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XIV - STATION SAFETY ANALYSIS

1.0 SAFETY OBJECTIVE

The objective of this Station Safety Analysis is to evaluate the adequacy of the plant protective features and event mitigation capability to ensure that the consequences of specific abnormal events remain within the limits of applicable regulations and license commitments. Specifically, the off-site release of radioactive materials and public radiation exposure, and the on-site radiation exposure of plant personnel must meet the requirements of 10CFR20, 10CFR50, and 10CFR100.

Previous sections of this report provide the objective, design basis, and description of each major system and component. Systems that have unique requirements arising from consideration of nuclear safety are evaluated in the safety evaluation portions of those sections of the report. The safety evaluations consider the effects of failures within the system being investigated. Systems essential to safety are capable of performing their functions in adverse circumstances.

1.1 Background and Supporting Information

Definitions for key terms used in this section are presented in USAR Section I-2. A list of references is provided in Section XIV-11.

The Station Safety Analysis presented in the SAR was performed using the initial cycle fuel design and conservative plant parameters. The SAR transient analyses were made at a power level corresponding to 105% of Nuclear Boiler Rated (NBR) steam flow in order to allow the plant to uprate to the turbine-generator design power without invalidating the original licensing analysis. As fuel type and core loading arrangement changes, the fuel reload analysis is performed to establish the plant safety licensing basis and to ensure the plant will continue to meet the requirements of the Code of Federal Regulations for the fuel reload cycle. Analyses presented in USAR Section XIV-5 were performed at various power levels, see Tables XIV-5-1 and XIV-5-2 for actual values used.

Included as part of the fuel reload analysis is a transient analysis which updates the limiting transients of the USAR to reflect changes in the core and plant design characteristic. Approved methodologies described in References 48 and 49 are used in performing the reload transient analysis. Updated transient performance for the current reload cycle is contained in the Supplemental Reload Licensing Report.

Recent reload licensing analysis results for Cooper Nuclear Station are reported in Reference 35. This evaluation utilized several methodologies and boundary conditions that are different from the initial cycle analysis as described in the SAR:

(1) GEMINI/ODYN transient analysis methodology.

(2) Improved scram times (ODYN Option B) and/or scram times per Technical Specifications (ODYN Option A).

(3) GEXL-Plus thermal correlations.

(4) Power-dependent Minimum Critical Power Ratio (MCPR) thermal limits.

(5) Flow-dependent Minimum Critical Power Ratio (MCPR) thermal limits.

(6) Average Power Range Monitor, Rod Block Monitor and Technical Specification (ARTS) Improvement Program.

(7) An expansion of the power/flow operating region through load line limit, extended load line limit, maximum extended load line limit (MELLL), and increased core flow (ICF) analyses.

(8) Statistically based Rod Withdrawal Error (RWE) analysis.

(9) Statistically based Control Rod Drop Accident (CRDA) analysis.

(10) SAFER/GESTR Loss of Coolant Accident (LOCA) analysis methods.

(11) Single loop operation.

(12) Recirculation pump trip (MG set field breaker) on reactor vessel high pressure.

GEXL and GEXL-Plus thermal correlations, power and flow-dependent MCPR limits, and the improved scram time option are discussed in USAR Section III-7.6.1.

The current MELLL and ICF Analysis operating map is shown in Figure III-7-1c and is discussed in USAR Section III-7.6.4.

A discussion of the SAFER/GESTR methods for LOCA analysis is provided in USAR Section VI-5.2.

The SAR Station Safety Analysis categorized off-normal plant operation as either Abnormal Operational Transients or Accidents. Current safety analysis terminology utilizes three groups based on the frequency of occurrence[49]:

(1) Incidents of moderate frequency - These are incidents that may occur with a frequency greater than once per 20 years for a particular plant. These events are referred to as "anticipated (expected) operational occurrences."

(2) Infrequent incidents - These are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). These events are referred to as "abnormal (unexpected) operational occurrences."

(3) Limiting faults - These are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. These events are referred to as "design basis accidents."

The evaluation of BWR operating experience demonstrates that the actual frequency of occurrence of several of the events initially categorized as moderate frequency events is less than once in 20 years. However, these events are currently analyzed as if they were moderate frequency events and there are no operational occurrence analyses performed that fall in the infrequent event category.

Therefore, the SAR Station Safety Analysis utilization of only two categories is still valid. Based upon the acceptance criteria used, the following groups of classification terminologies used in the USAR are interchangeable:

(1) Abnormal Operational Transients, Incidents of Moderate Frequency, and Anticipated Operational Occurrences

(2) Accident and Limiting Fault

The USAR Station Safety Analysis includes a number of Special Events such as Station Shutdown from outside the Control Room, ATWS, etc. These events are described in USAR Section XIV-5.9.

2.0 SAFETY DESIGN BASES FOR ABNORMAL OPERATIONAL TRANSIENTS

1. Release of radioactive material to the environs will not exceed the limits of 10CFR20.

2. Expected (analyzed) abnormal operational transients will not result in any fuel failures.

3. Transient effects to the reactor coolant pressure boundary (including static and differential stresses, temperatures, and cooldown rates) will not exceed those allowed by applicable industry codes for transient.

2.1 Safety Design Bases for Analyzed Special Plant Events

1. Ability to bring the reactor to the shutdown condition by manipulation of the local controls and equipment which are available outside of the control room.

2. Ability to bring the reactor to the cold shutdown condition from outside the control room.

3. Ability to shutdown the reactor independent of control rods.

4. Consequences of an ATWS will not exceed the limits based on 10CFR50.62.

5. Ability to withstand and recover from a Station Blackout for the required coping duration.

3.0 SAFETY DESIGN BASES FOR ACCIDENTS

1. Release of Radioactive material will not exceed the guideline values of 10CFR100, or limits of 10 CFR 50.67 (Fuel Handling Accident or Loss of Coolant Accident).

2. Catastrophic failure of the fuel barrier will not occur as a result of exceeding mechanical or thermal limits.

3. Nuclear system stresses will not exceed those allowed for accidents by applicable industry codes.

4. Containment stresses will not exceed those allowed for accidents by applicable industry codes when containment is required.

5. Overexposure to radiation of station personnel in the control room will not occur.

4.0 APPROACH TO SAFETY ANALYSIS

4.1 General

The following safety analysis investigates two basic design basis groups of events pertinent to safety: abnormal operational transients (including Special Events) and accidents. The safety design bases for the first group are described in USAR Section XIV-2. Analysis of this group of events evaluates the station features that protect the first two radioactive material barriers. Analysis of the events in the second group, accidents, evaluates situations that require functioning of the engineered safeguards including containment.

In considering the various abnormal operational transients and accidents, the full spectrum of conditions in which the core may exist is considered. This is accomplished by investigating the differing safety aspects of the four BWR operating states, as described in Appendix G. In general, only the most severe event of a given type is described in detail.

The OPL-3 Design Guide, is used by the General Electric Company (GE) and Cooper Nuclear Station to establish the important plant characteristic data which are to be used in the transient analyses for reload licensing applications. Included in the data are parameters and functions which are used to verify the SAR transient analysis. As part of the reload analysis, changes in the SAR applicable functions and parameter values are evaluated with respect to the SAR as a licensing basis and the selection of the potential limiting events considered in the reload transient analysis.

4.2 Abnormal Operational Transients

Figure XIV-4-1 shows, in block form, the general method of identifying and evaluating abnormal operational transients. Eight nuclear system parameter variations are listed as potential initiating causes of threats to the fuel and the reactor coolant pressure boundary. The parameter variations are as follows:

1. Nuclear system pressure increase: A nuclear system pressure increase threatens to rupture the reactor coolant pressure boundary. Increasing pressure also collapses the voids in the core moderator which increases core reactivity. This could increase power and lead to fuel cladding damage.

2. Reactor vessel water (moderator) temperature decrease: Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could increase power and lead to fuel cladding damage.

3. Positive reactivity insertion: Positive reactivity insertion is possible from causes other than nuclear system pressure increase or reactor vessel moderator temperature reduction. Such reactivity insertions could increase power and lead to fuel cladding damage.

4. Reactor vessel coolant inventory decrease: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

5. Reactor core coolant flow decrease: A reduction in core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

6. Reactor core coolant flow increase: Increased core flow reduces the void content of the moderator and increases core reactivity. This could increase core power level.

7. Core coolant temperature increase: A core coolant temperature increase could result in overheating of the fuel cladding.

8. Excess of coolant inventory: Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

These eight parameter variations include all of the effects within the nuclear system caused by abnormal operational transients that threaten the integrities of the reactor fuel or nuclear system process barrier. The variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat. For example, positive reactivity insertions resulting from sudden pressure increases are evaluated in the group of threats stemming from nuclear system pressure increases.

Abnormal operational transients may be the result of single equipment failures or single operator errors that can be reasonably expected during any mode of station operations. The following types of operational single failures and operator errors are identified:

1. The opening or closing of any single valve (a check valve is not assumed to close against normal flow).

2. The starting or stopping of any single component.

3. The malfunction or maloperation of any single control device.

4. Any single electrical failure.

5. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or standard operating practices. A single operator error is the set of actions which is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

1. Those actions that could be performed by one person.

2. Those actions that would have constituted a correct procedure had the initial decision been correct.

3. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the station, but are not necessarily directly related to the operator error.

Examples of single operator errors are as follows:

1. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

2. The selection and withdrawal of a single control rod out of sequence.

3. An incorrect calibration of an average power range monitor.

4. Manual isolation of the main steam lines due to operator misinterpretation of an alarm or indication.

The five types of single errors or single malfunctions are applied to the various station systems with a consideration for a variety of station conditions to discover events that directly result in any of the listed undesired parameter variations. Once discovered, each event is evaluated for the threat it poses to the integrities of the radioactive material barriers. Generally, the most severe event of a group of similar events is described.

Four additional events are analyzed as special events:

1. Station shutdown from outside the main control room: The capability provided by the station design to perform the operations required to place and maintain the station in a safe shutdown condition from outside the main control room is demonstrated.

2. Reactor shutdown without control rods: The capability provided by the station design to shut down the reactor with the Standby Liquid Control System is demonstrated. This is described in USAR Section III-9.3.

3. Anticipated Transients Without Scram: In order to determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation situation is postulated.

4. Station Blackout: In order to demonstrate compliance with 10CFR50.63, the capability provided by station design to withstand and recover for a 4-hour coping duration without reliance on the off-site or on-site AC power sources is demonstrated.

4.3 Accidents

Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and which are not expected during the course of plant operations. The following types of accidents are considered:

a. Mechanical failure of various components leading to the release of radioactive material from one or more barriers. The components referred to here are not components that act as radioactive material barriers. Examples of mechanical failures are breakage of the coupling between a control rod drive and the control rod, failure of a crane cable, and failure of a spring used to close an isolation valve.

b. Overheating of the fuel barrier. This includes overheating as a result of reactivity insertion or loss of cooling. Other radioactive material barriers are not considered susceptible to failure due to any potential overheating situation.

c. The arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the nuclear system process barrier.

Figure XIV-4-2 shows, in block form, the method of identifying and evaluating accidents. For analysis purposes, accidents are categorized as follows:

a. Accidents that result in radioactive material release from the fuel within the nuclear system process barrier, primary containment, and secondary containment initially intact.

b. Accidents that result in radioactive material release directly to the primary containment.

c. Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact.

d. Accidents that result in radioactive material release directly to the secondary containment with the primary containment not intact.

e. Accidents that result in radioactive material release outside the secondary containment.

The effects of the various accident types are investigated, with a consideration for a variety of plant conditions, to examine events that result in the release of radioactive material. The accidents resulting in potential radiation exposures greater than any other accident considered under the same general accident assumptions are designated Design Basis Accidents (DBA) and are described in detail.

To incorporate additional conservatism into the accident analyses, consideration is given to the effects of an additional, unrelated, unspecified fault in some active component or piece of equipment. Such a fault is assumed to result in the maloperation of a device which is intended to mitigate the consequences of the accident. The assumed result of such an unspecified fault is restricted to such relatively common events as an electrical failure, instrument error, motor stall, breaker freeze-in, or valve maloperation. Highly improbable failures, such as pipe breaks, are not assumed to occur coincident with the assumed accident. The additional failures to be considered are in addition to failures caused by the accident itself.

In the analyses of the Design Basis Accidents a variety of single additional failures is considered by making analysis assumptions that are sufficiently conservative to include the range of effects from any single additional failure. Thus, no single additional failure of the types to be considered could worsen the computed radiological effects of the Design Basis Accidents.

4.4 Barrier Damage Evaluations

4.4.1 Fuel Damage

Fuel damage is defined for design purposes as perforation of the cladding which permits release of fission products (see USAR Section III-2, "Fuel Mechanical Design"). The mechanisms which could cause fuel damage in reactor transients are:

(1) Rupture of the fuel cladding due to strain caused by relative expansion of the uranium dioxide pellet and the fuel cladding.

(2) Severe overheating of the fuel cladding caused by inadequate cooling.

Steady-state operating limits have been established to assure that sufficient margin exists between the steady-state operating condition and any fuel damage condition to accommodate the worst abnormal operational transient without experiencing fuel damage throughout the life of the fuel.

For the worst abnormal operational transient, it must be demonstrated that nucleate boiling heat transfer is maintained and the fuel rods do not experience a boiling transition. The analyses originally presented in the SAR expressed the required margin between steady-state conditions and those which would produce a boiling transition in terms of the minimum critical heat flux ratio (MCHFR). If MCHFR was maintained above 1.0 during the abnormal operational transient, no fuel damage would be calculated to occur as a result of inadequate cooling.

For current plant operation, analyses have been performed which show that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition (and, therefore, cladding damage due to overheating) if the MCPR is equal to or greater than the fuel cladding integrity safety limit value (see USAR Section III-7, "Thermal and Hydraulic Design"). The current fuel cladding integrity safety limit MCPR for two recirculation loop operation and for single-loop operation are contained in the current cycle Supplemental Reload Licensing Report[35]. Therefore, MCPR operating limits are established by addition of the maximum ΔCPR value for the most limiting abnormal operational transient to the safety limit MCPR value.

4.4.2 Reactor Coolant Pressure Boundary Damage

Safety Design Basis 3 for abnormal operational transients and for accidents is assessed by comparing peak internal pressure with the overpressure transient allowed by the applicable industry code. The only significant areas of interest for internal pressure damage are the high-pressure portions of the Reactor Coolant Pressure Boundary, the reactor vessel and the high-pressure pipelines attached to the reactor vessel. The overpressure protection provided for the high-pressure portions of the Reactor Coolant Pressure Boundary is described in USAR Section IV-4. The event hypothesized to evaluate the adequacy of the overpressure protection is more severe than any of the abnormal operational transients described in this chapter. The abnormal operational transients, therefore, will not result in nuclear system stress in excess of that allowed for transients by applicable industry codes.

The only design basis accident (Table XIV-7-2) that results in a nuclear system pressure increase is the rod drop accident. An analysis performance measurement, which is discussed in USAR Section III-6 ("Nuclear Design"), is used to evaluate whether reactor coolant pressure boundary damage occurs as a result of reactivity accidents. If peak fuel enthalpy remains below 280 calories per gram no reactor coolant pressure boundary damage results from nuclear excursion accidents.

4.4.3 Containment Damage

Safety Design Bases 1 and 4 for accidents requires that the primary and secondary containments retain their structural integrities for certain accident situations. Containment integrity is maintained as long as internal pressures remain below the maximum allowable values as indicated below:

Primary Containment	62 psig
Secondary Containment	7 inches H_20

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the Safety Analysis Report where the mechanical design features of systems and components are described. Design Basis Accidents are used in determining the sizing and strength requirements of much of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals that either the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

5.0 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS[3]

The potential adverse effects to the fuel and Nuclear Steam Supply System of the eight nuclear system variations described in USAR Section XIV-4.2 are analyzed in this section. Only a few of these transients result in a significant reduction in MCPR and therefore pose a threat to the integrity of the fuel cladding.

To determine the limiting transient events that threaten fuel damage, the relative dependency of critical power ratio (CPR) upon various thermal-hydraulic parameters has been examined. A sensitivity study was performed to determine the effect of changes in bundle power, bundle flow, subcooling, R-factor, and pressure on CPR.

CPR is most responsive to fluctuations in the R-factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet subcooling was also shown. The R-factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout a transient. Therefore, transients which would be limiting because of MCPR would primarily involve significant changes in power. Based on this, the transients most likely to limit operation because of MCPR considerations are:

1. Turbine trips or generator load rejection without bypass,

2. Loss of feedwater heating or inadvertent HPCI startup,

3. Feedwater controller failure (maximum demand), and

4. Control rod withdrawal error.

The Cooper fuel reload analysis is performed considering the above sensitivity analysis results as well as the other transient categories described in this chapter. The potentially limiting events are evaluated to determine the required MCPR operating limits.

Many of the events described in the following sections have not been reanalyzed for any of the reload cycles, because they are bounded by other events which have been analyzed. These events, including the associated assumptions and conclusions, continue to be part of the plant's licensing basis. The conclusions of these analyses are still valid; however, specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions.

The core-wide Abnormal/Anticipated Operational Occurrence analyses for MELLL and ICF were performed for CNS Cycle 30 reload transients. These transient events, documented in Reference 35 include:

1. Generator load reject without bypass (LRNBP) (Section 5.1.1.3)

2. Turbine trip without bypass (TTNBP) (Section 5.1.2.3)

3. Loss of feedwater heating (LFWH) (Section 5.2.1)

4. Feedwater controller failure (FWCF) – maximum demand (Section 5.8.1)

5. Inadvertent HPCI startup (Section 5.2.3)

In addition, CNS Cycle 20 analyses for MELLL and ICF were performed for Anticipated Transients Without Scram (ATWS). ATWS results are also documented in Reference 83.

The reactor pump seizure event is analyzed for single loop operation (Section 5.5.3) for an equilibrium cycle GNF2 core with transient analysis inputs that are consistent with the Cycle 28 analysis as documented in Reference 98. The cycle independent result is then scaled to the cycle specific SLMCPR value in each reload cycle as documented in Reference 35.

5.1 Events Resulting in a Nuclear System Pressure Increase

The SAR categorized the following events as those that result directly in significant nuclear system pressure increases due to a sudden reduction of steam flow while the reactor is operating at power:

a. Generator trip/load rejection (turbine control valve fast closure).

b. Turbine trip (turbine stop valve closure).

c. Closure of the main steam isolation valves.

d. Failure of the turbine bypass valves to open when required.

e. Loss of main condenser vacuum.

A consideration of the last two varieties of events shows that turbine bypass valve failure and loss of condenser vacuum are specific cases of the first two event types. A failure of the turbine bypass valves to open when required is analyzed as the most severe form of a generator load rejection or turbine trip. Instantaneous loss of condenser vacuum is nearly identical to the turbine trip without bypass transient, with scram from the turbine stop valve position indicating signals. For the loss of vacuum, the feedwater turbines would also be tripped. However, the parameters of main concern, fuel thermal limit margins and vessel overpressure are not significantly different from the analysis performed for the turbine trip (no bypass) transient (USAR Section XIV-5.1.2.3).

A full closure of all main steam isolation valves (MSIVs) without direct (from position switch) scram is used to evaluate the performance of the nuclear system pressure relief system. This analysis is included in USAR Section IV-4 "Nuclear System Pressure Relief System".

In addition to the above events, the potential for system pressure increase due to a malfunction in which the Digital Electro-Hydraulic (DEH) pressure controller output fails low will be discussed.

5.1.1 Generator Load Rejection (Turbine Control Valve Fast Closure)

5.1.1.1 Identification of Causes

A loss of generator load causes the turbine generator to overspeed. As described in USAR Section VII-11.3.3 "Overspeed Protection," the Digital Electro-Hydraulic (DEH) Control System provides an Overspeed Protection Control (OPC) system to control turbine overspeed in the event that turbine speed reaches or exceeds 103 percent of rated. When this occurs, the OPC solenoid valves energize, draining the governor valve emergency trip header, causing the governor and intercept valves to close. The closing causes a sudden reduction of steam flow, which results in a nuclear system pressure increase. Low governor valve emergency trip header pressure serves as the input signal for the control valve fast closure scram.

5.1.1.2 Generator Load Rejection (Turbine Control Valve Fast Closure) With Bypass

This transient was initially analyzed in the SAR. It has not been reanalyzed for any of the subsequent reload cycles because it is bounded by the generator load rejection without bypass event.

5.1.1.2.1 Frequency Classification

This event is classified as an incident of moderate frequency.

5.1.1.2.2 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the generator load rejection with bypass transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating in the manual flow control mode when load rejection occurs.

5.1.1.2.3 Sequence of Events and Systems Operation

Figure XIV-5-1 is the transient simulation of the generator load rejection with bypass at an initial operating steam flow condition of 105 percent NBR. Complete loss of generator load produces the following sequence of events:

a. Turbine-generator overspeed protection initiates fast (about 0.2 second) governor valve closure.

b. Fast valve closure is sensed by the reactor protection system causing an immediate reactor scram (for initial power levels above 30 percent).

c. The bypass valves are opened simultaneously with the governor valve closure.

d. Reactor pressure rises to the relief valve setpoints, causing them to open for a short period, discharging some of the stored energy to the suppression pool.

e. The turbine bypass system controls nuclear system pressure after the relief valves close.

TABLE XIV-5-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR SAR TRANSIENT ANALYSES
FOR INITIAL CORE ONLY (See USAR Section XIV-5.0 for Explanation)

Parameter	Analysis Value
Thermal Power Level (MWt)	2482
Jet Pump M Ratio	1.27
Feedwater Capacity (Percent of Rated Steam Flow)	115
Steam Bypass Capacity (Percent of Rated Steam Flow)	26
Relief Valve Capacity (Percent of Rated Steam Flow)	61
Relief Valve Setpoints (psig)	1091, 1101, 1111
Safety Valve Capacity (Percent of Rated Steam Flow)	15
Safety Valve Setpoint (psig)	1240
Void Coefficient (¢/percent)	$\leq$ -8
Vessel Dome Pressure (psig)	1020
Core Flow (Mlb/hr)	73.5
Steam Flow (Mlb/hr)	10.04
Feedwater Temperature (°F)	391
APRM Neutron Flux Scram Setpoint at Rated Drive Flow (percent rated)	125.1
Vessel Dome Pressure Scram Setpoint (psig)	1070
Response Time of CRD During Scram	Tech. Spec. Limits
Change in Feedwater Temperature-Worst Single Failure of Feedwater Heaters (°F)	100
Response Time (Delay) of RPS Logic (sec.)	0.07
Response Time for Full Stroke of TCV Fast Closure (sec.)	0.2
Response Time for Full Stroke of TSV Fast Closure (sec.)	0.1
Turbine Throttle Pressure (psig)	960
Turbine Power/Scram Bypass Setpoint Analysis Basis (percent rated)	30
Low Turbine Throttle Pressure Setpoint for MSIV Closure (psig)	850
Rate of Change of Recirculation MG Set Coupler Scoop Tube (percent/sec)	25
MSIV Position Switch Setpoint (percent open)	90
RPV Low Water Level for RCIC Initiation, MSIV Isolation, and Recirculation Pump Trip (inches above vessel zero)	479.25

Important Protection System Functions Assumed in Analysis:

Scram due to turbine stop valve closure or fast control valve closure.

Scram due to high neutron flux (120 percent of operating power).

Scram due to high vessel dome pressure (50 psi above operating pressure).

Scram due to low reactor water level.

Scram due to main steam isolation valve closure.

Scram and main steam line isolation due to loss of auxiliary power.

Main steam line isolation due to low turbine inlet pressure.

Main steam line isolation, recirculation pump trip, and reactor core isolation cooling system initiation due to low reactor water level (Level 2). (MSIV closure was changed to Level 1 in License Amendment 83. See discussion in USAR Section XIV-5.3.4.7.)

5.1.1.2.4 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. REDY does not consider the effects of steam line pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). Therefore, the transient is expected to be less severe than predicted.

The transient is essentially the same as the similar case resulting from a turbine trip (USAR Section XIV-5.1.2.2). Below about 25 percent of rated power, the bypass system will transfer steam around the turbine and avoid reactor scram. Between about 25 percent and 30 percent power, a high RPV pressure scram will result unless operator action can reduce power to within the bypass capacity.

As shown in Figure XIV-5-1, as soon as turbine control valve fast closure is sensed, a scram is initiated. This occurs in advance of the high neutron flux and high RPV pressure scram signals thereby limiting the peak neutron flux to about 180 percent of rated. The average surface heat flux reaches a peak of about 115 percent of rated. The small increase in average surface heat flux coupled with the slight increase in core flow ensures that nucleate boiling is maintained throughout the transient.

5.1.1.3 Generator Load Rejection (Turbine Control Valve Fast Closure) Without Bypass

The core-wide Abnormal/Anticipated Operational Occurrence analyses for MELLL and ICF are performed for each reload cycle. These transient events include the generator load rejection without bypass (LRNBP). This analysis is documented in Reference 35.

5.1.1.3.1 Frequency Classification

This event is classified as an infrequent incident. However, this event is currently analyzed, and corresponding limitations defined as if it was a moderate frequency event.

5.1.1.3.2 Starting Conditions and Assumptions

The parameter values used in the analysis are presented in Table XIV 5-2. The following additional plant operating conditions and assumptions form the principal bases for analysis of the generator load rejection without bypass transient:

a. The reactor and turbine generator are initially operating at rated power when the load rejection occurs.

b. All of the plant control systems continue normal operation.

c. Auxiliary power is continuously supplied at rated frequency.

d. The reactor is operating in the manual flow control mode when load rejection occurs.

TABLE XIV-5-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENT ANALYSES FOR RELOAD LICENSING ANALYSIS (RLA)

Transient Analyses at 100 Percent Rated Power

When GEMINI calculation methods are used for the plants licensed at 100 percent NBR power level, the pressurization transients (generator load rejection without bypass, turbine trip without bypass, inadvertent HPCI initiation and feedwater controller failure - maximum demand) for which the MCPR operating limit is a requirement and the loss of feedwater heating event with PANACEA are analyzed at 100 percent NBR power.

Parameter	Analysis Value
Thermal Power Level (MWt)	2419
Vessel Dome Pressure (psig)	1005
Vessel Core Pressure (psia) (ICF/MELLLA)	1036/1031
Steamline Pressure Drop (psid)	52
Core Flow (Mlb/hr) (ICF/MELLLA)	77.2/56.4
Steam Flow (Mlb/hr) (ICF/MELLLA)	9.72/9.70
Feedwater Temperature (°F)	367.1
Core Coolant Inlet Enthalpy (BTU/lb) (ICF/MELLLA)	521.3/511.5
Change in Feedwater Temperature-Worst Single Failure of Feedwater Heaters (°F)	100
Recirculation System Pump Operating Conditions (99.4 Percent Power/101.2 Percent Flow)	
Pump Head (psid) - Pump A/Pump B	135/140
Pump Flow (Mlb/hr) - Pump A/Pump B	16.9/17.3
Pump Speed (rpm) - Pump A/Pump B	1570.5/1572
Core Plate Pressure Drop (psid)	19.5
Normal Vessel Water Level - Wide and Narrow Range at Rated Power (inches above vessel zero)	552.0
Licensed Power/Flow Operating Map (Operating Flexibility Option Maximum Extended Load Line Limit Analysis (MELLLA) and Increased Core Flow (ICF)	ICF/MELLLA
APRM Neutron Flux Scram Setpoint at Rated Drive Flow (percent rated)	123
Vessel Dome Pressure Scram Setpoint (psig)	1060
Response Time of Pressure Scram Sensor (sec.)	0.5
MSIV Position Switch Setpoint (percent open)	90
Turbine Stop Valve (TSV) Position Switch Setpoint (percent open)	90
Response Time of TCV Fast Closure Sensor (sec.)	0.03
Response Time (Delay) of RPS Logic (sec.)	0.05

TABLE XIV-5-2 (CONT'D)

Transient Analyses at 100 Percent Rated Power

Parameter	Analysis Value
Response Time of CRD During Scram	ODYN Option B and/or Tech Spec (Option A)
ATWS Recirculation System (MG Set Field Breaker) Trip Pressure Setpoint (psig)	1095
High RPV Water Level Trip-Main Turbine, RFP, RCIC, HPCI (inches above vessel zero)	575.25
Low Water Level Alarm Setpoint (inches above vessel zero)	544.25
Main Steamline Parameters	
Average Length-Vessel to Inboard MSIV (ft)	95.3
Combined Volume-Vessel to Inboard MSIV (ft^3)	967.3
Average Length-Inboard MSIV to TSV (ft)	348.9
Combined Volume-Inboard MSIV to TSV (ft^3)	4094.9
Bypass Steamline Parameters	
Average Length-Header/Tap to BPV (ft)	56.5
Combined Volume-Header/Tap to BPV (ft^3)	93.6
Maximum RFP Runout Flow at FW Design Pressure (percent rated)	128 percent at 1070 psig
Maximum RFP Runout Flow at RLA Dome Pressure (percent rated)	144 percent at 1005 psig
Design Capacity of HPCI at Rated Dome Pressure (gpm)	4250
Design Capacity of RCIC at Rated Dome Pressure (gpm)	400
Minimum Temperature of Condensate Storage Tank Water (°F)	40
Total BPV Capacity at Rated Throttle Pressure (percent rated)	25
BPV Delay for Fast Opening from Event Initiation to Start of BPV Opening (sec.)	0.1
Total Response Time of BPV to 80 percent BPV Flow (sec.)	0.3
Response Time for Full Stroke of TCV Fast Closure (sec.)	0.15
Response Time for Full Stroke of TSV Fast Closure (sec.)	0.1
TCV’s Position Measured at Rated Power (percent open)	63.5%
Turbine Throttle Pressure (psig)	953
RPV Low Water Level Scram and Isolation (except MSIV) Setpoint (inches above vessel zero)	517.25

TABLE XIV-5-2 (CONT'D)

Transient Analyses at 100 Percent Rated Power

Parameter	Analysis Value
Response Time of RPV Low Water Level Scram Sensor (sec.)	1
Turbine Power/Scram Bypass Setpoint Analysis Basis (percent rated)	30
Scram Setpoint Intercept at Zero Drive Flow (percent rated)	65.6
Response Time of Drive Flow Sensor (sec.)	1.00
RPV Low Water Level for RCIC/HPCI Initiation and Recirculation Pump Trip (inches above vessel zero)	467.75
RPV Low Water Level for ADS, LPCI, LPCS, MSIV Closure (inches above vessel zero)	358.56
Low Turbine Throttle Pressure Setpoint for MSIV Closure (psig)	825
Rate of Loss of Condenser Vacuum if CW Pumps Fail (inches Hg/sec.)	2.00
Low Condenser Vacuum Protection Setpoints (inches Hg vacuum)	
Initiate Turbine Trip	17
Initiate BPV and MSIV Closure	7
Rate of Change of Recirculation MG Set Coupler Scoop Tube (percent/sec)	25
Non-Fuel Power Fraction	0.038

Average Scram Insertion Times

Percent Inserted From Fully Withdrawn	ODYN Option B (sec.)	Technical Specifications (sec.)
5	0.324	0.490
20	0.694	0.900
50	1.459	2.000
90	2.535	3.500

Safety Relief Valve and Spring Safety Valve Parameters

Valve Setpoint	Valve Capacity
Two S/RVs at 1112.4 psig Three S/RVs at 1122.7 psig Three S/RVs at 1133 psig	Two S/RVs have a certified capacity of 862,100 lb/hr at a reference pressure of 1080 psig + 3 percent accumulation. Three SR/Vs have a certified capacity of 870,000 lb/hr at a reference pressure of 1090 psig +3 percent accumulation. Three SR/Vs have a certified capacity of 877,900 at a reference pressure of 1100 psig +3 percent accumulation.(1)
Three SSVs at 1277.2 psig	Each SSV has a certified capacity of 644,543 lb/hr at a reference pressure of 1240 psig +3 percent accumulation (1).

(1) Accumulation corresponds to the overpressure above the setpoint, required to assure the valve is fully open. Example: A valve which has a certified capacity of 870,000 lb/hr at a reference pressure of 1090 psig +3 percent accumulation means that the valve is certified to flow at least 870,000 lb/hr at a valve inlet pressure of (1090)(1.03) = 1122.7 psig.

TABLE XIV-5-2 (CONT'D)

Transient Analyses at 102 Percent Rated Power

The MSIV closure with flux scram is analyzed at a 102 percent of original licensed power level.*

Parameter	Analysis Value
Thermal Power Level (MWt)	2428.6
Vessel Dome Pressure (psig)	1045
Steamline Pressure Drop (psid)	52
Core Flow (Mlb/hr)	73.5
Steam Flow (Mlb/hr)	9.79
Feedwater Temperature (°F)	369.1
Void Fraction (percent)	43.66

* These input values are used in the analysis of the MSIV closure with flux scram transient which is described in USAR Section IV-4.6.

e. The turbine bypass valve system is failed in the closed position.

5.1.1.3.3 Sequence of Events and Systems Operation

Figures XIV-5-2a-h provide the transient simulation of the generator load rejection without bypass initiated at 100 percent NBR steam flow. Complete loss of the generator load produces the following sequence of events:

a. The loss of generator load results in the turbine accelerating at a maximum rate until the OPC system starts to close the governor and intercept valves. The turbine governor (control) valves will close at a rate of 0.150 seconds for the full valve stroke.

b. Reactor scram is initiated upon sensing low governor valve emergency trip header pressure which serves as the input signal for the control valve fast closure scram.

c. If RPV pressure rises above 1095 psig (value used in analysis), an ATWS-RPT trip of the recirculation pump MG set field breakers occurs.

d. If steam line pressure rises above 1113 psig (value used in analysis), some or all of the relief valves open, discharging steam to the suppression pool.

5.1.1.3.4 Core and System Performance

The generator load rejection without bypass and generator load rejection with bypass events are similar events categorized in this transient category. The more limiting of the two events has been identified by Reference 49 as most likely to limit power generation when MCPR limits are considered. Therefore, it is analyzed as a licensing basis transient for the fuel reload cycle.

For reload cores, an evaluation is performed to determine if the generator load reject without bypass transient could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the Supplemental Reload Licensing Report.

Plant/cycle-specific analyses for Cooper Nuclear Station using the GEMINI/ODYN transient analysis methodology as described in References 48 and 49 were reported in Reference 35 and are summarized in Table XIV-7-3. The cycle-specific reload analysis demonstrates that MCPR would not exceed the safety limit MCPR.

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. A large data base was established by analyzing limiting transients over a range of power and flow conditions.

This data base shows that two power ranges must be examined. The first power range is between rated power and the power level where reactor scram on turbine control valve fast closure is bypassed (P_{Bypass}, 30 percent of rated power). In this range, the generator load rejection with no bypass becomes less severe as power decreases since the reduced steam flow rate at low power level results in milder reactor pressurization.

The second power range is between P_{Bypass} and 25 percent of rated power. No thermal limit monitoring is required below 25 percent power. Below

P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on closure of the turbine control valves. This delays the scram signal until the vessel pressure reaches the high pressure scram setpoint. The extensive transient data base also shows a significant sensitivity to the initial core flow for transients initiated below P_{Bypass}. Sensitivity analyses on the limiting transients at below rated power conditions have shown that the maximum ΔCPR at any power level occurs at the maximum core flow condition.

The results of a generator load rejection without bypass transient analysis at initial conditions of 30 percent rated power and 100 percent rated core flow are presented in Reference 53. The analysis confirms that this transient, where reactor scram occurs due to high RPV pressure, is bounded by the feedwater controller failure (FWCF) maximum demand event. Below P_{Bypass}, the actual bounding MCPR limit is chosen with sufficient conservatism such that it is independent of the cycle-specific operating limits.

5.1.2 Turbine Trip (Turbine Stop Valve Closure)

5.1.2.1 Identification of Causes

A variety of turbine, electrical, or nuclear system malfunctions will initiate a main turbine trip. Some examples are: loss of control fluid pressure (bearing oil or auto-stop oil), low condenser vacuum, electrical distribution system faults, and reactor high water level. When the turbine trips, the turbine stop valves close causing a sudden reduction in steam flow which results in a nuclear system pressure increase. Turbine stop valve position serves as an input to the reactor protection system to initiate the shutdown of the reactor.

5.1.2.2 Turbine Trip (Turbine Stop Valve Closure) With Bypass

This transient was initially analyzed in the SAR. It has not been reanalyzed for any of the subsequent reload cycles because it is bounded by the turbine trip without bypass event.

5.1.2.2.1 Frequency Classification

This event is classified as an incident of moderate frequency.

5.1.2.2.2 Starting Conditions and Assumptions

The plant operating conditions and assumptions are identical to those of the generator load rejection with bypass presented in USAR Section XIV-5.1.1.2.2.

5.1.2.2.3 Sequence of Events and Systems Operation

The sequence of events for a turbine trip with bypass is very similar to that for a generator load rejection with bypass. However, the valve closure time assumed for the turbine stop valves (0.1 second) is slightly faster than that assumed for the turbine control valves (0.2 second).

Position switches at the stop valves provide the means of sensing the turbine trip and initiating immediate reactor scram. The turbine control system immediately initiates bypass valve opening in an attempt to control pressure. The safety/relief valves open for a short time to help relieve the pressure transient, and then the bypass valves control reactor pressure following the transient.

5.1.2.2.4 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. REDY does not consider the effects of steam line pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). Therefore, the transient is expected to be less severe than predicted.

Turbine trips from lower initial power levels decrease in severity to the point where scram may even be avoided within the bypass capacity if auxiliary power is available from an external source.

Figure XIV-5-3 is the transient simulation of the turbine trip with bypass at an initial operating steam flow condition of 105 percent NBR. The fuel thermal transient is mild, with peak heat flux less than 107 percent of the initial fuel surface heat flux. Peak neutron flux is held to 175 percent of rated by the fast action of the stop valve scram.

