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ADVANCED NUCLEAR FUELS CORPORATION

MILLSTONE UNIT 2
PLANT TRANSIENT ANALYSIS REPORT
ANALYSIS OF CHAPTER 15 EVENTS

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MILLSTONE UNIT 2 PLANT TRANSIENT ANALYSIS REPORT - ANALYSIS OF CHAPTER 15 EVENTS

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TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

LIST OF TABLES

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

1.0 INTRODUCTION

This report provides a review of the Standard Review Plan(1) Chapter 15 Events for Millstone Unit 2. The event review is being performed to support the first reload of Advanced Nuclear Fuels Corporation (ANF) fuel. The review also supports Technical Specification changes required to increase plant operating flexibility (e.g., longer cycles).

In accordance with ANF methodology(2), all events described in the Standard Review Plan (SRP) have been reviewed and placed (dispositioned) into one of the following four categories:

(1) The event initiator or controlling parameters have been changed from the analysis of record so that the event needs to be reanalyzed for the current licensing action;

(2) The event is bounded by another event which is to be reanalyzed;

(3) The event causes and principal variables which control the results of the event are unchanged from or bounded by the analysis of record; or

(4) The event is not in the licensing basis for the plant.

The review of the current plant safety analysis for the event disposition process incorporated the following revisions to acceptable plant operating conditions:

(1) Increased maximum radial peaking factor.

(2) Extension of the cycle length to 18 months.

  (a) Increased both positive and negative bounds on the moderator temperature coefficient.

  (b) Increased shutdown margin requirements.

(3) Operation over an inlet temperature range.

Section 2.0 provides a summary of the event disposition. In order to facilitate review, the events are numbered in accordance with the SRP and cross-referenced to the pertinent updated FSAR sections. Section 3.0 presents the results of the analysis and justifications. Section 4.0 presents the references used in this report.

In the event disposition, all of the reactor operating conditions allowed by the plant Technical Specifications[3] are examined to insure that the bounding subevents are identified for each SRP event category. This insures that the subsequent safety analysis will support the complete range of allowable operating conditions. The reactor operating modes allowed for Millstone Unit 2 by the plant Technical Specifications are listed in Table 1.1.

Table 1.1 Reactor Operating Modes for Millstone Unit 2

	Mode	Reactivity Condition, K_{eff}	% Rated Thermal Power*	Average Coolant Temperature
1.	Power Operation	≥ 0.99	$> 5\%$	$\geq 300°F$
2.	Startup	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3.	Hot Standby	< 0.99	0	$\geq 300°F$
4.	Hot Shutdown	< 0.99	0	$300°F > T_{avg} > 200°F$
5.	Cold Shutdown	< 0.98	0	$\leq 200°F$
6.	Refueling**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

2.0 SUMMARY OF DISPOSITION OF EVENTS

Table 2.1 presents a summary of results of the event disposition. In accordance with ANF methodology, the events are placed in one of the four categories identified in Section 1.0.

Table 2.1 Disposition of Events Summary

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3	14.1.4, 14.11.2
	15.1.2	Increase in Feedwater Flow			
		1) Power	Bounded	15.1.3	14.1.4, 14.8, 14.11.2
		2) Startup	Bounded	15.1.3	
	15.1.3	Increase in Steam Flow	Analyze		14.8, 14.11.1
	15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Bounded	15.1.3	14.8, 14.11.1
	15.1.5	Steam System Piping Failures Inside and Outside of Containment	Analyze		14.1.4, 14.12
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	15.2.1	Loss of External Load	Analyze		14.9
	15.2.2	Turbine Trip	Bounded	15.2.1	14.9
	15.2.3	Loss of Condenser Vacuum	Not in Licensing Basis		
	15.2.4	Closure of the Main Steam Isolation Valves (MSIVs)	Analyze		14.8
	15.2.5	Steam Pressure Regulator Failure	Not applicable; BWR Event		

Table 2.1 Disposition of Events Summary (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
	15.2.6	Loss of Nonemergency A.C. Power to the Station Auxiliaries	Not in Licensing Basis		
	15.2.7	Loss of Normal Feedwater Flow	Analyze		14.8, 14.10, 14.A
	15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Not in Licensing Basis		
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW				
	15.3.1	Loss of Forced Reactor Coolant Flow	Analyze		14.6
	15.3.2	Flow Controller Malfunction	Not Applicable		
	15.3.3	Reactor Coolant Pump Rotor Seizure	Analyze		14.6
	15.3.4	Reactor Coolant Pump Shaft Break	Bounded	15.3.3	14.6
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES				
	15.4.1	Uncontrolled Control Rod/Bank Withdrawal from a Subcritical or Low Power Condition	Analyze		14.2
	15.4.2	Uncontrolled Control Rod/Bank Withdrawal at Power	Analyze		14.2
	15.4.3	Control Rod Misoperation			
		1) Dropped Control Rod/Bank	Analyze		14.4
		2) Dropped Part-Length Control Rod	Not Applicable		14.4

Table 2.1 Disposition of Events Summary (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
		3) Malpositioning of the Part-Length Control Rod Group	Not Applicable		14.4.5
		4) Statically Misaligned Control Rod/Bank	Not in Licensing Basis		
		5) Single Control Rod Withdrawal	Analyze		14.2
		6) Reactivity Control Device Removal Error During Refueling	Not Applicable		14.1.4
		7) Variations in Reactivity Load to be Compensated by Burnup or On-Line Refueling	Not Applicable		14.1.4
	15.4.4	Startup of an Inactive Loop	Not Applicable		14.7
	15.4.5	Flow Controller Malfunction	Not applicable; No Flow Controller		
	15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Analyze, Modes 1-6		14.3
	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Not in Licensing Basis		

Table 2.1 Disposition of Events Summary (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
	15.4.8	Spectrum of Control Rod Ejection Accidents	Analyze		14.13
	15.4.9	Spectrum of Rod Drop Accidents (BWR)	Not applicable; BWR Event		
15.5	INCREASES IN REACTOR COOLANT INVENTORY				
	15.5.1	Inadvertent Operation of the ECCS that Increases Reactor Coolant Inventory	Not in Licensing Basis		
	15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Not in Licensing Basis		
15.6	DECREASES IN REACTOR COOLANT INVENTORY				
	15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	Analyze		14.5
	15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment	Not Applicable		14.1.4
	15.6.3	Radiological Consequences of Steam Generator Tube Failure	Bounded	14.14	14.14
	15.6.4	Radiological Consequences of a Main Steamline Failure Outside Containment	Not applicable; BWR Event		

Table 2.1 Disposition of Events Summary (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
	15.6.5	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Analyze		14.15.3, 14.15.4
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT				
	15.7.1	Waste Gas System Failure	Bounded	14.17	14.17
	15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Not in Licensing Basis		
	15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Not in Licensing Basis		
	15.7.4	Radiological Consequences of Fuel Handling Accidents	Bounded	14.19	14.19
	15.7.5	Spent Fuel Cask Drop Accidents	Not in Licensing Basis		

Table 2.1 Disposition of Events Summary (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event	Updated FSAR Designation
	FSAR EVENTS NOT CONTAINED IN THE STANDARD REVIEW PLAN				
	(1)	Effects of External Events	Bounded	14.21	14.21
	(2)	Failures of Equipment Which Provide Joint Control/Safety Functions	Bounded	14.1.4	14.1.4
	(3)	Containment Pressure Analysis	Bounded	14.16	14.16
	(4)	Hydrogen Accumulation in Containment	Bounded	14.18	14.18
	(5)	Radiological Consequences of the Design Basis Incident (DBI)	Bounded	14.20	14.20

3.0 BASIS AND JUSTIFICATION FOR DISPOSITION OF EVENTS

This section presents the basis and justification for the disposition of events. The section numbers and event names are in accordance with those events described in the SRP.

Each event described in the SRP (and events in the FSAR but not in the SRP) is considered in accordance with the plant licensing basis and dispositioned into one of the four categories described in Section 1.0. Events which are not bounded by other events or by existing accepted analyses, and are in the plant licensing basis, are dispositioned to be analyzed. In the event disposition process, the event initiator is identified for each event. The magnitude of the initiator for each event is calculated and compared to the magnitude of the initiator for other events. The comparison basis includes all the plant operating conditions. This allows, in several cases, a ranking of the event initiators as to severity, allowing the lesser events to be dispositioned as bounded by the greater event. Similar logic is applied in determination of the applicability and bounding nature for existing accepted analyses.

The licensing basis for Millstone Unit 2 is as stated in the Updated Final Safety Analysis Report.[4] The formulation of event scenarios to be considered in the safety analysis depends on single failure criteria established by the plant licensing basis. Examination of the Millstone 2 licensing basis yields the following single failure criteria:

(1) The Reactor Protection System (RPS) is designed with redundancy and independence to assure that no single failure or removal from service of any component or channel of a system will result in the loss of the protection function.

(2) Each Engineered Safety Feature (ESF) is designed to perform its intended safety function assuming a failure of a single active component.

(3) The onsite power system and the offsite power system are designed such that each shall independently be capable of providing power for the ESF assuming a failure of a single active component in either power system.

The safety analysis is structured to demonstrate that the plant systems design satisfies these single failure criteria. The following assumptions result:

(1) The ESFs required to function in an event are assumed to suffer a worst single failure of an active component.

(2) Reactor trips occur at the specified setpoint within the specified delay time assuming a worst single active failure.

(3) The following postulated accidents are considered assuming a concurrent loss of offsite power: main steam line break, control rod ejection, and LOCA.

(4) The loss of normal feedwater, an anticipated operational occurrence, is analyzed assuming a concurrent loss of offsite power.

The requirements of 10 CFR 50, Appendix A, Criteria 10, [illegible] 25 and 29 require that the design and operation of the plant and the reactor protective system assure that the Specified Acceptable Fuel Design Limits (SAFDLs) not be exceeded during Anticipated Operational Occurrences (AOOs). As per the definition of AOO in 10 CFR 50, Appendix A, "Anticipated Operational Occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The Specified Acceptable Fuel Design Limits (SAFDLs) are that: 1) the fuel shall not experience centerline melt (~21 kW/ft); and 2) the departure from nucleate boiling ratio (DNBR) shall have a minimum allowable limit such that there is a 95% probability with a 95% confidence interval that departure from nucleate boiling (DNB) has not occurred (XNB DNBR of 1.17).

As indicated, three revisions to acceptable plant operating conditions are planned. The three revisions are:

(1) The maximum Technical Specification radial peaking factor is being increased from the current limit of 1.537 to 1.61.

(2) Plant cycle length is being increased from 12 to 18 month cycles.

   (a) The moderator temperature coefficient (MTC) in the Technical Specification is being changed in order to accommodate the increased cycle length. The MTC change for power less than or equal to 70% is from +5 to +7 pcm/°F. The MTC change for power greater than 70% is from $+4 \geq MTC \geq -24$ pcm/°F to $+4 \geq MTC \geq -28$ pcm/°F.

   (b) The shutdown margin requirements are being changed to offset the more negative end of cycle MTC. The change in MTC results in the modes 1, 2, 3 and 4 shutdown margin going from the current limit of $\geq 2.9\%$ to $\geq 3.6\%$.

(3) In order to cover end of cycle coastdowns, the analysis will also support plant operation at reduced inlet temperatures. The current nominal inlet temperature is 549°F and the analysis will support up to a 12°F inlet temperature reduction at full power. Greater temperature reduction is acceptable if concurrent with reduced power and pressurizer level during an EOC coastdown.

The event review and event analyses will be performed to insure that these revisions to acceptable plant operating conditions are supported.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Event Initiator - A decrease in feedwater temperature may be caused by loss of one of several feedwater heaters. The loss could be due to the interruption of steam extraction flow or to an accidental opening of a feedwater heater bypass line. The worst case loss of feedwater heaters would occur if all of the low pressure heaters were bypassed. The effects of any decrease in the feedwater temperature due to flow increases (Main or Auxiliary Feedwater) are discussed in Section 15.1.2.

15.1.1.2 Event Description - Due to a malfunction in the feedwater heater system, the enthalpy of the feedwater being injected into the steam generators is reduced. The increased subcooling of the feedwater reduces the secondary system average fluid enthalpy and increases the energy removal rate from the primary system. The increase in primary to secondary heat transfer causes the reactor coolant temperature at the outlet of the steam generator to decrease. This causes a corresponding decrease in the core inlet coolant temperature. With a negative moderator reactivity temperature coefficient, the reactor core power will begin to increase as the cooler moderator fluid reaches the core.

15.1.1.3 Reactor Protection - Reactor protection is provided by the variable overpower, thermal margin/low pressure, local power density, and low steam generator pressure trips. Reactor protection for the decrease in feedwater temperature event is summarized in Table 15.1.1-A.

15.1.1.4 Disposition and Justification - For operating Modes 1-3, the response of the nuclear steam supply system is governed by the magnitude of the overcooling introduced by the initiating event. There is no extraction to the feedwater heaters for operating Modes 4-6. As such, there is not a credible event for these reactor operating conditions.

The most limiting case for Mode 1 is from rated power conditions because the feedwater flow rate and heater duty decrease with load. Also, at rated power conditions, the initial DNBR margin is minimized. The consequences of the event in Modes 2 and 3 are bounded by those of Mode 1 because the magnitude of the initiating event in Modes 2 or 3 is much smaller than in Mode 1.

