combinations were tested to identify undesirable flow patterns. The applicant states that the test results show that the containment sump hydraulic performance is adequate at

* LPM to provide section numbers.

water levels above the (-)22'-6" elevation without vortex suppression grating. Since the minimum LOCA water elevation during recirculation pump operation is estimated at (-)23'-10", vortex suppression is required and will be provided for Millstone 3. The applicant also indicates that tests with the vortex suppression grating in place show that the sump performance is acceptable at the minimum estimated sump water level. To assure an acceptable pressure drop across the fine mesh sump screening during the recirculation mode of operation, the applicant states that the design velocity through the screens is limited to 0.2 ft/sec. assuming 50 percent of the available screen area is blocked.

With regard to debris and fallen thermal insulation which could block the trash rack or screen, the applicant indicates that the allowance for 50 percent plugging or blockage of the sump has been assumed in the design. The 50 percent blockage assumption is conservative since lighter particles will float on the water surface which will be above the screen assembly. Heavier particle will sink to the containment floor and will not be drawn into the screen due to the low inlet velocities used in the design of the sump.

The effects of primary coolant sources outside containment, NUREG-0737, Item III.D.1 are discussed in Section *.

* LPM to provide section numbers.

The safety injection lines are protected from intersystem leakage by relief valves in both suction header and discharge lines, Intersystem leakage detection is described in Section * for the RHR and safety injection pumps.

As specified in SRP Section 6.3 subsection II.B, no ECCS components are shared between units, which meets the requirements of GDC 5.

6.3.2 Evaluation of Single Failures

As specified in SRP Section 6.3.II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided during the initial injection phase with and without offsite power, assuming a single failure. The cold leg accumulators have a normally open motor-operated isolation valve and two check valves in series in their discharge lines. When the RCS pressure falls below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the RCS.

During plant startup, the operator is instructed, via operating procedures, to energize and open these valves when the RCS pressure reaches the safety injection setpoint. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint. Power is disconnected after valves are opened.

* LPM to provide section numbers.

Certain safety injection systems are blocked to preclude unwanted automatic actuation during normal shutdown and startup conditions. Failure to unblock these systems could seriously impair the reactor safety. We have requested the applicant to describe the alarms available to alert the operator to accidents when certain safety injection systems are blocked, as well as the operator actions and time frame available for the operator to mitigate such accidents, and the consequences of the accident. We will report he resolution of this issue in a supplement to this SER.

Power lockouts are provided in the control room for each valve whose spurious movement could result in degraded ECCS performance. The applicant's proposed method for locking out power to valves is discussed Section *.

Three active injection systems are available, each system having two pumps. The pumps in each system are connected to separate power buses and are powered from separate diesel generators in the event of loss of

* LPM to provide section numbers.

offsite power, as required by GDC 17. Thus, at least one pump in each injection system would be actuated. The high head injection systems contain parallel valves in the suction and discharge lines, thus ensuring operability of one train even in the event that one valve fails to open. The low and intermediate head injection systems are normally aligned so that valve actuation is not required during the injection phase.

The staff has expressed concern with regard to excessive boron concentration in the reactor vessel and hot leg recirculation flushing related to long term cooling following a LOCA. The applicant indicated in a letter dated August 29, 1983, that a response would be provided in a later date. We will review the material when it becomes available and report our findings in a supplement to the SER.

Flooding of ECCS components inside containment following a LOCA has been evaluated. No ECCS LOCA-related instruments or valve operators will be flooded following a postulated accident. All electrically-operated valves in the ECCS required to be functional during and following a LOCA are located outside containment. All other electrical equipment in the ECCS that is required during post-LOCA is either located outside containment or above the maximum calculated water level inside containment.

Based on staff review of the design features and with satisfactory resolution of the unresolved items discussed above, the staff concludes that the ECCS complies with the single-failure criterion of GDC 35.

6.3.3 Qualification of Emergency Core Cooling System

The ECCS is designed to seismic category I requirements, its compliance with Regulatory Guide 1.29, "Seismic Design Classification", and the design requirements of the structure which house the ECCS are discussed in Section *. The equipment design quality classification, its compliance with Regulatory Guide 1.26, "Quality Group Classification and Standard for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants," is discussed in Section *.

The ECCS protection against missiles inside and outside containment by the design of suitable reinforced concrete barriers, which include reinforced concrete walls and slabs (conformance to GDC 4), is discussed in Section *. The protection of the ECCS from pipe whip inside and outside of containment is discussed in Section *.

The active components of the ECCS design to function under the most severe duty loads including safe-shutdown-earthquake is discussed in Section *. The ECCS design to permit periodic inspection in accordance with ASME Code, Section XI, which constitutes compliance with GDC 36, is discussed in Section *. This meets the intend of SRP 6.3.III.23.c.

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* LPM to provide section numbers.