5.1.2.3 Turbine Trip (Turbine Stop Valve Closure) Without Bypass

The core-wide Abnormal/Anticipated Operational Occurrence analyses for MELLL and ICF are performed for each reload cycle. These transient events include the turbine trip without bypass (TTNBP). This analysis is documented in Reference 35.

5.1.2.3.1 Frequency Classification

This event is classified as an infrequent incident. However, this event is currently analyzed, and corresponding limitations defined as if it was a moderate frequency event.

5.1.2.3.2 Starting Conditions and Assumptions

The plant operating conditions and assumptions are identical to those of the generator load rejection without bypass presented in USAR Section XIV-5.1.1.3.2.

5.1.2.3.3 Sequence of Events and Systems Operation

Figure XIV-5-4a-h provides the transient simulation of the turbine trip without bypass initiated at 100 percent NBR steam flow. The sequence of events for a turbine trip is similar to those for a generator load rejection (USAR Section XIV-5.1.1.3.3). Stop valve closure occurs over a period of 0.10 second.

Position switches at the stop valves sense the turbine trip and initiate reactor scram. If the RPV pressure rises to the ATWS setpoint, a trip of the recirculation pump MG set field breakers occurs. If steam line pressure rises to the pressure relief setpoints, relief valves open, discharging steam to the suppression pool.

5.1.2.3.4 Core and System Performance

The turbine trip without bypass and the turbine trip with bypass events are similar events categorized in this transient category. The more limiting of the two events has been identified by Reference 49 as most likely to limit power generation when MCPR limits are considered. Therefore, it is analyzed as a licensing basis transient for the fuel reload cycle.

For reload cores, an evaluation is performed to determine if the turbine trip without bypass transient could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the Supplemental Reload Licensing Report.

Plant/cycle-specific analyses for Cooper Nuclear Station using the GEMINI/ODYN transient analysis methodology as described in References 48 and 49 were reported in Reference 35 and are summarized in Table XIV-7-3. The cycle-specific reload analysis demonstrates that MCPR would not exceed the safety limit MCPR.

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. A large data base was established by analyzing limiting transients over a range of power and flow conditions.

This data base shows that two power ranges must be examined. The first power range is between rated power and the power level where reactor scram on turbine stop valve closure is bypassed (P_{Bypass}, 30 percent of rated power). In this range, the turbine trip with no bypass becomes less severe as power decreases since the reduced steam flow rate at low power level results in milder reactor pressurization.

The second power range is between P_{Bypass} and 25 percent of rated power. No thermal limit monitoring is required below 25 percent power. Below P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on closure of the turbine stop valves. This delays the scram signal until the vessel pressure reaches the high pressure scram setpoint. The extensive transient data base also shows a significant sensitivity to the initial core flow for transients initiated below P_{Bypass}. Sensitivity analyses on the limiting transients at below rated power conditions have shown that the maximum ΔCPR at any power level occurs at the maximum core flow condition.

The results of a turbine trip without bypass transient analysis at initial conditions of 30 percent rated power and 100 percent rated core flow are presented in Reference 53. The analysis confirms that this transient, where reactor scram occurs due to high RPV pressure, is bounded by the feedwater controller failure (FWCF) maximum demand event. Below P_{Bypass}, the actual bounding MCPR limit is chosen with sufficient conservatism such that it is independent of the cycle-specific operating limits.

5.1.3 Main Steam Isolation Valve (MSIV) Closure

The transients in this section were initially analyzed in the SAR. These events were not identified as transients which are significantly affected by fuel reload in Reference 49. Therefore, they have not been reanalyzed for any of the subsequent reload cycles. The MSIV closure event without direct scram (from position switch) is currently analyzed, and results for this event are presented in USAR Section IV-4.

5.1.3.1 Identification of Causes

Various steam line and nuclear system malfunctions, or operator action, can initiate closure of the main steam isolation valves. Examples of conditions which cause automatic closure are low reactor water level, low steam line pressure, high steam line area temperature, high steam line flow, and low main condenser vacuum.

5.1.3.2 Frequency Classification

MSIV Closure events are classified as incidents of moderate frequency.

5.1.3.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the main steam isolation valve closure transients:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating in the manual flow control mode.

d. Main Steam Isolation Valves close in 3 to 5 seconds. A 3-second nonlinear valve closure was simulated, which is the fastest allowable closure time.

e. In the analysis for Closure of All MSIVs, it was conservatively assumed that feedwater flow was terminated within 5 seconds due to MSIV closure.

5.1.3.4 Closure of One Main Steam Isolation Valve

Closure of one main steam isolation valve is desirable for testing purposes. Position switches on the valves provide reactor scram signals if the valves are less than 90 percent open and the reactor mode switch is in the RUN position. However, the arrangement of the MSIV closure scram function in the Reactor Protection logic does permit the test closure of one valve without initiating scram from the position switches. Normal procedures for such a test will require an initial power reduction to below 70 percent of rated power in order to avoid high flux or pressure scram.

5.1.3.4.1 Sequence of Events and Systems Operation

Figure XIV-5-5 graphically shows the changes of important nuclear system variables during the simulated 3-second closure of an MSIV in one (out of four) main steam lines from 105 percent of rated power conditions. Closure of one main steam isolation valve produces the following sequence of events:

a. A 3-second closure of an isolation valve in one (out of four) main steam lines is initiated.

b. The steam flow disturbance raises vessel pressure and reactor power causing a high neutron flux scram.

c. The reactor vessel pressure rise is limited since steam can flow to the main turbine and bypass valves through the other three main steam lines.

5.1.3.4.2 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. REDY does not consider the effects of steam line pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). Therefore, the transient is expected to be less severe than predicted.

As shown in Figure XIV-5-5, the peak surface heat flux is 134 percent of rated and peak center fuel temperature increased only 65°F. Nucleate boiling is maintained throughout the transient (MCHFR remained above 1.8).

5.1.3.5 Closure of All Main Steam Isolation Valves (MSIVs)

A full closure of all MSIVs with direct scram is described in this section.

5.1.3.5.1 Sequence of Events and Systems Operation

Figure XIV-5-6 graphically shows the changes of important nuclear system variables during the simulated 3-second closure of all MSIVs at 105 percent of rated power conditions. Closure of all main steam isolation valves produces the following sequence of events:

a. A 3-second closure of all main steam isolation valves is initiated.

b. Reactor scram is initiated by the isolation valve position switches before the valves have traveled more than 10 percent from the open position.

c. Reactor pressure rises to the relief valve setpoints, causing them to open and discharging some of the stored energy to the suppression pool.

d. Feedwater flow is lost due to closure of the MSIVs.

5.1.3.5.2 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. REDY does not consider the effects of steam line pressure wave transmission and core axial power shape. However, these two effects tend to compensate each other to some degree. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). Therefore, the transient is expected to be less severe than predicted.

Results of startup tests at several plants with turbine-driven feed pumps have demonstrated that feedwater flow actually remained near rated for at least 20 seconds, and was sufficient to bring vessel water level back to normal before HPCI/RCIC flow was injected into the vessel.

As shown in Figure XIV-5-6, scram is initiated before any significant flow interruption occurs, therefore no fuel center temperature or fuel surface heat flux peaks occur. A neutron flux peak (about 173 percent of rated) occurs near 2.2 seconds due to the conservative scram worth characteristic assumed. The positive reactivity added by void collapse is greater than the negative control rod reactivity assumed for the first part of rod motion. Nucleate boiling is maintained throughout the transient (no reduction in MCHFR occurs).

5.1.4 DEH Pressure Controller Output Signal Fails Low

A triple redundant DEH control system is provided to maintain primary system pressure control. The pressure upstream of the main turbine stop valves is sensed by three redundant throttle pressure transmitters at the equalizing header and the control system uses a median select logic to determine which pressure transmitter is used to control throttle pressure. The pressure control system compares the detected throttle pressure to a pressure setpoint to control the position of the main turbine control (governor) and bypass valves in order to control pressure. If all three pressure transmitters are detected healthy by the main processors, the selected transmitter is equal to the median (middle) value. If only two pressure inputs are detected healthy, the selected output is equal to the highest input. If one pressure input is good, it is used.

It is assumed for purposes of this transient analysis that a single failure occurs on the controlling pressure transmitter which erroneously causes the DEH control system to close the turbine control (governor) valves and thereby increases reactor pressure. A failure of a DEH control system component that causes the turbine control (governor) valves or turbine bypass valves to move towards the closed position will momentarily result in an initial pressure increase because the reactor is still generating the initial steam flow. The DEH control system is self-diagnostic. It will detect the faulty component and disable it. The control system is redundant and will continue to perform its functions, and will restore steady state operation.

The magnitude of the difference between the transmitters output are maintained low so that this disturbance is mild, similar to a pressure setpoint change and no protective actions or significant reductions in fuel thermal margins occur. This transient is much less severe than the generator or turbine trip transients previously discussed.

This event was initially considered in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.2 Events Resulting in a Reactor Vessel Water Temperature Decrease

The SAR categorized the following events as those that result directly in a reactor vessel water temperature decrease due to either an increase in the flow of cold water to the vessel or a reduction in the temperature of the water being delivered to the vessel:

a. Loss of feedwater heating

b. Shutdown cooling malfunction - decreasing temperature

c. Inadvertent Start of HPCI Pump

5.2.1 Loss of Feedwater Heating

A decrease in feedwater temperature due to loss of feedwater heating would result in a core power increase due to the increase in core inlet subcooling and the reactivity effects of the corresponding increase in moderator density. The loss of feedwater heating transient is identified in Reference 49 to be one of the events that is most likely to limit operation because of MCPR consideration. As a result, this event is analyzed as a licensing basis transient for this fuel reload cycle.

The core-wide Abnormal/Anticipated Operational Occurrence analyses for MELLL and ICF are performed for each reload cycle. These transient events include loss of feedwater heating (LFWH). This analysis is documented in Reference 35.

5.2.1.1 Identification of Causes

Feedwater heating can be lost in at least two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater due to the stored heat capacity of the heater. In the second case, the feedwater bypasses the heater and the change in heating occurs during the stroke time of the bypass valve (about one minute, similar to the heater time constant). In either case, the reactor vessel receives cooler feedwater.

5.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being classified as an infrequent incident. However, because of the lack of a sufficient frequency database, this transient disturbance is analyzed as an incident of moderate frequency.

5.2.1.3 Starting Conditions and Assumptions

The parameter values used in the analysis are presented in Table XIV-5-2. The following additional plant operating conditions and assumptions form the principal basis for analysis of the loss of feedwater heating transient:

a. The plant is operating at rated power. This transient is less severe from lower initial power levels for two main reasons: (1) lower initial power levels will have initial MCPR values greater than the operating limit MCPR, and (2) the magnitude of the power rise decreases with lower initial power conditions.

b. The plant is operating in the manual flow control mode.

5.2.1.4 Sequence of Events and Systems Operation

The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations and the feedwater heaters are assumed to trip instantaneously. For Cooper, the assumed reduction in feedwater temperature for this transient is 100°F. Loss of feedwater heating results in a core power increase due to the increase in core inlet subcooling and the reactivity effects of the resulting increase in moderator density.

5.2.1.5 Core and System Performance

For reload cores, an evaluation is performed to determine if the loss of feedwater heating could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the Supplemental Reload Licensing Report.

Plant/cycle-specific analyses for Cooper Nuclear Station using the steady-state 3-D BWR Simulator Code as described in References 48 and 49 were reported in Reference 35 and are summarized in Table XIV-7-3. The cycle-specific reload analysis demonstrates that MCPR would not exceed the MCPR safety limit. The transient plots, flux, and Q/A normally reported in the Supplemental Reload Licensing Report are not outputs of the BWR Simulator code.

5.2.2 Shutdown Cooling Malfunction - Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHRS heat exchangers. At design power conditions no conceivable means of a malfunction in the shutdown cooling system causing temperature reduction is possible with a single failure event.

If the reactor were critical or near critical, the resulting temperature decrease causes a slow insertion of positive reactivity into the core and a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron flux reactor scram would terminate the transient without fuel damage and without any measurable nuclear system pressure increase.

This event was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.2.3 Inadvertent Start of HPCI Pump

5.2.3.1 Identification of Causes

Several systems are available for providing high pressure supplies of cold water to the vessel for normal or emergency functions. The control rod drive system and the makeup water system, normally in operation, can be postulated to fail in the high-flow direction introducing the possibility of increased power due to higher core inlet subcooling. The same type of transient would be produced by the inadvertent startup of the RCIC System or the HPCI System. The severity of the resulting transient is highest for the largest of these sources of cold water injection, the HPCI System.

5.2.3.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.2.3.3 Starting Conditions and Assumptions

The inadvertent start of the HPCI pump is evaluated on a cycle specific basis as one of the potentially limiting events. If it is determined that Level 8 turbine trip would not occur, then this event is evaluated against the subcooling change for the loss of feedwater heating event. The plant operating conditions and assumptions are identical to those for loss of feedwater heating. If it can not be determined that a Level 8 turbine trip is avoided for an inadvertent start of a HPCI pump, then this analysis is performed with the NRC approved ODYN code. This event with a Level 8 turbine trip is then very much like the feedwater controller failure with an initial subcooling portion, followed by a turbine trip.

5.2.3.4 Sequence of Events and Systems Operation

Figures XIV-5-28a-n show the transient simulation. The HPCI System introduces cold water through the feedwater sparger. The normal feedwater flow is correspondingly reduced by the water level controls. The increase in inlet subcooling due to the inadvertent HPCI start is slightly less than that produced by the loss of feedwater heating event. This event is very similar to a loss of feedwater heating if a Level 8 trip does not occur. The operator actions responding to this event would be similar to those for loss of feedwater heating. If the feedwater level controller can not respond fast enough to prevent excessive level, then a Level 8 turbine trip may occur. In addition, the operator should determine the reason why the HPCI flow was initiated and follow proper procedures to shut off the pumps.

5.2.3.5 Core and System Performance

For reload cores, an evaluation is performed to determine if this event could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Should this event not be bounded by the loss of feedwater heating event, the cycle-specific results are determined using the NRC approved ODYN model.

5.3 Events Resulting in a Positive Reactivity Insertion

The SAR categorized the following events as those that result directly in positive reactivity insertions as the result of rod withdrawal errors and errors during refueling operations:

a. Continuous rod withdrawal during power range operation.

b. Continuous rod withdrawal during reactor startup.

c. Control rod removal error during refueling.

d. Fuel assembly insertion error during refueling.

5.3.1 Continuous Rod Withdrawal During Power Range Operation

The analysis for this event considered power levels from the Rod Worth Minimizer (RWM) low power setpoint (10 percent of rated) to rated power. The consequences of this transient are relatively mild and neither localized nor gross core damage will occur. In the worst case situation, protective action from the Rod Block Monitor is required to prevent violating the safety limit MCPR.

5.3.1.1 Identification of Causes

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its rod block position. Due to the positive reactivity insertion, the core average power increases. More importantly, the local power in the vicinity of the withdrawn control rod increases and could potentially cause cladding damage due to overheating, which may accompany the occurrence of boiling transition, which is an assumed abnormal operational transient failure threshold.

5.3.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being classified as an infrequent incident. However, because of the lack of a sufficient frequency database, this transient disturbance is analyzed as an incident of moderate frequency.

5.3.1.3 Starting Conditions and Assumptions

The following assumptions form the principal bases for analysis of the continuous rod withdrawal during power range operation transient:

a. The reactor is operating at a power level above 30 percent of rated power at the time the control rod withdrawal error occurs.

b. The reactor operator has followed procedures and up to the point of the withdrawal error is in a normal mode of operation (i.e., the control rod pattern, flow set points, etc., are all within normal operating limits).

c. For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion will occur.

5.3.1.4 Sequence of Events and Systems Operation

The following list depicts the sequence of events for this transient:

a. Event begins, operator selects the maximum worth control rod, acknowledges any alarms and withdraws the rod at the maximum rod speed.

b. Core average power and local power increase causing local power range monitor (LPRM) alarm.

c. Event ends - rod block by Rod Block Monitor (RBM).

5.3.1.5 Core and System Performance

Rod withdrawal error (RWE) is analyzed in Reference 53. Implementation of the ARTS program modified the RBM from flow-biased to power-dependent trips to allow the use of a generic non-limiting analysis for the rod withdrawal error to determine the MCPR requirements and the corresponding RBM setpoints. The generic ARTS RWE analysis is a statistical evaluation of randomly occurring realistic RWE conditions. The analysis is applicable for fuel designs through GE8/8B. Applicability of the analysis to later fuel designs will be determined on a cycle specific basis until a sufficient database is established to determine generic application.

The generic RWE analysis results in the calculation of bounding values of ΔCPR as a function of rod block monitor setpoint. These values are reported in the Supplemental Reload Licensing Report for each fuel type. The ΔCPRs are conservative relative to the actual operating limit MCPR and are valid throughout the cycle. Plant/cycle-specific analyses for Cooper Nuclear Station were reported in Reference 35 and are summarized in Table XIV-7-3. The cycle-specific reload analysis demonstrates that MCPR would not exceed the safety limit MCPR.

5.3.2 Continuous Rod Withdrawal During Reactor Startup

This transient was initially analyzed in the SAR. It has not been reanalyzed for any of the subsequent reload cycles because it is not considered a credible event.

In addition, this event does not meet the single equipment failure or single operator error criteria for abnormal operational transients. This event is contingent upon the failure of the Rod Worth Minimizer (RWM) System, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator nonacknowledgement of continuous alarm annunciations prior to safety system actuation.

5.3.2.1 Identification of Causes

While performing a reactor startup, the reactor operator makes a procedural error and fully withdraws the maximum worth control rod. Due to the positive reactivity insertion, the core average power increases. More importantly, the local power in the vicinity of the withdrawn control rod increases and could potentially cause cladding damage due to overheating, which may accompany the occurrence of boiling transition, which is an assumed abnormal operational transient failure threshold.

5.3.2.2 Frequency Classification

The probability of this event is considered low enough to warrant its being classified as an infrequent incident. However, because of the lack of a sufficient frequency database, this transient disturbance is analyzed as an incident of moderate frequency.

5.3.2.3 Starting Conditions and Assumptions

The following plant operating conditions and assumptions form the principal bases for analysis of the continuous rod withdrawal during reactor startup transient:

a. The reactor is just subcritical with reactor power at an initial value of 10^{-8} percent of rated (equivalent to 3 cps in the Source Range).

b. The average moderator temperature is 82°F.

c. The Rod Worth Minimizer (RWM) is bypassed.

d. Prior to the error, control rods have been withdrawn in sequence and are in a pattern consistent with 75 percent rod density.

e. The IRM channel in each RPS trip system that monitors the core area closest to the rod withdrawal error is bypassed.

5.3.2.4 Sequence of Events and Systems Operation

A continuous rod withdrawal during startup produces the following sequence of events:

a. The operator selects an out-of-sequence rod. The rod is assumed to be adjacent to the last rod withdrawn in sequence. The location of this rod was selected to maximize the distance to the second nearest IRM detector assigned to each RPS trip system.

b. The error rod is fully withdrawn at the maximum rod drive speed of 3 inches/second and adds 0.025 $\Delta K/K$ reactivity.

c. A scram signal is initiated when one IRM detector in each RPS trip system reaches 120/125 of full scale.

5.3.2.5 Core and System Performance

At the time the scram occurs the peak flux in the core is 2.7 percent rated average flux. The core average power is 0.07 percent when scram occurs. Therefore, MCPR is maintained above the safety limit MCPR. No fuel damage is calculated to occur due to a continuous rod withdrawal during reactor startup.

5.3.3 Control Rod Removal Error During Refueling

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The nuclear characteristics of the core assure that the reactor is subcritical even in its most reactive condition with the most reactive control rod fully withdrawn during refueling.

As described in USAR Section VII-6.3, when the reactor mode switch is in REFUEL, only one control rod can be withdrawn. Selection of a second rod for movement with any other rod withdrawn initiates a rod block, thereby preventing the withdrawal of more than one rod at a time. Therefore, the Refueling Interlocks prevent any condition which could lead to inadvertent criticality due to a control rod withdrawal error during refueling.

In addition, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles, thus eliminating any hazardous condition.

Therefore, there is no postulated set of circumstances which results in an inadvertent rod removal error while in the REFUEL mode based on single equipment failure or single operator error under the assumption that refueling interlocks are not bypassed.

5.3.4 Fuel Assembly Insertion Error During Refueling

The event considered here is inadvertent criticality due to the possibility of loading fuel into a cell containing no control rod. The core is designed such that it can be made subcritical under the most reactive conditions with the strongest control rod fully withdrawn. Therefore, any single fuel bundle can be positioned in any available location without violating the shutdown criteria, providing all control rods are fully inserted.

As described in USAR Section VII-6.3, the Refueling Interlocks require that all control rods must be fully inserted before the refueling platform or service platform can be used to insert a fuel bundle into the core.

Therefore, there is no postulated set of circumstances which results in an inadvertent fuel assembly insertion error while in the REFUEL mode based on single equipment failure or single operator error under the assumption that refueling interlocks are not bypassed.

5.3.5 Loading Error

Fuel loading error was classified as an infrequent incident under Amendment 28 to GESTAR II provided that the plant confirms the requirements for application of the generic analysis that supported the amendment.

5.3.5.1 Starting Conditions and Assumptions

Proper location and orientation of the fuel assemblies in the reactor core is monitored during fuel movements and verified by procedures during core loading. Verification procedures address location, orientation, and seating through visual examinations of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated or misoriented bundle event. Plant operation with a mislocated or misoriented fuel bundle is a result of a failure in the core verification process following core refueling.

5.3.5.2 Event Description

The loading error involves either mislocation or misorientation. For mislocation, at least two fuel bundles are assumed to be mislocated. One location is loaded with a bundle that would potentially operate at a lower critical power than it would otherwise. The other location would operate at a higher critical power. The low critical power location could have less margin to boiling transition than other bundles in the core.

For misorientation, the loading error involves the misorientation of a single fuel bundle. The power distribution in the misoriented bundle would be affected as well as its neighbors. The resulting power distribution could reduce the margin to boiling transition.

For the infrequent incident classification, it is assumed that the loading event proceeds to cause fuel failures.

5.3.5.3 Identification of Operator Actions

There is a possibility that core monitoring will provide information that allows the operator or reactor engineer to recognize that an error exists and determine appropriate mitigating actions.

Where the high radial power mislocated bundle is adjacent to an instrument, the power adjustment in radially TIP or LPRM adapting monitoring systems will cause higher monitored bundle power. The reactor will be operated such that the most limiting of the bundles near the mislocation will be maintained below the operating limit MCPR. Where the mislocated bundle has a bundle between it and the instrument, the core monitoring may not recognize the mislocation.

If loading errors were made and have gone undetected, the operator would assume that the mislocated bundle would operate at the same power as the instrumented bundle in the mirror-image location and would operate the plant until end of cycle.

If misoriented loading errors were made and have gone undetected, the plant would also continue to operate until end of cycle.

Should fuel failure occur, the offgas activity quickly increases. At that point, the operator would take steps to reduce power or scram the reactor to reduce or terminate the release.

5.3.5.4 Results and Consequences

The generic analysis provides a bounding analysis based on a very conservative assumption of all of the fuel rods failing in five fuel bundles. Two scenarios for the fuel loading error were considered. The first assumed that the fission product activity is airborne in the turbine and condenser following Main Steam Isolation Valve (MSIV) closure and leaks directly from the condenser to the atmosphere. In the second scenario, it was assumed that no automatic MSIV closure occurred and that the activity was transported to an augmented offgas system.

Calculations of post-accident doses for the Exclusion Area Boundary (EAB) were performed for each scenario to compare radiological consequences with the applicable exposure limits. EAB doses were also calculated for both scenarios utilitizing the alternate source term methodology.

The plant-specific offgas system parameters and site atmospheric dispersion parameters are used to confirm the applicability of the EAB generic analysis. A conservative analysis for the control room dose was also established such that plant specific atmospheric dispersion parameters can also be used to confirm its applicability. Some items from the generic analysis must be confirmed and documented with the reload design documentation to support application of this infrequent incident option.

5.4 Events Resulting in a Reactor Vessel Coolant Inventory Decrease

The SAR categorized the following events as those that result directly in a decrease in reactor vessel coolant inventory either by restricting the normal flow of fluid into the vessel or by increasing the removal of fluid from the vessel:

a. Digital Electro-Hydraulic (DEH) pressure controller output fails high.

b. Inadvertent opening of a safety/relief valve.

c. Loss of feedwater flow.

d. Loss of auxiliary power.

5.4.1 DEH Pressure Controller Output Fails High

The pressure controller output fails high transient results in a loss of pressure control followed by an increase in steam flow demand to the main turbine that causes rapid depressurization in the vessel.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.4.1.1 Identification of Causes

A triple redundant DEH control system is provided to maintain primary system pressure control. The pressure upstream of the main turbine stop valves is sensed by three redundant throttle pressure transmitters at the equalizing header and the control system uses a median select logic to determine which pressure transmitter is used to control throttle pressure. The pressure control system compares the detected throttle pressure to a pressure setpoint to control the position of the main turbine control (governor) and bypass valves in order to control pressure. If all three pressure transmitters are detected healthy by the main processors, the selected transmitter is equal to the median (middle) value. If only two pressure inputs are detected healthy, the selected output is equal to the highest input. If one pressure input is good, it is used.

If the triple redundant DEH control system fails, the governor valves followed by the bypass valves open until limited by the maximum total flow limit setting for the DEH control system. This potential reactor depressurization threatens to impose serious stresses on the nuclear system.

5.4.1.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.4.1.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the DEH pressure controller output fails high transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating with the recirculation system in the manual flow control mode.

d. The normal maximum total flow limit setting for the DEH control system is 110 percent.

e. The low pressure Main Steam Isolation Valve closure setpoint is 850 psig.

f. It was conservatively assumed that feedwater flow was terminated within 5 seconds following MSIV closure.

5.4.1.4 Sequence of Events and Systems Operation

Figure XIV-5-8 graphically shows the changes of important nuclear system variables during the simulated pressure controller failure initiated at 105 percent of rated power conditions. The controller failure causes the following sequence of events:

a. A controller failure to 115 percent steam flow demand was simulated as a worst case since 110 percent is the normal maximum flow limit.

b. The rapid formation of voids in the core reduces reactor power quickly.

c. Main Steam Isolation Valve (MSIV) closure is initiated at about 15 seconds.

d. A scram is initiated on MSIV closure by valve position switches before the valves have traveled more than 10 percent from the open position.

e. Steam line and vessel pressures drop slightly over 100 psi before the isolation becomes effective near 18 seconds.

f. The MSIV closure results in the complete loss of feedwater flow by approximately 22 seconds.

g. Reactor vessel water level initiates RCIC, HPCI, and Recirculation Pump trip at approximately 22 seconds.

h. Following isolation, safety relief valves cycle to dissipate the stored and decay heat.

5.4.1.5 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-8, the rapid formation of voids in the core reduces reactor power and heat flux quickly. No reduction in fuel thermal margins occur.

5.4.1.6 Barrier Performance

The MSIV closure limits the duration of the depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier. Reference 54 addresses the change in the Main Steam Line isolation signal from 850 psig to 825 psig: "The 850 psig low pressure isolation was originally determined based on judgment and was chosen approximately 100 psi less than the turbine inlet pressure. The 100 psi number is not critical and a larger value would result in only small changes in the effects on saturation temperature and fuel duty (the difference in saturation temperature between 850 psig and 750 psig is approximately 15°F)". It was concluded that lowering the setpoint would result in a negligible added requirement in terms of fuel duty and vessel cooldown and would not invalidate the transient safety analyses.

5.4.2 Inadvertent Opening of a Safety/Relief Valve

5.4.2.1 Identification of Causes

An inadvertent Safety/Relief Valve opening transient resulting from the valve opening and then reclosing has no significant effect on the plant. When the valve opens and remains stuck in the open position, a mild depressurization transient is introduced.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is affected by the fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.4.2.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.4.2.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the inadvertent opening of a safety/relief valve transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating with the recirculation system in the manual flow control mode.

d. The capacity of the safety/relief valve is 7.6 percent of rated steam flow.

5.4.2.4 Sequence of Events and Systems Operation

Figure XIV-5-9 graphically shows the changes of important nuclear system variables during the simulated inadvertent opening of a 7.6 percent capacity safety/relief valve from 105 percent of rated power conditions. The inadvertent opening of a safety/relief valve causes the following sequence of events:

a. The opening of a safety/relief valve allows steam to be discharged to the suppression pool.

b. The sudden increase in the rate of steam flow leaving the reactor vessel causes reactor vessel and steam line pressures to decrease.

c. The turbine pressure controller senses the pressure decrease and closes the turbine governor valves far enough to stabilize reactor vessel pressure at a slightly lower value.

d. The reactor settles out at nearly the initial power level.

5.4.2.5 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-9, thermal margins do not change significantly during the transient.

5.4.3 Loss of Feedwater Flow

5.4.3.1 Identification of Causes

A loss of feedwater flow may occur as a result of feedwater pump failures, condensate pump failures, feedwater controller failures, operator errors, or trip on reactor high water. Loss of auxiliary power (see USAR Section XIV-5.4.4) also produces a loss of feedwater along with the loss of many other plant functions.

The loss of feedwater flow event is a mild transient with respect to maintaining adequate fuel thermal margins and reactor vessel pressure margins. However, the loss of feedwater flow event is the most challenging abnormal operational transient with respect to coolant inventory control since it results in the most rapid reactor coolant inventory loss.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles. Additional transient analysis results for the loss of feedwater flow event, performed in support of lowering the reactor water level scram setpoint, are discussed in USAR Section XIV-5.4.3.6. Similar analysis results for lowering the reactor vessel water level for MSIV closure is discussed in USAR Section XIV-5.4.3.7.

5.4.3.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.4.3.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the loss of feedwater flow transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating with the recirculation system in the manual flow control mode.

d. Feedwater flow coastdown time is conservatively assumed to be 5 seconds.

5.4.3.4 Sequence of Events and Systems Operation

Figure XIV-5-10 graphically shows the changes of important nuclear system variables during the simulated loss of feedwater flow from an initial 105 percent of rated power condition. The loss of feedwater flow causes the following sequence of events:

a. The transient is initiated by the simultaneous trip of all the feedwater pumps. Feedwater flow drops to zero over a 5-second period.

b. The reduction in feedwater flow causes water level to drop at a rate of up to 6 inches/second.

c. The reduced injection of relatively cold feedwater lowers core inlet subcooling, increases core average void fraction, and reduces reactor power at an initial rate of about 5 percent per second.

d. A reactor vessel low water level scram shuts down the reactor at about 8 seconds.

e. Reactor vessel water level reaches the Level 2 setpoint at about 16 seconds. RCIC/HPCI initiation and MSIV closure occur at Level 2. (MSIV closure was changed to Level 1 in License Amendment 83. See discussion in Section XIV-5.3.4.7.)

f. Reactor pressure, which initially decreased with power, will rise following MSIV closure due to decay heat. Relief valve operation will control system pressure.

5.4.3.5 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in

Figure XIV-5-10, the reduction in core inlet subcooling and increased void fraction caused by the loss of feedwater flow results in an initial power reduction while core flow remains constant. The reactor is shutdown before core flow is reduced. Therefore, the fuel thermal margins are not threatened during the transient.

5.4.3.6 Effect of Reducing Reactor Water Level 3 Setpoint

As shown in Table XIV-7-1, the loss of feedwater flow is the only transient event that requires a reactor scram initiation on low reactor water level. Therefore, an additional transient analysis for loss of feedwater flow was performed in support of lowering the reactor water Level 3 scram and isolation setpoint. The results of this analysis are reported in Reference 42. The Level 3 setpoint was lowered to provide extra margin between the level at which a controlled manual scram is initiated and the level at which the Level 3 functions occur. The lower Level 3 setpoint was initiated with Cycle 15.

The analysis was performed using GE Nuclear Energy's 10CFR50 Appendix K model (SAFE/REFLOOD/CHASTE) to verify that the RCIC system will still perform its intended function of maintaining the reactor water level above Level 1 for this event with the following assumptions:

a. Conservative decay heat values (1971ANS + 20 percent) are used.

b. The initial reactor power corresponds to 105 percent of rated steam flow.

c. The initial water level in the reactor is at the normal water level.

d. The feedwater pumps coast down in 1 second.

e. The reactor scrams at the Level 3 setpoint.

f. Only RCIC initiates at Level 2.

The results of the analysis show that the lower Level 3 setpoint delays the time of scram by approximately 2.5 seconds for the loss of feedwater flow event. However, the RCIC system is still able to maintain the reactor water level above Level 1.

5.4.3.7 Effect of Changing to Reactor Water Level 1 for MSIV Closure

In the Loss of Feedwater event, the reactor water level decreases quickly causing a reactor low level scram at Level 3. After scram, the reactor water level continues to drop until it reaches Level 2 where HPCI and RCIC initiate. The scenario for the previous analyses proceeds with the MSIVs isolation on Level 2. This allows RCIC to maintain core cooling and some SRV actuations after isolation. With the MSIV water level trip lowered to Level 1, the reactor is not isolated while HPCI or RCIC is operating. Thus, the lower MSIV water level poses a different challenge to the HPCI and RCIC systems and a further evaluation is required.

This analysis was performed using GE Nuclear Energy's 10CFR50 Appendix K model (SAFE/REFLOOD/CHASTE) to verify that the RCIC system will still perform its intended function of maintaining the reactor water level above Level 1 for this event with the following assumptions:

a. Conservative decay heat values (1973ANS + 20 percent) are used to maximize the heat addition to the vessel, SRV challenges, and inventory loss.

b. The initial reactor power corresponds to 102 percent of licensed power, which also maximizes the above parameters.

c. The initial water level in the reactor is assumed to be at the scram level (Level 3) and the reactor is scrammed at time zero. This is consistent with the 10CFR50, Appendix K LOCA analysis.

d. The feedwater pumps coast down in 1 second. This is also consistent with the 10CFR50, Appendix K LOCA analysis.

e. Only RCIC initiates at Level 2. Since the HPCI injection rate is 10 times that of RCIC, this assumption provides the most severe challenge to the reactor cooling.

The results of the analysis show that RCIC is capable of providing adequate core cooling even if the MSIV water level is lowered to Level 1. The RCIC flow is sufficient to compensate for the steam flow through the turbine control valves to the main condenser. It also maintains the reactor water level above Level 1. Since the water level remains above Level 1 throughout the event, the reactor remains unisolated. The turbine control valve maintains the reactor pressure at approximately 950 psig and precludes any SRV or low level set release (LLS) actuation.

5.4.4 <u>Loss of Auxiliary Power</u>

5.4.4.1 <u>Identification of Causes</u>

Loss of Auxiliary Power is defined as an event which de-energizes all buses that supply power to the unit auxiliary equipment such as recirculation pumps, condensate pumps, and circulating water pumps. Two methods of experiencing this event are postulated:

a. Trip(s) or fault(s) occurring in the auxiliary power distribution system itself without transfer to outside sources.

b. Complete loss of all external connections to the grid.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.4.4.2 <u>Frequency Classification</u>

This event is classified as an incident of moderate frequency.

5.4.4.3 <u>Starting Conditions and Assumptions</u>

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the loss of auxiliary power transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

5.4.4.4 Sequence of Events and Systems Operation

Estimates of the responses of the various reactor systems provided the following simulation sequence for the transient initiated due to trips or faults occurring in the auxiliary power system:

a. All pumps supplied by auxiliary power trip at t=0. Normal coastdown times were used for the recirculation and feedwater pumps.

b. The Reactor Protection System MG sets were assumed to coastdown in 5 seconds to the point where scram and main steam isolation occurred.

c. Loss of the main condenser circulating water pumps was estimated to cause condenser vacuum to drop to the turbine trip setting by 6 seconds. The short period (about 2 seconds) during which bypass flow would have been permitted was neglected.

Figure XIV-5-11 graphically shows the changes of important nuclear system variables during the simulated loss of auxiliary power from an initial 105 percent of rated power condition. The initial portion of the transient is very similar to the simple loss of all feedwater (USAR Section XIV-5.4.3). Initiation of scram, isolation valve closure, and turbine trip all occur between 5 to 6 seconds and the transient changes to that of an isolation. The relief valves open for a short time then reclose as the remainder of the stored heat is dissipated.

An alternate transient results if complete connection with the external grid is lost at time = 0. The same sequence as above would be followed except that the reactor would also experience a generator load rejection and its associated scram at the beginning of the transient. Figure XIV-5-12 shows this simulated loss of auxiliary power event.

5.4.4.5 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-11, there is no significant increase in fuel temperature or surface heat flux during the transient. Nucleate boiling is maintained throughout the transient (MCHFR remains above 1.7). While Figure XIV-5-12 shows a small peak in neutron flux as a result of the load rejection, it is limited by scram and the recirculation pump trips. No increase in fuel surface heat flux occurs. No fuel damage occurs in either case.

5.5 Events Resulting in a Core Coolant Flow Decrease

Coolant flow into the core is of primary importance in reactor performance. Events which produce reductions in flow reduce the effectiveness of heat transfer from the fuel. Therefore, the ability of the fuel clad barrier to withstand the transients must be evaluated.

The SAR categorized the following events as those that affect the reactor recirculation system and result directly in a core coolant flow decrease:

a. Recirculation flow control failure - decreasing flow.

b. Trip of one recirculation pump.

c. Trip of two recirculation MG set drive motors.

d. Recirculation pump seizure.

The currently available operating plant history data for the frequency of occurrence of events indicates that the recirculation pump seizure event should be classified as a design basis (postulated) accident (limiting fault). However, the results of the recirculation pump seizure event are presented in this section, because this event is analyzed as an Abnormal Operational Transient as part of the reload analysis.

5.5.1 Recirculation Flow Control Failure - Decreasing Flow

5.5.1.1 Identification of Causes

Several possible flow controller malfunctions can cause a decrease in core coolant flow.

a. A malfunction of both flow controllers could call for zero speed for both recirculation system MG sets. However, rate limits included in the individual pump speed controllers are set so that this failure cannot produce a transient more severe than the trip of both drive motors.

b. A failure in a MG set controller could cause the scoop tube positioner for the fluid coupler to move at its maximum speed in the direction of zero pump speed and flow. The transient is very similar to the trip of one recirculation pump. However, the pump speed reduction is slower than simply opening a MG set generator field breaker.