This cooldown related to bypassing the feedwater heaters is bounded by the cooldown produced by the limiting event postulated in Section 15.1.3. As such, the consequences of the Increase in Steam Flow event (15.1.3) bound the consequences for the Decrease in Feedwater Temperature event discussed in this section (15.1.1). The disposition of events for the Decrease in Feedwater Temperature event is summarized in Table 15.1.1-B.

15.1.2 Increase in Feedwater Flow

15.1.2.1 Event Initiator - This event is initiated by a failure in the feedwater system which causes an increase in the feedwater flow to the steam generators. The initiators considered are complete opening of a feedwater control valve, overspeed of the feedwater pumps, inadvertent start of a second feedwater pump at low power, startup of the auxiliary feedwater system, and inadvertent opening of the feedwater control valve bypass line.

15.1.2.2 Event Description - The increased flow to the steam generators causes an increase in the energy removal capability of the steam generators by reducing the average fluid enthalpy in the steam generators. The increased energy removal from the primary system causes the reactor coolant temperature at the outlet of the steam generator to decrease. The core inlet temperature will correspondingly be reduced, which will cause the core power to increase if the moderator temperature coefficient is negative.

Because this event is characterized as a primary system overcooling event, the primary system pressure initially decreases along with the core inlet temperature. There is also a possibility for a core power increase in the presence of a negative moderator reactivity feedback coefficient. Increased reactor power reduces the core DNB margin. A potential exists that the net effect of these three factors will represent a challenge to the core DNB margin.

15.1.2.3 Reactor Protection - Reactor protection for the rated power and power operation conditions (Mode 1) is provided by the variable overpower trip, local power density trip, thermal margin/low pressure trip, low steam generator pressure trip, and reactor trip on turbine trip due to high steam generator water level.

For Modes 2 and 3, protection is provided by the low steam generator pressure trip, safety injection actuation signal, and the variable overpower trip. Reactor protection for the Increase in Feedwater Flow event is summarized in Table 15.1.2-A.

15.1.2.4 Disposition and Justification - The event consequences at rated power operating conditions will bound the consequences from all other power operating conditions. At rated power operating conditions, the initial thermal margin (DNBR) is minimized. Maximizing the increase in feedwater flow maximizes the load demand. This results in the maximum rate of moderator cooldown which, in the presence of a negative moderator temperature coefficient, results in the maximum challenge to the specified acceptable fuel design limits (SAFDLs). Therefore, the limiting consequences of the increase in feedwater flow will occur at the full load rated power conditions and will bound all other power operating conditions due to the initial steam generator inventory and initial margin to DNB. The greatest cooldown which can be postulated due to feedwater addition at full power is the inadvertent startup of all three auxiliary feedwater pumps. This cooldown is larger than that due to the full opening of both feedwater control valves but less than that calculated for Event 15.1.3, Increased Steam Flow.

The main feedwater system is off-line in Modes 4-6 but may be on-line in Mode 3. For Mode 3 operating conditions, the potential cooldown in conjunction with a negative moderator temperature coefficient may result in a return to power at reduced primary pressure, elevated all-rods-in peaking, and less than four reactor coolant pump conditions. This case may pose a greater challenge to the SAFDLs than the full power case, and would bound zero power operation in Mode 2. This is due to the potential for prompt criticality in Mode 3. The greatest increase in feed flow would result from the startup of an idle pump with both control valves full open. The cooldown rate is less than the rate computed for Event 15.1.3 in Mode 3, and consequently Event 15.1.2 in Mode 3 is bounded by Event 15.1.3 initiated from Mode 3.

In Modes 4-6, the only increased feed flow event initiator is inadvertent startup of an auxiliary feedwater pump since the main feedwater system is off-line. The startup of all three auxiliary feedwater pumps results in an increased energy removal rate, less than that computed for the Increase in Steam Flow event (15.1.3) for Modes 4-6.

The disposition of events for the Increase in Feedwater Flow event is summarized in Table 15.1.2-B.

15.1.3 Increase in Steam Flow

15.1.3.1 Event Initiator - This event is initiated by a failure or misoperation in the main steam system which results in an increase in steam flow from the steam generators. This event could be caused by the rapid opening of the turbine control valves, the atmospheric steam dump valves, the steam bypass to condenser valves, or the safety relief valves (SRVs).

15.1.3.2 Event Description - The increase in steam flow creates a mismatch between the energy being generated in the reactor core and the energy being removed through the secondary system. This mismatch results in a cooldown of the primary system. A power increase will occur if the moderator temperature reactivity feedback coefficient is negative. The power increase will cause a decrease in the DNB margin.

15.1.3.3 Reactor Protection - The main steam system is designed to accommodate a 10% increase in load (step increase). Reactor protection against a main steam flow increase greater than a 10% step is provided by the following trip signals: variable overpower trip, thermal margin/low pressure trip, local power density trip, low steam generator water level trip, and low secondary pressure trip. Reactor protection for the Increase in Steam Flow event is summarized in Table 15.1.3-A.

15.1.3.4 Disposition and Justification - The atmospheric steam dump valves are sized to accommodate 15% of the steam flow at 2700 MWt. The steam dump to condenser valves and the turbine bypass valve are sized to accommodate 41% of the steam flow at 2700 MWt, and each SRV will pass 6.75% of the steam flow at 2700 MWt. The capacities of the control valves for the main feedwater and auxiliary feedwater pump turbines are significantly less. As such, the simultaneous opening of the steam dump valves and the turbine bypass valve could result in an increased load as great as 41% of the steam flow above the rated power operating condition of 2700 MWt. This energy removal rate bounds the rated power operating conditions for Events 15.1.1, 15.1.2, and 15.1.4. Therefore, this event will be analyzed as part of the plant transient analysis for Millstone Unit 2. The consequences of this event for all other power operating conditions, including Mode 2, are bounded by the rated power operating condition due to the increased margin to DNB at the other power operating conditions.

Since the Mode 3 operating condition has a higher average coolant temperature and a larger potential for cooldown than the Modes 4-6 reactor operating conditions, this condition represents the bounding event for the "zero power" initial conditions. It is evaluated to assess the potential for a return to power at reduced pressure conditions. Whereas the steam dump valves are available in Mode 3, MSIV closure results in opening of the atmospheric dump valves being the only increased steam flow event initiator in Modes 4-6. The analysis is performed from a hot zero power, all-rods-in plant state assuming a 3.6% $\Delta\rho$ shutdown margin and one to four operating reactor coolant pumps as allowed by the Technical Specifications (Reference 3).

The disposition of events for the Increase in Steam Flow event is summarized in Table 15.1.3-B. The event analysis is performed to support increased radial peaking and a more negative EOC moderator temperature coefficient.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Event Initiator - This event is initiated by an increase in steam flow caused by the inadvertent opening of a secondary side safety or relief valve.

15.1.4.2 Event Description - The resulting mismatch in energy generation and removal rates results in an overcooling of the primary system. If the moderator temperature coefficient is negative, the reactor power will increase.

15.1.4.3 Reactor Protection - Reactor protection is provided by the variable overpower trip, local power density trip, thermal margin/low pressure trip, low secondary pressure trip, and low steam generator water level trip. Reactor protection for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event is summarized in Table 15.1.4-A.

15.1.4.4 Disposition and Justification - The inadvertent opening of a steam generator safety valve would result in an increased steam flow of approximately 6.75% of full rated steam flow. Each dump (relief) valve is sized for approximately 7.50% steam flow with the reactor at full rated power. As such, the consequences of any of these occurrences will be bounded by the events in Section 15.1.3. The disposition of events for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event is summarized in Table 15.1.4-B.

15.1.5 Steam System Piping Failures Inside and Outside of Containment

15.1.5.1 Event Initiator - This event is initiated by a rupture in the main steam piping upstream of the MSIVs which results in an uncontrolled steam release from the secondary system.

15.1.5.2 Event Description - The increase in energy removal through the secondary system results in a severe overcooling of the primary system. In the presence of a negative moderator temperature coefficient, this cooldown causes a decrease in the shutdown margin (following reactor scram) such that a return to power might be possible following a steam line rupture. This is a potential problem because of the high power peaking factors which exist, assuming the most reactive control rod to be stuck in its fully withdrawn position.

15.1.5.3 Reactor Protection - Reactor protection is provided by the low steam generator pressure and water level trips, variable overpower trip, local power density trip, thermal margin/low pressure trip, high containment pressure trip, and safety injection actuation signal. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 15.1.5-A.

15.1.5.4 Disposition and Justification - At rated power conditions, the stored energy in the primary coolant is maximized, the available thermal margin is minimized, and the pre-trip power level is maximized. These conditions result in the greatest potential for cooldown and provide the greatest challenge to the SAFDLs. Initiating this event from rated power also results in the highest post-trip power since it maximizes the concentration of delayed neutrons providing for the greatest power rise for a given positive reactivity insertion. Additional thermal margin is also provided at lower power levels by the automatically decreasing setpoint of the variable overpower trip. Thus, this event initiated from rated power conditions will bound all other cases initiated from at power operation modes.

For the zero power and subcritical plant states (Modes 2-6), there is a potential for a return-to-power at reduced pressure conditions. The most limiting steam line break event at zero power is one which is initiated at the highest temperature, thereby providing the greatest capacity for cooldown. This occurs in Modes 2 and 3. Thus, the event initiated from Modes 2 and 3 will bound those initiated from Modes 4-6. Further, the limiting initial conditions will occur when the core is just critical. These conditions will maximize the available positive reactivity and produce the quickest and largest return to power. Thus, the steam line break initiated from critical conditions in Mode 2 will bound the results of the event initiated from subcritical Mode 3 conditions.

The Technical Specifications (Reference 3) only require a minimum of one reactor coolant pump to be operating in Mode 3. One pump operation provides the limiting minimum initial core flow case. Minimizing core flow minimizes the clad to coolant heat transfer coefficient and degrades the ability to remove heat generated within the fuel pins. Conversely, however, a maximum loop flow will maximize the primary to secondary heat transfer coefficient, thus providing for the greatest cooldown. Higher loop flow will sweep the cooler fluid into the core faster, maximizing the rate of positive reactivity addition and the peak power level.

The worst combination of conditions is achieved for the four pump loss of offsite power case. In this situation, the initial loop flow is maximized resulting in the greatest initial cooldown, while the final loop flow is minimized providing the greatest challenge to the DNB SAFDL. Since the natural circulation flow which is established at the end of the transient will be the same regardless of whether one or four pumps were initially operating, the results of the four pump loss of offsite power case will bound those of the one pump case. Thus, only four pump operation need be analyzed for the Mode 2 case.

The event is analyzed to support a more negative moderator temperature coefficient. This event must be analyzed both with and without a coincident loss-of-offsite power. Typically, there are two single failures which are considered for the offsite power available case. The first is failure of a HPSI pump to start. The second is failure of an MSIV to close, resulting in a continued uncontrolled cooldown. However, Millstone 2 has combination MSIV/swing disc check valves. A double valve failure would thus be required for steam from the intact steam generator to reach the break. This is not deemed credible. Thus, the single failure to be considered with offsite power available is failure of a HPSI pump to start. For the loss-of-offsite power case, the limiting single failure is the failure of a diesel generator to start. This is assumed to result in the loss of one HPSI pump and one charging pump. The disposition of events for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 15.1.5-B.

TABLE 15.1.1-A AVAILABLE REACTOR PROTECTION FOR DECREASE IN FEEDWATER TEMPERATURE EVENT

Reactor Operating Conditions	Reactor Protection
1	Variable Overpower Trip Thermal Margin/Low Pressure Trip Local Power Density Trip Low Steam Generator Pressure Trip
2	Variable Overpower Trip Low Steam Generator Pressure Trip
3	Variable Overpower Trip
4-6	Not a credible event for these reactor operating conditions since there is no extraction steam to the feedwater heaters

TABLE 15.1.1-8 DISPOSITION OF EVENTS FOR THE DECREASE IN FEEDWATER TEMPERATURE EVENT

Reactor Operating Conditions	Disposition
1	Bounded by Event 15.1.3, Increase in Steam Flow Event
2, 3	Bounded by the above
4-6	No analysis required; not a credible event

TABLE 15.1.2-A AVAILABLE REACTOR PROTECTION FOR THE INCREASE IN FEEDWATER FLOW EVENT

Reactor Operating Condition	Reactor Protection
1	Variable Overpower Trip
	Local Power Density Trip
	Thermal Margin/Low Pressure Trip
	Low Steam Generator Pressure Trip
	Safety Injection Actuation Signal
	Reactor Trip on Turbine Trip due to High Steam Generator Water Level
2	Low Steam Generator Pressure Trip
	Variable Overpower Trip
	Safety Injection Actuation Signal
3	Variable Overpower Trip
	Safety Injection Actuation Signal
4	Tech. Spec. Requirements on Shutdown Margin
	Inherent Negative Doppler Feedback
5, 6	No analysis required; no significant consequences

TABLE 15.1.2-B DISPOSITION OF EVENTS FOR THE INCREASE IN FEEDWATER FLOW EVENT

Reactor Operating Condition	Disposition
1	Bounded by Event 15.1.3 (Increase in Steam Flow)
2	Bounded by the Mode 3 case
3-6	Bounded by Event 15.1.3