The ECCS incorporates two subsystems which serve other functions. The RHRS provides for decay heat removal during reactor shutdown, at other times the RHRS is aligned for ECCS operation. The centrifugal charging pumps are utilized for maintaining the required volume and water chemistry of primary fluid in the RCS. On an ECCS actuation signal, the system is aligned to ECCS operation and the CVCS function is isolated. The dual function of the RHRS and the centrifugal charging system does not effect its capability to function as an integral portion of the ECCS.

6.3.4 Testing

The applicant has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, as required by Regulatory Guides 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," and 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors," and GDC 37.

6.3.4.1 Preoperational Tests

One of these tests is to verify system actuation; namely, the operability of all ECCS valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valves interlocks.

Another test is to check the cold-leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each accumulator to confirm the line is clear and check the operation of the check valves.

The applicant will use the results of these preoperational tests to evaluate the hydraulic and mechanical performance of ECCS pumps delivering through the flow paths for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full-flow (through the actual piping) conditions.

The applicant has indicated to comply with the requirements of Regulatory Guide 1.79 and GDC 37 that cover testing of the ECCS.

Based on the review of the test programs discussed above, we conclude that the ECCS test program for Millstone 3 is acceptable. Additional discussion of the preoperational test program is presented in Section * of this SER.

6.3.4.2 Periodic Component Tests

Routine periodic testing of the ECCS components and all necessary support systems will be performed. Valves that actuate after a LOCA are operated through a complete cycle. Pumps are operated individually in this test on their miniflow lines except the charging pumps which are tested by their normal charging function. The applicant has stated that these tests will be performed in accordance with ASME Code, Section XI.

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* LPM to provide section numbers.

6.3.5 Performance Evaluation

The ECCS has been designed to deliver fluid to the RCS to limit the fuel cladding temperature following transients and accidents that require ECCS actuation. The ECCS is also designed to remove the decay and sensible heat during the recirculation mode. 10 CFR 50.46 lists the acceptance criteria for an ECCS. These criteria include the following:

(1) The calculated maximum fuel cladding temperature does not exceed 2200°F.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In addition, 10 CFR 50.46 states: "ECCS cooling performance shall be calculated in accordance with an acceptable model, and shall be calculated for a number of postulated loss-of-coolant accidents. Appendix K to 10 CFR 50, ECCS evaluation models, sets forth certain required and acceptable features of evaluation models."

The applicant has examined a spectrum of large breaks in RCS piping and these analyses indicate that the most limiting event is a cold leg guillotine break with a discharge coefficient of 0.6. The applicant took credit for one train of active ECCS components and three of the four accumulators in the analysis. In the large break analysis the worst case break was assumed which resulted in decreasing RCS pressure. ECCS was assumed to be initiated by Low Pressure Reactor Trip. The analysis results demonstrated that adequate core cooling is provided assuming the worst single failure with no credit taken for nonsafety related equipment.

The large-break LOCA evaluation model utilized in this analysis is described in WCAP-9220. This model was approved by NRC (letter from J.F. Stolz (NRC) to T.M. Anderson (W) dated April 29, 1978) and is used in large-break LOCA analyses for Westinghouse plants.

Containment parameters are chosen to minimize containment pressure so that core reflood calculations are conservative. Fuel rod initial conditions are chosen to maximize clad temperature and oxidation. Calculations of core geometry are carried out past the point where temperatures are decreasing. The most limiting break with respect to peak clad temperature is the double-ended guillotine break in the reactor coolant pump discharge piping. The peak clad temperature is 1960°F, which is below the 2200°F limit.

The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1.0 percent conclusion of 10 CFR 50.46. The maximum local water reaction is 8.67 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46.

6.3.5.2 Small-Break LOCA

The LOCA sensitivity studies determined the limiting small break to be less than a 10 inch diameter rupture of the RCS cold leg. A range of small break analysis were presented which established the limiting 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 less than that calculated for a large break and below the limits of 10 CFR 50.46.

The applicant has analyzed a spectrum of small-break LOCA (3-in, 4-in, and 6-in). With regard to peak clad temperature and metal water reaction, the analyses identify that the 4-in break is the limiting small break, the calculated peak cladding temperature is 1485°F, the total metal water reaction is less than 3 percent. The results also indicate that for a 6-in line break, the core mixture height falls to a minimum of 3.5ft at approximately 3 minutes following the accident and the level begins to recover afterward.

The applicant has analyzed the performance of the ECCS in accordance with the criteria set forth in Section 50.46 and Appendix K to 10 CFR 50. The staff has reviewed the applicant's evaluation, and concludes that it is acceptable and meets the criteria of 10 CFR 50.46.

6.3.5.3 Conclusions

The emergency core cooling system (ECCS) includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transport heat from the reactor core following a loss-of-coolant-accident. The scope of review of the ECCS for the Millstone-3 plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The staff review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

Pending resolution of the aforementioned concerns, the staff concludes that the design of the Emergency Core Cooling System is acceptable and meets the requirements of General Design Criteria 2, 5, 17, 27, 35, 36, and 37. This conclusion is based on the following.

(1) The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety or portions thereof which could have an adverse effect on ECCS by meeting position C.2 of Regulatory Guide 1.29.