This second event, a failure which involves a single MG set speed controller, was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.5.1.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.5.1.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the single recirculation flow controller failure - decreasing flow transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating in the manual flow control mode.

d. The scoop tube positioner associated with the failed flow controller moves towards minimum speed at its maximum rate assumed to be 25 percent/second.

5.5.1.4 Sequence of Events and Systems Operation

Figure XIV-5-13 graphically shows the changes of important nuclear system variables during the simulated single recirculation flow control failure with decreasing flow event.

The sequence of events for this transient is similar to, and can never be more than, that listed in USAR Section XIV-5.5.2.4 for the trip of one recirculation pump.

5.5.1.5 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-13, there is no increase in fuel temperature or surface heat flux during the transient. Nucleate boiling is maintained throughout the transient (MCHFR is greater than 1.5). This transient is less severe than the trip of one recirculation pump event.

5.5.2 Recirculation Pump Trips

This section covers the transients in which one or both recirculation pumps are independently de-energized. Complete loss of plant auxiliary power, which causes pump trip plus other simultaneous events, is presented in USAR Section XIV-5.4.4. Of chief concern are the fuel thermal margins which are experienced throughout these transients. An abrupt reduction in core flow increases the core void fraction and thereby decreases power. The fuel surface heat flux decreases more slowly than the flow because of the fuel time constant, so thermal margins momentarily decrease. The reactor returns to the steady-state power/flow characteristic with thermal margins greater than the initial high power condition. The inertias of the pumping system are chosen to provide acceptably slow flow reductions for all pump trip possibilities.

The transients in this section were initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.5.2.1 Identification of Causes

Recirculation pump motor operation can be tripped off by design for intended objectives as well as randomly by unpredictable operational failures.

Some of the conditions that will produce a recirculation pump trip by design for intended objectives are: reactor vessel water Level 2 (ATWS-RPT) trip, reactor vessel high pressure (ATWS-RPT) trip, recirculation loop suction/discharge valves not fully open, MG set generator lockout, MG set high air temperature, MG set low lube oil pressure, and MG set high fluid drive oil temperature.

Unpredictable operational failures include: operator error, loss of electrical power source to the pumps, equipment or sensor failures and malfunctions which initiate the above intended trip response.

5.5.2.2 Frequency Classification

These events are classified as incidents of moderate frequency.

5.5.2.3 Starting Conditions and Assumptions

The parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the recirculation pump trip transients:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating in the manual flow control mode.

5.5.2.4 Trip of One Recirculation Pump

Normal trip of one recirculation loop is accomplished through the drive motor breaker. However, a worse coastdown transient occurs if the generator field excitation breaker is opened, separating the pump and its motor from the inertia of the MG set. The instantaneous collapse of the generator field was assumed.

5.5.2.4.1 Sequence of Events and Systems Operation

Figure XIV-5-14 graphically shows the changes of important nuclear system variables during the simulated single recirculation pump trip initiated from 105 percent rated power conditions. The recirculation pump trip causes the following sequence of events:

a. Trip of one recirculation pump is initiated by opening the generator field breaker.

b. The core flow decrease lowers the core pressure drop and the flow through the operating loop jet pumps increases.

c. Jet pump flows on the loop with the tripped recirculation pump reverse at about 4 seconds.

d. Conditions stabilize with the operating loop jet pumps providing about 136 percent of their normal flow with core flow at about 58 percent of rated.

5.5.2.4.2 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-14, there is essentially no increase in fuel temperature or surface heat flux during the transient. Nucleate boiling is maintained throughout the transient (MCHFR remains above 1.3).

5.5.2.5 Trip of Two Recirculation Pumps

The two-loop trip provides the evaluation of the fuel thermal margins maintained by the rotating inertia of the recirculation drive equipment. No single operator act or equipment malfunction can produce simultaneous trip of the generator field breakers for both MG sets. Plant protection action can, however, simultaneously trip either the MG set drive motors or the generator field breakers. The trip of both generator field breakers in response to an ATWS condition is discussed in USAR Section IV-3.5.

5.5.2.5.1 Sequence of Events and Systems Operation

Figure XIV-5-15 graphically shows the changes of important nuclear system variables during the simulated trip of both recirculation pumps initiated from 105 percent rated power conditions. No scram is initiated directly by the simultaneous pump trip and the power will settle out at part-load, natural circulation conditions.

5.5.2.5.2 Core and System Performance

This transient was analyzed using the REDY transient model[50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (0.95, 1.05, and 0.8 respectively). As shown in Figure XIV-5-15, there is essentially no increase in fuel temperature or surface heat flux during the transient. Nucleate boiling is maintained throughout the transient (MCHFR = 1.24) and no fuel damage occurs.

5.5.3 One Recirculation Pump Seizure

Even though the one recirculation pump seizure is classified as a design basis accident (limiting fault) based on its frequency of occurrence, it is analyzed as an Abnormal Operational Transient.

5.5.3.1 Two Loop Operation

This event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power.

The pump seizure event is very mild in relation to other accidents such as the LOCA. This is easily verified by consideration of the two events. In both accidents, the recirculation driving loop flow is lost extremely rapidly - in the case of the seizure, stoppage of the pump occurs; for the LOCA, the severance of the line has a similar, but more rapid and severe influence. Following a pump seizure event, flow continues, water level is maintained, the core remains submerged, and this provides a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases due to loss of coolant resulting in uncovery of the reactor core and subsequent overheating of the fuel rod cladding. In addition, for the pump seizure event, reactor pressure does not significantly decrease, whereas complete depressurization occurs for the LOCA. Clearly, the increased temperature of the cladding and reduced reactor pressure for the LOCA both combine to yield a much more severe stress and potential for cladding perforation for the LOCA than for the pump seizure. Therefore, it can be concluded that the potential effects of the hypothetical pump seizure accident are very conservatively bounded by the effects of a LOCA and specific analyses of the pump seizure accident are not required.

5.5.3.2 Single Loop Operation

The one-pump seizure event is also a relatively mild event during single recirculation pump operation[36][37]. Analyses were performed for CNS to determine the impact this accident would have on one recirculation pump operation. Figure XIV-5-15a shows key system parameters during this event. The analyses were initialized from steady-state operation at the following initial conditions, with the added condition of one inactive recirculation loop. The following initial conditions were assumed:

Thermal Power = 68.5% and core flow = 57.1%

These conditions were chosen because they bound the SLO operating domain.

The anticipated sequence of events following a recirculation pump seizure which occurs during plant operation with the alternate recirculation loop out of service is:

(1) The recirculation loop flow in the loop in which the pump seizure occurs drops rapidly.

(2) Core voids increase which results in a negative reactivity insertion and a sharp reduction in neutron flux and heat flux.

(3) Key parameters settle to new steady-state within 6 seconds.

(4) Neutron flux, heat flux, reactor water level, steam flow, and feedwater flow all exhibit transient behaviors. However, it is not anticipated that the increase in water level will cause a turbine trip and result in scram.

The transient will terminate at a condition of natural circulation and reactor operation will continue with a small decrease in system pressure.

The rated equivalent Operating Limit MCPR is documented in the cycle specific Supplemental Reload Licensing Report in Reference 35. This OLMCPR value establishes the minimum CPR for the pump seizure accident for CNS to ensure that the fuel cladding integrity safety limit is not exceeded; therefore, no fuel failures are expected to occur as a result of this analyzed event.

A one recirculation pump seizure event during single loop operation could place reactor operation in the region of potential thermal-hydraulic instability. Therefore, plant procedures require reactor shutdown (scram) if neither recirculation pump is in operation while above 1 percent of rated thermal power. Subsequent operator actions would be similar to any of those previously identified for transients which result in automatic scrams.

5.6 Events Resulting in a Core Coolant Flow Increase

Coolant flow into the core is of primary importance in reactor performance. Events which produce fast increases in flow will result in an increase in reactor power. Therefore, the ability of the fuel clad barrier to withstand the transients must be evaluated.

The SAR categorized the following events as those that affect the reactor recirculation system and result directly in a core coolant flow increase:

a. Recirculation flow control failure - increasing flow.

b. Startup of idle recirculation loop.

5.6.1 Recirculation Flow Control Failure - Increasing Flow

5.6.1.1 Identification of Causes

Controller malfunctions could result in maximum pump speed. The most severe case is the failure of one of the MG set speed controllers since the speed controller rate limits are adjusted to keep multiple controller failure less severe.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

5.6.1.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.6.1.3 Starting Conditions and Assumptions

The most severe case occurs from the lowest initial power/flow conditions along the flow control line. Initial reactor conditions of 47 percent of rated core flow and 65 percent of rated power were selected to bound the expected normal operating conditions. Other parameter values and the available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the single recirculation flow controller failure - increasing flow transient:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The scoop tube positioner associated with the failed flow controller moves towards maximum speed at its maximum rate assumed to be 25 percent/second.

5.6.1.4 Sequence of Events and Systems Operation

Figure XIV-5-16 graphically shows the changes of important nuclear system variables during the simulated single recirculation pump flow control failure - increasing flow event initiated from 65 percent rated power conditions. The flow controller failure causes the following sequence of events:

a. The flow control failure produces rapidly increasing recirculation loop flow and core flow.

b. The rapid increase in core flow reduces void content and causes an increase in neutron flux.

c. An APRM high flux scram shuts down the reactor.

5.6.1.5 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). As shown in Figure XIV-5-16, the transient fuel surface heat flux reached 81 percent of rated conditions, but it barely exceeded the steady-state flow control power/flow characteristic. Nucleate boiling is maintained throughout the transient (MCHFR remained above 2.5) and fuel center temperature increased only 267°F. Therefore, no fuel barrier damage occurs.

5.6.2 Startup of Idle Recirculation Pump

5.6.2.1 Identification of Causes

This event considers an improper startup of an idle recirculation pump (assuming the other recirculation loop is operating) without first warming the idle loop. The increase in core flow and introduction of colder water will reduce the void content in the core. The void reactivity causes the neutron flux to increase.

This transient was initially analyzed in the SAR. This event was not identified as one of the transients that is significantly affected by fuel reload in Reference 49. Therefore, it has not been reanalyzed for any of the subsequent reload cycles.

Subsequent to the SAR, this transient has been generically reanalyzed without the conservative assumption 'a.' in Section 5.6.2.3. Even without this conservative assumption, multiple operator errors in recirculation loop valve line-up are required to start the idle loop as analyzed. The reanalysis of this transient retains the single more conservative assumption 'f.', thus maximizing the increase in core flow and subsequent neutron flux increase. The result of the reanalysis is that the transient remains non-limiting with respect to the more relaxed Power and Flow dependent thermal limits adopted by CNS.

5.6.2.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.6.2.3 Starting Conditions and Assumptions

The initial conditions which were assumed for the SAR analysis are:

a. One recirculation loop is shut down and filled with cold water (100°F). (Normal procedure requires warming this loop to within 50°F of the active loop prior to placing it in service).

b. The other recirculation pump is operating at a speed producing 90 percent of rated loop diffuser flow (45 percent of rated total diffuser flow) in the active jet pumps.

c. The core is receiving 41 percent of its rated flow, while the remainder of the active jet pump flow bypasses the core as reverse flow through the inactive jet pumps.

d. Reactor power is 60 percent of rated, a high initial value which misses the protection provided by the high neutron flux scram.

e. The idle loop recirculation pump suction valve is open, but the pump discharge valve is closed.

f. The scoop tube positioner for the idle recirculation pump MG set is at a setting which approximates 50 percent generator speed demand.

Other parameter values and available protective functions used in the analysis are presented in Table XIV-5-1. The following additional plant operating conditions and assumptions form the principal bases for analysis of the recirculation pump trip transients:

a. All of the plant control systems continue normal operation unless specifically designated to the contrary.

b. Auxiliary power is continuously supplied at rated frequency to power all auxiliary power equipment.

c. The reactor is operating in the manual flow control mode.

5.6.2.4 Sequence of Events and Systems Operation

Figure XIV-5-17 graphically shows the changes of important nuclear system variables during the simulated startup of the cold idle recirculation loop initiated from 60 percent rated power conditions. The loop startup transient causes the following sequence of events:

a. The drive motor breaker for the idle recirculation pump MG set is closed at t = 0.

b. The drive motor reaches near synchronous speed quickly, while the generator approaches full speed in about 5 seconds.

c. Next the generator field breaker closes automatically, loading the generator and applying starting torque to the pump motor. Generator speed will be drawn down as it tries to free the stopped rotor of the pump. Pump breakaway is modeled to occur in 8 seconds. Speed demand is sequentially programmed back to 20 percent of rated speed.

d. The pump discharge valve is started open as soon as its interlock with the drive motor breaker is cleared. (Normal procedure would delay valve opening to separate the two portions of the flow transient and make sure the drive loop is properly mixed with vessel-temperature water.) A nonlinear 30-second valve opening characteristic was used.

e. Shortly after the pump begins to move, a surge in flow from the activated diffusers gives the core inlet flow a sharp rise. (Diffuser flow achieves its normal flow direction at less than 10 seconds.)

5.6.2.5 Core and System Performance

This transient was analyzed using the REDY transient model [50, 51, and 52]. Calculations using the REDY model are based upon end-of-cycle conditions and utilize conservative multipliers on void, Doppler and scram reactivities (1.25, 0.95, and 0.8 respectively). As shown in Figure XIV-5-17, a short-duration neutron flux peak of about 118 percent is produced. The average surface heat flux, however, follows the slower response of the fuel and nucleate boiling is maintained throughout the transient (MCHFR remains above 2.2). No damage occurs to the fuel clad barrier.

5.7 Event Resulting in a Core Coolant Temperature Increase

The SAR categorized the following event as one which can cause directly a reactor vessel water temperature increase, one in which hotter water is returned to the reactor vessel without changing the coolant flow rate. This event is loss of shutdown cooling.

5.7.1 Loss of RHR Shutdown Cooling

The loss of RHR shutdown cooling can only occur during the low pressure portion of a normal reactor shutdown and cooldown. For most single failures which could result in loss of shutdown cooling, no unique safety actions are required. In these cases shutdown cooling is simply re-established using other normal shutdown cooling equipment.

In cases where the RHR shutdown cooling suction line becomes inoperative, a unique requirement for cooling arises. Under conditions when the reactor vessel head is off, either loop of the RHR system can be operated in the LPCI mode to maintain water level to assure continued core cooling. If the reactor vessel head is on, decay heat will generate steam which will pressurize the system. Any of the normal methods of conducting plant cooldown can be used under these conditions with makeup provided from normal or emergency makeup sources.

5.8 Event Resulting in Excess of Coolant Inventory

5.8.1 Feedwater Controller Failure - Maximum Demand

The feedwater controller failure - maximum demand transient event is determined in Reference 49 to be likely to limit operation based on MCPR considerations. As a result, this event is also analyzed as a licensing basis transient for this fuel reload cycle.

The core-wide Abnormal/Anticipated Operational Occurrence analyses for MELLL and ICF are performed for each reload cycle. These transient events include the feedwater controller failure (FWCF) maximum demand. This analysis is documented in Reference 35.

5.8.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

5.8.1.2 Frequency Classification

This event is classified as an incident of moderate frequency.

5.8.1.3 Starting Conditions and Assumptions

The parameter values used in the analysis are presented in Table XIV-5-2. The following additional plant operating conditions and assumptions form the principal bases for analysis of the feedwater controller failure - maximum demand transient:

a. Feedwater controller fails during maximum flow demand.

b. Maximum feedwater pump runout is assumed. (An absolute 5 percent of the maximum runout flows given in Table XIV-5-2 is added as a conservative measure to cover uncertainties in the value.)

c. The reactor is operating in a manual flow control mode which provides for the most severe transient.

d. Initial reactor water level is at the low water level alarm point.

5.8.1.4 Sequence of Events and Systems Operation

Figures XIV-5-18a-f provide is the transient simulation of a feedwater controller failure - maximum demand initiated at 100 percent NBR steam flow. A feedwater controller failure during maximum demand produces the following sequence of events:

a. The reactor vessel receives an excess of feedwater flow.

b. This excess flow results in an increase in core subcooling, which results in a core power rise, and an increase in the reactor vessel water level.

c. The rise in the reactor vessel water level eventually leads to high water level turbine trip, feedwater pump trip and reactor scram trip.

5.8.1.5 Core and System Performance

For reload cores, an evaluation is performed to determine if the feedwater controller failure - maximum demand transient could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the Supplemental Reload Licensing Report.

Plant/cycle-specific analyses for Cooper Nuclear Station using the GEMINI/ODYN transient analysis methodology as described in References 48 and 49 were reported in Reference 35 and are summarized in Table XIV-7-3.

The feedwater controller failure - maximum demand transient is terminated when the reactor water high level results in a main turbine trip which produces a turbine stop valve closure scram signal. Therefore, the transient must also be considered at power levels where the direct scram from turbine stop valve closure is bypassed.

Extensive transient analyses at a variety of power and flow conditions were performed during original development of the ARTS improvement program. A large data base was established by analyzing limiting transients over a range of power and flow conditions.

This data base shows that two power ranges must be examined. The first power range is between rated power and the power level where reactor scram on turbine stop valve closure is bypassed (P_{Bypass}, 30 percent of rated power). In this range, for the feedwater controller failure, the power decrease results in a greater mismatch between feedwater pump runout and the initial feedwater flow. This results in an increase in reactor subcooling and more severe changes in thermal limits during the event.

The second power range is between P_{Bypass} and 25 percent of rated power. No thermal limit monitoring is required below 25 percent power. Below P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on closure of the turbine stop valves. The extensive transient data base also shows a significant sensitivity to the initial core flow for transients initiated below P_{Bypass}. Sensitivity analyses on the limiting transients at below rated power conditions have shown that the maximum ΔCPR at any power level occurs at the maximum core flow condition.

The results of a feedwater controller failure - maximum demand transient analysis at initial conditions of 25 percent rated power and 100 percent rated core flow are presented in Reference 53. The analysis confirms that the MCPR change for this transient bounds the other transients analyzed in this power range (25-30 percent). Below P_{Bypass}, the actual bounding MCPR limit is chosen with sufficient conservatism such that it is independent of the cycle-specific operating limits.

5.9 SPECIAL EVENTS

5.9.1 Station Shutdown from Outside the Control Room

This special event is presented to demonstrate the capability to perform the operations required to maintain the station in a safe condition from outside the control room.

5.9.1.1 Criteria for Station Shutdown from Outside the Control Room

a. In the event that the control room becomes inaccessible, it shall be possible to bring the reactor from power range operation to a hot shutdown condition by manipulation of the local controls and equipment which are available outside the control room.

b. It shall be possible to bring the reactor to a cold shutdown condition by using controls and equipment outside the control room.

5.9.1.2 Assumptions

a. The station is operating initially at full power.

b. Station personnel evacuate the control room taking time only for those immediate actions within the control room that can be accomplished in seconds. (The procedures for this event provide guidance for the case where there is insufficient time to carry out actions affecting the plant prior to evacuation).

c. Station personnel take all subsequent action required to bring the reactor to a cold shutdown condition using controls and equipment located outside the control room.

5.9.1.3 Evaluation - Achievement of Cold Shutdown Condition

The Alternate Shutdown capability of the plant is described in USAR Section VII-18. Cold shutdown from outside the control room will be achieved by a series of actions as defined in predetermined procedures for such shutdown. Specifically for the case where immediate evacuation of the control room was required, this will include, but not be limited to, the following typical actions:

a. Insertion of the control rods by interruption of power to the reactor protection system motor-generator sets.

b. Operation of the HPCI system to maintain reactor level, temperature, and pressure control using controls and indications located in the Alternate Shutdown Room.

c. Operation of RHR for torus cooling using controls and indications located in the Alternate Shutdown Room and local control of the pump breaker.

d. Establishing long term cooling using LPCI injection, the ADS safety relief valves, RHR heat exchangers, and Service Water System by means of controls and indications provided in the Alternate Shutdown Room as well as other local actions required for system operations.

e. Monitoring reactor temperature, pressure, and level from indicators provided in the Alternate Shutdown Room and local indicators and devices inside the reactor building.

When the above steps are completed, the reactor has been brought to the cold shutdown condition and could remain in this condition for an unlimited period of time without requiring access to the control room. Analyses demonstrate that Primary Containment design temperature and torus design pressure are not exceeded[84], and that necessary torus area equipment remains functional[89].

It is concluded that the criteria of USAR Section XIV-5.9.1.1 are satisfied by station design.

5.9.2 Reactor Shutdown Without Control Rods

This special event is presented to demonstrate the capability of the standby liquid control (SLC) system to shutdown the reactor and maintain the shutdown condition as the reactor is cooled to cold shutdown conditions. The SLC system (discussed in USAR Section III-9) is manually initiated and controlled and is not intended to replace control rods for fast scram of the reactor.

Two cases are postulated to evaluate the capability of the standby liquid control system to shutdown the reactor:

a. The reactor is scrammed and it is postulated that some of the control rods malfunction and are not fully inserted.

b. The reactor is operating normally and it is postulated that all control rods malfunction and remain fixed at their present position.

The maximum rate of core reactivity increase for Case 1 conditions would result if the reactor was scrammed from full power, held at the hot standby condition for about one day until the rate of xenon decay was maximum, and then depressurized at the maximum allowable cooldown rate of 100°F per hour. Following scram, the available shutdown margin would actually increase as xenon poisoning in the core increases. The maximum xenon decay rate occurs after xenon poisoning has decreased again to values below those present at equilibrium xenon conditions at full power. The combined reactivity effects of maximum xenon decay rate and maximum reactor cooldown rate result in a maximum rate of change of core reactivity which is approximately one-fifth of the rate of change of core reactivity using the SLC system.

The maximum number of control rods are withdrawn when the reactor is at full power with equilibrium xenon poisoning and this condition establishes the maximum total reactivity control requirement for the SLC system based upon the reactivity control required to achieve the cold shutdown condition from the initial conditions assumed in Case 2 above. The SLC system, as designed, has sufficient capacity to control the reactivity difference between the steady state, full power operating condition of the reactor with voids and the cold shutdown conditions, including shutdown margin, to assure complete shutdown from the most reactive condition at any time in the core life.

The maximum rate of core reactivity increase for Case 2 conditions would result if the reactor was scrammed, held at the hot standby condition until the xenon concentration is maximum, and then returned to the full power condition. This sequence maximizes the rate of xenon depletion (burnup) after return to power and results in the maximum rate of increase of core reactivity from inherent nuclear processes. This sequence results in a maximum rate of core reactivity change which is approximately one-fifteenth of the rate of change of core reactivity using the SLC system.

The standby liquid control system provides a minimum boron injection rate which substantially exceeds the maximum rate of reactivity insertion based upon the worst possible conditions associated with Case 1 or Case 2 above. It is concluded that the design of the SLC system is adequate to satisfy the requirements of this special event.

5.9.3 Anticipated Transients Without Scram

This section covers the events which result in an anticipated transient without scram (ATWS). A more detailed discussion of this material is provided in Reference 55. The current Cooper Nuclear Station design utilizes diverse, highly redundant, and very reliable scram systems. These systems are frequently tested and would insert the control rods even if multiple component failures should occur, thus making the probability of an ATWS event extremely remote.

Anticipated transients without scram events are not design basis accidents. ATWS events are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) as required. The failure of the reactor to scram quickly during these transients could lead to unacceptable reactor coolant system pressures and to fuel damage. Mitigation of the lack of scram must involve insertion of negative reactivity into the reactor, thereby terminating the long-term aspects of the event.

A postulated failure to scram the reactor following an anticipated transient has been considered by the NRC. As a result of its assessment, the NRC issued a final rule, 10CFR50.62, in June 1984. The NRC ruling is of a prescriptive nature, specifying the requirements for an automatic recirculation pump trip (RPT), alternate rod insertion (ARI), and a standby liquid control (SLC) system.

5.9.3.1 ATWS Features

The purpose of the RPT function is to reduce core flow and create additional voids in the core, thereby decreasing power generation and limiting any power or pressure disturbance. The RPT system (refer to USAR Section VII-9.4.4.2) trips the field breakers on the recirculation motor-generator sets producing a rapid recirculation pump and core flow coastdown. The RPT system functions early in the transient on signals of either high RPV pressure or low RPV water level (Level 2). Instruments, power supplies, and cables used for RPT are separate from those used by the reactor protection system (RPS). This equipment is designed to perform its function in a reliable manner.

The ARI function (see USAR Section III-5.5.3.4) utilizes redundant scram air header exhaust valves to provide an alternate path for control rod insertion which is diverse and independent from the reactor protection system. Opening the ARI exhaust valves depressurizes the scram air header which results in control rod insertion. ARI is automatically initiated by the same reactor vessel high pressure and low water level signals that initiate RPT. The ARI function can also be initiated manually.

The SLC system (discussed in detail in USAR Section III-9) is a completely diverse method, utilizing injection of a soluble boron solution, for shutting down the reactor. The SLC system is initiated manually. In order to comply with the NRC ATWS Rule, the CNS SLC system must supply a minimum 11.5 percent concentration by weight sodium pentaborate solution at a flow rate of at least 76.4 gpm. Operation of both SLC system pumps is required to meet the minimum flow rate requirement.

It is extremely unlikely that the control rods will not be inserted by either the normal RPS trip system or ARI. However, for this event, the combination of RPT to initially reduce reactor power and SLC system operation to bring the reactor to hot shutdown ensure acceptably low suppression pool temperatures, adequate core coverage, and acceptable core temperatures for ATWS events.

5.9.3.2 <u>Acceptance Criteria for ATWS Analyses</u>

The NRC ATWS Rule does not specifically establish performance criteria. However, the analyses to demonstrate the acceptable response of the plant to ATWS conditions can be compared to the following historically applied criteria.

a. Fuel Integrity - The long-term cooling capability is assured by meeting the cladding temperature and oxidation criteria utilized for loss of coolant accidents as specified in 10CFR50.46 (i.e., peak cladding temperature not exceeding 2200°F, and the local oxidation of the cladding not exceeding 17 percent of the total cladding thickness).

b. Containment Integrity - The long-term containment integrity is maintained by demonstrating that the following criteria are met. The calculated containment pressure does not exceed the design pressure of 56 psig for the containment structure. The calculated maximum bulk suppression pool temperature is limited to 281°F, consistent with the containment design capabilities.

c. Primary System - The system transient pressure is limited such that the maximum primary stress within the reactor coolant pressure boundary does not exceed the Service Level C Limits as defined in the ASME Code, Section III. This was taken as a maximum reactor vessel pressure of 1500 psig. The functional capability of those components whose operation is required during or after the transient will not be impaired by the ATWS conditions.

d. Long-Term Shutdown Cooling - Subsequent to an ATWS event, the reactor can be brought to a safe shutdown condition without depending on control rod insertion and can be further cooled down and maintained in a cold shutdown condition.

e. Radiological Consequences - The calculated release of radioactivity does not exceed ten percent of the total radioactivity within the fuel rods.

5.9.3.3 <u>ATWS Analysis Methods and Assumptions</u>

5.9.3.3.1 <u>Updated Analysis</u>

The ATWS analysis was updated upon introduction of MELLL. The ATWS analysis was evaluated for the change to a 24 month operating cycle using GNF2 fuel. The two most limiting ATWS transients, MSIV closure and pressure regulator failure, have been analyzed for having up to 3 SRVs out of service. This three SRV out of service analysis bounds the NRC one SRV out of service allowance permitted in Technical Specification 3.4.3. Since only limiting portions of the ATWS analysis was updated, the previous ATWS analyses have been retained in the USAR. Response to selected ATWS events was evaluated with approved analytical models. The initial plant operating conditions used for the ATWS transient evaluations are listed in Table XIV-5-4.

All of the updated analysis assumes that ARI fails. The system response for each of the events was based on the same assumptions as the original analysis except for item a:

a. Thirty seconds after an MSIV closure, the feedwater turbines have exhausted their steam supply and subsequently the feedwater flow is reduced to zero.

5.9.3.3.2 Original Analysis

Response to the postulated ATWS events was evaluated with approved analytical models. The initial plant operating conditions used for the ATWS transient evaluations are listed in Table XIV-5-4. It should be noted that some setpoints and initial conditions differ from those currently in effect for CNS. The differences include a shorter rod insertion time than specified in the current design of the ARI system. The conclusions of these analyses, therefore, are not completely applicable to the current plant design and fuel cycle. Specific details contained in the descriptions and associated figures should be used only to understand the analysis and its conclusions. These specific details should not be used as sources of current design information.

The system response for each of the events with ARI was obtained utilizing the following assumptions:

a. Ninety seconds after an MSIV closure, the feedwater turbines have exhausted their steam supply and subsequently the feedwater flow is ramped to zero over a period of 5 seconds.

b. Realistic decay heat values based on the May-Witt correlation are used.

c. The pool cooling mode of the RHR system is effective by eleven minutes after the start of each event. The eleven minutes is comprised of a ten minute operator initiation time plus one additional minute for the operator to manually line-up the RHR pumps and heat exchangers. (Except for inadvertent opening of a relief valve event in which pool cooling is assumed to begin when the bulk pool temperature reaches 110°F). Sensitivity studies were performed to determine the impact of delayed suppression pool cooling initiation time. An evaluation was performed to establish the bounding value for the worst case ATWS Special Event—Pressure Regulator Failure Open. Establishment of suppression pool cooling as late as 43.5 minutes will not challenge containment design limits.[97]

5.9.3.4 ATWS - Main Steam Isolation Valve (MSIV) Closure

This transient produces high neutron flux, high heat flux from the fuel, and the potential for high vessel pressure and suppression pool temperature. The maximum values from this event are, in most cases, bounding of all the events considered. This transient was analyzed assuming 3 SRVs were out of service and with a maximum adder to the SRV setpoints and with inputs assumed in Table XIV-5-4.

5.9.3.4.1 Identification of Causes

Various steam line and nuclear system malfunctions, or operator action, can initiate closure of the main steam isolation valves. Examples of conditions which cause automatic closure are low reactor water level, low steam line pressure, high steam line area temperature, high steam line flow, and low main condenser vacuum.

Scram signal paths that must be assumed to fail include MSIV position switches, high APRM neutron flux, high RPV pressure, and all manual scram attempts.

5.9.3.4.2 Sequence of Events and Systems Operation

Closure of the MSIVs would produce an immediate increase in reactor pressure, which would result in a reduction in moderator voids and a rapid increase in reactor power. In the absence of normal scram, the fuel temperature would rise and the negative Doppler reactivity would limit the power. The opening of relief valves would tend to curtail the increase in reactor pressure and power. At about 4 seconds, the vessel dome pressure reaches the ATWS setpoint and both recirculation pumps trip. The ARI logic is initiated at the same pressure that trips the recirculation pumps. Should the ARI function fail, the SLC system would be initiated to shutdown the reactor.

TABLE XIV-5-4

CONDITIONS FOR ATWS EVENT ANALYSES

Parameter	Original Analysis Value	Updated Analysis Value	3 SRVOOS Updated Analysis Value
Thermal Power Level (MWt)	2381	2381	2419
Vessel Dome Pressure (psig)	1005	1005	1005
Core Flow (Mlb/hr)	73.5	55.13 MELLL 73.5 Rated	56.45 MELL 77.18 ICF (Note 3)
Steam Flow (Mlb/hr)	9.56	9.56	9.72
Feedwater Temperature (°F)	367	367	367.1
Initial Vessel Water Level (ft above separator skirt)	2.2	--	--
Initial Vessel Water Level (inches above vessel zero)	--	551.8	551.8
Vessel Inventory (lbs)	490,000	--	--
Void Reactivity Coefficient (¢/percent)	-11	-12	-12.8
Doppler Coefficient (¢/°F)	-0.28	-0.14	--
Sodium Pentaborate Solution Concentration in the Storage Tank at High Level (Percent by Weight)	12	11.5	11.5
Suppression Pool Liquid Volume (ft^3)	87,650	87,650	87,650
Suppression Pool Temperature (°F)	90	100[90]	100[90]
Condensate Storage Volume (Gal)	59,400	59,400	11754 (Note 4)
Condensate Storage Temperature (°F)	120	120	100
Core Average Void Fraction (percent)	39.3	46.7	44.1
Closure Time of MSIV (sec)	4	4	4.0
Relief Valve System Capacity (Percent NBR Steam Flow/Number of Valves)	70.0/8	73/8 (Note 1)	45/5 (Note 7)
Relief Valve Setpoint Range (psig)	1091/1111	1127/1178	1195/1205 PRFO 1246 MSIVC (Note 5)
Relief Valve and Sensor Time Delay (sec)	0.4	0.3	0.3
Relief Valve Opening Time (sec)	0.15	0.3	0.3
Safety Valve System Capacity (Percent NBR Steam Flow/Number of Valves)	19.8/3	20/3 (Note 2)	20/3
Safety Valve Setpoint Range (psig)	1253	1292	1292
Safety Valve Time Constant (sec)	0.2	0.2	0.2
Safety Valve Opening Time (sec)	0	0	0
Pressure Drop Below Setpoint for Relief Valve Closure (psi)	Low-Low-Set Relief	--	--
SRV Closing Setpoint as Fraction of Opening Setpoint	--	0.97	0.97
Relief Valve Closure Time Delay (sec)	0.3	0.3	0.3
Relief Valve Closure Time Constant (sec)	0.3	0.3	0.3
SLC Pump Start and Transport Time (sec)	30	30	30

TABLE XIV-5-4 (Cont'd)

CONDITIONS FOR ATWS EVENT ANALYSES

Parameter	Original Analysis Value	Updated Analysis Value	3 SRVOOS Updated Analysis Value
SLC Injection Rate - Number x Flow per Pump (gpm)	2x53	2x38.2	2x38.2
HPCI/RCIC Low Water Level Initiation Setpoint	Level 2	Level 2	Level 2 (Note 6)
HPCI/RCIC High Water Level Shutoff Setpoint	Level 8	Level 8	Level 8 (Note 6)
HPCI Start Time (sec)	20	20	20 (Note 6)
HPCI Flow Rate (gpm)	4,250	3,825	3825
RCIC Start Time (sec)	20	20	20 (Note 6)
RCIC Flow Rate (gpm)	400	360	360
ATWS High Pressure Setpoint (psig)	1071	1120	1120
ATWS Dome Pressure Sensor and Logic Time Delay (sec)	0.5	0.5	0.5
ATWS Low Water Level Setpoint	Level 2	Level 2	Level 2 (Note 6)
Recirculation Pump System Inertia Constant (sec)	5	5	5 (Note 6)
Delay Before Start of ARI Control Rod Insertion (sec)	15	--	--
Control Rod Insertion Time During ARI (sec)	10	--	--
RHR Pool Cooling Capacity (BTU/sec-°F)	486	354	354
Service Water Temperature (°F)	75	95[90]	95
Boron Required for Hot Shutdown (ppm)	355	522	522
Setpoint for Low Water Level Closure of MSIV	Level 1	Level 1	Level 1 (Note 6)
Setpoint for Low Steamline Pressure Closure of MSIV (psig)	825	825	825

Footnotes:

1. Total capacity for RV is evaluated at a reference pressure of 1090 psig.
2. Total capacity for SSV is evaluated at a reference pressure of 1240 psig.
3. A limited set of cases were performed at ICF flow to determine Peak Clad Temperature.
4. Emergency CST volume from OPL-3A.
5. The 3 lowest set SRVs are assumed to be OOS. For the PRFO transient, the analyzed setpoints were +70 PSI above normal setpoints * 1.03 and also included an applied statistical spread. For the MSIVC transient, the analyzed setpoints for all SRVs were at a maximum value of 1246.3.
6. Value not reported in analysis report, but is same as used in previous analyses.
7. The CNS analysis of record assumes up to 3 SRVs out of service. This is conservative to the approved Technical Specification 3.4.3 which allows only one SRV to be out of service.

5.9.3.4.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. In case of an apparent ATWS, certain manual actions would be required to be performed by the operator if automatic features do not function as designed. Possible operator actions would include manual initiation of a reactor scram, trip of the reactor recirculation pumps, manual initiation of ARI, and actuation of SLC.

Operator action would be required to initiate the suppression pool cooling mode of the RHR system. Control of RPV level could be optimized through manual control of the RCIC/HPCI systems. Following reactor shutdown, normal procedures would be utilized to bring the plant to cold shutdown.

5.9.3.4.4 Core and System Performance

5.9.3.4.4.1 Current Analysis Based on 3 SRV OOS

Figures XIV-5-21a and XIV-5-21b illustrate the transient behavior. It provides the time history of key parameters for the event. A sharp neutron flux peak reaches a maximum of 259 percent of the initial value at about 4 seconds into the event and rapidly decreases thereafter. The maximum average fuel surface heat flux of 142 percent NBR is reached about 1 second later. The peak cladding temperature of 1240°F resulting from the reduced core flow occurs at 55.5 seconds. This value is well below the historical 2200°F limit of 10CFR50.46. Generic analyses indicate that the local cladding oxidation is well below the 17 percent limit.

The operator is assumed to manually initiate the SLC system at two minutes after the ATWS high pressure setpoint is reached. The water level is reduced after the feedwater pump trip due to the exhaustion of steam supply to the turbines. The operators reduce the water level by controlling HPCI and RCIC flow. After the hot shutdown boron weight has been injected into the vessel, the reactor water level is raised to the normal range. Hot shutdown occurs at about 24 minutes into the event.

5.9.3.4.4.2 Original Analysis

Reactor Shutdown by RPT and ARI

Figure XIV-5-19 illustrates the transient behavior. A sharp neutron flux peak reaches a maximum of 805 percent of the initial value at about 4 seconds into the event and rapidly decreases thereafter. The maximum average fuel surface heat flux of 144 percent NBR is reached about 1 second later. The peak cladding temperature of 1359°F resulting from the neutron flux spike occurs at 7.5 seconds. This value is well below the historical 2200°F limit of 10CFR50.46. Generic analyses indicate that the local cladding oxidation is well below the 17 percent limit, and that the estimated release of radioactivity is below 1 percent of the total inventory in the rods. Hot shutdown occurs due to ARI at about 30 seconds into the event.

