Docket No. 50-336
B17413

Attachment 3

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Control Room Ventilation System
Marked Up Pages

September 1998

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

4.3.2.1.4 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $X/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and ~~apriculate~~ (Half Lives greater than 8 days) radioactivity shall be ~~asusmed~~ to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5×10^5 cpm.

assumed particulate

Amendment No. 49

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

0.035

c. ~~1~~ GPM ~~total~~ primary-to-secondary leakage through ~~both steam generators and 0.10 GPM through~~ any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE

.1

4.4.6.2 Reactor Coolant System ~~leakages~~ shall be demonstrated to be within ~~each of the above~~ limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

INSERT A

INSERT A - Page 3/4 4-9

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ 100/Ē μCi/gram of gross specific activity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours. Specification 3.0.4 is not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

c. With the specific activity of the primary coolant > 100/Ē μCi/gram of gross specific activity, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/Ē μCi/gram of gross specific activity, perform the sampling and analysis requirements of item 4 a) of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits.

*With T_{avg} ≥ 515°F.

NO CHANGE
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SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-2.

~~August 1, 1975~~

TABLE 4.4-2

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS ~~MINIMUM~~ FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days
3. Radiochemical Analysis for $\bar{E}$ Determination	1 per 6 months*
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the Specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131, or 100/$\bar{E}$ μCi/gram of gross specific activity, and ~~DOSE EQUIVALENT I-131 exceeds 1.0 µCi/gram, and~~ b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer. The provisions of Specification 4.0.4 are not applicable.

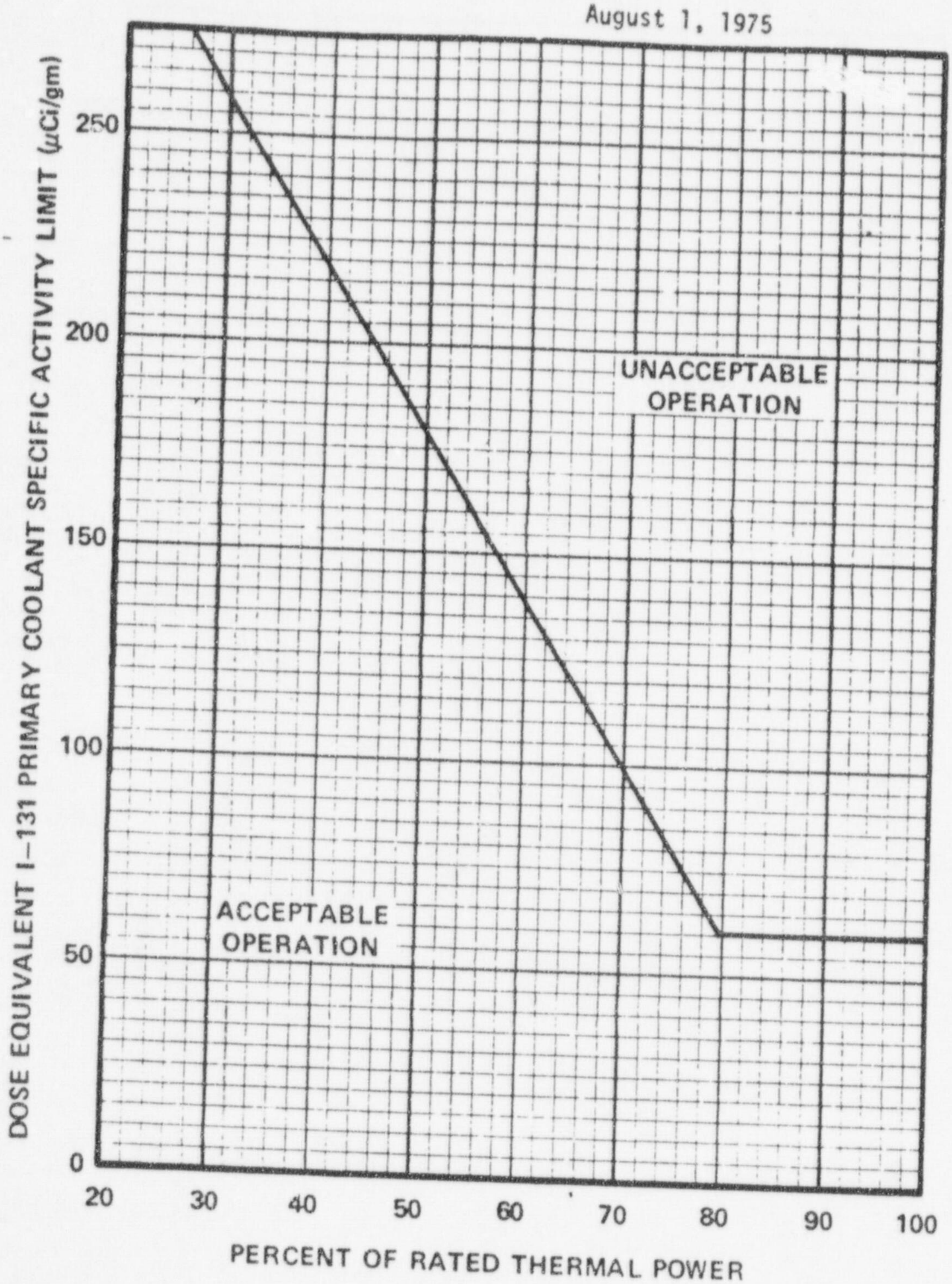

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FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity $> 1.0 \mu Ci/gram$ Dose Equivalent I-131

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

72 hours

Inoperable Equipment	Required Action
a. One containment spray train	a.1 Restore the inoperable containment spray train to OPERABLE status within ~~7 days~~ or be in HOT SHUTDOWN within the next 12 hours.
b. One containment cooling train	b.1 Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c. One containment spray train AND One containment cooling train	c.1 Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d. Two containment cooling trains	d.1 Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
e. All other combinations	e.1 Enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each spray pump from the control room,

2. Verifying, that on recirculation flow, each spray pump develops a discharge pressure of ≥ 254 psig,

*The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump operates for at least 15 minutes,

4. Cycling each testable, automatically operated valve in each spray train flow path through at least one complete cycle,

5. Verifying that upon a sump recirculation actuation signal the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established, and

6. Verifying that all accessible manual valves not locked, sealed or otherwise secured in position and all remote or automatically operated valves in each spray train flow path are positioned to take suction from the RWST on a Containment Pressure--High-High signal.

b. At least once per 18 months, during shutdown, by cycling each power operated valve in the spray train flow path not testable during plant operation through at least one complete cycle of full travel.

c. At least once per 18 months by verifying a total leak rate less than or equal to 12 gallons per hour in conjunction with the high pressure safety injection system (reference Specification 4.5.2.c.5) at:

1) Discharge pressure of greater than or equal to 254 psig on recirculation flow for those parts of the system between the pump discharge and the header isolation valve, including the pump seals.

2) Greater than or equal to 22 psig at the pump suction for the piping from the containment sump check valve to the pump suction.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

4.6.2.1.2 Each containment air recirculation and cooling unit shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting, in low speed, each unit from the control room,

b. Verifying that each unit operates for at least 15 minutes, and

c. Verifying a cooling water flow rate of $\geq$ 500 gpm to each cooling unit.

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CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent Enclosure Building Filtration Trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one Enclosure Building Filtration Train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each Enclosure Building Filtration Train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal absorber train and verifying that the train operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ ~~6~~ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the train starts on an Enclosure Building Filtration Actuation Signal (EBFAS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

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SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

~~October 29, 1990~~

PLANT SYSTEMS

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent ~~c~~ontrol ~~r~~oom ~~e~~mergency ~~v~~entilation ~~systems~~ shall be OPERABLE. C R E V Trains

APPLICABILITY: ALL MODES

ACTION:

Modes 1, 2, 3, and 4: train Ventilation Train

With one Control Room Emergency ~~Air Clean-up System~~ inoperable, restore the inoperable ~~system~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6*: train

a. With one Control Room Emergency ~~Air Clean-Up System~~ inoperable, restore the inoperable ~~system~~ to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency ~~Air Clean-Up System~~ in the recirculation mode. Ventilation Train

b. With both Control Room Emergency ~~Air Clean-Up Systems~~ inoperable, or with the OPERABLE Control Room Emergency ~~Air Clean-Up System~~ required to be in the recirculation mode by ACTION (a.) not capable of being powered by an OPERABLE normal and emergency power source; suspend all operations involving CORE ALTERATIONS or positive reactivity changes. Ventilation Trains

* In Modes 5 and 6, when a Control Room Emergency ~~Air Clean-up system~~ is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then Limiting Condition for Operation (LCO) 3.7.6.1.a or 3.7.6.1.b shall be invoked as applicable.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

C R E V Train

4.7.6.1 Each control room emergency ventilation ~~system~~ Train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is $\leq$ 100°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room, flow through the HEPA filters and charcoal absorber train and verifying that the ~~system~~ train operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the ~~system~~ train by:

   1. Verifying that the cleanup ~~system~~ train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the ~~system~~ train flow rate is 2500 cfm $\pm$ 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.* The carbon sample shall have a removal efficiency of $\geq$ 95 percent.

   3. Verifying a ~~system~~ train flow rate of 2500 cfm $\pm$ 10% during ~~system~~ train operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~6~~ 3.4 inches Water Gauge while operating the ~~system~~ train at a flow rate of 2500 cfm ± 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the ~~system~~ train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENT (Continued)

Control Room Emergency Ventilation

3. Verifying that control room air in-leakage is less than 130 ~~100~~ SCFM with the ~~Control Air Conditioning~~ System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ~~system~~ train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ~~system~~ train at a flow rate of 2500 cfm ± 10%.

NO CHANGE
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STORAGE POOL AREA VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.15 At least one Enclosure Building Filtration Train shall be OPERABLE and capable of automatically initiating operation in the auxiliary exhaust mode and exhausting through HEPA filters and charcoal adsorbers on a storage pool area high radiation signal.

APPLICABILITY: WHENEVER IRRADIATED FUEL IS IN THE STORAGE POOL.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Enclosure Building Filtration Train is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.9.15 The above required Enclosure Building Filtration Train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the train operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ ~~6~~ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that on a Spent Fuel Storage Pool Area high radiation signal, the train automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

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SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

REACTOR COOLANT SYSTEM

BASES

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

0.035

The ~~total~~ steam generator tube leakage limit of ~~1~~ GPM ~~for all~~ per steam generator~~s~~ ensures that the dosage contribution from the tube leakage will be ~~limited to a small fraction of Part 100 limits~~ in the event of either a steam generator tube rupture or steam line break. The ~~1~~ GPM limit is consistent with the assumptions used in the analysis of these accidents.

less than the limits of General Design Criteria 19 of 10CFR50 Appendix A

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

The 0.10 GPM Primary to Secondary leakage limitation assures structural integrity. A tube with a through-wall circumferential crack which leaks at 0.10 GPM under normal operating conditions retains the structural margins recommended in Regulatory Guide 1.121. In addition, the total leakage under accident conditions would remain below the 1 GPM limit.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses. The leak rate surveillance requirements assure that the leakage assumed for the system outside containment during the recirculation phase will not be exceeded.

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray system during post-LOCA conditions.

To be OPERABLE, the two trains of the containment spray system shall be capable of taking a suction from the refueling water storage tank on a containment spray actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal. Each containment spray train flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

The containment cooling system consists of two containment cooling trains. Each containment cooling train has two containment air recirculation and cooling units. For the purpose of applying the appropriate action statement, the loss of a single containment air recirculation and cooling unit will make the respective containment cooling train inoperable.

Either the containment spray system or the containment cooling system has sufficient heat removal capability to handle any design basis accident. However, the containment spray system is more effective in dealing with the superheated steam from a main steam break inside containment. ~~Therefore, at least one train of containment spray is always required to be OPERABLE, when pressurizer pressure is ≥ 1750 psia.~~

INSERT B

3/4.6.3 CONTAINMENT ISOLATION VALVES

The Technical Requirements Manual contains the list of containment isolation valves (except the containment air lock and equipment hatch). Any changes to this list will be reviewed under 10CFR50.59 and approved by the Plant Operations Review Committee (PORC).

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within

INSERT B - Page B 3/4 6-3

In addition, the containment spray system provides a mechanism for removing iodine from the containment atmosphere. Therefore, at least one train of containment spray is required to be OPERABLE when pressurizer pressure is $\geq$ 1750 psia, and the allowed outage time for one train of containment spray reflects the dual function of containment spray for heat removal and iodine removal.

PLANT SYSTEMS

BASES

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 FLOOD LEVEL

The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the C ~~c~~ontrol R ~~r~~oom E ~~e~~mergency V ~~v~~entilation S ~~s~~ystem ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria ~~10~~ 19 of Appendix "A", 10 CFR 50.

INSERT C

INSERT C - Page B 3/4 7-4

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flow rate surveillance requirement of 2500 cfm $\pm$ 10%.

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), operator action will be required to return to a full train line up. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train line up. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until 10 minutes into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

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Attachment 4

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Control Room Ventilation System
Retyped Pages

September 1998

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

4.3.2.1.4 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $X/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and particulate (Half Lives greater than 8 days) radioactivity shall be assumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5 $x10^5$ cpm.

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 0.035 GPM primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE shall be demonstrated to be within limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

4.4.6.2.2 Primary to secondary leakage shall be demonstrated to be within the above limits by performance of a primary to secondary leak rate determination at least once per 72 hours. The provisions of Specification 4.0.4 are not applicable for entry into MODE 4.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram of gross specific activity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours. Specification 3.0.4 is not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram of gross specific activity, be in HOT STANDBY with T_{avg} < 515°F within 4 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/gram of gross specific activity, perform the sampling and analysis requirements of item 4 a) of Table 4.4-2 until the specific activity of the primary coolant is restored to within its limits.

*With $T_{avg} \geq$ 515°F.

TABLE 4.4-2

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days
3. Radiochemical Analysis for $\bar{E}$ Determination	1 per 6 months*
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131, or 100/$\bar{E}$ μCi/gram of gross specific activity, and b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer. The provisions of Specification 4.0.4 are not applicable.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains, with each cooling train consisting of two containment air recirculation and cooling units, shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

Inoperable Equipment	Required Action
a. One containment spray train	a.1 Restore the inoperable containment spray train to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
b. One containment cooling train	b.1 Restore the inoperable containment cooling train to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.
c. One containment spray train AND One containment cooling train	c.1 Restore the inoperable containment spray train or the inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
d. Two containment cooling trains	d.1 Restore at least one inoperable containment cooling train to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.
e. All other combinations	e.1 Enter LCO 3.0.3 immediately.

SURVEILLANCE REQUIREMENTS

4.6.2.1.1 Each containment spray train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each spray pump from the control room,

2. Verifying, that on recirculation flow, each spray pump develops a discharge pressure of $\geq$ 254 psig,

*The Containment Spray System is not required to be OPERABLE in MODE 3 if pressurizer pressure is < 1750 psia.

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that the train starts on an Enclosure Building Filtration Actuation Signal (EBFAS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent Control Room Emergency Ventilation Trains shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

Modes 1, 2, 3, and 4:

With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6*

a. With one Control Room Emergency Ventilation Train inoperable, restore the inoperable train to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation Train in the recirculation mode.

b. With both Control Room Emergency Ventilation Trains inoperable, or with the OPERABLE Control Room Emergency Ventilation Train required to be in the recirculation mode by ACTION (a.) not capable of being powered by an OPERABLE normal and emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

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* In Modes 5 and 6, when a Control Room Emergency Ventilation Train is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of 3.7.6.1 Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system (s), subsystem (s), train (s), component (s) and device(s) are OPERABLE, or likewise satisfy the requirements of the specification. Unless both conditions (1) and (2) are satisfied within 2 hours, then Limiting Condition for Operation (LCO) 3.7.6.1.a or 3.7.6.1.b shall be invoked as applicable.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each Control Room Emergency Ventilation Train shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is $\leq$ 100°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room, flow through the HEPA filters and charcoal absorber train and verifying that the train operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the train by:

   1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 2500 cfm $\pm$ 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.* The carbon sample shall have a removal efficiency of $\geq$ 95 percent.

   3. Verifying a train flow rate of 2500 cfm $\pm$ 10% during train operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89.

SURVEILLANCE REQUIREMENTS

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.4 inches Water Gauge while operating the train at a flow rate of 2500 cfm ± 10%.

2. Verifying that on a recirculation signal, with the Control Room Emergency Ventilation Train operating in the normal mode and the smoke purge mode, the train automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that control room air in-leakage is less than 130 SCFM with the Control Room Emergency Ventilation System operating in the recirculation/filtration mode.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

g. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 2500 cfm ± 10%.

1. Verifying that the cleanup train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the train flow rate is 9000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a train flow rate of 9000 cfm ± 10% during train operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is ≤ 2.6 inches Water Gauge while operating the train at a flow rate of 9000 cfm ± 10%.

2. Verifying that on a Spent Fuel Storage Pool Area high radiation signal, the train automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the train at a flow rate of 9000 cfm ± 10%.

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* ASTM D3803-89 shall be used in place of ANSI N509-1976 as referenced in table 2 of Regulatory Guide 1.52. The laboratory test of charcoal should be conducted at a temperature of 30°C and a relative humidity of 95% within the tolerances specified by ASTM D3803-89. Additionally, the charcoal sample shall have a removal efficiency of ≥ 95%.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The steam generator tube leakage limit of 0.035 GPM per steam generator ensures that the dosage contribution from the tube leakage will be less than the limits of General Design Criteria 19 of 10CFR50 Appendix A in the event of either a steam generator tube rupture or steam line break. The 0.035 GPM limit is consistent with the assumptions used in the analysis of these accidents.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses. The leak rate surveillance requirements assure that the leakage assumed for the system outside containment during the recirculation phase will not be exceeded.

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray system during post-LOCA conditions.

To be OPERABLE, the two trains of the containment spray system shall be capable of taking a suction from the refueling water storage tank on a containment spray actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal. Each containment spray train flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

The containment cooling system consists of two containment cooling trains. Each containment cooling train has two containment air recirculation and cooling units. For the purpose of applying the appropriate action statement, the loss of a single containment air recirculation and cooling unit will make the respective containment cooling train inoperable.

Either the containment spray system or the containment cooling system has sufficient heat removal capability to handle any design basis accident. However, the containment spray system is more effective in dealing with the superheated steam from a main steam break inside containment. In addition, the containment spray system provides a mechanism for removing iodine from the containment atmosphere. Therefore, at least one train of containment spray is required to be OPERABLE when pressurizer pressure is $\geq$ 1750 psia, and the allowed outage time for one train of containment spray reflects the dual function of containment spray for heat removal and iodine removal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The Technical Requirements Manual contains the list of containment isolation valves (except the containment air lock and equipment hatch). Any changes to this list will be reviewed under 10CFR50.59 and approved by the Plant Operations Review Committee (PORC).

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 FLOOD LEVEL

The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm ± 10%.

Currently there are some situations where the CREV System may not automatically start on an accident signal, without operator action. Under most situations, the emergency filtration fans will start and the CREV System will be in the accident lineup. However, a failure of a supply fan (F21A or B) or an exhaust fan (F31A or B), operator action will be required to return to a full train lineup. Also, if a single emergency bus does not power up for one train of the CREV System, the opposite train filter fan will automatically start, but the required supply and exhaust fans will not automatically start. Therefore, operator action is required to establish the whole train lineup. This action is specified in the Emergency Operating Procedures. The radiological dose calculations do not take credit for CREV System cleanup action until 10 minutes into the accident to allow for operator action.

When the CREV System is checked to shift to the recirculation mode of operation, this will be performed from the normal mode of operation, and from the smoke purge mode of operation.

Docket No. 50-336
B17413

Attachment 5

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
FSAR Change
Main Steam Line Break and Radiological Consequences

September 1998

6.4.4 Availability and Reliability

6.4.4.1 Special Features

The components of the containment spray system are designed to general requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physically separating duplicate equipment, as described in Section 6.1.

To assure the availability of water to the pumps, separate suction headers from the refueling water storage tank are provided for the spray pump located in the two separate and shielded pump rooms, which house the pumps of the engineered safety features systems. Each of the two pump rooms contains one spray pump, one low-pressure safety injection pump and one high-pressure safety injection pump. Two separate headers, one to each of these pump rooms, are also provided from the containment sump.

The containment spray pumps are located in the lowest elevation of the auxiliary building at Elevation (-) 45-6 to assure a flooded suction. This assures pump priming and protects the mechanical seals in the spray pumps. In this location, the available NPSH is always greater than the required NPSH (see Table 6.4-1).

To assure adequate design margins, the available NPSH for the containment spray pumps is conservatively calculated at 27 feet during the recirculation mode in accordance with Safety Guide 1. This is based on a containment sump water level at Elevation (-) 22-6, neglecting containment pressure and assuming that all the safety injection and containment spray pumps are operating at their respective runout conditions. The calculated minimum containment sump water level is at Elevation (-) 15-6. The peak calculated containment pressure is 51.2 psig with 250 F sump water and 14.7 psig with 92 F sump water. 96-6 3/96

To increase system reliability, the containment spray pump motors have the capacity to start with the motor-operated valves on the discharge header fully opened.

The refueling water storage tank (RWST) and containment sump assure sources of water for the containment spray system. These components are described in Section 6.2.

The liberation of combustible gases, resulting from metal corrosion in the containment postaccident environment, is described in Subsection 14.8. The contribution to metal corrosion from a continuous borated water spray within the containment is less than 200 mils/yr. Therefore, with the brief exposure to the containment sprays during postincident conditions, this corrosion is negligible.

~~No credit is taken for the reduction of fission product concentration in the containment atmosphere from the borated spray.~~

A failure mode analysis is given in Table 6.4-2.

Inadvertent initiation of the spray system does not affect the safety of the unit, since within the containment all the instruments are dripproof or weatherproof, all the motors are dripproof or totally enclosed and signal cable runs are enclosed in waterproof jackets. All piping or equipment insulation which may come in contact with sprays are of the metal high radiation levels. Operation of the CRFS is monitored by filter bank differential pressure and temperature indication. Fan operation is monitored by motor trip alarms.

In the event of a LOCA or a fuel handling accident, the control room air conditioning system is automatically switched to the isolation/ recirculation mode. Tests show that the unfiltered in-leakage is less than ~~100 cfm.~~ 130 scfm. (97-72 11/97) (97-72)

A fresh air make-up system will not be used to maintain a positive pressure differential with respect to the external environment or the adjacent internal spaces at any time during the normal or emergency modes of operation.

The control room air conditioning system mode of operation includes an automatic isolation of the system to the complete recirculation mode and automatic initiation of the bypass filtering operation. This automatic switchover to the complete recirculation mode and filtering mode is initiated by the EBFAS or the AEAS.

The post-accident mode of operation is a closed cycle with air intakes and outlets isolated. The control room atmosphere is exhausted from the space, filtered, and cooled as required and returned to the space. Outside air is not introduced into the system unless required for personnel safety. (95-13 8/95)

9.9.10.4 Availability and Reliability

9.9.10.4.1 Special Features

The components of the control room air conditioning system are designed to engineered safety feature requirements including seismic response as described in Section 6.1. All components are protected from missile damage and pipe whip by physical separation of duplicate equipment, as described in Section 6.1.

Each air conditioning subsystem is capable of maintaining a suitable environment within the control room. Each system is designed for the normal control room cooling load which is greater than the cooling requirements under post-accident operation. Each system is completely independent, including the control and filtration systems with the exception of some common ductwork and dampers. Common components such as dampers are isolated during post-accident operation. Control inputs to these devices are overridden. (96-49 1/97) Each subsystem is powered by a separate emergency source (Section 8.3). A failure mode analysis for the control room air conditioning system is given in Table 9.9-17. Although there are common plenums, all ductwork is considered a passive component not subject to a single failure mode. (96-49)

The charcoal filter elements within the CRFS are analyzed to ensure adequate residual heat removal capabilities following any single failure. The analysis concludes that the maximum temperature calculated, based on a radioactive filter inventory which was conservatively assumed to be ten (10) times greater than the maximum inventory calculated resulting from a design basis accident at the site, was less than 212°F (100°C). This is substantially below the charcoal ignition temperature, thus filter bed isolation should not constitute a fire hazard. Temperature indication is provided to alert personnel of excessive charcoal bed temperature. (97-39 12/97)

For the hot shutdown case, the event is initiated by a rapid opening of the atmospher , dump valves or the turbine bypass valves resulting in a steam flow increase of 41% of the nominal full power steam flow. A bounding value for the negative MTC was assumed as was the technical specification value of the shutdown margin. The results of this event for both the one pump and four pump case were found to be bounded by the full power, full flow event.

The responses of key system variables are given in Figures 14.1.3-1 to 14.1.3-7 for the rated power case. The sequence of events is given in Table 14.1.3-3.

14.1.3.7 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted for the full power case is greater than the safety limit. The correlation limit assures that with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold of 21 kW/ft is not violated during this event.

14.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

14.1.4.1 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. 5/90

14.1.4.2 Event Description

The resulting mismatch in energy generation and removal rates results in an overcooling of the primary system. If the MTC is negative, the reactor power will increase.

14.1.4.3 Reactor Protection

Reactor protection is provided by the variable overpower trip, LPD trip, TM/LP 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 14.1.4-1.

14.1.4.4 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 14.1.3. The disposition of events for the Inadvertent Opening of a Steam Generator Relief or Safety Valve event is summarized in Table 14.1.4-2.

~~14.1.5 Steam System Piping Failures Inside and Outside of Containment~~ INSERT C

~~14.1.5.1 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.

14.1.5.2 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 MTC, 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.

14.1.5.3 Reactor Protection

Reactor protection is provided by the low steam generator pressure and water level trips, variable overpower trip, LPD trip, TM/LP trip, high containment pressure trip, and SIAS. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5-1. 5/90

14.1.5.4 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 (SLB) 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 SLB initiated from critical conditions in Mode 2 will bound the results of the event initiated from subcritical Mode 3 conditions.

The technical specifications (Reference 14.1-1) only require a minimum of one RCP 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. 93-18 4/93 Admin

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 MTC. 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 High Pressure Safety Injection (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 14.1.5-2.

14.1.5.5 Definition of Events Analyzed

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The SLB event is initiated by a double ended guillotine break of the main steam line at its largest point between the steam generators and the flow restrictors. This break location leads to an uncontrolled steam release from the secondary. The event occurs concurrent with the most reactive control rod stuck out of the core.

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. In the presence of a negative MTC, this cooldown results in a large decrease in the shutdown margin and a return to power. This return to power is exacerbated because of the high power peaking factors which exist, with the most reactive control rod stuck in its full withdrawn position.

The consequences for the event are bounded by analyzing at both HZP and Hot Full Power (HFP) conditions. At HFP 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. Initiating this event from rated power also has the potential for the highest post-trip power since it maximizes the concentrations of delayed neutrons thus providing for the greatest power rise for a given positive reactivity insertion. If the event occurred at lower power, additional thermal margin is provided by the automatically decreasing setpoint of the variable overpower trip. Thus this event, initiated from full rated power conditions, will bound all other cases initiated from at power operation modes or power levels.

For the zero power and subcritical plant states there is also a potential for a return-to-power. The most limiting SLB event at zero power is one which is initiated at the highest temperature and pressure, thereby providing the greatest capacity for cooldown. The most limiting conditions will occur when the core is critical. This condition will maximize the available positive reactivity and therefore produce the quickest and largest return to power. Thus, the SLB occurring from critical conditions will bound the results of the event initiated from subcritical conditions.

As outlined in References 14.1-2 and 14.1-3, three computerized calculations are required prior to the final calculation of the Minimum Departure From Nucleate Boiling Ratio (MDNBR) values and the maximum Linear Heat Generation Rate (LHGR) values utilized in the determination of fuel failure. The NSSS response is computed with Siemens Power Corporation (SPC-RELAP) (developed from RELAP5/MOD2 (Reference 14.1-4), with SPC modifications (Reference 14.1-2), the detailed core and hot assembly power distributions and reactivity spot checks are computed with the SPC three-dimensional core simulator model, XTG (Reference 14.1-5), and the detailed core and hot assembly flow and enthalpy distributions are computed with XCOBRA-IIIC (Reference 14.1-6). The modified Barnett correlation was utilized to calculate MDNBR due to the reduced pressures occurring during the SLB event. 93-18 4/93

14.1.5.5.1 Analysis of Results

The SPC-RELAP analysis provides the NSSS boundary conditions for the XTG and the XCOBRA-IIIC (Reference 14.1-6) calculations. This section presents a description of the treatment of factors which can have a significant impact on NSSS response and resultant MDNBR and LHGR values. The plant specific parameters used in this analysis are listed in Tables 14.1.5-3 to 14.1.5-5. Conservatisms are included in parameters or factors known to have significant effects on the NSSS performance and resulting MDNBR and LHGR values. 93-18

14.1.5.5.1.1 Break Location, Size, and Flow Model 5/90

The limiting break, a double ended guillotine break, is located inside containment between the steam generator outlet and the flow restrictors. This break location results in the largest cross sectional flow area and will therefore produce the most rapid cooldown and the highest return to power. The break flow areas for the affected and intact steam generators are listed in Table 14.1.5-3. These areas correspond to the locations in the flow path where choked flow will occur.

The SPC-RELAP break mass flow rate is computed using the Moody critical flow model modified such that only steam flows out the break. 93-18

14.1.5.5.1.2 Boron Injection

Boron injection into the primary system acts to mitigate the return to power. Injection of boron is modeled from two sources, the HPSI and the charging system. The characteristics of the HPSI and charging systems are listed in Table 14.1.5-3. Both systems are conservatively assumed in this analysis to take suction from the Refueling Water Storage Tank (RWST). The line volume between the check valves isolating these systems pumps and the cold leg injection location is assumed to initially contain no boron. The time required to sweep this unborated water from these lines with borated water is included as an integral part of the SPC-RELAP NSSS calculation. The delivery curve for the HPSI system used in this analysis is given in Figure 14.1.5-1. 93-18

14.1.5.5.1.3 Single Failure Assumption

The single failure assumed in the engineered safeguards system was the failure of one of the two HPSI pumps required to be in service during normal operation. Also only one charging pump is assumed to be available. This assumption results in an additional delay in the time required for the boron to reach the reactor core. This delay is further amplified when combined with the assumption of a stagnant upper head which serves to maintain primary system pressure due to flashing of the hot fluid in the upper head. Although one charging pump was assumed available, the impact of crediting charging has been evaluated and determined not to invalidate the conclusions of this analysis. 93-4 (6/9

14.1.5.5.1.4 Feedwater

For the HZP scenarios the AF flow was initialized such that steam flow equaled the heat generated by the RCPs. No decay heat is assumed to maximize the cooldown. After the initiation of the transient, the AF flow is allowed to increase with decreasing steam generator pressure assuming a fixed control valve setting. At 180 seconds (conservatively based on technical specifications (Reference 14.1-1), the AF is increased to pump runout flow. All flow is directed into the affected steam generator to maximize the cooldown rate.

In the HFP cases the main feedwater flow will be terminated 30 seconds after the reactor trip occurs due to closure of the feedwater regulator valves. After reactor scram, the feedwater flow increases as the secondary pressure decreases at the lowest possible fluid temperature until the regulators are closed. Fluid temperature is determined by assuming all heating of feedwater ceases after the time of the break. The AF is modeled as in the HZP cases after 180 seconds.

14.1.5.5.1.5 Trips and Delays

Trips for the HPSI, charging system, main feedwater valves, and MSIVs are given in Table 14.1.5-4. Biases to account for uncertainties are included in the trip setpoints as shown. For the steam and feedwater valves, the delay times given are between the time the trip setpoint is reached and the time full valve closure is reached. For the HPSI and charging pumps, the delay time given is from the time the setpoint is reached until the pumps have accelerated to rated speed. Additional delay time required to sweep the lines of unborated water is accounted for by setting the boron concentration of the injected flow to zero until the volume of the injection lines has been injected.

14.1.5.5.1.6 Neutronics

The core kinetics input for this calculation consisted of the minimum required control rod shutdown worth at the EOC, and EOC values associated with the reactivity feedback curves, delayed neutron fraction, delayed neutron fraction distribution and related time constants, and prompt neutron generation time. The SPC-RELAP default fission product and actinide decay constants were utilized for this calculation.

The core reactivity is derived from input of several functions. These include effects from control rod worth, moderator density changes, boron concentration, and Doppler effects. The reactivity is weighted between the core sectors. Different reactivity functions were utilized where necessary for the HZP and the HFP cases. The SPC-RELAP analyses were performed with an MTC of -28 pcm/°F. A summary of the nuclear input and assumptions is given in Table 14.1.5-5.

14.1.5.5.1.7 Decay Heat

The presence of radioisotope decay heat at the initiation of the SLB event will reduce the rate and the extent of cooldown of the primary system. For the HFP case, decay heat was calculated on the basis of infinite irradiation time prior to transient initiation. For the HZP case, there was no decay heat at the transient initiation, but decay heat was calculated based on the core power and the time at power calculated during the transient. This treatment of decay heat serves to maximize the stored energy in the HFP cases and to minimize it in the HZP cases. This treatment provides limiting stored energy conditions for the SLB cases.

14.1.5.5.1.8 Nodalization

The NSSS transient calculations presented in this report utilized the nodalization model described in Reference 2. The nodalization treats all major NSSS components and subcomponents as discrete elements, with the exception of the secondary side of the steam generators. In addition, all components with long axial dimensions are divided into subcells adequate to minimize numerical diffusion and smearing of gradients. 5/90

In order to simulate the asymmetric thermal hydraulic and reactivity feedback effects that occur during an SLB transient, the core is nodalized into three radial sectors. One sector corresponds to the region immediately surrounding the assembly where the most reactive control rod is assumed stuck out of the core. This sector is termed the "stuck rod" sector. The remainder of the region of the core which is directly affected by the loop containing the break is the second sector and is termed the "affected" sector. The remainder of the core and the other loop is termed either "unaffected" or "intact" sector or loop.

14.1.5.5.1.9 Interloop Mixing

During an actual SLB transient, some mixing between the parallel channels within the reactor pressure vessel will occur in the downcomer, the lower plenum, the core, and the upper plenum due to lateral momentum imbalances, and turbulence or eddy mixing. The mixing will act to reduce the positive reactivity feedback effects due to a reduced rate and magnitude of cooldown of the affected loop and associated core sector.

In this analysis, no credit is taken for turbulent or eddy mixing of coolant between loops or the parallel flow channels within the reactor pressure vessel (RPV). However, interloop mixing is calculated to occur due to flow in interloop junctions in the upper and lower plenums. Mixing in the lower plenum was reduced to a minimum by using an extremely high loss coefficient between the affected and intact sectors.

14.1.5.5.2 Minimum Departure From Nucleate Boiling and Linear Heat Generation Rate

MDNBR calculations require determination of the power, enthalpy, and flow distributions within the highest power assembly of the stuck rod core sector. Similarly, determination of the maximum LHGR also requires characterization of the power distribution. The power distribution within the core, including the highest powered assembly within the stuck rod core sector, is calculated with XTG (Reference 14.1-5). Flow and enthalpy distributions within the core, including the highest powered assembly within the stuck rod core sector, are calculated with XCOBRA-IIIC (Reference 14.1-6). In order to obtain compatible flows, moderator densities, and powers within the high power assemblies, iteration between XTG and XCOBRA-IIIC is conducted. 93-18 4/93 Admin 93-18 Admin

For this calculation, the modified Barnett correlation was found to be suitable for the MDNBR calculation. The modified Barnett correlation is based upon closed channels and primarily uniform power distribution data. The correlation is based on assembly inlet (or upstream) fluid conditions rather than on local fluid conditions as is the case with subchannel based correlations. Use of the correlation is limited to the range of the data base unless conservative extrapolations can be made.

14.1.5.6 Analysis Results

A summary of calculated results important to this analysis is presented in Table 14.1.5-6 for the four scenarios analyzed. The MDNBR values are listed together with the corresponding core power values at the time of MDNBR which corresponds to the maximum post scram power level. For cases where offsite power is available for operation of the primary coolant system pumps, MDNBR and maximum LHGR occurs at the maximum power condition. For cases where offsite power is lost and the primary system pumps coast down, the maximum LHGR and MDNBR, however, occur when the worst combination of core power, flow, inlet temperature, and pressure is present. These conditions occurred at the time of peak power in this analysis. 5/90

The scenario which results in the highest post scram power level and largest LHGR is that initiated from HZP with offsite power available for operation of the primary coolant pumps. The general post trip response of the NSSS for the HFP scenario with offsite power available is comparable to that for the HZP scenario. One exception is the post scram subcritical core power response during the initial portion of the transient. The post scram subcritical power response is different for the HFP case due to delayed neutron and stored energy effects. In the HFP case the scram shutdown margin is large enough that by the time the reactivity reaches zero most of the delayed neutrons are no longer in the system inhibiting a return to power. Because the HZP case results in a higher power level and higher LHGR, it is presented in detail.

