Attachment 2 to AEP:NRC:1140

Proposed Revised Technical Specifications Pages

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux (High and Low Setpoint)	Less than or equal to 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature delta T	Less than or equal to 6.0 seconds*
8. Overpower delta T	Less than or equal to 6.0 seconds*
9. Pressurizer Pressure--Low	Less than or equal to 1.0 seconds
10. Pressurizer Pressure--High	Less than or equal to 1.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	Less than or equal to 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	Less than or equal to 1.0 seconds
14. Steam Generator Water Level--Low-Low	Less than or equal to 1.5 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	Less than or equal to 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	Less than or equal to 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Service Water System	Not Applicable
	Containment Air Recirculation Fan	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	Less than or equal to 27.0@@/27.0++
b.	Reactor Trip (from SI)	Less than or equal to 3.0
c.	Feedwater Isolation	Less than or equal to 8.0
d.	Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	Less than or equal to 13.0#/48.0##

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/27.0++
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 48.0++/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low	
a. Safety Injection (ECCS)	Less than or equal to 29.0@@/39.0@
b. Reactor Trip (from SI)	Less than or equal to 5.0
c. Feedwater Isolation	Less than or equal to 10.0
d. Containment Isolation-Phase "A"	Less than or equal to 20.0#/30.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Less than or equal to 15.0#/50.0##
h. Steam Line Isolation	Less than or equal to 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
6.	Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a.	Safety Injection (ECCS)	Less than or equal to 27.0@@/37.0@
b.	Reactor Trip (from SI)	Less than or equal to 3.0
c.	Feedwater Isolation	Less than or equal to 8.0
d.	Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	Less than or equal to 14.0#/48.0##
h.	Steam Line Isolation	Less than or equal to 11.0
7.	Containment Pressure--High-High	
a.	Containment Spray	Less than or equal to 45.0
b.	Containment Isolation-Phase "B"	Not Applicable
c.	Steam Line Isolation	Less than or equal to 10.0
d.	Containment Air Recirculation Fan	Less than or equal to 660.0
8.	Steam Generator Water Level--High-High	
a.	Turbine Trip	Less than or equal to 2.5
b.	Feedwater Isolation	Less than or equal to 11.0
9.	Steam Generator Water Level--Low-Low	
a.	Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b.	Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10.	4160 volt Emergency Bus Loss of Voltage	
a.	Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11.	Loss of Main Feedwater Pumps	
a.	Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12.	Reactor Coolant Pump Bus Undervoltage	
a.	Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

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DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements, with one exception. This exception is the CVCS boron makeup system and the BIT.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks.

c.1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel with nominal enrichment above 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU. In Region 1, fuel shall be stored in a three-out-of-four cell configuration with one symmetric cell location of each 2 x 2 cell array vacant.

2. The boundary between the Region 1 mentioned above and the rest of the spent fuel storage racks (defined as Region 2) shall be such that the three-out-of-four storage requirement shall be carried into Region 2 by, at least, one row as shown in Figure 5.6-1.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the reactor protection system.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of 10 R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

Table 3.3-6 is based on the following Alarm/Trip Setpoints and Measurement Ranges for each instrument listed. For the unit vent noble gas monitors, it should be noted that there is an automatic switchover from the low/mid-range channels to the high-range channel when the upper limits of the low- and mid-range channel measurement ranges are reached. In this case there is no flow to the low- and mid-range channels from the unit vent sample line. This is considered to represent proper operation of the this monitor. Therefore, if automatic switchover to the high-range should occur, and the low- and mid-range detectors are capable of functioning when flow is re-established, the low- and mid-range channels should not be declared inoperable and the ACTION statement in the Technical Specification does not apply. This is also true while purging the low- and mid-range chambers following a large activity excursion prior to resumption of low-level monitoring and establishment of a new background.

INSTRUMENT	ALARM/TRIP SETPOINT	MEASUREMENT RANGE*
1) Area Monitor- Upper Containment (VRS 1101/1201)	The monitor trip setpoint is based on 10 CFR 20 limits. A homogeneous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate.	10^{-4}R/hr to 10R/hr.