The reactor water level is initially maintained by feedwater flow, but MSIV closure will ultimately result in the termination of feedwater flow. When Level 2 is reached at about 45 seconds, HPCI and RCIC automatically start and replenish the vessel inventory from the condensate storage tank, up to the high level (Level 8) trip setpoint.

Reactor Shutdown by RPT and SLC (No ARI)

In order to demonstrate the SLC system effectiveness the ATWS high pressure ARI signal is also assumed to fail. Figure XIV-5-20 provides the time history of key parameters for the first 100 seconds of the event; Figure XIV-5-21 plots the same parameters over a longer time period.

The peak values of neutron flux, fuel heat flux, and peak cladding temperature are identical to those obtained for the MSIV closure with ARI. This is because all of these parameters peak very early in the event (prior to when either ARI or SLC injection becomes effective).

The operator is assumed to manually initiate the SLC system at two minutes after the ATWS high pressure setpoint is reached. At about 4.5 minutes the boron solution reaches the core and begins shutting down the reactor. Hot shutdown is achieved by 14 minutes into the event.

The reactor water level is initially maintained by feedwater flow. However, the flow is terminated at 95 seconds due to MSIV closure. At around 130 seconds the HPCI and RCIC flows begin entering the vessel to make up the inventory lost through the SRV actuations.

5.9.3.4.5 Barrier Performance

5.9.3.4.5.1 Current Analysis Based on 3 SRV OOS

Figures XIV-5-21a and XIV-5-21b illustrate the transient behavior. Near 4 seconds, the pressure setpoint of the relief valves and safety valves is reached and the valves begin to open to relieve pressure. The vessel pressure continues to rise for a short time following RPT until, at approximately 15 seconds into the event, it reaches its peak and begins to decrease due to the decrease in core power and steam generation that results from the RPT, and due to relief/safety valve actuation. The maximum pressure at the vessel bottom is 1472 psig, which is well below the ASME service Level C limit of 1500 psig for the vessel. The relief valves begin to close at about 33 seconds; pressure is then controlled by relief valve cycling.

With SLC manually initiated at two minutes after the ATWS high pressure setpoint is reached, hot shutdown is achieved by 24 minutes into the event. Decay heat continues to generate a small amount of steam which flows through the relief valves following hot shutdown. The peak suppression pool temperature of 196°F is reached at 73 minutes. The corresponding maximum containment pressure is 13 psig. These values are well below the containment design limits.

5.9.3.4.5.2 Original Analysis

Reactor Shutdown by RPT and ARI

Figure XIV-5-19 illustrates the transient behavior. Near 4 seconds, the pressure setpoint of the relief valves and safety valves is reached and the valves begin to open to relieve pressure. The vessel pressure continues to rise for a short time following RPT until, at approximately 9 seconds into the event, it reaches its peak and begins to decrease due to the decrease in core power and steam generation that results from the RPT, and due to relief/safety valve actuation. The maximum pressure at the vessel bottom is 1325 psig, which is well below the Service Level C limit of 1500 psig for the vessel. Following hot shutdown, the decay heat continues to generate a small amount of steam which flows through the relief valves. The relief valves begin to close at about 33 seconds; pressure is then controlled by relief valve cycling.

At 11 minutes, the pool cooling mode of the RHR system becomes effective. The suppression pool bulk temperature reaches its maximum value of

129°F in 183 minutes. The corresponding maximum containment pressure is 2.5 psig. These values are well below the containment design limits. (For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

Reactor Shutdown by RPT and SLC (No ARI)

For the case where ARI is assumed to fail, Figure XIV-5-20 provides the time history of key parameters for the first 100 seconds of the event; Figure XIV-5-21 provides the same over a longer time period.

The peak value of vessel pressure is identical to that obtained for the MSIV closure with ARI. Vessel pressure peaks very early in the event (prior to when either ARI or SLC injection becomes effective).

With SLC manually initiated at two minutes after the ATWS high pressure setpoint is reached, hot shutdown is achieved by 14 minutes into the event. Decay heat continues to generate a small amount of steam which flows through the relief valves following hot shutdown. The peak suppression pool temperature of 187°F is reached at 22 minutes. The corresponding maximum containment pressure is 11 psig. These values are well below the containment design limits.

Sensitivity studies were performed to determine the impact of delayed SLC initiation on both hot shutdown time and suppression pool response. The cases analyzed assumed five and ten minute SLC initiation times (from reaching the ATWS high pressure setpoint) rather than the two minute time in the base case. The five minute delay in SLC initiation resulted in an increase in hot shutdown time of roughly 4 minutes and an increase of 19°F in the suppression pool temperature as compared to the base case. The ten minute delay increased the hot shutdown time by 9 minutes and increased the pool temperature by 45°F as compared to the base case event.

5.9.3.5 ATWS - Turbine Trip with Bypass

This event results in values for fuel cladding temperature, primary system pressure, suppression pool temperature, and primary system pressure which are below those in the MSIV closure ATWS event. The availability of the condenser heat sink limits the suppression pool heatup. This event is bounded by the MSIV Closure event for all the ATWS acceptance criteria due to the availability of the bypass valves and main steam condenser. Therefore, an updated analysis is not performed for this event.

5.9.3.5.1 Identification of Causes

A variety of turbine, electrical, or nuclear system malfunctions will initiate a turbine trip. Some examples are: loss of control fluid pressure (bearing oil or auto-stop oil), low condenser vacuum, electrical distribution system faults, and reactor vessel high water level. When the turbine trips, the turbine stop valves close causing a sudden reduction in steam flow which results in a nuclear system pressure increase.

Scram signals are initiated from turbine stop valve position, high APRM neutron flux, high reactor vessel pressure, or manual action. All must be assumed to fail in the subsequent discussion.

5.9.3.5.2 Sequence of Events and Systems Operation

As with the MSIV closure event, the plant behavior is separable into an early transient period with rapidly changing power and pressure, and a slightly longer term period of plant shutdown. The key difference is that the main condenser remains available for this event. From the time that the steam flow is within the bypass capacity, the main condenser is capable of removing all the steam from the vessel.

The turbine trip event begins with the rapid closure of the turbine stop valves and the opening of the turbine bypass valves. After the stop valves close in 0.1 seconds, the pressure immediately begins to rise, resulting in a reduction in void fraction and rapid increase in power. The opening of relief valves would tend to curtail the increase in reactor pressure and power. Shortly after 1 second, the vessel dome pressure reaches the ATWS setpoint and both recirculation pumps trip. The ARI logic is initiated at the same pressure that trips the recirculation pumps. Should the ARI function fail, the SLC system would be initiated to shutdown the reactor.

5.9.3.5.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. The general actions required for this event would be similar to those described for the MSIV closure ATWS event.

5.9.3.5.4 Core and System Performance

Figure XIV-5-22 illustrates the transient behavior. The neutron flux reaches a maximum of 686 percent of the initial value at 1 second into the event and rapidly decreases thereafter. The resulting peak average fuel surface heat flux occurs at about 3 seconds at a value of 144 percent NBR. The fuel cladding temperature peaks at approximately 4 seconds with a value of 1143°F.

Hot shutdown occurs due to ARI at about 30 seconds into the event. In the event that insertion of the control rods via ARI is not achievable, the SLC system would be utilized as an alternative method of achieving reactor shutdown. The peak cladding temperature would not be affected because it occurs early in the transient before either ARI or SLC systems can have an effect.

Unlike the MSIV closure event, the feedwater system remains available in this event. Level 2 is reached at about 40 seconds, initiating HPCI and RCIC.

5.9.3.5.5 Barrier Performance

Figure XIV-5-22 illustrates the transient behavior. At approximately 2 seconds, the pressure setpoint of the relief valves is reached and the valves open to arrest the pressure rise. Pressure continues to rise while the relief valves are opening and peaks at approximately 5 seconds. The maximum pressure at the vessel bottom is 1190 psig.

The availability of the bypass valves and the main condenser limits the suppression pool heatup. At 11 minutes, the pool cooling mode of the RHR system becomes effective. The suppression pool bulk temperature reaches its maximum value of 126°F at 273 minutes into the event with a corresponding containment pressure of 2.3 psig. Suppression pool temperature and containment pressure would be higher if reactor shutdown was achieved using SLC rather than ARI. The peak values would be less than those reached in the MSIV closure event (SLC without ARI). (For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

5.9.3.6 ATWS - Inadvertent Opening of a Relief Valve

This event has no rapid excursion in reactor pressure and power, but is merely a long-term suppression pool heatup and eventual vessel depressurization. Depending on the safety relief valve capacity and the operator action taken following the failure of manual scram, this transient may result in the highest suppression pool temperature of any ATWS event.

5.9.3.6.1 Identification of Causes

In this postulated event, a relief valve is assumed to fail open during full power operation. All manual attempts to close the relief valve and scram the reactor are assumed to fail.

5.9.3.6.2 Sequence of Events and Systems Operation

5.9.3.6.2.1 Updated Analysis

The event is depicted in Figures XIV-5-23a and XIV-5-23b. This event begins when one of the primary relief valves on the main steam lines inadvertently opens, releasing steam into the suppression pool which is assumed to be at 95°F. When the valve opens, there is a small pressure perturbation until the turbine control valves close slightly to control pressure. The operator receives an alarm that a relief valve is open and observes the suppression pool heating up. If attempts to close the valve are unsuccessful, the pool temperature reaches 110°F in 5.8 minutes. At this point the operator attempts to manually scram the reactor, and when the scram fails, activates ARI. Since insertion of the control rods via ARI is assumed to fail, the SLC system would be utilized as an alternative method of effecting reactor shutdown. The operators reduce the water level by controlling HPCI and RCIC flow. After the hot shutdown boron weight has been injected into the vessel, the reactor water level is raised to the normal range. Hot shutdown is achieved at about 25 minutes into the event.

5.9.3.6.2.2 Original Analysis

The event is depicted in Figure XIV-5-23. (Please note that the time histories shown in the figure correspond to the transient being initiated with a suppression pool temperature of 95°F. Since the initial suppression pool temperature was assumed to be 90°F, 117 seconds should be added to all time values on the plot. This represents the time required to heat the pool from 90 to 95°F).

This event begins when one of the primary relief valves on the main steam lines inadvertently opens, releasing steam into the suppression pool which is assumed to be at 90°F. When the valve opens, there is a small pressure perturbation until the turbine control valves close slightly to control pressure. The operator receives an alarm that a relief valves is open and observes the suppression pool heating up. If attempts to close the valve are unsuccessful, the pool temperature reaches 110°F in 7.2 minutes. At this point the operator attempts to manually scram the reactor, and when the scram fails, activates ARI. In the event that insertion of the control rods via ARI is not achievable, the SLC system would be utilized as an alternative method of effecting reactor shutdown.

5.9.3.6.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. The general actions required for this event would attempt to close the open relief valve, attempt a manual scram when the relief valve could not be closed, manually initiate ARI, and initiate suppression pool cooling mode of RHR.

In this event the operator must manually initiate ARI, since the event does not trip either the high pressure or low water level initiation signals. Should insertion of rods with ARI fail, the operator would need to manually initiate SLC.

5.9.3.6.4 Core and System Performance

5.9.3.6.4.1 Updated Analysis

The suppression pool is the only system exposed to abnormal conditions. Once ARI fails to activate, the operator would actuate SLC system, reduce water level, bypass low water level MSIV closure, and restore the water level after the hot shutdown boron weight is injected into the vessel. Hot shutdown is achieved by about 25 minutes into the event. Decay heat continues to generate steam, and the reactor continues to depressurize.

5.9.3.6.4.2 Original Analysis

The suppression pool is the only system exposed to abnormal conditions. Once ARI is activated, hot shutdown is achieved by 7.6 minutes into the event. Decay heat continues to generate steam, and the reactor continues to depressurize. If the operator has not switched the reactor mode switch out of the RUN mode, main steam line isolation occurs on low steam line pressure at 9.1 minutes. The feedwater pumps trip 1.5 minutes later.

Loss of inventory and vessel depressurization cause a decrease in water level until the low level setpoint is reached causing a recirculation pump trip and the initiation of the HPCI and RCIC systems. The HPCI and RCIC systems continue to cycle off at Level 8 and on at Level 2 to maintain the vessel inventory which is lost through the stuck open relief valve.

5.9.3.6.5 Barrier Performance

5.9.3.6.5.1 Updated Analysis

All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. Once the reactor is in hot shutdown, the event depressurizes the reactor. There is no challenge to the reactor coolant pressure boundary during this event.

Since the ARI fails to actuate, the SLC system is activated. A peak suppression pool temperature of 164°F occurs at 155 minutes, and the associated peak containment pressure is 6.2 psig. This is below the containment design limits.

5.9.3.6.5.2 Original Analysis

All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. Once the reactor is in hot shutdown, the event depressurizes the reactor. There is no challenge to the reactor coolant pressure boundary during this event.

Suppression pool cooling is assumed to begin when suppression pool temperature reaches 110°F. A peak suppression pool temperature of 156°F occurs at 79 minutes, and the associated peak containment pressure is 5.3 psig. If the use of SLC was required to shutdown the reactor these values would be higher, but still below the containment design limits.

5.9.3.7 ATWS - Pressure Regulator Failure Open

This transient, like the MSIV closure, produces high neutron flux, heat flux, vessel pressure, and suppression pool temperature. Depending on the plant configuration, peak values from this event have exceeded those for the MSIV closure. For this reason, the analysis of this event is included. This transient was analyzed assuming 3 SRVs were out of service and with a +70 psi adder to the nominal SRV setpoints and with inputs assumed in Table XIV-5-4.

5.9.3.7.1 Identification of Causes

This event begins with the postulated failure of the Digital Electro-Hydraulic (DEH) pressure control system in a manner such that maximum steam flow is demanded. Scram signals that are assumed to fail in this event are the MSIV position switches, APRM high neutron flux, and high pressure.

5.9.3.7.2 Sequence of Events and Systems Operation

5.9.3.7.2.1 Current Analyses Based on 3 SRV OOS

Figures XIV-5-24a and XIV-5-24b show the system response for this event with SLC system. The failure of the pressure controller to its maximum steam flow position causes a quick increase in vessel steam flow, which results in a rapid decrease in vessel pressure. At about 22.7 seconds the low pressure MSIV isolation setpoint is reached and the MSIVs begin to close.

Once the MSIVs close, the transient is essentially like the MSIV closure event. The isolation is followed by a rapid rise in power and pressure. High RPV pressure initiates RPT and ARI, though ARI is assumed to fail for the updated analysis. Relief and safety valves open in response to high vessel pressure.{AN1067}

5.9.3.7.2.2 Original Analysis{S}

Figure XIV-5-24 shows the system response for this event. The failure of the pressure controller to its maximum steam flow position causes a quick increase in vessel steam flow, which results in a rapid decrease in vessel pressure. At about 10 seconds the low pressure MSIV isolation setpoint is reached and the MSIVs begin to close.

Once the MSIVs close, the transient is essentially like the MSIV closure event. The isolation is followed by a rapid rise in power and pressure. High RPV pressure initiates RPT and ARI. Relief and safety valves open in response to high vessel pressure.

5.9.3.7.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. The general actions required for this event would be similar to those described for the MSIV closure ATWS event.

5.9.3.7.4 Core and System Performance

5.9.3.7.4.1 Current Analysis Based on 3 SRV OOS

The neutron flux reaches a peak of 263 percent NBR near 27 seconds, and the peak fuel average heat flux of 149 percent NBR occurs at about 29 seconds. At approximately 40 seconds into the event, the maximum pressure at the vessel bottom is reached at 1489 psig. The peak cladding temperature of 1442°F occurs at 51 seconds. The SLC system serves as a backup to shutdown the reactor since ARI fails.

The water level is reduced after the feedwater pump trip due to the exhaustion of steam supply to the turbines. The operators maintain the water level at a reduced level with HPCI and RCIC. The SLC system is manually initiated at two minutes after the ATWS high pressure setpoint is reached. After the hot shutdown boron weight has been injected into the vessel, the reactor water level is raised to the normal range. The hot shutdown is achieved by 1886 seconds into the event. Decay heat continues to generate a small amount of steam which flows through the relief valves following hot shutdown.

5.9.3.7.4.2 Original Analysis

The neutron flux reaches a peak of 540 percent NBR near 17 seconds, and the peak fuel average heat flux of 122 percent NBR occurs at about 19 seconds. The peak cladding temperature of 1323°F occurs at 37 seconds and hot shutdown by ARI is achieved at about 43 seconds into the event. The SLC system serves as a backup to shutdown the reactor should ARI fail.

5.9.3.7.5 Barrier Performance

5.9.3.7.5.1 Current Analysis Based on 3 SRV OOS{S}

For BOC conditions, the peak vessel pressure occurs at 40 seconds into the event and is 1489 psig. This is within the ASME Service Level C limit of 1500 psig mainly due to the automatic action of RPT, which occurs at about 29 seconds and reduces core flow and power, and the relieving action of the SRVs.

For BOC conditions, the peak cladding temperature of 1442°F resulting from the reduced core flow occurs at 51 seconds. This value is well below the historical 2200°F limit of 10CFR50.46. Generic analyses indicate that the local cladding oxidation is well below the 17 percent limit.

For BOC conditions, with suppression pool cooling placed in operation at 11 minutes into the event, a peak suppression pool temperature of 183°F is reached at about 5013 seconds with a corresponding pressure of 10.3 psig. These values are well below the containment design limits. They would be somewhat lower if the ARI succeeded to achieve hot shutdown.(For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

5.9.3.7.5.2 Original Analysis

The peak vessel pressure occurs at 23 seconds into the event and is 1293 psig. This is within the Service Level C limit of 1500 psig mainly due to the automatic action of RPT, which occurs at about 18 seconds and reduces core flow and power, and the relieving action of the SRVs.

With suppression pool cooling placed in operation at 11 minutes into the event, a peak suppression pool temperature of 128°F is reached at 203 minutes with a corresponding pressure of 2.4 psig. These values would be somewhat higher if the ARI failed and SLC was needed to achieve hot shutdown.(For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

5.9.3.8 ATWS - Loss of Normal Feedwater

This event has generally been shown to result in the lowest vessel water level of any event. It is also the only ATWS event in which the primary initiation signal is low water level. This event is bounded by the MSIV Closure and the pressure regulator failure open events for all the ATWS acceptance criteria. Therefore, an updated analysis is not performed for this event.

5.9.3.8.1 Identification of Causes

A loss of feedwater flow may occur as a result of feedwater pump failures, condensate pump failures, feedwater controller failures, operator errors, or trip on reactor high water. Loss of auxiliary power (see USAR Section XIV-5.11.9) also produces a loss of feedwater along with the loss of many other plant functions.

5.9.3.8.2 Sequence of Events and Systems Operation

Figure XIV-5-25 shows the reactor response to this event. All feedwater flow is assumed to be lost in about 5 seconds. After the loss of feedwater has taken place; the pressure, water level and neutron flux begin to fall. Recirculation runback occurs at 5 seconds on coincident signals of loss of feedwater pumps and low water level alarm (Level 4). At around 23 seconds, low water level (Level 2) is reached. This trips the recirculation pumps, initiates ARI, and initiates the HPCI and RCIC systems. By about 50 seconds, the ARI function has achieved a hot shutdown condition in the reactor. The SLC system is available in the unlikely event that ARI fails.

At about 45 seconds, the HPCI and RCIC flows enter the vessel. They replace the main feedwater system flow and begin to overcome the inventory loss; the minimum level is reached near 67 seconds. Since by this time the neutron flux has already been decreased well below 1 percent NBR by ARI action, there is no significant steam generation. The vessel pressure drops slowly as quenching by the RCIC and HPCI continues.

5.9.3.8.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. In case of an apparent ATWS, certain manual actions would be required to be performed by the operator if automatic features do not function as designed. Possible operator actions would include manual initiation of a reactor scram, trip of the reactor recirculation pumps, manual initiation of ARI, and actuation of SLC.

Operator action would be required to initiate the suppression pool cooling mode of the RHR system. Control of RPV level could be optimized through manual control of the RCIC/HPCI systems. Following reactor shutdown, normal procedures would be utilized to bring the plant to cold shutdown.

5.9.3.8.4 Core and System Performance

Neutron flux and average surface heat flux decrease throughout the transient. Operation of the HPCI and RCIC systems will restore level to its normal range, where either automatic cycling occurs between Level 2 and 8 setpoints, or the operator manually controls level using RCIC (the preferred method).

5.9.3.8.5 Barrier Performance

This event has no rapid excursions as in some of the other events, but results in a long-term power reduction and depressurization. Since the pressure begins to fall at the outset of the transient, the need for relief valves does not arise. The containment limits are not approached.

5.9.3.9 ATWS - Loss of Normal AC Power

This transient, when coupled with the normal scram system failure, may produce high vessel pressure and pool temperature. In addition, it results in low vessel water level following feedwater flow cessation due to the loss of power to the condensate pumps.

5.9.3.9.1 Identification of Causes

The loss of normal AC power would generally be caused by large grid disturbances which in turn would de-energize buses that supply power to auxiliary equipment such as the recirculation pumps, condensate pumps, and circulating water pumps.

The MSIV closure which results from the loss of normal AC power would normally initiate a reactor scram if it had not yet occurred from the loss of power to the reactor protection system. If these signals fail to cause scram, additional scram signals occur from high flux, high pressure, and low water level.

5.9.3.9.2 Sequence of Events and Systems Operation

5.9.3.9.2.1 Updated Analysis

Figures XIV-5-26a and XIV-5-26b show the reactor response to this event. The event begins with the loss of recirculation pumps and feedwater (feedwater flow is lost due to loss of power to the condensate pumps). This leads to an initial reduction in power and pressure. At 2 seconds, the start of MSIV closure is assumed to take place, which results in a rapid rise in power and pressure.

The power and pressure increases are limited on this event by the action of the SRVs and the trip of the recirculation pumps which occurred at the beginning of this event due to the loss of power. The high pressure logic should activate ARI. However, the failure of ARI to function prompts the operators to manually initiate the SLC system (assumed to be 2 minutes after the dome pressure reaches the high pressure setpoint). The operators maintain the water level at a reduced level with HPCI and RCIC. After the hot shutdown boron weight has been injected into the vessel, the reactor water level is raised to the normal range. Hot shutdown is achieved at 21 minutes.

5.9.3.9.2.2 Original Analysis

Figure XIV-5-26 shows the reactor response to this event. The event begins with the loss of recirculation pumps and feedwater (feedwater flow is lost due to loss of power to the condensate pumps). This leads to an initial reduction in power and pressure. At 2 seconds, the start of MSIV closure is assumed to take place, which results in a rapid rise in power and pressure.

The power and pressure increases are limited on this event by the action of the SRVs and the trip of the recirculation pumps which occurred at the beginning of this event due to the loss of power. The high pressure logic activates ARI and all rods are inserted shortly after 30 seconds. The relief valves close around 90 seconds.

5.9.3.9.3 Identification of Operator Actions

The detailed and specific guidance for operator actions is provided by the CNS Emergency Procedures. The general actions required for this event would be similar to those described for the MSIV closure ATWS event.

5.9.3.9.4 Core and System Performance

5.9.3.9.4.1 Updated Analysis

The neutron flux peaks at 150 percent NBR near 6 seconds. The heat flux does not exceed 100 percent NBR due to the recirculation pump trip at the beginning. A peak cladding temperature of 1050°F is reached at about 50 seconds. This value is well below the historical 2200°F limit of 10CFR50.46. Generic analyses indicate that the local cladding oxidation is well below the 17 percent limit. With normal scram assumed to have failed, the long-term power shutdown is achieved with SLC when ARI fails.

The operator would control the reactor water level at reduced level with RCIC and HPCI flow until hot shutdown boron weight is injected into the vessel. The water level would then be restored to its normal range. The hot shutdown is achieved at 61 minutes.

5.9.3.9.4.2 Original Analysis

The neutron flux peaks at 412 percent NBR near 7 seconds. A peak cladding temperature of 1051°F is reached at about 21 seconds. With normal scram assumed to have failed, the long-term power shutdown is achieved by ARI which is initiated by the ATWS high reactor pressure signal. SLC remains as a backup to ARI.

Reactor water level is restored quickly to its normal range by RCIC and HPCI flow.

5.9.3.9.5 Barrier Performance

5.9.3.9.5.1 Updated Analysis{S}

Peak vessel pressure occurs in 8.6 seconds at vessel bottom and is 1204 psig. The transient pressure is within the ASME Service Level C overpressure limit of 1500 psig. This is due to the trip of the recirculation pumps which occurs at the start of the transient and the relieving action of the SRVs which all open.

The RHR is activated in the pool cooling mode at eleven minutes into the event. The suppression pool temperature peaks in 86 minutes at 176°F. The corresponding peak containment pressure is 8.0 psig. If the ARI succeeds to achieve reactor shutdown, these peak values would be lower.(For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

5.9.3.9.5.2 Original Analysis

Peak vessel pressure occurs in 10 seconds at vessel bottom and is 1201 psig. The transient pressure is within the Service Level C overpressure limit of 1500 psig. This is due to the trip of the recirculation pumps which occurs at the start of the transient and the relieving action of the SRVs which all open, then start reclosing near 28 seconds.

The RHR is activated in the pool cooling mode at eleven minutes into the event. The suppression pool temperature peaks in 208 minutes at 128°F. The corresponding peak containment pressure is 2.4 psig. If the initiation of SLC was required to achieve reactor shutdown, these peak values would be higher, but less than the similar case for MSIV closure.(For sensitivity of delayed suppression pool cooling initiation time refer to Section 5.9.3.3.2c.)

5.9.3.10 Impact of ECCS Relaxations on ATWS

An evaluation was performed to determine the impact of the ECCS parameter relaxations on the ATWS analysis. It was previously determined that the Main Steam Isolation Valve (MSIV) Closure event resulted in the most severe heat flux, maximum vessel pressure, and maximum suppression pool temperature."[55] For the ECCS parameter relaxations study, both the MSIV Closure and Loss of Feedwater Flow ATWS events were evaluated.

To determine the impact of the ECCS parameter relaxations, a review of the assumptions and inputs used in the MSIV Closure analysis was conducted. This event assumes 2 RHR pumps in pool cooling mode. Since the 2 pump flow in RHR pool cooling mode is less then the relaxed RHR flow rate, the peak suppression pool temperature results of this analysis are not impacted by the relaxation in LPCI flow rate. Therefore, only the effects of HPCI Flow Rate and the Level 1 and Level 2 low water setpoint relaxations need to be reconsidered. The calculated peak vessel bottom pressure and peak cladding temperature remain the same because they occur early in the event before there is any impact of the relaxations. Closing the MSIV results in a feedwater trip, so HPCI flow becomes the main source of water to the core. The water added by HPCI is turned to steam that is sent to the suppression pool, causing the suppression pool temperature to increase. Therefore, delaying HPCI start and decreasing the flow actually causes a decrease in the peak suppression pool temperature. It was determined that relaxing the ECCS parameters would not adversely impact the MSIV closure ATWS event.

The ATWS/LOFW event causes a rapid drop in water level. A comparative analysis was performed to verify that the relaxed HPCI flow rate and start time will perform its intended function of maintaining the reactor water level above the Top of Active Fuel. An ATWS LOFW analysis was performed for a similar BWR/4 plant with a higher power level. Normalizing the HPCI flow to rated feedwater flow provides a basis for comparing plant power levels and HPCI responses. The normalized HPCI flow for the analyzed BWR/4 plant is slightly less (<1% of rated feedwater flow loss) than the CNS relaxed value, so it can be used conservatively as a benchmark for the comparison. In the BWR/4 plant analysis, the LOFW event was significantly less severe than the MSIV Closure, the limiting event, and the minimum water level was above the Top of Active Fuel. The ECCS parameter relaxations have an insignificant impact due to the large margin between the ATWS analysis results for LOFW and the limiting criteria.

5.9.3.11 Impact of SRVs Out-of-Service

The licensing basis for Anticipated Transients Without Scram (ATWS) overpressure protection[94] requires 7 of 8 SRVs to be operable. Approval was based on an analysis[95] that showed up to 3 SRVs could be out of service (OOS) in an ATWS and still assure a sufficient margin to the ASME Service Level C limit (1500 psig). Nevertheless, allowing only one SRV OOS is conservative since ATWS margin is expanded even more.

This SRV OOS analysis used comparable core initial conditions and equipment performance characteristics as the analysis referred to in Table XIV-5-4 as the Updated Analysis Value. The evaluation process is essentially identical to that previously used. The minor differences are in the boundary conditions of power, SRV setpoints and availability, initial water temperatures and the initial nuclear condition.[95] The major difference from the updated analysis is that this study considered SRVs OOS.

The evaluation only considered the Pressure Regulator Fails Open (PRFO) and the MSIV Closure events. The previous analysis of record showed that the other ATWS events (loss-of-auxiliary power (LOAP) and inadvertent opening of relief valve (IORV)) are less limiting than the PRFO and MSIV Closure.

Starting from an initial reactor power of 2419 MWt (1.016*Licensed Thermal Power), analysis success goals for ATWS vessel pressure were set less than the 1500 psig ASME limit for the vessel. This success goal used a reasonable margin to account for future changes (fuel and hardware). Fuel cladding temperature limit is 2200°F and Local Cladding Oxidation limit is 17%, consistent with 10 CFR 50.46. The Containment Pressure limit of 56 psig and Suppression Pool Temperature limit of 208°F were consistent with the previous analyses.

5.9.4 Station Blackout (SBO)

This special event is presented to demonstrate the capability of CNS to meet the requirements of 10CFR50.63 for a 4-hour coping duration. The SBO scenario was developed based on reasonably expected operator actions and automatic system responses, for the purpose of describing the plant behavior during such an event.

The 4-hour SBO coping duration was established using the methodology of NUMARC 87-00 based on CNS site-specific characteristics. The following results were determined:

a. Offsite Power Design Characteristic Group P1-I1/2
b. Emergency AC Power Supply System Configuration C
c. Emergency Diesel Generator Target Reliability 0.95

5.9.4.1 SBO Initial Plant Conditions

The station blackout special event occurs while the reactor is operating at 100% rated thermal power and has been at this power level for at least 100 days. Immediately prior to the SBO, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level. Plant equipment is either normally operating or available from the standby state.

5.9.4.2 SBO Sequence of Events

The initiating event is assumed to be a sudden loss of off-site power with a resulting turbine trip. Fire or natural phenomena (such as flooding and earthquakes) are not considered as initiators. No other independent design basis accidents or other events are assumed to occur either prior to or during the SBO, unless they are resulting from the loss of AC power. Similarly, no independent failures (other than those that initiate the SBO) are assumed to occur.

The following occurs at time t = 0:

a. Reactor scram and Group I, II, and III isolations occur.
b. Auto-start signals to both emergency diesel generators.

The following occurs at time t = 0 to t = 10 minutes:

a. All control rods are verified full in; power reduces to decay heat levels.
b. The emergency diesel generators fail to start.
c. The recirculation pumps trip on loss of power.
d. Safety relief valves control reactor pressure in low-low set mode.
e. Reactor water level decreases due to loss of condensate and feedwater systems. At Level 2, both RCIC and HPCI auto start on low water level. Water level is restored, and HPCI is manually secured after cycling off automatically after

reaching Level 8 (i.e., within approximately 10 minutes of HPCI operation).

The following occurs at time t = 10 minutes to t = 2 hours:

a. Reactor Operators enter the CNS Emergency Procedures which result in load stripping and compensatory measures to enhance cooling to heated rooms which contain equipment credited with coping with the SBO.
b. The RCIC system is used to control reactor water level for the remainder of the SBO event until on-site or off-site electrical power can be restored. After recovery of the vessel water level during the first cycle of HPCI operation, only the RCIC system is required to cope with an SBO, and it alone is adequate for level control. Credit is taken for manual RCIC control in maintaining a steady reactor water level.

The following occurs at time t = 2 hours to t = 4 hours:

a. Reactor level and pressure continue to be maintained by RCIC.
b. Activities continue to restore both onsite and offsite AC power.

5.9.4.3 Primary Containment Heatup Analysis

The SBO scenario has been analyzed using the Modular Accident Analysis Program (MAAP) computer model to bound the effects of Primary Containment during the 4-hour SBO event (see Figure XIV-5-27). The transient modeling follows the event time line previously described. Additionally, at time 0 a concurrent small break LOCA of 66 gpm is modeled. This simulates recirculation pump seal leakage (2 pumps at 18 gpm each) plus 30 gpm from identified sources of leakage (see Section IV-10.3.1).

The peak calculated drywell temperature after 4-hours is 272°F, which is less than the CNS drywell design temperature of 281°F. The peak calculated torus pressure is 34 psia, which is below the minimum RCIC turbine exhaust high pressure trip operating setpoint. The maximum calculated suppression pool temperature is approximately 179°F, which is well below the temperature where emergency procedures would require ADS actuation. Based on these results, emergency reactor depressurization would not be required, and the RCIC system would operate satisfactorily for the 4-hour coping duration. In addition, essential electrical equipment inside the drywell would not exceed their environment qualification.

6.0 ANALYSIS OF DESIGN BASIS ACCIDENTS

6.1 Introduction

The methods for identifying and evaluating accidents (USAR Section XIV-4.3) have resulted in the establishment of design basis accidents for the various accident categories as follows:

	Accident Category	Design Basis Accident
a.	Accidents that result in radioactive material release from the fuel with the Reactor Coolant Pressure Boundary, Primary Containment, and Secondary Containment intact.	Control Rod Drop Accident (single control rod)
b.	Accidents that result in radioactive material release directly to the primary containment.	Loss-of-Coolant Accident (rupture of one recirculation loop)
c.	Accidents that result in radioactive material release directly to the Secondary Containment with the Primary Containment initially intact.	Accidents in this category are less severe than those in Categories "d" and "e" below
d.	Accidents that result in radioactive material releases directly to the Secondary Containment with the Primary Containment not intact.	Fuel Handling Accident (fuel assembly drops on core during refueling)
e.	Accidents that result in radioactive material releases outside the Secondary Containment.	Main Steam Line Break Accident (main steam line breaks outside of Secondary Containment)

An investigation of accident possibilities reveals that accidents in Category "c" are less severe than those in Categories "d" and "e." Category "c" includes two varieties of accidents: failures of the Reactor Coolant Pressure Boundary inside the Secondary Containment and failures involving fuel that is located outside the Primary Containment but inside the Secondary Containment. Under the accident selection rules described in USAR Section XIV-4.3, a main steam line break inside the Reactor Building is the most severe accident of the first variety, but this accident results in a radioactive release to the environs no greater than that resulting from the main steam line break outside the Secondary Containment. Similarly, the most severe accident of the second variety is the dropping of a fuel assembly into the fuel pool, but this results in a smaller radioactivity release to the environs than that resulting from dropping a fuel assembly on the fuel in the reactor vessel during refueling. Because the consequences of accidents in Category "c" are less severe than those resulting from similar accidents in other categories, the accidents in Category "c" are not described.

Two of the more significant design bases accidents were analyzed for CNS Cycle 20. These are:

1. The short term Primary Containment response to the LOCA (Section 6.3.7.1) was analyzed for CNS Cycle 20 which includes the effects of MELLL and ICF and the use of improved vessel blowdown models.

2. In addition, the reactor pump seizure event was analyzed for single loop operation (Section 5.5.3).

Note that the design basis radiological consequences are presented in the following sections of this chapter. Supplemental radiological consequences descriptions and methods are described in USAR Section XIV-8.

6.2 Control Rod Drop Accident

The accidents that result in releases of radioactive material from the fuel with the Reactor Coolant Pressure Boundary, Primary Containment, and Secondary Containment initially intact are the results of various failures of the Control Rod Drive System. Examples of such failures are collet finger failures in one control rod drive mechanism, a control drive system pressure regulator malfunction, and a control rod drive mechanism ball check valve failure. None of the single failures associated with the control rods or the control rod system results in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this Control Rod Drop Accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with all other barriers initially intact.

6.2.1 Identification of Causes

There are many ways of inserting reactivity into a boiling water reactor; however, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. It is possible, however, that a rapid removal of a high worth control rod could result in a potentially significant excursion; therefore, the accident which has been chosen to encompass the consequences of a reactivity excursion is the Control Rod Drop Accident (CRDA).

6.2.2 Frequency Classification

The Control Rod Drop Accident is classified as a design basis accident (limiting fault).

6.2.3 Starting Conditions and Assumptions

A highly improbable combination of actual events would be required for the design basis Control Rod Drop Accident to occur. The following events are required:

a. The complete rupture, breakage or disconnection of a fully inserted control rod drive from its cruciform control blade at or near the coupling.

The design of the drive and its coupling uses high quality materials and it receives stringent quality control and testing procedures appropriate to other equipment typically listed in the critical component list for the plant. Additionally, tests conducted under both simulated reactor conditions and conditions more extreme than those expected in reactor service have shown that the drive (or coupling) retains its integrity even after thousands of scram cycles. Tests also show that the drive and coupling do not fail when subjected to forces 20 times greater than can be achieved in a reactor.

b. Sticking of the control rod blade in its fully inserted position as the drive is withdrawn.

Sticking of the control blade in its fully inserted position is highly unlikely because each blade is equipped with rollers that make contact with the nearly flat fuel channel walls, traveling in a gap of approximately 1/2 in. clearance. Since a control blade weighs approximately 186 pounds, even if it separates from its drive, gravity forces would tend to make the blade follow its drive movement as if it were connected. Control blades of the current design in use in operating reactors have exhibited no tendency to stick. Starting with Cycle 24, the control blades used will have spacer pads in lieu of rollers at the top of the blade. The blades will also weigh approximately 205 pounds. As the control rod drop accident assumes the blade is stuck and a higher weight would enhance the blade's ability to follow its drive movement, these changes will not affect the above assumptions.

c. Full withdrawal of the control rod drive within the constraints of Banked Position Withdrawal Sequence (BPWS).