The NSSS responses for the scenarios with loss of offsite power for operation of the primary system coolant pumps are different from those scenarios where offsite power is available throughout the transient due to the pump coastdown and subsequent natural circulation of the primary coolant. Post scram maximum power levels attained during the transient are significantly lower. Lower power levels result from lower positive moderator feedback. The positive moderator feedback is reduced due to the coolant density reductions that occur axially upwards in the core at low core flow rates, even for low core power levels. Lower power levels cause MDNBR values to increase, but lowering flow rates cause MDNBR values to decrease. Overall, the combination of factors results in lower MDNBR values for the reduced flow condition than for the full flow condition.

Of the two loss of offsite power scenarios analyzed, the HZP case results in lower MDNBR values. The general response of the HFP and HZP cases with loss of offsite power is comparable. Again, the exception is the post scram subcritical core power response during the initial portion of the transient. The post scram subcritical power response is different for the HFP case due to delayed neutron and stored energy effects. Because the two scenarios are quite similar in terms of their general response, only the limiting MDNBR case (i.e., HZP without offsite power) is presented in detail.

14.1.5.6.1 Hot Zero Power with Offsite Power Available

The SPC-RELAP simulation of the NSSS during the HZP transient with offsite power available is illustrated in Figures 14.1.5-2 through 14.1.5-7. A tabulation of the sequence of events is presented in Table 14.1.5-7. The SPC-RELAP computation was terminated 600 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR. Beyond 600 seconds, core reactivity would become more subcritical as dryout of the steam generator occurs following AF termination. Termination of the AF by manual operator action was assumed to occur 600 seconds after initiation of the break. 93-18 4/93 93-18

14.1.5.6.1.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Steam flow for the affected generator is plotted in Figure 14.1.5-2. The affected steam generator continues to blow down through the break throughout the transient. The pressure and mass flow rate drop rapidly at first and then proceed downward at a slower decay rate until 238 seconds. At that time, the cool AF condenses a significant quantity of steam and the break flow essentially goes to zero. The cooldown of the secondary side produces a change in heat transfer regimes between the primary and secondary which results in a heatup of the primary coolant. The higher temperature reduces the reactivity present and power drops rapidly. 5/90

The intact steam generator blows down for a short period until the MSIVs completely close approximately 10.5 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system and then slowly decays as the intact steam generator begins to act as a heat source to the primary system.

14.1.5.6.1.2 Primary System Thermal Hydraulic Parameters

The primary system coolant temperature and pressure responses resulting from the break flow are illustrated in Figures 14.1.5-3 through 14.1.5-5. The primary system pressure decays rapidly as the coolant contracts due to cooldown and the pressurizer liquid empties. The MSIVs close at 10.5 seconds, ending the blowdown of the intact steam generators and reducing the rate of energy removal from the primary fluid. Primary system pressure recovers somewhat at that point, and then increases slowly for the duration of the transient.

14.1.5.6.1.3 Reactivity and Core Power

The reactivity transient calculated by SPC-RELAP is illustrated in Figure 14.1.5-6. Initially, the core is assumed to be critical at HZP. All control rods, except the most reactive one, are assumed to be inserted into the core following the first reactor trip signal. The reactivity transient then proceeds. Cooldown of both the coolant and fuel brings the core critical due to moderator and Doppler reactivity feedback. Shortly thereafter, power begins to rise steadily due to the dominating positive reactivity feedback from the moderator. 93-18

The HPSI and charging flow is actuated 45.2 seconds after the break. Borated water passes through the core 153 seconds after the break initiation following a 30 second HPSI delay, a 40 second charging pump startup delay, a line flushing delay, and a transport delay between the cold leg injection point and the core. Entry of borated water into the core helps override the positive moderator feedback and helps to terminate the increase in core power. Core power then begins a decline as the concentration of boric acid increases with time. At 238 seconds, the power drops rapidly due to increasing primary coolant temperature. Following the rapid power drop, the power declines much slower as the boron concentration increases. Terminating AF 600 seconds after the break will subsequently cause the primary coolant to heat up. This, combined with the ever increasing boron concentration, will terminate the SLB event.

Figure 14.1.5-7 shows the transient reactor power. The maximum power level is 686 MWt or 25% of rated power at 153 seconds after the break initiation. 5/90

14.1.5.6.1.4 XTG and XCOBRA-IIIC Results

The XTG calculation is made initially on the basis of SPC-RELAP input. Each assembly within the three channels is assumed to have a uniform flow corresponding to the sector flows calculated with SPC-RELAP. Due to high power peaking in the region of the stuck control rod, large moderator density reductions are calculated to occur in the top portions of several assemblies in this region of the core in the XTG calculation. This moderator density decrease is a major factor in the flattening of the axial and radial profiles, and the significant reduction in reactivity observed when XTG is compared to SPC-RELAP. 93-18 4/93 93-18 93-18

The SPC-RELAP reactivity and power calculation has considerable inherent conservatism. To demonstrate this, a comparison of the change in reactivity at the maximum LHGR time is made. A comparison of the overall change in reactivity between SPC-RELAP and XTG shows that SPC-RELAP conservatively underestimates the negative reactivity by 2.31 \$ at the start of the transient and overestimates the reactivity at maximum LHGR time by 8.74 \$, thus indicating that the SPC-RELAP power calculation is conservative. It should be noted that the XTG calculated reactivities are best estimate at both the initial and maximum LHGR conditions. 93-18 93-18 93-18

An XCOBRA-IIIC core analysis was conducted to define the flow and enthalpy distribution within the high power assembly. The XCOBRA-IIIC core flow distribution analysis indicates that the flow, and therefore moderator density, in the upper elevations of the high power assembly is greater in the closed channel XTG calculation than the open channel XCOBRA-IIIC calculation. The power distribution and reactivity calculated by XTG are therefore conservative.

14.1.5.6.1.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate

For the MDNBR portion of the calculation, the radial power distribution was modified to conservatively account for local rod power distribution affects within the hot assembly. This was done by raising the power of the hot assembly by an additional 15% to bound the peak rod power.

On the bases of these conservative assumptions, the MDNBR value was calculated to be 2.40. This compares to a 95/95 DNBR limit of 1.135 for the modified Barnett correlation.

Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR standpoint.

The analysis of the peak LHGR also comes from the XTG and XCOBRA-IIIC analysis. The peak LHGR is calculated from the SPC-RELAP total core power and the XTG radial and axial peaking. The peak LHGR, 20.994 kW/ft, was calculated for the case of HZP with offsite power available. Comparing this LHGR with a centerline melt criteria of 21 kW/ft, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria. 93-18 4/93

14.1.5.6.2 Hot Zero Power with Loss of Offsite Power

The SPC-RELAP NSSS simulation of the most limiting SLB scenario from an MDNBR standpoint (i.e., HZP with loss of offsite power) is illustrated in Figures 14.1.5-8 through 14.1.5-13. A tabulation of the sequence of events is presented in Table 14.1.5-8. Termination of the AF by manual operator action was assumed to occur 600 seconds after initiation of the break. This is well beyond the time of MDNBR and maximum LHGR. Following termination of AF, core reactivity would become more subcritical due to continued addition of boron and eventual dryout of the affected steam generator. 93-18

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14.1.5.6.2.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Steam flow for the affected steam generator is plotted in Figure 14.1.5-8. The affected steam generator continues to blow down through the break throughout the transient. The pressure and mass flow rate drop rapidly at first and then proceed downward at a slower decay rate.

The intact steam generators blow down for a short period until the MSIVs completely close approximately 10.5 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system. Subsequently, the intact steam generator pressure remains essentially constant as the primary intact coolant loop approaches natural circulation conditions.

14.1.5.6.2.2 Primary System Thermal Hydraulic Parameters

The primary system core coolant temperature and pressure responses resulting from the break flow are illustrated in Figures 14.1.5-9 through 14.1.5-11. The primary system pressure decays rapidly as the coolant contracts due to the cooldown and the pressurizer empties. Continued pressure reduction in the primary system causes the relatively hot stagnant liquid in the head of the RPV vessel to flash. The flashing in the upper head, coupled with near equilibration of other NSSS parameters, retards the pressure decay from that point forward. The elevated pressure acts to limit the delivery of boron into the core due to the pressure versus flow characteristics of the HPSI system.

A comparison of intact and affected core sector inlet temperatures throughout the transient indicates significant differences due to the limited cross flow allowed between loops. The core sector flows all show the same trend due to the coastdown of the primary coolant pumps. That is, all flows decrease rapidly until natural circulation conditions are achieved in the two flow loops.

14.1.5.6.2.3 Reactivity and Core Power

The reactivity transient calculated by SPC-RELAP is illustrated in Figure 14.1.5-12. Initially, the core is assumed to be critical at HZP. All control rods, except the most reactive one, are assumed to be inserted into the core at the start of the transient. Cooldown of both the coolant and fuel brings the core critical due to moderator and Doppler reactivity feedback. After reaching criticality, the power spikes momentarily, but fuel temperature rises rapidly and Doppler feedback effects rapidly reduce core reactivity and power. Shortly thereafter, power rises again, then stabilizes as the affected core sector average moderator temperature stabilizes. Although the affected core sector inlet temperature continues to decrease during this period, the flow rate is also decreasing, thus stabilizing the affected core sector average moderator temperature. 93-18 4/93

The HPSI and charging flow is actuated 48.7 seconds after the break and the shutdown effect of boron is superimposed upon the other reactivity feedback effects. Borated water passes through the core 152 seconds after the break initiation, following a line flushing delay and a transport delay between the cold leg injection point and the core. Entry of borated water into the core initiates a general power descent which would ultimately bring the reactor to a shutdown condition as the concentration of boron increases with time. Terminating AF 600 seconds after the break will subsequently cause the primary to heat up. This, combined with the ever increasing boron concentration, will finally terminate the SLB event. 5/90

The transient experienced by the core power is illustrated in Figure 14.1.5-13. A small power spike is calculated to occur at 63 seconds after the break is initiated. However, it is of such short duration that fuel temperatures and core heat flux do not increase sufficiently to cause any DNB concern at that particular point in time. The next maximum power level is 293 MWt or 11% of rated at 169 seconds after the break initiation.

14.1.5.6.2.4 XTG and XCOBRA-IIIC Results

The XTG calculation is initially made on the basis of SPC-RELAP predicted core power, flow, pressure, and inlet temperatures. The XTG calculations provide the radial and axial power distributions for use in the XCOBRA-IIIC code. Due to the high power peaking in the region of the stuck control rod, and the low core average natural circulation flow rates, large moderator density decreases are calculated in several assemblies in this region in the XTG calculation. This is a major factor in the flattening of the axial and radial profiles, and the significant reduction in reactivity observed when XTG is compared to SPC-RELAP. XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly. 93-18 93-18

The absolute difference in reactivity between SPC-RELAP and XTG indicates that the SPC-RELAP power calculation is conservative. The SPC-RELAP reactivities at HZP and the MDNBR point are calculated to be -8.00 \$ and 0.046 \$, respectively and the XTG values are calculated to be -10.31 \$ and -9.91 \$. 93-18

14.1.5.6.2.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

A similar approach to that taken for the HZP power available case was utilized for the most limiting MDNBR case, which is the HZP scenario with loss of offsite power. The MDNBR of the hot fuel assembly is calculated to be 1.18. This compares to a 95/95 DNBR limit of 1.135 for the modified Barnett correlation. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR standpoint.

As before the analysis of the peak LHGR comes from the XTG and XCOBRA-IIIC analysis. The peak LHGR was 16.5 kW/ft. Comparing this LHGR with a centerline melt criteria of 21 kW/ft, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.6.3 Hot Full Power Without Offsite Power Available

The sequence of events for the case is presented in Table 14.1.5-9. For reasons presented in Section 14.1.5.6 this case was not discussed in detail.

14.1.5.6.4 Hot Full Power With Offsite Power Available

The sequence of events for the case is presented in Table 14.1.5-10. For reasons presented in Section 14.1.5.6 this case was not discussed in detail.

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14.1.5.7 Conclusions

The HZP scenario with loss of offsite power was determined to be the most limiting in this analysis from an MDNBR standpoint. The HFP and HZP scenarios, with offsite power maintained for operation of the primary coolant pumps resulted in a return to higher power levels than the scenarios where offsite power is lost. However, these scenarios provide substantially greater margin to the MDNBR limit because of the higher coolant flow rate. In no scenario evaluated, however, was fuel failure calculated to occur as a result of penetration of the MDNBR safety limit.

The HZP scenario with offsite power available was determined to be the most limiting in this analysis from the standpoint of centerline melt. This scenario results in the highest return to power and highest calculated LHGR of 20.9 kW/ft. The HFP and HZP scenarios with offsite power maintained for operation of the primary coolant pumps returned to higher power levels than the scenarios where offsite power is lost. Even though these scenarios have substantially greater margin to the MDNBR limit because of a higher coolant flow rate, the higher power levels in combination with the highly skewed power distribution due to the assumed stuck rod cluster resulted in them having the least margin to the fuel centerline melt limit.

REFERENCES

14.1-1 Technical Specifications for Millstone Unit 2, Docket No. 50-336, Updated through Amendment No. 116.

14.1-2 SPC-84-93(P), Supp. 1, "Steam Line Break Methodology for PWRs," Advanced Fuels Company Richland, WA, June 1988.

14.1-3 XN-NF-84-93(P), "Steamline Break Methodology for PWRs," Exxon Nuclear Company, Richland, WA, November 1984.

14.1-4 "RELAP5/MOD2 Code Manual, Volume 1: System Models and Numerical Methods; Volume 2, Users Guide and Input Requirements," NUREG/CR-4312, EGG-2396, Revision 1, EG&G Idaho, Inc., Idaho Falls, ID 83415, March 1987.

14.1-5 "XTG-- A Two-Group Three Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing," XN-CC-28(A), Revision 3, Exxon Nuclear Company, Richland, WA 99352, October 1978.

14.1-6 "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.

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The following pages are "Insert C," Section 14.1.5 and Section 14 References:

14.1.5 Steam System Piping Failures Inside and Outside of Containment

Two separate analyses have been performed for the steam line break event. Section 14.1.5.1 describes the pre-scram analysis performed to determine Departure from Nucleate Boiling Ratio (DNBR) and Linear Heat Generation Rate (LHGR) up to and including reactor trip. This time period represents the highest reactor power condition and the assumptions have been selected to minimize DNBR and maximize LHGR during this time frame. Section 14.1.5.2 describes the post-scram analyses performed to determine MDNBR and LHGR during the return to power caused by the overcooling. A different set of assumptions and single failure were determined to minimize MDNBR and maximize LHGR for the return to power time frame.

14.1.5.1 Pre-Scram Analysis

14.1.5.1.1 Event Initiator

The pre-scram SLB analysis is initiated by a rupture in the main steam piping which results in an uncontrolled steam release from the secondary system.

14.1.5.1.2 Event Description

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. With a negative MTC, the primary system cooldown causes the reactor power level to increase. If the break is not large enough to trip the reactor on a Low Steam Generator Pressure signal, the cooldown will continue until the reactor is tripped on a Variable Overpower or TM/LP signal (for breaks outside containment) or a High Containment Pressure signal (for breaks inside containment) or until the reactor reaches a new steady-state condition at an elevated power level.

Although the SLB calculation is typically a cooldown event, for the pre-scram analysis the cooldown event is not significant for the limiting pre-scram case. The case with a loss of offsite power, also known as a "pumps off" case, credits the low reactor coolant flow trip for harsh conditions. In this case, the Reactor Coolant Pumps (RCPs) are tripped shortly after the initiation of the transient. The sharp reduction in reactor coolant flow causes the pre-scram pumps off calculation to become a heat up transient very similar to a Loss of Coolant Flow (LOCF). Therefore, the conditions for this case are biased as if it were a LOCF (i.e. BOC neutronics). This case becomes a combination of an MSLB and an LOCF event.

14.1.5.1.3 Reactor Protection

Reactor protection is provided by the low steam generator pressure and water level trips, variable overpower trip, LPD trip, TM/LP trip, high containment pressure trip, low reactor coolant flow, and SIAS. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.1-1.

14.1.5.1.4 Disposition and Justification

HFP initial conditions are limiting for the pre-scram SLB cases since this is the highest power condition.

The outside containment breaks do not cause harsh conditions inside containment, and therefore, do not cause the Low Reactor Coolant Flow trip to be degraded. If a loss of offsite power were concurrent with an outside containment break, the primary coolant flow rate would coastdown similar to an LOCF event, without the Low Reactor Coolant Flow trip being degraded. The outside containment break case with loss of offsite power is therefore bounded by the LOCF event.

The inside containment breaks do cause harsh conditions inside containment, and therefore, an increased allowance for instrument uncertainty was applied for the Low Reactor Coolant Flow trip. Therefore, only the inside containment breaks will be analyzed with a loss of offsite power.

The following pre-scram HFP Steam Line Break cases for break sizes ranging up to a double-ended guillotine break in a main steam line were analyzed, with the effects of power decalibration and harsh containment conditions (where applicable) included in the analysis:

1. Breaks outside containment and downstream of the check valves (symmetric cases)
2. Breaks outside containment and upstream of a check valve (asymmetric cases)
3. Breaks inside containment with RCPs on (asymmetric cases)
4. Breaks inside containment with RCPs off (asymmetric cases)

The event is analyzed to support the technical specification EOC MTC limit. This event must be analyzed both with and without a coincident loss-of-offsite power.

The single failure assumed in this analysis is the loss of one channel of Nuclear Instrumentation (NI) which provides power indication to the RPS. If one channel is out of service, the three remaining NI safety channels will be in a 2-out-of-3 coincidence mode. With the assumption of a failure in one of these channels, both of the remaining channels are required for a trip, relying on the lowest power indication for the safety function.

The disposition of events for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.1-2.

14.1.5.1.5 Definition of Events Analyzed

The pre-scram SLB event is initiated by a rupture in the main steam piping. The break location is downstream of the steam generator integral flow restrictor and either

1. outside containment and upstream of the main steam line check valves (asymmetric break), or

2. outside containment and downstream of the main steam line check valves (symmetric break), or

3. inside containment and upstream of the main steam check valves (asymmetric break).

Steam released through a break located downstream of the main steam line check valves flows to the break from both steam generators and, therefore, results in a symmetric transient. However, steam released through a break located upstream of one of the check valves flows to the break from the upstream steam generator only (because the check valve precludes backflow to the break from the other steam generator) and, therefore, results in an asymmetric transient.

Power decalibration is caused by density-induced changes in the reactor vessel downcomer shadowing of the power-range ex-core detectors during heatup or cooldown transients. The nuclear power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full-power operation (and higher when the vessel inlet coolant is warmer than the normal full-power temperature). This effect is included in the modeling of any power-dependent reactor trips credited in the analysis of full-power cooldown events and low-power events. The Variable Overpower trip, the Thermal Margin/Low Pressure (TM/LP) trip function, and the Local Power Density (LPD) trip all depend on the indicated nuclear power level.

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

As outlined in Reference 14.1-1 , three computerized calculations are required prior to the final calculation of the Minimum Departure From Nucleate Boiling Ratio (MDNBR) values and the maximum Linear Heat Generation Rate (LHGR) values utilized in the determination of fuel failure. The NSSS response is computed using the Siemens Power Corporation (SPC) ANF-RELAP code (Reference 14.1-2), the detailed core and hot assembly power distributions and the reactivity at the time of peak post-scram power are calculated using the SPC XTGPWR code (Reference 14.1-3), and the detailed core and hot assembly flow and enthalpy distributions are calculated using the SPC XCOBRA-IIIC code (Reference 14.1-4). The SPC XNB correlation was utilized to calculate MDNBR.

14.1.5.1.5.1 Analysis of Results

The ANF-RELAP analysis provides the NSSS boundary conditions for the XTGPWR and the XCOBRA-IIIC calculations. This section presents a description of the treatment of factors which can have a significant impact on NSSS response and resultant MDNBR and LHGR values. The plant specific parameters used in this analysis are listed in Tables 14.1.5.1-3 to 14.1.5.1-5. Conservatisms are included in parameters or factors known to have significant effects on the NSSS performance and resulting MDNBR and LHGR values.

14.1.5.1.5.1.1 Break Location, Size, and Flow Model

The pre-scram SLB event analyzes breaks outside containment both downstream (symmetric cases) and upstream (asymmetric cases) of the main steam line check valves and breaks inside containment (asymmetric cases). A full range of break sizes, up to the double-ended guillotine break of a main steam line, were considered.

The ANF-RELAP break mass flow rate is computed using the Moody critical flow model modified such that only steam flows out the break.

14.1.5.1.5.1.2 Power Decalibration

Power decalibration is caused by density-induced changes in the reactor vessel downcomer shadowing of the power-range ex-core detectors during heatup or cooldown transients. The nuclear power levels indicated by those instruments are lower than the actual reactor power levels when the coolant entering the reactor vessel is cooler than the normal temperature for full-power operation (and higher when the vessel inlet coolant is warmer than the normal full-power temperature). This effect is included in the modeling of any power-dependent reactor trips credited in the analysis of full-power cooldown events and low-power events. The Variable Overpower trip, the Thermal Margin/Low Pressure (TM/LP) trip function, and the Local Power Density (LPD) trip all depend on the indicated nuclear power level.

14.1.5.1.5.1.3 Harsh Containment Conditions

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

14.1.5.1.5.1.4 Boron Injection

Boron injection into the primary system acts to mitigate the return to power. Injection of boron is modeled from the HPSI system. The HPSI system is conservatively modeled to take suction from the Refueling Water Storage Tank (RWST) at 35°F with a boron concentration of 1720 ppm. Initially, the line volume between the check valves isolating the system pumps and the cold leg injection location is assumed to be filled with unborated water. The time required to flush this unborated water from the safety injection lines is included as an integral part of the ANF-RELAP NSSS calculation. In the pre-scram SLB event, the analysis is terminated shortly after reactor trip, therefore injection of borated water is not a factor in the analysis.

14.1.5.1.5.1.5 Single Failure Assumption

In order to simulate the asymmetric thermal-hydraulic and reactivity feedback effects that occur during the pre-scram SLB event, the core is divided into an affected sector (1/2 of the core) and an unaffected sector (1/2 of the core). The single failure assumed in this analysis is the loss of one channel of Nuclear Instrumentation (NI) which provides power indication to the Reactor Protection System (RPS). If one channel is out of service, the three remaining NI safety channels will be in a 2-out-of-3 coincidence mode to cause a reactor trip. The excore detectors are placed around the reactor vessel is positions that result in one detector seeing the flux only from the affected region, one seeing the flux only from the unaffected region, and two detectors seeing nearly equal flux from both regions. If one of these latter two is out of service, and the other is assumed to be a single failure, the remaining two channels will be required to cause an RPS trip (high power or TM/LP). Since the power in the affected region will always be higher than in the unaffected region, it is sufficient to model the NI channel reading the unaffected region only.

14.1.5.1.5.1.6 Feedwater

Normal MFW flow is assumed to be delivered to both SGs. The MFW flow increases as the secondary pressure decreases at the lowest possible fluid temperature until the feedwater regulator valve closes. Fluid temperature is determined by assuming heating of the feedwater ceases at the same time the break is initiated. The MFW flow is terminated 14 seconds after receiving the isolation signal.

14.1.5.1.5.1.7 Trips and Delays

Actuation signals and delays are given in Table 14.1.5.1-4. Biases to account for uncertainties are included in the trip setpoints as shown. In the pre-scram SLB event, the analysis is terminated shortly after reactor trip, therefore injection of borated water is not a factor in the analysis.

14.1.5.1.5.1.8 Neutronics

The core kinetics input for this calculation consisted of the minimum required control rod shutdown worth at EOC, and EOC values associated with the reactivity feedback curves, delayed neutron fraction, delayed neutron fraction distribution and related time constants, and prompt neutron generation time. The ANF-RELAP default fission product and actinide decay constants were utilized for this calculation.

The core reactivity is derived from input of several functions. These include effects from control rod worth, moderator density changes, boron concentration, and Doppler effects. The reactivity is weighted between the core sectors. The ANF-RELAP analyses for cases with offsite power available were performed with an MTC of -28 pcm/°F. The ANF-RELAP analyses for cases with a loss of offsite power were performed with an MTC of +4.0 pcm/°F. A summary of the nuclear input and assumptions is given in Table 14.1.5.1-5.

14-1.5.5.1.9 Decay Heat

The presence of radioisotope decay heat at the initiation of the SLB event will reduce the rate and the extent of cooldown of the primary system. The initial decay heat is calculated on the basis of infinite irradiation time at a power of 2754 MW prior to transient initiation. This treatment of decay heat serves to maximize the stored energy and provide limiting stored energy conditions for the SLB cases.

14.1.5.1.5.1.10 Nodalization

The NSSS transient calculations utilized the nodalization model described in Reference 14.1-1. The nodalization treats all major NSSS components and subcomponents as discrete elements, with the exception of the secondary side of the steam generators. In addition, all components with long axial dimensions are divided into subcells adequate to minimize numerical diffusion and smearing of gradients.

In order to simulate the asymmetric thermal-hydraulic and reactivity feedback effects that occur during the pre-scram SLB event, the core is divided into an affected sector (1/2 of the core) and an unaffected sector (1/2 of the core).

14.1.5.1.5.1.11 Interloop Mixing

During an actual SLB transient, some mixing between the parallel channels within the reactor pressure vessel will occur in the downcomer, the lower plenum, the core, and the upper plenum due to lateral momentum imbalances, and turbulence or eddy mixing. The mixing will act to reduce the positive reactivity feedback effects due to a reduced rate and magnitude of cooldown of the affected loop and associated core sector.

In this analysis, no credit is taken for turbulent or eddy mixing of coolant between loops or the parallel flow channels within the reactor pressure vessel. However, interloop mixing is calculated to occur due to flow in interloop junctions in the upper and lower plenums. Mixing in the lower plenum was effectively reduced to zero by using an extremely high loss coefficient between the affected and intact sectors.

14.1.5.1.5.2 Minimum Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Analysis

The XTGPWR (Reference 14.1-3) core neutronics code is used to calculate the core radial power distributions for XCOBRA-IIIC (Reference 14.1-4) during the asymmetric transients with offsite power available only. The XTGPWR model is a three-dimensional representation of the entire core, with four radial nodes and 24 axial nodes for each fuel assembly.

Based on the overall core conditions calculated by ANF-RELAP for the symmetric cases (or ANF-RELAP and XTGPWR for the asymmetric cases with offsite power available) at the peak heat flux time-point, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single "channel") linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single "channel"). The limiting assembly DNBR calculations are performed using the XNB DNB correlation (Reference 14.1-4).

For the asymmetric transients, the radial power peaking is augmented above the Technical Specification limit to account for the increase in radial power peaking which occurs during the transient. The increase in peaking is determined by XTGPWR.

14.1.5.1.6 Analysis Results

A summary of calculated results important to this analysis is presented in Table 14.1.5.1-6 for the limiting MDNBR and LHGR cases. The MDNBR values are listed together with the corresponding core power values at the time of MDNBR which corresponds to the maximum power level. For cases where offsite power was available for operation of the primary coolant system pumps, the MDNBR and the maximum LHGR occurred at the time of the maximum power condition. For cases where offsite power is lost and the primary system pumps coast down, the maximum LHGR and the MDNBR occur when the worst combination of core power, flow, inlet temperature, and pressure are present. These conditions occurred at the time of peak power in this analysis.

The scenario which resulted in the highest power level and the largest LHGR is the HFP 3.50 ft^2 symmetric break outside containment with offsite power available for operation of the primary coolant pumps. This case is presented in detail.

The scenario which resulted in the limiting MDNBR is the HFP case with a loss of offsite power and is also presented in detail.

14.1.5.1.6.1 Hot Full Power 3.50 ft^2 Break Outside Containment and Downstream of a Check Valve with Offsite Power Available

The ANF-RELAP simulation of the NSSS during the HFP symmetric break transient with offsite power available is illustrated in Figures 14.1.5.1-1 through 14.1.5.1-6. A tabulation of the sequence of events is presented in Table 14.1.5.1-7. The ANF-RELAP computation was terminated 60 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR. The general response of the reactor was the same for all the symmetric break sizes but the occurrence of events was delayed as the break size decreased.

14.1.5.1.6.1.1 Secondary System Parameters

Upon break initiation the break flow increased sharply and then began to decline in response to falling secondary side pressure. When the turbine trip occurred, the break flow increased due to a local pressure increase. The main steam line flow rate from each generator initially increased (see Figure 14.1.5.1-6) in response to the break and the assumed instantaneous full opening of the turbine control valves. The increased steam flow creates a mismatch between the core heat generation rate and the steam generator heat removal rate. This power mismatch causes the primary-to-secondary heat transfer rate to increase, which in turn causes the primary system to cool down (see Figure 14.1.5.1-2). When the reactor scram occurred, the turbine valves closed and steam flow declined sharply. At this point, the MFW flow may exceed the steam flow as the control system attempts to restore steam generator mass. Both steam flow and MFW flow were terminated when the main steam isolation valves closed.

14.1.5.1.6.1.2 Primary System Parameters

Approximately five seconds after the break occurred, the core inlet temperature began to decline. With a negative MTC (see Figure 14.1.5.1-3), the primary system cooldown caused the reactor power level to increase. The core power continues to increase until reactor scram on low steam generator pressure occurs. This terminated the power excursion. The pressurizer pressure and level began to decline as the volume of water in the primary system shrank. The core inlet mass flow rate increased due to the increasing density of the primary system fluid while the reactor coolant pumps' speed remained constant.

14.1.5.1.6.1.3 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for this scenario was calculated to be 1.298 which is above the 95/95 XNB correlation limit. Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The peak LHR for the LHR-limiting case (3.50 ft^2 break outside containment and downstream of a check valve) is calculated to be 19.7 kW/ft. Comparing this LHGR value with a centerline melt criteria of 21 kW/ft, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.1.6.2 Hot Full Power 3.51 ft^2 Inside Containment Asymmetric Break Concurrent with a Loss of Offsite Power

The ANF-RELAP NSSS simulation of the most limiting pre-scram SLB scenario from an MDNBR standpoint (i.e., HFP 3.51 ft^2 inside containment asymmetric break concurrent with a loss of offsite power) is illustrated in Figures 14.1.5.1-7 through 14.1.5.1-11. A tabulation of the sequence of events is presented in Table 14.1.5.1-8. The ANF-RELAP computation was terminated 60 seconds after break initiation. This is well beyond the time of MDNBR or peak LHGR.

The transient is initiated by the opening of the break. The RCPs tripped shortly after transient initiation. The sharp reduction in the reactor coolant flow causes this pre-trip pumps off calculation to become a heat up transient very similar to a Loss of Coolant Flow event. Typically, the Steam Line Break calculation is a cooldown event. Because this case is a heat up event the most positive BOC neutronics conditions are used, and the maximum inside containment asymmetric break size is used. The maximum break size causes the biggest decrease in primary pressure. Maximizing the primary system pressure decrease causes the maximum decrease in moderator density and the maximum positive moderator feedback. The RCP trip causes the RCS flow to decrease rapidly throughout this transient. The decreasing RCS flow causes the transient time of the fluid in the core to increase and the fluid temperature begins to rise. The increasing fluid temperature causes positive moderator feedback, which in turn causes an increase in core power. However, the decreasing RCS flow causes the heat transfer to the fluid to decrease. The increase in core power is offset by the decrease in heat transfer from the fuel rods, such that, the fuel rod heat flux decreases slightly until reactor scram. The reactor scrams on the low reactor coolant flow trip signal.

14.1.5.1.6.2.3 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR value for the pre-scram 3.51 ft^2 asymmetric break inside containment with a loss of offsite power was calculated to be 0.88 which is below the 95/95 XNB correlation limit. The number of failed assemblies is determined by comparing the core power distribution to the assembly power where DNB occurs. This results in a predicted failure of 3.7% of the fuel rods in the core.

The peak LHR for this case is bounded by the 3.50 ft^2 outside containment symmetric break. Therefore, the LHGR for this case is below the criteria of 21.0 kW/ft and no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.1.7 Conclusions

The HFP 3.50 ft^2 break outside containment and downstream of a check valve (symmetric break) with offsite power available was determined to be the most limiting in this analysis from an LHGR standpoint (19.7 kW/ft). In no scenario evaluated, however, was fuel failure calculated to occur as a result of violating the 21 kW/ft fuel centerline melt criteria.

The HFP 3.51 ft^2 asymmetric break inside containment coincident with a loss of offsite power was determined to be the most limiting in this analysis from the standpoint of MDNBR. The MDNBR was calculated to be 0.88 which is below the 95/95 XNB correlation limit. This results in a predicted failure of 3.7% of the fuel rods in the core.

14.1.5.2 Post-Scram Analysis

14.1.5.2.1 Event Initiator

This event is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the MSIVs which results in an uncontrolled steam release from the secondary system.

14.1.5.2.2 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 MTC, 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.

14.1.5.2.3 Reactor Protection

Reactor protection is provided by the low steam generator pressure and water level trips, variable overpower trip, LPD trip, TM/LP trip, high containment pressure trip, and SIAS. Reactor protection for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.2-1.

14.1.5.2.4 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 (SLB) 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 SLB initiated from critical conditions in Mode 2 will bound the results of the event initiated form subcritical Mode 3 conditions.

The technical specifications only require a minimum of one RCP 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 the technical specification EOC MTC limit. 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 High Pressure Safety Injection (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. The disposition of events for the Steam System Piping Failures Inside and Outside of Containment event is summarized in Table 14.1.5.2-2.

14.1.5.2.5 Definition of Events Analyzed

The post-scram SLB is initiated by a rupture in the main steam piping downstream of the integral steam generator flow restrictors and upstream of the MSIVs which results in an uncontrolled steam release from the secondary system. The effects of harsh containment conditions (where applicable) are included in the following analyses:

1. HFP and HZP breaks outside containment with offsite power available
2. HFP and HZP breaks outside containment with a loss of offsite power
3. HFP and HZP breaks inside containment with offsite power available
4. HFP and HZP breaks inside containment with a loss of offsite power

The event is analyzed to support the technical specification EOC MTC limit. This event must be analyzed both with and without a coincident loss-of-offsite power.

The single failure assumed in this analysis results in the disabling of one of the two HPSI pumps required to be in service during normal operation. In addition to the single failure, there is no credit taken for the charging pump system. This assumption results in an additional delay in the time required for boron to reach the core. The delay is amplified when combined with the assumption of a stagnant upper head which serves to maintain the primary system pressure due to flashing of the hot fluid in the upper head.

The increase in energy removal through the secondary system results in a severe overcooling of the primary system. In the presence of a negative MTC, this cooldown results in a large decrease in the shutdown margin and a return to power. This return to power is exacerbated because of the high power peaking factors which exist, with the most reactive control rod stuck in its full withdrawn position.

As outlined in Reference 14.1-1 , three computerized calculations are required prior to the final calculation of the Minimum Departure From Nucleate Boiling Ratio (MDNBR) values and the maximum Linear Heat Generation Rate (LHGR) values utilized in the determination of fuel failure. The NSSS response is computed using the Siemens Power Corporation (SPC) ANF-RELAP code (Reference 14.1-2), the detailed core and hot assembly power distributions and the reactivity at the time of peak post-scram power are calculated using the SPC XTGPWR code (Reference 14.1-3), and the detailed core and hot assembly flow and enthalpy distributions are calculated using the SPC XCOBRA-IIIC code (Reference 14.1-4). The modified Barnett correlation was utilized to calculate MDNBR due to the reduced pressures occurring during the SLB event.

14.1.5.2.5.1 Analysis of Results

The ANF-RELAP analysis provides the NSSS boundary conditions for the XTGPWR and the XCOBRA-IIIC calculations. This section presents a description of the treatment of factors which can have a significant impact on NSSS response and resultant MDNBR and LHGR values. The plant specific parameters used in this analysis are listed in Tables 14.1.5.2-3 to 14.1.5.2-5. Conservatisms are included in parameters or factors known to have significant effects on the NSSS performance and resulting MDNBR and LHGR values.

14.1.5.2.5.1.1 Break Location, Size, and Flow Model

The post-scram SLB event is initiated by a double ended guillotine break of a main steam line downstream of the integral steam generator flow restrictors and upstream of the MSIVs. The flow is choked at the integral steam generator flow restrictor, which has an area of 3.51 ft^2. On the steam generator side of the break, steam flows out of the break throughout the entire transient. On the MSIV side of the break, break flow terminates after the MSIVs are fully closed. As an added conservatism, the main steam check valves are not credited in the analysis. The event occurs concurrent with the most reactive control rod stuck out of the core. The break flow areas for the affected and intact steam generators are listed in Table 14.1.5.2-3. These areas correspond to the locations in the flow path where choked flow will occur.

The ANF-RELAP break mass flow rate is computed using the Moody critical flow model modified such that only steam flows out the break.