TABLE 15.1.3-A AVAILABLE REACTOR PROTECTION FOR THE INCREASE IN STEAM FLOW EVENT

Reactor Operating Condition	Reactor Protection
1	Low Steam Generator Pressure Trip
	Low Steam Generator Water Level Trip
	Thermal Margin/Low Pressure Trip
	Local Power Density Trip
	Variable Overpower Trip
	Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip
	Low Steam Generator Water Level Trip
	Variable Overpower Trip
	Safety Injection Actuation Signal
3	Variable Overpower Trip
	Safety Injection Actuation Signal
4	Tech. Spec. Requirements on Shutdown Margin
	Inherent Negative Doppler Feedback
5, 6	No analysis required; no significant consequences

TABLE 15.1.3-8 DISPOSITION OF EVENTS FOR THE INCREASE IN STEAM FLOW EVENT

Reactor Operating Condition	Disposition
1	Analyze
2	Bounded by the above
3	Analyze
4	Bounded by the above
5, 6	No analysis required; no significant consequences

TABLE 15.1.4-A AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE EVENT

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Variable Overpower Trip Local Power Density Trip Thermal Margin/Low Pressure Trip Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Variable Overpower Trip Safety Injection Actuation Signal
3, 4	Tech. Spec. requirements on shutdown margin, inherent negative Doppler feedback
5, 6	No analysis required; not a credible event

TABLE 15.1.4-B DISPOSITION OF EVENTS FOR THE INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE EVENT

Reactor Operating Conditions	Disposition
1-4	Bounded by analyses presented for Event 15.1.3
5, 6	Not a credible event; no analysis required

TABLE 15.1.5-A AVAILABLE REACTOR PROTECTION FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT EVENT

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Pressure Trip
	Low Steam Generator Water Level Trip
	Variable Overpower Trip
	Local Power Density Trip
	Thermal Margin/Low Pressure Trip
	High Containment Pressure Trip
	Safety Injection Actuation Signal
2	Low Steam Generator Pressure Trip
	Low Steam Generator Water Level Trip
	Variable Overpower Trip
	High Containment Pressure Trip
	Safety Injection Actuation Signal
3-6	Tech. Spec. requirements on shutdown margin, inherent negative Doppler feedback

TABLE 15.1.5-B DISPOSITION OF EVENTS FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT EVENT

Reactor Operating Conditions	Disposition
1	Analyze
2	Analyze
3-6	Bounded by the above

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.2.1 Loss of External Load

15.2.1.1 Event Initiator - A major loss of load can be initiated as the result of a loss of external electrical load or a turbine trip. Turbine stop valve closure is assumed as the initiator of this event because this is the fastest load rejection which can be postulated which will challenge the plant overpressure and SAFDL protection features. The assumed fast valve closure time (0.1 sec.) and the assumed unavailability of the steam dump system allow this event to bound the effects of Events 15.2.2 (Turbine Trip - Steam bypass system available) and 15.2.4 (Closure of the MSIV - Valve closure time >0.1 sec.).

15.2.1.2 Event Description - For a full load reduction at power, the primary to secondary heat transfer would be severely diminished because of the increase in secondary side temperature. Initially, in response to the load reduction and diminished energy removal through the secondary system, the primary system temperatures begin to increase. The increasing primary system average temperature causes an insurge into the pressurizer due to the expanding primary fluid. The primary system pressure increases as the pressurizer steam space is compressed by the insurging liquid. Primary system overpressure protection is afforded by the pressurizer power operated relief valves and the primary safety valves. Eventually, the secondary system pressure reaches the opening setpoint of the secondary side safety valves and steam discharge occurs to limit the secondary side pressure rise. Energy removal through the steam generator and pressurizer safety valves mitigates the consequences of the load reduction. However, in analyzing the overpressurization aspects of this event, no credit is taken for the power operated relief valves on the primary or secondary systems.

15.2.1.3 Reactor Protection - Reactor protection is provided by the high pressurizer pressure, variable overpower, thermal margin/low pressure, and low steam generator water level trips. If the turbine is tripped at the initiation of this event, a direct reactor trip signal would be generated and the effects of this event would be mitigated. However, no credit is taken for a direct reactor trip on turbine trip. Additionally, reactor protection is provided by the primary and secondary safety valves. Because of the potential for increasing the primary system temperatures, with small increases in pressure, this event can challenge the SAFDLs as well as the overpressure criteria mentioned above. Reactor protection for the Loss of External Load event is summarized in Table 15.2.1-A.

15.2.1.4 Disposition and Justification - This event is only credible for rated power and power operating conditions because there is no load on the turbine at other reactor conditions. The consequences of this event for rated power operation bound the consequences for other reactor conditions because of the maximum stored energy in the primary coolant, minimum initial thermal margin and maximum power to load mismatch which occurs upon load loss.

This event will be analyzed for Millstone Unit 2 from rated power conditions to support the proposed increased radial peaking and a more positive BOC moderator temperature coefficient. Additionally, the analysis of the overpressurization aspects must consider plant operating conditions representative of the end of cycle (EOC) coastdown.

There is no single failure considered which could worsen the results. The disposition of events for the Loss of External Load event is summarized in Table 15.2.1-B.

15.2.2 Turbine Trip

15.2.2.1 Event Initiator - This event is initiated by a turbine trip which results in closure of the main steam stop valves and a rapid reduction in energy removal through the steam generators.

15.2.2.2 Event Description - The reactor protection system is designed to generate a reactor trip signal automatically when the turbine is tripped. Following reactor trip, there would be a rapid decrease in the energy being generated in the primary system. This would mitigate the consequences of the turbine trip event. Primary and secondary system overpressurization protection is provided by the code safety valves on both the primary and secondary systems and the secondary atmospheric dump valves. Also, if the condenser was available, the steam bypass system would be activated to reduce the secondary system pressure.

15.2.2.3 Reactor Protection - Reactor protection is provided by the high pressurizer pressure trip, variable overpower trip, thermal margin/low pressure trip, low steam generator water level trip, and a nonsafety grade reactor trip on turbine trip. Additional protection is also provided by the primary and secondary side safety valves. Reactor protection for the Turbine Trip event is summarized in Table 15.2.2-A.

15.2.2.4 Disposition and Justification - This event is only credible for rated power and power operating conditions since the turbine will either be in a tripped condition or there will be no load on the steam generators for other reactor operation conditions. The consequences of this event for rated power operation bound the event consequences for other operating conditions because of the higher initial stored energy in the primary system, maximum power to load mismatch potential, and the reduced SAFDL margin for rated power operation. Because of the limiting assumptions used in the analysis of the consequences of the Loss of External Load event (15.2.1), the consequences of the Turbine Trip event are bounded by the consequences of Event 15.2.1, which is to be analyzed for Millstone Unit 2. The major assumptions used in Event 15.2.1 are the conservatively rapid turbine stop valve closure time, the failure to trip the reactor on turbine trip, and the assumed unavailability of the atmospheric steam dump system. The disposition of events for the Turbine Trip event is summarized in Table 15.2.2-B.

15.2.3 Loss of Condenser Vacuum

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.2.3-A and 15.2.3-B, respectively.

15.2.4 Closure of the Main Steam Isolation Valves (MSIVs)

15.2.4.1 Event Initiator - The event postulated is the loss of control air to the MSIV valve operator. Either one or both MSIVs may inadvertently close. The valves are swinging disc type check valves, installed in a reversed position and held open against steam flow by a pneumatically operating cylinder assembly. The valves are spring-loaded to the closed position.

15.2.4.2 Event Description - The inadvertent MSIV closure is primarily of concern in boiling water reactors as indicated in the SRP (Ref. 1), but closure of the MSIVs in a PWR would cause a drastic reduction in the load on the reactor. As such, the consequences of a dual MSIV closure are similar to the consequences of Event 15.2.1. Although the valve closure time for the MSIVs is less than 6 seconds, this is much longer than the turbine stop valve closure time assumed in Event 15.2.1 (0.1 seconds); as such, the transient events will proceed somewhat slower and be less severe than in the case of Event 15.2.1.

A single MSIV closure will result in an asymmetric primary temperature distribution. Upon cessation of steam flow, the pressure in the affected steam generator will increase to the opening setpoint of the steam line safety valves. The primary to secondary heat transfer will be diminished, resulting in a heatup of the associated primary side loop. In response to the drop in steam flow through the turbine control valves, the steam flow out of the unisolated steam generator will increase. Depressurization of the steam generator will result, causing an increase in heat removal from the primary. The associated primary side loop will thus experience a cooldown. The side of the ... ected to the cooldown will experience a power rise in the pre... negative moderator temperature coefficient.

15.... actor Protection - Reactor protection for the dual MSIV closure is provided by the high pressurizer pressure trip, variable overpower trip, thermal margin/low pressure trip, low steam generator pressure trip, and low steam generator water level trip. Additional protection is provided by the primary and secondary side safety valves.

Reactor protection for the single MSIV closure is provided by the low steam generator level and low steam generator pressure trips. Due to the location of the excore detectors and the asymmetries associated with this event, the variable overpower and TM/LP trip may not get the required 2 out of 4 channels tripped. Further, since one loop will be cooling down and one will be heating up, the pressure may be either increasing or decreasing. Thus, this event cannot take credit for the variable high power, TM/LP or high pressure trips. Additional protection continues to be provided by the primary and secondary side safety valves.

15.2.4.4 Disposition and Justification - This event is not credible in Modes 4-6 as the MSIVs are closed. For simultaneous closure of both MSIVs, the event will progress very similarly to Event 15.2.1. As such, the limiting case is obtained when the event is initiated from rated full power conditions. The turbine stop valve closure time employed in the 15.2.1 analysis (0.1 sec) is much smaller than the MSIV closure time (6 sec). Thus, the consequences of Event 15.2.1 will bound those of the dual MSIV closure event.

The asymmetric conditions resulting from the closure of only one of the two MSIVs is similar to that predicted for a steam line break. That is, the primary coolant loop associated with the closed MSIV experiences a heatup due to the loss of heat sink and the primary coolant loop associated with the open MSIV experiences a cooldown due to the perceived load increase. The temperature increase seen by the hot loop will be limited by the actuation of the steam generator safety valves. The temperature decrease seen by the cooling loop will continue until such time as a reactor trip is generated.

Since the loop experiencing the cooldown will see the larger temperature change, the limiting conditions for the event are at end of cycle. The end of cycle moderator temperature coefficient (MTC) is larger in absolute magnitude than the beginning of cycle MTC. When the larger MTC is coupled with the larger temperature change in the cooling loop, a larger overall increase in core power will be predicted. This larger increase in core power will produce the limiting DNB conditions for the event.

Since the asymmetries associated with the event preclude taking credit for the high pressure or variable overpower trip, the single MSIV closure cannot be bounded a priori by the loss of load, Event 15.2.1. Further, since the loop that is experiencing the heatup is actually driving the event instead of the loop experiencing the cooldown, the event cannot be bounded a priori by the steam line break. Thus, it is concluded that the event will be analyzed.

The limiting single MSIV closure case is that which is initiated from rated power in Mode 1. There is no Mode 3 concern, as in this mode the magnitude of interruption in steam flow is not great enough to cause any significant consequences. There is also no potential for a post-trip return to power since the remaining MSIV and the turbine stop valves provide redundant means for terminating the remaining steam flow. There is no single failure considered which could worsen the results. The disposition of events for the Closure of the Main Steam Isolation Valves event is summarized in Table 15.2.4-B.

15.2.5 Steam Pressure Regulator Failure

Millstone Unit 2 does not have any steam line pressure regulators, so this event is not credible for this plant. No analysis needs to be considered for this event.

15.2.6 Loss of Nonemergency A.C. Power to the Station Auxiliaries

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.2.6-A and 15.2.6-B, respectively.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Event Initiator - The Loss of Normal Feedwater Flow transient is initiated by a trip of the main feedwater pumps or a malfunction in the feedwater control valves.

15.2.7.2 Event Description - The Loss of Normal Feedwater Flow event results in a total loss of all main feedwater flow to the steam generators. Because the main feedwater system is supplying subcooled water to the steam generators, the loss of main feedwater flow will result in a reduction of the secondary system heat removal capability. The decrease in energy removal rate from the primary system causes the primary system fluid temperature to increase. The resulting primary system fluid expansion results in an insurge into the pressurizer, compressing the steam space and causing the primary system pressure to increase.

The RCS pressure and temperature rise until a reactor trip occurs either due to low steam generator water level or high pressurizer pressure. Assuming the steam bypass control system is in the manual mode of operation, termination of main steam flow due to closure of the turbine stop valves following reactor trip temporarily causes steam generator and RCS pressurization. The decrease in core heat rate after insertion of the CEAs, in combination with the main steam safety valves opening, restores the RCS to a new steady state condition. Auxiliary feedwater flow is automatically initiated on a low steam generator water level assuring sufficient steam generator inventory to prevent steam generator dryout and provide for core decay heat removal.

15.2.7.3 Reactor Protection - System overpressure protection is provided by the primary and secondary system safety valves. A reactor trip occurs on low steam generator level with additional reactor protection provided by the high pressurizer pressure trip, variable overpower trip, and the thermal margin/low pressure trip. Reactor protection for the Loss of Normal Feedwater Flow event is summarized in Table 15.2.7-A.

15.2.7.4 Disposition and Justification - This event is only credible for rated power and power operating conditions because the main feedwater system is not required to provide feedwater to the steam generators for other reactor operating conditions. The consequences of this event for rated power operation bound the consequences for other conditions because of the higher initial stored energy in the primary system, the minimum steam generator inventory, and the greater impact of the loss of feedwater flow on the secondary system.