(2) The applicant has met the requirements of GDC 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(3) The applicant has met the requirements of GDC 17 with respect to providing sufficient capacity and capability to assure that (a) specified acceptable fuel design ,limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

(4) The applicant has met the requirements of GDC 27 with regard to providing combined radioactivity control system capability to assure that under postulated accident conditions and with appropriate margin for stuck-rods the capability to cool the core is maintained and the applicant's design meets the guidelines of Regulatory Guide 1.47.

(5) The applicant has met the requirements of GDC 35 to provide abundant cooling capability for ECC by providing redundant safety-grade systems that meet the recommendations of Regulatory Guide 1.1.

(6) The applicant has met the requirements of GDC 36 with respect to the design of ECCS to permit appropriate periodic inspection of important components of the system.

(7) The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

(8) The applicant has provided an analysis of the proposed ECCS relative to the acceptance criteria of 10 CFR Part 50.46, and Appendix K to demonstrate that their ECCS designs for peak cladding temperature, maximum calculated cladding oxidation, maximum hydrogen generation, coolable core geometry and long-term cooling are in accordance with the acceptable evaluation model.

15 Accident Analysis

The accident analyses for Millstone Unit-3, have been reviewed in accordance with Section 15 of the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800. A review of each of the areas listed in the "Areas of Review" portion of the appropriate SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the appropriate SRP section. Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed was found to be acceptable for Millstone.

15.1 General

In accordance with SRP Section 15.1.1.I the applicant evaluated the ability of the Millstone Station to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10 and 15.

For each event analyzed, the worst operating condition and single failure were assumed and credit was taken for minimum engineered safeguards response. Parameters specific to individual events were conservatively selected. Two types of events were analyzed:

(1) Those incidents that might be expected to occur during the lifetime of the reactor (anticipated transients)

(2) Those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents).

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control insertion, accounts for a stuck rod, in accordance with GDC 26.

Review of thermal hydraulic code THINC-IV is described in Section * of this SER. The staff review of the FACTRAN code has progressed to the

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* LPM to provide section numbers.

point that there is reasonable assurance that analyses results dependent on the codes will not be appreciably altered by any revisions that may be required by the staff.

For transients and accidents, the applicant utilized a method which conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. DNBRs were calculated using the W-3 correlation, with a minimum DNBR of 1.3 used as the threshold for fuel failure.

The applicant accounts for variations in initial conditions by making the following assumptions as appropriate for the event being considered:

(1) Core power, 3425 MWt, ±2 percent

(2) Average reactor vessel temperature (T_{avg}), 587.1±6.5°F

(3) Pressure (at pressurizer), 2250 ± 30 psi.

The staff concludes the assumption for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects. For transients and accidents used to verify the ESF design, the applicant has utilized the safeguards power design value of 3579 MWt.

15.2 Normal Operation and Operational Transients

The applicant has analyzed several events expected to occur one or more times in the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the plant lifetime. Specific events were reviewed to ensure conformance with the acceptance criteria provided in the Standard Review Plan (SRP).

The acceptance criteria for transients of moderate frequency in the Standard Review Plan include the following considerations:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of design values (Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code).

(2) Fuel clad integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in SRP Section 4.4 provides a 95 percent probability, at a 95 percent confidence level, that no fuel rod in the core experiences a departure from nucleate boiling.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Conformance with SRP acceptance criteria constitutes compliance with GDC 10, 15 and 26 of Appendix A to 10 CFR 50. See Section * of this SER for a discussion of auxiliary feedwater system conformance to TMI Action Plan Item II.E.1.1 and Section * for a discussion of compliance with TMI Action Plan Item II.E.1.2.

The transients analyzed are protected by the following reactor trips:

(1) Power range high neutron flux

(2) High pressure

(3) Low pressure

(4) Overpower ΔT

(5) Overtemperature ΔT

(6) Low reactor coolant flow

(7) Reactor coolant pump shaft low speed

(8) Low steam generator water level

(9) High steam generator water level.

Time delays to trip, calculated for each trip signal, are included in the analyses. See Section * of this SER for a discussion of the staff review of reactivity control system functional design.

All of the transients which are expected to occur with moderate frequency can be grouped according to the following plant process disturbances: undercooling transients, increased cooling transients, changes in coolant inventory, and changes in core reactivity. Design-basis accidents have been evaluated separately as indicated in Section *.

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15.2.1 Increase in Heat Removal by the Secondary System

The applicant has analyzed the following events that produce increased heat removal by the secondary system:

(1) Decrease in feedwater temperature (SRP Section 15.1.1),

(2) Increase in feedwater flow (SRP Section 15.1.2),

(3) Excessive increase in steam flow (SRP Section 15.1.3), and

(4) Inadvertent opening of a steam generator relief valve or safety valve (SRP Section 15..1.4).