14.1.5.2.5.1.2 Boron Injection

Boron injection into the primary system acts to mitigate the return to power. Injection of boron is modeled from the HPSI system. The HPSI system is conservatively modeled to take suction from the Refueling Water Storage Tank (RWST) at 35°F with a boron concentration of 1720 ppm. Initially, the line volume between the check valves isolating the system pumps and the cold leg injection location is assumed to be filled with unborated water. The time required to flush this unborated water from the safety injection lines is included as an integral part of the ANF-RELAP NSSS calculation. The characteristics of the HPSI system are listed in Table 14.1.5.2-3. The delivery curve for the HPSI system used in this analysis is given in Figure 14.1.5.2-1.

14.1.5.2.5.1.3 Single Failure Assumption

The single failure assumed in the engineered safeguards system results in the disabling of one of the two HPSI pumps required to be in service during normal operation. In addition to the single failure, there is no credit taken for the charging pump system. This assumption results in an additional delay in the time required for boron to reach the reactor core. The delay is further amplified when combined with the assumption of a stagnant upper head which serves to maintain the primary system pressure due to flashing of the hot fluid in the upper head.

14.1.5.2.5.1.4 Feedwater

For the HFP scenarios, normal MFW flow is assumed to be delivered to both SGs. The MFW flow increases as the secondary pressure decreases at the lowest possible fluid temperature until the feedwater regulating valve closes. Fluid temperature is determined by assuming heating of the feedwater ceases at the same time the break is initiated. The MFW flow is terminated 14 seconds after receiving the isolation signal.

For the HFP scenarios, the AFW flow is assumed to be zero at break initiation. After 180 seconds, AFW is delivered at the maximum capacity of the AFW system with flow restrictors installed on the AFW delivery lines. For the HZP scenarios, the AFW flow is increased to the maximum capacity immediately at break initiation. For all scenarios, all of the AFW flow is directed to the affected steam generator to maximize the cooldown rate. The operator is assumed to terminate the AFW flow to the affected steam generator at 600 seconds.

14.1.5.2.5.1.5 Trips and Delays

Trips for the HPSI, main feedwater valves, and MSIVs are given in Table 14.1.5.2-4. Biases to account for uncertainties are included in the trip setpoints as shown. For the steam and feedwater valves, the delay times given are between the time the trip setpoint is reached and the time full valve closure is reached. For the HPSI system, the delay time given is from the time the setpoint is reached until the pumps have accelerated to rated speed. Additional delay time required to sweep the lines of unborated water is accounted for by setting the boron concentration of the injected flow to zero until the volume of the injection lines has been cleared.

14.1.5.2.5.1.6 Neutronics

The core kinetics input for this calculation consisted of the minimum required control rod shutdown worth at the EOC, and EOC values associated with the reactivity feedback curves, delayed neutron fraction, delayed neutron fraction distribution and related time constants, and prompt neutron generation time. The ANF-RELAP default fission product and actinide decay constants were utilized for this calculation.

The core reactivity is derived from input of several functions. These include effects from control rod worth, moderator density changes, boron concentration, and Doppler effects. The reactivity is weighted between the core sectors. Different reactivity functions were utilized where necessary for the HZP and the HFP cases. The ANF-RELAP analyses were performed with an MTC of -28 pcm/°F. A summary of the nuclear input and assumptions is given in Table 14.1.5.2-5.

14.1.5.2.5.1.7 Decay Heat

The presence of radioisotope decay heat at the initiation of the SLB event will reduce the rate and the extent of cooldown of the primary system. For the HFP scenarios, the initial decay heat is calculated on the basis of infinite irradiation time at a power of 2700 MW prior to transient initiation. For the HZP scenarios, the initial decay heat is calculated on the basis of infinite irradiation time at a power of 1 W prior to transient initiation. For both scenarios, decay heat generated from return to power is calculated. This treatment of decay heat serves to maximize the stored energy in the HFP cases and to minimize it in the HZP cases. This treatment provides limiting stored energy conditions for the SLB cases.

14.1.5.2.5.1.8 Nodalization

The NSSS transient calculations utilized the nodalization model described in Reference 14.1-1. The nodalization treats all major NSSS components and subcomponents as discrete elements, with the exception of the secondary side of the steam generators. In addition, all components with long axial dimensions are divided into subcells adequate to minimize numerical diffusion and smearing of gradients.

In order to simulate the asymmetric thermal hydraulic and reactivity feedback effects that occur during an SLB transient, the core is nodalized into three radial sectors. One sector corresponds to the region immediately surrounding the assembly where the most reactive control rod is assumed stuck out of the core. This sector is termed the 'stuck rod' sector. The remainder of the region of the core which is directly affected by the loop containing the break is the second sector and is termed the 'affected' sector. The remainder of the core and the other loop is termed either the 'unaffected' or the 'intact' sector or loop.

14.1.5.2.5.1.9 Interloop Mixing

During an actual SLB transient, some mixing between the parallel channels within the reactor pressure vessel will occur in the downcomer, the lower plenum, the core, and the upper plenum due to lateral momentum imbalances, and turbulence or eddy mixing. The mixing will act to reduce the positive reactivity feedback effects due to a reduced rate and magnitude of cooldown of the affected loop and associated core sector.

In this analysis, no credit is taken for turbulent or eddy mixing of coolant between loops or the parallel flow channels within the reactor pressure vessel (RPV). However, interloop mixing is calculated to occur due to flow in interloop junctions in the upper and lower plenums. Mixing in the lower plenum was reduced to a minimum by using an extremely high loss coefficient between the affected and intact sectors.

14.1.5.2.5.1.10 Harsh Containment Conditions

Harsh containment conditions can be caused by the release of steam within the reactor containment. Under such conditions, only those trips which have been qualified for harsh environments are credited, and increased uncertainties are included in the setpoints of all environmentally qualified trips which are credited.

14.1.5.2.5.2 Minimum Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Analysis

MDNBR calculations require determination of the power, enthalpy, and flow distributions within the highest power assembly of the stuck rod core sector. Similarly, determination of the maximum LHGR also requires characterization of the power distribution. The power distribution within the core, including the highest powered assembly within the stuck rod core sector, is calculated with XTGPWR (Reference 14.1-3). Flow and enthalpy distributions within the core, including the highest powered assembly within the stuck rod core sector, are calculated with XCOBRA-IIIC (Reference 14.1-4). In order to obtain compatible flows, moderator densities, and powers within the high power assemblies, iteration between XTGPWR and XCOBRA-IIIC is conducted.

For this calculation, the modified Barnett correlation was found to be suitable for the MDNBR calculation. The modified Barnett correlation is based upon closed channels and primarily uniform power distribution data. The correlation is based on assembly inlet (or upstream) fluid conditions rather than on local fluid conditions as is the case with subchannel based correlations. Use of the correlation is limited to the range of the data base unless conservative extrapolations can be made.

14.1.5.2.6 Analysis Results

A summary of calculated results important to this analysis is presented in Table 14.1.5.2-6 for the limiting MDNBR and LHGR scenarios. The MDNBR values are listed together with the corresponding core power values at the time of MDNBR which corresponds to the maximum post-scram power level. The outside containment cases, regardless of whether or not offsite power was or was not available, were found to be the most limiting. For cases where offsite power was available for operation of the primary coolant system pumps, the MDNBR and the maximum LHGR occurred at the time of the maximum power condition. For cases where offsite power is lost and the primary system pumps coast down, the maximum LHGR and the MDNBR occur when the worst combination of core power, flow, inlet temperature, and pressure are present. These conditions occurred at the time of peak power in this analysis.

The scenario which resulted in the highest post-scram power level and the largest LHGR is that initiated from HFP with the break occurring outside containment and with offsite power available for operation of the primary coolant pumps. This case is presented in detail.

The NSSS responses for the scenarios with loss of offsite power for operation of the primary system coolant pumps are different from those scenarios where offsite power is available throughout the transient due to the pump coastdown and subsequent natural circulation of the primary coolant. Post-scram maximum power levels attained during the transient are significantly lower. Lower power levels result from lower positive moderator feedback. The positive moderator feedback is reduced due to the coolant density reductions that occur axially upwards in the core at low core flow rates, even for low core power levels. Lower power levels cause MDNBR values to increase, but lowering flow rates cause MDNBR values to decrease. Overall, the combination of factors results in lower MDNBR values for the reduced flow condition than for the full flow condition.

Of the two loss of offsite power scenarios analyzed, the HFP break occurring outside containment case resulted in lower MDNBR values. The general response of the HFP and HZP cases with loss of offsite power is comparable. Because the two scenarios are quite similar in terms of their general response, only the limiting MDNBR case (i.e., HFP break outside containment and without offsite power) is presented in detail.

14.1.5.2.6.1 Hot Full Power Outside Containment with Offsite Power Available

The ANF-RELAP simulation of the NSSS during the HFP transient with offsite power available is illustrated in Figures 14.1.5.2-2 through 14.1.5.2-9. A tabulation of the sequence of events is presented in Table 14.1.5.2-7. The ANF-RELAP computation was terminated 600 seconds after break initiation. This is well beyond the time of

MDNBR or peak LHGR. AFW termination of the AFW by manual operator action was assumed to occur 600 seconds after initiation of the break.

14.1.5.2.6.1.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Break flow for the steam generators is plotted in Figure 14.1.5.2-2. Secondary pressure for the steam generators is plotted in Figure 14.1.5.2-3. After break initiation, the pressure in the affected steam generator decreased immediately and then stabilized around 180 seconds. The mass inventory in both steam generators decreased throughout the transient. The relatively high reactor power level caused the affected steam generator to dry out by 490 seconds. The affected steam generator drying out caused the primary-to-secondary heat transfer to deteriorate. As a result, the primary system temperature rose, the secondary side pressure decreased, and, since the break flow is determined by the secondary system pressure, the break flow also declined. The heatup of the primary coolant reduced the reactivity present and power dropped rapidly.

The intact steam generator blows down for a short period until the MSIVs completely close approximately 17 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system and then slowly increases as the primary system begins to heat up.

14.1.5.2.6.1.2 Primary System Thermal Hydraulic Parameters

The primary system coolant temperature and pressure responses resulting from the break flow are illustrated in Figures 14.1.5.2-4 through 14.1.5.2-6. The primary system pressure decays rapidly as the coolant contracts due to cooldown and the pressurizer empties. The MSIVs close at 17 seconds, ending the blowdown of the intact steam generators and reducing the rate of energy removal from the primary fluid. The pressurizer emptied at approximately 60 seconds and system pressure (which increased slowly for the duration of the transient) was thereafter established by the saturation temperature of the primary coolant in the upper head of the reactor vessel.

14.1.5.2.6.1.3 Reactivity and Core Power

The reactivity transient calculated by ANF-RELAP is illustrated in Figure 14.1.5.2-8. Initially, the core is assumed to be at full power. All control rods, except the most reactive one, are assumed to be inserted into the core following the reactor trip signal. The reactivity transient then proceeds. The total core reactivity, initially at 0.00$, decreased instantly due to the scram worth at reactor trip, but then steadily increased due to moderator and Doppler feedback associated with the primary system cooldown. Shortly thereafter, power begins to rise steadily due to the dominating positive reactivity feedback from the moderator. The reactor soon achieves a quasi-steady-state power level where the Doppler and the moderator reactivities balance the scram reactivity.

Fifty-five seconds after break initiation, the RCS pressure dropped below the shutoff head of the HPSI system and HPSI flow to the RCS began. But, the elevated primary pressure limited the delivery of boron into the core due to the pressure versus flow characteristics of the HPSI system and unborated water never cleared the safety injection lines during the transient.

Figure 14.1.5.2-9 shows the transient reactor power. The reactor power initially declined due to insertion of the control rods. The severe cooldown resulted in power increasing after 52 seconds. A quasi steady-state reactor power level was established by 260 seconds and a maximum power level of 378 MW or 14% of rated power occurred at 462 seconds.

14.1.5.2.6.1.4 XTGPWR and XCOBRA-IIIC Results

The XTGPWR calculation is made initially on the basis of ANF-RELAP input. Each assembly within the three channels is assumed to have a uniform flow corresponding to the sector flows calculated with ANF-RELAP. Due to high power peaking in the region of the stuck control rod, large moderator density reductions are calculated to occur in the top portions of several assemblies in this region of the core in the XTGPWR calculation. This moderator density decrease is a major factor in the flattening of the axial and radial profiles, and the significant reduction in reactivity observed when XTGPWR is compared to ANF-RELAP. An XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly.

The ANF-RELAP reactivity and power calculation has considerable inherent conservatism. To demonstrate this, a comparison of the change in reactivity at the maximum LHGR time is made. A comparison of the overall change in reactivity between ANF-RELAP and XTGPWR shows that ANF-RELAP conservatively underestimates the negative reactivity by 1.01$ at the time of maximum LHGR thus indicating that the ANF-RELAP power calculation is conservative.

14.1.5.2.6.1.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

For the MDNBR portion of the calculation, the radial power distribution was modified to conservatively account for local rod power distribution affects within the hot assembly. This was done by raising the power of the hot assembly to bound the peak rod power.

On the bases of these conservative assumptions, the MDNBR value was calculated to be 2.28. This compares to a 95/95 DNBR limit of 1.135 for the modified Barnett correlation.

Therefore, no fuel rods would be expected to fail during this transient scenario from an MDNBR stand point.

The analysis of the peak LHGR also comes from the XTGPWR and XCOBRA-IIIC analysis. The peak LHGR is calculated from the ANF-RELAP total core power and the XTGPWR radial and axial peaking. The peak LHGR, 24.27 kW/ft, was calculated for the HFP outside containment break with offsite power available event. When compared to a centerline melt criteria of 21.0 kW/ft, four assembly quadrants (one full assembly) or 0.46% of the core, are predicted to fail due to violation of the centerline melt criteria.

14.1.5.2.6.2 Hot Full Power Outside Containment with Loss of Offsite Power

The ANF-RELAP NSSS simulation of the most limiting SLB scenario from an MDNBR standpoint (i.e., HFP outside containment break with a loss of offsite power) is illustrated in Figures 14.1.5.2-10 through 14.1.5.2-16. A tabulation of the sequence of events is presented in Table 14.1.5.2-8. Termination of the AFW by manual operator action was assumed to occur 600 seconds after initiation of the break. This is well beyond the time of MDNBR and maximum LHGR. Termination of AFW would cause the affected SG to dry out and an increase in the primary system temperature. The increase in primary temperature, will drive the reactor subcritical and restore shutdown.

14.1.5.2.6.2.1 Secondary System Thermal Hydraulic Parameters

Steam flow out the break is the source of the NSSS cooldown. Steam flow for the affected steam generator is plotted in Figure 14.1.5.2-10. Secondary pressure for the steam generators is plotted in Figure 14.1.5.2-11. The affected steam generator blows down through the break throughout the transient. The pressure and mass flow rate dropped rapidly at first and then proceeded downward at a slower decay rate until natural circulation flow was established by approximately 250 seconds.

The intact steam generators blow down for a short period until the MSIVs completely close approximately 16 seconds after the break is initiated. The pressure recovers as the intact steam generator equilibrates with the primary system. Subsequently, the intact steam generator pressure remains essentially constant as the primary intact coolant loop approaches natural circulation conditions.

14.1.5.2.6.2.2 Primary System Thermal Hydraulic Parameters

The primary system core coolant temperature and pressure responses resulting from the break flow are illustrated in Figures 14.1.5.2-12 through 14.1.5.2-14. The primary system pressure decays rapidly as the coolant contracts due to the cooldown and the pressurizer empties. Continued pressure reduction in the primary system causes the relatively hot stagnant liquid in the head of the RPV vessel to flash. The flashing in the upper head, coupled with near equilibration of other NSSS parameters, retards the pressure decay from that point forward.

A comparison of intact and affected core sector inlet temperatures throughout the transient indicates significant differences due to the limited cross flow allowed between loops. The core sector flows all show the same trend due to the coastdown of the primary coolant pumps. That is, all flows decrease rapidly until natural circulation conditions are achieved in the two flow loops.

14.1.5.2.6.2.3 Reactivity and Core Power

The reactivity transient calculated by ANF-RELAP is illustrated in Figure 14.1.5.2-15. Initially, the core is assumed to be at full power. All control rods, except the most reactive one, are assumed to be inserted into the core following the reactor trip signal. The reactivity transient then proceeds. The total core reactivity, initially at 0.00$, decreased instantly due to the scram worth at reactor trip, but then steadily increased due to moderator and Doppler feedback associated with the primary system cooldown. Shortly thereafter, power begins to rise steadily due to the dominating positive reactivity feedback from the moderator. The reactor soon achieves a quasi-steady-state power level where the Doppler and the moderator reactivities balance the scram reactivity.

Ninety seconds after break initiation, the RCS pressure dropped below the shutoff head of the HPSI system and HPSI flow to the RCS began. But, the elevated primary pressure limited the delivery of boron into the core due to the pressure versus flow characteristics of the HPSI system and unborated water never cleared the safety injection lines during the transient.

The transient experienced by the core power is illustrated in Figure 14.1.5.2-16. The reactor power declined to a decay heat level during the first 150 seconds of the transient. The maximum peak power level of 207 MW or 7.7% of rated power occurred at 488 seconds.

14.1.5.2.6.2.4 XTGPWR and XCOBRA-IIIC Results

The XTGPWR calculation is initially made on the basis of ANF-RELAP predicted core power, flow, pressure, and inlet temperatures. The XTGPWR calculations provide the radial and axial power distributions for use in the XCOBRA-IIIC code. Due to the high power peaking in the region of the stuck control rod, and the low core average natural circulation flow rates, large moderator density decreases are calculated in several assemblies in this region in the XTGPWR calculation. This is a major factor in the flattening of the axial and radial profiles, and the significant reduction in reactivity observed when XTGPWR is compared to ANF-RELAP. An XCOBRA-IIIC analysis is also conducted to define the flow and enthalpy distribution within the high power assembly.

A comparison of the overall change in reactivity between ANF-RELAP and XTGPWR shows that ANF-RELAP conservatively underestimates the negative reactivity by 1.00$ at the time of MDNBR thus indicating that the ANF-RELAP power calculation is conservative.

14.1.5.2.6.2.5 Departure From Nucleate Boiling Ratio and Linear Heat Generation Rate Results

The MDNBR of the hot fuel assembly is calculated to be 1.71 which is above the modified Barnett 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

As before, the analysis of the peak LHGR comes from the XTGPWR and the XCOBRA-IIIC analysis. The peak LHGR was 17.96 kW/ft, Comparing this LHGR with a centerline melt criteria of 21 kW/ft, it is apparent that centerline melt is not predicted to occur. Thus, no fuel failures are predicted to occur due to violation of the centerline melt criteria.

14.1.5.2.7 Conclusions

The HFP and HZP scenarios, with offsite power maintained for operation of the primary coolant pumps resulted in a return to higher power levels than the scenarios where offsite power is lost. However, these scenarios provide substantially greater margin to the MDNBR limit because of the higher coolant flow rate. In no scenario evaluated, however, was fuel failure calculated to occur as a result of penetration of the MDNBR safety limit. The HFP and HZP scenarios with offsite power maintained for operation of the primary coolant pumps returned to higher power levels than the scenarios where offsite power is lost. Even though these scenarios have substantially greater margin to the MDNBR limit because of a higher coolant flow rate, the higher power levels in combination with the highly skewed power distribution due to the assumed stuck rod cluster resulted in them having the least margin to the fuel centerline melt limit.

The HFP outside containment break scenario concurrent with a loss of offsite power was determined to be the most limiting in this analysis from an MDNBR standpoint. The MDNBR of the hot fuel assembly is calculated to be 1.71 which is above the modified Barnett 95/95 DNBR correlation limit. Therefore, no fuel rods are expected to fail from an MDNBR standpoint.

The HFP outside containment break scenario with offsite power available was determined to be the most limiting in this analysis from the standpoint of centerline melt. This scenario results in the highest return to power and highest calculated LHGR of 24.27 kW/ft. When compared to a centerline melt criteria of 21.0 kW/ft, four assembly quadrants (one full assembly) or 0.46% of the core, are predicted to fail due to violation of the centerline melt criteria.

14.1.5.3 Radiological Consequences of a Main Steam Line Break

The main steam line break is postulated to occur in a main steam line outside the containment. The radiological consequences of a main steam line break inside containment is bounded by the main steam line break outside containment. The plant is assumed to be operating with Technical Specification coolant concentrations and primary to secondary leakage. A 0.035 gpm primary to secondary leak is assumed to occur in both steam generators.

Two separate main steam line break cases are analyzed. In the first case, associated with this accident is that 1 fuel assembly experiences melting and releases the melted fuel into the RCS at the onset of the accident. One fuel assembly is equivalent to 0.46% melt. The activity associated with the melt condition is therefore available for release to the atmosphere via primary to secondary leakage. In the second case a pre-accident iodine spike is assumed to occur. In this case the primary coolant iodine concentrations are 60 times the plant technical specification activity level of 1 uCi/gm DE I-131. In addition, the noble gas activity in the primary coolant is assumed to be at technical specification levels.

The noble gases and iodines in the primary coolant that leak into the faulted steam generator during the transient are released directly to the environment without holdup or decontamination. An iodine partition factor of 0.01 is used for the releases from the unaffected steam generator. Off-site power is assumed to be lost, thus making the condenser unavailable. The steam releases from the main steam line break are from the turbine building blowout panels as the atmospheric dispersion factor is greater for this release point than the enclosure building blowout panels. The steam releases from the intact steam generator are from the MSSVs/ADVs.

The radiological consequences of a main steam line break to the EAB, LPZ and Millstone 2 Control Room are reported in Tables 14.1.5.3-2 and 14.1.5.3-3. The assumptions used to perform this evaluation are summarized in Table 14.1.5.3-1.

The resulting doses to the EAB and LPZ do not exceed the limits specified in 10CFR100. The resulting doses to the Control Room do not exceed the limits specified in GDC19.

REFERENCES

14.1-1 "Steam Line Break Methodology for PWRS," EMF-84-093(P), Revision 1, Siemens Power Corporation, Richland, WA, June 1998.

14.1-2 "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," ANF-89-151(P)(A), Advanced Nuclear Fuels Corporation, May 1992.

14.1-3 "XTG-- A Two-Group Three Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing (PWR Version)," XN-CC-28, Volume 1, Revision 5, Exxon Nuclear Company, Richland, WA 99352, July 1979.

14.1-4 "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, January 1986.

TABLE ~~14.1.5-1~~ 14.1.5.1-1

AVAILABLE REACTOR PROTECTION FOR STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT ~~EVENT~~

PRE-SCRAM ANALYSIS

Reactor Operating Conditions	Reactor Protection
1	Low Steam Generator Pressure Trip Low Steam Generator Water Level Trip Low Reactor Coolant Flow 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 Low Reactor Coolant Flow Variable Overpower Trip High Containment Pressure Trip Safety Injection Actuation Signal
3-6	Technical Specification Requirements on Shutdown Margin, Inherent Negative Doppler Feedback

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TABLE ~~14.1.5-2~~ 14.1.5.1-2

DISPOSITION OF EVENTS FOR STEAM SYSTEM
PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT ~~EVENT~~
PRE-SCRAM ANALYSIS

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

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TABLE 14.1.5-3

ADVANCED NUCLEAR FUELS RELAP THERMAL-HYDRAULIC INPUT
(STEAM LINE BREAK)

Iinitial Condition Thermal-Hydraulic Input	HZP	HFP
Total Core Power (watt)	1	$2700 \cdot 10^6$
Primary Pressure (psia)	2250	2250
Core Inlet Temperature (°F)	532	549
Primary Flow Rate (gpm)	401,000	398,000
Pressurizer Level (% of span)	40.0	65.0
Secondary Pressure (psia)	891	867
Secondary Temperature (°F)	531	527
Steam Flow Rate (lb/s) - per Steam Generator	7	1631
Feedwater Flow Rate (lb/s) - per Steam Generator	7	1631
Feedwater Enthalpy (Btu/lb)	0.1	410.7
Secondary Fluid Mass (lb)	223,000	143,000
Break Characteristics		
Minimum Flow Area		
Affected Steam Generator	6.31 ft^2	
Intact Steam Generator	2.35 ft^2	
Location of Pipe Break	Upstream of Affected Steamline Flow Restrictor	
Injection Systems		
Total HPSI Pumps (2 normal, 1 mounted spare)	3	3
Active HPSI Pumps	2	2
Single Failure (No credit for mounted spare)	1 HPSI pump	1 HPSI pump
Active Charging Pumps	1	1

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TABLE 14.1.5-3

ADVANCED NUCLEAR FUELS RELAP THERMAL-HYDRAULIC INPUT (STEAM LINE BREAK)

	HZP	HFP
Refueling Water Storage Tank Boron Concentration (ppm)	1720	1720
HPSI Delivery Curve	Fig. 14.1.5-1	Fig. 14.1.5-1
Feedwater		
Auxiliary		
Flow, maximum (lbm/sec)	229.5	229.5
Temperature (°F)	32.1	32.1
Main		
Initial Flow/Steam Generator (lbm/sec)	0.0	1631.1
Initial Temperature (°F)	N/A	432.1

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TABLE 14.1.5-4

ACTUATION SIGNALS AND DELAYS
(STEAM LINE BREAK)

Parameter Setpoints	Analysis Setpoint	Uncertainty	Value
1. Low Steam Line Pressure	500 psia	-22 psi	478 psia
2. Low Pressurizer Pressure	1600 psia	-22 psi	1578 psia

MSIV Closure

Required Actuation Signal

A. (1) Above

Delay - 6.9 seconds

HPSI Actuation

Required Actuation Signal

A. (2) Above

Delay - 30 seconds

Main Feedwater Valve Closure

Required Actuation Signal

A. (1) Above

Delay - 30 seconds

Reactor Scram

Required Actuation Signal

A. (1) or (2) Above

Delay - .9 second instrument delay; 3.0 second insertion time.

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TABLE 14.1.5-4

ACTUATION SIGNALS AND DELAYS
(STEAM LINE BREAK)

Charging Pump Actuation

Required Actuation Signal

A. (2) Above

Delay - 40 seconds

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TABLE 14.1.5-5

ADVANCED NUCLEAR FUELS-RELAP NUCLEAR INPUT AND ASSUMPTION (STEAM LINE BREAK)

Point Kinetics Input	Value
Effective Delayed Neutron Fraction	.0049
Effective Neutron Lifetime (micro sec)	22.0
Minimum Shutdown Reactivity Requirement	3.6% delta rho

Stuck Rod Location

Within half core section cooled by the affected loop

Fission Product and Actinide Decay Constants

Default values in ANF-RELAP utilized

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TABLE 14.1.5-6

STEAM LINE BREAK ANALYSIS SUMMARY

Initial Power Level	Offsite Power Available	Maximum Post Scram Return to Power (MWt)	MDNBR	Maximum LHGR (kW/ft)
HZP	Yes	686	2.40	< 21.0
HZP	No	294	1.18	16.5
HFP	Yes	394	3.00	17.1
HFP	No	147	4.60	5.7

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TABLE 14.1.5-7

STEAM LINE BREAK SEQUENCE OF EVENTS--HOT ZERO POWER--POWER AVAILABLE

Time*	Event
0.	Reactor at hot zero power.
0.+	Double-ended guillotine break located between affected steam generator and the flow restrictors.
3.6	Main steam isolation valve closure signal generated by low steam generator pressure.
10.5	Main steam line isolation valves stop blowdown from intact steam generator 6.9 seconds after low steam generator pressure signal.
15.2	Safety injection signal generated by low primary coolant pressure.
32.	Reactor becomes critical.
45.2	HPSI and charging pumps actuated.
153.	Thermal power reaches maximum level at 25% of rated power.
153.	First boron has passed through core.
180.	Auxiliary feedwater initiated to affected steam generator.
600.	Auxiliary feedwater isolated manually.
600.+	Primary system temperature increase due to steam generator dryout and additional boron injection will terminate power excursion.

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*Time after break, seconds

TABLE 14.1.5-8

STEAM LINE BREAK SEQUENCE OF EVENTS--
HOT ZERO POWER--WITHOUT OFFSITE POWER

Time*	Event
0.	Reactor at hot zero power.
0.+	Double-ended guillotine break located between affected steam generator and the flow restrictor.
3.6	Main steam isolation signal generated by low steam generator pressure.
10.5	Main steam line isolation valves stop blowdown from intact steam generator 6.9 seconds after low steam generator pressure signal.
18.7	Safety injection signal generated by low primary coolant pressure.
48.7	HPSI and charging pumps actuated.
50.	Reactor becomes critical.
152.	First boron has passed through core.
169.	Thermal power reaches maximum level at 11% of rated power.
180.	Auxiliary feedwater initiated to affected steam generator.
600.	Auxiliary feedwater isolated manually.
600.+	Primary system temperature increase due to steam generator dryout and additional boron injection will terminate power excursion.

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*Time after break, seconds

TABLE 14.1.5-9

STEAM LINE BREAK SEQUENCE OF EVENTS--HOT FULL POWER--POWER AVAILABLE

Time*	Event
0.	Reactor at hot full power.
0.+	Double-ended guillotine break located between affected steam generator and the flow restrictor.
3.5	Reactor trip and main steam isolation valve closure signal generated by low steam generator pressure.
10.4	Main steam line isolation valves stop blowdown from intact steam generator 6.9 seconds after low steam generator pressure signal.
13.8	Safety injection signal generated by low primary coolant pressure.
43.8	HPSI and charging pumps actuated.
174.	Reactor becomes critical.
180.	Auxiliary feedwater initiated to affected steam generator.
204.	Thermal power reaches maximum level at 15% of rated power.
204.	First boron has passed through core.
600.	Auxiliary feedwater isolated manually.
600.+	Primary system temperature increase due to steam generator dryout and additional boron injection will terminate power excursion.

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*Time after break, seconds

TABLE 14.1.5-10

STEAM LINE BREAK EVENT SEQUENCE - HOT FULL POWER - WITHOUT OFFSITE POWER

Time*	Event
0.	Reactor at hot full power.
0.+	Double-ended guillotine break located between affected steam generator and the flow restrictor.
3.6	Reactor trip and main steam isolation signal generated by low steam generator pressure.
10.5	Main steam line isolation valves stop blowdown from intact steam generator 6.9 seconds after low steam generator pressure signal. 5/90
16.0	Safety injection signal generated by low primary coolant pressure.
46.0	HPSI and charging pumps actuated.
180.	Auxiliary feedwater initiated to affected steam generator.
224.	First boron has passed through core.
235.	Reactor becomes critical.
250.	Thermal power reaches maximum level at 5.4% of rated power.
600.	Auxiliary feedwater isolated manually.
600.+	Primary system temperature increase due to steam generator dryout and additional boron injection will terminate power excursion.

*Time after break, seconds

Table 14.1.5.1-3

ANF-RELAP Thermal-Hydraulic Input (Pre-Scram Steam Line Break)

Initial Condition Thermal-Hydraulic Input	HFP
Reactor Power (MW)	2754
Pressurizer Pressure (psia)	2250
Pressurizer Level (%)	65
Cold Leg Coolant Temperature (°F)	549
Total Primary Flow Rate (lbm/sec)	37,640
Secondary Pressure (psia)	881
Core Bypass Flow Rate (lbm/sec) per Loop	753
Main Feedwater Temperature (°F)	432
Steam Generator Mass Inventory (lbm)	167,237

Table 14.1.5.1-4

Actuation Signals and Delays (Pre-Scram Steam Line Break)

Reactor Trip	Non-Harsh Containment Condition Setpoint	Harsh Containment Condition Setpoint	Delay
Variable Overpower (ceiling)	111.6% of rated	Not credited	0.9 s
Low Reactor Coolant Flow	Credited	85% flow	0.65 s
High Containment Pressure	Not applicable	5.83 psig	0.9 s
Low Steam Generator Pressure	658	550	0.9 s
TM/LP (floor)	1728 psia	1700 psia	0.9 s
TM/LP (function)	Evaluated from function given in Technical Specification	Not credited	0.9 s

Table 14.1.5.1-5

ANF-RELAP Neutronics Input and Assumptions (Pre-Scram Steam Line Break)

Point Kinetics Input	Value
Effective Delayed Neutron Fraction	0.0054
Moderator Temperature Coefficient (pcm/°F)	
Offsite Power Available (Technical Specification most negative limit)	-28
Loss of Offsite Power (Technical Specification most positive limit above 70 % RTP)	+4
HFP Scram Worth (pcm)	6628
Shutdown Margin Requirement (pcm)	3600
Doppler Coefficient	
Offsite Power Available	1.20 x most-negative value at EOC
Loss of Offsite Power	0.80 x least-negative value at BOC

Fission Product and Actinide Decay Constants

Default values in ANF-RELAP utilized

Table 14.1.5.1-6

MDNBR and Peak Reactor Power Level Summary (Pre-Scram Steam Line Break)

Location of Break	Type of Cooldown	Size of Break	MDNBR	Peak Reactor Power (% of rated)
Outside containment, downstream of check valves	Symmetric	2.40 ft^2	1.332	126.90%
		3.00 ft^2	1.310	130.01%
		3.50 ft^2	1.298	**130.91%**[1]
Outside containment, upstream of check valve	Asymmetric	1.20 ft^2	1.254	124.42%
		1.40 ft^2	1.244	126.06%
		1.60 ft^2	1.302	124.87%
		1.80 ft^2	1.334	124.92%
Inside containment, upstream of check valve	Asymmetric	0.40 ft^2	1.299	117.85%
		0.60 ft^2	1.258	121.53%
		0.80 ft^2	1.262	122.26%
		1.80 ft^2	1.318	125.51%
Inside containment, upstream of check valve with loss of offsite power	Asymmetric	**3.51 ft^2**	**0.[illegible]8**[2]	106.86%

1 The peak LHRs for all pre-scram breaks are bounded by the peak LHR for the 3.50 ft^2 break outside containment and downstream of a check valve.

2 The MDNBRs for all pre-scram breaks are bounded by the MDNBR for the 3.51 ft^2 break inside containment and upstream of a check valve with the loss of offsite power.

Table 14.1.5.1-7

LHGR-Limiting Pre-Scram Steam Line Break Sequence of Events: HFP 3.50 ft^2 Symmetric Break Outside Containment with Offsite Power Available

Time (sec)	Event
0	Break downstream of main steam line check valves opens
0	Turbine control valves open fully
7	Low steam generator pressure trip setpoint reached
8	Turbine trips on reactor scram signal
9	Scram CEA insertion begins
9	Reactor power reaches maximum value
10	MDNBR occurs

Table 14.1.5.1-8

MDNBR-Limiting Pre-Scram Steam Line Break Sequence of Events: HFP 3.51 ft^2 Asymmetric Break Inside Containment with Loss of Offsite Power

Time (sec)	Event
0	Break occurs
0	RCPs trip
0	Peak LHGR (kW/ft)
2	Scram signal on low flow trip
3	Scram CEA Insertion begins
3	Max Power (Fraction of RTP)
4	MDNBR

TABLE ~~14.1.5-1~~ 14.1.5.2-1

AVAILABLE REACTOR PROTECTION FOR STEAM SYSTEM
PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT ~~EVENT~~

POST-SCRAM ANALYSIS

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	Technical Specification Requirements on Shutdown Margin, Inherent Negative Doppler Feedback

5/90

TABLE ~~14.1.5-2~~ 14.1.5.2-2

DISPOSITION OF EVENTS FOR STEAM SYSTEM
PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT ~~EVENT~~

POST - SCRAM ANALYSIS

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

5/90

Table 14.1.5.2-3

ANF-RELAP Thermal-Hydraulic Input (Post-Scram Steam Line Break)

Initial Condition Thermal-Hydraulic Input	HFP	HZP
Core Power (MW)	2700	1E-6
Primary Pressure (psia)	2250	2250
Pressurizer Level (%)	65	40
Cold Leg Temperature (°F)	549	532
Primary Flow Rate per Loop (lbm/sec)	18,820	19,241
Secondary Pressure (psia)	880	892
Steam Generator Mass Inventory (lbm)	167,237	253,989
Total Steam Flow (lbm/sec) per Steam Generator	1634	4
Total MFW Flow (lbm/sec) per Steam Generator	1634	4
MFW Temperature (°F)	432	432
Total AF Flow (lbm/sec)	184	184
RWST Boron Concentration (ppm)	1720	1720
AF Temperature (°F)	32	32

Break Characteristics	
Minimum Flow Area	
Affected Steam Generator (ft^2)	3.51
Unaffected Steam Generator (ft^2)	3.51
Location of Pipe Break	Downstream of steam generator integral flow restrictor and upstream of MSIV

Table 14.1.5.2-3

ANF-RELAP Thermal-Hydraulic Input (Post-Scram Steam Line Break)

Injection Systems	HFP	HZP
Total HPSI Pumps	3	3
Active HPSI Pumps	2	2
Single Failure (No credit for mounted spare)	1 HPSI pump	1 HPSI pump
Active Charging Pumps	0	0
Refueling Water Storage Tank Boron Concentration (ppm)	1720	1720
HPSI Delivery Curve	Fig. 14.1.5.2-1	Fig. 14.1.5.2-1

Feedwater		
Auxiliary		
Flow, maximum (lbm/sec)	183.6	183.6
Temperature (°F)	32.1	32.1
Main		
Initial Flow per Steam Generator (lbm/sec)	1634.1	0.0
Initial Temperature (°F)	432.4	N/A

Table 14.1.5.2-4

Actuation Signals and Delays (Post-Scram Steam Line Break)

Parameter Setpoints	Inside Containment	Outside Containment
1. Low Steam Generator Pressure Trip	550 psia	658 psia
2. Low Pressurizer Pressure SIAS	1500 psia	1578 psia
3. Low Steam Generator Pressure MSI	370 psia	478 psia

MSIV Closure

Required Actuation Signal

(3) Above

Delay - 6.9 seconds

HPSI Actuation

Required Actuation Signal

(2) Above

Delay - 25.0 seconds

Main Feedwater Valve Closure

Required Actuation Signal

(3) Above

Delay - 14.0 seconds

Reactor Scram

Required Actuation Signal

(1) Above

Delay - 0.9 second instrument delay
3.0 second insertion time

Table 14.1.5.2-5

ANF-RELAP Neutronics Input and Assumptions (Post-Scram Steam Line Break)

Point Kinetics Input	Value
Effective Delayed Neutron Fraction	0.0054
Moderator Temperature Coefficient (pcm/°F)	-28.0
HFP Scram Worth (pcm)	6438.0
Shutdown Margin Requirement (pcm)	3600.0

Stuck Rod Location

Within half-core section cooled by affected loop

Fission Product and Actinide Decay Constants

Default values in ANF-RELAP utilized

Table 14.1.5.2-6

Post-Scram Steam Line Break Analysis Summary

Initial Power Level	Offsite Power Available	Break Location	Maximum Post-Scram Return to Power (MW)	MDNBR	Maximum LHGR (kW/ft)	Fuel Failure (% of Core)
HFP	No	outside containment	207.5	1.71	17.96	0.0
HFP	Yes	outside containment	378.0	2.28	24.27	0.5
HZP	No	outside containment	182.9	1.89	15.76	0.0
HZP	Yes	outside containment	343.5	2.37	23.47	0.3

Initial Power Level	Offsite Power Availability	Break Location	ANF-RELAP Reactivity Change ($)	XTGPWR Reactivity Change ($)	Conservatisms in Input Parameters (MTC, Doppler, and Scram Worth Bias) ($)	Net Conservatism in ANF-RELAP model ($)
HFP	No	outside containment	+0.00	-6.30	+5.30	+1.00
HFP	Yes	outside containment	+0.00	-5.87	+4.86	+1.01
HZP	No	outside containment	+6.69	+3.00	+2.72	+0.97
HZP	Yes	outside containment	+6.68	+3.43	+2.34	+0.91

Table 14.1.5.2-7

LHGR-Limiting Post-Scram Steam Line Break Sequence of Events: HFP Outside Containment Break with Offsite Power Available

Time (sec)	Event
0.	Reactor at HFP
0.+	Double ended guillotine break.
4	Low steam generator pressure trip, Reactor trip
11	MSIV and MFW valves closure trip signal
16	SI signal
17	MSIVs closed
25	MFW valves closed
41	SI pumps at rated speed (25 s delay)
180	AFW starts
462	Peak post-scram power reached (378.03 MW)
N/A	SI lines cleared. Boron begins to enter primary system
490	Steam generator dry out
600	Calculation terminated. Power decreasing.