In order to limit the worth of the rod which could be dropped, the Rod Worth Minimizer (RWM) is used below 10 percent power to control the sequence of rod withdrawal. The RWM is programmed to follow the BPWS as described in Section VII-16.3.3. When the BPWS is followed, a rod drop would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/g.

In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10 percent of rated. For all normal and abnormal rod patterns, including those which maximize the worth of individual control rods, it is impossible to reach 280 cal/g in the event of a control rod drop occurring at power greater than 10 percent.

The RWM serves as a backup to procedural control on control rod sequences. In the event that the RWM is out of service, when required, the BPWS is enforced administratively, as provided for in Technical Specification 3.3.2.1.

d. Loss of offsite power is assumed coincidental with the CRDA. Loss of offsite power results in a loss of cooling water to the main condenser and sealing steam to the turbine with eventual loss of condenser vacuum. On loss of condenser vacuum the main steam lines will isolate to prevent overpressurization of the main condenser. However, for purposes of the dose analysis credit is not taken for MSIV closure.

This unlikely set of circumstances makes possible the rapid removal of a control rod. The dropping of the rod results in a high local reactivity in a small region of the core and for large, loosely coupled cores, significant shifts in the spatial power generation during the course of the excursion. Therefore, the method of analysis must be capable of accounting for any possible effects of the power distribution shifts.

6.2.4 Sequence of Events and Systems Operation

The sequence of events and approximate time of occurrence for this accident are described below.

	Event	Approximate Elapsed Time
(a)	Reactor is at a control rod pattern corresponding to maximum incremental worth.	
(b)	Rod worth minimizer or operators are functioning within constraints of BPWS. Maximum worth control blade that can be developed at any time in core life under any operating conditions while employing the BPWS becomes decoupled from the control rod drive.	

	Event	Approximate Elapsed Time
(c)	Operator selects and withdraws the control rod drive of the decoupled maximum worth rod along with the other rods assigned to the Banked-position group such that the proper core geometry for the maximum incremental rod worth exists.	
(d)	Decoupled control blade sticks in the fully inserted position.	
(e)	Blade becomes unstuck and drops at the nominal measured velocity determined from experimental data (3.11 fps).	0
(f)	Reactor goes on a positive period and initial power burst is terminated by the Doppler Reactivity Feedback.	< 1 sec
(g)	APRM High Neutron Flux signal scrams reactor (Analysis assumes 120% which is conservative because the IRM scram would be operative and supports the analytical APRM scram setpoint of 123%).[69]	
(h)	Scram terminates accident.	< 5 sec

6.2.5 Core and System Performance

Since peak fuel enthalpy is the most important single parameter for determining the severity of the CRDA and the onset of fuel rod failure, results are presented as a function of the resultant peak fuel enthalpy. As reference points, the following design and fuel failure criteria have been accepted by the NRC (NUREG-0800, Section 15.4.9).[61]

(1) Enthalpy = 170 cal/g, cladding failure threshold

(2) Enthalpy = 280 cal/g, specific energy design limit

(3) Enthalpy = 425 cal/g, prompt fuel dispersal threshold

The nuclear excursion analysis methods used for the CRDA evaluation are in accordance with NSSS Supplier guidance.[49] Techniques and models used to analyze the CRDA have been documented by General Electric.[4][5][6][7][56] This documented information has been used for the development of design approaches to make the consequences of a CRDA acceptable.

The analysis of the CRDA was performed by General Electric on a generic (bounding analysis) basis for plants with BPWS.[4] Rod drop accident calculations using the low incremental rod worths which result from following the BPWS indicate that peak fuel enthalpy is well below the 280 cal/g design limit and rarely exceeds the 170 cal/g fuel cladding failure threshold.

CRDA results from BPWS plants have been statistically analyzed.[57] The results show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/g limit even with a maximum incremental rod worth corresponding to 95 percent probability at the 95 percent confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA from the standard GE BWR reload package for the BPWS plants.

The rod drop excursion model can readily accommodate different fuel types and covers all fuel types up to GNF2.[48][88][96] The evaluation of the CRDA thus consists of ensuring that the appropriate parameters of the new reloaded core are bounded by the input parameter values used in the generic analysis.

The generic analysis assumes the slowest scram allowed by the Technical Specifications (and that the dropped rod does not scram), the most rapid credible rod drop velocity, and the smallest (i.e., high exposure) value for delayed neutron fraction. The remaining parameters of interest include the Doppler feedback, the scram reactivity, and the accident reactivity characteristics.

The effects of axial gap formation due to fuel densification on the CRDA results have been evaluated.[58] Based on this evaluation, it has been established with 99 percent probability that increased local peaking in any fuel rod due to the formation of axial gaps will be less than 5 percent. This effect has been accommodated in the analysis by adjusting the local peaking factor.

If any of the key CRDA inputs for a plant-specific reload application do not fall within the bounding analysis inputs then the accident analysis must be reanalyzed on a plant specific basis. In such a case, the plant specific analyses will be performed using an actual hot and/or cold Doppler coefficient of reactivity corresponding to the beginning of cycle, which is most limiting for this accident since the Doppler coefficient is least negative at BOC. The other key parameters will also be at their worst case plant specific values.

The results of the combined generic and/or plant-specific CRDA analyses must show that the positive reactivity insertion rate of the dropped rod is compensated sufficiently by the negative Doppler and scram reactivity effects to limit the peak pellet enthalpy to a maximum of 280 cal/g.

Based on the bounding analysis, it was conservatively determined that the following number of fuel rods would reach a fuel enthalpy of 170 cal/g, which is the enthalpy limit for eventual cladding perforation:

GNF2 fuel: 1200 fuel rods fail

This analysis is generic for all BWRs and verification that the results are bounding for CNS are supplied in the cycle specific supplemental reload licensing report.[35] These include comparisons to show that the Doppler reactivity coefficient, accident reactivity functions, and scram reactivity functions are all within the generic boundary values provided by the NSSS supplier.[49]

6.2.6 Barrier Performance

As described in USAR Section III-6.7, maintaining the peak fuel enthalpy less than the design value of 280 cal/g ensures that the reactor coolant pressurization rate will be less than 50 psi/sec and pose no threat to the Reactor Coolant Pressure Boundary.

6.2.7 Radiological Consequences

The radiological consequences of the CRDA are based on an assumed fuel failure occurring as a result of a rapid removal of a high worth control rod. This results in a high local reactivity in a small region of the core. The radiological consequences methodology uses NUREG-0800 Standard Review Plan 15.4.9 as a basis for the evaluation. As a result of the accident, radionuclides are released from the damaged fuel rods to the Main Condenser. A single release path is modeled from the Main Condenser to the Turbine Building.

From this release, doses are calculated for individuals offsite. The release of radionuclides to the environment can also result in a dose to the Control Room occupants due to intake of contaminated air via the Main Control Room Air Conditioning System.

The radiological consequences of the CRDA were assessed using the AXIDENT software code. This code was used to calculate the Control Room occupant, Exclusion Area Boundary, and Low Population Zone whole body, beta, and thyroid doses.[79]

6.2.7.1 Fission Products Released From the Fuel to the Main Condenser

The radionuclide source term during a CRDA is based on the fraction of the reactor core that is damaged as a result of the accident. The bounding source term considers a reactor core containing GNF2 fuel and a 24 month fuel cycle.

The core inventory source term was calculated using the isotope generation and depletion code ORIGEN2, which incorporates the BWR extended burnup library BWRUE. The core is assumed to have been operated at rated thermal power (plus 0.398% to account for uncertainties in power measurement).

As discussed in USAR Section XIV-6.2.5, 1200 10x10 (GNF2) fuel rods are assumed to fail. Each GNF2 fuel bundle has an equivalent of 85.6 fuel rods per assembly, some of which are part length, resulting in a fractional number of equivalent fuel pins per assembly. With 548 fuel bundles in the core the fraction of failed fuel is calculated to be 1200/(85.6 rods x 548 fuel bundles) or 0.0256. The mass fraction of the fuel in the damaged rods that melt is 0.0077, resulting in an overall core melt fraction of 1.97E-4 (0.0256 damaged rods per core x 0.0077 melted rods per damaged rods). The resulting core damage fraction that does not melt is 0.0254.

The activity released from the melted fuel is the product of the core melt fraction (1.97E-4), the radial peaking factor, and the radionuclide release fraction. A radial peaking factor of 2.0 is applied to the radionuclide inventory calculation for GNF2 fuel. The fraction of radionuclides released from the damaged rods is 100% of the total noble gases, and 50% of the total radioactive iodine in the rods at the time of the accident. The activity released from the damaged (but not melted) fuel is the product of the unmelted core damage fraction (0.0254), the radial peaking factor (2.0), and the fuel-clad gap fraction. The gap fraction used is 0.1 for noble gas and iodine.

Following the rod drop event, the activity from the failed fuel is assumed to mix instantaneously with the reactor coolant. The analysis assumes that 10% of all the iodines and 100% of all the noble gases are transported to the Main Condenser. A conservative Decontamination Factor of 10 is applied for all of the iodines in the release path to reflect the partitioning and plateout that occurs in the Main Condenser. All of the noble gases are available for release from the Main Condenser. Table XIV-6-1 describes the resulting source term that reaches the Main Condenser by isotope.

6.2.7.2 Fission Products Released From the Turbine Building

The fission product transport to the environs is modeled as a single release path resulting from leakage from the Main Condenser to the

TABLE XIV-6-1

DETERMINATION OF THE CRDA SOURCE TERM

Isotope	ORIGEN2 Ci/MWt (1)	MWt	Uncertainty Factor	Total Core Inventory	Activity Released From Damaged Fuel Pins (Ci)	Activity Released From Melted Fuel Pins (Ci)	Total Activity Released From Fuel Pins (Ci)	Fraction Transported to the Main Condenser	Fraction Released From Main Condenser	AXIDENT Correction Factor (2)	CRDA Source Term (Ci)
I-131	2.72E+04	2419	1.00398	6.61E+07	3.35E+05	1.30E+04	3.48E+05	0.1	0.1	4	1.39E+04
I-132	3.96E+04	2419	1.00398	9.62E+07	4.88E+05	1.89E+04	5.07E+05	0.1	0.1	4	2.03E+04
I-133	5.48E+04	2419	1.00398	1.33E+08	6.76E+05	2.62E+04	7.02E+05	0.1	0.1	4	2.81E+04
I-134	6.04E+04	2419	1.00398	1.47E+08	7.45E+05	2.89E+04	7.74E+05	0.1	0.1	4	3.09E+04
I-135	5.16E+04	2419	1.00398	1.25E+08	6.36E+05	2.47E+04	6.61E+05	0.1	0.1	4	2.64E+04
Xe—131m	3.04E+02	2419	1.00398	7.38E+05	3.75E+03	2.91E+02	4.04E+03	1	1	1	4.04E+03
Xe—133m	1.73E+03	2419	1.00398	4.20E+06	2.13E+04	1.66E+03	2.30E+04	1	1	1	2.30E+04
Xe-133	5.45E+04	2419	1.00398	1.32E+08	6.72E+05	5.21E+04	7.24E+05	1	1	1	7.24E+05
Xe—135m	1.10E+04	2419	1.00398	2.67E+07	1.36E+05	1.05E+04	1.46E+05	1	1	1	1.46E+05
Xe-135	2.04E+04	2419	1.00398	4.95E+07	2.52E+05	1.95E+04	2.71E+05	1	1	1	2.71E+05
Xe-138	4.49E+04	2419	1.00398	1.09E+08	5.54E+05	4.30E+04	5.97E+05	1	1	1	5.97E+05
Kr-83m	3.25E+03	2419	1.00398	7.89E+06	4.01E+04	3.11E+03	4.32E+04	1	1	1	4.32E+04
Kr-85m	6.75E+03	2419	1.00398	1.64E+07	8.32E+04	6.46E+03	8.97E+04	1	1	1	8.97E+04
Kr-85	4.26E+02	2419	1.00398	1.03E+06	5.25E+03	4.08E+02	5.66E+03	1	1	1	5.66E+03
Kr-87	1.28E+04	2419	1.00398	3.11E+07	1.58E+05	1.22E+04	1.70E+05	1	1	1	1.70E+05
Kr-88	1.81E+04	2419	1.00398	4.40E+07	2.23E+05	1.73E+04	2.40E+05	1	1	1	2.40E+05

Note (1): The radionuclide release to the environs used in AXIDENT is based on the ORIGEN2 generated source term, as compiled in this table.

Note (2): The AXIDENT code core release fractions are based on Loss-of-Coolant Accident assumptions. To obtain the correct iodine release for a Control Rod Drop Accident the iodine source term used as input to the AXIDENT code must be increased by a factor of 4 to compensate for the AXIDENT code iodine release fractions.

Turbine Building. The CRDA source term will not reach the environment via the offgas treatment system flow paths as justified by the following:

1. If the event occurred at low reactor power levels, when the Steam Jet Air Ejectors (SJAEs) are not in service, the mechanical vacuum pumps are used to remove noncondensables from the Main Condenser. After the event occurs, a Main Steam Line high radiation signal resulting from the CRDA would immediately trip the mechanical vacuum pumps and close the mechanical vacuum pump inlet and outlet valves. Thus, there would be no motive force to draw the noncondensables from the Main Condenser into the offgas system.

2. If the event occurred at other power levels when the SJAEs were in service, a 30-minute holdup line downstream of the SJAE exhaust provides for decay of fission gases. A CRDA would cause a SJAE offgas radiation monitor high radiation signal to initiate a 15-minute timer. After 15 minutes, this isolates the offgas system downstream of the 30-minute holdup line. Thus, the fission gas released from the CRDA will be isolated prior to exiting the 30-minute holdup line.

The analysis assumes that the Main Condenser leaks directly to the Turbine Building atmosphere at a rate of 1% of the volume per day for an evaluation period of 24 hours. Credit is taken for radioactive decay during this period of holdup in the Main Condenser. The leakage to the Turbine Building is assumed to pass directly to the environment with no mixing or holdup in the Turbine Building volume. The activity is distributed throughout the Turbine Building and egresses as a diffuse ground level release.

6.2.8 Radiological Effects

6.2.8.1 Offsite Consequence Results

The offsite consequences in terms of the radiological doses resulting from the activity released to the environment during a Control Rod Drop Accident have been determined based on the calculated Turbine Building atmospheric dispersion factors (X/Q) for the Exclusion Area Boundary and Low Population Zone shown on Table XIV-6-2. These X/Q values were generated using the methodology presented in Regulatory Guides 1.3 and 1.25. Building wake and fumigation considerations are factored into the atmospheric dispersion factor determinations.

Two dose periods were evaluated, a 2-hour dose period at the Exclusion Area Boundary (EAB) and a 30-day dose period for the Low Population Zone (LPZ). The EAB and LPZ radiological consequences for the Control Rod Drop Accident have been assessed using the AXIDENT software code. The code was used to calculate the whole body, beta, and thyroid doses at these receptor locations. The AXIDENT assessment results are shown on Table XIV-6-3, and are within the SRP 15.4.9 dose acceptance criteria of 25% of the 10CFR100 guidelines values.

It is concluded that this accident will not result in any radiological doses which endanger the health and safety of the public.[79]

6.2.8.2 Onsite (Control Room Personnel) Consequence Results

The Control Room occupant radiological doses from a Control Rod Drop Accident were assessed using the AXIDENT software code. The doses have been determined based on the calculated Turbine Building atmospheric dispersion factors (X/Q) for the Control Room Air Conditioning System ventilation intake, with consideration of the effects of the Control Room Emergency Filter System (CREFS). The Turbine Building X/Q values for the Control Room dose calculations were generated using the ARCON96 software code

TABLE XIV-6-2

X/Q VALUES FOR THE EXCLUSION AREA BOUNDARY AND LOW POPULATION ZONE

X/Q Values for the Exclusion Area Boundary

Time Period	X/Q Value (sec/m^3)	Comments
0 to 2 hours	5.2E-4	Turbine Building Ground level release

X/Q Values for the Low Population Zone

Time Period	X/Q Value (sec/m^3)	Comments
0 to 8 hours	2.9E-4	Turbine Building Ground level release
8 to 24 hours	7.3E-5	Turbine Building Ground level release

X/Q Values for the Control Room Intake

Time Period	X/Q Value (sec/m^3)	Occupancy Factor	X/Q Value (sec/m^3) Adjusted for Occupancy	Comments
0 to 2 hours	9.54E-4	1	9.54E-4	Turbine Building Vent level release
2 to 8 hours	4.93E-4	1	4.93E-4	Turbine Building Vent level release
8 to 24 hours	2.69E-4	1	2.69E-4	Turbine Building Vent level release

TABLE XIV-6-3

CRDA EXCLUSION AREA BOUNDARY, LOW POPULATION ZONE, AND CONTROL ROOM RADIOLOGICAL DOSE CONSEQUENCES

Exclusion Area Boundary Doses Over 2 Hours[1]

Thyroid (rem)	Whole Body (rem)	Beta (rem)
0.788	0.105	0.049

Low Population Zone Doses Over 30 Days[1]

Thyroid (rem)	Whole Body (rem)	Beta (rem)
2.06	0.111	0.081

Control Room Occupant Doses Over 30 Days[2]

Thyroid (rem)	Whole Body (rem)	Beta (rem)
6.53	0.0082	0.191

1. The regulatory acceptance limits for offsite dose is 75 rem Thyroid and 6 rem Whole Body.

2. The regulatory acceptance limit for Control Room occupant dose is 30 rem Thyroid, 5 rem Whole Body, and 30 rem Beta Skin dose.

and site specific meteorology for the years 1994-1998. The Turbine Building release path was modeled as a diffuse release using the vent level release mode of ARCON96. The results are shown on Table XIV-6-2.

During the first 24 hours of the accident, the release from the Turbine Building is assumed to enter the Control Room via the normal intake to the Control Room Air Conditioning System. For purposes of maximizing the calculated Control Room doses, the Control Room Emergency Filter System (CREFS) is assumed to not initiate prior to 24 hours after the accident. The resulting radionuclide concentration within the Control Room Envelope is diluted by the air space volume (64,640 ft^3 for the Control Room proper, and 141,860 ft^3 for the entire Control Room Envelope).

The AXIDENT software code was used to calculate the whole body, beta, and thyroid doses at the Control Room receptor location. The results of the AXIDENT assessment are presented on Table XIV-6-3. The results are within the dose limits of 10CFR50 Appendix A, GDC 19.[79]

6.3 Loss-of-Coolant Accident

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The information being presented in Section XIV-6.3.7.1 as historical has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR, as amended.

Accidents that could result in release of radioactive material directly into the Primary Containment are the result of postulated nuclear system pipe breaks inside the drywell. There are no realistic, identifiable events which would result in a pipe break inside the Primary Containment of the magnitude required to cause a Loss-of-Coolant Accident (LOCA).

However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the cause being identified. These accidents result in the heating and pressurization of the Primary Containment, a challenge of the Emergency Core Cooling Systems, and the potential release of radioactive material to the environs.

The entire spectrum of break sizes is evaluated to determine the most limiting break size and location. Instantaneous guillotine double-ended breaks of the Reactor Recirculation piping, Main Steam Line at RPV nozzle, Feedwater Line at RPV nozzle, and Core Spray Line at RPV nozzle are investigated.

The following LOCA analysis describes the impact of the LOCA on the Primary Containment and the radiological consequences. There are two basic types of analyses, short-term and long-term. The short-term analysis describes the calculation of peak Primary Containment pressure which occurs within the first 30 seconds. The long-term analysis describes the calculation of pressure and temperature history in the drywell and wetwell for several minutes following the accident.

Other LOCA analyses in USAR Chapter VI describe the impact on the core, ECCS equipment and suppression pool temperature for NPSH. The ECCS system LOCA analysis in USAR Section VI-5.2 demonstrates CNS conformance with the requirements of 10CFR50.46 including specific inputs documented in 10CFR50 Appendix K. The NPSH LOCA analysis in USAR Section VI-5.3 demonstrates the adequacy with regard to NPSH at the various ECCS pumps and is used to calculate the suppression pool temperature and the minimum Primary Containment pressure following the design basis LOCA.

The small steam line break LOCA and the impact on Primary Containment temperature is described in USAR V-2.4.2.1. The use of containment spray ensures that the containment structural design temperature of 281°F is not exceeded.

Diesel generator electric loading profiles required to support ECCS operation following a LOCA are evaluated in USAR Section VIII-5. Service Water System flow requirements following a LOCA are discussed in USAR Section X-8.

6.3.1 Identification of Causes

All possibilities for pipe break sizes and locations have been investigated including the severance of small lines, the main steam lines upstream and downstream of the flow restrictors, and the Reactor Recirculation loop lines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the Reactor Recirculation loop lines. This accident is established as the design basis LOCA.

6.3.2 Frequency Classification

The LOCA is classified as a design basis accident (limiting fault).

6.3.3 Starting Conditions and Assumptions

The starting conditions and assumptions for the design basis LOCA analyses are found in the USAR Sections XIV-6.3.7 and XIV-6.3.8.

6.3.4 Sequence of Events and Systems Operation

The sequence of events and system operation for each scenario for the design basis LOCA are described in USAR Section XIV-6.3.7.

6.3.5 Identification of Operator Actions

Safety related equipment is designed so that no operator action is required for 10 minutes following the LOCA.

The small steam line break LOCA and the impact on Primary Containment temperature is described in USAR V-2.4.2.1. At 600 seconds, the containment spray mode of RHR is used. The use of containment spray at 600 seconds ensures that the containment structural design temperature of 281°F is not exceeded.[99]

From the DBA LOCA containment analysis, the RHR pumps are switched from the LPCI mode to containment cooling at 600 seconds.[85]

From the Mark I Containment Program Load Definition Report, for the Small Break Accident scenario, the operator is evaluated for rapidly depressurizing the reactor via ADS after 10 minutes. This is not an anticipated operator action following the SBA, but ensures a conservative evaluation of the hydrodynamic loads on the Wetwell and Vent System Structures following a LOCA.[31]

From LOCA dose calculations, one of the running Standby Gas Treatment trains is secured within one hour of the accident. This provides margin and ensures the regulatory dose limits will not be exceeded following a DBA LOCA.[80]

As part of the licensing of the LOCA dose calculations, the NRC has required that the MSIV leakage pathway be evaluated as being capable of withstanding the seismic loadings of a postulated Safe Shutdown Earthquake. In order to minimize the leakage past the seismic analysis boundaries and to establish the pathway as shown on drawing CNS-MS-43, certain post-LOCA valve manipulations are performed. These manual actions are initiated after alarms are received in the Control Room indicating a LOCA with core damage.

Manual injection of SLC is also credited as part of the licensing of the LOCA dose calculations. Injection of sodium pentaborate using the SLC system within approximately 12 hours after a LOCA is sufficient to maintain pH above 7.0 for the 30-day duration of the accident. Consequently, iodine re-evolution from the suppression pool need not be considered in the LOCA dose analysis with the Alternative Source Term. SLC injection will be initiated within 6 hours in order to ensure the injection is completed prior to the area becoming an EQ harsh environment.

6.3.6 Core and System Performance

For the short-term DBA scenario, the reactor will automatically scram due to high drywell pressure. Main steam line isolation will occur due to low reactor water level. No mechanical S/RV actuation will occur because of the rapid reactor vessel depressurization and large rate of reactor fluid and energy inventory loss through the break. Shortly after the postulated pipe break, the ECCS automatically begins to pump water from the plant emergency condensate storage tank and/or the suppression pool into the reactor pressure vessel to flood the reactor core. Following vessel flooding and drywell/wetwell airspace pressure equalization, suppression pool water is continually recirculated from the pool to the reactor vessel by the ECCS pumps. The core decay power results in a slow heat-up of the suppression pool. The suppression pool cooling mode of the RHR system is manually actuated to remove energy from the suppression pool to return the Primary Containment to normal temperature conditions.

For the long-term DBA scenario, the Core Spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water.

As described in Chapter VI, analysis of the consequences of this accident demonstrates that the integrated performance of the ECCS in conjunction with restrictions on MAPLHGR will ensure compliance to the acceptance criteria of 10CFR50.46. These criteria have been established to prevent fuel damage as a result of this accident.

6.3.7 Barrier Performance (Primary Containment Response)

This section describes the analysis of Primary Containment response to a DBA LOCA event for Cooper Nuclear Station. There are two basic types of analyses, which use slightly different methods for the analysis. The short-term analysis, in USAR Section XIV-6.3.7.1, describes the calculation of peak Primary Containment pressure which occurs within the first 30 seconds. The long-term analysis, in USAR Section XIV-6.3.7.2, describes the calculation of pressure and temperature history in the drywell and suppression chamber for several minutes following the accident.

6.3.7.1 Short-Term Primary Containment Response Analysis

The following assumptions and initial conditions were used in calculating the effects of a LOCA on the Primary Containment during the initial 30 seconds following a DBA LOCA for the Mark I Containment Program.

a. The reactor is assumed to be scrammed at the time of accident initiation. Actually, scram will occur in less than one second from receipt of the high drywell pressure signal, but the difference in shutdown time between zero and one second is negligible.

b. The main steam line isolation valves were assumed to start closing at 0.5 seconds after the accident, and the valves were assumed to be fully closed in the shortest possible time of three seconds following closure initiation. Actually, the closures of the main steam line isolation valves are expected to be the result of low water level, so these valves may not receive a signal to close for over four seconds, and the closing time could be as high as 10 seconds. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the Primary Containment.

c. The reactor control volume is assumed to include the fluid and structural masses and energy of the feedwater line to the point in the feedwater system where the temperature during normal operation is equal to the saturation temperature at the final calculated reactor vessel pressure. Feedwater mass below this temperature will not flash during reactor vessel depressurization and therefore will not discharge to the Primary Containment.

d. The blowdown model is the Homogeneous Equilibrium Model (HEM) and the blowdown model replaces the slip flow model.

e. The pressure response of the Primary Containment is calculated assuming:

1. During the time at which the peak drywell pressure occurs, the thermodynamic condition in the drywell can be either a homogeneous mixture of air, saturated vapor and water, or a homogeneous mixture of air and superheated vapor. The type of break together with the assumptions made concerning the energy exchange between the two phases existing in the drywell determines which of the two homogeneous mixtures will exist at any particular time in the transient. As the numerical integration proceeds, there is a continuous check of the drywell atmosphere. The bulk wetwell pool and airspace temperatures are assumed equal throughout the transient. This assumption maximizes the wetwell airspace temperatures and pressures.

2. The constituents of the fluid flowing in the drywell to suppression chamber vents are based on a homogeneous mixture of the fluid in the drywell. The consequences of this assumption result in complete liquid carryover into the drywell vents. Actually, some of the liquid will remain behind in a pool on the drywell floor so that the calculated drywell pressure is conservatively high.

3. The flow of liquid, steam and air in the vent system is assumed to be a homogeneous mixture based on the instantaneous mass fractions in the drywell. This assumption yields increased vent system flow density resulting in higher flow losses and therefore higher drywell pressure. These conservative drywell pressures were used as the forcing function for the plant unique pool swell tests. Steam condensation on drywell structures and internal components is conservatively neglected.

4. No heat loss from the gases inside the Primary Containment is assumed. This adds extra conservatism to the analysis, i.e., the analysis will tend to predict higher Primary Containment pressures than would actually result.

The Cooper Primary Containment response for the LOCA has been evaluated using the model presented in NEDO-10320, 10329 and 21888. The assumptions, predicted Primary Containment response, deviations, justifications and conclusions are given below.[12]

The current analyses include the effects of Maximum Extended Load Line Limit (MELLL) and Increased Core Flow (ICF) and use improved vessel blowdown models. The purpose of MELLL and ICF is to provide operating flexibility by expanding the power/flow map. Only the short-term Primary Containment response, which determines the peak drywell pressure and temperature, as well as the containment hydrodynamic loads, could potentially be impacted as a result of increased vessel subcooling which will increase the blowdown flow rates during a postulated DBA LOCA. The peak wetwell and suppression pool temperatures are reached due to the long-term release of the decay heat and the sensible energy from the reactor vessel to the Primary Containment. Since MELLL and ICF do not increase the reactor power level nor the vessel operating pressure, neither the decay heat nor the vessel sensible energy is increased. Thus, the peak wetwell and suppression pool temperature are not impacted by the MELLL and ICF operation. Therefore, no reanalysis of the long-term Primary Containment response is necessary.

The MELLL and ICF analysis, with improved vessel modeling[82] for calculating the blowdown flow rates and flow enthalpies, gives a peak drywell pressure of 54.4 psig and a peak drywell temperature of 301.4°F. These peak values were obtained for the power/flow point of 102%P/75%F (MELLL point). The peak drywell pressure of 54.4 psig is below the previously calculated value of 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission)[59], and the peak drywell temperature of 301.4°F is maintained for a short time and does not raise the structural drywell temperature above the design value of 281°F.[85] The Primary Containment hydrodynamic loads also remain below the values previously defined in the PULD[30] and CNS Plant Unique Analysis Report.

Blowdown flow rates and flow enthalpies for the MELLL point of 102%P/75%F are given below:

Time (seconds)	Liquid Flow (lb/sec)	Liquid Enthalpy (BTU/lb)	Vapor Flow (lb/sec)	Vapor Enthalpy (BTU/lb)
0	0	568.2	0	1190
0.0713	45931	515.7	69	1198
0.3525	42168	516.9	132	1198
0.9775	41587	521.8	113	1197
1.6025	40109	524.7	91	1197
3.8057	28659	507.8	1041	1200
4.9307	25059	498.4	1341	1201
7.0713	25644	516.9	2456	1198
10.6025	7148	439.1	3452	1205
15.8525	6453	412.2	2547	1204
30.1000	2191	317.6	1009	1192

Figure XIV-6-16a shows the Primary Containment pressure response for the MELLL analysis at the power/flow point of 102%P/75%F.

Immediately prior to the postulated design basis LOCA, the following conditions are assumed to exist in the Primary Containment:[30]

	Drywell	Wetwell
Air Volume, Ft^3	132,465	106,850
Downcomer Submergence (ft)	-	3.333
Suppression Pool Level		Highest level within the normal operating range
Temperature, °F	135	95
Pressure, psig	0	0
Relative Humidity %	20	100

The overall vent resistance factor for the Cooper Primary Containment is 5.51.[31]

The results of the short-term Primary Containment analyses (see also Table V-2-1) are:

Maximum Accident Pressures for the Drywell / Suppression Chamber

Design Basis Accident (original analysis)........ 46.2 psig / 29.0 psig

NEDO 10320 Accident Analysis................. 58.2 psig / Not Available

Accepted value of peak calculated containment pressure for 10 CFR Part 50, Appendix J leakage rate testing (P_a) .... 58 psig[59]

[The peak containment pressure of 58.2 psig was determined using the methodology of NEDO-10320. This methodology has been confirmed to be excessively conservative. The current licensing basis calculation yields a more realistic, yet conservative, value of 54.4 psig. Based on this, 58 psig is considered to be a very conservative value of peak containment pressure following a postulated LOCA.]

Mark I Containment Program DBA.................. 51.4 psig / 24.3 psig

Mark I Containment Program IBA.................. 31.2 psig / 28.9 psig

Mark I Containment Program SBA.................. 23.2 psig / 21.3 psig

The short-term transient response of the drywell and wetwell to a DBA was reevaluated under the Mark I Containment Program.[29] The models used and the assumptions made are briefly described in the Mark I Containment Program Load Definition Report.[31] The models were verified by comparison with model tests performed at the Bodega Bay and Humboldt Bay test facilities.

The DBA peak calculated drywell pressure was 51.4 psig and the DBA peak drywell temperature was 295°F.[30] Figures XIV-6-18 and XIV-6-19 show the Primary Containment pressure and temperature responses using this approach.

The peak pressure presented in the FSAR was 46.2 psig. In Figure XIV-6-20, it shows that the primary containment pressures reduces to 29 psig shortly after the discharge of the primary coolant from the reactor vessel into the drywell.

Figure XIV-6-16 shows the calculated short-term response of the containment using the above numerical values as input to the analytical model described in NEDO-10320. The peak calculated pressure is 58.2 psig. Section 6.2.1 of the Safety Evaluation of the Cooper Nuclear Station, dated February 14, 1973,[59] issued by the U.S. Atomic Energy Commission, cited 58 psig as the peak pressure in the drywell following a design basis LOCA. There is 6 percent margin between this and the maximum allowable pressure of 62 psig.

Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the suppression chamber via the vacuum-breaker system as the drywell pressure decreases due to steam condensation.

It will be shown later in this response, that the NEDO-10320 model overpredicts peak containment pressures whereas the analytical techniques used prior to the introduction of the NEDO-10320 model showed good agreement with available test data. The following is a discussion of both the design bases for the Cooper containment and the conservatisms inherent in the NEDO-10320 model. It will be shown that the increase in peak pressure from 46.2 to 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission)[59] is not real and that in fact there would be a very considerable margin between the actual peak pressure and the design pressure. This margin appears as a difference between the peak calculated pressure and the design pressure.

The apparent reduction in this margin resulting from the use of the NEDO-10320 model is due entirely to the considerable analytical conservatism of this model (as described later) and does not represent any reduction of the real margin.

The Cooper containment geometry is based directly on the full scale Bodega Bay Pressure Suppression Tests (Preliminary Hazards Summary Report, Bodega Bay Atomic Park, Unit 1, Pacific Gas and Electric Company, December 1962, Docket No. 50-205). The highest drywell pressure obtained during any test which had the design basis blowdown area was 52 psig. (Test No. 40). This test had a period of deliberate steam prepurging of the drywell immediately prior to the reactor blowdown, and this prepurging resulted in a drywell pressure of 20 psig at the time the blowdown started. The prepurged condition does not form the design basis for the full scale containments, so Test No. 40 is not really applicable to the containment design. However, despite this inapplicability, it was the conservative basis upon which the Cooper design pressure of 56 psig was selected.

In some respects, the test rig represented a more severe test than the design basis accident insofar as peak pressures are concerned. The vent geometry and downcomer design of the Cooper containment are the same as the Bodega Bay test facility but other design parameters of the full scale containment are more favorable than those which were tested. The following table (Table XIV-6-4) lists these parameters.

The primary system mass to pool mass ratio and primary system break area to pool volume ratio are both indicators of the thermal loading on the suppression pool. In both instances, the test facility has a less favorable ratio than the Cooper containment and thus a more severe pool thermal loading transient. Similarly, the Bodega Test rig had a greater ratio of drywell air to wetwell air. The significance of this is that following the start of blowdown most of the drywell air is rapidly pushed over into the wetwell; because of the adverse volume ratio, the resultant suppression chamber pressure in the tests was higher than that which would occur in the full scale containment (28 psig and 23 psig respectively). Since the drywell pressure is equal to the wetwell pressure plus the vent pressure losses, the net effect is to make the measured drywell pressure conservative.

Further conservatism is added by the fact that the reactor vessel for the tests was initially at 1250 psig. This is 200 psi higher than the maximum pressure at which the Cooper reactor will ever operate on a steady state basis. Thus, all else being equal, the mass velocity of the blowdown flow to the drywell was higher for the tests than for the full scale reactor. This would tend to make the measured values of vent pressure losses high which would in turn make the observed peak drywell pressure high.

The Cooper containment has a primary system break area to vent area ratio of 0.02 whereas the test facility had a ratio .0194. This is non-conservative only by a factor of 3 percent.

The preceding discussion has shown that Bodega test number 40 was a more severe transient than the DBA for the full scale containment and that basing the Cooper containment design conditions on the results of this test is a conservative procedure. Even for the unrepresentative conditions of test #40, the peak drywell pressure was only 52 psig. The average peak drywell pressure for all tests having a 3.24 inch blowdown orifice (equivalent to the DBA) and 4 feet vent submergence was 41.5 psig. This average drops to 35.3 psig if the unrepresentative prepurge tests are ignored. Thus it can be concluded that despite the predictions of the NEDO-10320 model, test data shows that the peak drywell pressure in the Cooper containment will not exceed 52 psig.

<u>NEDO-10320 CONSERVATISMS</u>

The CNS peak containment pressure calculated in the original analysis was 46.2 psig (see Figure XIV-6-20); the following is a discussion of the model changes which have been made in the interim and which have resulted in the peak calculated containment pressure increasing from 46.2 psig to 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission)[59].

The key difference between the initial calculational model and the model described in NEDO-10320 is in the treatment of the flow in the drywell to wetwell vent system. Both models assume that the flow is a homogeneous mixture of the drywell inventory of air, steam and water. However, when calculating the irreversible pressure losses associated with this flow, the initial model assumed that the liquid constituent of the flow was carried along as a fine mist contributing to the inertia of the flow but not to the irreversible losses. The rationale for this assumption was based on the fact that the void fraction of the vent flow would be .995. Figures XIV-6-1/2/3/4 demonstrate the good agreement between this model and available test data.

TABLE XIV-6-4

COMPARISON OF COOPER CONTAINMENT AND BODEGA BAY TEST FACILITY

	COOPER	BODEGA BAY
Drywell Free Volume, ft^3	132,250	1100
Wetwell Water Volume, ft^3	91,100	339
Wetwell Air Volume, ft^3	106,850	670
Total Vent Area, ft^2	215	3.012
Break Area, ft^2	4.285	.0573
Vent Submergence, ft (minimum)	4.0	3 to 5
Downcomer Diameter, ft	2	2
Primary System Blowdown Mass, lb	$.51x10^6$	$.24x10^4$
Reactor Pressure, psig	1050	1250
Ratio of Primary System Mass to Pool Mass	.090	.114
Ratio of Break Area to Wetwell Water Volume ft^{-1}	4.7×10^{-5}	16.7×10^{-5}
Ratio of Drywell to Wetwell Air Volume	1.24	1.64
Ratio of Break Area to Vent Area	.020	.0194

The vent flow model described in NEDO-10320 uses the density of the homogeneous vent flow when calculating the irreversible pressure losses. Since this typically represents a 300 percent increase in density compared to the previous model, the consequence of the change is a substantial increase in the calculated values of the irreversible pressure losses in the vents. This is the reason that the peak calculated pressure of 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission)[59] is higher than the 46.2 psig presented in the FSAR. The increase can be attributed entirely to the model change; there has been no change in the reactor containment system. As pointed out in NEDO-10320, the higher peak calculated containment pressure would not actually occur in the event of a LOCA. The increase is associated entirely with the added conservatism in the model. This conservatism is clearly demonstrated in Figures 5.21/22/23/24 of NEDO-10320, which indicates that the "new" model over predicts available data by anywhere up to 50 percent. This should be compared to the initial CNS model which showed good agreement with the data.