* This is the minimum required sensitivity of the instrument. Indicated values on these instruments above or below these minimum sensitivity ranges are acceptable and indicate existing conditions not instrument inoperability.

Radiation Monitoring Instrumentation (Continued)

INSTRUMENT	ALARM/TRIP SETPOINT	MEASUREMENT RANGE*
2) Area Monitor Containment High Range (VRA 1310/1410)	The monitor setpoint was selected to reflect the guidance provided in Generic Letter 83-37 for NUREG-0737 Technical Specifications	1R/hr to 1×10^7 R/hr Photons.
3) Process Monitor Particulate (ERS 1301/1401)	The monitor trip setpoint is based on 10 CFR 20 The setpoint was determined using the Noble gas setpoint and historical monitor data of the ratio of particulates to Noble gases.	1.5×10^{-4} uCi/cc to 7.5 uCi/cc
4) Process Monitor Noble Gas (ERS 1305/1405)	The monitor trip setpoint is based on 10 CFR 20 limits. A homogeneous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate.	5.8×10^{-7} uCi/cc to 2.7×10^{-2} uCi/cc
5) Steam Generator PORV (MRA 1601) (MRA 1602) (MRA 1701) (MRA 1702)	Not Applicable.**	0.1uCi/cc to $1.0\text{x}10^2$ uCi/cc.

* This is the minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

Radiation Monitoring Instrumentation (Continued)

INSTRUMENT	ALARM/TRIP SETPOINT	MEASUREMENT RANGE*
6) Noble Gas Unit Vent Monitors		
a) Low Range (VRS 1505)	See Bases Section 3/4.3.3.10	5.8×10^{-7} uCi/cc to 2.7×10^{-2} uCi/cc
b) Mid Range (VRS 1507)	Not Applicable**	1.3×10^{-3} uCi/cc to 7.5×10^{2} μCi/cc
c) High Range (VRS 1509)	Not Applicable**	2.9×10^{-2} uCi/cc to 1.6×10^{4} uCi/cc
7) Gland Steam Condenser Vent Noble Gas Monitor		
a) Low Range (SRA 1805)	See Bases Section 3/4.3.3.10	5.8×10^{-7} uCi/cc to 2.7×10^{-2} uCi/cc
8) Steam Jet Air Ejector Vent Noble Gas Monitor		
a) Low Range (SRA 1905)	See Bases Section 3/4.3.3.10	5.8×10^{-7} uCi/cc to 2.7×10^{-2} uCi/cc.
b) Mid Range (SRA 2907)	Not applicable.**	1.3×10^{-3} uCi/cc to 7.5×10^{2} uCi/cc.
c) High Range (SRA 2909)	Not Applicable.**	2.9×10^{-2} uCi/cc to 1.6×10^{4} uCi/cc.
9) Spent Fuel Storage (RRC-330)	The monitor setpoint is selected to alarm and trip consistent with 10 CFR 70.24(a) (2)	1×10^{-1} mR/hr to 1×10^{4} mR/hr

* This is minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

Radiation Monitoring Instrumentation (Continued)

The Radiation Monitoring Instrumentation Surveillance Requirements per Table 4.3-3 are based on the following interpretation:

1) The CHANNEL FUNCTIONAL TEST is successfully accomplished by the injection of a simulated signal into the channel, as close to the detector as practical, to verify the channel's alarm and or trip function only.