Table 14.1.5.2-8

MDNBR-Limiting Post-Scram Steam Line Break Sequence of Events: HFP Outside Containment Break with Loss of Offsite Power

Time (sec)	Event
0.	Reactor at HFP
0.+	Double ended guillotine break. Loss of offsite power.
4	Low steam generator pressure trip, Reactor trip
9	MSIV and MFW valves closure trip signal
16	MSIVs closed
18	SI signal
23	MFW valves closed
43	SI pumps at rated speed (25 s delay)
180	AFW starts
488	Peak post-scram power reached (207.47 MW)
N/A	SI lines cleared. Boron begins to enter primary system
600	Calculation terminated. Power decreasing.

TABLE 14.1.5.3-1

ASSUMPTIONS USED IN MAIN STEAM LINE BREAK ANALYSIS

Core Power Level (MWt)	2754
Primary to Secondary Leak Rate per Steam Generator	0.035 gpm
Primary Coolant Iodine Concentration	1 uCi/gm DE I-131
Secondary Coolant Iodine Concentration	0.1 uCi/gm DE I-131
Primary Coolant Noble Gas Concentration	$100/E_{bar}$
Pre-accident Spike Iodine Concentration	60 uCi/gm DE I-131
Melted Fuel Percentage	0.46%
Peaking Factor	1.45
Reactor Coolant Mass	430,000 lbs
Intact Steam Generator Minimum Mass	100,000 lbs
Safety Injection Signal Response	85 seconds
Site Boundary Breathing Rate (m^3/sec)	
0 - 8 hr	3.47E-04
8 - 24 hr	1.75E-04
24 - 720 hr	2.32E-04
Site Boundary Dispersion Factors (sec/m3)	
EAB: 0 - 2 hr	3.66E-04
LPZ: 0 - 4 hr	4.80E-05
4 - 8 hr	2.31E-05
8 - 24 hr	1.60E-05
24 - 96 hr	7.25E-06
96 - 720 hr	2.32E-06
Control Room Breathing Rate	3.47E-04 m^3/sec
Control Room Damper Closure Time	5 seconds
Control Room Intake Prior To Isolation	800 cfm
Control Room Inleakage During Isolation	130 cfm
Control Room Emergency Filtered Recirculation Rate (t=10 min)	2,250 cfm
Control Room Intake Dispersion Factors (sec/m3)	
PORVs/ADVs: 0 - 8 hr	3.19E-03
8 - 24 hr	2.05E-03
24 - 96 hr	7.61E-04
96 - 720 hr	2.13E-04
Turbine Building Blowout Panels: 0 - 8 hr	4.23E-03
8 - 24 hr	2.85E-03
24 - 96 hr	1.12E-03
96 - 720 hr	3.63E-04
Control Room Free Volume	35,650 ft^3
Control Room Filter Efficiency (all iodines)	90%
Thyroid Dose Conversion Factors	ICRP 30

TABLE 14.1.5.3-2

SUMMARY OF MILLSTONE 2 MSLB ACCIDENT DOSES
(0.46% Melted Fuel)

Location	Thyroid (rem)	Whole Body (rem)	Beta (rem)
EAB	4.8	0.06	N/A
LPZ	2.3	0.02	N/A
Control Room	29	0.03	0.5

TABLE 14.1.5.3-3

SUMMARY OF MILLSTONE 2 MSLB ACCIDENT DOSES
(Pre-accident Iodine Spike)

Location	Thyroid (rem)	Whole Body (rem)	Beta (rem)
EAB	0.935	0.010	N/A
LPZ	0.176	0.002	N/A
Control Room	5.314	0.003	0.039

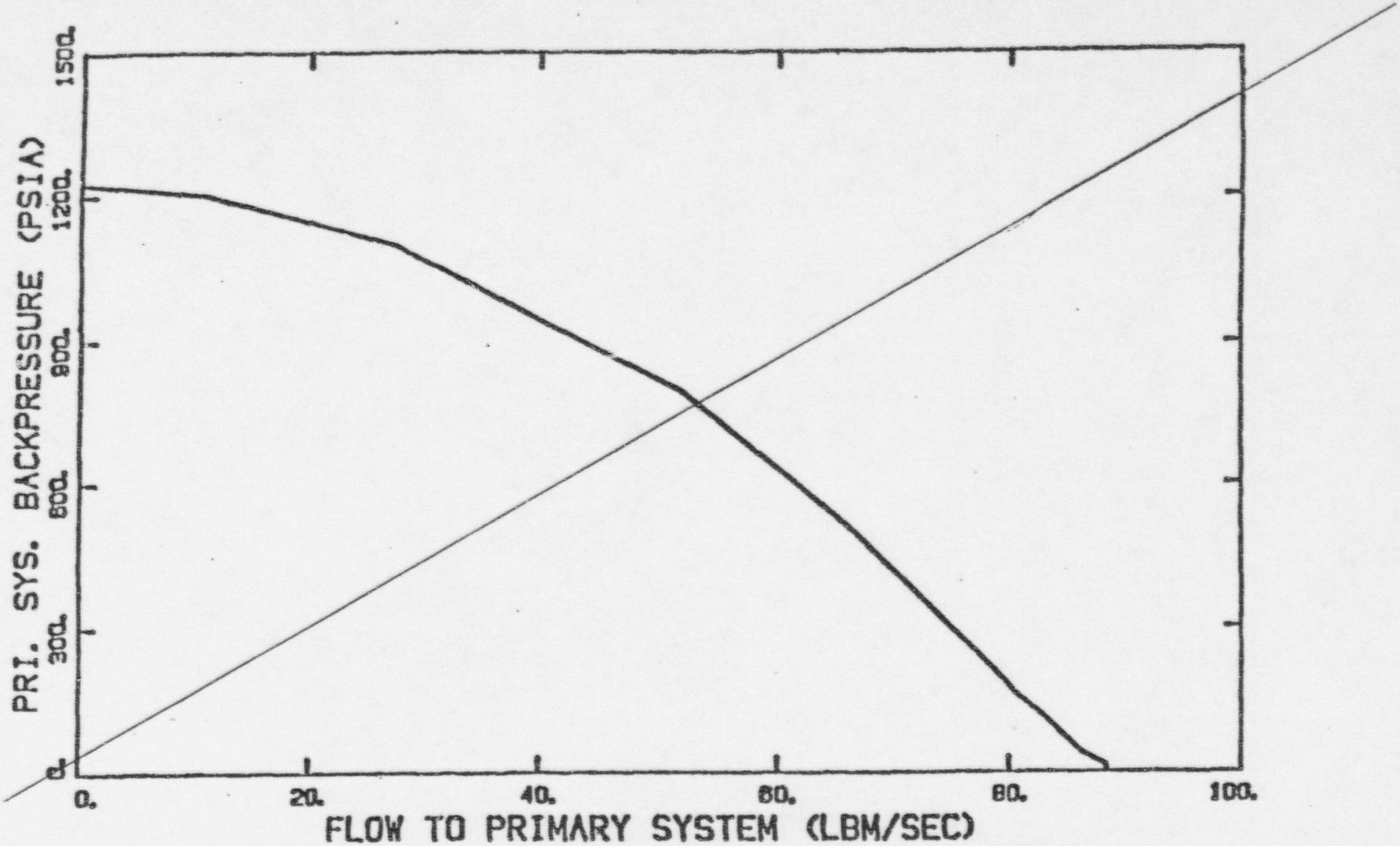

FIGURE 14.1.5-1 ONE PUMP HIGH PRESSURE SAFETY INJECTION SYSTEM DELIVERY VS. PRIMARY PRESSURE

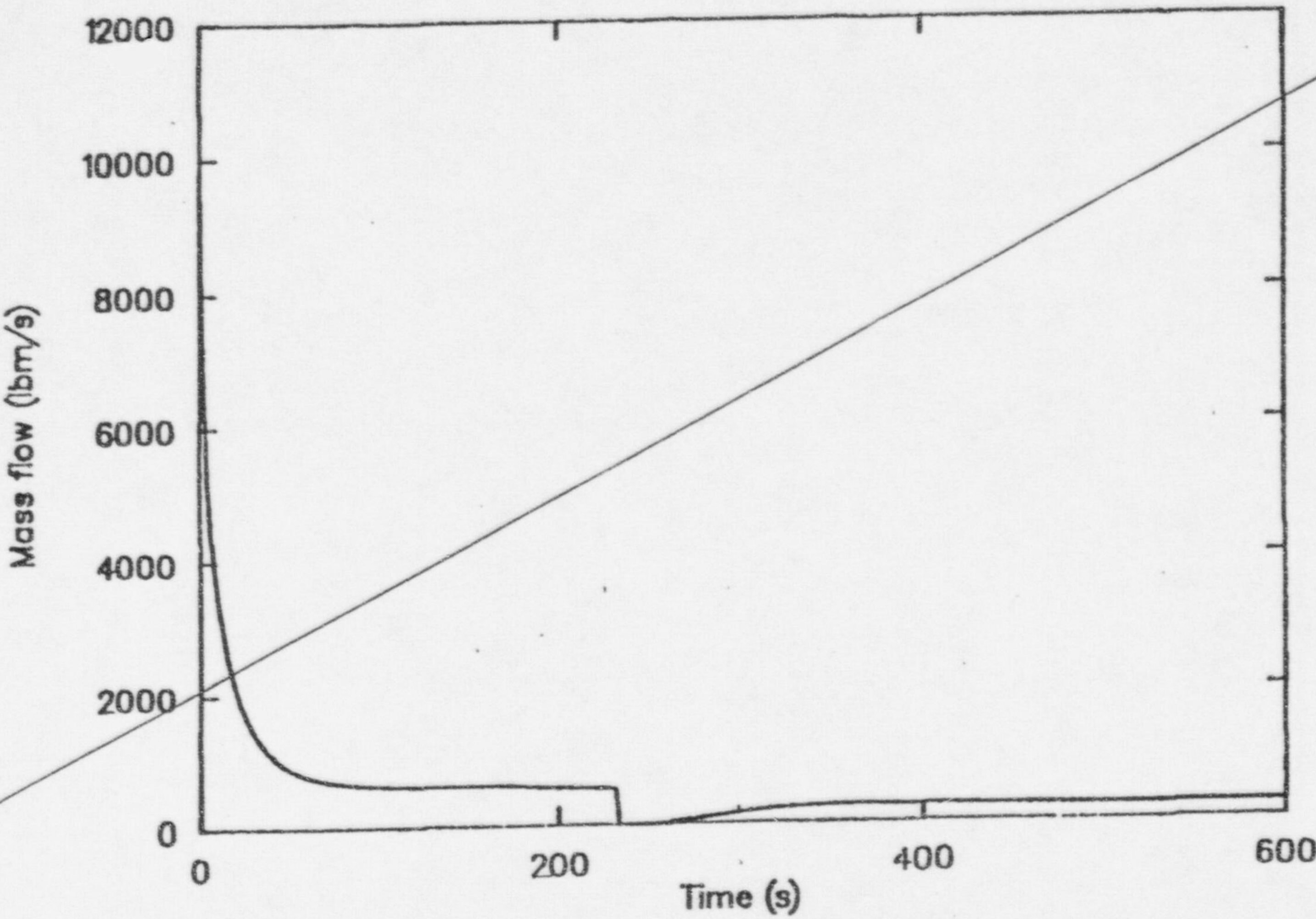

FIGURE 14.1.5-2 AFFECTED STEAM GENERATOR BREAK FLOW VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

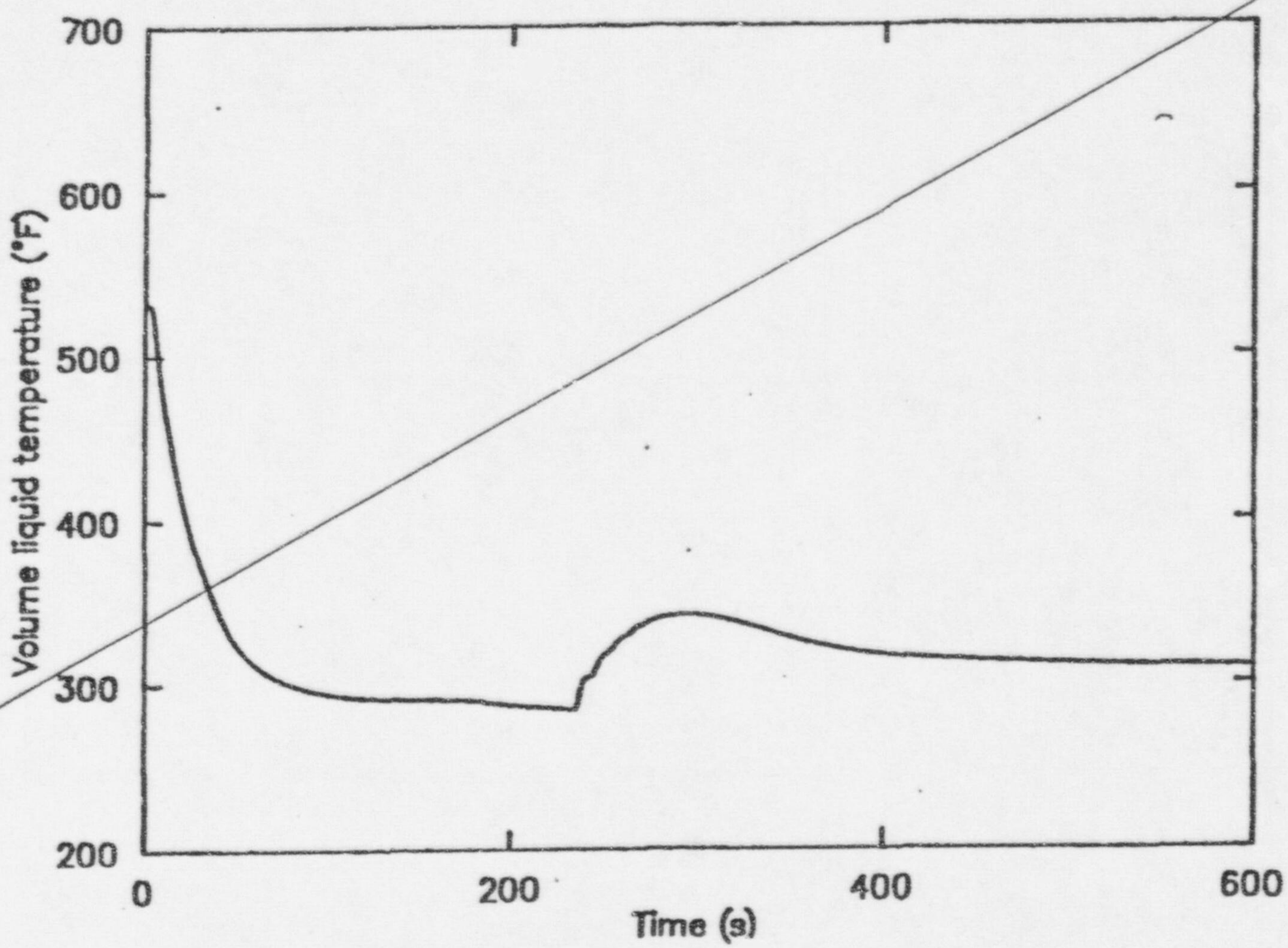

FIGURE 14.1.5-3 AFFECTED CORE SE. ,OR INLET TEMPERATURE VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

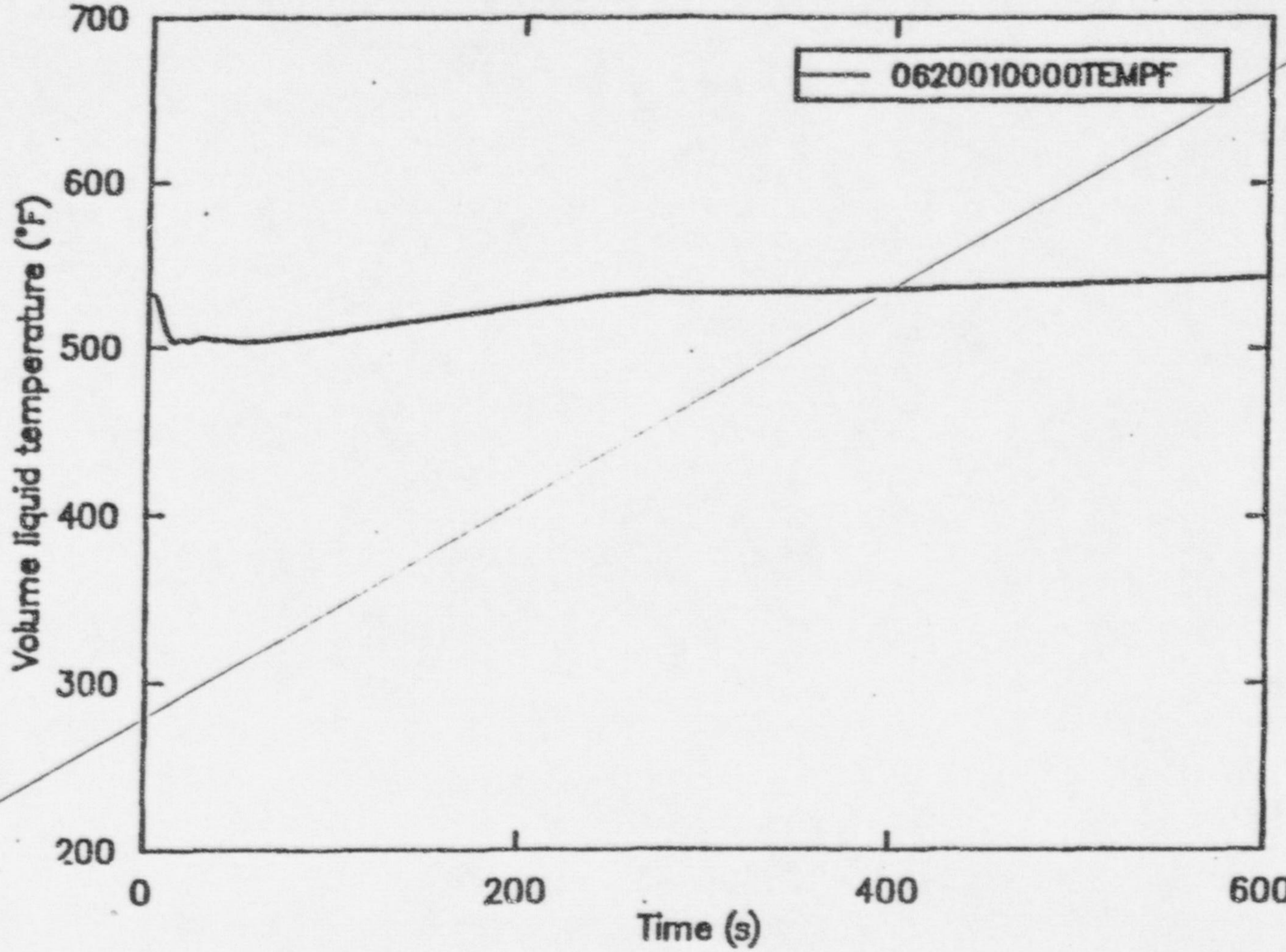

FIGURE 14.1.5-4 INTACT CORE SECTOR INLET TEMPERATURE VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

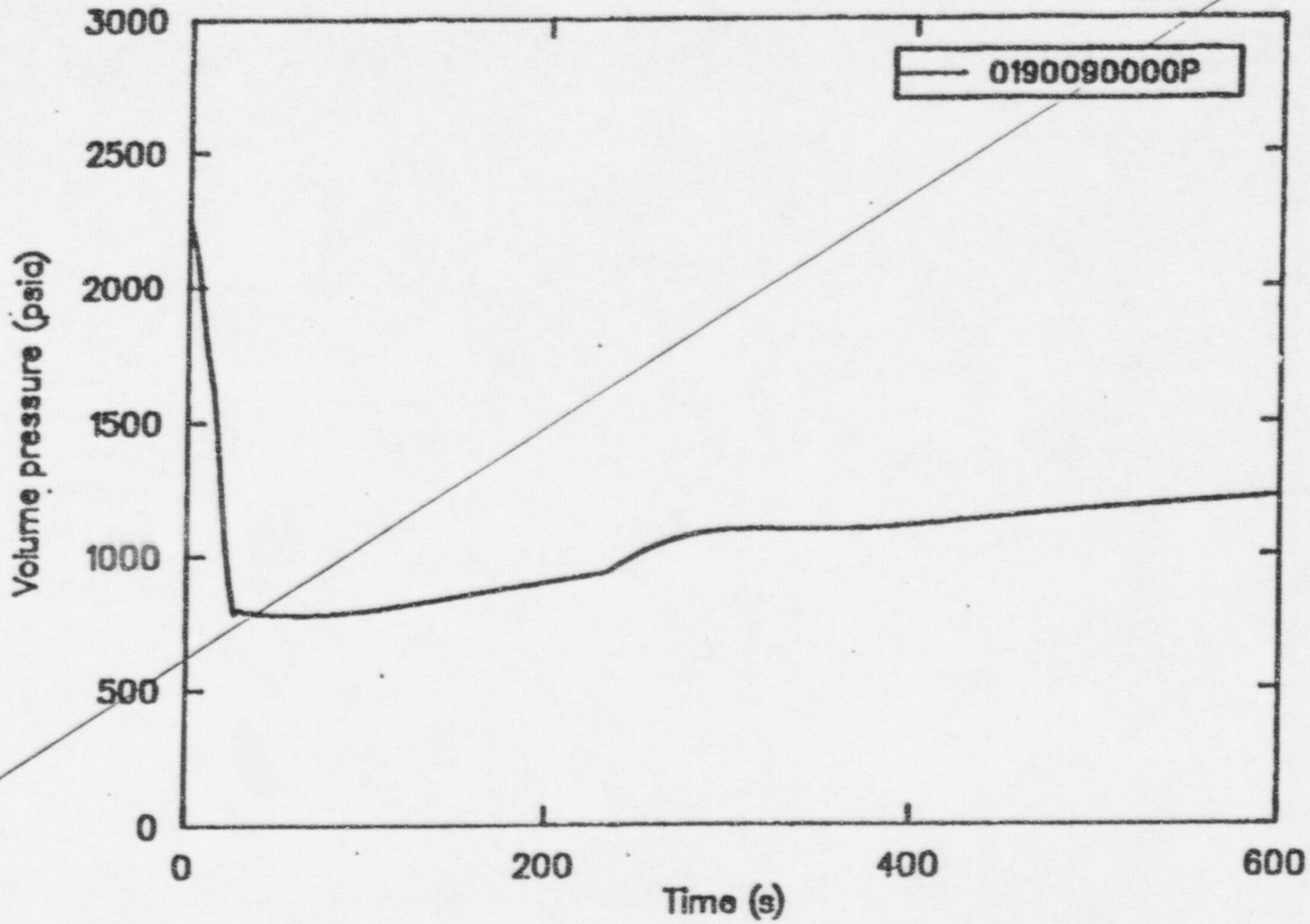

FIGURE 14.1.5-5 PRESSURIZER PRESSURE VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

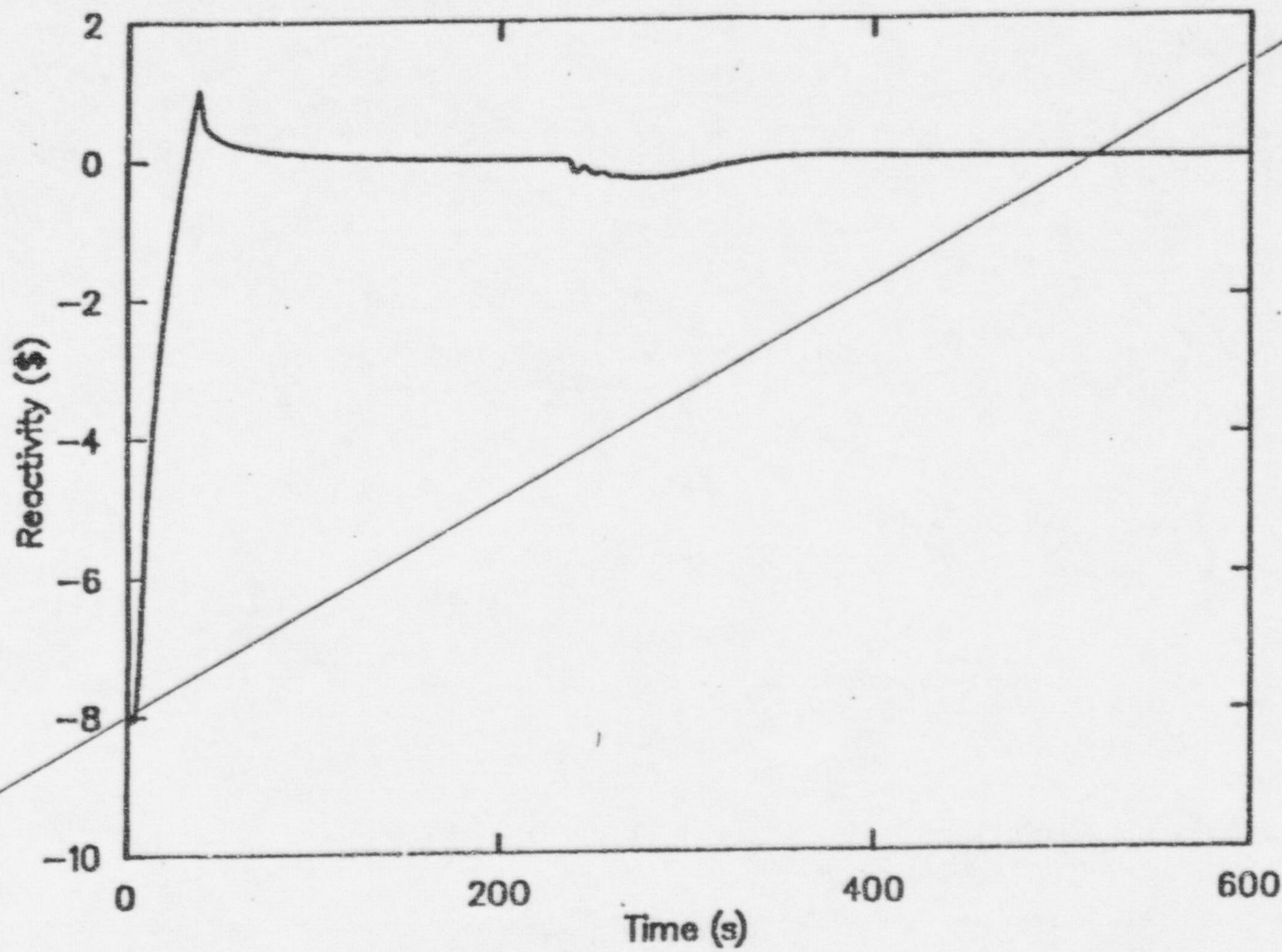

FIGURE 14.1.5-6 ADVANCED NUCLEAR FUELS - RELAP CALCULATED CORE REACTIVITY VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

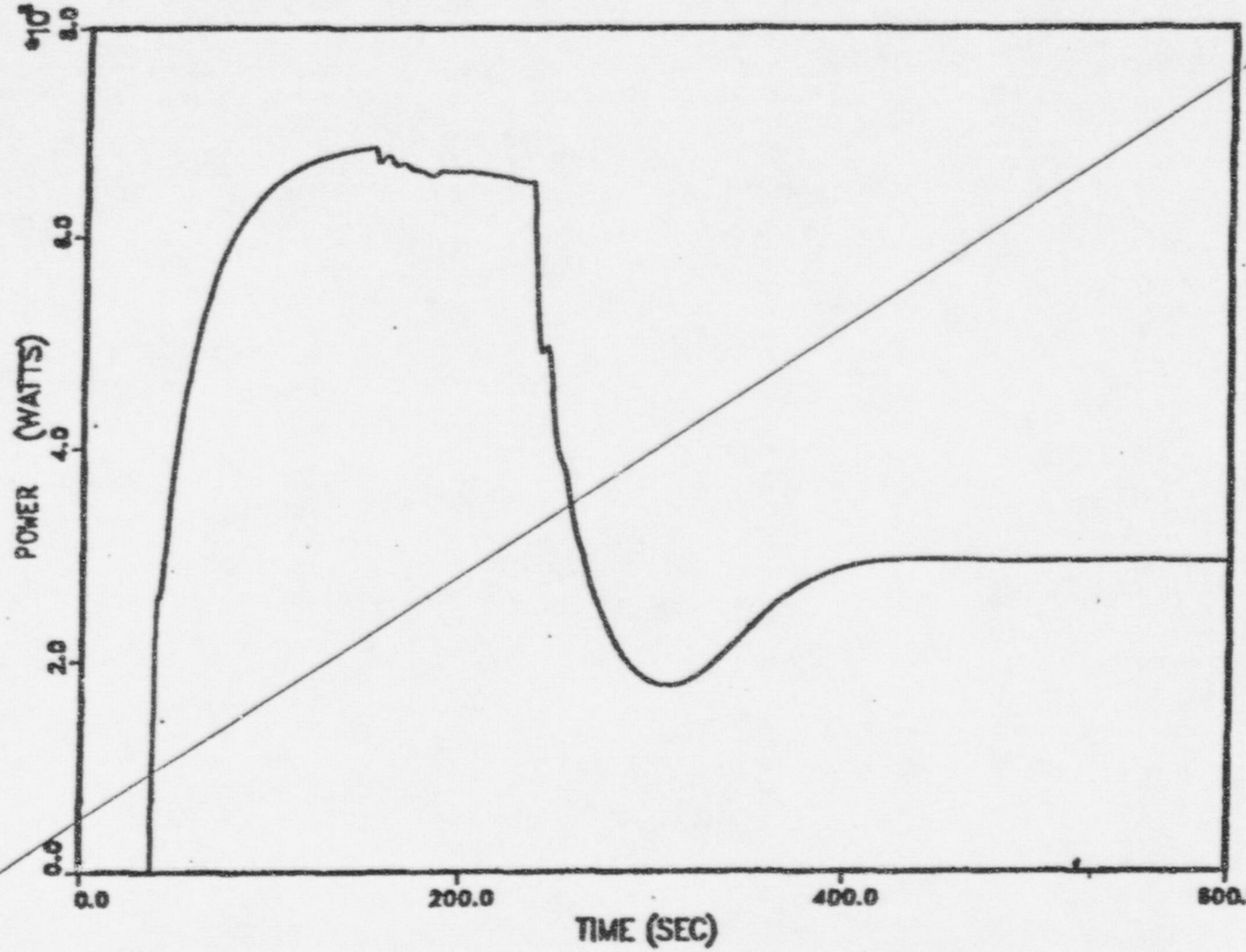

FIGURE 14.1.5-7 CORE POWER VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