The near term pressurization and DNB aspects of this event are bounded by those of Event 15.2.1. This is due to fact that in the analytical methodology for Event 15.2.7 given in Reference 2, it is indicated that reactor trip occurs at time zero coincident with turbine trip on a low steam generator water level signal. In Event 15.2.1, reactor trip is delayed until a high pressurizer pressure signal is received. This results in a higher power level at trip, greater pressurization and greater challenge to the SAFDLs than in Event 15.2.7. Long term pressurization, if it occurs, is very gradual and is arrested by opening of the pressurizer code safety valves.

The Loss of Normal Feedwater event will be analyzed to assess the maximum expected pressurizer level swell and the long term adequacy of the auxiliary feedwater system to restore and maintain steam generator inventory and prevent steam generator dryout. The maximum level swell will be examined to assure that the pressurizer does not become water solid. The analysis must consider plant operating conditions representative of the end of cycle coastdown as described for Event 15.2.1.

The analysis will support a more positive BOC moderator temperature coefficient. The single failures considered in this analysis are failure of an auxiliary feedwater pump to start, or a loss of offsite power resulting in coastdown of the reactor coolant pumps. The disposition of events for the Loss of Normal Feedwater Flow event is summarized in Table 15.2.7-B.

15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.2.8-A and 15.2.8-B, respectively.

TABLE 15.2.1-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF EXTERNAL LOAD EVENT

Reactor Operating Conditions	Reactor Protection
1	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	Low Steam Generator Water Level Trip
2	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Low Steam Generator Water Level Trip
3-6	No analysis required; not a credible event.

TABLE 15.2.1-B DISPOSITION OF EVENTS FOR THE LOSS OF EXTERNAL LOAD EVENT

Reactor Operating Conditions	Disposition
1	Analyze
2	Bounded by the above, no analysis required.
3-6	No analysis required; not a credible event.

TABLE 15.2.2-A AVAILABLE REACTOR PROTECTION FOR THE TURBINE TRIP EVENT

Reactor Operating Conditions	Reactor Protection
1	High Pressurizer Pressure Trip
	Nonsafety Grade Reactor Trip on Turbine Trip
	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	Low Steam Generator Water Level Trip
2	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Low Steam Generator Water Level Trip
3-6	No analysis required; not a credible event.

TABLE 15.2.2-B DISPOSITION OF EVENTS FOR THE TURBINE TRIP EVENT

Reactor Operating Conditions	Disposition
1	Bounded by Event 15.2.1 for the rated power operating condition (#1).
2	Same as above.
3-6	No analysis required; not a credible event.

TABLE 15.2.3-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF CONDENSER VACUUM EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.2.3-B DISPOSITION OF EVENTS FOR THE LOSS OF CONDENSER VACUUM EVENT

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis; not analyzed.

TABLE 15.2.4-A AVAILABLE REACTOR PROTECTION FOR THE CLOSURE OF THE MSIVS EVENT

Reactor Operating Conditions	Reactor Protection
1	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	Low Steam Generator Water Level Trip
2	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Low Steam Generator Water Level Trip
3	Variable Overpower Trip
4-6	No analysis required; not a credible event.

TABLE 15.2.4-B DISPOSITION OF EVENTS FOR THE CLOSURE OF THE MSIVs EVENT

Reactor Operating Conditions	Disposition
1	Dual MSIV closure: bounded by Event 15.2.1. Single MSIV closure: analyze
2, 3	Bounded by Mode 1.
4-6	No analysis required; not a credible event.

TABLE 15.2.5-A AVAILABLE REACTOR PROTECTION FOR THE STEAM PRESSURE REGULATOR FAILURE EVENT

Reactor Operating Conditions	Reactor Protection
1-6	None required, not a credible event for this plant.

TABLE 15.2.5-B DISPOSITION OF EVENTS FOR THE STEAM PRESSURE REGULATOR FAILURE EVENT

Reactor Operating Conditions	Disposition
1-6	No analysis required.

TABLE 15.2.6-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF NONEMERGENCY A.C. POWER TO THE STATION AUXILIARIES EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.2.6-B DISPOSITION OF EVENTS FOR THE LOSS OF NONEMERGENCY A.C. POWER TO THE STATION AUXILIARIES EVENT

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis; not analyzed.

TABLE 15.2.7-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF NORMAL FEEDWATER FLOW EVENT

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Water Level Trip
	High Pressurizer Pressure Trip
	Thermal Margin/Low Pressure Trip
	Variable Overpower Trip
2	High Pressurizer Pressure Trip
	Variable Overpower Trip
	Low Steam Generator Water Level Trip
3	Variable Overpower Trip
4-6	No analysis required; not a credible event.

TABLE 15.2.7-B DISPOSITION OF EVENTS FOR THE LOSS OF NORMAL FEEDWATER FLOW EVENT

Reactor Operating Conditions	Disposition
1	Analyze to assess maximum pressurizer level swell and long term adequacy of AFW. Pressurization and DNB aspects bounded by Event 15.2.1.
2, 3	Bounded by the above, no analysis required.
4-6	No analysis required; not a credible event.

TABLE 15.2.8-A AVAILABLE REACTOR PROTECTION FOR THE FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.2.8-B DISPOSITION OF EVENTS FOR THE FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT EVENT

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis; not analyzed.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW

15.3.1 Loss of Forced Reactor Coolant Flow

15.3.1.1 Event Initiator - The loss of forced reactor coolant flow in the primary system may result from a mechanical or electrical failure in a main reactor coolant pump or in the power supply to these pumps. Forced coolant flow may be completely or partially lost. The limiting event initiator is that which results in the trip of all four reactor coolant pumps.

15.3.1.2 Event Description - The immediate result of the loss of forced coolant flow is an increase in the coolant temperature as it flows through the reactor core. If the reactor is at power, this temperature increase could challenge the specified acceptable fuel design limits. DNB could result if the reactor is not tripped.

15.3.1.3 Reactor Protection - Reactor protection is provided by the following reactor trips:

(1) Low reactor coolant flow;
(2) Low reactor coolant pump speed;
(3) Thermal margin/low pressure; and
(4) High pressurizer pressure trip.

Reactor protection for the Loss of Forced Reactor Coolant Flow event is summarized in Table 15.3.1-A.

15.3.1.4 Disposition and Justification - The power sources for the main reactor coolant pumps are the most likely initiator for a loss of flow event involving more than one pump. A mechanical or electrical fault in one of the pumps will only result in a single pump loss of forced coolant flow transient. The normal power supplies for the pumps are from two buses which receive power from the main generator. Two pumps, in opposite loops, are powered from each bus. If there is a generator trip, the pumps are automatically transferred to a bus supplied from the external power lines. A generator trip with the failure of this transfer could result in a loss of power to all four pumps.

In the case of four pump operation, two situations must be considered: two pump coastdown and a total loss of forced coolant flow. Considering first the total loss of flow cases, the consequences of this postulated event are bounded by rated power operation. The rated power case is bounding because of the reduced DNB margin for this initial state combined with tne highest power to flow ratio during coastdown.

For the two pump loss of flow cases, the magnitude of the coastdown is less severe than the four pump coastdown, and the consequences of this event are bounded by the four pump loss of flow event. For the two pump flow coastdown cases, there is always some degree of forced reactor coolant flow. These events are, therefore, not as challenging as the four pump coastdown events. A comparison of the governing parameters indicates that these events are bounded by the four pump loss of flow event from full rated power conditions.

In summary, the four pump loss of flow event is the bounding event for the 15.3.1 events in all modes of operation. It will be analyzed to support increased radial peaking and a more positive BOC moderator temperature coefficient. The only active system challenged is the reactor protection system which is redundant and single failure proof.

The disposition of events for the Loss of Forced Reactor Coolant Flow event is summarized in Table 15.3.1-B.

15.3.2 Flow Controller Malfunction

There are no flow control devices on the primary reactor coolant system of Millstone Unit 2. This event is therefore not credible and need not be analyzed.

15.3.3 Reactor Coolant Pump Rotor Seizure

15.3.3.1 Event Initiator - This event is initiated by an instantaneous seizure of a reactor coolant pump rotor.

15.3.3.2 Event Description - Flow in the affected loop will be rapidly reduced causing the core flow to also decrease rapidly. As in the 15.3.1 events, the reduction in primary reactor coolant flow will result in the increase in primary coolant temperatures and a challenge to the DNB margin. A pressurization of the primary system will also occur due to the heatup of the primary coolant which causes a rapid insurge into the pressurizer. A low reactor coolant flow trip will be generated.

15.3.3.3 Reactor Protection - Reactor protection for the reactor coolant pump rotor seizure event is provided by the low reactor coolant flow trip, thermal margin/low pressure trip, and the high pressurizer pressure trip.

Reactor protection for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 15.3.3-A.

15.3.3.4 Disposition and Justification - This event is a concern for only rated power and power operating conditions because for other reactor operating conditions there is sufficient thermal margin so there will not be a challenge to the fuel design limits. The core heat flux to flow ratio is an excellent indicator of the potential DNB challenge for a loss of flow event. The highest ratios for this event are predicted to occur during the first few seconds of the transient from full power rated operating conditions. The consequences of this event will therefore be bounded by a pump rotor seizure event initiated from full power rated conditions. There is no single failure considered which could worsen the results. The event is analyzed to support increased radial peaking and a more positive BOC moderator temperature coefficient.

The disposition of events for the Reactor Coolant Pump Rotor Seizure event is summarized in Table 15.3.3-B.

15.3.4 Reactor Coolant Pump Shaft Break

This event is not in the current licensing basis for Millstone Unit 2 and is, therefore, not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.3.4-A and 15.3.4-B.

TABLE 15.3.1-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF FORCED REACTOR COOLANT FLOW EVENT

Reactor Operating Conditions	Reactor Protection
1 (4 pump operation)	Low Reactor Coolant Flow Trip
	Low Reactor Coolant Pump Speed Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
2 (4 pump operation)	High Pressurizer Pressure Trip Tech. Spec. requirements on number of operating pumps
3-6 (less than 4 pump operation)	High Pressurizer Pressure Trip Tech. Spec. requirements on number of operating pumps

TABLE 15.3.1-B DISPOSITION OF EVENTS FOR THE LOSS OF FORCED REACTOR COOLANT FLOW EVENT

Reactor Operating Conditions	Disposition
1	Analyze
2-6	Bounded by the above, no analysis required.

TABLE 15.3.2-A AVAILABLE REACTOR PROTECTION FOR THE FLOW CONTROLLER MALFUNCTION EVENT

Reactor Operating Conditions	Reactor Protection
1-6	No analysis required, event not credible.

TABLE 15.3.2-B DISPOSITION OF EVENTS FOR THE FLOW CONTROLLER MALFUNCTION EVENT

Reactor Operating Conditions	Disposition
1-6	Not a credible event, no analysis required.

TABLE 15.3.3-A AVAILABLE REACTOR PROTECTION FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

Reactor Operating Conditions	Reactor Protection
1	Low Reactor Coolant Flow Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
2	High Pressurizer Pressure Trip
	Available Thermal Margin
3-6	Available Thermal Margin

TABLE 15.3.3-B DISPOSITION OF EVENTS FOR THE REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

Reactor Operating Conditions	Disposition
1	Analyze
2	Bounded by the above.
3-6	No analysis required.

TABLE 15.3.4-A AVAILABLE REACTOR PROTECTION FOR THE REACTOR COOLANT PUMP SHAFT BREAK EVENT

Reactor Operating Conditions	Reactor Protection
1-6	No analysis required; not in licensing basis.

TABLE 15.3.4-B DISPOSITION OF EVENTS FOR THE REACTOR COOLANT PUMP SHAFT BREAK EVENT

Reactor Operating Conditions	Disposition
1-6	No analysis required; not in licensing basis.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Uncontrolled Control Rod/Bank Withdrawal From a Subcritical or Low Power Startup Condition

15.4.1.1 Event Initiator - Initiated by the uncontrolled withdrawal of a control rod/bank, this event results in the insertion of positive reactivity and consequently a power excursion. This event could be caused by a malfunction in the reactor control or rod control systems. The consequences of a single bank withdrawal from operating Modes 3-6 are considered in this event category; the consequences at rated power and power operating initial conditions are considered in Event 15.4.2.

The control rods are wired together into preselected bank configurations. These circuits prevent the control rods from being withdrawn in other than their respective banks. Power is supplied to the banks in such a way that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence.

15.4.1.2 Event Description - The neutron flux rises very rapidly in response to the continuous positive reactivity insertion. The initial rapid rise is terminated by the reactivity feedback effect of the negative Doppler coefficient. The number of reactor coolant pumps in operation can significantly affect the ability to remove the heat generated in the fuel due to the power increase.

15.4.1.3 Reactor Protection - The power transient is eventually terminated (as well as the control rod withdrawal) by the reactor protection system on one of the following signals:

(1) Variable overpower trip;
(2) High pressurizer pressure trip; or
(3) Variable overpower pre-trip alarm, which initiates Rod Withdrawal Prohibit Action.

Reactor protection for the Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Startup Condition event is summarized in Table 15.4.1-A.

15.4.1.4 <u>Disposition and Justification</u> - The Technical Specifications for Millstone Unit 2 require that the control rod drives be de-energized in Modes 4-6 whenever the Reactor Coolant System boron concentration is less than the refueling concentration of 1720 ppm (Reference 3, pg. 3/4.1-31). A rod withdrawal from these modes is therefore not considered a credible event.