The most limiting transient with respect to fuel performance is the inadvertent opening of the steam generator relief or safety valve. The increase steam demand causes a reactor power increase which results in a reactor trip. The continued steam flow through the open valve will cause additional cooldown and additional positive reactivity insertion to the primary coolant system. The Safety Injection System (SIS) will inject highly concentrated boric acid from the RWST into the primary coolant system on either two out of four pressurizer low pressure signals, two out of three high containment pressure signals or two out of three low steam line pressure signals in any one loop. This insures the reactor will remain shutdown with any subsequent cooldown. The normal steam generator feedwater supply will be isolated automatically upon SIS initiation and then an orderly cooldown would be affected. The transient is terminated with the utilization of only safety related equipment. DNB does not occur during this transient.

* LPM to provide section numbers.

The transient which is most limiting of these with respect to the peak pressure is the increase in feedwater flow transient. The applicant has calculated a peak pressure of 2287 psia during this transient which is well below the system design pressure of 2485 psig.

15.2.2 Decrease in Heat Removal by the Secondary System

The applicant has analyzed the following events which result in a decrease in heat removal by the secondary system:

(1) Loss of external load (SRP Section 15.2.1),
(2) Turbine trip (SRP Section 15.2.2),
(3) Loss of Condenser Vacuum (SRP Section 15.2.3),
(4) Inadvertent closure of Main Steam Isolation Valve (SRP Section 15.2.4),
(5) Steam Pressure Regulator Failure (SRP Section 15.2.5),
(6) Loss of nonemergency power to the station auxiliaries (SRP Section 15.2.6),
(7) Loss of Normal Feedwater Flow (SRP Section 15.2.7).

Plant transients which result in an unplanned decrease in heat removal by the secondary system that might be expected to occur with moderate frequency are identified in the above list. All these postulated transients have been reviewed. It was found that the most limiting event in this group of events in regard to the maximum pressure within the reactor coolant and main steam systems was the lost of normal feedwater caused by a loss of offsite power. The reactor is tripped on the high pressurizer pressure signal and the peak pressure during the transient is 2565 psia, well below the ASME requirements for maximum pressure to be limited to 110% of design pressure.

The applicant states in Section 15.2.7 of the FSAR that the most limiting event with respect to fuel performance and maximum pressure within the reactor coolant and main steam system is the loss of normal feedwater caused by a loss of offsite AC power. In this transient, the loss of offsite power is closely followed by a turbine trip and reactor trip. The emergency feedwater system is automatically started but only one emergency feedwater pump is assumed to be feeding all four steam generators. It is assumed that only safety related equipment is used to mitigate the events, the primary system pressurizer relief valves are assumed fail to operate therefore all residual heat must be removed though the steam generator atmospheric steam dump valves, which are safety related components. The first few seconds after the loss of normal feedwater transient will closely resemble a simulation of the complete loss of forced reactor coolant flow event (discussed in Section 15.3.2 of the FSAR.) The DNBR is always greater than 1.30. The peak pressure during the transient is 2565 psia well below the ASME requirements for maximum pressure to be limited to 110% of design pressure.

15.2.3 <u>Decrease in Reactor Coolant Flow Rate</u>

The applicant has analyzed the total loss of forced reactor coolant flow event which bounds partial loss of forced reactor coolant flow. This event is reviewed using the review procedures and acceptance criteria set forth in SRP Section 15.3.1 and 15.3.2.

The loss of off-site power and resulting loss of all forced coolant flow through the reactor core causes an increase in the average coolant temperature and a decrease in the margin to DNB. The reactor is tripped from an under-voltage trip monitoring the RCP power supply and a minimum DNBR of 1.32 is reached approximately 3 seconds into the transient. The maximum calculated RCS pressure is 2330 psia during the transient.

We conclude that the results of the analysis meet the guidelines of SRP Section 15.3.1 and 15.3.2 and is acceptable.

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Start of an Inactive Reactor Coolant Pump at an Incorrect Temperature

In Section 15.4.4 of the FSAR, the applicant provides the results of an analysis for startup of an Inactive Reactor Coolant Pump event. This event is reviewed using the review procedures and acceptance criteria set forth in SRP Section 15.4.4.

During the first part of the transient, the increase in core flow with cold water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is due to the decrease in core inlet water temperature. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable to the staff. The maximum calculated RCS pressure is 2350 psia and the minimum DNBR is above 1.93 during the transient.

We conclude that the results of the analysis meet the criteria in SRP Section 15.4.4 and is acceptable.

15.2.4.2 Inadvertent Boron Dilution

Section 15.4.6 of the Standard Review Plan requires that at least 15 minutes should be available from the time the operator is made aware of an unplanned boron dilution event to the time a loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. Thirty minutes warning is required during refueling. The staff has requested that control room alarms be available to alert the operator to boron dilution events in all modes of operation. If a second alarm is not provided, the applicant must show that the consequences of the most limiting unmitigated boron dilution event meet the staff criteria and are acceptable. The staff requires that the applicant provide analyses in accordance with the guidelines of SRP, Section 15.4.6, boron dilution events in each of the six operational modes. The analyses should confirm that time intervals meet the SRP criteria. Also, technical specifications should be established consistent with the analyses assumptions.

The applicant stated in the letter dated August 29, 1983 that a response to our request would be provided at a later date. We will review the information when it becomes available and report the staff evaluation in a supplement to this SER.