2) The CHANNEL CALIBRATION as defined in T/S Section 1.9 permits the "known values" generated from radioactive calibration sources to be supplemented with "known values" represented by simulated signals for that subset of "known values" required for calibration and not practical to generate using the radioactive calibration sources.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. For the meteorological instrumentation, the required channel check consists of a qualitative assessment of channel behavior during operation by observation. For the 10 m wind speed and wind direction instruments the channel check also includes, when possible, a comparison of channel indications.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.5.1 APPENDIX R REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Appendix R remote shutdown instrumentation ensures that sufficient instrumentation is available to permit shutdown of the facility to COLD SHUTDOWN conditions at the local shutdown indication (LSI) panel. In the event of a fire, normal power to the LSI panels may be lost. As a result, capability to repower the LSI panels from Unit 2 has been provided. If the alternate power supply is not available, fire watches will be established in those fire areas where loss of normal power to the LSI panels could occur in the event of fire. This will consist of either establishing continuous fire watches or verifying OPERABILITY of fire detectors per Specification 4.3.3.7 and establishing hourly fire watches. The details of how these fire watches are to be implemented are included in a plant procedure.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection systems/detectors ensures that adequate detection capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of the fires will reduce the potential for damage to safety related systems or components in the areas of the specified systems and is an integral element in the overall facility fire protection program. In the event that a portion of the fire detection systems is inoperable, the action statements provided maintain the facility's fire protection program and allows for continued operation of the facility until the inoperable system(s)/detector(s) are restored to OPERABILITY. However, it is not our intent to rely upon the compensatory action for an extended period of time and action will be taken to restore the minimum number of detectors to OPERABLE status within a reasonable period.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

The containment water level and containment sump level transmitters will be modified or replaced and OPERABLE by the end of the refueling outage scheduled to begin in February 1989.

*Amendment 112 (Effective before startup following refueling outage currently scheduled in 2/89).

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

3/4.3.3.9 The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

3/4.3.3.10 The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria specified in Section 11.3 of the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.

BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 170°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of $70^{\circ}F$. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to $80^{\circ}F$ to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

TABLE 3.3-2
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	Less than or equal to 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux High Negative Rate	Less than or equal to 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature Delta T	Less than or equal to 6.0 seconds*
8. Overpower Delta T	Less than or equal to 6.0 seconds*
9. Pressurizer Pressure--Low	Less than or equal to 2.0 seconds
10. Pressurizer Pressure--High	Less than or equal to 2.0 seconds
11. Pressurizer Water Level--High	Less than or equal to 2.0 seconds

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1. Manual		
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Essential Service Water System	Not Applicable
	Containment Air Recirculation Fan	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Purge and Exhaust Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2. Containment Pressure-High		
a.	Safety Injection (ECCS)	Less than or equal to 27.0@@/27.0++
b.	Reactor Trip (from SI)	Less than or equal to 3.0
c.	Feedwater Isolation	Less than or equal to 8.0
d.	Containment Isolation-Phase "A"	Not Applicable
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/27.0++
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation - Phase "A"	Less than or equal to 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
g. Essential Service Water System	Less than or equal to 48.0++/13.0#
4. Differential Pressure Between Steam Lines - High	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation - Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
g. Essential Service Water System	Less than or equal to 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable
h. Steam Line Isolation	Less than or equal to 13.0

TABLE 3.3-5 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	Less than or equal to 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
g. Essential Service Water System	Less than or equal to 14.0#/48.0##
h. Steam Line Isolation	Less than or equal to 11.0
7. Containment Pressure--High-High	
a. Containment Spray	Less than or equal to 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	Less than or equal to 10.0
d. Containment Air Recirculation Fan	Less than or equal to 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE NOTATION

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

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Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T'') is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. The overpower delta T reactor trip provides protection or back-up protection for at-power steam line break events. Credit was taken for operation of this trip in the steam line break mass/energy releases outside containment analysis. In addition, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the uncontrolled control rod assembly bank withdrawal at-power event.

BASES

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of 10 R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

Table 3.3-6 is based on the following Alarm/Trip Setpoints and Measurement Ranges for each instrument listed. For the unit vent noble gas monitors, it should be noted that there is an automatic switchover from the low/mid-range channels to the high-range channel when the upper limits of the low- and mid-range channel measurement ranges are reached. In this case there is no flow the low- and mid-range channels from the unit vent sample line. This is considered to represent proper operation of this monitor. Therefore, if automatic switchover to the high-range should occur, and the low- and mid-range detectors are capable of functioning when flow is re-established, the low- and mid-range channels should not be declared inoperable and the ACTION statement in the Technical Specification does not apply. This is also true while purging the low- and mid-range chambers following a large activity excursion prior to resumption of low-level monitoring and establishment of a new background.