Whenever the rod control system is energized, Technical Specifications require 4 operating reactor coolant pumps, although pressure is allowed to be as low as 2000 psia in Mode 3 per Reference 3, pg. 3/4.1-3. The greatest power rise for this event is obtained when it is initiated from the lowest power.

Therefore, the event initiated from a low power critical Mode 2 condition at 2000 psia will bound all other low power or subcritical cases. The only active system challenged in this event is the reactor protection system, which is redundant and single failure proof. The event will be analyzed to support increased radial peaking and a more positive moderator temperature coefficient.

The disposition of events for the Uncontrolled Control Rod/Bank Withdrawal from a Subcritical or Low Power Startup Condition event is summarized in Table 15.4.1-B.

15.4.2 <u>Uncontrolled Control Rod/Bank Withdrawal at Power</u>

15.4.2.1 <u>Event Initiator</u> - This event is initiated by an uncontrolled control rod/bank withdrawal from power operating conditions.

15.4.2.2 Event Description - Positive reactivity is added to the reactor core due to the uncontrolled bank withdrawal resulting in a power transient. The increase in core power results in an increase in the core heat flux creating a challenge to the DNB margin. The challenge to the DNB margin is further accentuated by the mismatch between the energy removal from the steam generators and the power produced in the core. This mismatch in power causes the primary system temperatures to rise, reducing the DNB margin.

15.4.2.3 Reactor Protection - The challenge to the fuel design limits is terminated by the automatic action of the reactor protection system which terminates the bank withdrawal and inserts negative reactivity to terminate the power transient. The automatic action of the reactor protection system is initiated as the result of one of the following signals:

(1) Variable overpower trip;
(2) Local power density trip;
(3) Thermal margin/low pressure trip; or
(4) High pressurizer pressure trip.

Reactor protection for the Uncontrolled Control Rod/Ban Withdrawal at Power event is summarized in Table 15.4.2-A.

15.4.2.4 Disposition and Justification - This event is designed to address the safety challenge posed by an uncontrolled control rod/bank withdrawal transient from power conditions. This event addresses all the power operating conditions and the rated power operating conditions. It is performed to test the adequacy of the variable overpower and thermal margin/low pressure (TM/LP) trip setpoints in mitigating the challenge to the SAFDLs.

A rod withdrawal initiated from lower powers will provide less of a challenge to the SAFDLs due to increased initial thermal margin, a lesser amount of setpoint overshoot, and a decreased variable overpower trip setpoint resulting in a greater thermal margin at trip. The event initiated from full power will then bound those initiated from lower power conditions. This event will therefore be analyzed at full power for conditions ranging from BOC to EOC, for a spectrum of reactivity insertion rates. The only active system challenged by this event is the reactor protection system, which is redundant and single failure proof. This analysis will support increased radial peaking and a more positive BOC and a more negative EOC moderator temperature coefficient.

The disposition of events for the Uncontrolled Control Rod/Bank Withdrawal at Power event is summarized in Table 15.4.2-B.

15.4.3 Control Rod Misoperation

15.4.3.1 Event Initiator - The control rod misoperation event encompasses a number of transients resulting from different event initiators. The specific events addressed under this event category include the following:

(1) Dropped control rod or control rod bank;
(2) Dropped part-length control rod;
(3) Malpositioning of the part-length control rod group;
(4) Statically misaligned control rod/control rod bank;
(5) Single control rod withdrawal;
(6) Reactivity control device removal error during refueling; and
(7) Variations in reactivity load to be compensated by burnup or on-line refueling.

(1) Dropped Control Rod/Bank

15.4.3.1(1) Event Initiator - The dropped control rod and dropped control bank events are initiated by a de-energized control rod drive mechanism or by a malfunction associated with a control rod bank.

15.4.3.2(1) Event Description - In the dropped control rod event, the reactor power initially drops in response to the insertion of negative reactivity.

However, the local peaking increases due to the local effect on the power distribution. The reactor core will attempt to return to a new equilibrium at the original power level as a result of moderator and Doppler reactivity feedback. Because of the increased peaking and the potential return to the initial power level, the dropped control rod event poses a severe challenge to the DNB margin.

15.4.3.3(1) Reactor Protection - If the amount of reactivity is large enough to cause a significant reduction in core power, a reactor trip would be generated by the variable overpower trip. Reactor protection for the Control Rod Misoperation (Dropped Control Rod/Bank) event is summarized in Table 15.4.3(1)-A.

15.4.3.4(1) Disposition and Justification - Since the control rod drive mechanisms are de-energized in Modes 4-6 and reactor power is limited to zero percent with k_{eff} <.99 in Mode 3, there will be no consequences of this event for these modes. Ultimately, the consequences of this event are a return to power at elevated peaking conditions. Thus, the worst case is obtained when the final power level, increased peaking, and core inlet temperature are maximized. This occurs for dropped rod/bank events initiated from full power. The full power case thus bounds all other power operation conditions.

In general, a bank drop will cause a reactor trip and, as such, poses no challenge to the DNB margin. However, the event is analyzed from full power to assure that even the minimum worth bank when dropped will cause a reactor trip prior to a significant return to power. For a single dropped control rod, a reactor trip is not expected. Thus, a DNB evaluation assuming a return to full power at maximum dropped rod peaking will be performed to demonstrate that the SAFDLs are not violated. The analysis will support increased radial peaking and a more negative EOC moderator temperature coefficient. It should be noted that the operator will have multiple indications that a dropped rod/bank has occurred via CEA deviation circuit alarms and rod bottom signals. The only active system challenged in this event is the reactor protection system, which is redundant and single failure proof. The disposition of events for the Control Rod Misoperation (Dropped Control Rod/Bank) event is summarized in Table 15.4.3(1)-B.

(2) Dropped Part-Length Control Rod, and

(3) Malpositioning of the Part-Length Control Rod Group

All part-length control rods have been removed from the Millstone Unit 2 core. Therefore, these events are not applicable. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.4.3(2)-A, 15.4.3(2)-B, 15.4.3(3)-A, and 15.4.3-(B).

(4) Statically Misaligned Control Rod/Bank

These events are not in the current licensing basis for Millstone Unit 2 and therefore are not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.4.3(4)-A and 15.4.3(4)-B, respectively.

(5) Single Control Rod Withdrawal

15.4.3.1(5) Event Initiator - This event is initiated by the inadvertent withdrawal of a single CEA from the core. No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single CEA from the inserted CEA bank during full power operation. Procedures are available to permit the operator to withdraw a single CEA in the control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. The event can occur only as the result of multiple wiring failures or multiple operator actions in disregard of available event indication.

In the extremely unlikely event of simultaneous electrical failures which could result in single CEA withdrawal, the rod position indicators and deviation alarms would indicate the relative positions of the assemblies in the bank. Withdrawal of a single CEA by operator action, whether deliberate or by a combination of errors, would similarly result in the same visual indications. The CEA Motion Inhibit prevents further rod control motion upon detection of CEA malpositioning.

15.4.3.2(5) Event Description - The withdrawal of a single full length CEA is initiated by the inadvertent withdrawal of a single control rod. The ensuing reactivity insertion causes core power to increase. In the event that the secondary steam dump control system does not respond to the increased power production, secondary system temperature and pressure will increase, causing a corresponding increase in primary coolant temperature. This increase in primary coolant temperature occurs slowly enough that the pressurizer pressure control system, if available, is capable of suppressing the primary pressure increase. The degradation of coolant conditions coupled with the power increase is essentially the same as expected for CEA bank withdrawals at power, and may approach DNB conditions in the hot channel.

The single CEA withdrawal is distinguished from the withdrawal of a CEA bank by a severe radial power redistribution. High radial power peaking is localized in the region of the single withdrawn CEA and may, in severe cases, surpass the design limits. Thus, assemblies in the immediate vicinity of the withdrawn CEA may experience boiling transition. Such exposure would be limited to short time periods. Some fuel damage might occur.

15.4.3.3(5) Reactor Protection - The challenge to the fuel design limits is terminated by the automatic action of the reactor protection system which terminates the CEA withdrawal and inserts negative reactivity to terminate the power transient. The automatic action of the reactor protection system is initiated as the result of one of the following signals:

(1) Variable overpower trip;
(2) Local power density trip;
(3) Thermal margin/low pressure trip;
(4) High pressurizer pressure trip; or

(5) Variable overpower pre-trip alarm, which initiates Rod Withdrawal Prohibit action.

Reactor protection for the Uncontrolled Control Rod Bank Withdrawal at Power event is summarized in Table 15.4.3(5)-A.

15.4.3.4(5) Disposition and Justification - The overall system response to the withdrawal of a single CEA will be identical to the response to a withdrawal of a CEA bank. The only difference will be that the core will experience localized peaking in the vicinity of the withdrawn CEA that is not present if an entire bank is withdrawn. Therefore, the disposition of the single CEA withdrawal will be identical to that of the CEA bank withdrawal.

The disposition of the low to zero power bank withdrawal is addressed in Event 15.4.1. The disposition of the bank withdrawal from power operating conditions is addressed in Event 15.4.2. The disposition of events for the Single Control Rod Withdrawal event is summarized in Table 15.4.3(5)-B.

(6) Reactivity Control Device Removal Error During Refueling

Millstone Unit 2 has no reactivity control devices which are used during refueling and could inadvertently be removed. Boron dilution during refueling is considered in Event 15.4.6. Therefore, this event is not applicable.

(7) Variations In Reactivity Load to be Compensated by Burnup or On-Line Refueling

This event considered the anticipated variations in the reactivity load of the reactor, to be compensated by means of action such as buildup and burnup of xenon poisoning, fuel burnup, on-line refueling, fuel followers, temperature moderator and void coefficients.

Provisions for xenon changes and fuel burnup are described in Section 3 of Reference 4. On-line refueling will not be performed on Millstone Unit 2. The core design does not include fuel followers. The safety analyses are based upon the most adverse combination of temperature, moderator and void coefficients. Therefore, this event has no significant consequences and is not analyzed.

15.4.4 Startup of an Inactive Loop

15.4.4.1 Event Initiator - This event is initiated by the startup of an inactive reactor coolant pump.

15.4.4.2 Event Description - Each primary coolant loop is equipped with two single suction centrifugal pumps, one per cold leg, which are located between the steam generator outlet and the reactor vessel inlet nozzles. A nonreversing mechanism is provided to prevent reverse rotation of the pump rotor. This feature also limits backflow through the pump under non-operating conditions. Note: there is no backflow in the hot leg (or steam generator) associated with the side of the plant that has the inactive reactor coolant pump. The inadvertent actuation of an inactive pump would therefore lead to a decrease in moderator temperature and, with a negative moderator coefficient, an increase in core reactivity with a potential increase in core power level.

15.4.4.3 Reactor Protection - Reactor protection for this event is afforded by Technical Specification requirements on shutdown margin and reactor coolant pump operation. Reactor protection for the Startup of an Inactive Loop event is summarized in Table 15.4.4-A.

15.4.4.4 Disposition and Justification - This event is not credible in operating Modes 1 and 2 because Technical Specifications require all four reactor coolant pumps to be operating (Reference 3, pg. 3/4.4-1). It is not credible in Mode 6 due to administrative procedures requiring that the pumps be prevented from starting.

Technical Specification requirements on shutdown margin in Modes 3-5 are such that any reactivity insertion due to an inactive loop start is not great enough to reach criticality. Thus, the consequences of this event in Modes 3-5 are minimal and no analysis is required. The disposition of events for the Startup of an Inactive Loop event is summarized in Table 15.4.4-B.

15.4.5 Flow Controller Malfunction

Millstone Unit 2 does not have any flow control devices on the primary reactor coolant loops so this event is not credible and does not need to be analyzed.

15.4.6 CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant

15.4.6.1 Event Initiator - A dilution of the primary system boron concentration can occur as a result of adding primary grade water into the reactor coolant system via the Chemical Volume and Control System (CVCS) or the Precise Control of Reactivity System (PCRS). The greatest dilution rate occurs for operation of the CVCS charging pumps. The three available charging pumps can inject water into the primary system at a maximum rate of 132 gpm.

15.4.6.2 Event Description - Dilution of the primary coolant Boron concentration results in the insertion of positive reactivity. For reactor operation Modes 3-6, the event can result in a gradual erosion of available shutdown margin which, if unchecked, can cause a return to criticality. In the case of a boron dilution at rated power and power operation reactor operating conditions, the consequences are very similar to the consequences of a slow control rod withdrawal.

15.4.6.3 Reactor Protection - Reactor protection for the boron dilution event during operating Modes 3-6 is provided by Technical Specification Shutdown Margin requirements, Administrative procedures, and sufficient time for the operator to take the appropriate action in the unlikely event that a boron dilution should occur. Reactor protection for the reactor critical, power operation, and rated power operating conditions is provided by various trips and operator response time. Reactor protection for the CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant event is summarized in Table 15.4.6-A.

15.4.6.4 Disposition and Justification - For boron dilutions in reactor Modes 1-6, the challenge to the SAFDLs is very similar to that of slow control rod withdrawals and can be bounded by the consequences of control rod withdrawal events as analyzed for events 15.4.2 and 15.4.1. A spectrum of control rod withdrawal reactivity addition rates is considered for Events 15.4.2 and 15.4.1, so the range of reactivity addition rates will be established to encompass the predicted reactivity addition rates for boron dilution events in Modes 1-6.