15.2.5 Increase in Reactor Coolant System Inventory

The applicant has analyzed the following events that result in increase in the primary system inventory:

(1) Inadvertent actuation of emergency core coolant system during power operation (SRP Section 15.5.1)
(2) Chemical and volume control system malfunction (SRP Section 15.5.2).

Emergency core cooling system operation could be initiated by a spurious signal or operator error. Two cases were examined, one in which reactor trip occurs simultaneously as a result of the safety injection signal, the other in which reactor trip occur later in the transient on RCS low pressure. The reactor pressure decreases during initial phase of the transient and reach a minimum pressure of 1850 psia at 100 seconds into the transient then recovers slightly to approximately 2000 psia. The DNBR never drops below its initial value for this transient.

The applicant's evaluation of the chemical and volume control system malfunction event is presented in Section * and the staff evaluation is addressed in Section * of this SER.

15.2.6 Decrease in Reactor Coolant Inventory

In Section 15.6.1 of the FSAR, the applicant provides the results of an analysis for inadvertent opening of a pressurizer safety or relief valve. During this event, nuclear power remains at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially as a result of the reduction in RCS pressure, but increase rapidly following the trip. The minimum DNBR of 1.5 occurred at 24 seconds into the transient. The RCS pressure decreases throughout the transient.

15.3 Design-Basis Accidents

The staff has reviewed the postulated events with regard to the facility design basis. These events have been classified in the Standard Review Plan as postulated accidents. The acceptance criteria specified in the SRP for evaluation of the consequences of the postulated accidents include the following:

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* LPM to provide section numbers.

(1) Pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design pressure, except that calculated pressures of 120 percent of design may be permitted for very low probability events.

(2) The potential for core damage should be evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 limit as discussed in SRP Section 4.4. If the DNBR fails below these values, fuel damage (rod perforation) should be assumed unless it can be shown, based on an acceptable fuel damage model, that no fuel failure results. If fuel damage is calculated to occur, it should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

(3) Any activity release must be such that the calculated doses at the site boundary are within the guidelines of 10 CFR 100 (see Section *). Conformance with the SRP acceptance criteria constitutes compliance with GDC 27, 28 and 31.

Other aspects of the staff review included evaluation of protection against conditions which might lead to brittle fracture of the reactor system pressure boundary low-temperature operation for compliance with GDC 31 (see SER Section *). Staff review of emergency core cooling system functional design for compliance with GDC 35 is discussed in Section * of this SER. The staff

* LPM to provide section numbers.

coordinated its review of Chapter 15 events with the review of the auxiliary feedwater system. Section * of the SER discusses compliance of the AFW design with the requirements in Item II.E.1.1 of NUREG-0737 and Section * discusses compliance with Item II.E.1.2.

In the analysis of the events, the applicant investigated a broad spectrum of related events to determine the bounding case, including the worst single active failure. Sensitivity studies were performed to identify parameters for initial conditions and appropriate credit for systems and their performance during the limiting events in terms of protection of various barriers.

15.3.1 Loss of Coolant Accident (SRP Section 15.6.5)

The applicant has analyzed the double ended cold leg guillotine (DECLG) break as the most limiting large break LOCA. The analysis is done using three different flow coefficients. The results of these show that the DECLG with a Moody break discharge coefficient of 0.6 is the worst case. In this analysis peak clad temperature reached is 1960°F. For the small break LOCA the applicant has determined that a cold leg rupture of less than 10" diameter is the most limiting. The analysis was performed for a 3 inch, 4 inch and 6 inch diameter break. The results show that the 4 inch diameter break is the worst case and it results in a peak clad temperature 1495°F. Both of these accidents are terminated by Safety Injection System and Emergency Core Cooling System operation. Only safety grade equipment is utilized to mitigate the accident.

* LPM to provide section numbers.

The staff concludes that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR Part 50, 50.46 and Appendix K, GDC 35, and 10 CFR Part 100. This conclusion is based on the following:

The applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR Part 50, 50.46 and Appendix K to 10 CFR Part 50). The analysis considered a spectrum of postulated break size and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff and described in NUREG-0390 and SER for licensing the Sequoyah (NUREG-0011) and McGuire (NUREG-0422) plants. The results of the analyses show that the ECCS satisfy the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

(6) The applicant has met the requirements of TMI Action Plan Item II.E.2.3, II.K.3.5, II.K.3.25, II.K.3.30 and II.K.3.31.

Pending resolution of the aforementioned concerns, the staff concludes that the calculated performance of the ECCS following a postulated LOCA and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulation, the applicable regulatory guide and the staff technical position. Therefore, the staff concludes the ECCS is considered acceptable.

15.3.2 Steamline Rupture (SRP Section 15.1.5)

The applicant has submitted analyses of postulated steamline breaks that show no fuel failures attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse four-loop plants.