INSTRUMENTATION

BASES

Radiation Monitoring Instrumentation (Continued)

INSTRUMENT	ALARM/TRIP SETPOINT	MEASUREMENT RANGE*
1) Area Monitor-Upper Containment (VRS 2101/2201)	The monitor trip setpoint is based on 10 CFR 20 limits. A homogenous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate.	10^{-4}R/hr to 10R/hr.
2) Area Monitor-Containment High Range (VRA 2310/2410)	The monitor setpoint was selected to reflect the guidance provided in Generic Letter 83-37 for NUREG-0737 Technical Specifications.	1R/hr to 1 x 10^{7}R/hr Photons.
3) Process Monitor Particulate (ERS 2301/2401)	The monitor trip setpoint is based on 10 CFR 20 The setpoint was detemined using the Noble gas setpoint and historical monitor data of the ratio of particulate o Noble gases.	1.5×10^{-4}uCi to 7.5 uCi.
4) Process Monitor Noble Gas (ERS 2305/2405)	The monitor trip setpoint is based on 10 CFR 20 limits. A homogenous mixture of the containment atmosphere is assumed. The setpoint value is defined as the monitor reading when the purge is operating at the maximum flow rate.	5.8×10^{-7}uCi/cc to 2.7×10^{-2}uCi/cc
5) Steam Generator PORV (MRA 2601) (MRA 2602) (MRA 2701) (MRA 2702)	Not Applicable.**	0.1uCi/cc to 1.0×10^{2} uCi/cc.

*This is the minimum required sensitivity of the instrument. Indicated values on these instruments above or below these minimum sensitivity ranges are acceptable and indicate existing conditions not instrument inoperability.

Radiation Monitoring Instrumentation (Continued)

INSTRUMENT	ALARM/TRIP SETPOINT	MEASUREMENT RANGE*
6) Noble Gas Unit Vent Monitors		
a) Low Range (VRS 2505)	See Bases Section 3/4.3.3.10	$5.8x10^{-7}$uCi/cc to $2.7x10^{-2}$ uCi/cc.
b) Mid Range (VRS 2507)	Not Applicable**	$1.3x10^{-3}$uCi/cc to $7.5x10^{2}$uCi/cc
c) High Range (VRS 2509)	Not Applicable**	$2.9x10^{-2}$uCi/cc to $1.6x10^{4}$uCi/cc
7) Gland Steam Condenser Vent Noble Gas Monitor		
a) Low Range (SRA 2805)	See Bases Section 3/4.3.3.10	$5.8x10^{-7}$uCi/cc to $2.7x10^{-2}$uCi/cc.
8) Steam Jet Air Ejector Vent Noble Gas Monitor		
a) Low Range (SRA 2905)	See Bases Section 3/4.3.3.10	$5.8x10^{-7}$uCi/cc to $2.7x10^{-2}$uCi cc.
b) Mid Range (SRA 2907)	Not applicable.**	$1.3x10^{-3}$uCi/cc to $7.5x10^{2}$uCi/ cc.
c) High Range (SRA 2909)	Not Applicable.**	$2.9x10^{-2}$uCi/cc to $1.6x10^{4}$uCi/ cc.
9) Spent Fuel Storage (RRC-330)	The monitor setpoint is selected to alarm and trip consistent with 10 CFR 70.24(a) (2)	$1x10^{-1}$mR/hr to $1x10^{4}$mR/hr

* This is minimum sensitivity of the instrument for normal operation, to follow the course of an accident, and/or take protective actions. Values of the instrument above or below this minimum sensitivity range are acceptable.