Thus, there is little technical justification for using the NEDO-10320 model beyond providing additional conservatism in the peak calculated containment pressure. Despite this added calculational conservatism, there is still 6 percent margin between the calculated peak pressure of 58.2 psig (accepted as 58 psig by the U.S. Atomic Energy Commission)[59] and the maximum allowable pressure of 62 psig.

In addition, it should also be noted that both the initial model and the new model discussed in NEDO-10320 ignore many phenomena which would, in the event of an actual LOCA, result in a peak containment pressure less than that predicted by either model. Examples of the phenomena are (1) the abundant evidence[13] that actual reactor primary system blowdown flow rates will be 60 percent to 80 percent of the theoretical maximum rates predicted by the Moody model and used in the containment calculations; (2) vapor entrainment in the blowdown flow rate and a more detailed modeling of the reactor primary system would both result in a less severe blowdown to the containment than that assumed for the peak pressure calculation; (3) condensation of blowdown steam on the containment walls and structures will certainly occur, but is ignored; (4) the models assume complete carryover of reactor blowdown liquid to the suppression pool. This maximizes both the vent pressure losses and the suppression chamber vapor pressure, and is thus conservative. Tests indicate that in practice there would be some liquid remaining in the drywell, (see Supplement 1 to NEDO-10320 for a discussion of water carryover fractions); (5) nucleate boiling cooling of the core is assumed throughout the transient; this maximizes the primary system pressure and, thus, the blowdown flow rate to the containment. However, this assumption is inconsistent with the results of the LOCA analysis presented in Section VI of the FSAR.

6.3.7.2 Long Term Containment Response Analysis

In event of a design basis LOCA, the Core Spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal-water reaction to less than 0.1 percent. The Core Spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression chamber via the drywell-to-suppression chamber vent pipes. Steam flow is negligible. The energy transported to the suppression chamber water is then removed from the Primary Containment by the RHR heat exchangers.

Prior to activation of the RHR containment cooling mode (arbitrarily assumed at 600 seconds after the accident), the RHR pumps (LPCI mode) have been adding liquid to the reactor vessel. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow, in addition to heat losses to the drywell walls, offers considerable cooling to the drywell and causes a depressurization of the Primary Containment as the steam in the drywell is condensed. At 600 seconds, the RHR pumps are assumed to be switched from the LPCI mode to the RHR containment cooling mode.

To assess the Primary Containment long term response after the accident, an analysis was made of the effects of various containment spray and cooling combinations and parameter relaxations.[47], [68] and [85] For all cases, one of the Core Spray loops is assumed to be in operation. The long-term pressure and temperature response of the Primary Containment to the DBA-LOCA was analyzed for the following RHR containment cooling mode conditions. The limiting cases are Cases E and F. Cases A-D have not been reanalyzed with the latest assumptions.

Case A - Operation of both RHR cooling loops – 4 RHR pumps, 4 RHR Service Water Booster pumps, 3 Service Water pumps, and 2 RHR heat exchangers - with containment spray.

Case B - Operation of one RHR cooling loop with 2 RHR pumps, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - with containment spray.

Case C - Operation of one RHR cooling loop with 1 RHR pump, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - with containment spray.

Case D - Operation of one RHR cooling loop with 1 RHR pump, 2 RHR Service Water Booster pumps, 2 Service Water pumps, and 1 RHR heat exchanger - no containment spray.

Case E - Operation of one RHR cooling loop with 1 RHR pump, 1 RHR Service Water Booster pump, 1 Service Water pump, and 1 RHR heat exchanger - with containment spray.

Case F – Operation of one RHR cooling loop with 1 RHR pump, 1 RHR Service Water Booster pump, 1 Service Water pump, and 1 RHR heat exchanger using suppression pool cooling.

The major assumptions used for Cases E and F in the long-term DBA-LOCA Primary Containment analysis were as follows:

1. The reactor has been operating at a power of 2429 MWt (equal to the Original Licensed Thermal Power of 2381 MWt plus 2% to account for power measurement uncertainties) per Regulatory Guide 1.49. The ANS 5.1 decay heat model is used assuming an exposure of 38,000 MWd/ST (amount of energy generated per unit metric-ton fuel mass), which represents a high fuel burnup, and, therefore, a high decay power condition.

2. Offsite power is assumed lost at the initiation of the accident and is not restored during the entire event.

3. The power required to operate Core Spray pump(s) and LPCI/Containment Cooling pump(s) is added to the Primary Containment heat load by increasing water temperature at the pump discharge accordingly.

4. At the initiation of the accident, the suppression pool has the minimum water volume of 87,650 ft^3.

5. The Service Water temperature remains at 95°F throughout the event.

6. During the event, the portion of feedwater in the Feedwater system that is higher in temperature than peak pool temperature is assumed to continue to return to the reactor vessel.

7. No heat loss from the Primary Containment to the Reactor Building airspace is assumed.

8. The minimum thermal capabilities specified in General Electric Drawing 729E211BB are used for the RHR heat exchangers. The

RHR Heat Exchanger K-value per loop is 218.5 Btu/sec-°F for Cases A and B, 198 Btu/sec-°F for Cases C and D, and 185 Btu/sec-°F for Case E.

9. The initial air space pressure is 0.75 psig in both the drywell and suppression chamber.

10. The initial air space relative humidity is 20 percent for the drywell and 100 percent for the suppression chamber.

11. Initially, the drywell temperature is 135°F, while the suppression pool temperature is at its maximum limit of 95°F.

The Primary Containment responses for the five DBA cases are evaluated with the above assumptions, utilizing the SHEX computer code[76] and an analytical model based on NEDO-10320 and NEDO-20533. Case E, the most limiting case, was reanalyzed to evaluate the impact of the ECCS parameter relaxations. A description of the long-term response model is provided in NEDO-10320, NEDO-20533, GENE-673-020-0993[76] and GENE-637-045-1293.[47] The values of key parameters used as input for long-term DBA-LOCA containment analysis are given in Table XIV-6-5. The calculated long-term pressure and temperature responses for the five cases are shown in Figures XIV-6-5 through XIV-6-9.

The initial pressure response of the Primary Containment (the first 600 seconds after break) is essentially the same for each of the five cases. During the long-term Primary Containment response (after depressurization of the reactor vessel is complete) the suppression pool is assumed to be the only heat sink in the Primary Containment. The effects of decay energy, stored energy, and energy from the metal-water reaction on the suppression pool temperature are considered.

<u>Case A</u>

This case assumes that both RHR loops are operating in the containment cooling mode. This includes two RHR heat exchangers, 4 RHR pumps, 4 RHR Service Water Booster pumps,[78] and 3 Service Water pumps. The RHR pumps draw suction from the suppression pool, pump the water through the RHR heat exchangers and return the cooled water to the Primary Containment through the containment spray headers. This forms a closed cooling loop with the suppression pool with about 95 percent of the flow being sprayed into the drywell and the remaining 5 percent being sprayed into the suppression pool. This containment cooling condition is arbitrarily assumed to start at 600 seconds after the accident. Prior to this time the RHR pumps are used to flood the core (LPCI mode).

The containment pressure and temperature response to this set of conditions is shown as Figure XIV-6-5. After the initial rapid depressurization by the HPCI and pipe break and subsequent further depressurization due to core spray and LPCI core flooding, energy addition due to core decay heat results in a gradual pressure and temperature rise in the Primary Containment. When the energy removal rate of the RHR exceeds the energy addition rate from the decay heat, the Primary Containment pressure and temperature decrease to their pre-accident values. Table XIV-6-6 summarizes the cooling equipment operation, the peak Primary Containment pressure following the initial blowdown peak, and the peak suppression pool temperature.

TABLE XIV-6-5

INPUT PARAMETERS USED FOR DBA LOCA PRIMARY CONTAINMENT ANALYSIS[47, 85]

Parameters	Units	Values Used in Analysis
Core Thermal Power (102% of rated)	MWt	2,429
Drywell Free Volume	ft^3	132,250
Wetwell Free Volume at Low Water Level (LWL) (above suppression pool)	ft^3	112,240
Suppression Pool Volume at LWL	ft^3	87,650
Initial Pool Temperature	°F	100
Supp. Chamber-to-Drywell Vacuum Breaker Opening Diff. Pressure Setpoint	psid	0.5
Service Water Temperature	°F	95

Case B

This case assumes that only one RHR subsystem is operating in the containment cooling mode. This includes one RHR heat exchanger, 2 RHR pumps, 2 RHR Service Water Booster pumps,[78] and 2 service water pumps. As in the previous case, the RHR containment cooling mode is assumed to be activated at 600 seconds after the accident. The containment pressure and temperature response to this set of conditions is shown as Figure XIV-6-6. A summary of this case is shown in Table XIV-6-6.

Case C

This case assumes that one RHR subsystem is operating in the containment cooling mode at partial pumping capacity. This includes one RHR heat exchanger, one RHR pump, two RHR Service Water Booster pumps,[78] and two Service Water pumps. This reduction in RHR flow results in a decrease in the heat removal capacity of the RHR heat exchanger, which in turn results in slightly higher Primary Containment temperature and pressure. It is assumed that this cooling condition is established at 600 seconds after the accident.

The Primary Containment response to this set of conditions is shown in Figure XIV-6-7. A summary of this case is shown in Table XIV-6-6.

Case D

This case is exactly the same as the preceding one except that the drywell spray is not operating. During this mode of operation RHR pumps draw suction from the suppression pool and discharge flow through the RHR heat exchangers where it is cooled and then returned directly to the pool. It is assumed that this cooling condition is established at 600 seconds after the accident.

The Primary Containment response to this set of conditions is shown in Figure XIV-6-8. A summary of this case is shown in Table XIV-6-6.

When comparing "spray" case with the "no spray" case, the suppression pool temperature response is virtually the same. This is because the same amount of energy is removed from the pool whether the exit flow from the RHR heat exchanger is returned to the pool or injected into the drywell as spray. However, the peak Primary Containment pressure is higher for the "no spray" case. This, however, is of no consequence because the pressure is still much less than the Primary Containment maximum allowable pressure of 62 psig. Figure XIV-6-10 illustrates the slight effect on Primary Containment leakage rate due to the higher pressure.

Case E

This case assumes that one RHR subsystem is operating in the containment cooling mode with only one RHR heat exchanger, one RHR pump, one RHR Service Water Booster pump, and one Service Water pump. This case represents the most degraded condition of heat removal while operating in the containment cooling mode. It is assumed that this condition is established at 600 seconds after the accident.

The Primary Containment response to this set of conditions is shown as Figure XIV-6-9. A summary of this case is shown in Table XIV-6-6.

Case E was reanalyzed to evaluate the impact of using 95°F Service Water temperature.[85]

TABLE XIV-6-6

LOSS-OF-COOLANT ACCIDENT
PRIMARY CONTAINMENT RESPONSE SUMMARY[47,85]

Long-Term Response (to 10^5 sec after Accident)

Case	RHR Subsystem	RHR Pumps per Subsystem	Service Water Booster Pumps per Subsystem	Containment Spray Flow per Subsystem, gpm	Core Spray Flow, gpm	Suppression Pool Cooling Flow, gpm	Peak Pool Temp., °F	Secondary (Long-Term) Peak Pressure, psig
A***	2	2	2	11,500	4,720	---	167	7.5
B***	1	2	2	11,500	4,720	---	187	10.8
C***	1	1	2	7,700	4,720	---	190	11.0
D***	1	1	2	---	4,720	7,700	189	14.3
E*, **	1	1	1	6,500	4,720	---	208.7	14.9
F**	1	1	1	---	4,720	7,700	208.2	22.7

NOTES:

* Case E has been reanalyzed with 6500 gpm containment spray flow to account for maximum potential nozzle plugging.[86]

** Cases E and F are the "worst case" scenarios for CNS, since they rely on the minimum amount of equipment that would be available assuming a loss of offsite power and the failure of one diesel generator.

*** Cases A-D have not been reanalyzed using the latest assumptions.

For short term responses, refer to USAR Section XIV-6.3.7.1

The drywell pressure response model has been checked with good results against both the Humboldt Bay and Bodega Bay Pressure Suppression tests for a wide range of break sizes.

Case F

This case assumes that one RHR subsystem is operating in the containment cooling mode with only one RHR heat exchanger, one RHR pump, one RHR Service Water Booster pump, and one Service Water pump. This case represents the most degraded condition of heat removal while operating in the suppression pool cooling mode. It is assumed that this condition is established at 600 seconds after the accident.

The Primary Containment response to this set of conditions is shown as Figure XIV-6-9A. A summary of this case is shown in Table XIV-6-6.

6.3.7.3 Metal Water Reaction Effects on the Primary Containment

If Zircaloy in the reactor core is heated to temperatures above about 2000°F in the presence of steam, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2800 BTU per pound of zirconium reacted. The energy produced is accommodated in the suppression chamber pool. The hydrogen formed, however, will result in an increased drywell pressure due simply to the added volume of gas to the fixed Primary Containment volume. Although very small quantities of hydrogen are produced during the accident, the Primary Containment has the inherent ability to accommodate a much larger amount as discussed below. This discussion covers the impact of hydrogen generation on containment pressures; combustibility issues and conformance to 10CFR50.44 requirements are described in USAR Chapter V.

The basic approach to evaluating the capability of a containment system with a given containment spray design is to assume that the energy and gas are liberated from the reactor vessel over some time period. The rate of energy release over the entire duration of the release is arbitrarily taken as uniform, since the capability curve serves as a capability index only, and is not based on any given set of accident conditions as an accident performance evaluation might be.

It is conservatively assumed that the suppression pool is the only body in the system which is capable of storing energy. The considerable amount of energy storage which would take place in the various structures of the containment is neglected. Hence, as energy is released from the core region, it is absorbed by the suppression pool. Energy is removed from the pool by heat exchangers which reject heat to the service water. Because the energy release is taken as uniform and the service-water temperature and exchanger flow rate are constant, the temperature response of the pool can be determined. It is assumed that the suppression chamber gases are at the suppression chamber water temperature.

The metal-water reaction during core heatup was calculated by the core heat-up model described in APED-5454.[74] The extent of the metal-water reaction thus calculated is less than 0.1 percent of all the zirconium in the core. As an index of the containment's ability to tolerate postulated metal-water reactions, the concept of "Containment Capability" is used. Since this capability depends on the time domain, the duration over which the metal-water reaction is postulated to occur is one of the parameters used.

Containment capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction during a specified duration without exceeding the maximum allowable pressure of the containment. To evaluate the containment capability, various percentages of metal-water reaction are assumed to take place over certain time periods. This analysis presents a method of measuring system capability without requiring prediction of the detailed events in a particular accident condition.

Since the percent metal-water reaction capability varies with the duration of the uniform energy and gas release, the percent metal-water reaction capability is plotted against the duration of release. This constitutes the containment capability curves as shown in Figure XIV-6-11. All points below the curves represent a given metal-water reaction and a given duration which will result in a Primary Containment peak pressure which is below the maximum allowable pressure. The calculations are made at the end of the energy release duration because the number of moles of gases in the system is then at a maximum, and the suppression pool temperature is higher at this time than at any other time during the energy release.

It should be noted that the curves are actually derived from separate calculations of two conditions; the "steaming" and the "non-steaming" situations.

The maximum amount of metal-water reaction which the Primary Containment can tolerate without sprays for a given duration is given by the condition where all of the non-condensible gases are stored in the suppression chamber. This condition assumes that "steaming" from the drywell to the suppression chamber results in washing all of the non-condensible gases into the suppression chamber. This is shown as the flat portion of the containment capability characteristic curve. Activation of containment sprays condense the drywell steam so that no steaming occurs, thus allowing non-condensibles to also be stored in the drywell. This is denoted by the rising (spray) curve. The intersection between the no spray curve and the spray curve represents the duration and metal-water reaction energy release which just raises all the spray water to the saturation temperature at the maximum allowable Primary Containment pressures.

For durations to the left of the intersection some steam is generated and all the gases are stored in the suppression chamber. For durations to the right of the intersection, the spray flow is subcooled as it exits from drywell by increasing amounts as the duration is increased.

The energy release rate to the Primary Containment is calculated as follows:

$$q_{IN} = \frac{Q_O + Q_{MW} + Q_S}{T_D}$$

where:

q_{IN} = Arbitrary energy release rate to the Primary Containment, Btu per sec

Q_O = Integral of decay power over selected duration of energy gas release, Btu

Q_{MW} = Total chemical energy released exothermically from selected metal-water reaction, Btu

Q_S = Initial internal sensible energy of core fuel and cladding, Btu

T_D = Selected duration of energy and gas release, seconds

The total chemical energy released from the metal-water reaction is proportional to the percent metal-water reaction. The initial internal sensible energy of the core is taken as the difference between the energy in the core after the blowdown and the energy in the core at a datum temperature of 250°F.

The temperature of the drywell gas is found by considering an energy balance on the spray flows through the drywell.

Based upon the drywell gas temperature, suppression chamber gas temperature, and the total number of moles in the system, as calculated above, the containment pressure is determined. The containment capability curves in Figure XIV-6-11 present the results of the parametric investigation.

6.3.8 Radiological Consequences

The radiological consequences of the LOCA are based on an assumed fuel failure occurring as a result of a guillotine break in the Reactor Recirculation System. This is conservative since the ECCS has been analyzed as meeting the requirements of 10CFR50.46 (see Section VI-5). The transport methodology is based on Regulatory Guide 1.183 and 10CFR50.67 "Accident Source Term." As a result of the accident, radionuclides are assumed to be released from the damaged fuel rods to the Primary Containment. Four pathways of radiological releases from Primary Containment are considered. The first pathway is direct leakage to Secondary Containment from Primary Containment which is released to the environment via the Standby Gas Treatment System to the Elevated Release Point. The second pathway is direct leakage from Primary Containment to Secondary Containment which is released to the environment directly from the Reactor Building during the brief period of time prior to the SGT system establishing a negative pressure condition in the Secondary Containment. The third pathway is leakage to Secondary Containment via Engineered Safety Features circulating reactor coolant outside Primary Containment. This leakage pathway also egresses via the Standby Gas Treatment System to the Elevated Release Point. The fourth pathway is the leakage past the closed Main Steam Isolation Valves (MSIVs) through the Turbine Building.

From these releases, doses are calculated for individuals offsite. The release of radionuclides to the environment can also result in a dose to the Control Room occupants due to intake of contaminated air via the Main Control Room Air Conditioning System. The release can also result in a dose to the Control Room occupants due to gamma shine from various sources including the Primary Containment, the Reactor Building, the passing cloud, a Core Spray line, and the CREFS filter.

The radiological consequences of the LOCA were assessed using the RADTRAD software code. This code was used to calculate the Control Room occupant, Exclusion Area Boundary, and Low Population Zone TEDE doses.[80] The additional gamma shine dose contribution to Control Room personnel was calculated using the Microshield software code.

6.3.8.1 Fission Products Released to Primary Containment

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the Reactor Coolant Pressure Boundary to the drywell:

a. The reactor has been operating at a power of 2429 MWt (equal to the Original Licensed Thermal Power of 2381 MWt plus 2% to account for power measurement uncertainties) for an extended period prior to the recirculation line break. This assumption results in equilibrium concentrations of fission products in the fuel. For radionuclides which have not reached equilibrium the core inventory at time of shutdown is used.

b. The source term used for this accident is based on the core inventory source term for GNF2 fuel and a 24 month fuel cycle. The core inventory source term was calculated using the isotope generation and depletion code ORIGEN2, which incorporates the BWR extended burnup library BWRUE.

c. Fission products from the damaged fuel are released into the Reactor Coolant System and then into the Primary Containment (i.e., drywell and wetwell). The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. For conservatism, the CNS LOCA analysis excludes this two-minute delay. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term released during each of these two phases. The inventory in each release phase is released at a constant rate over the duration of the phase, starting at the onset of the phase. The release to Primary Containment during the two phases includes 100% of the noble gas fission products and 30% of the iodine fission products in the core.

d. The chemical form of radioiodine released from the fuel to the drywell is set at 95% particulate (cesium iodide, CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gas, fission products are assumed to be in particulate form. The release mixes homogeneously with the drywell air space. No credit is assumed for mixing with the wetwell air space or for Suppression Pool scrubbing. The analysis includes decay and daughtering of radionuclides and natural deposition inside the Primary Containment.

e. For the purposes of assessing the consequences of leakage from the ECCS, it is assumed that 100% of the radioiodines released from the fuel are transported to the Suppression Pool as they are released. This source term assumption is conservative in that 100% of the radioiodines released from the fuel are assumed to be available for ECCS leakage, Primary Containment leakage, and MSIV leakage. In actuality, the radioiodines in the Primary Containment atmosphere would relocate to the Suppression Pool over time. Noble gases released from the fuel are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to become airborne on release from the ECCS, they are not included in the ECCS source term. The iodine released from the ECCS leakage is assumed to be in the form of 97% elemental and 3% organic iodine. Also, because the pool pH is kept greater than 7 for the 30-day accident duration by use of SLC injection during the first 8 hours, any iodine that enters the pool will not re-evolve into a more volatile form and potentially escape to the environment.

6.3.8.2 Fission Product Release to Secondary Containment

As discussed in Section XIV-6.3.8, fission product activity is released to the Secondary Containment via two paths. The first pathway is direct leakage to Secondary Containment from Primary Containment air volume.

The activity released is based on 100% mixing with the drywell air volume and a leakage rate of 0.635 %-vol/day for the first day and 0.3175%-vol/day for the remainder of the 30-day accident duration. The second pathway is via ECCS system and subsystem leakage, when in post-LOCA recirculation cooling. The activity released to the Suppression Pool is mixed homogeneously with the Suppression Pool water volume. The assumed ESF leakage rate is 45,000 cc/min during postulated LOCA conditions, which is equivalent to a leakage of approximately 3000 cc/min during a test of the ESF system. Of this leakage, with the Suppression Pool water circulating outside Primary Containment not exceeding 212°F, a flash factor of 10% is assumed. Accordingly, 10% of the iodine in the ESF leakage becomes airborne.

6.3.8.3 Fission Product Release to Environs

As stated in Section 6.3.8, the fission product activity is released to the environs via four pathways. Two pathways from Primary Containment leakage and ECCS leakage are Secondary Containment atmosphere releases to the Elevated Release Point via the Standby Gas Treatment System. The third pathway is a Secondary Containment atmosphere release from the Reactor Building prior to the SGT system establishing a negative pressure in the Secondary Containment. The fourth pathway consists of Main Steam Isolation Valve leakage to the Main Condenser, and subsequent leakage to the Turbine Building.

6.3.8.3.1 Secondary Containment Release to the Environs

The fission product activity released to the environs from Secondary Containment is dependent upon the fission product inventory airborne in the Secondary Containment, the volumetric flow from the Secondary Containment and the efficiency of the various components of the Standby Gas Treatment System.

The following assumptions and initial conditions were used in calculating the fission products released to the environs from Secondary Containment.

a. Low reactor water level (Level 2) or high drywell pressure will isolate the Reactor Building HVAC system and initiate SGTS. A brief period of time may exist prior to the SGT system establishing a negative pressure condition in the Secondary Containment. A CNS pressurization analysis indicates that the Reactor Building may have a positive pressure for 210 seconds in the worst case of a failure of the intake damper to close. The dose analysis conservatively assumes that the drywell releases directly to the environment at the Primary Containment leakage rate of 0.635%-vol/day for the first 5 minutes as a ground release from the Reactor Building. The fission products released to Secondary Containment from Primary Containment or ECCS leakage are immediately available for release via SGTS with no dilution or holdup in Secondary Containment. Accordingly, a conservatively high value of 1 Secondary Containment air volume per second is assumed in the transport modeling. All fission products released from the Secondary Containment egress via SGTS and the Elevated Release Point.

b. The Standby Gas Treatment System (SGTS), in addition to iodine and particulate filters, also contains a demister for the removal of entrained water droplets and electric heaters for heating the incoming air, upstream of the particulate and iodine filters to a ΔT of 10°F above the temperature of the mixture entering the Standby Gas Treatment System. A ΔT of 10°F will reduce the relative humidity of the incoming mixture to approximately 70 percent.

c. Both SGT trains start upon a Group 6 PCIS signal (assumed from time = 0 seconds to 1 hour), with heater power to one train assumed to have failed. At the 1-hour point, the SGT train with the failed heater is manually secured. Table XIV-6-7 identifies the SGT flow and iodine removal efficiencies that are assumed. This includes a –1% correction to account for SGT filter bypass.

6.3.8.3.2 MSIV Leakage Pathway Release to the Environs

The MSIV leakage was quantified by assuming that the MSIV seats leaked at a total leakage of 300 scfh at accident pressure (equivalent to the aggregate Technical Specification limit of 212 scfh when tested at reduced pressure). The MSIV leakage is reduced 50%, from 300 scfh to 150 scfh, at 24 hours due to reduced drywell pressure. All of the MSIV seat leakage is directed to the Turbine Building via the Main Condenser. This is conservative since any iodines leaked to the Reactor Building (e.g., MSIV stem leakage in the Steam Tunnel) would be processed by SGTS prior to release. Holdup and deposition were credited in one shell of the condenser, but holdup and deposition in the Main Steam lines were conservatively neglected. The main condenser effective filter efficiency (radionuclide removal efficiency) is 94.91% for the initial 24-hour post-LOCA period and is 97.39% for the remainder of the 30-day accident duration. The activity in the Main Condenser is assumed to egress to the Turbine Building at a rate equal to the assumed MSIV leakage rate. This activity instantaneously exits the Turbine Building as a ground release without any credit for holdup or mixing in the Turbine Building.

6.3.8.4 Radiological Effects

6.3.8.4.1 Offsite Consequence Results

The offsite consequences in terms of the radiological doses resulting from the activity released to the environment during a Loss-of-Coolant Accident have been determined based on the calculated Reactor Building, Turbine Building and Elevated Release Point atmospheric dispersion factors (X/Q) for the Exclusion Area Boundary and Low Population Zone shown on Table XIV-6-8. These X/Q values were generated using the methodology presented in Regulatory Guides 1.3 and 1.25. Building wake and fumigation considerations are factored into the atmospheric dispersion factor determinations. The effect of fumigation was included between 1.3 and 1.8 hours in order to correspond to the first half hour of the worst 2-hour period for the Exclusion Area Boundary dose, which occurs from 1.3 to 3.3 hours.

Two dose periods were evaluated, the worst 2-hour dose period at the Exclusion Area Boundary and a 30-day dose period for the Low Population Zone. The Exclusion Area Boundary and Low Population Zone radiological consequences for the Loss-of-Coolant Accident have been assessed using the RADTRAD software code. The code was used to calculate the TEDE doses at these receptor locations. The RADTRAD assessment results are shown on Table XIV-6-10, and are within the 10CFR50.67 guideline values of 25 rem TEDE.

It is concluded that this accident will not result in any radiological doses which endanger the health and safety of the public.

6.3.8.4.2 Onsite (Control Room Personnel) Consequence Results

The Control Room occupant radiological doses from a Loss-of-Coolant Accident were assessed using the RADTRAD software code. The doses have been determined based on the calculated Turbine Building, Reactor Building, and Elevated Release point atmospheric dispersion factors (X/Q) for the Control Room Air Conditioning System ventilation intake, with consideration of the effects of the Control Room Emergency Filter System (CREFS). The Turbine Building, Reactor Building, and Elevated Release Point X/Q values for the Control Room dose calculations were generated using the ARCON96 software code and site specific meteorology for the years 1994-1998. The Turbine Building release path was modeled as a diffuse ground level release from the wall closest to the Control Room air intake. The Reactor Building release path was modeled as a ground level release emanating from the Reactor Building ventilation exhaust point. The Elevated Release Point was modeled as a single release point using the elevated release mode of ARCON96. The ARCON96 generated elevated release X/Q value for the Control Room, during the 30-minute period from 1.3 to 1.8 hours following the accident, was replaced by the ground-level fumigation X/Q value used in response to NUREG-0737 Item III.D.3.4. Occupancy factors for Control Room occupants after one day are applied to allow for actual time that the occupants are assumed to be present in the Control Room. The results are shown on Table XIV-6-9.

Within 11 seconds of the Group 6 PCIS isolation signal, CREFS initiates, which isolates the normal unfiltered Control Room Air Conditioning System supply. An isolation time of 1 minute was conservatively assumed in the analysis. Prior to isolation, the total air intake rate is 3635 cfm (which includes normal air intake flow, infiltration leakage, and inleakage through opening and closing of doors). No credit is taken for filtration in the first 60 seconds. After isolation, the total air intake rate is 1210 cfm, which includes CREFS intake flow (810 cfm), and unfiltered inleakage (400 cfm unfiltered inleakge is assumed even when the isolated Control Room is at positive pressure and includes an unfiltered ingress/egress inleakage of 10 cfm). CREFS filter efficiency is specified as 99 percent for particulate and 90 percent for elemental and organic material. These are further reduced by 1% to account for bypass. The resulting radionuclide concentration within the Control Room Envelope is diluted by the air space volume (141,860 ft^3 for the entire Control Room Envelope).

The RADTRAD software code was used to calculate the TEDE doses at the Control Room receptor location. The results of the RADTRAD assessment are presented on Table XIV-6-10. The post-LOCA 30-day gamma shine dose to Control Room personnel is also included in Table XIV-6-10. The shine contributors to Control Room dose were from the most significant sources: outside cloud, the Reactor Building, a Core Spray line, the CREFS filter, and the Primary Containment. The total EAB, LPZ, and Control Room doses are within the dose limits of 10CFR50.67.

6.4 Fuel Handling Accident

Accidents that result in the release of radioactive materials directly to the Secondary Containment can occur when the drywell is open. A survey of the various conditions that could exist when the drywell is open reveals that the greatest potential for the release of radioactive material occurs when the drywell head and reactor vessel head have been removed. In this case, radioactive material released as a result of fuel failure is available for transport directly to the Secondary Containment.

TABLE XIV-6-7

SGT SYSTEM FLOWS AND IODINE REMOVAL EFFICIENCIES

Time = 0 seconds – 1 hour (Both SGT trains running)

	Active Heater	Failed Heater
SGT Flow (cfm)	1492	1492
Elemental Iodine Efficiency	94%	89%
Particulate Iodine Efficiency	98%	98%
Organic Iodine Efficiency	94%	29%

Time = 1 hour – 30 days (One SGT train running, one train secured)

	Single Train	Single Train (cross-tie flow)
SGT Flow (cfm)	1492	288
Elemental Iodine Efficiency	94%	89%
Particulate Iodine Efficiency	98%	98%
Organic Iodine Efficiency	94%	29%

TABLE XIV-6-8

X/Q VALUES FOR THE EXCLUSION AREA BOUNDARY
AND LOW POPULATION ZONE

X/Q Values for the Exclusion Area Boundary

Time Period	X/Q Value (sec/m^3)	Comments
0 to 10 hours	5.2E-4	Turbine Building Ground Level Release
0 to 1.3 hours	1.6E-5	SGT System Elevated Release Point
1.3 to 1.8 hours	1.2E-4	SGT System Elevated Release Point with fumigation
1.8 to 10 hours	1.6E-5	SGT System Elevated Release Point

X/Q Values for the Low Population Zone

Time Period	X/Q Value (sec/m^3)	Comments
0 to 8 hours	2.9E-4	Turbine Building Ground Level release
8 to 24 hours	7.3E-5	Turbine Building Ground Level release
1 to 4 days	2.5E-5	Turbine Building Ground Level release
4 to 30 days	5.2E-6	Turbine Building Ground Level release
0 to 1.3 hours	4.0E-5	SGT System Elevated Release Point
1.3 to 1.8 hours	1.4E-4	SGT System Elevated Release Point with fumigation
1.8 to 8 hours	4.0E-5	SGT System Elevated Release Point
8 to 24 hours	1.6E-5	SGT System Elevated Release Point
1 to 4 days	5.8E-6	SGT System Elevated Release Point
4 to 30 days	1.7E-6	SGT System Elevated Release Point

TABLE XIV-6-9

X/Q VALUES FOR THE CONTROL ROOM INTAKE

Time Period	X/Q Value (sec/m^3)	Occupancy Factor	Comments
0 to 2 hours	8.64E-4	1	Turbine Building vent ground level release
2 to 8 hours	4.66E-4	1	Turbine Building vent ground level release
8 hours to 24 hours	2.32E-4	1	Turbine Building vent ground level release
1 to 4 days	1.53E-4	0.6	Turbine Building vent ground level release
4 to 30 days	1.25E-4	0.4	Turbine Building vent ground level release
0 to 5 minutes	4.15E-3	1	Reactor Building ground level release
0 to 1.3 hours	1.00E-10	1	SGT System Elevated Release Point
1.3 to 1.8 hours	3.03E-4	1	SGT System ground level release with fumigation
1.8 to 2 hours	1.00E-10	1	SGT System Elevated Release Point
2 to 8 hours	8.58E-10	1	SGT System Elevated Release Point
8 to 24 hours	1.41E-8	1	SGT System Elevated Release Point
1 to 4 days	5.62E-9	0.6	SGT System Elevated Release Point
4 to 30 days	5.69E-9	0.4	SGT System Elevated Release Point

TABLE XIV-6-10

LOSS-OF-COOLANT ACCIDENT EXCLUSION AREA BOUNDARY,
LOW POPULATION ZONE, AND CONTROL ROOM
RADIOLOGICAL DOSE CONSEQUENCES
TEDE (rem)

	EAB	LPZ	Control Room
Primary Containment	0.469	1.593	0.392
Main Steam Pathway	0.390	2.407	2.497
ESF Leakage	0.17	1.726	0.102
Shine	N/A	N/A	0.334
Total	1.029	5.726	3.325
Dose acceptance criteria (10CFR50.67)	25	25	5

Various mechanisms for fuel failure under this condition have been investigated. With the current fuel design the refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. Loss of refueling cavity inventory due to a seal failure[71] was evaluated that, if refueling cavity seal failed, the Core Spray and/or the Reactor Heat Removal systems would allow ample time to place fuel in a safe location per CNS Emergency Operating Procedures. In addition, the Reactor Protection System can initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products directly to the Secondary Containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

6.4.1 Identification of Causes

This event occurs under non-operating conditions for the fuel. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel bundle being dropped on the core while in the cold condition.

6.4.2 Frequency Classification

The Fuel Handling Accident is classified as a design basis accident (limiting fault).

6.4.3 Starting Conditions and Assumptions

The assumptions and analyses applicable to the Fuel Handling Accident are described below.

(1) The fuel assembly is dropped from 32.95 feet (the maximum height allowed by the fuel handling equipment).

(2) The entire amount of potential energy, including the energy of the entire assemblage falling on its side from a vertical position (referenced to the top of the reactor core) is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core. Of the possible ways that a fuel assembly could be dropped, the most potential energy would be involved if the grapple cable breaks, allowing the grapple head and three sections of the telescoping mast to remain attached to the falling assembly.

(3) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

(4) All fuel rods, including tie rods, were assumed to fail by 1 percent plastic strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

(5) Because the event occurs under non-operating conditions, fuel densification considerations do not enter into or affect the accident results.

(6) The RPV water level is at least 21 feet above the top of RPV flange. This assures that there will be a water height over Fuel Zone Zero and the top of the dropped bundle that exceeds the 23 feet assumed by Regulatory Guide 1.183 for decontamination factor considerations.

(7) At least 24 hours has elapsed since shutdown. Secondary Containment Integrity is not required during the movement of fuel that has had at least a 24 hour decay time.

(8) A loss of offsite or onsite power is not assumed for this accident.

(9) For fuel movements that have decayed less than 7 days, either a Reactor Building exhaust fan or an SGT System fan is running. Otherwise, the Control Room Emergency Filter System is manually initiated prior to the start of the fuel handling evolution.

6.4.4 Sequence of Events and Systems Operation

The most severe Fuel Handling Accident from a radiological release viewpoint is the drop of a channeled exposed fuel bundle onto other fuel in the reactor vessel. In the hypothesized accident, the fuel grapple cable breaks, allowing the fuel bundle, grapple head, and three sections of the telescoping mast to remain attached to the falling assembly. On impact, rods in both the dropped and struck bundles fail, releasing radioactive gases to the water in the reactor vessel. From there the gases are released to the refueling floor of the Reactor Building. With the Primary Containment and the reactor vessel open, the reactor cavity water pool and the Secondary Containment (Reactor Building) serve as the major barriers to the release of radioactive materials.

6.4.5 Core and System Performance

The severity of the Fuel Handling Accident is directly based on the number of fuel rods damaged and the radial peaking factor for each fuel type. This analysis addresses the worst case fuel drop for the fuel types used in the CNS core. The limiting case is a Global Nuclear Fuels (GNF) 10x10 fuel bundle dropped over GNF 10x10 bundles, which results in 150 fuel rods that are assumed to be damaged. The methodology used to determine the number of damaged fuel rods is contained in GNF NEDC-33270P, "GNF2 Advantage Generic Compliance With NEDE-24011-P-A (GESTAR II)."[96]

6.4.6 Barrier Performance (Secondary Containment Response)

Secondary Containment integrity is not required for this accident. However, consistent with NUMARC 93-01, Revision 3, Section 11.3.6.5, NPPD implements a Secondary Containment breach control strategy during the movement of irradiated fuel inside Secondary Containment (see Section V-3.4). This ensures that Secondary Containment is structurally intact during the accident. With Reactor Building ventilation exhaust flow established prior to the start of the event, high radiation levels in the Reactor Building exhaust plenum will initiate the Control Room Emergency Filter System.