A postulated double-ended rupture at hot standby power with offsite power available was analyzed as the worst case. The applicant referenced WCAP-9227 as justification for this selection. WCAP-9227 is currently under review by the staff. The review of WCAP-9227 has progressed to the point that there is reasonable assurance that analysis results presented in this topical report will not be appreciably altered by any revisions that may be required by the staff. The double ended rupture would cause the reactor to increase in power due to the decrease in reactor coolant temperature. The reactor would be tripped by either reactor overpower ΔT or by the actuation of the safety injection system. The safety injection system will be actuated by any of the following: Two out of four low pressurizer pressure signals, two out of three high containment pressure signals or two out of three low steam line pressure signals in any one loop.

Although a return of criticality occurs, there is no fuel damage since the minimum DNBR remains greater than 1.30.

The staff concludes that the consequences of postulated steamline breaks meet the relevant requirements set forth in GDC 27, 18, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan items. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was limited so that control rod insertability would be maintained and no loss of core cooling capability resulted. The minimum DNBR experienced by any fuel rod was $>$ 1.30, resulting in no rod experiencing cladding perforation.

(2) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency core cooling systems to provide abundant core cooling and reactivity control (via boron injection).

(4) The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite

power, have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

(6) The applicant has met the requirements of Task Action Plan Items II.E.1.1, and II.E.1.2, with respect to demonstrating the adequacy of the auxiliary feedwater system design to remove decay heat following steam system piping failure.

(7) The applicant has met the requirements of Task Action Plan Items II.K.3.25 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

(8) The applicant has met the requirements of Task Action Plan Items II.K.3.25 with respect to the operation and tripping of the reactor coolant pumps. The assumptions used are conservative and consistent with the generic resolution to Item II.K.3.5.

15.3.3 Feedwater System Pipe Break (SRP Section 15.2.8)

The applicant has provided a feedwater line break analysis for Millstone using assumptions that would minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the calculated primary system pressure. A double ended rupture of the largest feedwater line was assumed, as well as failure of one intact steam generator feedwater control valve to open and supply emergency feedwater to the steam generator.

The system code used to perform these analyses is LOFTRAN (discussed in Section *). The analysis assumed the most restrictive single failure of the auxiliary feedwater system, emergency feedwater flow is supplied to only two intact steam generators. This is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of only safety related equipment is sufficient to mitigate this accident. No fuel damage was calculated to occur, and the peak calculated pressurizer pressure was about 2500 psia. The staff finds these results to be within the required limits.

15.3.4 Reactor Pump Rotor Seizure and Shaft Break
(SRP Section 15.3.3/15.3.4)

The applicant's analyses for locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in Section 15.3 of the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3, 15.3.4 and GDC 17, we require that this event be analyzed assuming turbine trip and consequential loss of offsite power to the plant auxiliaries and resulting coastdown of all undamaged pumps. Appropriate delay times may be assumed for loss of offsite power if suitably justified.

The event should also be analyzed assuming the worst single failure of a safety active component. Maximum technical specification primary system activity and steam generator tube leakage at the rate specified in the Technical Specifications should be assumed. The results of the analyses should demonstrate that offsite doses following the accident are less than the 10 CFR 100 guideline values.

* LPM to provide section numbers.

In response to the staff request, the applicant indicate in a letter dated August 29, 1983, that additional information will be provided at a later date. We will review the information when it becomes available and report our resolution of this issue in a supplement to this SER.

15.3.5 Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) accident releases primary coolant to the secondary side of the steam generator, thus providing a pathway for radiological releases to the environment. This SER section evaluates the system aspects of the SGTR analysis. The radiological consequences of this accident are discussed in Section * of this SER.

The accident examined in the FSAR involves a complete severance of a single steam generator tube. The applicant's description of the accident was reviewed, including the sequence of events, bases for operator action, and the effects of loss of offsite power. The accident scenario involves reactor and turbine trip and subsequent safety injection (SI) actuation initiated by low pressurizer pressure. Emergency feedwater system startup, initiated by SI, was also examined. If offsite power is available, the turbine bypass valves would close and the steam would discharge to the atmosphere via the steam generator atmospheric relief and/or safety valves.

The applicant states that the operator is expected to determine that a steam generator tube rupture has occurred and to identify and isolate the faulty steam generator on a restricted time scale to minimize contamination of the

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* LPM to provide section numbers.

secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The applicant further states that "consideration of the indications provided the control board together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed within 30 minutes of initiation for the design basis events."

However, SGTR events at operating reactors generally indicate a longer time period than 30 minutes for pressure equilization e.g., 3 hours at Ginna. The applicant subsequently provided additional information in response to the staff's questions on this subject. We have reviewed the applicant's response and conclude that the applicant's analysis is incomplete with regard to demonstrating that the acceptance criteria in SRP Section 15.6.3 are fully met and that there are discrepancies in the applicant's submittal. We require the following additional information to fully evaluate this analysis:

(1) The applicant should submit an evaluation of operators actions necessary to effect pressure equalization, and a conservative time estimate for each action, as well as initial delay time.