The operator must have sufficient time to terminate the dilution prior to reaching Tech. Spec. shutdown margin requirements and/or losing minimum shutdown margin. These response times will be calculated for Millstone Unit 2 to bound the predicted critical boron concentrations. The disposition of events for the CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant event is summarized in Table 15.4.6-B.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.4.7-A and 15.4.7-B, respectively.

15.4.8 Spectrum of Control Rod Ejection Accidents

15.4.8.1 Event Initiator - This accident is initiated by a failure in the control rod drive pressure housing which could result in the rapid ejection of a control rod.

15.4.8.2 Event Description - Ejection of the control rod from the reactor core results in a rapid loss of negative reactivity causing a nuclear power transient. In addition to the power transient, the ejected rod results in a highly perturbed power distribution which, coupled with the power transient, could possibly lead to localized fuel damage. Also, the rapid nuclear power excursion can result in a significant short term heatup of the coolant with a resultant reactor coolant system pressure increase, although on the long term the reactor coolant system will depressurize due to the break in the reactor coolant pressure boundary.

15.4.8.3 Reactor Protection - Reactor protection for the Spectrum of Control Rod Ejection Accidents is summarized in Table 15.4.8-A. Doppler feedback inherent in the fuel also limits the nuclear power excursion.

15.4.8.4 Disposition and Justification - This event is not a concern in Modes 4-6 as all control rods are required to be fully inserted per Technical Specifications and no one control element assembly possesses enough reactivity worth to overcome the minimum allowed shutdown margin. The fuel energy content is maximized by starting from rated power initial conditions, so the consequences of this event are bounding for power operating initial conditions. However, because of the complex interaction of the ejected rod worth, and ejected peaking factor (which are maximized at hot critical operating conditions), and Doppler feedback effects, it is difficult to a priori bound the consequences of the event for either rated power or hot critical operating conditions. Therefore, the consequences of this event are analyzed for both rated power and hot critical operating conditions. The analysis is performed to support a more positive BOC and a more negative EOC moderator temperature coefficient.

In addition to the rod ejection, this event is characterized by a small break LOCA (SBLOCA) as the failure of the pressure housing is assumed to result in a breach of the primary coolant pressure boundary. The short term aspects of the event are dominated by the rod ejection, while the long term aspects are dominated by the SBLOCA. The limiting SBLOCA is evaluated in

Event 15.6.5 and is typically a cold leg break. In the rod ejection, the break is more characteristic of a hot leg break and therefore will be bounded by the SBLOCA. Also in the rod ejection, a much earlier reactor trip occurs, resulting in lower powers and temperatures than in Event 15.6.5. It is concluded that the long term aspects of the rod ejection are bounded by those of Event 15.6.5 for small breaks. Thus, only the short term rod ejection consequences need be evaluated. Note also that the limiting 15.6.5 event occurs for rated power operating conditions.

The disposition of events for the Spectrum of Control Rod Ejection Accidents is summarized in Table 15.4.8-B.

15.4.9 Spectrum of Rod Drop Accidents (BWR)

Millstone Unit 2 is not a Boiling Water Reactor (BWR) and as such this event is not applicable.

TABLE 15.4.1-A AVAILABLE REACTOR PROTECTION FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION EVENT

Reactor Operating Conditions	Reactor Protection
1	Not considered in this section
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit on Variable Overpower Pre-Trip Alarm
3	Variable Overpower Trip
	Rod Withdrawal Prohibit on Variable Overpower Pre-Trip Alarm
4-6	Not a credible event; no analysis required

TABLE 15.4.1-B DISPOSITION OF EVENTS FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION EVENT

Reactor Operating Conditions	Disposition
1	Not considered in the section
2	Analyze at 2000 psia
3	Bounded by the above
4-6	Not a credible event; no analysis required

TABLE 15.4.2-A AVAILABLE REACTOR PROTECTION FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL AT POWER EVENT

Reactor Operating Conditions	Reactor Protection
1	Variable Overpower Trip
	Local Power Density Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit Action on Variable Overpower or TM/LP Pre-Trip Alarm
2-6	Not considered in this section

TABLE 15.4.2-B DISPOSITION OF EVENTS FOR THE UNCONTROLLED CONTROL ROD/BANK WITHDRAWAL AT POWER EVENT

Reactor Operating Conditions	Disposition
1	Analyze at rated power
2-6	No analysis required; not considered in this section

TABLE 15.4.3(1)-A AVAILABLE REACTOR PROTECTION FOR THE DROPPED CONTROL ROD/BANK EVENT

Reactor Operating Conditions	Reactor Protection
1	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	Local Power Density Trip
	Available Thermal Margin
2	Variable Overpower Trip
	Available Thermal Margin
3-6	No significant consequences for these reactor operating conditions

TABLE 15.4.3(1)-B DISPOSITION OF EVENTS FOR THE DROPPED CONTROL ROD/BANK EVENT

Reactor Operating Conditions	Disposition
1	Analyze
2	Bounded by the above; no analysis required
3-6	No analysis required

TABLE 15.4.3(2)-A AVAILABLE REACTOR PROTECTION FOR THE DROPPED PART-LENGTH CONTROL ROD EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not a credible event; part-length control rods have been removed

TABLE 15.4.3(2)-B DISPOSITION OF EVENTS FOR THE DROPPED PART-LENGTH CONTROL ROD EVENT

Reactor Operating Conditions	Disposition
1-6	No analysis required; part-length control rods have been removed

TABLE 15.4.3(3)-A AVAILABLE REACTOR PROTECTION FOR THE MALPOSITIONING OF THE PART-LENGTH CONTROL ROD GROUP EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not a credible event; part-length control rods have been removed

TABLE 15.4.3(3)-B DISPOSITION OF EVENTS FOR THE MALPOSITIONING OF THE PART-LENGTH CONTROL ROD GROUP EVENT

Reactor Operating Conditions	Disposition
1-6	Not a credible event; no analysis required

TABLE 15.4.3(4)-A AVAILABLE REACTOR PROTECTION FOR THE STATICALLY MISALIGNED CONTROL ROD/BANK EVENT

Reactor Operating Conditions	Reactor Protection
1-6	No analysis required; not in licensing basis

TABLE 15.4.3(4)-B DISPOSITION OF EVENTS FOR THE STATICALLY MISALIGNED CONTROL ROD/BANK EVENT

Reactor Operating Conditions	Disposition
1-6	No analysis required; not in licensing basis

TABLE 15.4.3(5)-A AVAILABLE REACTOR PROTECTION FOR THE SINGLE CONTROL ROD WITHDRAWAL EVENT

Reactor Operating Conditions	Reactor Protection
1	Variable Overpower Trip
	Local Power Density Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit Action on Variable Overpower or TM/LP Pre-Trip Alarm
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
	Rod Withdrawal Prohibit on Variable Overpower Pre-Trip Alarm
3	Variable Overpower Trip
	Rod Withdrawal Prohibit on Variable Overpower Pre-Trip Alarm
4-6	Not a credible event; no analysis required

TABLE 15.4.3(5)-B DISPOSITION OF EVENTS FOR THE SINGLE CONTROL ROD WITHDRAWAL EVENT

Reactor Operating Conditions	Disposition
1	Analyze at rated power
2	Analyze at 2000 psia
3	Bounded by the above
4-6	Not a credible event; no analysis required

TABLE 15.4.3(6)-A AVAILABLE REACTOR PROTECTION FOR THE REACTIVITY CONTROL DEVICE REMOVAL ERROR DURING REFUELING EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not a credible event

TABLE 15.4.3(6)-B DISPOSITION OF EVENTS FOR THE REACTIVITY CONTROL DEVICE REMOVAL ERROR DURING REFUELING EVENT

Reactor Operating Conditions	Disposition
1-6	No analysis required; not a credible event

TABLE 15.4.3(7)-A AVAILABLE REACTOR PROTECTION FOR THE VARIATIONS IN REACTIVITY LOAD TO BE COMPENSATED BY BURNUP OR ON-LINE REFUELING EVENT

Reactor Operating Condition	Reactor Protection
1-6	No analysis required

TABLE 15.4.3(7)-B DISPOSITION OF EVENTS FOR THE VARIATIONS IN REACTIVITY LOAD TO BE COMPENSATED BY BURNUP OR ON-LINE REFUELING EVENT

Reactor Operating Condition	Disposition
1-6	No analysis required; no significant consequences

TABLE 15.4.4-A AVAILABLE REACTOR PROTECTION FOR THE STARTUP OF AN INACTIVE LOOP EVENT

Reactor Operating Conditions	Reactor Protection
1, 2, 6	Not applicable
3-5	Technical Specification requirements on shutdown margin and reactor coolant pump operation

TABLE 15.4.4-B DISPOSITION OF EVENTS FOR THE STARTUP OF AN INACTIVE LOOP EVENT

Reactor Operating Conditions	Disposition
1, 2, 6	Not applicable
3-5	No analysis required; minimal consequences

TABLE 15.4.5-A AVAILABLE REACTOR PROTECTION FOR THE FLOW CONTROLLER MALFUNCTION EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Event is not credible

TABLE 15.4 5-B DISPOSITION OF EVENTS FOR THE FLOW CONTROLLER MALFUNCTION EVENT

Reactor Operating Conditions	Disposition
1-6	Event is not credible; no analysis required

TABLE 15.4.6-A AVAILABLE REACTOR PROTECTION FOR THE CVCS MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT EVENT

Reactor Operating Conditions	Reactor Protection
1	Local Power Density Trip
	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
3-6	Technical Specification Shutdown Margin Requirements
	Administrative Procedures
	Operator Response Time

TABLE 15.4.6-B DISPOSITION OF EVENTS FOR THE CVCS MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT EVENT

Reactor Operating Conditions	Disposition
1-6	Analyze for loss of shutdown margin

TABLE 15.4.7-A AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis

TABLE 15.4.7-B DISPOSITION OF EVENTS FOR THE INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION EVENT

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis

TABLE 15.4.8-A AVAILABLE REACTOR PROTECTION FOR THE SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Reactor Operating Conditions	Reactor Protection
1	Variable Overpower Trip
	Thermal Margin/Low Pressure Trip
	High Pressurizer Pressure Trip
2	Variable Overpower Trip
	High Pressurizer Pressure Trip
3	Variable Overpower Trip
4-6	No reactor protection required; ejected rod worth less than the Technical Specification minimum shutdown margin. No significant consequence for this operating condition.

TABLE 15.4.8-B DISPOSITION OF EVENTS FOR THE SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

Reactor Operating Conditions	Disposition
1	Analyze for short term response. Long term bounded by Event 15.6.5.
2, 3	Analyze
4-6	No analysis required

TABLE 15.4.9-A AVAILABLE REACTOR PROTECTION FOR THE SPECTRUM OF ROD DROP ACCIDENTS (BWR)

Reactor Operating Conditions	Reactor Protection
1-6	Event is not applicable.

TABLE 15.4.9-B DISPOSITION OF EVENTS FOR THE SPECTRUM OF ROD DROP ACCIDENTS (BWR)

Reactor Operating Conditions	Disposition
1-6	Event is not applicable; no analysis required.

15.5 INCREASES IN REACTOR COOLANT SYSTEM INVENTORY

15.5.1 Inadvertent Operation of the ECCS That Increases Reactor Coolant Inventory

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.5.1-A and 15.5.1-B, respectively.

15.5.2 CVCS Malfunction That Increases Reactor Coolant Inventory

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Sumary Tables 15.5.2-A and 15.5.2-B, respectively. The potential consequences of diluting the primary system boron concentration are addressed in Event 15.4.6.

TABLE 15.5.1-A AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT OPERATION OF THE ECCS THAT INCREASES REACTOR COOLANT INVENTORY EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.5.1-B DISPOSITION OF EVENTS FOR THE INADVERTENT OPERATION OF THE ECCS THAT INCREASES REACTOR COOLANT INVENTORY EVENT

Reactor Operating Conditions	Disposition
1-G	Not in licensing basis; not analyzed.

TABLE 15.5.2-A AVAILABLE REACTOR PROTECTION FOR THE CVCS MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.5.2-B DISPOSITION OF EVENTS FOR THE CVCS MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY EVENT

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis; not analyzed.

15.6 DECREASES IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve

15.6.1.1 Event Initiator - The event is postulated to occur as a result of the inadvertent opening of a pressurizer pressure relief or safety valve due to an electrical or mechanical failure. The limiting event is obtained by assuming the inadvertent opening of both pressurizer power operated relief valves.

15.6.1.2 Event Description - The event initiator results in a blowdown of primary coolant as steam through the faulted valves. Primary system pressure drops rapidly until the pressurizer liquid is depleted, and then quite rapidly to a pressure determined by the saturation curve at the temperature of the coolant in the upper vessel head. Reactor scram will occur on thermal margin/low pressure before the pressurizer liquid is depleted, terminating the challenge to SAFDLs. In this initial stage, pressurizer heaters would actuate in an attempt to maintain pressure, but would be turned off on a low level signal before the heater elements were uncovered.

15.6.1.3 Reactor Protection - The thermal margin/low pressure trip provides initial protection against loss of thermal margin and possible fuel damage. Reactor protection for the Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve event is summarized in Table 15.6.1-A.