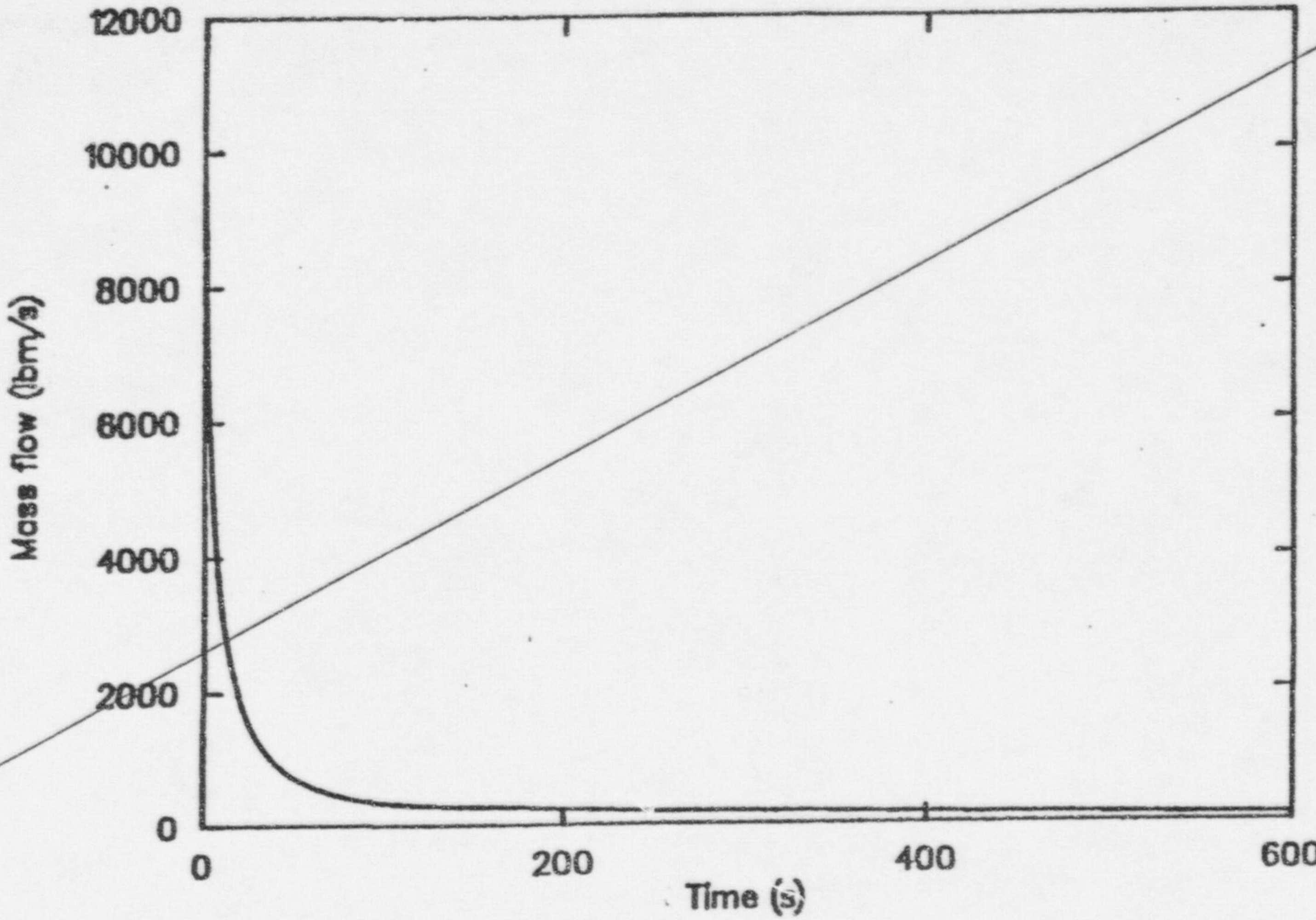

FIGURE 14.1.5-8 AFFECTED STEAM GENERATOR BREAK FLOW VS. TIME - HOT ZERO POWER WITHOUT OFFSITE POWER

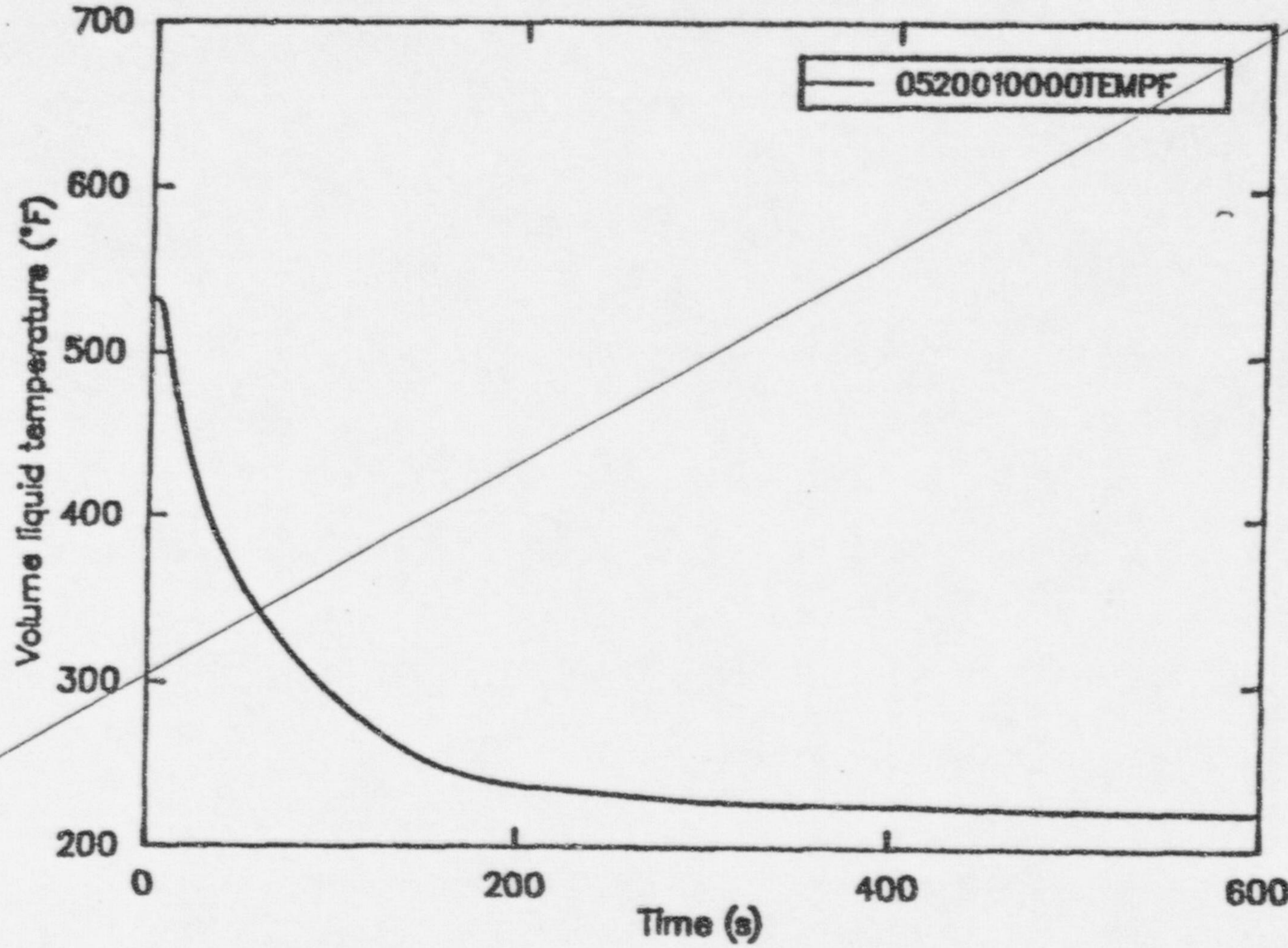

FIGURE 14.1.5-9 AFFECTED CORE SECTOR INLET TEMPERATURE VS. TIME – HOT ZERO POWER WITHOUT OFFSITE POWER

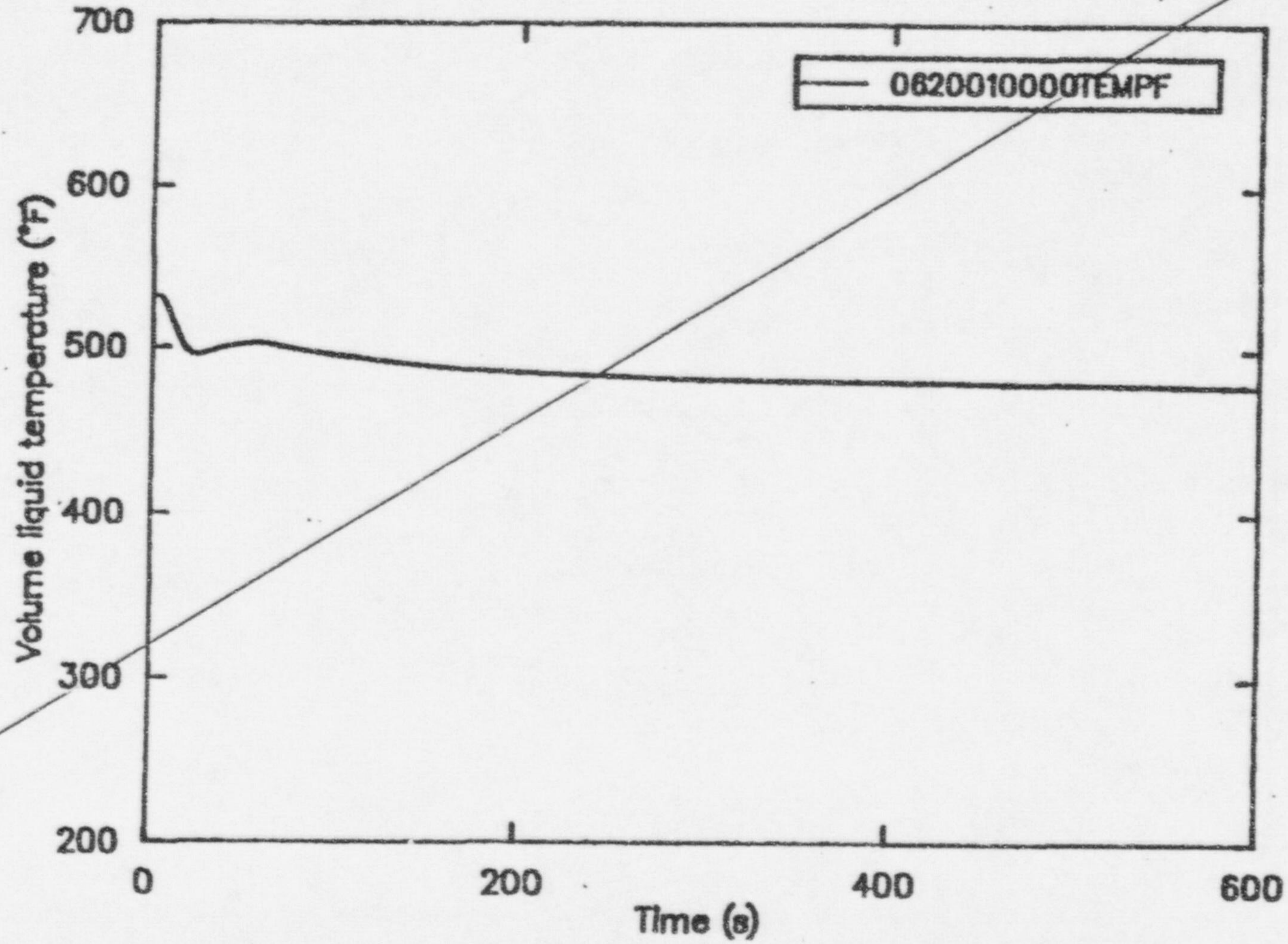

FIGURE 14.1.5-10 INTACT CORE SECTOR INLET TEMPERATURE VS. TIME - HOT ZERO POWER WITHOUT OFFSITE POWER

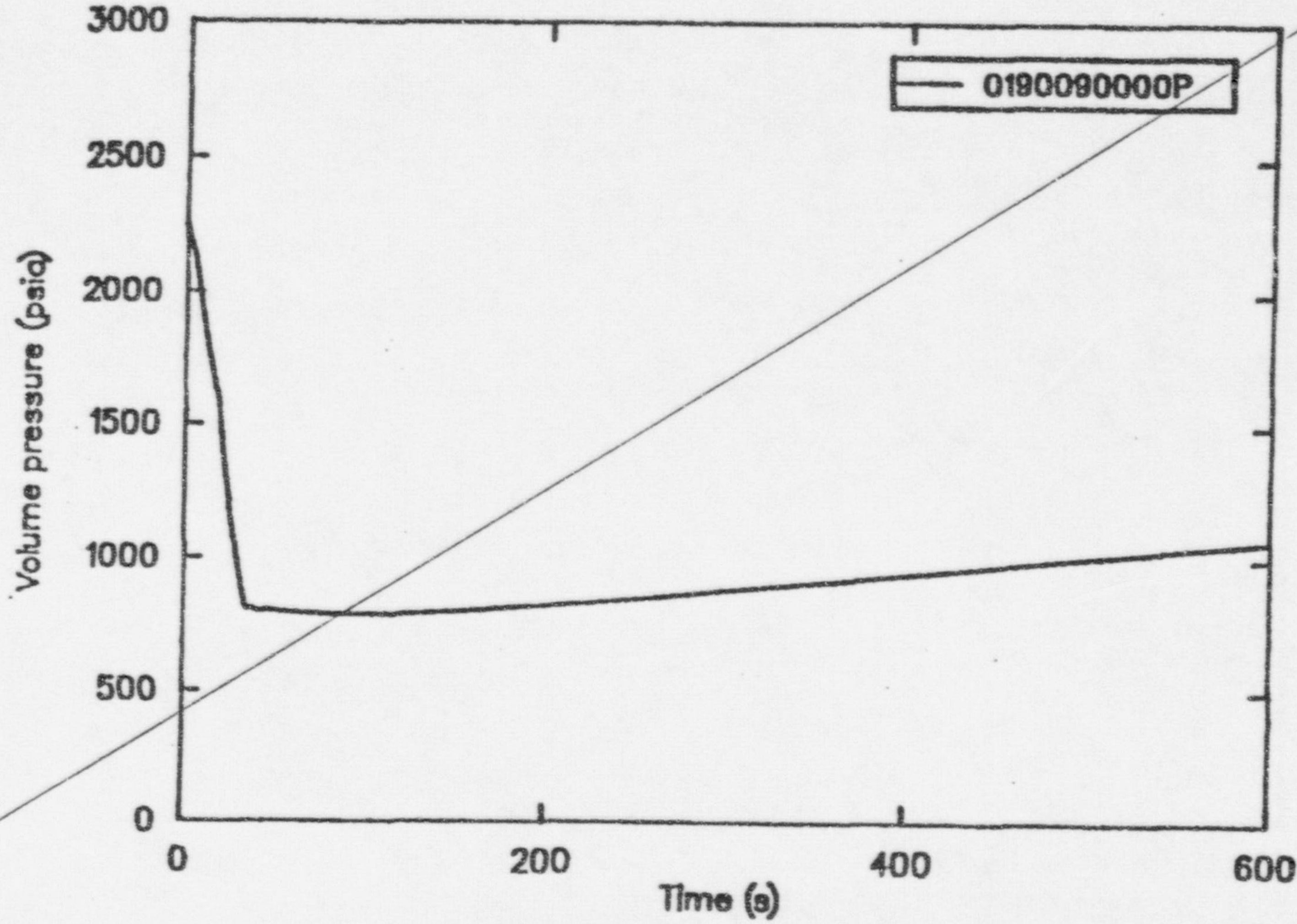

FIGURE 14.1.5-11 PRESSURIZER PRESSURE VS. TIME - HOT ZERO POWER WITHOUT OFFSITE POWER

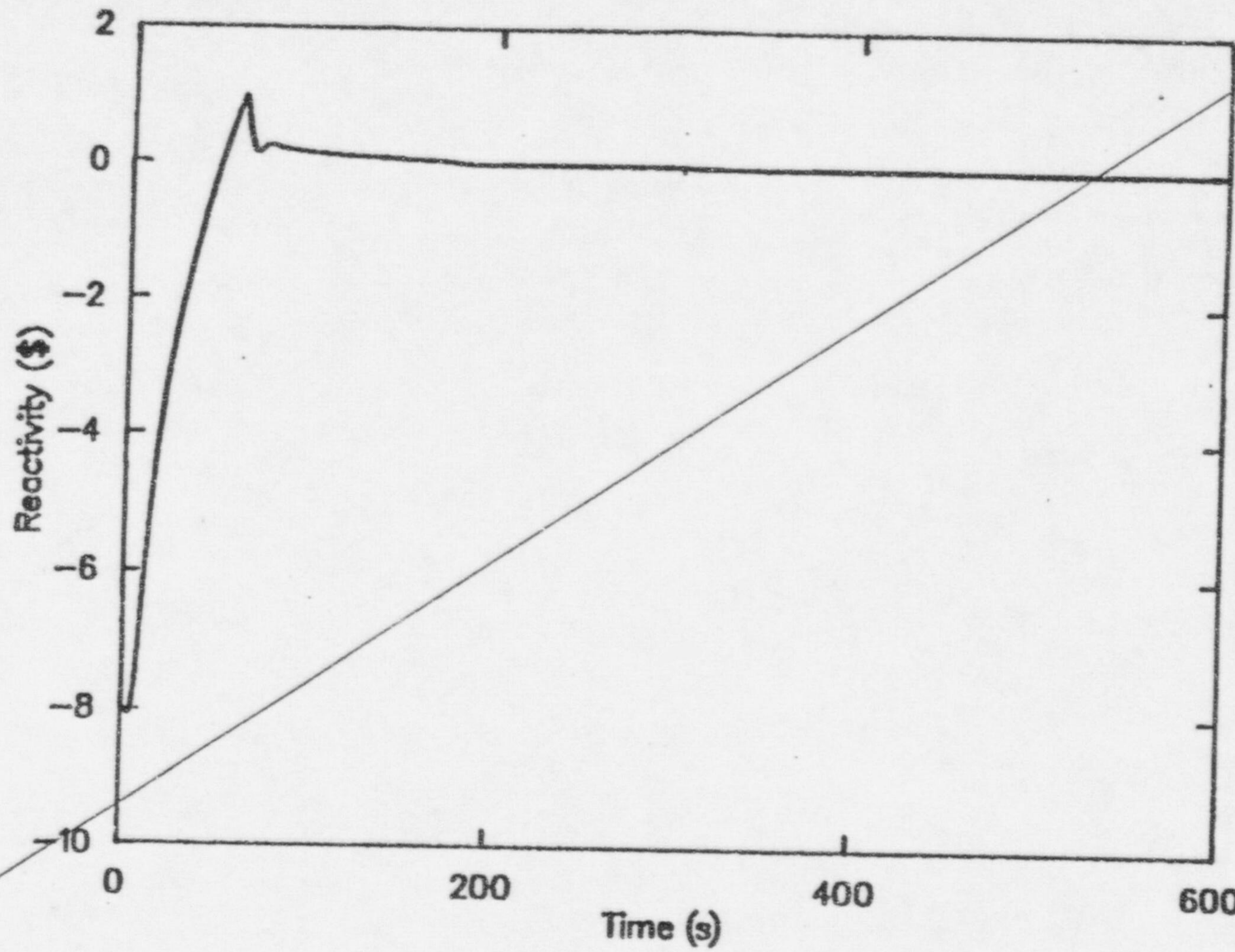

FIGURE 14.1.5-12 ADVANCED NUCLEAR FUELS – RELAP CALCULATED CORE REACTIVITY VS. TIME – HOT ZERO POWER WITHOUT OFFSITE POWER

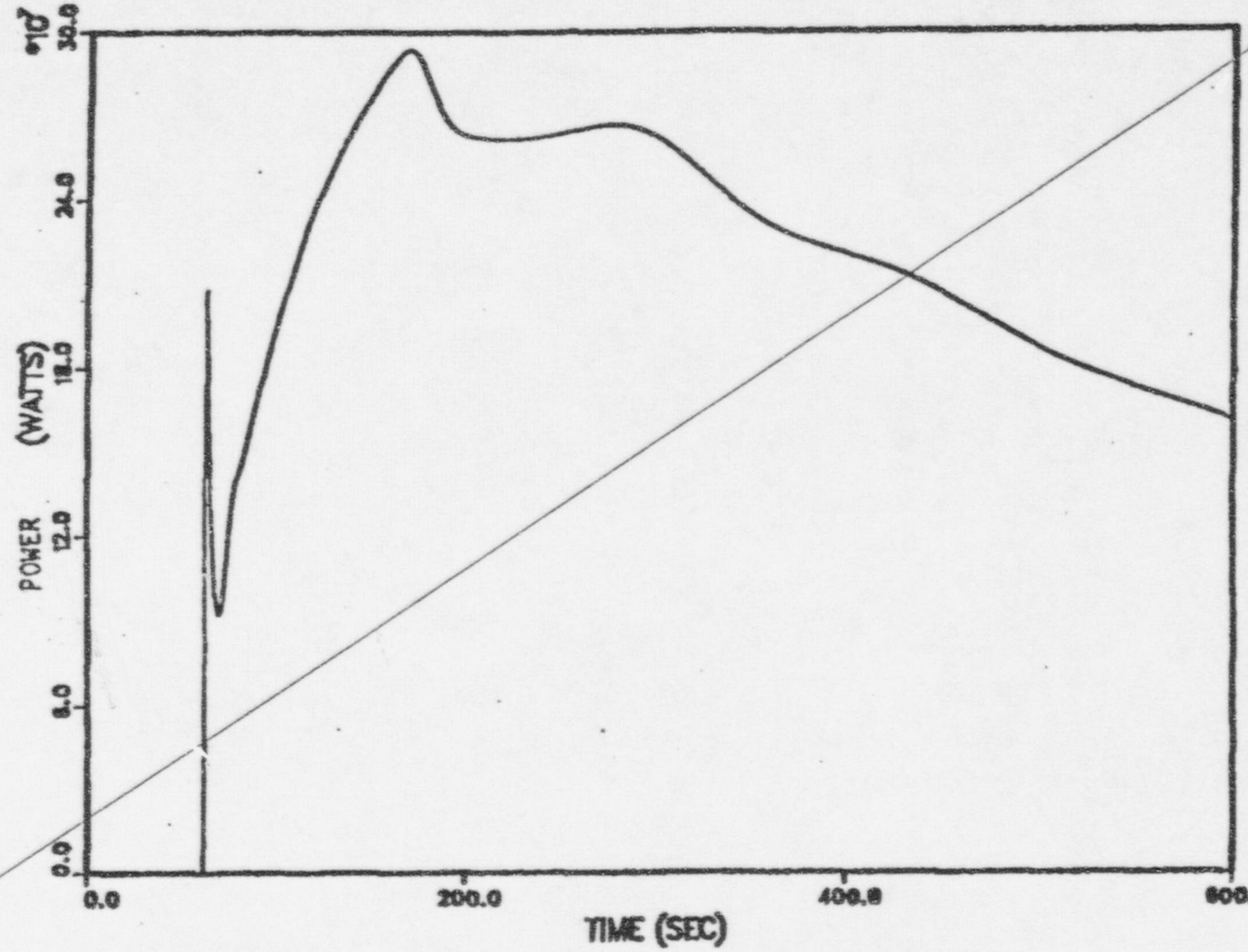

FIGURE 14.1.5-13 CORE POWER VS. TIME - HOT ZERO POWER WITHOUT OFFSITE POWER

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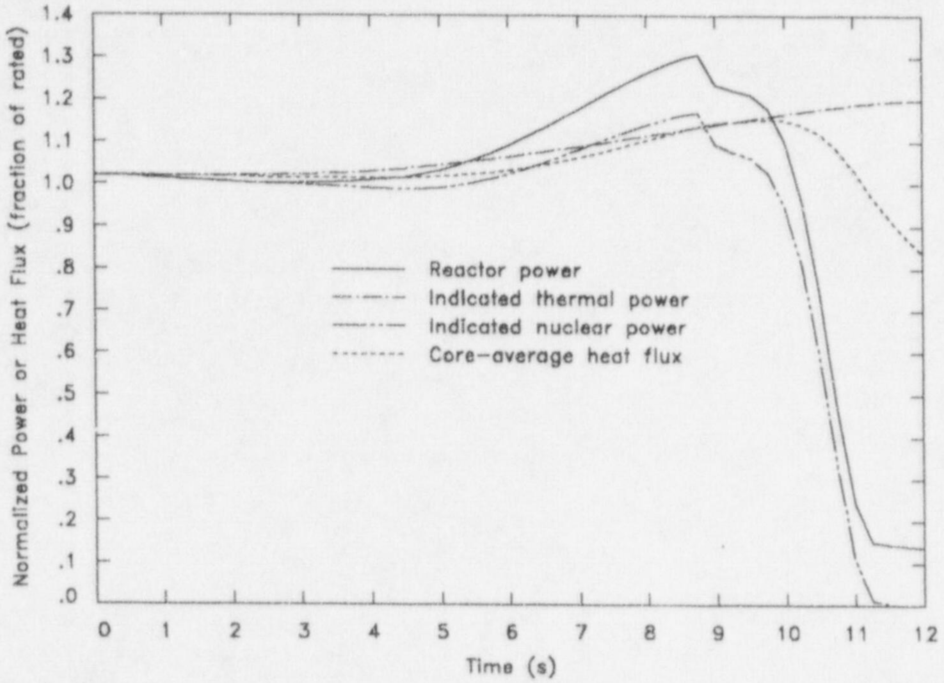

Figure 14.1.5.1-1 Normalized Core Power (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

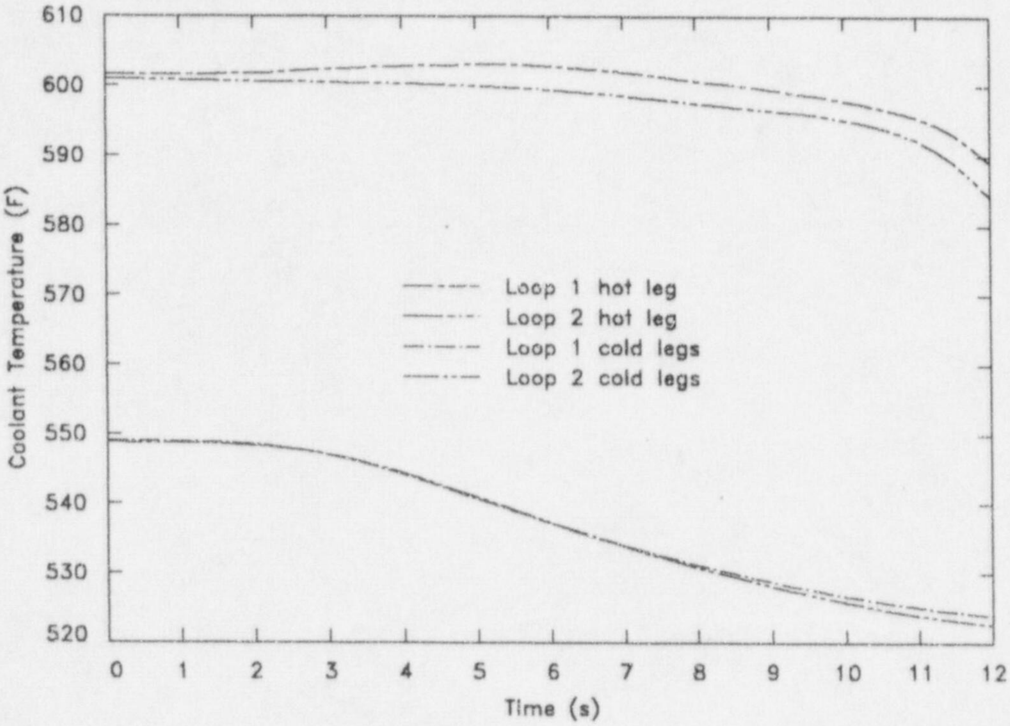

Figure 14.1.5.1-2 Core Inlet Temperatures (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

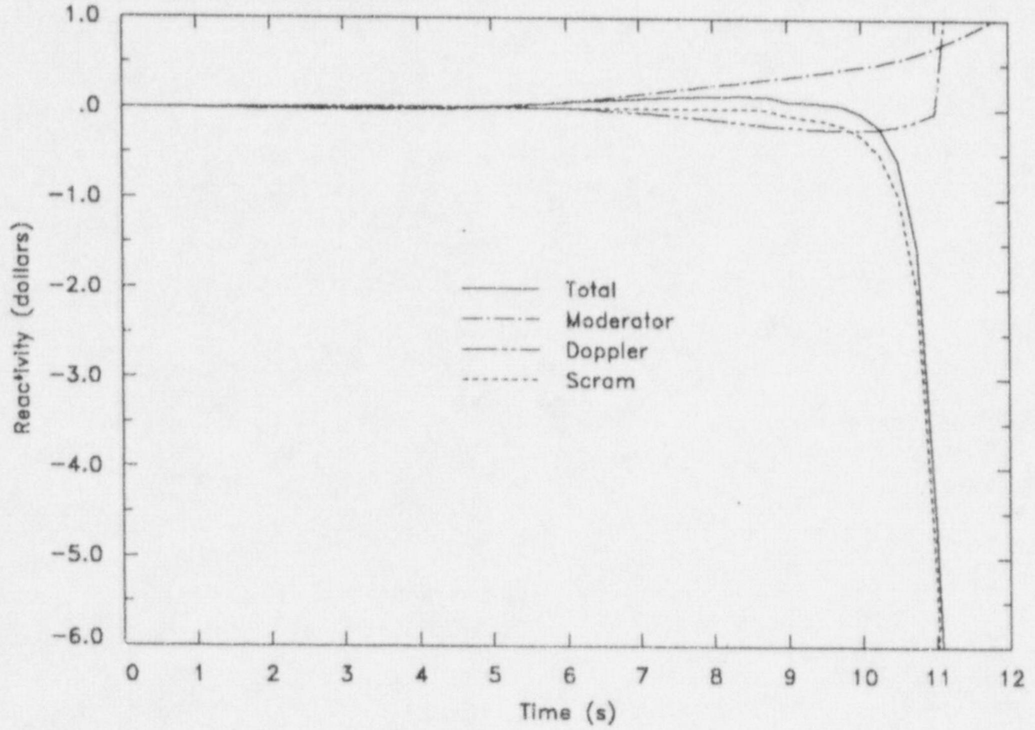

Figure 14.1.5.1-3 Reactivity Feedback (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

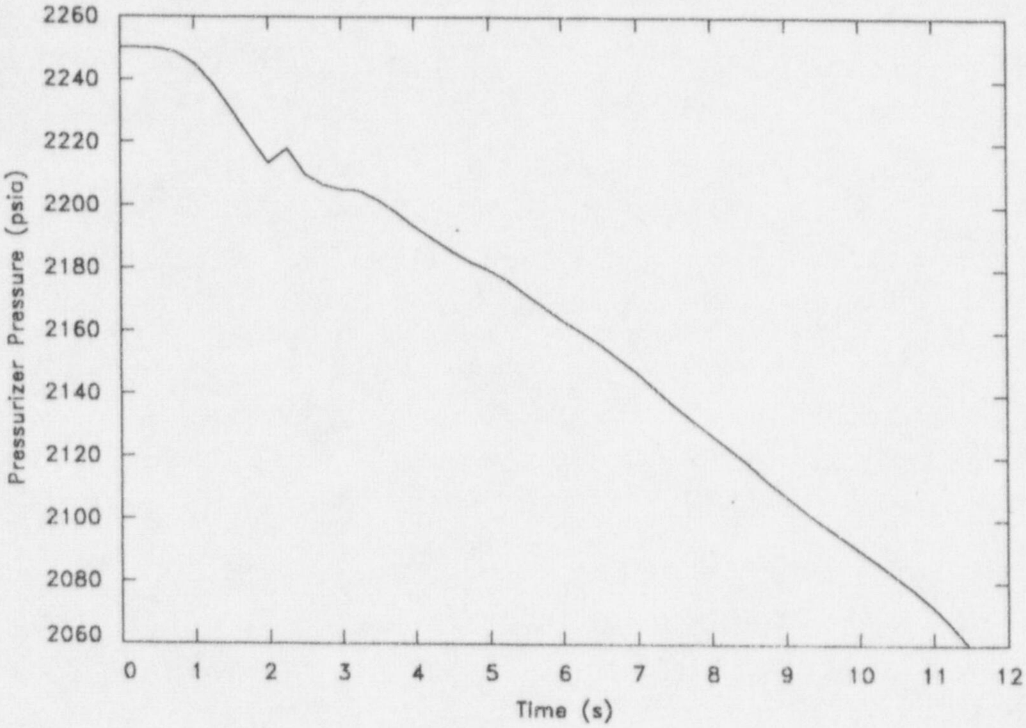

Figure 14.1.5.1-4 Pressurizer Pressure (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

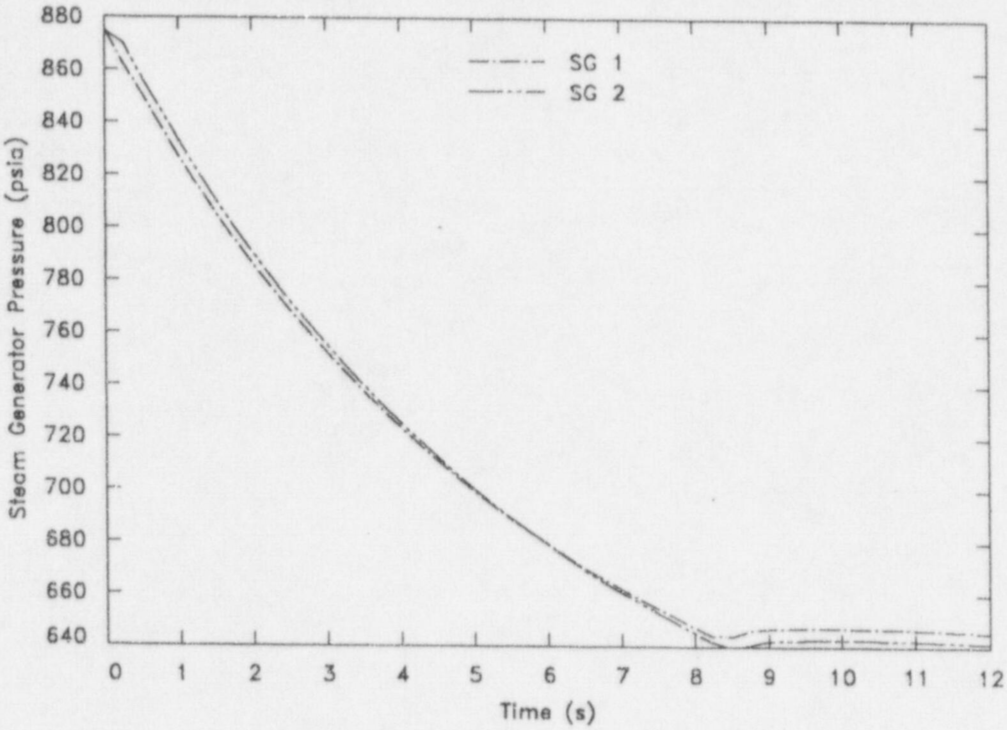

Figure 14.1.5.1-5 Steam Generator Pressures (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

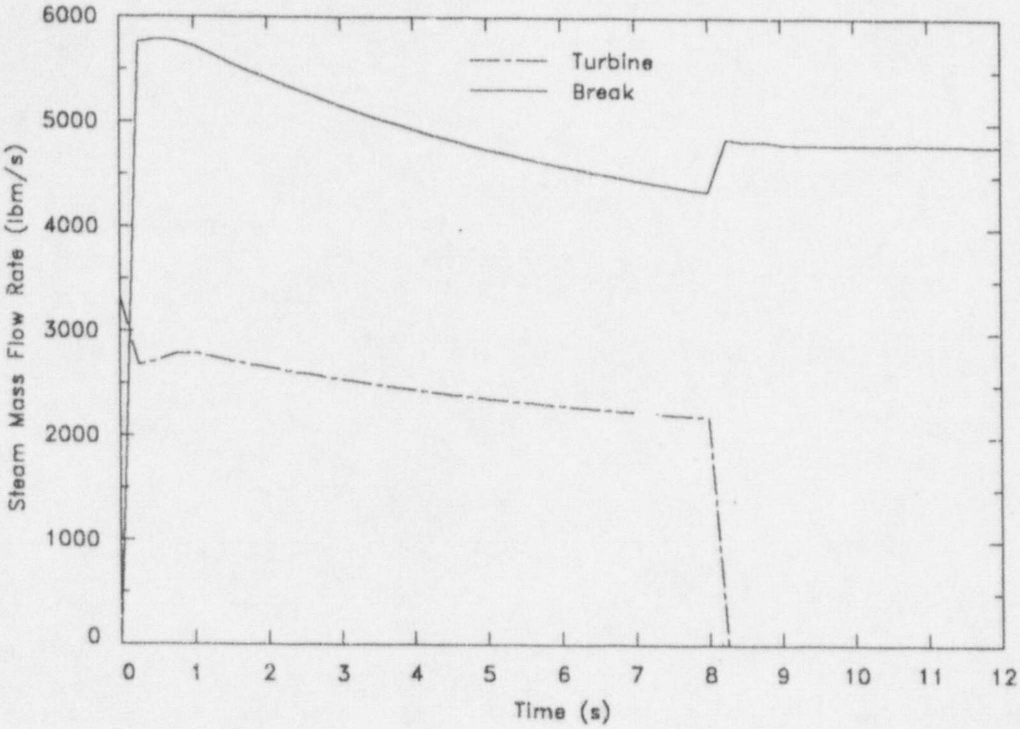

Figure 14.1.5.1-6 Main Steam Line Flow (Symmetric 3.50 ft^2 Break Outside Containment with Offsite Power Available)

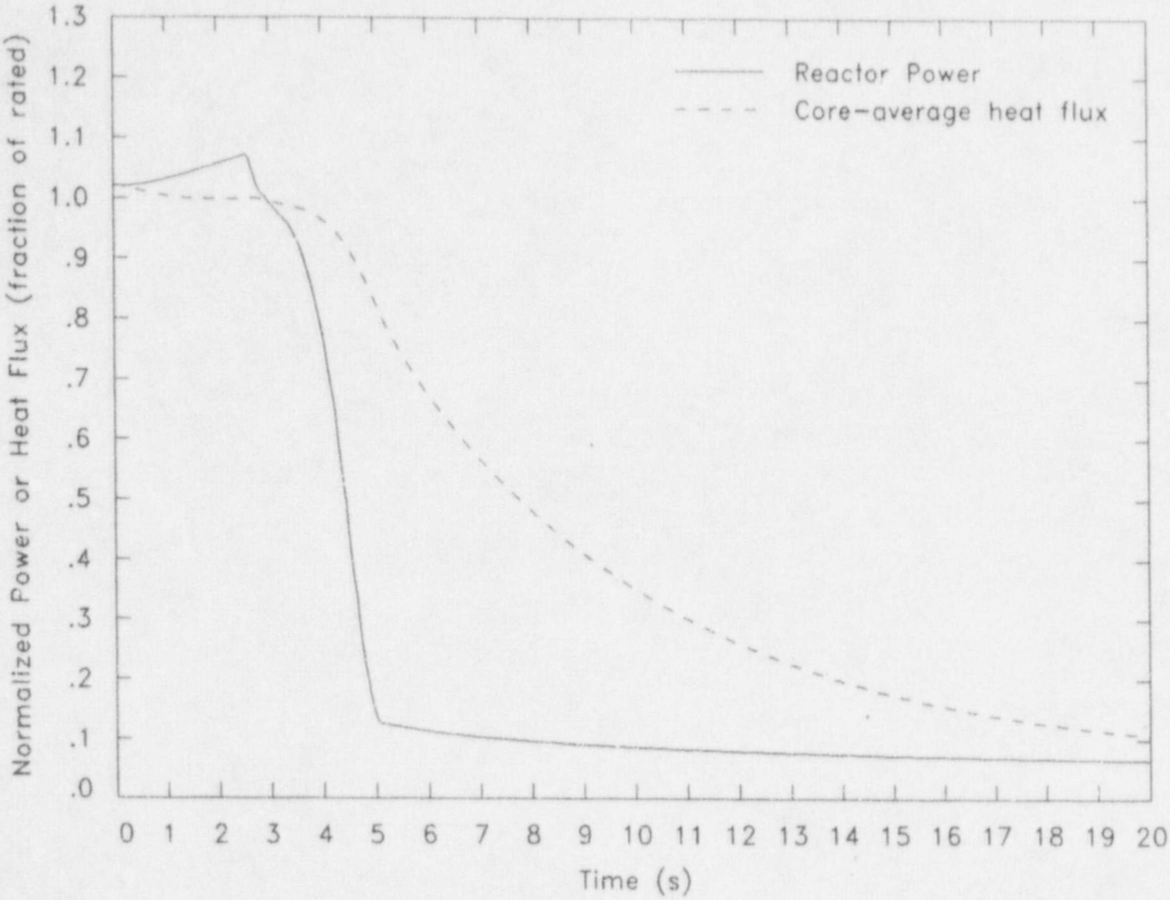

Figure 14.1.5.1-7 Normalized Power and Heat Flux (Asymmetric 3.51 ft^2 Break Inside Containment with Loss of Offsite Power)