(2) FSAR Section 15.6.3 indicates equalization of primary and secondary pressure 30 minutes after the SGTR event, with consequent termination of steam generator tube leakage. However, Fig. 15.6-3A and 15.6-3C indicate a pressure differential of 950 psi at 1800 seconds. The applicant should explain this discrepancy and modify the analysis of this event accordingly, utilizing the evaluation of operator actions discussed above.

(3) The applicant is requested to discuss (a) whether, as a result of possible modification of its analysis, including consideration of longer leak times, liquid can enter the main steamlines, and (b) what would the effects be on the integrity of the steam piping and supports, considering both the liquid dead weight and the possibility of water hammer. Unless the applicant can demonstrate that the incident will be terminated within a time period sufficiently short to avoid steam generator over fill, the applicant should submit the results of an analysis that demonstrates that the integrity of the steamlines and supports will be maintained.

(4) The applicant should verify that primary components that are credited in the analysis to mitigate the consequences of the SGTR, including the component power and motive sources, are classified as safety related, meet applicable General Design Criteria, including GDC 1, 2, and 4, are seismically and environmentally qualified, and have sufficient capability to equalize primary and secondary pressure within the time period postulated in the response to items 1 and 3 above.

Satisfactory applicant responses to these concerns will enable performance of an independent dose consequence analysis by the staff. The results of this analysis will be reported in SER Section *.

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* LPM to provide section numbers.

TMI Action Plan Requirements (RSB Scope)

II.B.1 Reactor Coolant System Vents

In response to the above requirements, the applicant has stated that the reactor vessel vent system design is in accordance with the requirements of NUREG-0737. However, the applicant did not provide sufficient information on the RCS high point venting for the staff to complete the review. We require the applicant to provide a description of the system and flow diagram for staff to review. We will report the resolution of this issue in a supplement to this SER.

II.K.1.5 Review ESF Value Positions, Controls and Related Test and Maintenance Procedures to Assure Proper ESF Functioning

In response to the above requirements, the applicant has stated that Millstone 3 meets the requirements of NUREG-0737.

II.K.1.10 Review and Modify Procedures for ESF From Service to Assure Operability Status is Known

In response to the above requirements the applicant has stated that Millstone 3 meets the requirements of NUREG-0737.

II.K.2.13 Thermal Mechanical Report Effect of High Pressure Injection On Vessel Integrity for Small Break LOCA With No Auxiliary Feedwater

Staff review of this item is covered in NRC unresolved safety issue A-49 "Pressurized Thermal Shock".

II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Westinghouse has performed a study which addresses the potential for void formation in Westinghouse design NSSS during natural circulation cooldown/-depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group. The staff is currently reviewing this report. When the staff has completed the evaluation of this report, Millstone 3 may be required to modify the operating procedures if needed.

II.K.2.19 Sequential Auxiliary Flow Analysis

Sequential auxiliary feedwater flow analytical requirements is only of concern to once-through steam generator design. Since Westinghouse utilizes inverted U-tube steam generator designs, requirements set forth by Item II.K.2.19 are not applicable.

II.K.3.1 Installation and Testing of Automatic PORV Isolation System

The applicant has stated, without justification, that the addition of an automatic isolation system for the PORVs will not be utilized for Millstone 3. The staff takes exception to the statement. However, the staff is currently reviewing the Westinghouse Owners Group generic report regarding this issue, unless the findings of our generic review reconcile with the applicant's view, NRC will require further consideration of the modification on Millstone 3 installation.

II.K.3.3 Reporting SV and PORV challenges and Failures

In response to the above requirements, the applicant states that any failure of a PORV or safety valve to close will be reported promptly to NRC. All challenges to the PORVs or safety valves will be documented in the annual report. Based on the above, we conclude that the Millstone 3 procedures meet the requirements of this item and is acceptable.

II.K.3.5 Automatic Trip of RCPs During LOCA

In response to the above requirements, the applicant states that according to Westinghouse analyses (WCAP-9584, WCAP-9585) that sufficient time is available for manual tripping of the pumps, therefore automatic RC pump trip is not necessary.

The staff will be issuing criteria for RCP trip in the near future for the applicant to implement.

II.K.3.10 Proposed Anticipatory Trip Modification

In response to this requirement, the applicant states that an analysis has been performed using realistic yet conservative values for the core physics parameters, and a conservatively high initial power, average reactor temperature and pressurizer pressure level. The transient was initiated from 50 percent of the reactor fuel power level plus 2 percent for power measurement uncertainty. The applicant concluded that based on the results from the analysis the peak pressure reached in the pressurizer would be 2302 psia. The transient will not cause the pressurizer PORVs to be challenged since the set point for these PORVs is 2350 psia. However we will require the applicant to provide details of the above stated analysis for staff review. We will report the resolution of this item in a supplement to this SER.

II.K.3.17 Report On Outages of ECCS

In response to this requirement, the applicant indicates that Northeast Nuclear Energy Company will compile information to determine the frequency and duration of ECCS outage and will use this information to determine if future systems or Tech Spec modifications are necessary. This commitment does not meet NUREG-0737 requirements for this item. We will require the applicant to submit a report in accordance with the requirements of NUREG-0737 to address ECCS outage. We will report our resolution of this item in a supplement to this SER.