15.6.1.4 Disposition and Justification - The event proceeds as a depressurization of the primary coolant system with a loss of inventory. The core power and primary loop temperatures are relatively unaffected by the pressure drop. Thus, a short term challenge to the SAFDLs exists due to the depressurization prior to scram. There is also a long term concern in that if primary inventory cannot be restored and maintained, core uncovery may result.

The greatest challenge to core uncovery exists at rated power conditions when the core power and primary coolant stored energy are maximized. The greatest challenge to the SAFDLs occurs for the event initiated at rated power where the margin to DNB is minimized. This analysis will support increased radial peaking.

An evaluation of the SAFDL challenge will also be made for 5% power operating conditions in Mode 2 when the TM/LP trip may be bypassed. In this mode, the primary system may depressurize below the TM/LP setpoint pressure without an automatic reactor trip occurring. The Safety Injection System will, however, be available to inject boron and provide for inventory makeup.

The disposition of events for the Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve event is summarized in Table 15.6.1-B.

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment

The disposition of this event is provided in Section 14.1.4 of the Updated Millstone Unit 2 FSAR, Reference 4. This disposition is not dependent on either fuel type, power distribution, or reactor protection system modifications. It is therefore not affected by the current licensing action and remains valid for this event. This is reflected in the Available Reactor Protection and Event Disposition Summary Tables 15.6.2-A and 15.6.2-B, respectively.

15.6.3 Radiological Consequences of Steam Generator Tube Failure

15.6.3.1 Event Initiator - This event is initiated by the complete severance of a single generator tube.

15.6.3.2 Event Description - Experience with nuclear steam generators indicates that the probability of complete severance of a tube is small. The more probable modes of failure are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet.

A leaking steam generator tube would allow transport of primary coolant into the main steam system. Radioactivity contained in the primary coolant would mix with shell side water in the affected steam generator. Some of this radioactivity would be transported by steam to the turbine and then to the condenser. Noncondensible radioactive materials would then be passed to atmosphere through the condenser air ejector discharge via the Plant stack.

The radioactive products would be sensed by the condenser air ejector radiation monitor or the stack radiation monitor. These monitors have audible alarms that will be annunciated in the control room to alert the operator to abnormal activity levels so that corrective action could be taken.

The behavior of the systems will vary depending upon the size of the steam generator tube failure. For small leaks the chemical and volume control charging pumps will be able to maintain the necessary primary coolant inventory and an automatic reactor trip will not occur. The gaseous fission products will be released from the main steam system at the air ejector discharge and will be discharged via the Plant stack. Nonvolatile fission products will tend to concentrate in the water of the steam generators.

For leaks larger than the capacity of the charging pumps, the pressurizer water level and pressure will decrease and a reactor trip will occur. Upon reactor trip, the turbine will trip and the steam system atmospheric dump valves and the turbine bypass valve will open. In this case it is possible that in addition to the noble fission gases a substantial amount of the radioiodines contained in the secondary system may also be released through the steam dump valves.

The amount of radioactivity released increases with break size. For this analysis, a double-ended break of one tube was assumed. The selection of one double-ended break as an upper limit is conservatively based upon the experience obtained with other steam generators. No double-ended failure has ever occurred in such units.

15.6.3.3 Reactor Protection - The leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the charging pumps; therefore, the Primary Coolant system pressure will decrease and a low pressurizer pressure trip or thermal margin/low-pressure trip will occur. The thermal margin trip has a low-pressure floor, set at 1,750 psia, below which trip will always occur. Following the reactor trip the Primary Coolant System is cooled down by exhausting steam through the atmospheric steam dump valves and turbine bypass valve. The radioactivity exhausted through the steam dump valves passes directly to atmosphere. The radioactivity exhausted through the bypass valve flows to the condenser where the gaseous products remaining are vented to the atmosphere through the condenser air ejector and Plant stack.

Reactor protection for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 15.6.3-A.

15.6.3.4 Disposition and Justification - The radiological consequences of a steam generator tube rupture incident are maximized at rated power operation due to the stored energy in the primary coolant which must be removed by the intact steam generator in order to bring the primary and secondary systems into pressure equilibrium terminating the primary to secondary leak. The only proposed licensing action which would impact the radiological consequences of this event is the allowance of a slightly more negative EOC moderator temperature coefficient. However, since this event is basically a depressurization of the primary at power, there is very little reactivity feedback to affect power. Thus, the change in moderator temperature coefficient will have an insignificant effect on the transient results. The radiological consequences of record will therefore remain bounding for this event.

The challenge to the SAFDLs exists due to the depressurization prior to scram. As such, this challenge is very similar to that which exists due to the inadvertent opening of a pressurizer relief valve (Event 15.6.1). Since the depressurization rates associated with Event 15.6.1 are substantially larger than those encountered for this event, the corresponding pressure undershoot will also be greater. Event 15.6.1 will thus be characterized by lower pressures at the time of MDNBR than those obtained for this event. Therefore, the DNB aspects of this event will be bounded by those of Event 15.6.1.

The disposition of events for the Radiological Consequences of Steam Generator Tube Failure event is summarized in Table 15.6.3-B.

15.6.4 <u>Radiological Consequences of a Main Steam Line Failure Outside Containment (BWR)</u>

This event is only applicable to Boiling Water Reactors (BWRs). As such, this event is not applicable to Millstone Unit 2.

15.6.5 <u>Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary</u>

15.6.5.1 <u>Event Initiator</u> - This event is initiated by a breach in the Primary Coolant System pressure boundary. Basically, a range of break sizes from small leaks up to a complete double-ended severance of a Primary Coolant System pipe must be considered. Typically, these breaks are classified as Large Breaks or Small Breaks.

15.6.5.2 <u>Event Description</u>

(1) <u>Large Breaks</u>. The Large Break LOCA events are characterized by four sequential phases: 1) blowdown, 2) refill, 3) reflood, and 4) long term cooling.

<u>The blowdown</u> phase immediately follows the initiation of a large break.

Primary system water is discharged through the break into containment. The system pressure decreases rapidly during the initial subcooled blowdown. As the saturation pressure is approached, local boiling and flashing takes place in the core and the reactor goes subcritical via the negative moderator reactivity feedback. The blowdown flow becomes a water-vapor mixture. The depressurization rate is reduced when core pressure falls below the saturation pressure. The water level continues to decrease until a large amount of water from the ECCS passive accumulators reaches the lower plenum.

<u>The refill phase</u> starts when the accumulator water begins to fill the lower plenum. At this time, the core is uncovered by water and the fuel rods are cooled primarily by thermal radiation.

<u>The reflood phase</u> begins when the water level reaches the bottom of the core.

<u>The long term cooling phase</u> starts after the core has quenched to the point where the zircaloy-water reaction is suppressed, or the water level covers the active fuel. During this phase, the water inventory is controlled by the safety injection pumps. The continuous operation of these pumps ensures the long term dissipation of the decay heat.

(2) <u>Small Breaks</u>. The small break LOCA, as generally defined, includes any break in the pressure boundary that has an area of 0.5 ft^2 or less. The principal PWR design feature for mitigating the consequences of a small break LOCA is the ECCS which maintains the water inventory. Its major subsystems for restoring water inventory are the high pressure safety injection (HPSI) system, and the low pressure safety injection (LPSI) system and the safety injection (accumulator) tanks.

A small break LOCA is characterized by slow RCS depressurization rates and mass transfer rates within the RCS relative to similar parameters calculated for large break LOCA. If the break area is large enough that the HPSI pumps cannot maintain the reactor coolant inventory and allow RCS pressure control, the RCS will depressurize. The depressurization produces a low pressurizer pressure (thermal margin/low pressure) reactor trip and a safety injection actuation signal (SIAS). The rate of RCS depressurization following SIAS depends on the break area and the HPSI shutoff head. With a combination of a very small break and a sufficiently high HPSI shutoff head, the depressurization may be arrested.

If the break area is sufficiently large to allow continued depressurization and net loss of coolant inventory even with the HPSI pumps in operation, the coolant level in the reactor vessel may recede below the top of the reactor core. If sufficient steam is produced in the RCS, natural circulation (the reactor coolant pumps will have been tripped by this time to reduce coolant loss out of the break) around the RCS loops will cease. Eventually, loss of reactor coolant inventory is arrested by ECCS flow exceeding the flow out the break. In either case, the coolant level within the reactor vessel will rise, and the RCS will eventually be refilled (although leaking).

15.6.5.3 Reactor Protection

(1) Large Breaks. Basically no credit is taken for a reactor trip by the Reactor Protection System (RPS) due to the rapid depletion of the moderator which shuts down the reactor core almost immediately, followed by ECCS injection which contains sufficient boron to maintain the reactor core in a subcritical configuration. Technical Specification limits on hot rod power serve to limit the peak cladding temperature.

(2) Small Breaks. Primary reactor protection for this event is provided by the low pressurizer pressure (low pressure/thermal margin) trips and the Safety Injection Actuation Signal (SIAS) on a low pressurizer pressure signal.

Reactor protection for the Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary event is summarized in Table 15.6.5-A.

15.6.5.4 <u>Disposition and Justification</u> - Section 15.6.5 of Reference 1 indicates that the primary acceptance criteria for this event are to limit offsite doses, to limit fuel clad oxidation, and to keep peak cladding temperatures below 2200°F. Offsite doses are maximized by assuming the highest concentration of radionuclides contained within the fuel pins at event initiation. This is accomplished by assuming steady state radionuclide concentrations characteristic of long term operation of the plant at full power. Fuel pin cladding temperatures and oxidation rates are maximized by initiating the event with the highest cladding temperatures and linear heat generation rates. Thus, the most limiting results for this event (both large and small break sizes) are obtained with the plant operating at full power in Mode 1. These results will bound those from Modes 2-6.

The parameters which are changing from those which were used in the reference LOCA analysis presented in Sections 14.15.3 and 14.15.4 of the Updated Millstone Unit 2 FSAR (Reference 4) are the radial peaking factor and the loading of ANF supplied reload fuel. The radial peaking factor is increasing. Both the large and small break LOCA events will be reanalyzed.

Disposition of events for the Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary event is summarized in Table 15.6.5-B.

TABLE 15.6.1-A AVAILABLE REACTOR PROTECTION FOR THE INADVERTENT OPENING OF A PWR PRESSURIZER PRESSURE RELIEF VALVE EVENT

Reactor Operating Conditions	Reactor Protection
1	Thermal Margin/Low Pressure Trip
	Safety Injection Actuation Signal
2, 3	Safety Injection Actuation Signal
	Available Thermal Margin
4	Available Thermal Margin
5, 6	No significant consequences for these reactor operating conditions

TABLE 15.6.1-B DISPOSITION OF EVENTS FOR THE INADVERTENT OPENING OF A PWR PRESSURIZER PRESSURE RELIEF VALVE EVENT

Reactor Operating Conditions	Disposition
1, 2	Analyze for DNBR
3-6	Bounded by the above

Table 15.6.2-A AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE OF CONTAINMENT EVENT

Reactor Operating Conditions	Reactor Protection
1-6	None required; not a credible event for this plant.

TABLE 15.6.2-B DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE OF CONTAINMENT EVENT

Reactor Operating Conditions	Disposition
1-6	Disposition of record provided in Section 14.1.4 of the Updated Millstone Unit 2 FSAR, Reference 4; not a credible event.

TABLE 15.6.3-A AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Reactor Operating Conditions	Reactor Protection
1	Thermal Margin/Low Pressure Trip
	Safety Injection Actuation Signal
2, 3	Safety Injection Actuation Signal
4-6	No significant consequences for these reactor operating conditions

TABLE 15.6.3-B DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE RUPTURE EVENT

Reactor Operating Conditions	Disposition
1	Bounded by the analysis of record and by that of Event 15.6.1
2-6	Bounded by the above

TABLE 15.6.4-A AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAMLINE FAILURE OUTSIDE CONTAINMENT (BWR) EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not applicable; a Boiling Water Reactor (BWR) event

TABLE 15.6.4-B DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAMLINE FAILURE OUTSIDE CONTAINMENT (BWR) EVENT

Reactor Operating Conditions	Disposition
1-6	Not applicable; a Boiling Water Reactor (BWR) event

TABLE 15.6.5-A AVAILABLE REACTOR PROTECTION FOR THE LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

Reactor Operating Conditions	Reactor Protection
1	Large Breaks - No credit taken for reactor trip by the Reactor Protection System (RPS) ECCS - short and long term cooling
1	Small Breaks - Thermal Margin/Low Pressure Trip Low Reactor Coolant Flow Trip Safety Injection Actuation Signal
2	Large Breaks - No credit taken for reactor trip by the Reactor Protection System (RPS) ECCS - short and long term cooling
2	Small Breaks - Safety Injection Actuation Signal
3-6	No significant consequences for these reactor operating conditions

TABLE 15.6.5-B DISPOSITION OF EVENTS FOR THE LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

Reactor Operating Conditions	Disposition
1	Large Break - Analyze
1	Small Break - Analyze
2-6	Large Break - Bounded by the event initiated from Mode 1
2-6	Small Break - Bounded by the event initiated from Mode 1

15.7 RADIOACTIVE RELEASES FROM A SUBSYSTEM OR COMPONENT

15.7.1 Waste Gas System Failure

The results of this event are not dependent on either fuel type, power distribution, or reactor protection system modifications. The reference analysis is therefore not affected by the current licensing action and remains the bounding analysis for this event. The reference analysis is provided in the Updated Millstone Unit 2 FSAR, Reference 4.