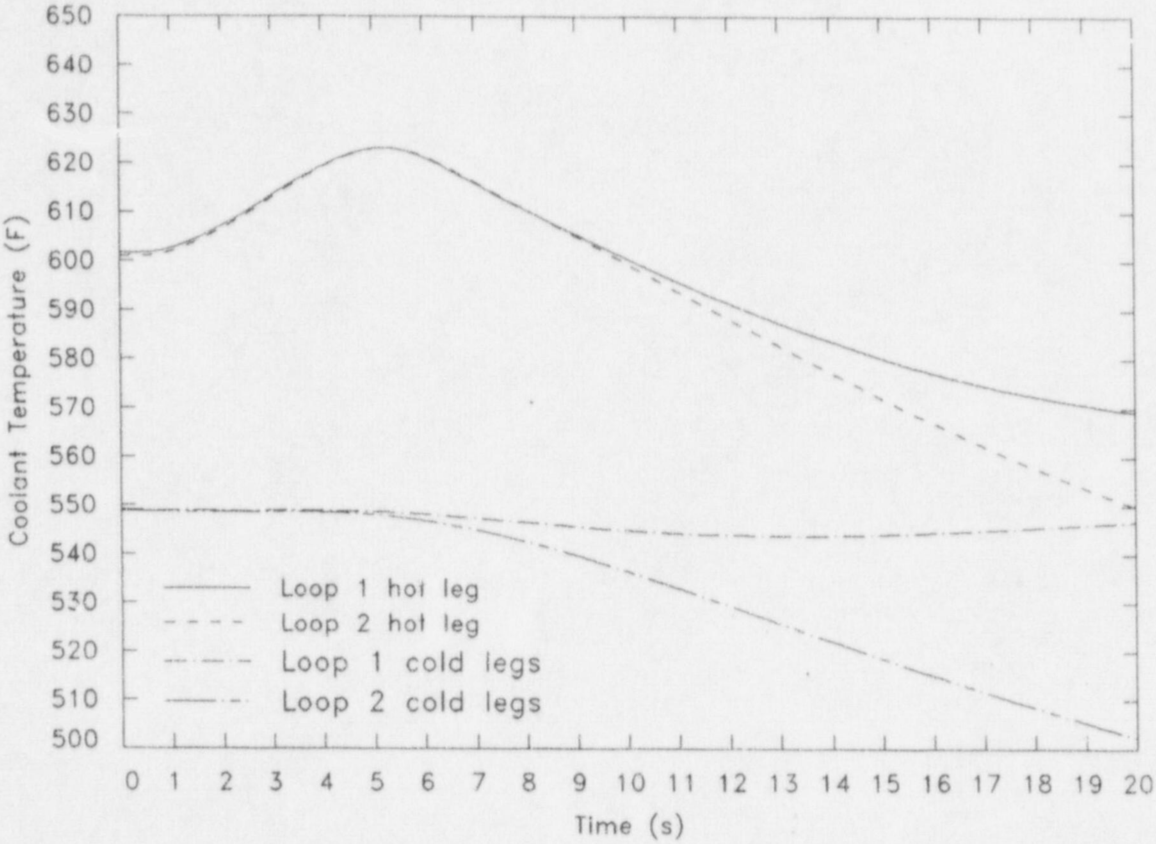

Figure 14.1.5.1-8 Reactor Coolant Temperatures (Asymmetric 3.51 ft^2 Break Inside Containment with Loss of Offsite Power)

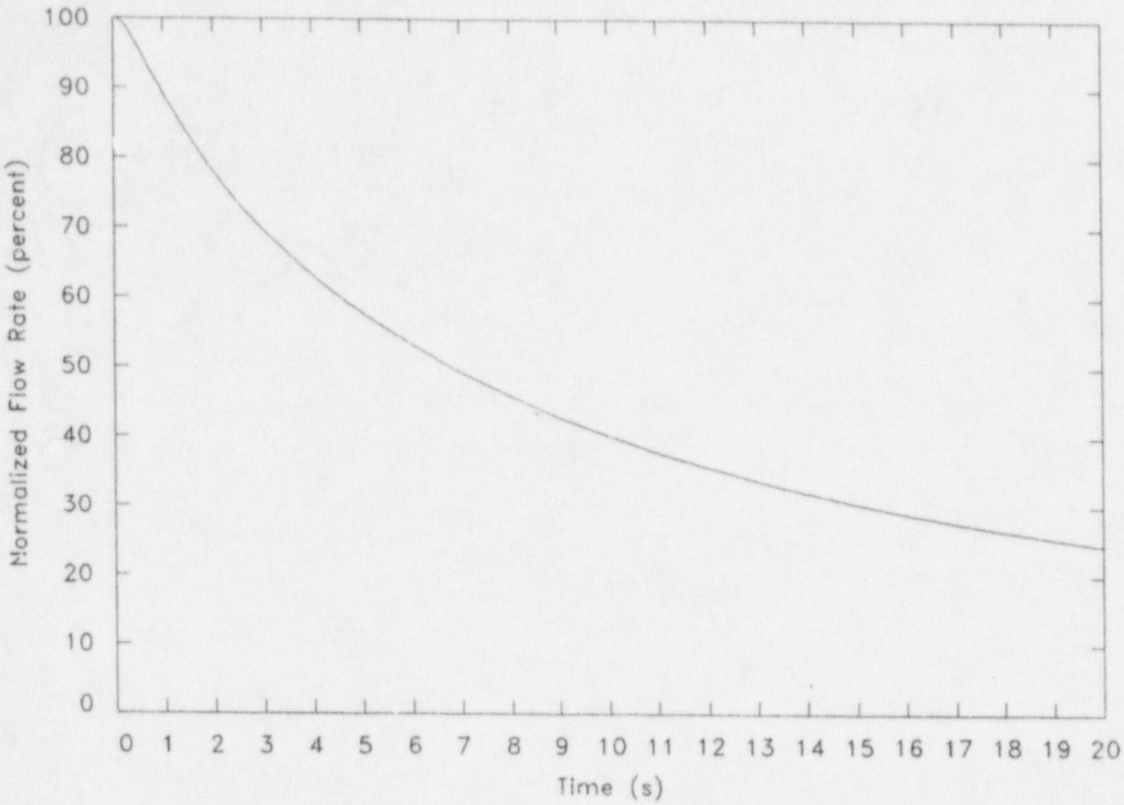

Figure 14.1.5.1-9 Normalized RCS Flow Rate (Asymmetric 3.51 ft^2 Break Inside Containment with Loss of Offsite Power)

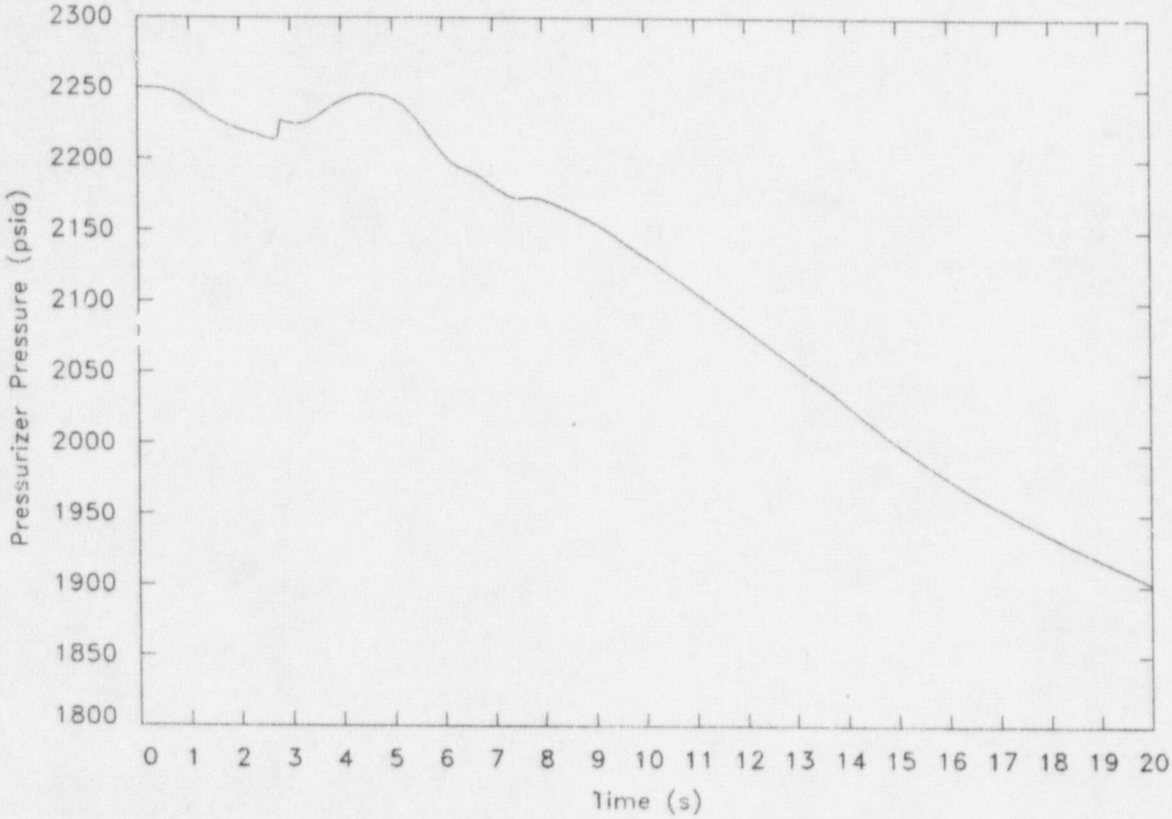

Figure 14.1.5.1-10 Pressurizer Pressure (Asymmetric 3.51 ft^2 Break Inside Containment with Loss of Offsite Power)

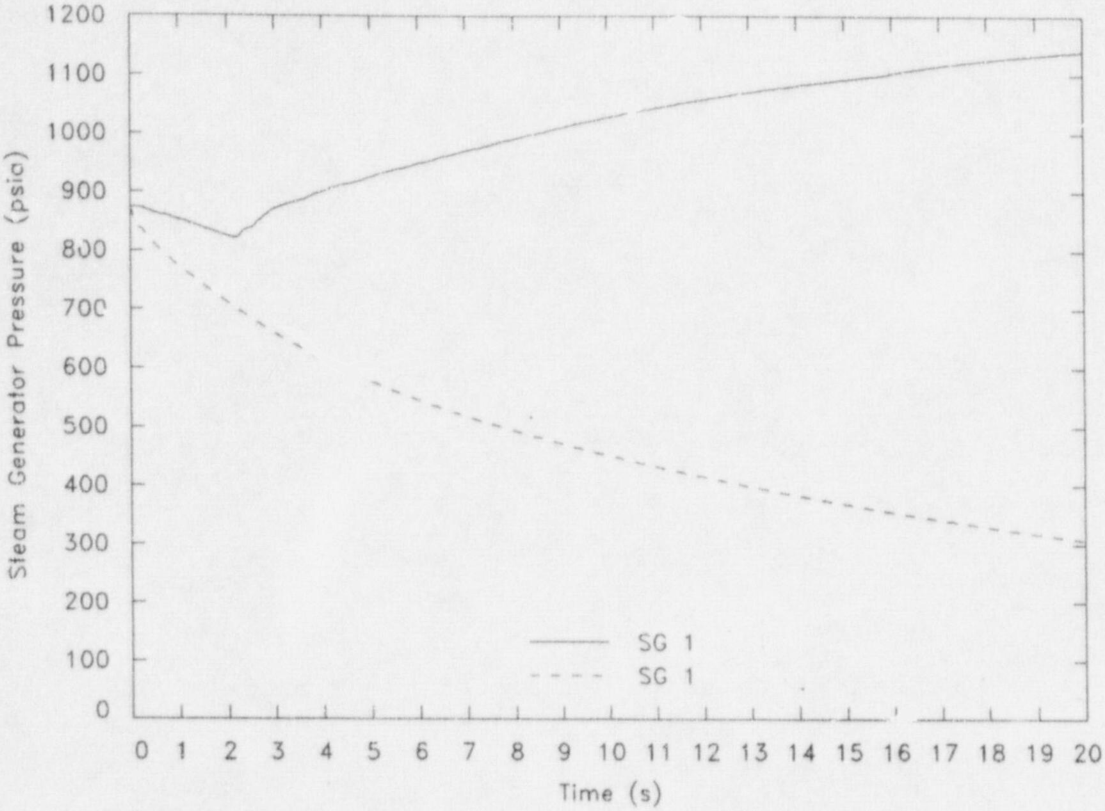

Figure 14.1.5.1-11 Steam Generator Pressures (Asymmetric 3.51 ft^2 Break Inside Containment with Loss of Offsite Power)

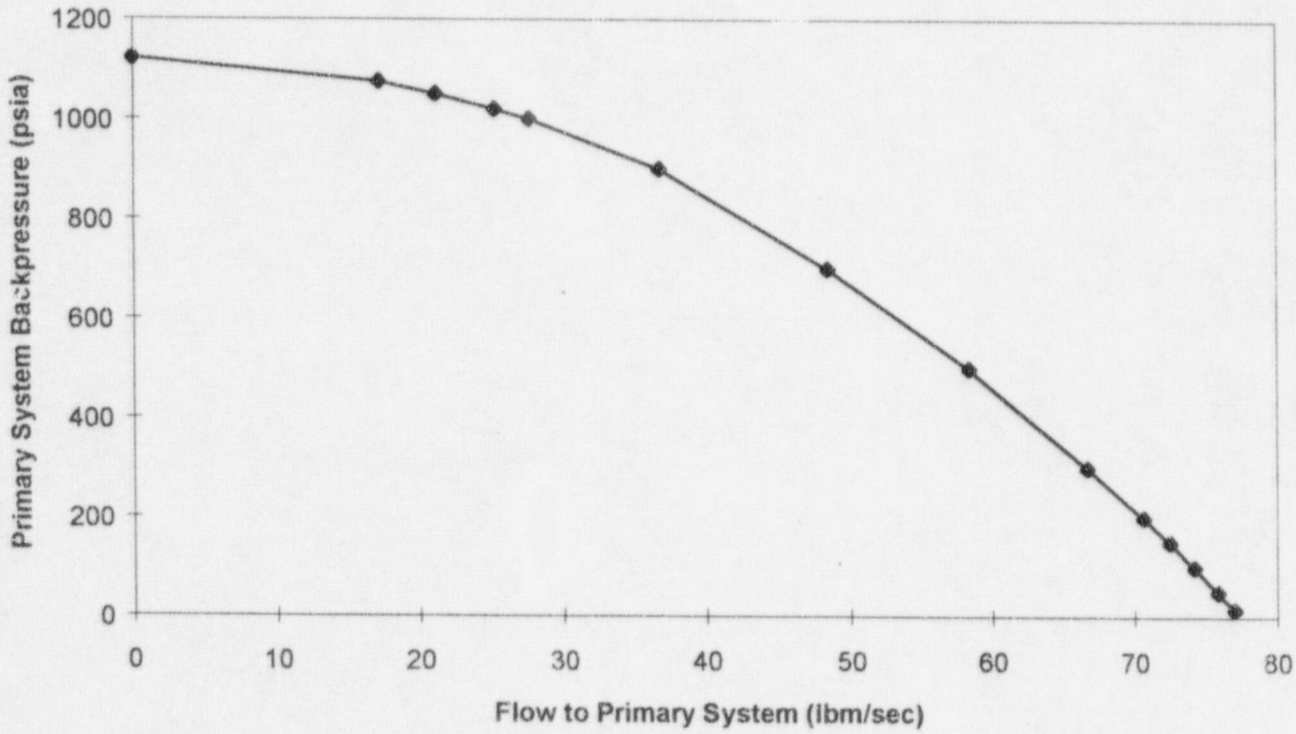

Figure 14.1.5.2-1 One Pump High Pressure Safety Injection System Delivery vs. Primary Pressure (Post-Scram Steam Line Break)

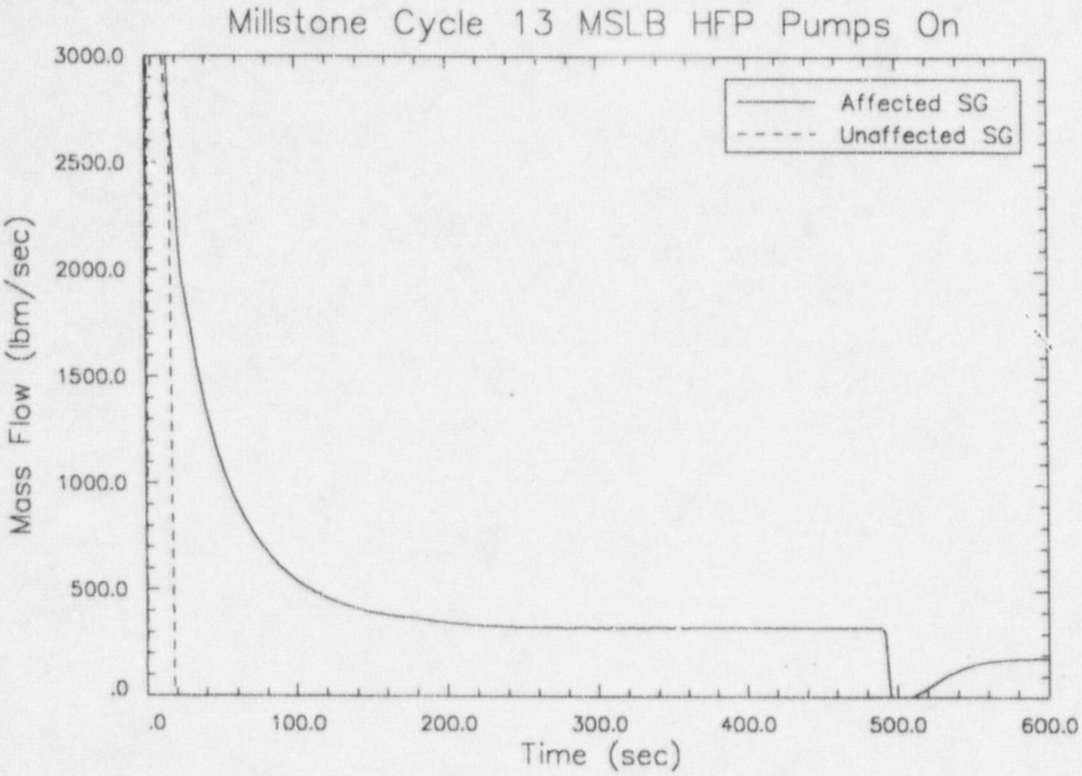

Figure 14.1.5.2-2 Steam Generator Break Flow (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

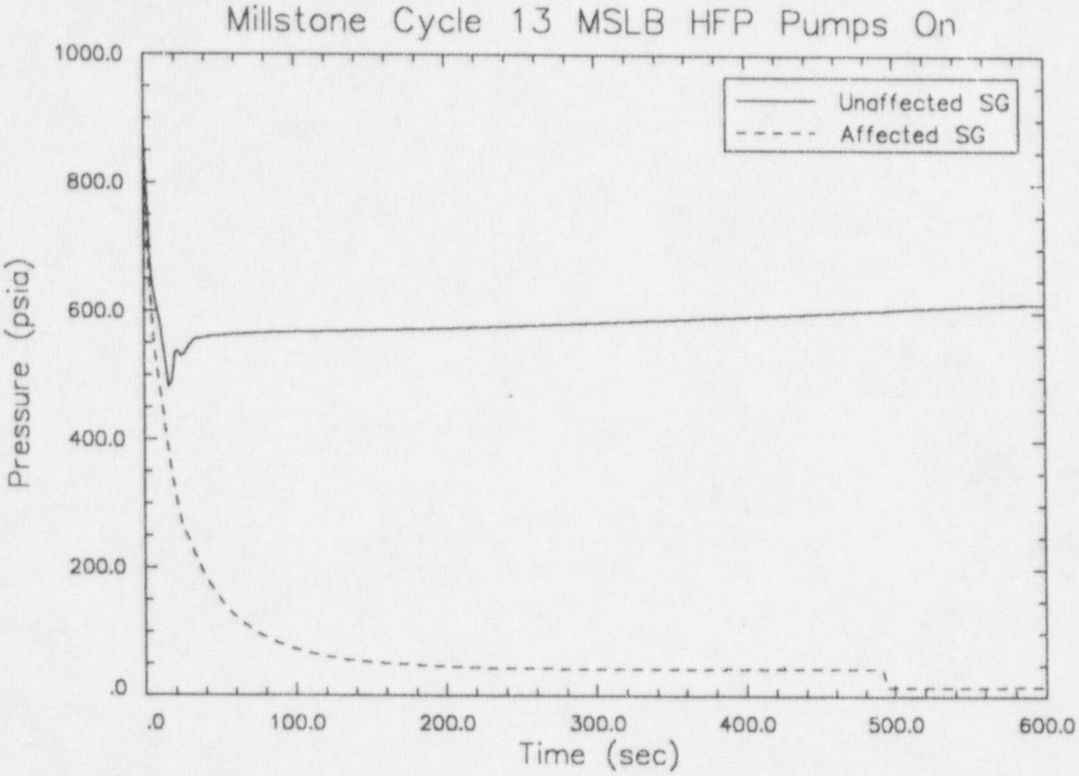

Figure 14.1.5.2-3 Steam Generators' Secondary Pressures (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

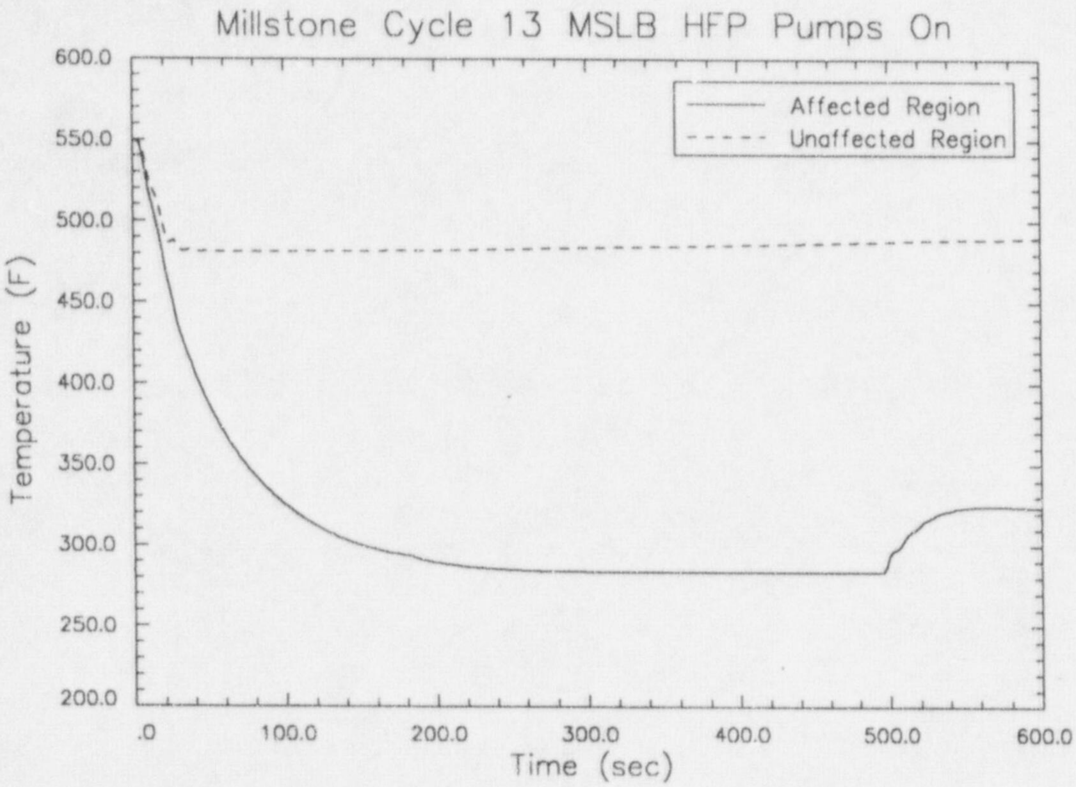

Figure 14.1.5.2-4 Core Inlet Temperatures (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

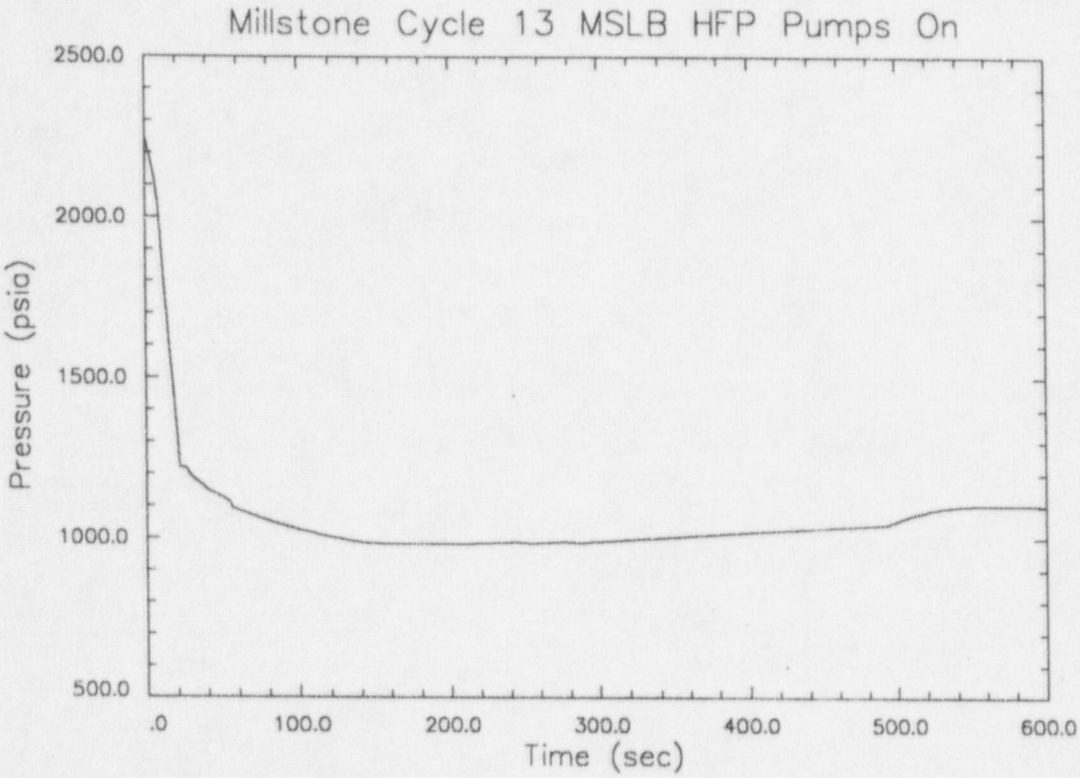

Figure 14.1.5.2-5 Pressurizer Pressure (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

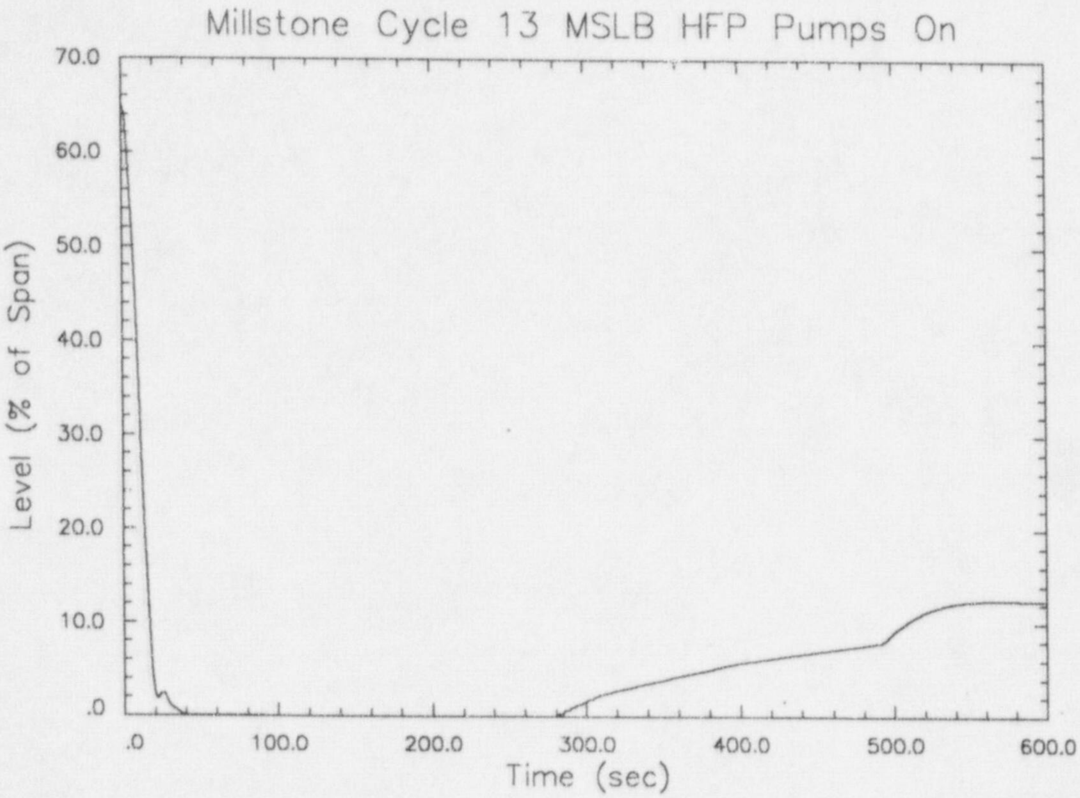

Figure 14.1.5.2-6 Pressurizer Level (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

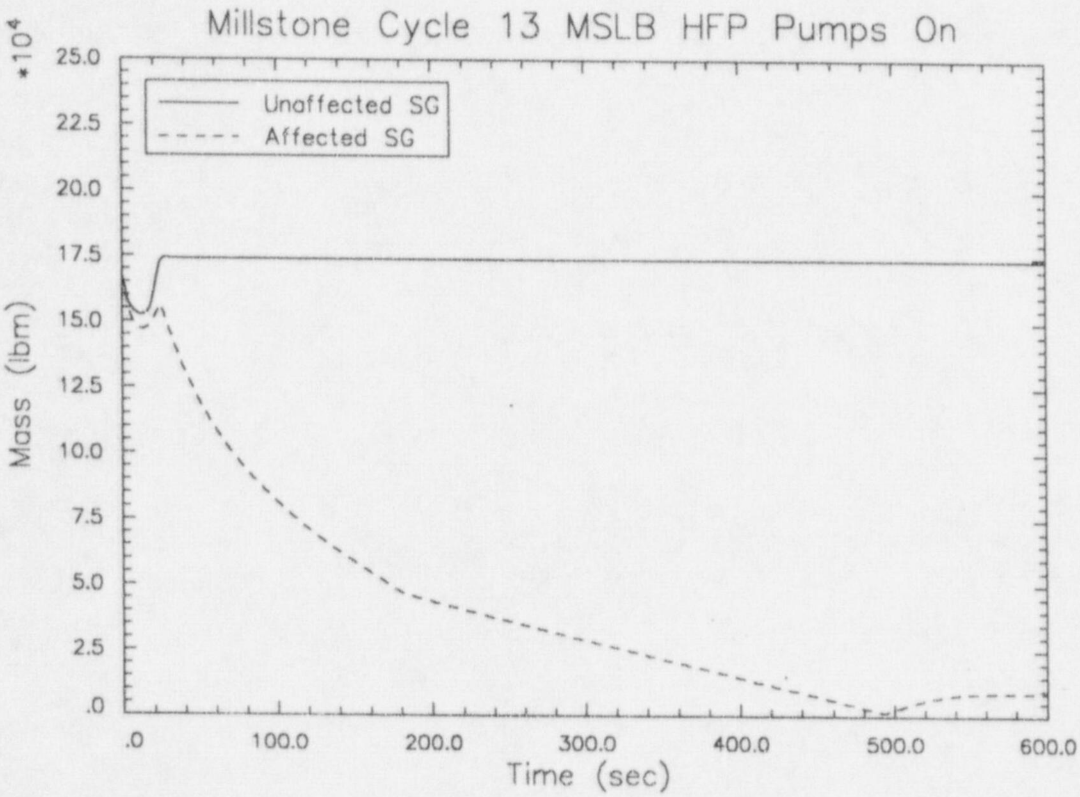

Figure 14.1.5.2-7 Steam Generators' Secondary Mass (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

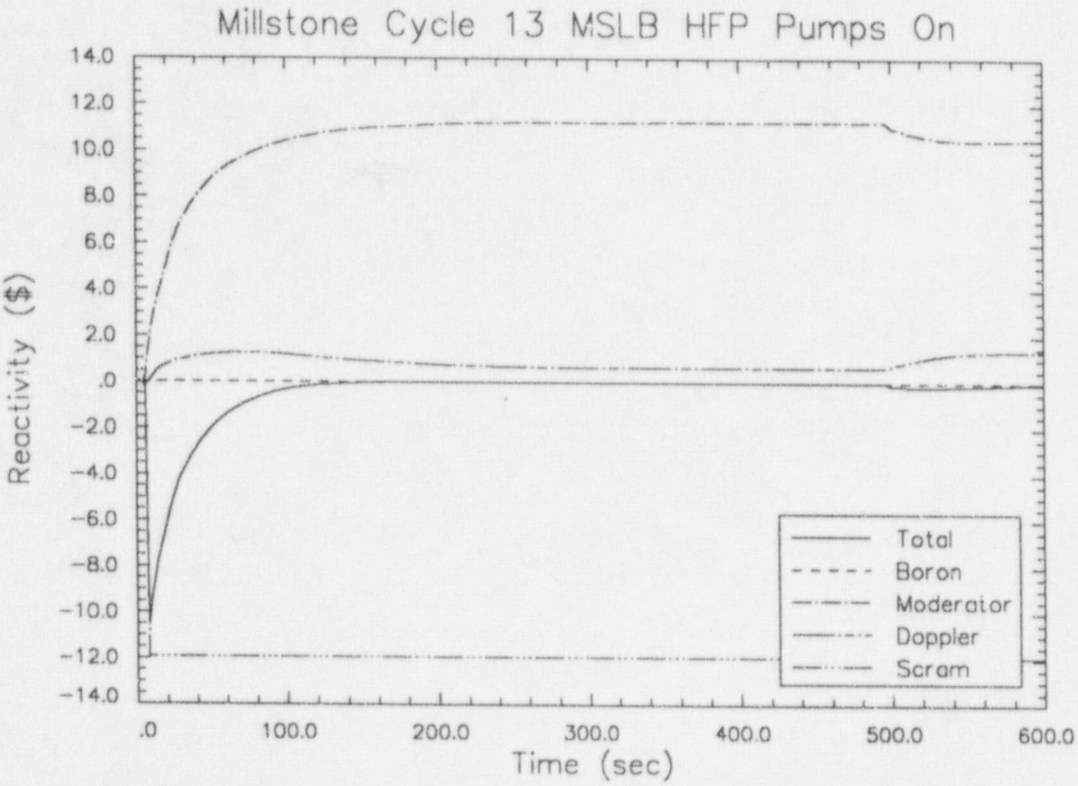

Figure 14.1.5.2-8 Reactivity Components (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