** These monitors are used to provide data to assist in post-accident off-site dose assessment.

BASES

The Radiation Monitoring Instrumentation Surveillance Requirements per Table 4.3-3 are based on the following interpretation:

1) The CHANNEL FUNCTIONAL TEST is successfully accomplished by the injection of a simulated signal into the channel, as close to the detector as practical, to verify the channel's alarm and/or trip function only.

2) The CHANNEL CALIBRATION as defined in T/S Section 1.9 permits the "known values" generated from radioactive calibration sources to be supplemented with "known values" required for calibration and not practical to generate using the radioactive calibration sources.

EMERGENCY CORE COOLING SYSTEMS

BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

[illegible]

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. Reactor coolant system cooldown can be caused by inadvertent depressurization, a LOCA or steam line rupture. The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

Attachment 3 to AEP:NRC:1140

Existing T/S Pages Marked to Reflect Proposed Changes

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION | PAGE

3/4.4 REACTOR COOLANT SYSTEM (Continued)

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux (High and Low Setpoint)	Less than or equal to 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature delta T	Less than or equal to 6.0 seconds*
8. Overpower delta T	~~NOT APPLICABLE~~ Less than or equal to 6.0 sec.*
9. Pressurizer Pressure--Low	Less than or equal to 1.0 seconds
10. Pressurizer Pressure--High	Less than or equal to 1.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	$\leq$ 1.0 seconds
13. Loss of Flow - Two loops (Above P-7 and below P-8)	$\leq$ 1.0 seconds
14. Steam Generator Water Level--Low-Low	$\leq$ 1.5 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE

write out signs

Write out symbols

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. **Manual**	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. **Containment Pressure-High**	
a. Safety Injection (ECCS)	$\leq 27.0^{*}$ @@/27.0 H
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{\#}/28.0^{\#\#}$
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	$\leq 13.0^{\#}/48.0^{\#\#}$

Write out symbols

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
3.	Pressurizer Pressure-Low	27.0@@/27.0++
a.	Safety Injection (ECCS)	~~$\leq$ 27.0*/13.0#~~
b.	Reactor Trip (from SI)	$\leq$ 3.0
c.	Feedwater Isolation	$\leq$ 8.0
d.	Containment Isolation-Phase "A"	$\leq$ 18.0#
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	$\leq$ 48.0*/13.0#
4.	Differential Pressure Between Steam Lines-High	$\leq$27.0@@/37.0@
a.	Safety Injection (ECCS)	~~$\leq$ 13.0#/23.0##~~
b.	Reactor Trip (from SI)	$\leq$ 3.0
c.	Feedwater Isolation	$\leq$ 8.0
d.	Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	$\leq$ 13.0#/48.0##
5.	Steam Flow in Two Steam Lines - High Coincident with T_{avg} --Low-Low	$\leq$29.0@@/39.0@
a.	Safety Injection (ECCS)	~~$\leq$ 15.0#/25.0##~~
b.	Reactor Trip (from SI)	$\leq$ 5.0
c.	Feedwater Isolation	$\leq$ 10.0
d.	Containment Isolation-Phase "A"	$\leq$ 20.0#/30.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	$\leq$ 15.0#/50.0##
h.	Steam Line Isolation	$\leq$ 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
6.	Steam Flow in Two Steam Lines-High Coincident With Steam Line Pressure-Low	
a.	Safety Injection (ECCS)	≤ 27.0@@/37.0@@ ~~≤ 13.0#/23.0##~~
b.	Reactor Trip (from SI)	≤ 3.0
c.	Feedwater Isolation	≤ 8.0
d.	Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e.	Containment Purge and Exhaust Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	Not Applicable
g.	Essential Service Water System	≤ 14.0#/48.0##
h.	Steam Line Isolation	≤ 11.0
7.	Containment Pressure--High-High	
a.	Containment Spray	≤ 45.0
b.	Containment Isolation-Phase "B"	Not Applicable
c.	Steam Line Isolation	≤ 10.0
d.	Containment Air Recirculation Fan	≤ 660.0
8.	Steam Generator Water Level--High-High	
a.	Turbine Trip	≤ 2.5
b.	Feedwater Isolation	≤ 11.0
9.	Steam Generator Water Level--Low-Low	
a.	Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b.	Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10.	4160 volt Emergency Bus Loss of Voltage	
a.	Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11.	Loss of Main Feedwater Pumps	
a.	Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12.	Reactor Coolant Pump Bus Undervoltage	
a.	Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A minimum contained volume of 900 gallons of borated water,

b. Between 20,000 and 22,500 ppm of boron, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the water level through a recirculation flow test at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