6.4.7 Radiological Consequences

The radiological consequences of a Fuel Handling Accident are based on a fuel failure due to the drop of a fuel assembly onto the core, in conjunction with a conservative transport methodology based on Regulatory Guide 1.183 and 10CFR50.67 “Accident Source Term.” As a result of the accident, radionuclides are released from the damaged fuel rods to the water pool above the core. Subsequently, the radionuclides are released to the refueling floor and then to the environment via the Reactor Building HVAC System. From this release, doses are calculated for individuals offsite. The release of radionuclides to the environment can result in a dose to the Control Room occupants due to intake of contaminated air via the Main Control Room Air Conditioning System.

The radiological consequences of the Fuel Handling Accident were assessed using the RADTRAD 3.03 software code. This code was used to calculate the Control Room occupant, Exclusion Area Boundary and Low Population Zone TEDE doses.[92]

6.4.7.1 Fission Product Release From Fuel

The radionuclide source term during a Fuel Handling Accident is based on the fraction of the reactor core that is damaged as a result of the accident. As discussed in USAR Section XIV-6.4.5, 150 GNF fuel rods are assumed to be damaged. Each fuel bundle has an equivalent of 85.6 full length fuel rods. With 548 fuel bundles in the core the fraction of the core failed is calculated to be 150/(548 x 85.6) or 0.0031977.

The core is assumed to have been operated at rated thermal power plus 2%, to account for uncertainties in power measurement for a sufficiently extended period (approximately 3 years) such that fission product equilibrium is reached. For radionuclides which have not reached equilibrium the core inventory at time of shutdown is used. The source term used for this accident is based on GNF NEDC-33270P, "GNF2 Advantage Generic Compliance With NEDE-24011-P-A (GESTAR II)."[96] This core inventory source term was calculated using the isotope generation and depletion code ORIGEN2, which incorporates the BWR extended burnup library BWRUE. Additionally, the calculated source term is based on a 24-hour decay period from when the fuel was last irradiated until the Fuel Handling Accident initiating event. A 7 day decay case was performed to demonstrate that CREFS is not needed for Control Room occupant dose mitigation.

A radial peaking factor of 1.95 is applied to the radionuclide inventory calculation to reflect a peaking factor that bounds core designs with GNF 10x10 fuel. The combination of the 1.00398 power uncertainity factor applied to the licensed thermal power of 2419 MW, and a radial peaking factor of 1.95 results in a very conservative source term.

The fuel gap fraction of radionuclides released from the damaged rods is 8% of the I-131, 10% of the KR-85, 5% of the other iodines and noble gases, and 0.12% alkali metals in the rods at the time of the accident per the assumptions of Regulatory Guide 1.183. The chemical forms of radioiodine released from the fuel to the pool are 95% aerosol (cesium iodine), 4.85% elemental and 0.15% organic. The particulate iodine is entirely retained within the reactor cavity pool.

6.4.7.2 Fission Product Release to Secondary Containment

The source term release to the Refueling floor is shown on Table XIV-6-11. Immediately after the fuel bundle drop, radionuclides are assumed to be released from the reactor cavity pool to the refueling floor in sufficient quantities to initiate CREFS due to high radiation (if decay time is less than 7 days). The following assumptions and initial conditions are used to calculate the fission product release to the Secondary Containment.

a. The fission product activity released to the Secondary Containment will be in proportion to the removal efficiency of the water in the refueling pool. The refueling cavity water height is at least 23 feet

TABLE XIV-6-11
FUEL HANDLING ACCIDENT
SECONDARY CONTAINMENT AIRBORNE FISSION PRODUCT INVENTORY 24 HOURS AFTER SHUTDOWN

Isotope	RADTRAD Ci/MWt(1)	MWt	Power Uncertainty Factor	Total Core Inventory (Ci)	Radioactive Half Life (sec)(2)	Time After Shutdown (sec)	Core Inventory at Time After Shutdown (Ci)(3)	Fraction of Core Inventory Released	Water Pool Decon Factor	Radial Peaking Factor	Fraction of Core Failed	FHA Source Term (Ci)
Br-82	1.870E+02	2419	1.00398	4.542E+05	1.271E+05	86400	2.835E+05	0.05	0.005	1.95	0.0031977	4.420E-01
Br-83	3.240E+03	2419	1.00398	7.869E+06	8.604E+03	86400	7.476E+03	0.05	0.005	1.95	0.0031977	1.165E-02
B4-84	5.560E+03	2419	1.00398	1.350E+07	1.908E+03	86400	3.175E-07	0.05	0.005	1.95	0.0031977	4.950E-13
Kr-83m	3.250E+03	2419	1.00398	7.893E+06	6.588E+03	86400	8.916E+02	0.05	1	1.95	0.0031977	2.780E-01
Kr-85	4.260E+02	2419	1.00398	1.035E+06	3.383E+08	86400	1.034E+06	0.10	1	1.95	0.0031977	6.450E+02
Kr-85m	6.750E+03	2419	1.00398	1.639E+07	1.613E+04	86400	4.003E+05	0.05	1	1.95	0.0031977	1.248E+02
Kr-87	1.280E+04	2419	1.00398	3.109E+07	4.578E+03	86400	6.493E+01	0.05	1	1.95	0.0031977	2.025E-02
Kr-88	1.810E+04	2419	1.00398	4.396E+07	1.022E+04	86400	1.258E+05	0.05	1	1.95	0.0031977	3.922E+01
I-128	4.330E+02	2419	1.00398	1.052E+06	1.499E+03	86400	4.728E-12	0.05	0.005	1.95	0.0031977	7.370E-18
I-130	1.100E+03	2419	1.00398	2.671E+06	4.450E+04	86400	6.956E+05	0.05	0.005	1.95	0.0031977	1.084E+00
I-131	2.720E+04	2419	1.00398	6.606E+07	6.947E+05	86400	6.060E+07	0.08	0.005	1.95	0.0031977	1.512E+02
I-132	3.960E+04	2419	1.00398	9.617E+07	8.280E+03	86400	6.959E+04	0.05	0.005	1.95	0.0031977	1.085E-01
I-133	5.480E+04	2419	1.00398	1.331E+08	7.488E+04	86400	5.982E+07	0.05	0.005	1.95	0.0031977	9.326E+01
I-134	6.040E+04	2419	1.00398	1.467E+08	3.156E+03	86400	8.453E-01	0.05	0.005	1.95	0.0031977	1.318E-06
I-135	5.160E+04	2419	1.00398	1.253E+08	2.380E+04	86400	1.012E+07	0.05	0.005	1.95	0.0031977	1.578E+01
Te-129(4)	8.840E+03	2419	1.00398	2.147E+07	4.176E+03	86400	1.273E+01	0.00	0.00	1.95	0.0031977	0.000E+00
Te-131(4)	2.420E+04	2419	1.00398	5.877E+07	1.500E+03	86400	2.714E-10	0.00	0.00	1.95	0.0031977	0.000E+00
Te-131m(4)	3.970E+03	2419	1.00398	9.642E+06	1.080E+05	86400	5.538E+06	0.00	0.00	1.95	0.0031977	0.000E+00
Te-132(4)	3.860E+04	2419	1.00398	9.375E+07	2.815E+05	86400	7.578E+07	0.00	0.00	1.95	0.0031977	0.000E+00
Te-133(4)	3.240E+04	2419	1.00398	7.869E+07	7.470E+02	86400	1.217E-27	0.00	0.00	1.95	0.0031977	0.000E+00
Te-133m(4)	1.970E+04	2419	1.00398	4.784E+07	3.324E+03	86400	7.193E-01	0.00	0.00	1.95	0.0031977	0.000E+00
Te-134(4)	4.480E+04	2419	1.00398	1.088E+08	2.508E+03	86400	4.660E-03	0.00	0.00	1.95	0.0031977	0.000E+00
Xe-129m	2.230E-01	2419	1.00398	5.416E+02	6.912E+05	86400	4.966E+02	0.05	1	1.95	0.0031977	1.548E-01
Xe-131m	3.040E+02	2419	1.00398	7.383E+05	1.028E+06	86400	6.965E+05	0.05	1	1.95	0.0031977	2.172E+02
Xe-133	5.450E+04	2419	1.00398	1.324E+08	4.532E+05	86400	1.160E+08	0.05	1	1.95	0.0031977	3.616E+04
Xe-133m	1.730E+03	2419	1.00398	4.202E+06	1.890E+05	86400	3.061E+06	0.05	1	1.95	0.0031977	9.543E+02
Xe-135	2.040E+04	2419	1.00398	4.954E+07	3.272E+04	86400	7.950E+06	0.05	1	1.95	0.0031977	2.479E+03
Xe-135m	1.100E+04	2419	1.00398	2.671E+07	9.174E+02	86400	1.208E-21	0.05	1	1.95	0.0031977	3.766E-25
Xe-138	4.490E+04	2419	1.00398	1.090E+08	8.502E+02	86400	2.835E-23	0.05	1	1.95	0.0031977	8.838E-27

(1) The radionuclide release to the refueling area used in RADTRAD 3.03 is based on the ORIGEN2 generated source term, as compiled in this table.
(2) From RADTRAD nuclide inventory file (NIF) Attachment B.
(3) Calculated from standard equation $N = N_i \times e^{-(.693/\text{Half Life}) \times \text{Decay Time}}$
(4) Tellurium Metals were included in the nuclide inventory file (NIF) based on their daughtering contribution to iodine as analyzed by RADTRAD.

above the top of the damaged fuel bundles. Therefore, the water decontamination factors provided in Regulatory Guide 1.183 are used.

b. The effective air volume of the refuel floor is 7.95×10^5 ft^3.

6.4.7.3 Fission Product Release to Environs

The following assumptions and initial conditions are used to calculate the fission product release to the environs.

a. High radiation levels in the reactor building exhaust plenum will start the Control Room Emergency Filter System, if less than a 7 day decay time has elapsed.

b. In accordance with RG 1.183, the radioactive material released to the refueling floor is released over a 2-hour period. Since Secondary Containment is not assumed to be functioning, the discharge is a ground level, unfiltered release from the ventilation exhaust plenum to the discharge point on the Reactor Building roof. This release point was determined to provide the most limiting dose consequences over other Reactor Building hatches, doors, and airlocks. Using the relationship for dilution with 100% makeup for an enclosed space, the resulting flow rate is 4.576×10^4 cubic feet/minute.

The chemical/physical form of iodine released to the refueling floor in RADTRAD 3.03 is apportioned among the three iodine species:

1. 0.57 for elemental iodine

2. 0.43 for organic iodine

As identified in Regulatory Guide 1.183, this analysis uses an effective pool decontamination factor of 200 (which encompasses a decontamination factor for elemental and organic species of 500 and 1, respectively).

6.4.7.4 Radiological Effects

6.4.7.4.1 Offsite Consequence Results

The offsite consequences in terms of radiological doses resulting from the activity released to the environment during a Fuel Handling Accident have been determined based on the calculated Reactor Building atmospheric dispersion factors (X/Q) for the Exclusion Area Boundary and Low Population Zone shown on Table X-6-14. The X/Q values were generated using the methodology presented in Regulatory Guides 1.3 and 1.25. Building wake effect is factored into the atmospheric dispersion factor determinations. Two dose periods were evaluated, the worst case 2-hour dose period at the Exclusion Area Boundary and a 30-day dose period for the Low Population Zone. The Exclusion Area Boundary and Low Population Zone radiological consequences of the Fuel Handling Accident have been assessed using the RADTRAD 3.03 software code. The code was used to calculate the TEDE at these receptor locations. The RADTRAD 3.03 assessment results are shown on Table XIV-6-16 and are well within the 10CFR50.67 dose limits.

6.4.7.4.2 Onsite (Control Room Occupant) Consequence Results

The Control Room occupant radiological doses from a Fuel Handling Accident were assessed using the RADTRAD 3.03 software code. The doses have been determined based on the calculated Reactor Building atmospheric dispersion factors (X/Q) for the Control Room Air Conditioning System ventilation intake, with consideration of the effects of the Control Room Emergency Filter System (CREFS) when damaged fuel has had less than a 7 day decay period. The Reactor Building X/Q values for the Control Room dose calculations were generated using the ARCON96 software code and site specific meteorology for the years 1994-1998. The Reactor Building exhaust vent was modeled as a single release point using the ground level release mode of ARCON96. Occupancy factors for Control Room occupants after 1 day are applied to the X/Q values to allow for actual time that the occupants are assumed to be present in the Control Room. The results are shown on Table XIV-6-15.

Within 60 seconds of the Group 6 PCIS isolation signal, CREFS initiates, which isolates the normal unfiltered Control Room Air Conditioning System supply. Prior to isolation, the total air intake rate is 3635 cfm (which includes normal air intake flow, infiltration leakage, and inleakage through opening and closing of doors). No credit is taken for filtration in the first 60 seconds. After isolation, the total air intake rate is 1210 cfm, which includes CREFS intake flow, ingress/egress inleakage, and unfiltered inleakage (400 cfm unfiltered inleakage is assumed even when the isolated Control Room is at positive pressure). CREFS filter efficiency is specified as 90 percent for all iodine species. This is reduced by 1% to account for bypass. The resulting radionuclide concentration within the Control Room Envelope is diluted by the air space volume. Assuming 20% of the volume of the Control Room Envelope (including the Control Room proper) includes walls floors, and equipment, the net volume is 64,640 ft^3 for the Control Room proper, and 141,860 ft^3 for the entire Control Room Envelope.

The RADTRAD 3.03 software code was used to calculate the TEDE doses at the Control Room receptor location. The Control Room occupant dose also includes gamma shine from both external cloud shine to the Control Room and CREFS filter shine. The results of the dose assessment are presented on Table XIV-6-16. The results are within the dose limits of 10CFR50.67.

6.5 Main Steam Line Break Accident

An evaluation has been performed to substantiate that the design of CNS is adequate to withstand the effects of a postulated rupture in any high energy fluid system outside the Primary Containment.

The following analysis presents the design basis accident analysis for radiological consequences and for compliance with 10CFR50.46 and 10CFR50 Appendix K. The design basis accident for this class of pipe breaks outside the Primary Containment is a complete severance of one main steam line outside the Secondary Containment. Figure XIV-6-12 shows the break location.

6.5.1 Identification of Causes

The main steam lines are designed to meet applicable industry codes and seismic design requirements. Therefore, there are no identifiable events which would result in a Main Steam Line Break Accident. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is assumed without the cause being identified.

TABLE XIV-6-14

X/Q VALUES FOR THE EXCLUSION AREA BOUNDARY
AND LOW POPULATION ZONE

X/Q Value for the Exclusion Area Boundary

Time Period	X/Q Value (sec/m^3)	Comments
0 to 2 hours	5.2E-4	Reactor Building Vent (Ground level release)

X/Q Values for the Low Population Zone

Time Period	X/Q Value (sec/m^3)	Comments
0 to 0.5 hours	2.9E-4	Reactor Building Vent (Ground level release)
0.5 – 8 hours	2.9E-4	Reactor Building Vent (Ground level release)
8 to 24 hours	7.3E-5	Reactor Building Vent (Ground level release)
1 to 4 days	2.5E-5	Reactor Building Vent (Ground level release)
4 to 30 days	5.2E-6	Reactor Building Vent (Ground level release)

TABLE XIV-6-15

X/Q VALUES FOR THE CONTROL ROOM INTAKE

Time Period	X/Q Value (sec/m^3)	Occupancy Factor	X/Q Value (sec/m^3) Adjusted for Occupancy	Comments
0 to 2 hours	4.15E-03	1	4.15E-03	Reactor Building Vent (Ground level release)
2 to 8 hours	3.24E-03	1	3.24E-03	Reactor Building Vent (Ground level release)
8 to 24 hours	1.32E-03	1	1.32E-03	Reactor Building Vent (Ground level release)
1 to 4 days	9.01E-04	0.6	5.41E-04	Reactor Building Vent (Ground level release)
4 to 30 days	7.22E-04	0.4	2.89E-04	Reactor Building Vent (Ground level release)

TABLE XIV-6-16

FUEL HANDLING ACCIDENT EXCLUSION AREA BOUNDARY,
LOW POPULATION ZONE, AND CONTROL ROOM
RADIOLOGICAL DOSE CONSEQUENCES

Case 1 - Damaged fuel decayed 24 hours to 7 days (no Secondary Containment)

Dose Location	Accumulated Dose (rem TEDE)	Accident Dose Criteria (rem TEDE)
Control Room	4.568*	5.0
Exclusion Area Boundary	1.45	6.3
Low Population Zone	0.809	6.3

Case 2 - Damaged fuel decayed 7 days or longer (no Secondary Containment or CREFS)

Dose Location	Accumulated Dose (rem TEDE)	Accident Dose Criteria (rem TEDE)
Control Room	4.393	5.0
Exclusion Area Boundary	0.622	6.3
Low Population Zone	0.347	6.3

*Includes 114 mrem due to gamma shine from external sources.

6.5.2 Frequency Classification

The Main Steam Line Break Accident is classified as a design basis accident (limiting fault).

6.5.3 Starting Conditions and Assumptions

6.5.3.1 Radiological Consequences Starting Conditions and Assumptions

The following assumptions are used to evaluate response of nuclear system parameters to the steam line break accident outside the Secondary Containment relative to the radiological consequences:

a. The reactor is operating at design power.

b. Reactor vessel water level is normal for initial power level assumed at the time the break occurs.

c. Reactor Coolant Pressure Boundary pressure, including reactor steam dome pressure, is normal for the initial power level.

d. The steam line is assumed to be instantly severed by a circumferential break. The break is physically arranged so that the coolant discharge through the break is unobstructed. These assumptions result in the most severe depressurization rate of the Reactor Coolant Pressure Boundary.

e. The Main Steam Line Isolation Valves (MSIVs) are assumed to be closed 10.5 seconds after the break. This assumption is based on a 0.5 second time required to develop the automatic isolation signal and a 10 second closure stroke time for the valves. The 0.5 second response time is conservative since Main Steam line flow restrictor differential pressure and main steam line low pressure generate MSIV closure signals, and have an actual response time of approximately 200 milliseconds after the break occurs. The 10-second closure time is longer than the maximum time of 5 seconds required by Technical Specifications. Figure XIV-6-13a describes the steam line break flow rate profile for this closure time. Faster MSIV closure could reduce the mass loss until finally some other process line break would become controlling. However, the resulting radiological dose for this break would be less than the main steam line break with a 10-second valve closure. Thus the postulated main steam line break outside the Primary Containment with a 10-second isolation valve closure results in maximum calculated radiological dose and is therefore the design basis accident.

f. In calculating the rate of water level rise inside the vessel, it is assumed that the steam bubbles formed during depressurization rise at an average velocity of about 1 foot per second relative to the liquid. This assumption is predicted by analysis[14][15] and confirmed experimentally.[16]

g. After the level of the mixture inside the reactor vessel rises to the top of the steam dryers, the quality of the two-phase mixture discharged through the break is assumed constant at its minimum value. This assumption maximizes the total mass of coolant discharged through the break because most of the mixture flow will actually be at a higher quality.

h. Feedwater flow is assumed to decrease linearly to zero over the first four seconds.

i. A loss of offsite AC power (LOOP) is assumed to occur simultaneously with the break. This results in the immediate loss of power to the reactor recirculation pumps. Recirculation flow is assumed to coast down according to momentum computations for the recirculation system. The LOOP is not a consequence of the steam line break, but is an additionally imposed assumption.

j. Reactor Recirculation system pump head is assumed to be zero when the coolant at the pump suction reaches 1 percent quality. This assumption accounts for the effects of cavitation on reactor recirculation pump capacity as the pumps coast down.

6.5.3.2 10CFR50.46 and Appendix K Starting Conditions and Assumptions

Table XIV-6-17 provides the initial conditions used for evaluating the postulated Main Steam Line Break Accident for 10CFR50.46 and 10CFR50 Appendix K compliance.

6.5.4 Sequence of Events and Systems Operation

6.5.4.1 Radiological Consequences Sequence of Events and Systems Operations

The sequence of events following the postulated main steam line break is as follows for the radiological consequence analysis:

The steam flow through both ends of the break increases to the value limited by critical flow considerations. The flow from the upstream side of the break is limited initially by the main steam line flow restrictor. The mass flow rate through the upstream side of the break is assumed not to be affected by isolation valve closure until the isolation valves are closed far enough to establish limiting critical flow at the valve location. After limiting critical flow is established at the isolation valve, the mass flow rate is assumed to decrease linearly as the valve is closed. This assumption results in an almost constant mass flow out of the break until the last 3 seconds of a 10-second valve closure. The flow area from the downstream side of the break is limited initially by the downstream break area. The mass flow rate through the downstream side of the break is assumed not to be affected by the closure of the isolation valves in the unbroken steam lines until those valves are far enough closed to establish limiting critical flow at the valves. After limiting critical flow is established at the isolation valve positions the mass flow is assumed to decrease linearly as the valves close. This assumption results in an almost constant mass flow through the break until the last 3 seconds of a 10-second valve closure.

A reactor scram is initiated as the MSIVs begin to close (see Section VII-2). In addition to the scram initiated from MSIV closure, voids generated in the moderator during depressurization contribute significant negative reactivity to the core even before the scram is complete. Because the main steam line flow restrictors are sized for the Main Steam Line Break Accident, reactor vessel water level remains above the top of the fuel throughout the accident.

6.5.4.2 10CFR50.46 and Appendix K Sequence of Events and Systems Operations

The sequence of events following the postulated main steam line break for the compliance with 10CFR50.46 and 10CFR50 Appendix K is shown in Table XIV-6-18.[77]

TABLE XIV-6-17

PLANT OPERATIONAL PARAMETERS USED IN COOPER SAFER/GESTR-LOCA ANALYSIS

Plant Parameters	Nominal Value
Core Thermal Power (MWt)	2381
Corresponding Power (% of Rated)	100.0
Vessel Steam Output (lbm/hr)	$9.56x10^6$
Core Flow (lb/hr)	$73.5x10^6$
Vessel Steam Dome Pressure (psia)	1020
Initial Water Level (inches above vessel zero)	551.8

TABLE XIV-6-18

MAIN STEAM LINE BREAK ACCIDENT SEQUENCE
FOR 10CFR50.46 AND 10CFR50 APPENDIX K

Event	Time (Sec)
Break Occurs	0.0
Level 8 Trip	1.6
Steamline Covers	3.7
Feedwater Flow Reaches Zero	5.0
MSIV Closed	5.5
Steamline Uncovers	6.3
Level 3 Trip	12.3
SRVs Open	~89
Level 2 Trip	409.9
Level 1 Trip	1758.5
Diesel Generator (DG) Initiation (Level 1)	1758.5
DG at Speed and Bus Powered	1785.5
CS Pump Start Signal (Level 1 + Bus Power)	1785.5
LPCI Pump Start Signal (Level 1 + Bus Power)	1785.5
LPCI Pump at Rated Speed	1818.5
CS Pump at Rated Speed	1818.5
ADS Valves Open	1878.5
CS/LPCI IV Pressure Permissive Reached	2051.9
CS Injection Valve Begins Opening (Permissive)	2052.9
LPCI Injection Valve Begins Opening (Permissive)	2052.9
CS Pump Shutoff Head Reached	2058.6
CS Injection Valve Full Open	2074.9
CS Injection Occurs	2074.9
Second Peak PCT (404°F) Occurs[(1)]	2075.0
LPCI Pump Shutoff Head Reached	2082.6
LPCI Injection Valve Full Open	2097.9
LPCI Injection Occurs	2097.9
CS at Rated flow	2147.1
ADS Valves Closed	2206.1
LPCI at Rated Flow	2328.7

(1) Heatup is less than initial cladding temperature of 584°F.

6.5.5 Core and System Performance

6.5.5.1 Radiological Consequences Core and System Performance

The system performance following the postulated main steam line break for the radiological consequence follows:

The steam flow rate through the upstream side of the break increases from the initial flow of 700 lb/sec in the line to approximately 1400 lb/sec (about 200 percent of design steam flow for one steam line) with critical flow initially occurring at the flow restrictor. The steam flow rate was calculated using an ideal nozzle model. That the flow model predicts the behavior of the flow limiter has been substantiated by tests conducted on a scale model over a variety of pressure, temperature, and moisture conditions.

The steam flow rate through the downstream side of the break consists of equal flow components from each of the unbroken lines.

The steam flow rate in each of the unbroken lines increases from an initial value of 700 lb/sec to approximately 1400 lb/sec.

The total steam flow rate leaving the vessel is approximately 5600 lb/sec, which is in excess of the steam generation rate of 2800 lb/sec. The steam flow-steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 60 psi/sec. The formation of bubbles in the reactor vessel water causes a rapid rise in the water level. The analytical model used to calculate level rise predicts a rate of rise of about 6 feet/second. Thus, the water level reaches the vessel steam nozzles at 2 seconds after the break, as shown in Figure XIV-6-13a. From that time on, a two-phase mixture is discharged from the break. The two-phase flow rates are determined by vessel pressure and mixture enthalpy.[17] The vessel depressurization is calculated using a digital computer model in which the reactor vessel is divided into five major nodes. The model includes the flow resistance between nodes, as well as heat addition from the core.

As shown in Figure XIV-6-13a, two-phase flow is discharged through the break at an almost constant rate until late in the transient. This is the result of not taking credit for the effect of valve closure on flow rate until isolation valves are closed far enough to establish critical flow at the valve locations. The slight decrease in discharge flow rate is caused by depressurization inside the reactor vessel. The linear decrease in discharge flow rate at the end of the transient is the result of the assumption regarding the effect of valve closure on flow rate after critical flow is established at the valve location.

The following total masses of steam and liquid are discharged through the break prior to isolation valve closure.

Steam	20,000 pounds
Liquid	120,000 pounds

6.5.5.2 10CFR50.46 and Appendix K Core and System Performance

The core and system performance following the postulated main steam line break for the compliance with 10CFR50.46 and 10CFR50 Appendix K follows:[77]

The non-recirculation line breaks, such as the steamline break outside containment, are generally not limiting, in terms of PCT, because the coolant inventory loss is not as great. For these breaks, the systems remaining available correspond to the systems available for the recirculation suction break (for the same single failure) less the ECCS in which the break occurs (see Table VI-5-2).

At the beginning of the event, the vessel depressurizes rapidly causing the downcomer level to swell. This level swell reaches the steamline elevation at about 4 seconds, changing the break flow from steam to a two-phase mixture. The change in the break flow decreases the vessel depressurization rate. At 5.5 seconds into the event and as shown in Figure XIV-6-13b, the MSIVs complete their closure and the break is isolated. The vessel then begins to pressurize. The pressurization causes the water level in the downcomer to drop rapidly as the voids in this region collapse. The vessel pressure rises until the SRV opening setpoints are reached at 89 seconds. The actuation of two of the lowest setpoint SRVs is sufficient to handle the decay heat. The SRVs cycle repeatedly slowly depleting the reactor inventory.

The CNS ADS logic only requires a low level signal to initiate the ADS timer. The low water level (Level 1) trip setpoint is reached at about 1759 seconds. The ADS valves open at about 1878 seconds following the expiration of the ADS delay timer. The vessel depressurization and inventory loss by the ADS actuation causes the bundle water level to decrease slightly, but this brief core uncovery does not result in fuel heatup. The Core Spray and LPCI system begin injecting coolant at 2075 and 2098 seconds, respectively. Injection by both systems is dictated by the opening time of the injection valve after the pressure permissive is reached. The core and vessel are rapidly refilled by the ECCS injection. The PCT for this break is 584°F and occurs at the event initiation.

Because no fuel damage is calculated to occur as a result of this accident, the insertion of the reload fuel designs will not change the results of this accident analysis.

6.5.6 Barrier Performance

The Main Steam Line Break Accident involves a break outside of the Primary Containment which depressurizes the reactor vessel. Therefore, the integrity of the reactor vessel and Primary Containment are not threatened by this accident.

6.5.7 Radiological Consequences

The radiological consequences of the Main Steam Line Break Accident were assessed using the AXIDENT software code. This code was used to calculate the Control Room occupant, Exclusion Area Boundary, and Low Population Zone whole body, beta, and thyroid doses. Since this event does not result in core damage or fuel failure the radiological results are independent of fuel type.[91]

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the Reactor Coolant Pressure Boundary outside the Secondary Containment:

a. The amounts of steam and liquid discharged are as calculated in USAR Section XIV-6.5.5.1.

b. For purposes of calculating the Control Room occupant dose, the Technical Specification dose equivalent Iodine-131 limit of ≤ 0.2 μCi/g is assumed. The maximum Low Population Zone (LPZ) and Exclusion Area Boundary (EAB) doses are calculated assuming the Technical Specification spike dose equivalent Iodine-131 limit of ≤ 4 μCi/g. Since the noble gas activity will contribute only a small fraction of the total dose to the Control Room occupant (and in turn to the LPZ and EAB dose) they are not considered.

The concentrations of biologically significant radionuclides contained in the coolant discharged as liquid (as ratioed to the Technical Specifications limit for dose equivalent Iodine-131) are as follows:

Iodine Isotope	Dose Equivalent I-131 ≤ 0.2 μCi/g
Iodine 131	8.83×10^{-2} μCi/ml
Iodine 132	2.93×10^{-3} μCi/ml
Iodine 133	8.86×10^{-2} μCi/ml
Iodine 134	7.02×10^{-4} μCi/ml
Iodine 135	1.94×10^{-2} μCi/ml

6.5.7.1 Fission Product Release From Break

Using the above assumptions, the following amounts of radioactive materials are released from the Reactor Coolant Pressure Boundary:

Iodine Isotope	Coolant Activity
Iodine 131	1.81 Ci
Iodine 132	10.03 Ci
Iodine 133	10.68 Ci
Iodine 134	14.39 Ci
Iodine 135	13.74 Ci

The above releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

6.5.7.2 Steam Cloud Movement

The following initial conditions and assumptions are used in calculating the movement of the steam cloud:

a. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition. The total calculated mass of the steam cloud is 64,800 lbs.

b. The pressure buildup inside the turbine building will result in release of the steam cloud to the environment in a matter of seconds.

6.5.7.3 Radiological Effects

6.5.7.3.1 Offsite Consequence Results

The offsite consequences in terms of radiological doses resulting from the activity released to the environment during the Main Steam Line Break Accident have been determined based on the calculated Turbine Building atmospheric dispersion factor (X/Q) for the Exclusion Area Boundary and Low Population Zone (3.729E-4 sec/m^3 and 2.22E-4 sec/m^3 respectively). These X/Q values were generated using the methodology presented in Regulatory Guide 1.78 for an instantaneous "puff" release. The Exclusion Area Boundary and Low Population Zone radiological consequences for the Main Steam Line Break Accident have been assessed using the AXIDENT software code. The code was used to calculate the whole body, beta, and thyroid doses at these receptor locations considering both the maximum normal Iodine-131 equivalent value and the pre-accident iodine spike contained in the Technical Specifications. The limiting doses for the Main Steam Line Break Accident are shown in Table XIV-6-19. The EAB and LPZ doses are presented for a post-accident time period of 2 hours and 30 days respectively. However, since all of the activity is released to the environs in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated. These results are well within the guideline values of 10CFR100 and the acceptance criteria of Standard Review Plan 15.6.4.[91]

It is concluded that the health and safety of the public is not endangered as a consequence of this postulated accident.

6.5.7.3.2 Onsite (Control Room Personnel) Consequence Results

Following a Main Steam Line Break Accident, steam from the break can enter the control room and cable spreading room through the intake to the Control Room Air Conditioning System. The Control Room occupant radiological doses from a Main Steam Line Break Accident were assessed using the AXIDENT software code. The X/Q value for the Control Room intake is set so that the whole activity released from the Secondary Containment (cloud) enters the Control Room. Based on a hemispherical cloud and wind speed of 1 m/s, the cloud will completely pass over the Control Room air intake in less than one minute (57.3 seconds). During this time period, the source term is treated as a constant flow of air with a uniform radionuclide concentration being drawn into the Control Room over a period of the passage of the cloud followed by a continuous flow of clean air at a constant flow rate. It is assumed that the leakage paths into the Control Room draw from the same radioactive cloud. The normal air intake flow into the Control Room is 3235 ft^3/min. The unfiltered leakage into the Control Room is 71 ft^3/min for infiltration, and 10 ft^3/min for ingress/egress. This provides for a total air flow of 3316 cfm prior to Control Room Emergency Filter System (CREFS) initiation.

At 60 seconds, the normal intake is conservatively assumed to isolate. From this point on, the air intake passes through a filter with a certain efficiency. The flow rate is changed to that of the emergency air supply (900 ft^3/min ± 10%). Since the cloud has passed over the air intake by this time, only clean air will go through the CREFS filter; therefore, no filter is assumed in the modeling. The air intake flow rate after 60 seconds is 891 cfm (810 cfm CREFS supply + 71 cfm infiltration + 10 cfm ingress/egress). The resulting radionuclide concentration within the Control Room Envelope is diluted by the air space volume (64,640 ft^3 for the Control Room proper and 141,860 ft^3 for the entire Control Room Envelope).

The AXIDENT software code was used to calculate the whole body, beta, and thyroid doses at the Control Room receptor location. The results of the AXIDENT assessment are presented on Table XIV-6-19. The results are within the dose limits of 10CFR50 Appendix A, GDC 19.[91]

6.5.8 Basis for Setting of Rated Flow for Automatic Isolation

Initially, the high steam flow isolation limiting setpoint was 140 percent of rated flow and was based on the following:[18]

A 120 percent high steam flow isolation signal was originally established on the two-steam line plants, Oyster Creek and Nine Mile Point. The basis for this setting was not due specifically to a safety limit, but was set high enough to avoid spurious trips during normal operation and be low enough to minimize the consequences of any breaks of any size in a MSL for a plant of that steam line configuration. The radiological consequences were also calculated with a technical specification MSIV closure of 10 seconds.Cooper is a four steam line plant with a Technical Specification of 5 seconds maximum closure time for the MSIVs. Thus for Cooper, the basis of the original setting of 140 percent of rated steam flow as a requirement for the automatic isolation was that this setting permitted the plant to continue to operate at full power with one of the four main steam lines isolated. This resulted in an average of 133 percent of rated steam flow in each of the remaining steam lines.

TABLE XIV-6-19

MAIN STEAM LINE BREAK ACCIDENT
RADIOLOGICAL DOSE CONSEQUENCES

	Control Room (30 days)			EAB (2 hours)		LPZ (30 days)	
	Thyroid	Whole Body	Beta	Thyroid	Whole Body	Thyroid	Whole Body
Dose[1] (rem)	6.09	1.65E-3	1.22E-2	0.568	7.66E-3	0.338	4.56E-3
Dose[2] (rem)	N/A	N/A	N/A	11.36	0.153	6.76	0.0913

1. Dose is based on a dose equivalent I-131 of 0.2 µCi/g.

2. Dose is based on a dose equivalent I-131 of 4 µCi/g, derived by multiplying the previous results by 20 (the ratio of 4 µCi/g to 0.2 µCi/g).

Steam leaks inside the tunnel leading from the drywell to the turbine building are detected in one of two ways. The first method is by steam flow measurements, and if any flow exceeds 150 percent (analytical limit) of rated flow, the high-flow sensors initiate closure of the isolation valves in 0.5 second. The valves close in another 3-5 seconds, isolating the main steam lines no later than 5.5 seconds after the break.

The second method of steam leakage detection relies on temperature sensors in the main steam tunnel. The response time of a sensor to a nearby steam line break would depend upon the size and location of the break in relationship to the nearest sensor.

Either high flow setting of 120 percent or 140 percent rated steam flow as shown in Table XIV-6-20 did not affect the maximum radiological dose previously calculated in USAR Section XIV-6.5.7.3 for steam line breaks in the turbine building.

For small and medium breaks less than about 0.85 ft^2, only steam is released and therefore the radiological dose is much less than the DBA shown in Table XIV-6-19. For breaks larger than 0.85 ft^2, water carryover, and consequently radiological dose, would increase to the maximum permitted by the MSIV closure and the MSL flow limiters which have a total throat area of 2.60 ft^2. Steam line breaks larger than 2.60 ft^2 would have the same radiological consequences as the 2.60 ft^2 break because of the flow limiting effect of the venturi throat. Therefore, it is concluded that the maximum DBA dose will not be affected by the original Technical Specification high steam flow setting of 120 percent or 140 percent rated steam flow.

The bases for the Table XIV-6-20 dose calculation was that for a break less than that required for automatic closure, operator action would occur to terminate the break five minutes after its initiation. It is also assumed that there is no coincident loss of AC power and therefore the feedwater system would continue to make up water to the reactor vessel, the core would remain covered, and the MSIVs would close in 10 seconds. Should a loss of AC power occur simultaneously with the break, a low water level trip would occur resulting in isolation and initiation of HPCI and radiation dose would be less than that presented in Table XIV-6-20. Even if one assumes that 30 minutes is required to determine there is a break and isolate the reactor, the resultant dose is two orders of magnitude less than that for the DBA.

There would be no measurable difference between the original high-flow setting of 140 percent and 120 percent for a guillotine break of a main steam line as assumed in the DBA. Thus, the difference in radiological consequences between 140 percent and 120 percent only becomes a consideration in the very unlikely event of a break of the main steam line which happens to be within 0.14 ft^2 (120 percent) and 0.30 ft^2 (140 percent). The resulting radiological consequences in either case would be orders of magnitude less than the design base accident considered in USAR Section XIV-6.5.3. Table XIV-6-20 also shows that the dose for a 120 percent setting is 0.000013 percent and for 140 percent is 0.000027 percent of that allowed by 10CFR100. For this unlikely event then, when the break did occur between 0.14 ft^2 and 0.30 ft^2, the maximum resulting dose would only be increased by 0.000014 percent of that allowed by 10CFR100.