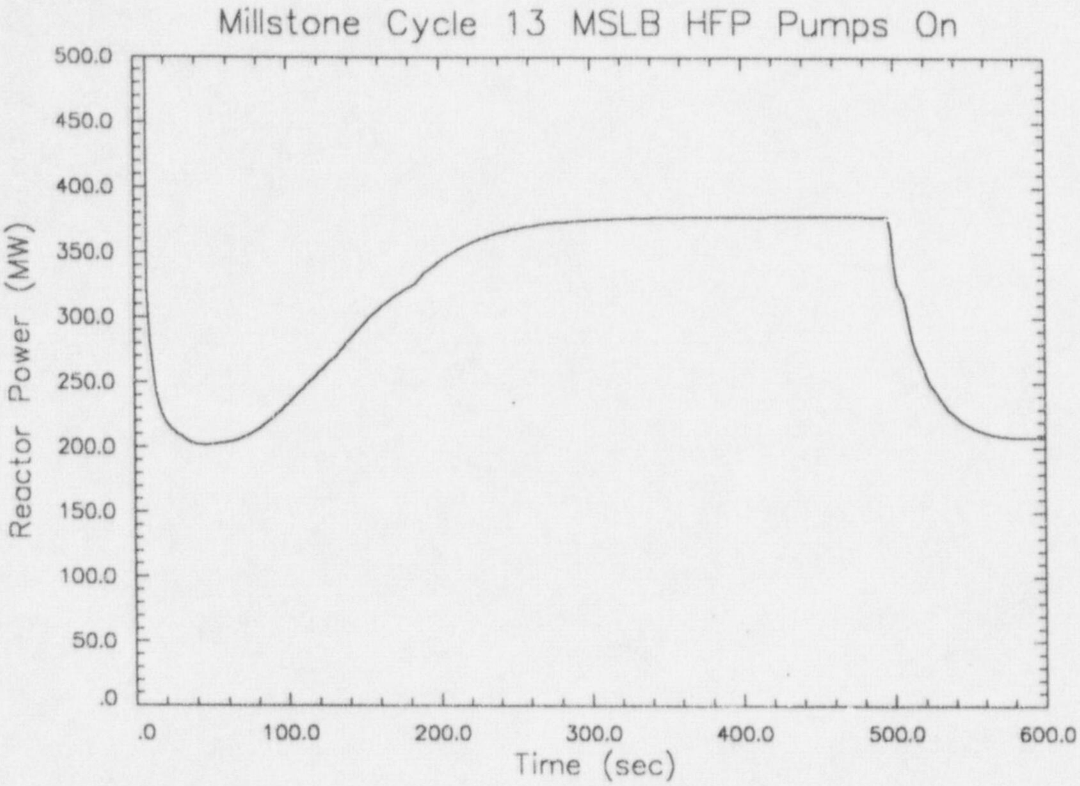

Figure 14.1.5.2-9 Reactor Power (HFP Post-Scram Steam Line Outside Containment Break with Offsite Power Available)

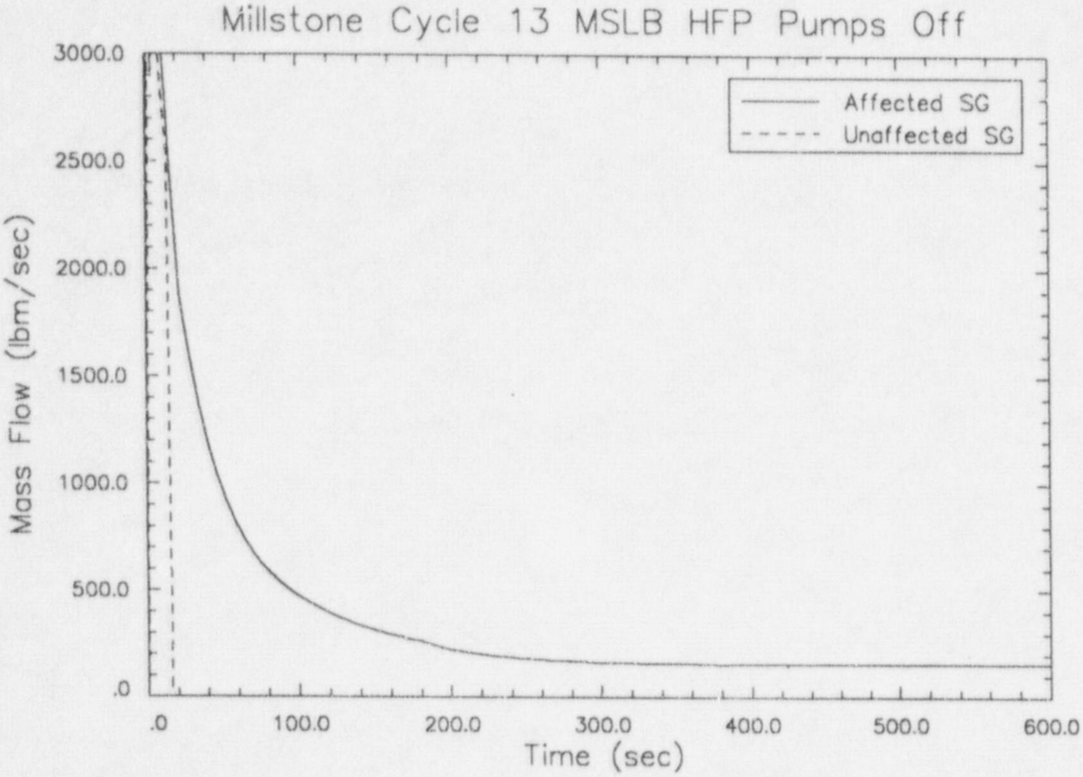

Figure 14.1.5.2-10 Steam Generator Break Flow (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

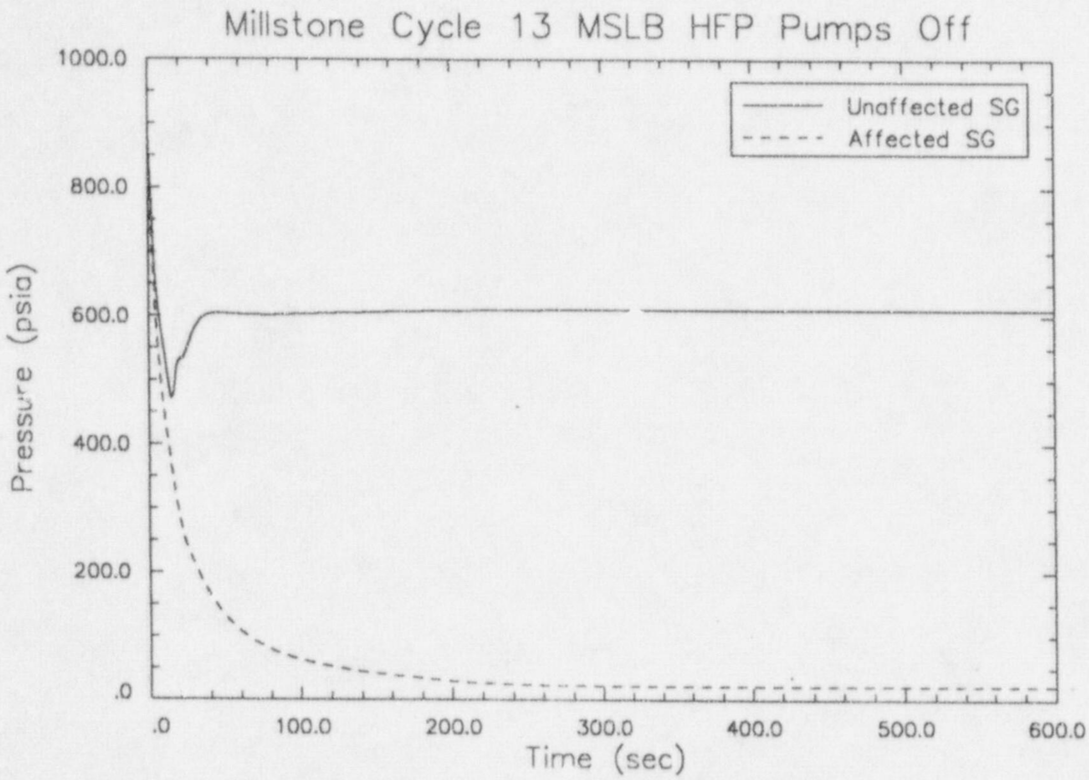

Figure 14.1.5.2-11 Steam Generators' Secondary Pressures (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

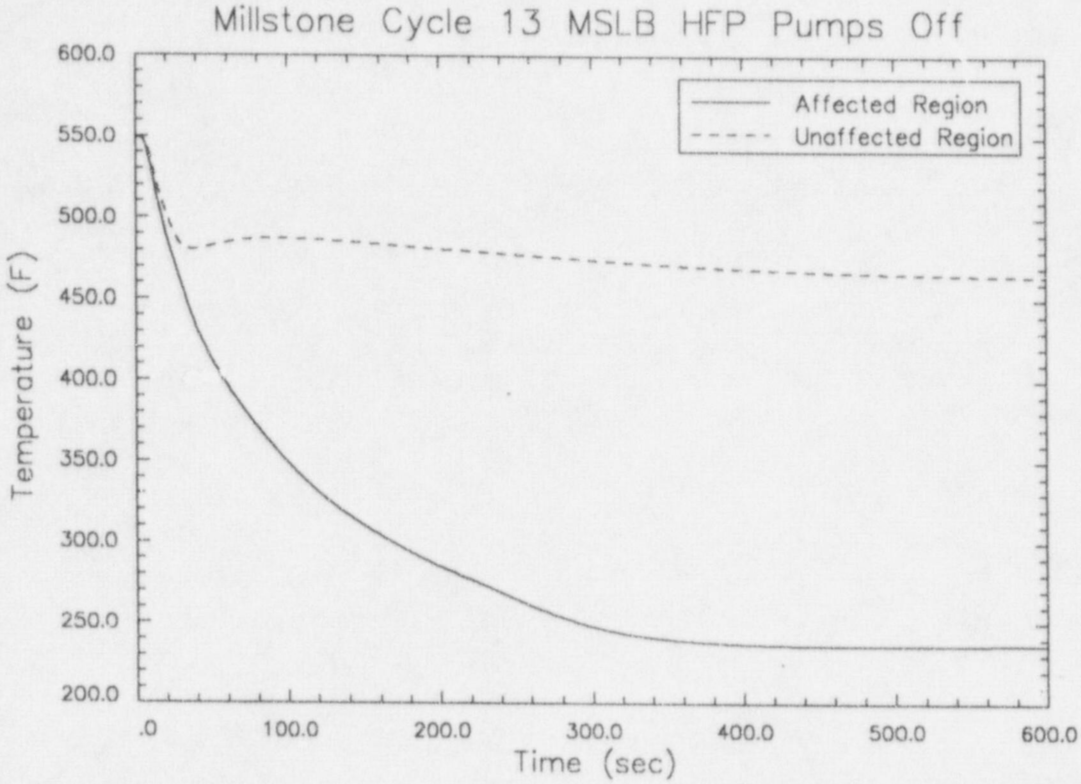

Figure 14.1.5.2-12 Core Inlet Temperatures (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

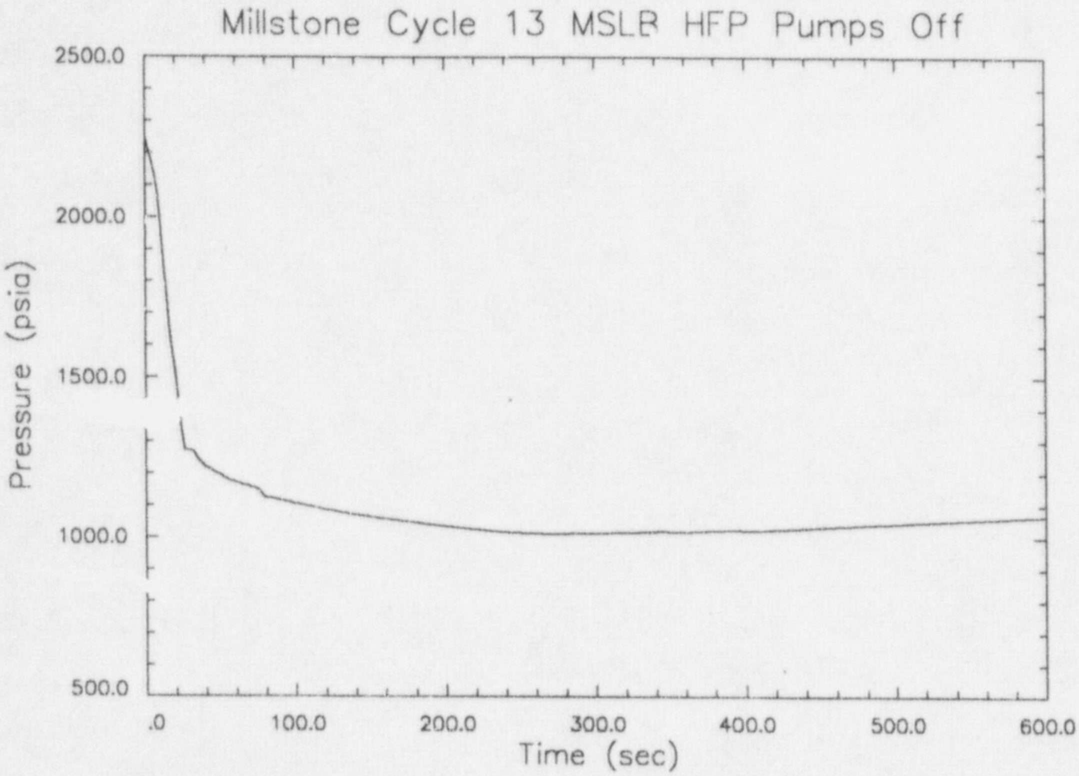

Figure 14.1.5.2-13 Pressurizer Pressure (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

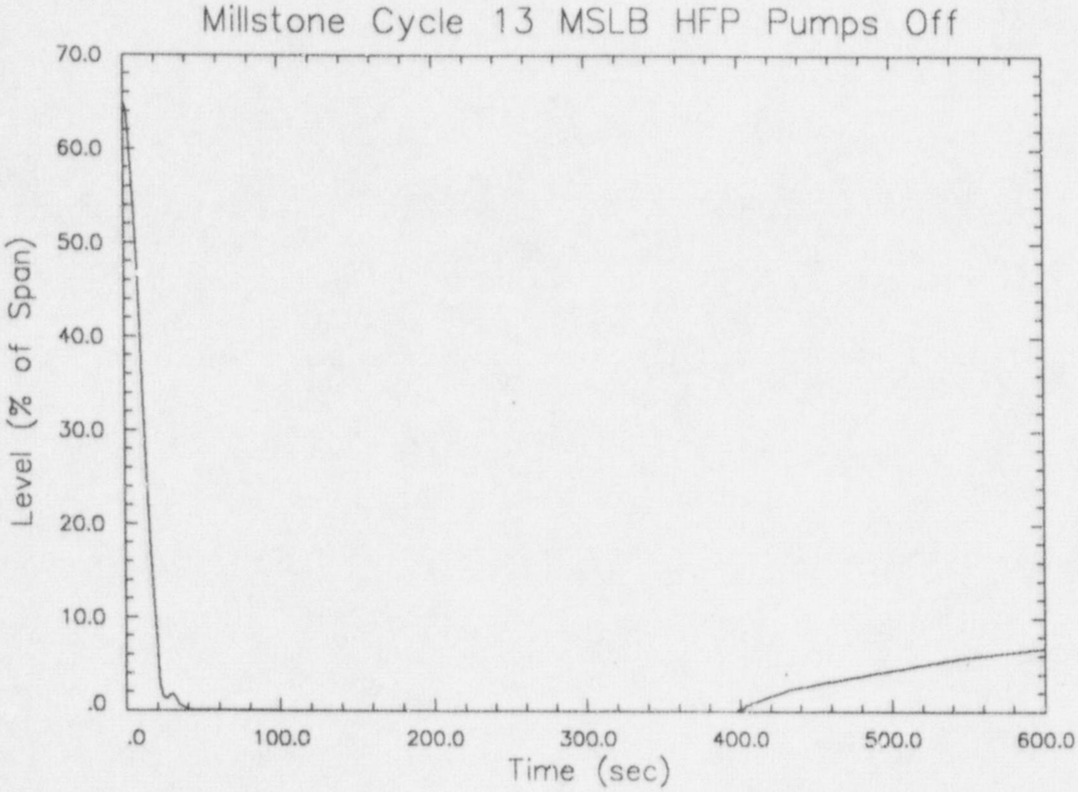

Figure 14.1.5.2-14 Pressurizer Level (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

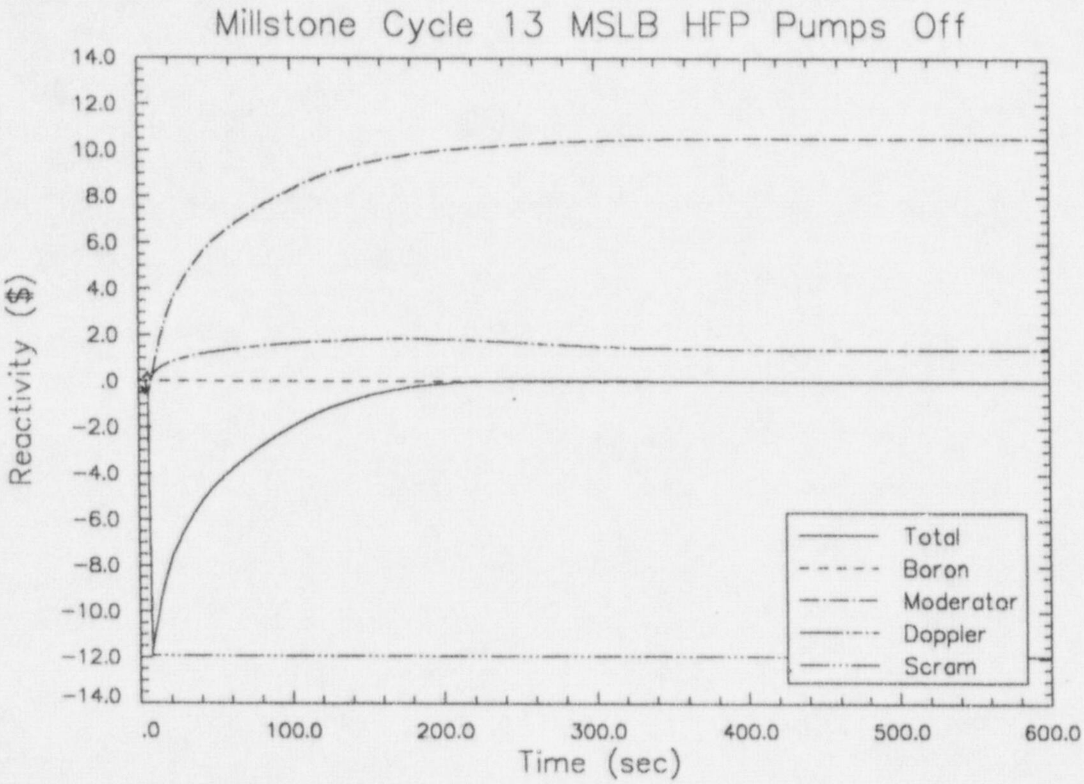

Figure 14.1.5.2-15 Reactivity Components (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

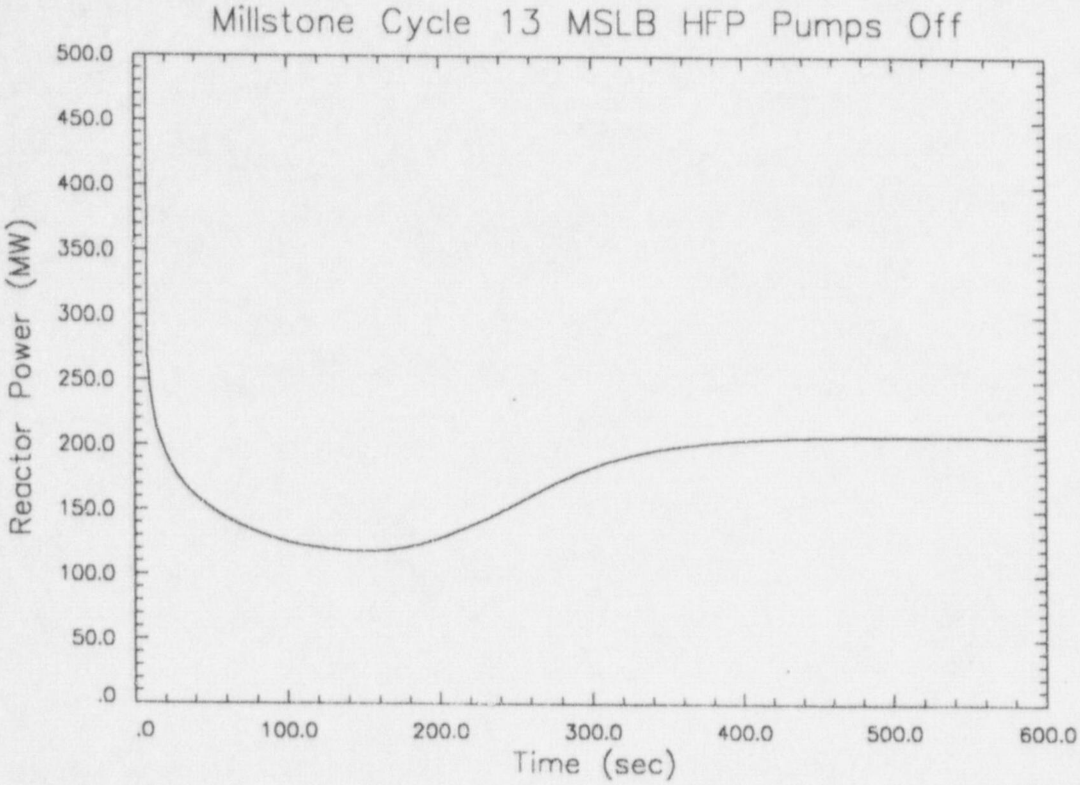

Figure 14.1.5.2-16 Reactor Power (HFP Post-Scram Steam Line Outside Containment Break with Loss of Offsite Power)

Assumption

(1) Activity in Containment available for release
100% Noble Gases
25% Iodines

(94-8 3/95)

(2) Initial Iodine Chemical Form:
91% elemental
4% organic
5% particulate

(3) Purging Occurs 5 days after initiation of LOCA

(4) Breathing Rate = 2.32×10^{-4} m^3/sec

(5) Power Level = 2700 MWt

(6) Dose Conversion Factors Reg. Guide 1.109

(7) Purge Rate = 50 ft^3/min

(8) Containment Building Volume = 1.899×10^{6} ft^3

(9) Release Point Unit 1 Stock

(10) Filter Efficiencies:
90% elemental iodine
70% organic iodine
90% particulate iodine

(11). Duration of Purge = 30 days

(12) X/Q (sec/m^3)
LPZ (0-30) days = 6.97×10^{-8}

14.8.4 Radiological Consequences of the Design Basis Accident

INSERT D

14.8.4.1 General

The DBA involves a gross release of activity from the fuel to the containment building. This section discusses the consequences of such a release.

14.8.4.2 Method of Analyses

The radiological consequences of a Design Basis LOCA at Millstone 2 were analyzed for a low wind speed condition and a high wind speed condition. These are represented by cases A and B, respectively.

Case A - Low Wind Speed Condition

This case assumes meteorological conditions exist which will give 95 percent highest X/Q values (e.g., low wind speeds). For this scenario the activity which leaks from the containment building enters the enclosure building where it is treated by the enclosure building filtration system (EBFS) before being released through the Unit 1 stack. A small percentage (1.69 percent) of the containment leakage bypasses the EBFS and is released at ground level for the entire accident (30 days). All containment leakage for the first 110 seconds is assumed to be a ground level release. This is due to the fact that it takes 110 seconds for the enclosure building to achieve negative pressure and thus assure that leakage will be into the enclosure building rather than out. All assumptions used in this analysis are given in Table 14.8.4-1.

The radiological evaluation used thyroid dose conversion factors consistent with those stated in Regulatory Guide 1.109 Rev. 1. In addition to the Staff's acceptance of these dose conversion factors in the published Safety Evaluation Report for Erie Nuclear Power Plant (NUREG-0423), NNECO offers the following justification:

(1) Dose Conversion Factors

The NRC has published a revised version of Regulatory Guide 1.109 (October, 1977) for use in Appendix I calculations. The inhalation dose conversion factors (DCFs) contained in this guide are lower than those previously used in radiological evaluations to meet reactor siting criteria of 10CFR100.

The source of the iodine DCF previously used in radiological off-site dose calculations is TID-14844 (March, 1962). They were derived using the acute intake model, a dose commitment of infinity and input parameters from ICRP-II. Parameters from ICRP-II such as effective half life, fraction of the isotope reaching the organ, and the effective energy released per disintegration were based on the best available data at that time (1959). Due to a lack of information, various conservatisms were employed in determining these parameters.

The DCF's in Regulatory Guide 1.109 were based on ICRP VI and X. They use a chronic intake model with a dose commitment extending for 50 years after intake. Credit is given for hold up of the nuclide in the lung before it reaches the thyroid. This has the effect of reducing the fraction of the nuclide reaching the organ of interest. The fraction was calculated based on information in ICRP X. The source of biological half lives was also based on ICRP X.

There is an appreciable difference between several factors upon which DCF's from TID-14844 and Regulatory Guide 1.109 were based. The first point of difference is the fraction (fa) of the nuclide which is deposited in the organ. As stated above, Reg. Guide 1.109 (Rev. 1) takes credit for retention of the iodine isotopes in the lung. This is a more realistic approach since it is expected that fewer of the short lived isotopes would be able to reach the thyroid than those that are long lived. Another factor where there are differences is the effective energy deposited in the organ.

The differences are only slight and are the result of different decay schemes used to compute this parameter. The Reg. Guide values were based on more recent data and hence are expected to be more accurate. It should also be pointed out that the Reg. Guide values are slightly higher than ICRP II values which tend to increase the DCF's. The third factor where differences occur is in the half lives. ICRP II reported a biological half life of 138 days while Reg. Guide 1.109 used a value of 100 days. Once again the Reg. Guide value was based on ICRP X. It is not clear what the reason is for the difference in this factor. The differences in biological half lives is not critical since the effective half life is directly proportional to the DCF's. The effective half lives do not significantly change. The change in biological half lives, therefore, has little effect on DCF's.

In summary, the major difference between the DCF's presented in TID-14844 and Reg. Guide 1.109 is the credit taken by the Reg. Guide for hold up of iodine in the lung. TID-14844 based its DCF's on the best information that was obtainable in 1959 whereas those presented in the Reg. Guide reflect the best information obtainable today. It is the conclusion of NNECO that the dose conversion factors in Reg. Guide 1.109 (Revision 1) are more applicable to offsite dose calculations and are therefore assumed in this dose analysis for Millstone Unit No. 2.

(2) Calculational Methods

In order to calculate offsite doses, the computer code TACT III was employed. This code evaluates the activities, and integrated doses at a site following the instantaneous or continuous release of halogens and noble gases from a control volume.

The input to the program consists of the time-dependent variables described below, the volume of the primary system, filter efficiencies, etc.

Eighteen isotopes are included in the model, including Kryptons, Xenons, and Iodines. The isotope inventory may be input, or the program will calculate it based on TID source terms; decay is evaluated, as well as filtration. The primary containment leak rate, atmospheric dispersion factors, and breathing rates may vary with time, at the option of the user. Site dose calculations use the semi-infinite cloud dose models suggested by Regulatory Guide 1.4.

Case B - High Wind Speed Conditions

An analysis was performed to determine the effect on the enclosure building of high wind speeds. The wind speed at which the enclosure building will begin to exfiltrate is one in which the corresponding wind velocity pressure is greater than the enclosure building negative pressure. The effect of wind on the enclosure building is discussed in Section 6.7.1.2.

The enclosure building filtration region (EBFR) design negative pressure is 0.25 in w.g. The wind velocity corresponding to a velocity pressure equivalent to this EBFR design pressure is 25 mph. Above this velocity displacement of the enclosure building atmosphere with outside air would begin. However, for conservatism it is assumed that this displacement would begin with a 23 mph wind.

The model formulated for site boundary and control room doses is as follows:

(1) The wind is from the (plant) North direction.

(2) The high wind condition exists for the first 36 hours following the incident (5 percent of thirty days).

(3) Only those areas of the EBFR above grade are exposed to the wind effects. Therefore, only the enclosure building structure is subject to air displacement due to wind effects.

(4) The amount of postaccident containment leakage is assumed as 0.5 volume percent per day per technical specification 3.6.1.

(5) The amount of exfiltration based conservatively on a 30 mph wind is less than 10 percent of the EBFS exhaust capabilities assuming only one fan operating. Therefore, 10 percent of the EBFR atmosphere is conservatively assumed to be an unfiltered ground release.

All other assumptions and methodologies are the same for Case A.

14.8.4.3 Dose Calculations

14.8.4.3.1 Thyroid Doses and Whole Body Exposures

The results of the calculated doses for Cases A and B are shown in Table 14.8.4-2 and are within the limits of 10CFR100.

14.8.4.3.2 Control Room Habitability

As a result of Three Mile Island (TMI) Action Plan Item III.D.3.4, the potential radiological doses to the MP-2 control room operators have been reevaluated. The analysis is based on the control room assumptions and meteorological parameters given in Tables 14.8.4-3 and 14.8.4-4.

The control room is designed to be occupied for the duration of the accident (30 days). Two (2) basic sources of radiation have been evaluated. They were: (1) direct dose from sources outside the control room, and (2) the dose received from airborne activity which enters the control room. The analyses ensure that the operators will be adequately protected from all sources of radiation.

The radiation design objective for the control room walls is to limit the whole body dose to personnel inside the control room to less than 5 rem during any DBA. The external sources considered in the shielding evaluation are: (1) containment, (2) enclosure/reactor building, (3) filtration systems, and (4) piping sources. The affect of external sources from DBA's at Millstone Units 1 and 2 were evaluated in the shielding analysis.

Because the containment as well as other sources at Millstone Unit No. 3 are separated from the Unit 2 control room by a relatively large distance as well as other structures, a shielding analysis from a Unit 3 accident was determined to be unnecessary.

The assumptions used in the shielding evaluation are listed in Table 14.8.4-5. The resulting doses from the shielding analysis are given in Table 14.8.4-6.

An EBFS signal from Millstone 2 initiates control room isolation and after a 42-second delay the control room emergency ventilation system will be operating at 2,500 cfm. Isolation of the control room will be complete within 5 seconds after reception of an isolation signal. During this 5 second interval the normal outside air flowrate through the damper was assumed to vary linearly from 2,000 cfm to 0 cfm. The MP2 control room emergency ventilation system recirculates air from inside the control room and filters the air through high efficiency particulate air (HEPA) and charcoal filters before returning it to the control room.

Two separate cases were analyzed for the design basis LOCAs at MP2. These cases are representative of high wind speed and low wind speed conditions. As described in Section 14.8.4.2 (Case B) it has been assumed that the high wind speed condition exists for 36 hours after the LOCA and 10 percent of the activity in the enclosure building bypasses the EBFS resulting in a ground level release to the environment. Displacement of the enclosure building atmosphere would begin at wind speeds above 25 mph. However, for conservatism, it is assumed that this displacement would begin with a wind speed of 23 mph. The low wind speed conditions used assumptions consistent with those for Case 1 and given in Table 14.8.4-1, except for the 1.69 percent bypass leakage. For the control room analysis, the bypass leakage was reevaluated without assuming a seismic event and was determined to be negligible.

The calculated whole body, beta and thyroid doses are presented in Table 14.8.4-8 and are below the General Design Criterion 19 limits. For accidents at either Unit 1 or Unit 3 (and for several Unit 2 accidents not involving a signal to automatically activate the EBFS) the Unit 2 control room will not automatically isolate and must rely on a high radiation signal to perform the isolation.

Under normal conditions, air is provided to the control room operators by the air intake duct. The duct is equipped with redundant radiation monitors which will automatically isolate the control room upon a high radiation signal. Approximately 23.1 seconds of continuous unfiltered air intake is assumed to enter the control room subsequent to isolation by a signal from either radiation monitor. After 42 seconds the control room emergency ventilation system will be operating. Control room air will be recirculated through HEPA and charcoal filters.

Since other operating reactors are located on the site, an assessment was made of the habitability of the Millstone 2 control room subsequent to an assumed design basis LOCA at either Millstone 1 or 3. Assumptions used for each of these plants are given in their respective FSARs. Because of the close proximity of the Millstone 1 turbine building with respect to the Unit 2 control room intake duct, an assessment was also made of a steam line break (SLB) accident at Unit 1 on the Unit 2 operators. The assumptions used in this accident are given in Table 14.8.4-7.

The calculated Millstone 2 control room dose from Millstone 1 and Millstone 3 released are presented in Table 14.8.4-8.

14.8.4.4 Conclusions

It is concluded that the exclusion boundary and low population zone (LPZ) guideline dose values of 10 CFR 100 would not be exceeded even for the DBA.

The following pages are "Insert D," Sections 14.8.4.1 through 14.8.4.5:

14.8.4.1 General

A LOCA would increase the pressure in the containment resulting in a containment isolation and initiation of the ECCS and containment spray systems. A SIAS signal automatically starts the Enclosure Building Filtration System (EBFS) which maintains a negative pressure within the enclosure building during accident conditions. The nuclide inventory assumed to be initially available for release from within containment consists of 100 percent of the core noble gasses and 25% of the iodines, as described in Regulatory Guide 1.4. A SIAS also isolates the control room by closing the fresh air dampers within 5 seconds. Within 10 minutes after control room isolation, the control room emergency ventilation (CREV) starts. CREV recirculates air within the control room through a 90 percent charcoal filter at 2,500 cfm ($\pm$ 10%) to remove iodines from the control room envelope.

The radiological consequences of a Design Basis LOCA at Millstone 2 were analyzed for a low and high wind speed condition. The low wind speed case was found to bound the high wind speed case. Therefore only the low wind speed case will be presented here.

14.8.4.2 Release Pathways

The release pathways to the environment subsequent to a LOCA are leakages from containment and the enclosure building , which are collected and processed by EBFS and leakages from containment and the RWST which bypass EBFS.

<u>Containment Leakage</u>
The containment is assumed to leak at the design leak rate for 24 hours after the accident. After 24 hours, since the pressure has been decreased significantly, Regulatory Guide 1.4 allows for the leak rate to be reduced to one-half the design leakage rate.

All containment leakage for the first 110 seconds is assumed to bypass EBFS and is released through the MP-2 vent. This is due to the fact that it takes 110 seconds for EBFS to achieve the required negative pressure in the enclosure building, thereby ensuring that leakage will be into the enclosure building rather than out.

EBFS collects most of the containment leakage and processes it through HEPA and charcoal filters and releases it up the Unit 1 stack. All containment leakage is collected and filtered by EBFS except for the small amount that is assumed to bypass EBFS and is released out the MP-2 vent.

Credit is taken for iodine removal due to containment sprays. The sprays are effective within 3 minutes post-LOCA and are assumed to shutoff 30 minutes later.

<u>ESF System Leakage Pathway</u>
Post-accident radioactive releases from the ESF system are derived from fluid leakages assumed during recirculation of the containment sump water through systems located

outside containment. The nuclide inventory assumed to be available for release from this pathway consists of 50% of the core iodines. The quantity of leakage is based on the assumption that the ESF equipment leaks at twice the maximum expected operational leak rate and that 10 percent of the iodine nuclides contained in the leakage fluid become airborne in the enclosure building. The nuclides which become airborne are collected and released to the environment through EBFS to the Unit 1 stack.

RWST Backleakage Pathway

Post-accident radioactive releases from the ECCS system are a result of ECCS subsystems containing recirculated sump fluid backleaking to the RWST. The backflow rate to the RWST, as a result of isolation valve leakage, is pre-defined and time dependent. Due to this time dependency, the contaminated sump fluid from backleakage does not enter into the RWST until 25.45 hours post-LOCA. Since the RWST is vented to atmosphere, the release is a result of the breathing rate of the RWST due to solar heating. The EAB dose is a 2 hour dose therefore it is not affected by backleakage.

14.8.4.3 Control Room Habitability

The radiation design objective of the control room is to limit the dose to personnel inside the control room to 5 rem whole body, or its equivalent, during a DBA. The potential radiation dose to a control room operator is evaluated for the LOCA. The analysis is based on the assumptions and meteorological parameters (X/Q values) given in Tables 14.8.4-3 and 14.8.4-4.

The control room is designed to be continuously occupied for the duration of the accident, 30 days. Two basic sources of radiation have been evaluated: leakage of airborne activity into the control room from sources described in 14.8.4.2 and direct dose from sources outside the control room. The control room shielding serves to protect the operators from direct radiation due to the passing cloud of radioactive effluent assumed to have leaked from containment, enclosure building and the RWST. The control room walls also provide shielding protection for radiation emanating from the CREV filters and containment shine.

A SIAS from Millstone 2 initiates control room isolation within 5 seconds by securing the fresh air intake dampers. Within 10 minutes CREV is operation recirculating air in the control room envelope through 90% efficient charcoal filters to remove radioactive iodines from the atmosphere. The calculated thyroid, whole body and skin doses from a Millstone 2 LOCA are presented in Table 14.8.4-8 and are below the General Design Criteria 19 limits.

Normally outside air is provided to the control room via an air intake duct, which is equipped with redundant radiation monitors. These radiation monitors isolate the control room within 10 seconds after a high radiation signal. This method of isolation will occur after a Millstone 3 LOCA. The calculated thyroid, whole body and skin doses from a Millstone 3 LOCA are presented in Table 14.8.4-8 and are below the General Design Criteria 19 limits.