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EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be ≥ 145°F at least once per 8 hours; otherwise, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be ≥ 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total contained volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal T_{avg} of 70°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable ~~Surveillance Requirements~~, Insert A

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1: The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel with nominal enrichment above 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU. In Region 1, fuel shall be stored in a three-out-of-four cell configuration with one symmetric cell location of each 2 x 2 cell array vacant.

2. The boundary between the Region 1 mentioned above and the rest of the spent fuel storage racks (defined as Region 2) shall be such that the three-out-of-four storage requirement shall be carried into Region 2 by, at least, one row as shown in Figure 5.6-1.

A

with one exception. This exception is the CVCS boron makeup system and the BIT.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. ~~No credit was taken for operation of this trip in the accident analyses; however,~~ its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

~~The reference average temperature (T'') is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis.~~ The overpower delta T reactor trip provides protection or back-up protection for at power steamline break events. Credit was taken for operation of this trip in the steamline break mass/energy releases outside containment analysis. In addition

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF proposes from diverse parameters.

redundancy

The OPERABILITY of these system is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these system is consistent with the assumptions used in the accident analyses.

should be systems

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

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See next page

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

Noble gas effluent monitors provide information, during and following an accident, which is considered helpful to the operator in assessing the plant condition. It is desired that these monitors be OPERABLE at all times during plant operation, but they are not required for safe shutdown of the plant.

In addition, a minimum of two in containment radiation-level monitors with a maximum range of 10^7 R/hr for photon only should be OPERABLE at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible.

INSERT B

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 170°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

~~3/4.5.4 BORON INJECTION SYSTEM~~

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

~~The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.~~

~~The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.~~

3/4.5.5 REFUELING WATER STORAGE TANK

Insert C

The OPERABILITY of the RWST as part of the ECCS ensures that a ~~sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA~~. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to 80°F to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.

INSERT C

...sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. RCS cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

BASES

SECTION PAGE

TABLE 3.3-2
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	Less than or equal to 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux High Negative Rate	Less than or equal to 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature Delta T	Less than or equal to 6.0 seconds*
8. Overpower Delta T	~~NOT APPLICABLE~~ Less than or equal to 6.0 seconds*
9. Pressurizer Pressure--Low	Less than or equal to 2.0 seconds
10. Pressurizer Pressure--High	Less than or equal to 2.0 seconds
11. Pressurizer Water Level--High	Less than or equal to 2.0 seconds

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Purge and Exhaust Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	Less than or equal to 27.0* @@/270H
b. Reactor Trip (from SI)	Less than or equal to 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	Less than or equal to ~~24.0*/12.0*~~ 27.0@@/37.0++
b. Reactor Trip (from SI)	Less than or equal to ~~2.0~~ 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
g. Essential Service Water System	Less than or equal to 48.0~~*~~ ++/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	Less than or equal to ~~12.0*/[illegible]~~ 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to ~~2.0~~ 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60..0
g. Essential Service Water System	Less than or equal to 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low	
a. Safety Injection (ECCS)	Not Applicable
b. Reactor Trip (from SI)	Not Applicable
c. Feedwater Isolation	Not Applicable
d. Containment Isolation-Phase "A"	Not Applicable
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	Not Applicable
h. Steam Line Isolation	Less than or equal to 13.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	Less than or equal to ~~12.0*/24.0**~~ 27.0@@/37.0@
b. Reactor Trip (from SI)	Less than or equal to ~~2.0~~ 3.0
c. Feedwater Isolation	Less than or equal to 8.0
d. Containment Isolation-Phase "A"	Less than or equal to 18.0*/28.0**
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
g. Essential Service Water System	Less than or equal to 14.0*/48.0**
h. Steam Line Isolation	Less than or equal to 11.0
7. Containment Pressure--High-High	
a. Containment Spray	Less than or equal to 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	Less than or equal to 10.0
d. Containment Air Recirculation Fan	Less than or equal to 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Less than or equal to 2.5
b. Feedwater Isolation	Less than or equal to 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	Less than or equal to 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps~~, SI and RHR pumps.~~