TABLE XIV-6-20

DOSE FOR VARIOUS MAIN STEAM HIGH FLOW SETTINGS

Main Steam High Flow Setting Analytical Limit (% of Rated Flow)	Detectible Break Size	Radiological Consequences for Thyroid Dose at Site Boundary	Percent of 10CFR 100 Limit for 5 Minute Closure
120%	0.14 ft^2	4E-5 Rems	.000013%
140%	0.30 ft^2	8E-5 Rems	.000027%
150%	0.38 ft^2	Note 3	Note 3

Note 1: Whole body gamma doses are much less than thyroid doses shown.

Note 2: Dose calculation assumed steam cloud released at height of turbine building with no down wash.

Note 3: The thyroid dose at 150% is estimated at 10% above the dose that would occur at the 140% flow setting.

Note 4: The total venturi area for the 4 Main Steam lines is 2.60 square feet. This is the break size at which choked flow occurs.

Note 5: The Main Steam line thyroid dose at the site boundary for the design bases case (full guillotine break) is described in Section XIV-6.5.7.3 and the EAB thyroid dose consequences are shown in Table XIV-6-19.

Note 6: For break sizes larger than 0.85 square feet, water carryover, and consequently radiological dose, will increase accordingly.

The current high flow analytical limit is 150 percent of rated steam flow. Analysis[33,34] shows this corresponds to an increase in break size detectibility from 0.30 ft^2 (140 percent) to 0.38 ft^2 (150 percent) for this instrument and that the ability of the other steam line break sensors (see USAR Section VII-3.4) to detect and isolate the break is not necessarily affected. The difference in total dose release between the two break sizes was conservatively calculated to be about 10 percent. This 10 percent increase in the extremely small dose calculated for the 140 percent setpoint does not significantly change the existing margin to the 10CFR100 limits.

7.0 RESULTS AND CONCLUSIONS

The following sections provide summaries of the results of the CNS safety analyses for abnormal operational transients, special events and accidents.

7.1 Abnormal Operational Transient Analysis Results and Conclusions

Table XIV-7-1 provides a summary of the results of the station safety analyses for abnormal operational transients. The spectrum of abnormal operational transients has been approached and analyzed by a method that included various combinations of plant problems and operating conditions. The abnormal operational transients include cycle-specific limiting transients and non-limiting transients.

The limiting transients have been analyzed to show that using the operating limit MCPRs in the COLR, the safety limit MCPR is not violated. The safety limit MCPR is set to correspond to the criterion that 99.9% of the fuel rods are expected to avoid boiling transition. Table XIV-7-2 lists the operating limit MCPRs for the current cycle based on the ΔCPR results from the abnormal operational transients analysis and the cycle specific safety limit MCPR. The non-limiting transients were evaluated using MCHFR verifying that no fuel failure would occur with MCHFR greater than 1.0 or that the transient is bounded by the results for another transient within its category. Using either the MCPR or MCHFR criterion, none of the abnormal operational transients results in fuel damage. Several transients are also analyzed with respect to the maximum thermal and mechanical overpowers to assure the MAPLHGR and LHGR limits for each segment of fuel bundle is not violated. The MAPLHGR and LHGR limits ensure that actual fuel operation is maintained within the fuel rod thermal-mechanical design and safety bases, i.e. fuel damage, failure and coolability, as described in USAR Chapter III. Assuming no fuel failures or less than 0.1% fuel failures occur, the 10 CFR Part 20 criteria will be met. Therefore, it can be concluded that safety design basis 1 and 2 for abnormal operational transients are avoided.

The abnormal operational transient, MSIV Closure without scram described in USAR Section IV-4.4.6, is the most limiting with respect to peak Reactor Coolant Pressure Boundary (RCPB) pressure. The analysis shows that the peak RCPB pressure does not exceed 1375 psig. No other transient analyzed in USAR Chapter XIV result in a higher peak pressure. Therefore, it can be concluded that safety design basis 3 for abnormal operational transients is met.

7.2 Special Events Analysis Results and Conclusions

The analysis described in USAR Section XIV-5.9.1 shows that CNS has the ability to bring the reactor to the hot and cold shutdown condition by manipulation of the local controls and equipment available outside of the control room. Therefore, the safety design bases 1 and 2 are met.

The analysis described in USAR Sections XIV-5.9.2 and 3 show that the reactor can be brought to a safe shutdown condition without depending on control rods using Standby Liquid control system. In order to demonstrate the SLC system effectiveness, the ATWS high pressure ARI signal is also assumed to fail. The analysis also shows that the consequences of an ATWS will not exceed the limits of 10CFR50.62. Therefore, the safety design bases 3 and 4 are met.

The analysis described in USAR Section XIV-5.9.4 shows that CNS has the ability to withstand and recover from a Station Blackout for the required coping duration. Therefore, the safety design basis 5 is met.

7.3 Accident Analysis Results and Conclusions

Table XIV-7-3 provides a summary of the results of the CNS safety analyses for accidents.

The accident analyses, described in USAR Section XIV-6, shows that the maximum offsite radiological consequences for all four (4) accidents are well within 10 CFR100 (10CFR50.67 for Fuel Handling Accident and LOCA) and, therefore, the safety design basis 1 is met.

The CRDA calculations using the low incremental rod worths which result from the BPWS indicate that peak fuel enthalpy is well below the 280 cal/g design limit and rarely exceeds the 170 cal/g fuel cladding failure threshold. The 10CFR50 Appendix K LOCA accident analysis, described in USAR Chapter VI, assumes the fuel MAPLHGRs. These MAPLHGRs ensure that actual fuel operation is maintained within the fuel rod safety bases, as described in USAR Chapter III. The 10CFR50 Appendix K LOCA analysis, described in USAR Chapter VI, shows that the peak cladding temperature is less than 2200°F and no fuel failure occurs during a LOCA. The MSLB analysis, provided in Section XIV-6.5, shows that the PCT is less than 2200°F and that fuel damage will not occur. Therefore, safety design basis 2 is met.

The transient effects to the RCPB (including static and differential stresses, temperatures, and cooldown rates) were analyzed during the CRDA. Maintaining the peak fuel enthalpy less than the design value of 280 cal/g ensures that the pressure rise rate in the primary system will be less than 50 psi/sec and pose no threat to the RCPB. Since the CRDA is the only accident for which pressurization of the RCPB is a factor, safety design basis 3 is met.

Primary Containment stresses were evaluated following a LOCA to assure that they will not exceed those allowed for accidents by applicable industry codes when Primary Containment is required. The peak calculated pressure following a LOCA is 54.4 psig (with MELLL and ICF) which is below the maximum allowable pressure of 62 psig. The peak DBA LOCA drywell airspace temperature was analyzed to be 301.4°F. The DBA LOCA peak suppression pool temperature was analyzed to be 208.7°F. The peak drywell temperature is maintained for a short time and does not raise the structural drywell temperature above the design value of 281°F[85]. Therefore, safety design basis 4 is met.

The accident analyses, described in USAR Section XIV-6, shows that the maximum onsite radiological consequences for all four (4) accidents are well within the General Design Criteria 19 (10CFR50.67 for Fuel Handling Accident and LOCA). Therefore, safety design basis 5 is met.

TABLE XIV-7-1
STATION SAFETY ANALYSIS
RESULTS OF ABNORMAL OPERATIONAL TRANSIENTS

Undesired Parameter Variation	Event Causing Transient	Scram Caused By	Typical Peak Nuclear System Pressure
CYCLE-SPECIFIC LIMITING TRANSIENTS			
Nuclear system pressure increase	Generator Trip without bypass****	Turbine control valve fast closure	*****
Nuclear system pressure increase	Turbine Trip without bypass****	Turbine stop valve closure	*****
Reactor water temperature decrease	Loss of feedwater heater****	None	*****
Positive reactivity insertion	Continuous rod withdrawal during power range operation****	None	Not applicable***
Core coolant flow decrease	Recirculation Pump Seizure****	None	Not applicable*
Coolant inventory excess	Feedwater controller failure maximum demand****	Turbine stop valve closure on high water level	*****
Reactor water temperature decrease	Inadvertent coldwater pump start	None	Not applicable***
NON-LIMITING TRANSIENTS			
Nuclear system pressure increase	Generator Trip with bypass	Turbine control valve fast closure	*****
Nuclear system pressure increase	Turbine Trip with bypass	Turbine stop valve closure	*****
Nuclear system pressure increase	Instantaneous loss of main condenser vacuum	Turbine stop valve closure	*****
Nuclear system pressure increase	DEH Pressure Controller Output signal fails	Main steam line isolation valve closure	Not applicable*
Nuclear system pressure increase	Main steam line isolation valve closure with scram	Main steam line isolation valve closure	*****
Reactor water temperature decrease	Shutdown cooling malfunction	None	Not applicable***
Positive reactivity insertion	Continuous rod withdrawal during reactor startup	High neutron flux	Not applicable

TABLE XIV-7-1
(CONTINUED)

Undesired Parameter Variation	Event Causing Transient	Scram Caused By	Typical Peak Nuclear System Pressure
Positive reactivity insertion	Continuous rod removal error during refueling	None	Not applicable
Positive reactivity insertion	Fuel Assembly Insertion Error	None	Not applicable
Coolant inventory decrease	DEH Pressure Controller Output Fails Open	Main steam line isolation valve closure	Not applicable*
Coolant inventory decrease	Open relief valve or safety valve	None	Not applicable*
Coolant inventory decrease	Total loss of feedwater flow	Reactor vessel low water level	Not applicable*
Coolant inventory decrease	Loss of all offsite power to station auxiliaries	Loss of power to reactor protection system	*****
Core coolant flow decrease	Recirculation flow control failure-decreasing flow	None	Not applicable*
Core coolant flow decrease	Trip of one recirculation pump	None	Not applicable*
Core coolant flow decrease	Trip of two recirculation pumps	None	Not applicable*
Core coolant flow increase	Recirculation pump flow control failure-increasing flow	High neutron flux	Not applicable***
Core coolant flow increase	Startup of idle recirculation pump	None	Not applicable***
Core coolant temperature increase	Loss of shutdown cooling	None	Not applicable***

* This transient results initially in depressurization of the reactor.
** This transient results in a depressurization.
*** This transient results in no significant change in RCPB pressure.
**** This transient involves significant changes in power and is limiting because of MCPR.[1]
***** Peak pressure on all abnormal operational transients is less than the MSIV closure with flux scram event used to evaluate the Pressure Relief System (see USAR Section IV-4.6).

TABLE XIV-7-2

SUPPLEMENTAL RELOAD LICENSING REPORT RESULTS FOR
COOPER NUCLEAR STATION RELOAD 30 CYCLE 31

BOC	Beginning of cycle exposure – 0.0 GwD/MtU.
MOC1	Middle of cycle exposure breakpoint #1 - Determined using the EOR value minus 5426 MwD/MtU.
MOC2	Middle of cycle exposure breakpoint #2 - Determined using the EOR value minus 3222 MwD/MtU.
EOR	End of Rated cycle exposure with initial condition 100% power, 100% flow and all rods withdrawn. Value in the SRLR is 14881 MwD/MtU. This value can be updated during the cycle based on actual core operation.
EOC	End of cycle.
ICF	Increased core flow - Initial condition with core flow of 105% rated at 100% power.
MELLLA	Maximum Extended Load Line Limit analysis - Initial condition with core flow of 76.8% rated at 100% power.
HBB	Hard Bottom Burn – very bottom peaked axial power shape was used in the cycle depletion.
UB	Under Burned – more mid peaked power shape was used in the cycle depletion.
1TBVOOS	One Turbine Bypass Valve Out Of Service – initial condition.
Flux	Thermal neutron flux.
Q/A	Thermal heat flux.
CPR	Critical Power Ratio.

Core-wide Anticipated Operational Occurrences Analysis Results

Methods used: GEMINI and GEXL-Plus

Event Exposure range/Initial conditions		Flux (Percent rated)	Q/A (Percent rated)	Maximum Uncorrected ΔCPR GNF2
Generator Load Reject without Bypass				
BOC to MOC1	HBB, MELLLA	143	101	0.15
MOC1 to MOC2	HBB, MELLLA	259	112	0.23
MOC2 to EOC	HBB, MELLLA	291	116	0.23
BOC to MOC1	HBB, MELLLA, 1TBVOOS	143	101	0.15
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	291	116	0.23
BOC to MOC1	HBB, ICF	184	105	0.16
MOC1 to MOC2	HBB, ICF	319	117	0.22
MOC2 to EOC	HBB, ICF	351	121	0.21
BOC to MOC1	HBB, ICF, 1TBVOOS	184	105	0.16
MOC1 to EOC	HBB, ICF, 1TBVOOS	351	121	0.21
MOC2 to EOC	UB, MELLLA	219	109	0.25
MOC1 TO EOC	UB, MELLLA, 1TBVOOS	219	109	0.25
MOC2 to EOC	UB, ICF	316	117	0.27
MOC1 to EOC	UB, ICF, 1TBVOOS	316	117	0.27
Turbine Trip without Bypass				
BOC to MOC1	HBB, MELLLA	137	100	0.14
MOC1 to MOC2	HBB, MELLLA	241	110	0.22
MOC2 to EOC	HBB, MELLLA	278	115	0.23
BOC to MOC1	HBB, MELLLA, 1TBVOOS	137	100	0.14
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	278	115	0.23
BOC to MOC1	HBB, ICF	173	103	0.15

TABLE XIV-7-2 (CONT'D)

Event Exposure range/Initial conditions		Flux (Percent rated)	Q/A (Percent rated)	Maximum Uncorrected ΔCPR GNF2
MOC1 to MOC2	HBB, ICF	301	115	0.22
MOC2 to EOC	HBB, ICF	337	120	0.21
BOC to MOC1	HBB, ICF, 1TBVOOS	173	103	0.15
MOC1 to EOC	HBB, ICF, 1TBVOOS	337	120	0.21
MOC2 to EOC	UB, MELLLA	205	107	0.25
MOC1 TO EOC	UB, MELLLA, 1TBVOOS	205	107	0.25
MOC2 to EOC	UB, ICF	290	115	0.27
MOC1 TO EOC	UB, ICF, 1TBVOOS	290	115	0.27
Feedwater Controller Failure				
BOC to MOC1	HBB, MELLLA	126	107	0.13
MOC1 to MOC2	HBB, MELLLA	223	114	0.21
MOC2 to EOC	HBB, MELLLA	255	118	0.21
BOC TO MOC1	HBB, MELLLA, 1TBVOOS	133	107	0.15
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	275	120	0.22
BOC to MOC1	HBB, ICF	149	110	0.16
MOC1 to MOC2	HBB, ICF	251	120	0.22
MOC2 to EOC	HBB, ICF	298	125	0.22
BOC TO MOC1	HBB, ICF, 1TBVOOS	164	110	0.19
MOC1 to EOC	HBB, ICF, 1TBVOOS	327	127	0.23
MOC2 to EOC	UB, MELLLA	187	110	0.22
MOC1 to EOC	UB, MELLLA, 1TBVOOS	202	112	0.24
MOC2 to EOC	UB, ICF	251	120	0.26
MOC1 to EOC	UB, ICF, 1TBVOOS	281	122	0.29
Inadvertent HPCI Initiation with Level 8 Turbine Trip				
BOC to MOC1	HBB, MELLLA	119	111	0.16
MOC1 to MOC2	HBB, MELLLA	214	117	0.23
MOC2 to EOC	HBB, MELLLA	247	122	0.23
BOC TO MOC1	HBB, MELLLA, 1TBVOOS	126	111	0.17
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	264	124	0.25
BOC to MOC1	HBB, ICF	145	113	0.17
MOC1 to MOC2	HBB, ICF	252	123	0.24
MOC2 to EOC	HBB, ICF	282	127	0.23
BOC TO MOC1	HBB, ICF, 1TBVOOS	154	113	0.19
MOC1 to EOC	HBB, ICF, 1TBVOOS	304	129	0.24
MOC2 to EOC	UB, MELLLA	177	113	0.24
MOC1 to EOC	UB, MELLLA, 1TBVOOS	189	115	0.26
MOC2 to EOC	UB, ICF	238	122	0.27
MOC1 to EOC	UB, ICF, 1TBVOOS	257	123	0.29

Cycle MCPR Values

Two recirculation loop operation safety limit: see Technical Specifications
Single recirculation loop operation safety limit: see Technical Specifications

Operating Limit MCPR = safety limit + ΔCPR

Non-pressurization Events[1]

Exposure range: BOC to EOC

Event	Operating Limit MCPR All Fuel Types
Loss of feedwater heating	1.24
Rod withdrawal error (for RBM setpoint to 114 percent)	1.37
Fuel loading error	N/A
Rated Equivalent SLO Pump Seizure[2]	1.46

Pressurization Events

Pressurization Events Domain Summary Table

		Operating Limit MCPR	
		Option A	Option B
Operating Domain	**Exposure Range**	**GNF2**	**GNF2**
MELLLA/ICF[3]	BOC to MOC1	1.42	1.32
	MOC1 to MOC2	1.49	1.39
	MOC2 to EOC	1.57	1.47
MELLLA/ICF+1TBVOOS[3]	BOC to MOC1	1.45	1.35
	MOC1 to EOC	1.59	1.49

Pressurization Events Domain Results

		Operating Limit MCPR	
-Event		**Option A**[4]	**Option B**[4]
Exposure Range/Initial Conditions		**GNF2**	**GNF2**
Generator Load Reject without Bypass			
BOC to MOC1	HBB, MELLLA	1.40	1.30
MOC1 to MOC2	HBB, MELLLA	1.48	1.38
MOC2 to EOC	HBB, MELLLA	1.51	1.41
BOC TO MOC1	HBB, MELLLA, 1TBVOOS	1.40	1.30
MOC1 TO EOC	HBB, MELLLA, 1TBVOOS	1.51	1.41
BOC to MOC1	HBB, ICF	1.41	1.31
MOC1 to MOC2	HBB, ICF	1.48	1.38
MOC2 to EOC	HBB, ICF	1.50	1.40
BOC TO MOC1	HBB, ICF, 1TBVOOS	1.41	1.31
MOC1 TO EOC	HBB, ICF, 1TBVOOS	1.50	1.40
MOC2 to EOC	UB, MELLLA	1.54	1.44
MOC1 TO EOC	UB, MELLLA, 1TBVOOS	1.54	1.44
MOC2 to EOC	UB, ICF	1.57	1.47
MOC1 TO EOC	UB, ICF, 1TBVOOS	1.57	1.47
Turbine Trip without Bypass			
BOC to MOC1	HBB, MELLLA	1.39	1.29
MOC1 to MOC2	HBB, MELLLA	1.47	1.37
MOC2 to EOC	HBB, MELLLA	1.51	1.41
BOC TO MOC1	HBB, MELLLA, 1TBVOOS	1.39	1.29
MOC1 TO EOC	HBB, MELLLA, 1TBVOOS	1.51	1.41
BOC to MOC1	HBB, ICF	1.40	1.30

		Operating Limit MCPR	
-Event		**Option A(4)**	**Option B(4)**
Exposure Range/Initial Conditions		**GNF2**	**GNF2**
MOC1 to MOC2	HBB, ICF	1.47	1.37
MOC2 to EOC	HBB, ICF	1.50	1.40
BOC TO MOC1	HBB, ICF, 1TBVOOS	1.40	1.30
MOC1 TO EOC	HBB, ICF, 1TBVOOS	1.50	1.40
MOC2 to EOC	UB, MELLLA	1.54	1.44
MOC1 TO EOC	UB, MELLLA, 1TBVOOS	1.54	1.44
MOC2 to EOC	UB, ICF	1.56	1.46
MOC1 TO EOC	UB, ICF, 1TBVOOS	1.56	1.46
Feedwater Controller Failure			
BOC to MOC1	HBB, MELLLA	1.38	1.28
MOC1 to MOC2	HBB, MELLLA	1.46	1.36
MOC2 to EOC	HBB, MELLLA	1.49	1.39
BOC to MOC1	HBB, MELLLA, 1TBVOOS	1.40	1.30
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	1.51	1.41
BOC to MOC1	HBB, ICF	1.41	1.31
MOC1 to MOC2	HBB, ICF	1.48	1.38
MOC2 to EOC	HBB, ICF	1.50	1.40
BOC TO MOC1	HBB, ICF, 1TBVOOS	1.44	1.34
MOC1 to EOC	HBB, ICF, 1TBVOOS	1.52	1.42
MOC2 to EOC	UB, MELLLA	1.50	1.40
MOC1 to EOC	UB, MELLLA, 1TBVOOS	1.53	1.43
MOC2 to EOC	UB, ICF	1.55	1.45
MOC1 to EOC	UB, ICF, 1TBVOOS	1.58	1.48
Inadvertent HPCI Initiation with Level 8 Turbine Trip			
BOC to MOC1	HBB, MELLLA	1.41	1.31
MOC1 to MOC2	HBB, MELLLA	1.48	1.38
MOC2 to EOC	HBB, MELLLA	1.52	1.42
BOC TO MOC1	HBB, MELLLA, 1TBVOOS	1.42	1.32
MOC1 to EOC	HBB, MELLLA, 1TBVOOS	1.54	1.44
BOC to MOC1	HBB, ICF	1.42	1.32
MOC1 to MOC2	HBB, ICF	1.49	1.39
MOC2 to EOC	HBB, ICF	1.52	1.42
BOC TO MOC1	HBB, ICF, 1TBVOOS	1.45	1.35
MOC1 to EOC	HBB, ICF, 1TBVOOS	1.53	1.43
MOC2 to EOC	UB, MELLLA	1.53	1.43
MOC1 to EOC	UB, MELLLA, 1TBVOOS	1.55	1.45
MOC2 to EOC	UB, ICF	1.56	1.46
MOC1 to EOC	UB, ICF, 1TBVOOS	1.59	1.49

(1) Fuel loading error not analyzed due to adoption of Amendment 28. The OLMCPR for the thermal-hydraulic stability analysis was determined to be non-limiting in cycle 31.

(2) The cycle-independent OLMCPR calculated for the recirculation pump seizure event while in SLO for GNF2 is 1.65, based on the cycle-specific SLO SLMCPR. When adjusted for the off-rated power/flow conditions of SLO, this limit corresponds to a rated OLMCPR of 1.46 for GNF2.

(3) Includes the extended operating domain for core flows from MELLLA up to the licensed ICF value.

(4) Option A and Option B refer to the adjustment method applied to the MCPR for each event to account for uncertainties and different scram speeds. Refer to USAR Section III-7.7.1.

TABLE XIV-7-3

STATION SAFETY ANALYSIS
RESULTS OF DESIGN BASIS ACCIDENTS

Design Basis Accident	Parameter for Fuel Rupture	Maximum Primary Containment Pressures/ Temperature	Peak RCPB Pressure	Off-Site Dose (rem) Exclusion Area Boundary (2 hours)		Off-Site Dose (rem) Low Population Zone (30 days)		Onsite Dose (rem) Control Room Occupant	
				Whole-Body	Thyroid	Whole-body	Thyroid	Whole-body	Thyroid
Control Rod Drop Accident	170 cal/gm < 280 cal/gm	Not applicable*	50 psi/sec < 1375 psig	0.105	0.788	0.111	2.06	0.0082	6.53
Loss of Coolant Accident	MAPLHGR < Fuel Design	54.4 psig drywell, 301.4°F drywell, 208.7°F suppression pool	Not applicable*	1.029 (TEDE)	N/A	5.726 (TEDE)	N/A	3.325 (TEDE)	N/A
Fuel Handling Accident	Not applicable	Not applicable*	Not applicable**	1.45 (TEDE)	N/A	0.809 (TEDE)	N/A	4.568 (TEDE)	N/A
Main Steam Line Break Accident	MCHFR > 1.0	Not applicable*	Not applicable*	0.153	11.36	0.0913	6.76	0.165	6.09

* This accident results in a depressurization.
** This accident occurs with the reactor vessel head off.

8.0 EVALUATION OF ENGINEERED SAFETY FEATURE SYSTEMS USING TID-14844 SOURCE TERMS

An evaluation was made of the adequacy of the Cooper Nuclear Station Primary Containment and engineered safety features using the assumptions of TID-14844 with regard to the fission product source term. (The SGT heat loading has been updated based on Regulatory Guide 1.183 AST LOCA release assumptions as discussed in Section 8.2 and Table XIV-8-2.)

8.1 Source Terms Assumptions (TID-14844)

For the purposes of calculating the dose, heat loading, air-borne or water-borne activity, the following assumptions were made:

a. The halogen and noble gas initial sources were taken directly from Table IV, External Gamma Dose Rates of TID-14844[25]. These were converted to activity by dividing by the "average" energy for each isotope taken from the same table.

b. The core particulate activity was taken from the ANS standard afterheat curve. The activity at any time was obtained by dividing the afterheat curve at that particular time by an average energy of 0.7 MeV.

c. The charcoal absorber iodine loading includes iodine-129 and iodine-127. The amount of each of these isotopes in the core was determined from NEDO 24782.

d. The activity in the suppression pool was assumed to be 50% of the core halogen inventory and 1% of the core particulate activity which are instantaneously released to the suppression pool.

e. The airborne activity in the primary containment was assumed to consist of 100% of the core noble gas activity, 25% of the core halogen activity, and 1% of the core particulate activity which are instantaneously released to the Primary Containment.

f. The airborne activity noted in item e is released at a constant leak rate of 0.635% per day to the Secondary Containment, uniformly mixed in the Secondary Containment and released to the Standby Gas Treatment System at the rate of 1.5 air changes per day.

g. For the determination of the activity and heat loading on the charcoal absorbers and the HEPA filters in the Standby Gas Treatment System the Primary Containment activity noted in item e above was assumed to be released at a constant leak rate of 0.635% per day which is taken directly to the Standby Gas Treatment System where the filter and absorber efficiency was assumed to be 100%.

The activities in the various systems at various times after the TID-14844 release accident are shown in Table XIV-8-1. The values in Table XIV-8-1 are based on a Primary Containment leak rate of 0.635% per day. The values are conservative in that the calculations did not take into account the depletion of the Primary Containment source strength due to the leakage from the Primary Containment.

8.2 Standby Gas Treatment System

The Standby Gas Treatment System contains two complete filtration trains where each train contains a moisture separator, a heater to control relative humidity, a roughing filter, a HEPA filter, charcoal absorbers, and a downstream HEPA filter. A complete description is given in Section V-3.3.4.

The HEPA filters are steel cased, open-faced, high efficiency, particulate filters. These filters are rated for 250°F continuous service. The Cooper Nuclear Station design incorporates two such filter banks in each filtration train. One filter bank is located upstream of the charcoal with the second HEPA bank downstream of the charcoal. In the analysis it was assumed that all particulate activity entering the Standby Gas Treatment System was deposited uniformly on the upstream HEPA filter bank.

The charcoal absorber is loaded into stainless steel drawers. The drawers are loaded with the charcoal in 2-inch deep horizontal beds. The inlet plenum is arranged such that the flow to each drawer is uniform and for the purpose of this analysis the activity was assumed to be loaded uniformly in the charcoal.

The Standby Gas Treatment System contains low leakage dampers necessary to isolate the trains and route the flow through either train. The system can be operated with a small flow through the shutdown filter train. This small flow will remove decay heat from the HEPA filters and charcoal absorber in the shutdown train. The heat load and air temperatures at specific points are listed in Table XIV-8-2. The SGT filter heat loads listed in Table XIV-8-1 were calculated based on TID-14844 release assumptions while the heat loads used in Table XIV-8-2 are updated based on Regulatory Guide 1.183 AST LOCA release assumptions.[93] The maximum air temperature listed in Table XIV-8-2 is less than 190°F. This is well below the continuous rated temperature for the HEPA filters and well below the charcoal ignition temperature of 625°F.

The charcoal absorber in each train contains 360 pounds of charcoal. The total iodine loading at the end of 30 days is 810 grams. This is predominantly iodine-127 and iodine-129. The resultant specific loading is 4.9 milligrams of iodine per gram of charcoal. This specific loading is within the recommended design loads quoted in reference [26].

The radiation dose to the critical components of the Standby Gas Treatment System has been investigated. There would not be significant radiation material damage in one train from the activity deposited on the absorbers and filters of the other train.

The highest doses in the Standby Gas Treatment System room occur in the vicinity of the charcoal absorbers. The charcoal drawer and the HEPA gaskets are closed cell neoprene sponge rubber. The 30-day integrated doses to the charcoal drawer gaskets and the HEPA gaskets are 1.6×10^8 Rads and 8.7×10^7

TABLE XIV-8-1

ACTIVITY, MASS LOADING AND HEAT LOADING AT VARIOUS LOCATIONS
FOR TID-14844 RELEASE ASSUMPTIONS

Primary Containment Leak Rate of 0.635% Day

	1 Hr	8 Hr	1 Day	10 Days	30 Days	Peak Valve & Time
Activity in S.P. (curies)	5.36×10^{8}	2.27×10^{8}	1.30×10^{8}	0.39×10^{8}	0.20×10^{8}	10.655×10^{8} at 0 hr
Heat Load S.P. (kW)	4.90×10^{3}	1.54×10^{3}	6.62×10^{2}	1.49×10^{2}	0.81×10^{2}	8.28×10^{3} at 0 hr
Activity Airborne P.C.(curies)	8.88×10^{8}	4.79×10^{8}	3.18×10^{8}	0.88×10^{8}	0.23×10^{8}	14.406×10^{8} at 0 hr
Heat Load S.P. (kW)	4.48×10^{3}	1.55×10^{3}	7.67×10^{2}	1.73×10^{2}	0.78×10^{2}	8.74×10^{3} at 0 hr
Activity Air S.C. (curies)	2.31×10^{5}	8.63×10^{5}	11.30×10^{5}	4.83×10^{5}	0.94×10^{5}	11.4×10^{5} at 32 hr
Heat Load S.C. (watts)	7.67×10^{2}	16.37×10^{2}	12.67×10^{2}	4.24×10^{2}	1.12×10^{2}	16.9×10^{2} at 12 hr
Activity on HEPA (curies)	219	4.72×10^{3}	2.13×10^{4}	1.49×10^{5}	4.31×10^{5}	Has not peaked at 30 days
Heat Load HEPA (watts) (Note 1)	0.88	19.5	89	787	1790	Has not peaked at 30 days
Activity on C.F. (curies)	1.0×10^{5}	3.6×10^{5}	5.8×10^{5}	1.1×10^{6}	6.0×10^{5}	11.3×10^{6} at 11.62 days
Heat C.F. (watts) (Note 1)	552	1.6×10^{3}	1.9×10^{3}	2.7×10^{3}	1.43×10^{3}	2.6×10^{3} at 11.62 days
Iodine Load on C.F. (grams)	1.24	9.90	29.65	288	810	

S.P. denotes Suppression Pool
P.C. denotes Primary Containment
S.C. denotes Secondary Containment
HEPA denotes High Efficiency Particulate Absolute Filter
C.F. denotes Charcoal Filter

Note 1 - The SGT HEPA and Carbon Filter heat loads have been updated based on Regulatory Guide 1.183 AST LOCA release assumptions and are listed in Table XIV-8-2.[93]

TABLE XIV-8-2

STANDBY GAS TREATMENT SYSTEM HEAT LOADS AND TEMPERATURES
FOR RG 1.183 AST LOCA RELEASE ASSUMPTIONS

Condition	1,492 ft^3/min	100 ft^3/min
Upstream HEPA Inlet Temperature	170°F	140°F
Upstream HEPA Heat Load	243 watts	243 watts
Temperature Rise Across HEPA	0.6°F	8.7°F
Upstream HEPA Exit Temperature	170.6°F	148.7°F
Charcoal Heat Load	1014 watts	1014 watts
Temperature Rise Across Charcoal	2.4°F	36.4°F
Charcoal Exit Temperature	173.0°F	185.1°F

Rads, respectively. While these doses are above the dose of 10^7 Rads normally considered acceptable for neoprene, this gasket material is still preferred. The basis for this preference stems from the excellent experience with this type gasket.[26] The ultra-conservative releases considered herein are far less probable than the highly improbable releases that would give doses up to the values normally considered acceptable for neoprene. Further, the highest iodine releases occur early in the accident before the neoprene has received a high dose; thus, the principal iodine activity would be absorbed on the charcoal before the gasket begins to degrade.

The dose to the Standby Gas Treatment System fan motors and fan drives is less than 10^6 Rads. This dose is well below the level of significant damage to all materials in these components.

The Standby Gas Treatment System contains a number of low leakage electrically operated dampers. The damper with the highest dose rate will be open while the activity is building on the HEPA filter and the charcoal absorber. It is not necessary to change the position of the damper to switch to the other train or to provide the small bypass cooling flow necessary to remove decay heat from the shutdown damper. Thus, even if this damper failed there would not be severe consequences. The damper that must operate to switch filter trains and to provide decay heat removal will receive a dose on the order of 10^6 Rads. The dampers contain several materials that are subject to radiation damage. The silicone rubber damper blade edge seals and the damper motor insulation will withstand the dose of 10^6 Rads without damage. The teflon bearings that support the end of the damper blades will suffer radiation damage. The dose will result in a 50% property damage in the tensile and shear strength. The elastic modulus will increase less than 50%. Even with this rather severe radiation damage these bearings should be functional.

8.3 ECCS Components

The ECCS components are located in compartments shielded from the torus and the drywell. All ECCS components that handle torus water after the Loss-of-Coolant Accident (LOCA) are located inside the Secondary Containment. There are no ECCS components located inside the Primary Containment that would be required to function following the postulated LOCA and that could suffer significant radiation damage. The ECCS components have been examined for materials that are subject to radiation damage.

The principal dose to the ECCS components located in the shielded compartments is from the radioactive torus water being circulated through the ECCS components. The RHR pump suction is 24-inch, the discharge is 18-inch. The dose rate on the surface of a 24-inch schedule 40 pipe is listed in Table XIV-8-3. The integrated dose is for various times after the accident based on the source terms listed in Table XIV-8-1. The doses to various ECCS components in the compartments are evaluated relative to the reference dose where the reference dose is equated to the 30-day dose on the surface of the 24-inch schedule 40 pipe. This reference dose is 6.7 x 10^6 Rads.

The RHR and Core Spray pumps must operate after the LOCA. These pumps contain seals made of a carbon washer moving relative to a metal seat. Both the washer and seat materials have thresholds for damage on the order of 10^{11} Rads.[27] This threshold for damage is orders of magnitude above the reference dose of 6.7 x 10^6 Rads. The seals also contain elastomer O-rings made of buna-N and a special elastomer material. One of the O-rings has a teflon retainer ring. All of the O-rings and seals are stationary. All of these materials suffer 25% damage from doses on the order of 4 x 10^6 rads which is less than the reference dose. The radiation dose to the teflon would be even a higher percentage of damage; however, if the O-rings and the teflon retainer failed there would be only a slight increase in leakage and the pump would continue to run and provide the essential ECCS cooling as all of the O-rings are stationary seals.

TABLE XIV-8-3

DOSE RATES AND INTEGRATED DOSES FOR VARIOUS EQUIPMENT AND LOCATIONS
BASED ON TID-14844 RELEASE ASSUMPTIONS

Equipment or Location	Maximum Dose Rate (Rad/Hr)	Integrated Dose (Rad)		
		12 Hours	2 Days	30 Days
Surface 24-inch Schedule 40 Pipe	3.4×10^{5}	1.1×10^{6}	2.1×10^{6}	6.7×10^{6}
Interior Surface of Drywell	8.2×10^{6}	2.2×10^{7}	4.3×10^{7}	1.5×10^{8}

The RHRS and Core Spray pump motors contain reservoirs of lubricating oil. The lubricating oil will stand a dose on the order of 10^8 Rads which is well above the reference dose of 6.7 x 10^6 Rads. The RHR and core spray pump motor insulation is a proprietary material with a radiation damage tolerance for 25% damage to the dielectric properties essentially the same as the reference dose. However, it should be noted that the motors are located several feet from the pipes and therefore should receive less than the reference dose.

8.4 Materials Within Primary Containment

Drywell Coating

The doses at the interior surface of the drywell are listed in Table XIV-8-3. These doses are based on the source strength listed in Table XIV-8-1. Interior surface of the drywell is coated with an inorganic zinc primer with an epoxy phenolic top coat. This material will be subjected to a 30-day dose of 1.5 x 10^8 Rads at the surface of the drywell. The coating material is capable of withstanding doses of over 10^9 Rads without any deterioration.[28] The material used for repair of coating has the property to withstand the same or higher dose as the original material.[87]

Electrical Penetrations

The Primary Containment electrical penetrations contain double seals. One seal is inside the primary shield and is subjected to a 30-day integrated dose of 1.5 x 10^8 Rads. The other seal which is outside the primary shield exists as a test barrier and would be subjected to a dose that is orders of magnitude less than at the interior surface of the drywell. Both seals are made of a ceramic material composed of a mixture of several metallic oxides. The ceramic material will stand doses of up to 10^{10} Rads without any significant damage to the sealing properties.

Thermocouple penetrations X-209A and C are designed with a flanged barrier with pressure seals provided by double O-rings and an epoxy material.

8.5 Summary

All of the safety-related core cooling functions are capable of operating and performing their intended functions while subject to the conservative design basis radiation sources. Using the ultraconservative sources postulated in TID-14844 all systems are capable of operating and performing their intended function even though some of the materials are marginal or relatively deteriorated at 30 days after the release. All of the questionable materials are used in static conditions where failure would not interfere with the operation of the active components. Thus, the system functions would continue. However, the material failure could result in increase of primary containment leakage through pump seals or in charcoal absorber and HEPA filter bypass. While other materials could be substituted for the questionable materials, it is felt that the substitute materials would not perform as well for the conditions where the dose rate is less. Thus, the use of substitute materials would compromise the ability to deal with the more probable lower activity release accident.

The temperatures of the charcoal and HEPA filters in the Standby Gas Treatment System are well below the service temperatures and the charcoal ignition temperatures when the Standby Gas Treatment System is operating on the minimum flow condition.

9.0 REFERENCES FOR CHAPTER XIV

1. NEDO 24011, Generic Reload Fuel Application.

2. Supplemental Reload Licensing Submittal for CNS, Unit 1, Reload 6, January, 1981.

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