14.8.4.4 Dose Computation

The radiological off-site dose consequences resulting from a postulated Millstone 2 LOCA are reported in Table 14.8.4-2. The off-site dose analysis show that the consequences to the EAB (0 - 2 hr) and LPZ (0 - 30 day) are less than the limits of 300 rem thyroid and 25 rem whole body as specified in 10CFR100. The assumptions used to perform the radiological analysis are summarized in Table 14.8.4-1.

14.8.4.5 Conclusion

Analysis shows that the off-site radiological consequences are within 10CFR100 guidelines and the control room radiological consequences are within GDC19 criteria.

TABLE 14.8.4.-1

5/90

LOSS OF COOLANT ACCIDENT (OFFSITE ASSUMPTIONS)

Assumption

(1) Core power level = ~~2700~~ 2754 MWt

~~(2) Operating time = 2 yr.~~

2 (~~3~~) Core released fractions: Noble gases = 100%, Iodines = 25%

3 (~~4~~) ~~Halogens~~ Iodine composition: 91% elemental
4% organic
5% particulate

4 (~~5~~) Reactor building leak rate: .5%/day ≤ 24 hrs.
.25%/day > 24 hrs.

5 (~~6~~) Enclosure Building Filtration System (EBFS) charcoal filter efficiencies:

90% for elemental
70% for organic
90% for particulate

6 (~~7~~) Bypass leakage fraction = ~~1.69~~ 1.7%

7 (~~8~~) EBFS ~~negative pressure initiation~~ Drawdown Time = 110 seconds

8 (~~9~~). X/Qs:

Location	Time Period	Elevated	Ground Release
~~SB~~ EAB	(0-2) hrs.	~~1.03×10^{-4}~~ 1.00 E-04	~~5.39×10^{-4}~~ 3.66 E-04
LPZ	(0-4) hrs.	~~3.41×10^{-5}~~ 2.69 E-05	~~2.12×10^{-5}~~ 4.80 E-05
	(4-8) hrs.	~~1.77×10^{-6}~~ 3.04 E-06	~~2.12×10^{-5}~~ 2.31 E-05
	(8-24) hrs.	~~2.62×10^{-7}~~ 2.17 E-06	~~4.76×10^{-6}~~ 1.60 E-05
	(24-96) hrs.	~~1.57×10^{-7}~~ 1.04 E-06	~~3.04×10^{-6}~~ 7.25 E-06
	(96-720) hrs.	~~6.97×10^{-8}~~ 3.63 E-07	~~1.3×10^{-6}~~ 2.32 E-06

9 (~~10~~) Thyroid Inhalation DCFs from ~~Reg. Guide 1.109~~ ICRP30

10 (~~11~~) Containment Free Air Volume = ~~1.899×10^{6} ft^3~~ 1.900 E+06 ft^3

11 (~~12~~) Breathing Rates (0-8) hr. = 3.47×10^{-4} m^3/sec

(8-24) hr. = 1.75×10^{-4} m^3/sec

(24-720) hr. = 2.32×10^{-4} m^3/sec

INSERT A

INSERT A

Assumption

12) Release Points: Filtered - MP1 Stack
Bypass - MP-2 Vent

13) Containment Spray Removal Coefficients: elemental = 3.827 per hour
particulate = 1.707 per hour

14) Containment Spray Effectiveness Time: 3 - 33 minutes post LOCA

15) ESF Leakage: 24 gallons per hour

16) ESF leakage begins at 25 minutes post LOCA

17) Sump Volume: 2.86E+5 gallons

18) RWST Backleakage begins at 25.45 hours

19) RWST Backleakage amount: 0.01 - 0.19 gpm

20) Iodine DF: 100

TABLE 14.8.4-2

5/90

SUMMARY OF DOSES FOR LOSS OF COOLANT ACCIDENT

DOSE (rems)

ORGAN	CASE A			CASE B
	SITE BOUNDARY	LPZ		SITE BOUNDARY
		HYDROGEN PURGE NOT INCLUDED	HYDROGEN PURGE NOT INCLUDED	
Thyroid	151	56.3	6.1	12
Whole Body	3.8	1.4	1.4	-

	Thyroid	Whole Body
EAB	4.58 E+01	2.46 E+00
LPZ	2.19 E+01	9.41 E-01

TABLE 14.8.4-3

5/90

LOSS OF COOLANT ACCIDENT → (CONTROL ROOM ASSUMPTIONS)

1. Control Room Volume = ~~7.7 x 10^4 ft^3~~ 3.565 E+04 ft^3
2. Control Room Unfiltered Inleakage in Recirculation Mode = ~~100~~ 130 cfm
3. Control Room Normal Makeup Air Flowrate = ~~2,000~~ 800 cfm
4. Time from ~~Start of Accident~~ MP-2 LOCA to Time when Dampers Close = 5 sec.
   Time from Control Room high radiation to Time when Dampers Close = 10 sec.
5. Time when Control Room Emergency Ventilation System Operating at Full ~~Speed~~ Capacity = ~~42 sec.~~ 10 min.
6. Control Room Emergency Ventilation (Filtration) System Flowrate = ~~2,500~~ 2250 cfm
7. Charcoal Iodine Filter Efficiency = 90 percent

***NOTES**: For the analysis of assumed LOCA at MP2. For accidents at either Unit 1 or 3 damper closure time is 23.1 sec. to account for monitor response and damper closure time. Other Unit 1 and Unit 3 assumptions are given in the following references.

REFERENCES:

1. Unit 1 - W. G. Counsil, 1981 (NUSCO) Letter to D. M. Crutchfield (NRC), transmitting Millstone Nuclear Power Station Unit 1, Systematic Evaluation Program, Section XV, Topics: Design Basis Events.
2. Unit 3 - Millstone Unit No. 3 FSAR.

8. Bypass Leakay Amount = 11 cc/hr
9. Control Room Shielding:

North Wall:	2' concrete
West Wall:	1.5' Concrete except 8' section which is 2' concrete
South Wall:	24.5' of 1' concrete except glass wall 86.75' long
East Wall:	2' Concrete
Roof:	2' Concrete
Floor:	1' concrete

TABLE 14.8.4 -4

ATMOSPHERIC DISPERSION DATA FOR MILLSTONE UNIT 2 CONTROL ROOM

5/90

MP-2 CONTROL ROOM X/Qs (sec./m^3)

Release Point	**MP-1**	**MP-2**	**MP-3**
a. Ground Level			
(0-8) hr.	4.43×10^{-2}	2.69×10^{-3}	4.78×10^{-4}
(8-24) hr.	3.05×10^{-2}	1.90×10^{-3}	3.29×10^{-4}
(1-4) day	1.20×10^{-2}	7.56×10^{-4}	1.20×10^{-4}
(4-30) day	3.68×10^{-3}	2.30×10^{-4}	2.26×10^{-5}
(0-24) hr.	N/A	9.14×10^{-5}	N/A
(24-36) hr.	N/A	9.14×10^{-5}	N/A
b. Elevated Release - Unit 1 Stack			
(0-8) hr.	2.0×10^{-9}	2.0×10^{-9}	N/A
(8-24) hr.	1.0×10^{-9}	1.0×10^{-9}	N/A
(1-4) day	2.5×10^{-10}	2.51×10^{-10}	N/A
(4-30) day	5.01×10^{-11}	5.01×10^{-11}	N/A

INSERT B

INSERT B

Release Point

MP-2 RWST:	
0 - 8 HR:	1.87E-3
8 - 24 HR:	1.20E-3
1 - 4 DAYS:	3.83E-4
4 - 30 DAYS:	5.81E-5

MP-2 Enclosure Building:	
0 - 8 HR:	5.46E-3
8 - 24 HR:	3.45E-3
1 - 4 DAYS:	1.27E-3
4 - 30 DAYS:	3.98E-4

MP-2 Vent	
0 - 8 HR:	2.92E-3
8 - 24 HR:	1.89E-3
1 - 4 DAYS:	6.18E-4
4 - 30 DAYS:	1.05E-4

MP-1 Stack	
0 - 4 hrs:	2.51E-4
4 - 8 hrs:	1.96E-5
8 - 24 hrs:	5.46E-6
24 - 96 hrs:	2.06E-7
96 - 720 hrs:	2.58E-9

MP-3 Containment	
0 - 8 hr:	9.19E-4
8 - 24 hr:	5.29E-4
24 - 96 hr:	1.65E-4
96 - 720 hr:	2.75E-5

MP-3 Ventilation Vent	
0 - 8 hr:	1.25E-3
8 - 24 hr:	7.49E-4
24 - 96 hr:	2.46E-4
96 - 720 hr:	4.08E-5

MP-3 MSVB	
0 - 8 hr:	2.47E-3
8 - 24 hr:	1.48E-3
24 - 96 hr:	4.87E-4

96 - 720 hr:	8.18E-5

MP-3 ESF Bldg

0 - 8 hr:	2.08E-3
8 - 24 hr:	1.18E-3
24 - 96 hr:	3.88E-4
96 - 720 hr:	6.12E-5

TABLE 14.8.4-5

ASSUMPTIONS USED TO CALCULATE DOSES FROM EXTERNAL SOURCES

5/90

1. Assumptions Used to Calculate Dose from Containment Source:

   a. Source Term: 100 percent core noble gas inventory released to containment

      50 percent core iodine inventory released to containment.

   b. Source assumed to be uniformly distributed to containment free air volume.

   c. Containment Volumes:

      Unit 1 = 2.568×10^5 ft^3
      Unit 2 = 1.899×10^6 ft^3

   d. Containment Concrete Wall Thickness:

      Unit 1 = 5'
      Unit 2 = 3.75' (walls), 3' (done)

   e. Control Room Concrete Wall Thickness:

      For wall facing Unit 1 containment = 3'-6"
      For wall facing Unit 2 containment = 2'-0"

2. Assumptions Used to Calculate Dose from Enclosure/Reactor Building Source:

   a. Containment Leak Rate:

      Unit 1 = 1.2 percent/day
      Unit 2 = 0.5 percent/day

   b. Volume of Enclosure/Reactor Building

      Unit 1 Reactor Building = 1.728×10^6 ft^3
      Unit 2 Enclosure Building = 1.44×10^6 ft^3

   c. Ventilation Rate

      Unit 1 Reactor Building = 100 percent/day
      Unit 2 Enclosure Building = 6,000 cfm

   d. Control Room Wall Thickness:

      Control Room Wall Facing Enclosure Building = 2'-0"
      Control Room Wall Facing Reactor Building = 3'-6"

TABLE 14.8.4-5

ASSUMPTIONS USED TO CALCULATE DOSES FROM EXTERNAL SOURCES

5/90

3. Assumptions Used to Calculate Dose from Filtration Systems:

   a. Filtration Systems Considered:

      - Millstone Unit 1 Standby Gas Treatment System (SGTS)
      - Millstone Unit 2 Enclosure Building Filtration System (EBFS)
      - Millstone Unit 2 Control Room Filters

   b. Thickness of Concrete Between Control Room and:

      - Millstone Unit 1 SGTS = 9'-0"
      - Millstone Unit 2 EBFS = 18'-0"
      - Millstone Unit 2 Control Room Filter = 2'-0"

4. Assumptions Used to Calculate Dose from Overhead Plume:

   a. Millstone Unit 2 Control Room Ceiling Concrete Thickness = 2'-0"

   b. Filtration System Filter Efficiencies:

      - SGTS = 90 percent (all forms of iodine)
      - EBFS = 90 percent (elemental and particulate iodine)
        = 70 percent (organic iodine)

   c. Plume Centerline X/Qs:

   (0-8) hr. = 4.84×10^{-3} sec/m^3

   (8-24) hr. = 4.19×10^{-4} sec/m^3

   (1-4) day = 1.65×10^{-4} sec/m^3

   (4-30) day = 9.92×10^{-3}

5. Assumptions Used to Calculate Dose from Piping Sources:

   a. Sources in the vicinity of the Unit 2 Control Room = Unit 1 core spray line.

   b. Source Term: 50 percent core iodine inventory
      1 percent solid fission products

   c. Concrete thickness of Control Room wall = 3'-6"

TABLE 14.8.4-6

SUMMARY OF DOSES FROM EXTERNAL SOURCES (1)

5/90

30-DAY DOSE (MREMS) TO MILLSTONE UNIT 2
CONTROL ROOM OPERATORS

ACCIDENT TYPE	OVERHEAD PLUME	DIRECT FROM PRIMARY CONT.	SECONDARY CONTAINMENT	UNIT 1 CORE SPRAY LINE	TOTAL
Millstone Unit 1 LOCA	7.440×10^{0}	3.288×10^{-1}	3.493×10^{0}	4.846×10^{1}	5.972×10^{1}
Millstone Unit 2 LOCA	3.424×10^{0}	4.044×10^{2}	5.214×10^{2}	-	9.292×10^{2}

TABLE 14.8.4-7

ASSUMPTIONS USED IN A MILLSTONE UNIT 1 MAIN STREAMLINE BREAK 5/90

1. Mass of Coolant/Steam Released = 1.753×10^7 gm

2. Coolant Concentration:

   DEQ (I-131) = 0.2 micro Ci/gm
   Noble Gas = 100/E micro Ci/gm

3. Duration of Release = 5.5 sec.

4. MP-2 Control Room Normal Ventilation System Flowrate = 2,000 ft^3/min. (Note: This flowrate assumed for entire 5.5 sec. since monitor response and damper closure = 23.1 sec.)

5. Time for MP-2 Control Room Ventilation System to Operate at Full Speed = 42 sec.

6. MP-2 Recirculation System Flowrate = 2,500 cfm

7. Time When Operators are Assumed to Don Scott Air Paks = 20 minutes

8. Effectiveness of Scott Air Paks = 10,000

9. Time When Operators are Assumed to Purge the Control Room = 30 min.

10. Purge Time Span = 30 min. to 4 hrs.

11. Purge Flowrate = 16,500

12. Control Room X/Q (0-8 hrs.) = 4.43×10^{-2} sec/m^3

TABLE 14.8.4-~~8~~ 5

5/90

DOSE TO MILLSTONE UNIT 2 CONTROL ROOM OPERATORS

Release	Thyroid Dose (Rems)	Whole Body[1] Gamma Dose (Rems)	Beta Skin Dose (Rems)
~~Millstone 1 (LOCA)[2]~~	~~2.27×10^{1}~~	~~7.58×10^{-2}~~	~~2.45×10^{-1}~~
~~Millstone 1 (MSLB)~~	~~2.62×10^{1}~~	~~4.23×10^{-1}~~	~~5.87×10^{0}~~
Millstone 2 (LOCA) ~~Low Wind Speed Conditions)~~	~~9.25×10^{0}~~ 2.58×10^{1}	~~9.45×10^{-1}~~ 7.18×10^{-1}	~~2.05×10^{-1}~~ 2.29×10^{0}
~~Millstone 2 (LOCA High Wind Speed Conditions)~~	~~2.62×10^{1}~~	~~9.43×10^{-1}~~	~~2.20×10^{-1}~~
Millstone 3 (LOCA)	~~2.48×10^{1}~~ 2.15×10^{1}	~~2.09×10^{-1}~~ 1.484×10^{0}	~~2.87×10^{0}~~ 1.40×10^{1}

NOTES: (1) Dose through wall and ceiling from external sources included.

~~(2) A time of 8 hours is assumed for Turbine Building Exhaust to be initiated.~~

Docket No. 50-336
B17413

Attachment 6

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Control Room Ventilation System
Requests for Additional Information

September 1998

Proposed Revision to Technical Specifications
Control Room Ventilation System
Requests for Additional Information

In a letter dated November 6, 1997,[1] the NRC requested the following additional information.

Question 1

Calculation M2CRM2-01156-R2 evaluates the control room doses from a Design Basis Accident Loss-of-Coolant Accident. A containment bypass fraction of 4.025E-7 is determined on page 9 of the calculation. This value was subsequently used to determine the containment bypass release rate of 8.388E-11 hr^{-1} for the period T = 110 seconds to T = 24 hours. Millstone 2 Technical Specification 3.6.1.2 establishes a bypass limit of 0.017 x LA (LA = 0.5%/day). The staff believes that the appropriate release rate to be:

0.5%/day x 1/100 x 0.017 x day/24 hr = 3.54E-6 hr^{-1}

The methodology used in the Millstone 2 calculation differs from the corresponding evaluation in Millstone 3 calculation M2CRM3-01146-R2, which developed a bypass leak rate of 0.0278%/day or 1.16E-5 hr^{-1}. A comparison of the staff exclusion boundary area and low-population zone analyses results with those documented in the Updated Final Safety Analysis Report suggests that the suspect release rate was not used in the evaluation of offsite doses.

Justify the use of the 4.025E-7 value when the containment could be considered operable with a bypass fraction as high as 0.017. If the bypass fraction assumption cannot be supported, the affected calculation should be revised and provided to the staff.

Reply

The current licensing basis for Millstone Unit No. 2 is presented below. (It is important to note that this information is based on historical documentation. It may not accurately reflect the current configuration of Millstone Unit No. 2. However, the conclusion reached is correct for the current Millstone Unit No. 2 configuration.)

Based upon the Atomic Energy Commission (AEC) requirements set in the

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[1] D. G. McDonald Jr. letter to NNECO, "Request for Additional Information Relating to the Control Room Ventilation System, Millstone Nuclear Power Station, Unit No. 2 (TAC NO. M92879)," dated November 6, 1997.

May 10, 1974,[2] Safety Evaluation Report (SER) and the correspondence leading up to the SER, the Control Dose assumptions of negligible bypass leakage ($6.44x10^{-6}$ scfm) were not questioned while a bypass leakage of 1.7% of the daily containment leakage rate was mandated for calculating the 10 CFR 100 offsite doses. The licensing basis for calculating the control room dose following an event is therefore $6.44x10^{-6}$ scfm.

Background

In a letter dated December 29, 1972,[3] the AEC sent the Millstone Point Company a Request for Additional Information, in the form of a series of questions, to support the AEC review of the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). AEC Questions 5.39 and 14.1 requested the following information:

5.39 Provide a detailed evaluation of penetration through-line leakage that can bypass the enclosure building. Detail each leakage path and describe provisions made for initial and periodic testing of these penetrations to measure leakage. Provide proposed Technical Specification limits for allowable leakage through these penetrations.

14.1 Discuss the dose calculational model used in Section 14.18.3.3 of the FSAR to determine the gamma and beta doses from iodine and noble gases inside the control room and report these doses separately. Differentiate the leakage and the doses associated with the leakage which bypasses the enclosure building from the leakage and the doses associated with the leakage which is filtered and passed out of the stack.

In a letter dated February 16, 1973,[4] the Millstone Point Company sent Amendment 15 to the MP2 License Application to the AEC. A response to questions 5.39 and 14.1 were included in this amendment:

5.39 - Response

To evaluate the through-line leakage that can bypass the enclosure building filtration region (EBFR), the fluid systems penetrating containment are categorized as follows:

1. Piping System open to the containment post-incident atmosphere.

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[2] O. D. Parr (AEC) letter to The Millstone Point Company, Safety Evaluation for Millstone Nuclear Power Station, Unit 2, dated May 10, 1974.

[3] K. R. Goller (AEC) letter to The Millstone Point Company, Request for additional Information, dated December 29, 1972.

[4] The Millstone Point Company letter to AEC, "Amendment No. 15 To Licensee Application In Docket No. 50-336," dated February 16, 1972.

2. Piping Systems which are closed and therefore not exposed to the containment post-incident atmosphere.

The following assumptions are made to postulate the maximum hypothetical conditions:

1. There is either a seismic occurrence and all Seismic Class 2 lines are broken or either there is no seismic occurrence and all Seismic Class 2 lines are intact.

2. The single failure applies to Seismic Class 1 components only.

....

The line from the normal sump to the aerated drain tank could provide a potential path for bypass leakage. Assuming the leakage through the valve is proportional to the square root of the pressure differential, the maximum leakage through the two (2) series containment is 3 cc/hr at containment post-incident design conditions. After approximately one (1) hour the containment pressure is less than 10 psig and the leakage rate is less than 1.0 cc/hr. This valve leakage is diluted in the aerated waste system.

....

The refueling water purification penetrations could provide a potential path of bypass leakage into the auxiliary building. The maximum leakage is approximately 4.0 cc/hr through each penetration during the first hour. After this the maximum leakage is approximately 1.0 cc/hr. Assuming normal system alignment (Figure 9.5-1), the leakage is diluted and contained within the closed process piping.

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From the preceding analysis, only the leakage through the normal sump to aerated drain tank and refueling water purification penetrations may be considered as through line leakage. The maximum potential leakage rate is 11.0 cc/hr during the first hour and less than 3.0 cc/hr thereafter. All other leakage is either contained within the process system or within the EBFR.

The provisions for initial and periodic leak testing of containment penetrations and maximum allowable leakage are specified in Subsection 15.4.5. and Table 5.2-11 of the FSAR.

14.1 - Response

The iodine and noble gas concentrations in the control room are calculated as a function of time. The thyroid and whole body doses for exposure to a semi-infinite cloud at these concentrations are calculated using the inhalation model (dose conversion factors tables 14.1-1 and 14.1-2) of TID14844, and the semi-infinite cloud model given in safety guide 4, and in Meteorology and Atomic Energy--1968. Additional information is given in Subsection 14.18.3.3.

....

The thyroid and whole body doses inside the control room previously stated in Subsection 14.18.3.3 were based isolating the control room 20 minutes after incident. The reduced doses given above results from design change in which control room is isolated on a EBFS or AEAS signal.

All leakage from the containment will be into the enclosure building. Since the enclosure building is kept at a slightly negative pres[illegible] all leakage from the containment will be exhausted through EBFS filters to the Unit 1 stack. Thus, the leakage and doses previously given are from the stack releases.

In a letter dated May 21, 1973,[5] the AEC sent the Millstone Point Company another Request for Additional Information, also in the form of a series of questions, to support the AEC review of the Millstone Unit No. 2 FSAR. Questions 6.16.1, 6.16.2, 6.16.3, and 6.16.4 concerned the possible leakage of radioactive materials from containment:

6.16 Provide the following information concerning the containment leakage.

6.16.1 List each potential leakage path which presents a potential pathway for release of radioactivity from the containment atmosphere (1) directly to the external atmosphere, (2) to the auxiliary building, and (3) to the enclosure building filtration area.

6.16.2 Provide an estimate of the fractions of the total amount of containment leakage which can bypass the enclosure building filtration area and be released (1) directly to the atmosphere and (2) to the auxiliary building. Describe the tests and test frequencies that will be used to detect and limit these leakage fractions and the total containment leakage.

6.16.3 Provide the Technical Specification limit that assures that the leakage through potential leak paths that bypass the enclosure building filtration

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[5] K. R. Goller (AEC) letter to The Millstone Point Company, Request for additional Information, dated May 21, 1973.

area will not exceed the fraction of total leakage assumed in the dose calculations.

6.16.4 In the response to Item 5.39 of Amendment 15, a seismic occurrence is not considered in the evaluation of through-line leakage that can bypass the enclosure building. Assuming a seismic occurrence, (1) describe any non-Category I system which could become open to the containment atmosphere, and terminate outside the enclosure building; (2) provide the information requested in 6.16.1 through 6.16.3 above for these additional pathways.

In a letter dated June 27, 1973,[6] the Millstone Point Company sent Amendment 16 to the MP2 License Application to the AEC. Responses to questions 6.16.1 through 6.16.4 were included in this amendment:

6.16.1 - Response

The detailed evaluation of penetration potential through-line leakage that could bypass the enclosure building filtration region (EBFR) was provided in the response to AEC Question 5.39 in Amendment 15. However, to supplement that response, containment potential leakage paths are again evaluated, as requested, on an individual case basis.

The following is the basis formulated for the analysis:

1. There is no seismic event, therefore all systems remain intact.

2. The model for liquid valve seat leakage through a closed valve is 2.0 cc/hr. per inch nominal valve diameter and 1.0 cc/hr. per inch diameter for two closed valves in series. The basis for this model is described in response to AEC Question 5.39 in Amendment 15.

3. The model for gaseous valve seat leakage through a closed valve is 0.10 SCFH per inch of nominal valve diameter and 0.05 SCFH per inch for two valves in series. The former is the acceptance criterion per Manufacturers Standardization Society SP-61, Hydrostatic Testing of Steel Valves, 1961 Edition.

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[6] The Millstone Point Company letter to AEC, "Amendment No. 16 To Licensee Application In Docket No. 50-336," dated June 27, 1973.

Attached to this response is Table 6.16-1 which states:

Potential Containment Leakage Paths
Leakage Path to Auxiliary Bldg

Penetration No. 14, Normal [Containment] Sump
Rate: 1.76×10^{-6} [SCFM]

Penetration No. 67, Refueling Water Purification
Rate: 2.34×10^{-6} [SCFM]

Penetration No. 68, Refueling Water Purification
Rate: 2.34×10^{-6} [SCFM]

Total Leakage Rate to Auxiliary Bldg.: 6.44×10^{-6} [SCFM]

Notes:

1. Potential containment leakage paths following a LOCI without assuming a seismic event.

2. Rate is expressed in units of standard cubic feet per minute.

3. The percentage of leakage through the given path compared to the assumed containment leakage rate during the first day following the incident.

6.16.2 - Response

The analysis for the design basis incident (DBI) is described in Section 14.16 of the FSAR. A containment leak rate of 1.5 volume percent per day was assumed during the first day and 0.75 volume percent per day thereafter as discussed in FSAR Subsection 14.18.2. The containment leakage is calculated as 48 SCFM thereafter.

In order to tabulate liquid leakages as a percentage of the total containment leakage a model is formulated. Due to the minute liquid leakages it is conservatively assumed that all the liquid evaporates into a vapor of the same constituents as the containment atmosphere. Therefore, these liquid leakages can be considered as vapor leakages. The percentages of containment leakages which can be released directly to the atmosphere, auxiliary building and enclosure building filtration region (EBFR) shown in the preceding Table 6.16-1.

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Attached to this response is Table 6.16-2 which states:

Potential Containment Leakage Paths
Leakage Path to Auxiliary Bldg

Total Leakage Rate: Negligible

Notes:

1. Potential containment leakage paths following a loss-of-coolant incident assuming a seismic event.

2. Rate is expressed in standard CFM.

6.16.3 - Response

The amount of containment leakage bypassing the EBFR is negligible as shown in the preceding Table 6.16-1. The potential containment post-incident leakage rate into the EBFR is calculated as 0.8 SCFM or approximately 100 lbs. during the first day. The allowable post-incident containment leakage rate per Technical Specification 15.4.5.A.2.a of the FSAR is approximately 1750 lbs. during the first day (approximately 1050 lbs. during the first day is allowed for the analyzed containment penetrations). The post-incident containment leakage rate assumed for the dose calculations was approximately 5200 lbs. during the first day. Therefore, the amount of leakage assumed for the post-incident dose calculations is approximately five times the Technical Specification limit and more than fifty times that calculated. The dose calculation assumed all potential containment post-incident leakage to be into the EBFR.

The detailed analyses in response to Question 5.39 and 6.16.1 in this Amendment indicate that the only potential leakage that can bypass the EBFR is a minute quantity of liquids. These leakages in actuality will be diluted into the process system fluids and therefore will be contained in that system. On this basis and on the results of the preceding analyses it is not deemed necessary to impose Technical Specification limitations on these three penetrations.

6.16.4 - Response

The analysis in the response to AEC Question 5.39 was based on the assumption that either there is a seismic occurrence and all Seismic Category 2 lines are broken or there is not a seismic occurrence and all Seismic Category 2 lines remain intact. The former case was not considered in the above analysis since any line that is Seismic Category 2 within containment was also Seismic Category 2 outside containment within the EBFR. Should any Seismic

Category 2 lines break within containment, these must also break outside containment. Since all containment penetrations pass through the EBFR prior to entering any other area, any containment leakage due to a broken line must then be vented to the EBFR. Therefore, for a valid analysis of throughline leakage the conservative approach should assume that Seismic Category 2 lines remain intact, as was done.

However, as requested, a re-evaluation of potential containment leakages on an individual penetration basis assuming that all Seismic Category 2 lines are broken following the postulated seismic event was done. The penetration leakage models as formulated in the response to AEC Question 6.16.1 in this Amendment are valid.

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As discussed in the response to AEC Question 6.16.2, the percentages of containment leakages which can be released directly to the atmosphere, auxiliary building and EBFR are shown in the preceding Table 6.16-2.

The above information establishes the Millstone Point Company position that leakage bypassing the containment was considered to be negligible. Both offsite and control room doses were originally calculated using this assumption. On May 10, 1974,[7] the AEC produced the original SER for Millstone Unit No. 2. Within the SER, the following statements were made concerning Control Room Habitability and Offsite Dose Calculations:

6.5 Habitability Systems

We have calculated the potential radiation doses to control room personnel following a LOCA. The resultant doses are within the guidelines of GDC 19. On this basis, we conclude that the design of the control room ventilation system is acceptable for the purpose of preventing significant radiological exposures to operating personnel.

15.2 Loss-of-Coolant Accident Dose Model

Unit 2 is a pressurized water reactor with a low leakage concrete primary containment and a sheet metal secondary structure forming an enclosure building. The enclosure building is maintained at a negative differential pressure after the postulated design basis LOCA. This assures treatment of most released activity by filtration systems and release from an elevated stack.

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[7] O. D. Parr (AEC) letter to The Millstone Point Company, Safety Evaluation for Millstone Nuclear Power Station, Unit 2, dated May 10, 1974.

Upon receipt of a high radiation signal, filtration units will be activated such that the enclosure building is drawn down to a negative differential pressure within one minute. In analyzing the capability of the proposed enclosure building to further minimize the direct outleakage of fission products, the staff considered three specific points: (1) the minimum negative differential pressure throughout the enclosure building, (2) the amount of time required to achieve a minimum negative differential pressure of .25 inches water gauge within the enclosure building, and (3) the fraction of the primary containment leakage that could bypass the filtration system and be released directly to the atmosphere.

We have evaluated the accident analyses for Unit 2 and have determined that the applicants' dose calculations were not consistent with staff reviews of other dual containment systems. Accordingly, we have calculated the loss-of-coolant accident (LOCA) dose for Unit 2 including an incremental dose resulting from the first minute after the postulated design basis loss-of-coolant accident. During this first minute, the staff assumes a direct ground level release from the containment with no holdup or filtering of fission products. After the first minute, the enclosure building region has reached a minimum negative differential pressure of 0.25 inches water gauge, sufficient to assure the retention and filtering of fission products assumed to be leaking from the reinforced concrete containment.

In order to limit the total dose to well within 10 CFR Part 100 guidelines we have advised the applicants that we will require an integrated containment leak rate of 0.5 percent per day. At the containment leak rate proposed by the applicants (1.5 percent per day), the total dose to the thyroid at the site boundary, including the incremental dose from the first minute and the dose resulting from bypass leakage, exceeds the guideline value specified by 10 CFR Part 100. It should be noted that based on a leak rate of 0.5 percent per day, the bypass leakage function was taken as 1.7 percent per day of the containment leak rate.

In order to achieve doses well within 10 CFR Part 100 guidelines, the staff will require the applicant to lower the containment leak rate to 0.5% per day. We believe that this value can be easily met. As can be seen from Table 15-1, the LOCA doses, assuming a primary containment leak rate of 0.5%/day, are well within the guidelines of 10 CFR Part 100.

Conclusion

Based upon the AEC requirements set in the May 10, 1974[8] SER and the correspondence leading up to the SER, the Control Dose assumptions of negligible bypass leakage (6.44×10^{-6} scfm) were not questioned while a bypass

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(8) O. D. Parr (AEC) letter to The Millstone Point Company, Safety Evaluation for Millstone Nuclear Power Station, Unit 2, dated May 10, 1974.

leakage of 1.7% of the daily containment leakage rate was mandated for calculating the 10 CFR 100 offsite doses. The Licensing Basis for calculating the Control Room dose following an event is therefore 6.44x10^{-6} scfm.

Question 2

The release activity in the Millstone 1 main steamline break analysis appears to be based, in part, on Appendix I source terms. The staff does not believe that this is an appropriate source of isotopic fractions for the design basis calculation. Appendix I source terms are typically based on projections of releases over a year and often discount nuclides with short half-life by including a short period of decay. The isotopic fractions are being used in an analysis involving the release of fresh reactor coolant system (RCS) activity. The Millstone 1 Updated Final Safety Analysis Report does not contain a listing of the design basis RCS activity.

Provide a listing of the Millstone 1 design basis RCS activity for the halogens, xenons, and kryptons; and a brief description of the basis (e.g., percent failed fuel, x.x μCi/gm, etc.).

Reply

Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1.[9]

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[9] B. D. Kenyon letter to the NRC, "Millstone Nuclear Power Station, Unit No. 1 Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated July 21, 1998.

In a letter dated November 25, 1997,[10] the NRC requested the following additional information.

Question 1

The proposed Technical Specification Bases contains the position that the nominal recirculation system flow rate of 2500 cubic feet per minute (cfm) can be used in lieu of the minimum acceptable flow of 2250 cfm since the range of flow fluctuation is overwhelmed by other conservatisms found in the control room dose calculations. The staff finds this position to be unacceptable. If the flow value of 2250 cfm is in some way limiting, the proper approach is to explicitly identify the site-specific analysis assumption value deemed to be conservative that can be modified to compensate for the reduced recirculation flow while still maintaining an adequate margin of safety.

The staff requests that the licensee delete this language from the proposed amendment.

Reply

The control room dose calculations have been revised and now use a value of 2250 cfm. The proposed change to the Bases of Technical Specification 3.7.6.1, "Plant Systems – Control Room Emergency Ventilation System," states that the minimum flow of 2250 cfm is used in the associated calculations.

Question 2

In the licensee's analysis of the Unit 3 loss-of-coolant accident, it was assumed that the activity released to the containment and available for release to the environment was 25% rather than the 50% assumed in prior versions of the calculation. However, the licensee's calculation tabulated spray parameters that are incompatible with the assumption of 25% core inventory. The difference between the assumed activities available for release is the method for crediting plate out of the activity released from the Reactor Coolant System (RCS). The spray parameters tabulated in the licensee's calculation appear to be based on Revision 2 to the Standard Review Plan (SRP) 6.5.2. However, it is the staff's position that Revision 2 to the SRP is appropriate only for use with an assumption of 50% core inventory available for release. In assuming 25% core inventory and the spray parameters used in the calculation, the licensee is crediting plate out twice -- the first being the 50% deterministic credit and the second being the plate out lambda of 3.1 hr^{-1}. Revision 0 of SRP 6.5.2 provided for an assumption of 25% core inventory available for release. However, this revision also limited the spray lambda to 10 hr^{-1} with an overall iodine decontamination factor limited to 100.

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(10) D. G. McDonald Jr. letter to NNECO, "Supplemental Request for Additional Information Relating to the Control Room Ventilation System, Millstone Nuclear Power Station, Unit No. 2 (TAC NO. M92879)," dated November 25, 1997.

The staff requests that the licensee correct this deficiency in the analysis.

Reply

Calculation #2 UR(B)-453, MP-2 Control Room Operator Doses Following a MP-3 LOCA, assuming Duct Leakage and Damper Bypass, uses Standard Review Plan (SRP) 6.5.2 Rev 2 and Regulatory Guide (RG) 1.4 in calculating doses. In accordance with the SRP and RG, the calculation assumes instantaneous plateout of 50% of elemental iodine initially released from the core and takes no credit for elemental iodine removal after a DF of 200 is reached relative to the amount of elemental iodines initially released from the core (i.e. the DF of 200 is relative to the initial release of 50% or a DF of 100 relative to the 25% iodines remaining after instantaneous plateout).

The Millstone Unit No. 2 LOCA calculations, which now take credit for sprays, starts with the instantaneous plateout of 50% and only assumes sprays operate for 30 minutes. The elemental iodine DF reached is 6.78 of the 25% iodines remaining after instantaneous plateout or 13.56 relative to the initial release of 50% of the elemental iodines.

Question 3

In the analysis of the Unit 1 main steamline break (MSLB), the licensee did not assess the dose consequences of an MSLB with the RCS activity at the maximum technical specification value of 4.0 μCi/g. This is provided for by Safety Guide 5, and by SRP 15.6.4. The licensee's analysis took the position that assuming a preincident spike was unnecessary because of the low probability of an MSLB accident in the 48-hour period when coolant activities are at 4.0 μCi/g. This position is not acceptable. Since the preincident iodine spike value represents an increase by a factor of 20 in concentration level, it is unlikely that doses will be acceptable.

The staff requests that the licensee provide an analysis of the control room dose associated with an MSLB with RCS concentration at 4.0 μCi/g dose equivalent I-131.

Reply

Millstone Unit No. 1 design basis accidents, loss of coolant and main steam line break, will no longer be evaluated for impact on Millstone Unit No. 2 control room habitability. This credits the decision to decommission Millstone Unit No. 1.[11]

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[11] B. D. Kenyon letter to the NRC, "Millstone Nuclear Power Station, Unit No. 1 Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor," dated July 21, 1998.