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging, SI, and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

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~~EMERGENCY CORE COOLING SYSTEMS~~

~~3/4.5.4 BORON INJECTION SYSTEM~~

~~BORON INJECTION TANK~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.5.4.1 The boron injection tank shall be OPERABLE with:~~

~~a. A minimum contained borated water volume of 900 gallons,~~

~~b. Between 20,000 and 22,500 ppm of boron, and~~

~~c. A minimum solution temperature of 145°F.~~

~~APPLICABILITY: MODES 1, 2 and 3.~~

~~ACTION:~~

~~With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:~~

~~a. Verifying the contained borated water volume at least once per 7 days,~~

~~b. Verifying the boron concentration of the water in the tank at least once per 7 days, and~~

~~c. Verifying the water temperature at least once per 24 hours.~~

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EMERGENCY CORE COOLING SYSTEMS

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be $\geq$ 145°F at least once per 8 hours; otherwise, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be $\geq$ 145°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. ~~No credit was taken for operation of this trip in the accident analyses; however,~~ the functional capability of the Overpower Delta T trip at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the uncontrolled control rod assembly bank withdrawal at power event.

The reference average temperature (T'') is set equal power indicated Tavg to ensure fuel integrity during conditions for the range of full power average tempe assumed in the safety analysis. The overpower delta trip provides protection or back-up protection for at steamline break events. Credit was taken for operation trip in the steamline break mass/energy releases outside containment analysis. In addition

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF proposes from diverse parameters.

The OPERABILITY of these system is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these system is consistent with the assumptions used in the accident analyses.

Protection has been provided for main feedwater system malfunctions in MODES 3 and 4. This protection is required when main feedpumps are aligned to feed steam generators in MODES 3 and 4. The availability of feedwater isolation on high-high steam generator level terminates the addition of cold water to the steam generators in any main feedwater system malfunction. The total volume that can be added to the steam generators by the main feedwater system in MODES 3 and 4 is limited by this safeguards actuation and the fact that feedwater isolation on low T_{avg} setpoint coincident with reactor trip can only be cleared above the low-low steam generator level trip setpoint.

The restrictions associated with bypassing ESF trip functions below either P-11 or P-12 provide protection against an increase in steam flow transient and are consistent with assumptions made in the safety analysis.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

Insert B

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ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes @ and @@) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times (Note ++), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumption used for VCT and RWST valves are valid.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

~~3/4.5.4 BORON INJECTION SYSTEM~~

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

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

~~The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.~~

3/4.5.5 REFUELING WATER STORAGE TANK

INSERT C

The OPERABILITY of the RWST as part of the ECCS ensures that ~~a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA.~~ The limits of RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

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...sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown, and ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. RCS cooldown can be caused by inadvertent depressurization, a loss of coolant accident or a steam line rupture.

Attachment 5 to AEP:NRC:1140

Main Steam Line Break Outside
Containment Analysis Summary

Main Steam Line Break Outside of Containment

Introduction

A break of a main steam line outside of containment in the main steam enclosure or main steam accessway presents a concern for the operability of equipment in those areas. The break would cause an immediate pressure peak. Later, as the steam generator tubes uncover, superheated steam release would lead to high local temperatures. The analysis of this event is considered in this section.