15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere), and

15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

These events are not in the current licensing basis for Millstone Unit 2 and therefore are not analyzed. Further, Event 15.7.2 has been deleted from the SRP, Reference 1.

15.7.4 Radiological Consequences of Fuel Handling Accident

15.7.4.1 Event Initiator - The event is initiated by a mishap either in containment or in the auxiliary building. The mishap results in the fuel assembly being dropped, causing damage to the fuel pins.

15.7.4.2 Event Description - For the purpose of defining the upper limit on fuel damage as the result of a fuel handling incident, it is assumed that the fuel assembly is dropped during handling. Interlocks, procedural and administrative controls make such an event unlikely. However, if an assembly is damaged to the extent that a number of fuel rods fail, the accumulated fission gases and iodines in the fuel element gap could be released to the surrounding water. Release of the fission products to the surrounding water is considered negligible as a result of reduced diffusion through the fuel due to the low fuel temperature during refueling.

15.7.4.3 Reactor Protection - No reactor protection is required for this event. This is summarized in the reactor protection for the Radiological Consequences of Fuel Handling Accident event in Table 15.7.4-A.

15.7.4.4 Disposition and Justification - Limiting analyses for the radiological consequences of fuel handling accidents both in the containment building and in the auxiliary building are presented in Section 14.19 of the FSAR (Ref. 4). The evaluation of the limiting modes of fuel bundle failure are unaffected by a fuel reload and thus remain bounding for Cycle 10. The radiological consequences of a fuel bundle failure were evaluated using methodology given in TID-14844[7] and are thus only a function of core power. None of the inputs to this analysis are impacted by a fuel reload, and thus the radiological consequences of the analysis of record also remain bounding for Cycle 10.

The disposition of the Radiological Consequences of Fuel Handling Accident event is summarized in Table 15.7.4-B.

15.7.5 Spent Fuel Cask Drop Accidents

This event is not in the current licensing basis for Millstone Unit 2 and therefore is not analyzed. This is shown in the Available Reactor Protection and Event Disposition Summary Tables 15.7.5-A and 15.7.5-B, respectively.

TABLE 15.7.1-A AVAILABLE REACTOR PROTECTION FOR THE WASTE GAS SYSTEM FAILURE EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Reactor Protection System (RPS) action not required

TABLE 15.7.1-B DISPOSITION OF EVENTS FOR THE WASTE GAS SYSTEM FAILURE EVENT

Reactor Operating Conditions	Disposition
1-6	Bounding analysis presented in Section 14.17 of the Updated Millstone Unit 2 FSAR, Reference 4

TABLE 15.7.2-A AVAILABLE REACTOR PROTECTION FOR THE RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE) EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; no analysis required.

TABLE 15.7.2-B DISPOSITION OF EVENTS FOR THE RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE) EVENT

Reactor Operating Condition	Disposition
1-6	Not in licensing basis; no analysis required.

TABLE 15.7.3-A AVAILABLE REACTOR PROTECTION FOR THE POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES EVENT

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; no analysis required.

TABLE 15.7.3-B DISPOSITION OF EVENTS FOR THE POSTULATED RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES EVENT

Reactor Operating Condition	Disposition
1-6	Not in licensing basis; no analysis required.

TABLE 15.7.4-A AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

Reactor Operating Conditions	Reactor Protection
1-6	Reactor Protection System (RPS) action not required

TABLE 15.7.4-B DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENTS

Reactor Operating Conditions	Disposition
1-6	Bounding analysis presented in Section 14.19 of the Updated Millstone Unit 2 FSAR, Ref. 4

TABLE 15.7.5-A AVAILABLE REACTOR PROTECTION FOR THE SPENT FUEL CASK DROP ACCIDENTS

Reactor Operating Conditions	Reactor Protection
1-6	Not in licensing basis; not analyzed.

TABLE 15.7.5-B DISPOSITION OF EVENTS FOR THE SPENT FUEL CASK DROP ACCIDENTS

Reactor Operating Conditions	Disposition
1-6	Not in licensing basis; not analyzed.

4.0 MILLSTONE UNIT 2 FSAR EVENTS NOT CONTAINED IN THE STANDARD REVIEW PLAN

4.1 EFFECTS OF EXTERNAL EVENTS

4.1.1 Event Initiator

The external events affecting the plant which were included in this category are as follows:

- High and Low Water;
- Storms;
- Tornadoes; and
- Earthquakes.

4.1.2 Event Description

The location of the buildings in this installation on the shore of Long Island Sound exposes them to several different natural events of varying intensity. These events include storms and tornadoes and their effects on the water level in the Sound. Also included are the effects of seismic tremors which could occur in this region.

Normal fluctuations in the bay water level caused by tides are very predictable and insignificant. Coupled, however, with strong winds, the water level could rise or fall appreciably more than usual. This may affect the intake to the safety-related service water cooling pumps.

The meteorology of the Millstone site is basically that of a seacoast transition zone which lies along a major storm track of extra-tropical cyclones and an occasional storm of tropical origin. These storms, along with seasonal thunderstorms, will produce intense rainfall and (during the winter) snow and freezing rain storms. These storms can produce high wind and snow loads on structures. Tornadoes have been reported due to occasional severe storms.

The earthquake history of the Southern New England area has been compiled to ascertain the maximum expected seismic acceleration.

4.1.3 Reactor Protection

No reactor protection is required for these events.

4.1.4 Disposition and Justification

A fuel reload will not affect any of the inputs to this analysis. Therefore, the analysis of record remains bounding for this event.

4.2 FAILURES OF EQUIPMENT PROVIDING JOINT CONTROL AND SAFETY FUNCTIONS

There is no equipment at Millstone Unit 2 which provides a joint control function and safety function. Therefore, this event is not credible.

4.3 CONTAINMENT PRESSURE ANALYSIS

4.3.1 Event Initiator

This event is initiated by a breach in the primary coolant system pressure boundary. Basically, a range of break sizes and types in both the hot legs and cold legs are considered. The event initiator is that break which results in the greatest mass and energy release to the containment.

4.3.2 Event Description

In the event of a LOCA, the release of primary coolant from the rupture area will cause the high pressure, high temperature liquid to enter the containment, rapidly flashing to steam and water within the containment. The addition of this mass and energy to the containment will result in a rise in both the pressure and temperature of the containment atmosphere. The containment building design is based on the mass and energy absorption capabilities of the volume enclosed by the containment structure.

A spectrum of break sizes for both the hot and cold primary coolant legs has been considered in the evaluation of the containment design to determine the most severe combination of reactor system mass and energy releases, sensible heat sources, and shutdown heat sources during the blowdown and reflood phases of the LOCA. Following the time of peak containment pressure, the safety injection systems and containment cooling systems reduce the containment pressure and temperature until the Refueling Water Storage Tank (RWST) water supply is exhausted.

4.3.3 Reactor Protection

This event considers the adequacy of the containment design to a LOCA event and, as such, does not address the need for reactor protection. The adequacy of available reactor protection is addressed in Event 15.6.5.

4.3.4 Disposition and Justification

A fuel reload will not affect the critical parameters of primary pressure, average primary coolant temperature or power which govern the mass and energy release to the containment. Since the fuel bundle design will not change substantially, the fuel performance during the event will be unaffected. Therefore, the analysis of record will remain bounding for this event.

4.4 HYDROGEN ACCUMULATION IN CONTAINMENT

4.4.1 Event Initiator

The event initiator is the Design Basis Incident (DBI). The DBI is the limiting Section 3 event producing the greatest mass and energy release to the containment.

4.4.2 Event Description

The significant sources of hydrogen following the Design Basis Incident are radiolysis of water from the decay of fission products, the zirconium-water reaction of the fuel cladding and corrosion of containment metals. If unchecked, the hydrogen generation may become great enough so that a flammable concentration would be present in the containment atmosphere. An electric hydrogen recombiner system is provided to reduce containment hydrogen concentrations.

4.4.3 Reactor Protection

No reactor protection systems are required for this event.

4.4.4 Disposition and Justification

The cladding thickness for ANF supplied fuel is slightly greater than that for either Combustion Engineering or Westinghouse fuel. Thus, the amount of zirconium clad in the core will increase slightly with ANF fuel (by less than 8%), resulting in a slight increase in the amount of hydrogen assumed to be generated from the zirconium-water reaction. The analysis of record has shown that with one of two hydrogen recombiners in operation, the maximum hydrogen concentration is well below the flammability limit. This limit will not be challenged by the expected increase in hydrogen concentration due to the reload fuel.

The analysis of record also evaluated the expected offsite doses resulting from a postulated containment purge operation, assuming failure of both hydrogen recombiners. Again, the resulting peak concentration level was well below the flammability limit and will not challenge the limit with ANF reload fuel.

The offsite dose analysis was performed using TID-14844[7] methodology. The proposed changes for Cycle 10 will not impact this methodology, and the current offsite dose analysis remains valid for Cycle 10.

We conclude that offsite doses will not exceed those in the analysis of record and that hydrogen flammability limits will not be challenged. A fuel reload will not affect any of the other inputs to this calculation. Thus, no reanalysis is performed.

4.5 RADIOLOGICAL CONSEQUENCES OF THE DESIGN BASIS INCIDENT (DBI)

4.5.1 Event Initiator

The event initiator is the Design Basis Incident (DBI). The DBI is the limiting Section 3 event producing the greatest mass and energy release to the containment.

4.5.2 Event Description

The DBI is assumed to result in a gross release of radioactivity from the fuel to the containment. The activity is assumed to leak from the containment directly to the atmosphere, into the surrounding enclosure building and into the control room. This results in offsite doses to the general population and can have an adverse effect on control room habitability.

4.5.3 Reactor Protection

No reactor protection systems are required for this event.

4.5.4 Disposition and Justification

This event assumes a gross release of radioactivity from the fuel to the containment building and ultimately to the atmosphere. Thus, the consequences of the event are driven by the assumed fission product inventory contained within the fuel. The analysis of record presented in Section 14.20 of Reference 4 evaluated the radiological consequences using methodology given in TID-14844(7), and are thus only a function of core power. None of the inputs to this analysis are impacted by a fuel reload, and thus the radiological consequences of the analysis of record remain bounding for Cycle 10.

TABLE 4.1-A AVAILABLE REACTOR PROTECTION FOR THE EFFECTS OF EXTERNAL EVENTS

Reactor Operating Condition	Reactor Protection
1-6	None required.

TABLE 4.1-B DISPOSITION OF EVENTS FOR THE EFFECTS OF EXTERNAL EVENTS

Reactor Operating Condition	Disposition
1-6	Bounded by the analysis of record.

TABLE 4.2-A AVAILABLE REACTOR PROTECTION FOR THE FAILURES OF EQUIPMENT PROVIDING JOINT CONTROL AND SAFETY FUNCTIONS EVENT

Reactor Operating Condition	Reactor Protection
1-6	Not a credible event; no analysis required.

TABLE 4.2-B DISPOSITION OF EVENTS FOR THE FAILURES OF EQUIPMENT PROVIDING JOINT CONTROL AND SAFETY FUNCTIONS EVENT

Reactor Operating Condition	Disposition
1-6	Not a credible event; no analysis required.

TABLE 4.3-A AVAILABLE REACTOR PROTECTION FOR THE CONTAINMENT PRESSURE ANALYSIS EVENT

Reactor Operating Condition	Reactor Protection
1-6	Addressed in Event 15.6.5

TABLE 4.3-B DISPOSITION OF EVENTS FOR THE CONTAINMENT PRESSURE ANALYSIS EVENT

Reactor Operating Condition	Disposition
1-6	Bounded by analysis of record.

TABLE 4.4-A AVAILABLE REACTOR PROTECTION FOR THE HYDROGEN ACCUMULATION IN CONTAINMENT EVENT

Reactor Operating Condition	Reactor Protection
1-6	None required.

TABLE 4.4-B DISPOSITION OF EVENTS FOR THE HYDROGEN ACCUMULATION IN CONTAINMENT EVENT

Reactor Operating Condition	Disposition
1-6	No analysis required; analysis of record provides offsite dose calculation; flammability limits not challenged

TABLE 4.5-A AVAILABLE REACTOR PROTECTION FOR THE RADIOLOGICAL CONSEQUENCES OF THE DESIGN BASIS INCIDENT EVENT

Reactor Operating Condition	Reactor Protection
1-6	None required.

TABLE 4.5-B DISPOSITION OF EVENTS FOR THE RADIOLOGICAL CONSEQUENCES OF THE DESIGN BASIS INCIDENT EVENT

Reactor Operating Condition	Disposition
1-6	Bounded by analysis of record.

5.0 REFERENCES

1. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

2. "Exxon Methodology for Pressurized Water Reactors - Analysis of Chapter 15 Events," ANF-84-73(P), Rev. 3, Advanced Nuclear Fuels Corp.

3. Technical Specifications for Millstone Unit 2, Docket No. 50-336, Updated through Amendment No. 116.

4. Millstone Unit 2, Updated Final Safety Analysis Report.

5. "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," XN-NF-82-21(A), Exxon Nuclear Company, September 1983.

6. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-NF-75-21(A), Revision 2, Exxon Nuclear Company.

7. "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, Reactor Technology, Division of Licensing and Regulation, U.S. Atomic Energy Commission, March 23, 1962.