II.K.3.25 Effect of Loss of Alternating Current Power On RCP Seals

In response to the above requirements, the applicant stated that in the event of loss of offsite power, the RCP motor is deenergized and both of these cooling supplies are terminated. However, the diesel generator are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during loss of offsite power for at least 2 hours. Based on the above, we conclude that the design meets the requirement of this item and is acceptable.

II.K.3.30 Revised Small Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K

In response to this requirement, the applicant stated that Westinghouse is committed to revise its small-break LOCA analysis model to address NRC concerns. This revised Westinghouse model scheduled for submittal to NRC for review by **. We conclude that the applicant commitment meets the requirement of this item and is acceptable.

II.K.3.31 Plant-Specific Calculations to Show Compliance with 10 CFR 50.46

In response to the requirement, the applicant states that a small break loss-of-coolant accident specific to Millstone 3 has been performed utilizing the present Westinghouse small break evaluation model. The results are in conformance with 10 CFR Part 50, Appendix K and 10 CFR Part 50.46. We conclude that the applicant analysis meets the requirements of this item and is acceptable. Nevertheless if the Westinghouse new model for small break LOCA evaluation yields more limiting results versus the current approved model, we will require the applicant to re-analyze the accident with the new model.

** LPM to determine the date of submittal.

ENCLOSURE 2

List of Unresolved Issues

Millstone Nuclear Power Station Unit 3

Reactor Systems Branch

1. Section 5.2.2 - Applicant to address the staff concern on failure in the temperature auctioneer for the PORV which could fail the low temperature overpressure protection system. (Q440.14)

2. Section 5.4.7 - Applicant to provide additional information regarding a potential problem on the loss of shutdown cooling during certain RCS maintenance evolutions. (Q440.15)

3. Section 5.4.7 - Applicant to address the most limiting single failure assumed in the analysis of evaluating the RHRS performance. Also identify and justify the assumptions used in the analysis which demonstrates that the RHRS meets the requirements of Branch Technical Position RSB 5-1. (Q440.24)

4. Section 6.3 - Applicant to provide information on the alarms available to alert the operator to a failure to block the safety injection system during normal shutdown and startup condition. Also the operator actions and time frame available for the operator to mitigate such incident and the consequences of the accident are required. (Q440.28)

5. Section 6.3 - Applicant to provide and justify the minimum time available to the operator to complete the switch over to the recirculation mode. (Q440.31)

6. Section 6.3 - Applicant to address the actions for switchover including actions to restore power to valves and provide an evaluation of the maximum time required for each operator action. (Q440.32)

7. Section 6.3 - Applicant to address the initiation and completion times of actions of the ECCS that were used in Chapter 15 analysis with and without offsite power (Q440.34)

8. Section 6.3 - Applicant to address the design consideration of the RWST. (Q440.36)

9. Section 6.3 - Applicant to address excessive boron concentration in the reactor vessel and hot leg recirculation flushing related to long-term cooling following a LOCA. (Q440.41)

10. Section 15.1.5 - Applicant to address the consequences of additional cooling caused by early introduction of AFW or failure of operator to isolate the AFW to the faulty steam generator. (Q440.46)

11. Section 15.2.5 - Applicant to address the operator actions assumed in the feedwater pipe break analysis. (Q440.49)

12. Section 15.3.3 - Applicant to provide the results of an analysis for RC pump shaft seizure assuming a loss of offsite power. (Q440.50)

13. Section 15.4.4 - Applicant to provide a description of the analytical model used to obtain the results for the analysis on startup of an inactive RC pump at an incorrect temperature and boron concentration. (Q440.55)

14. Section 15.4.6 - Applicant to provide the boron dilution analysis according to SRP Section 15.4.6 guidelines. (Q440.56)

15. Section 15.4.6 - Applicant to provide a list of all the instruments and alarms available to alert the operator of the boron dilution event.

16 Section 15.6.3 - Applicant to provide an analysis to demonstrate the operator ability to isolate and equalize the pressures of the faulty steam generator during a steam generator tube rupture event. (Q440.59)

17. TMI Action Plan Requirements (II.K.3.10) - Applicant to provide details of the analysis used to evaluate TMI action plant item II.K.3.10 with regard to proposed anticipatory trip modification.

18. TMI Action Plan Requirements (II.K.3.17) - Applicant to provide a report in accordance with the requirements of NUREG-0737 to address ECCS outage.

19. Section 5.2.2.2 - Applicant to provide analysis to evaluate the consequences of a vital DC bus failure which causes the RSRS to isolate as well as defeating the PORV.

20. Section 5.4.7.4 - Applicant to provide adequate means such as low flow alarms which alert the operator to take corrective actions.

21. Section 6.3.1 - Applicant to address the minimum temperature of the RWST content and how to prevent boron precipitation.

22. Section 5.2.2. - Applicant to provide analysis using the safety injection pump flow capacity as design basis to demonstrate the adequacy of the low-temperature overprotection system.