Appendix O of the original FSAR contains the results of the high energy line break analysis for the areas outside containment. In 1984, the issue of steam generator superheat with main steam line break (MSLB) outside of containment was raised in NRC IE Information Notice 84-90 (1). This notice described a potential problem pertaining to plant analysis and equipment qualification with respect to a MSLB with release of superheated steam. An analysis for the affected compartments was performed in response to this Notice (2), and is summarized in Section 14.4.6 of the Unit 1 FSAR. The FSAR analysis is based on mass and energy releases of a generic study performed by Westinghouse (3), and is applicable to both units.

As part of the rerating analysis (4), Westinghouse recalculated the mass and energy release rates to cover the range of conditions and plant parameter changes discussed in Section S-3.3.1 of their report. To support the new mass and energy releases, the temperature transient was reanalyzed by AEPSC using the RELAP5/Mod2 computer code (5) (the original analysis was performed using RELAP4/Mod5 (6); however, the codes are similar.) The analysis of the new mass and energy release rates shows that peak temperatures in the steam compartments are higher than previously calculated. However, these temperature peaks are brief, and the surface temperatures of the affected equipment do not exceed their applicable qualification temperatures because of the thermal lag between the equipment and its environment.

Description of Event

When the steam line breaks, the initial burst of steam creates a brief pressure peak. Air is flushed out, and the compartments fill with steam, the peak temperature being dictated by the steam enthalpy. Automated actuations close the main steam isolation valves, limiting the break flow to the affected loop, as well as initiating reactor trip and safety injection. The exact signals and actuations depend on the break size, and are discussed in the mass and energy release analysis. As the event progresses, the break flow rate decreases, and natural circulation begins to take effect. Cool air first enters the lower compartments, cooling the regions below the break. Above the break, cool air and steam mix to limit the temperature of those compartments.

Description of Analysis

The subcompartments of the east and west main steam enclosures and accessway are shown in Figures 1 and 2. These are nodalized consistently with the analysis of record. Since different but similar computer codes are used, a benchmark case was run to compare the two codes. As was expected, the results are similar. The effect of heat structures in removing energy is minimal for this analysis, so they were conservatively ignored.

The mass and energy release analysis (4) provides 68 separate cases which vary break size, unit, initial power level, and auxiliary feedwater flow rate. Several cases were reviewed, and from the resulting trend a limiting break was selected. The limiting break is defined as the break which causes the maximum equipment surface temperature, which is the criteria of interest. The limiting mass and energy release for all instruments was found to be a 1.2 square foot break on Unit 2 from 70% initial power.

The temperature response of the required instruments was analyzed by modeling the instrument as a heat structure. Additional margin was added to the heat transfer rate to the instrument to conservatively overpredict the surface temperature, consistent with the recommendations given in the applicable guidelines (7).

The mass release rate of the FSAR analysis in the first few seconds is greater than the release of any of the current cases, and the pressure transients for that analysis are limiting. The current FSAR pressure analysis remains bounding.

Results

Table 1 summarizes the qualification temperature, old and new peak compartment temperatures, and the maximum surface temperature reached by the equipment. Although the service condition temperature has increased from the FSAR analysis, the time at the very high temperatures is short, and the equipment has little time to respond. All instrument surface temperatures remain below the qualification temperatures, and are therefore acceptable for use.

References

1) NRC IE Information Notice No. 84-90, "MSLB Effect on Environmental Qualification of Equipment," December 7, 1984 (AEP:NRC:9216)

2) MSLB Environmental Analysis, Donald C. Cook Units 1 and 2 Impell Report No. 01-01200-1524, Revision 0, September 1986

3) Steam Line Break Mass/Energy Releases for Equipment Qualification Outside Containment, WCAP-10961-P, October 1985

4) Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2, Licensing Report, WCAP 11902, Supplement 1, 1989

5) RELAP5/Mod2 Code Manual, NUREG/CR-4312, August 1985

6) RELAP5/Mod5 - A Computer Program for Transient Thermal Hydraulic Analysis of Nuclear Reactors and Related Systems Vol 1-3, ANCR-NUREG-1335, September 1976

7) Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment, NUREG 0588, Rev. 1, July 1981

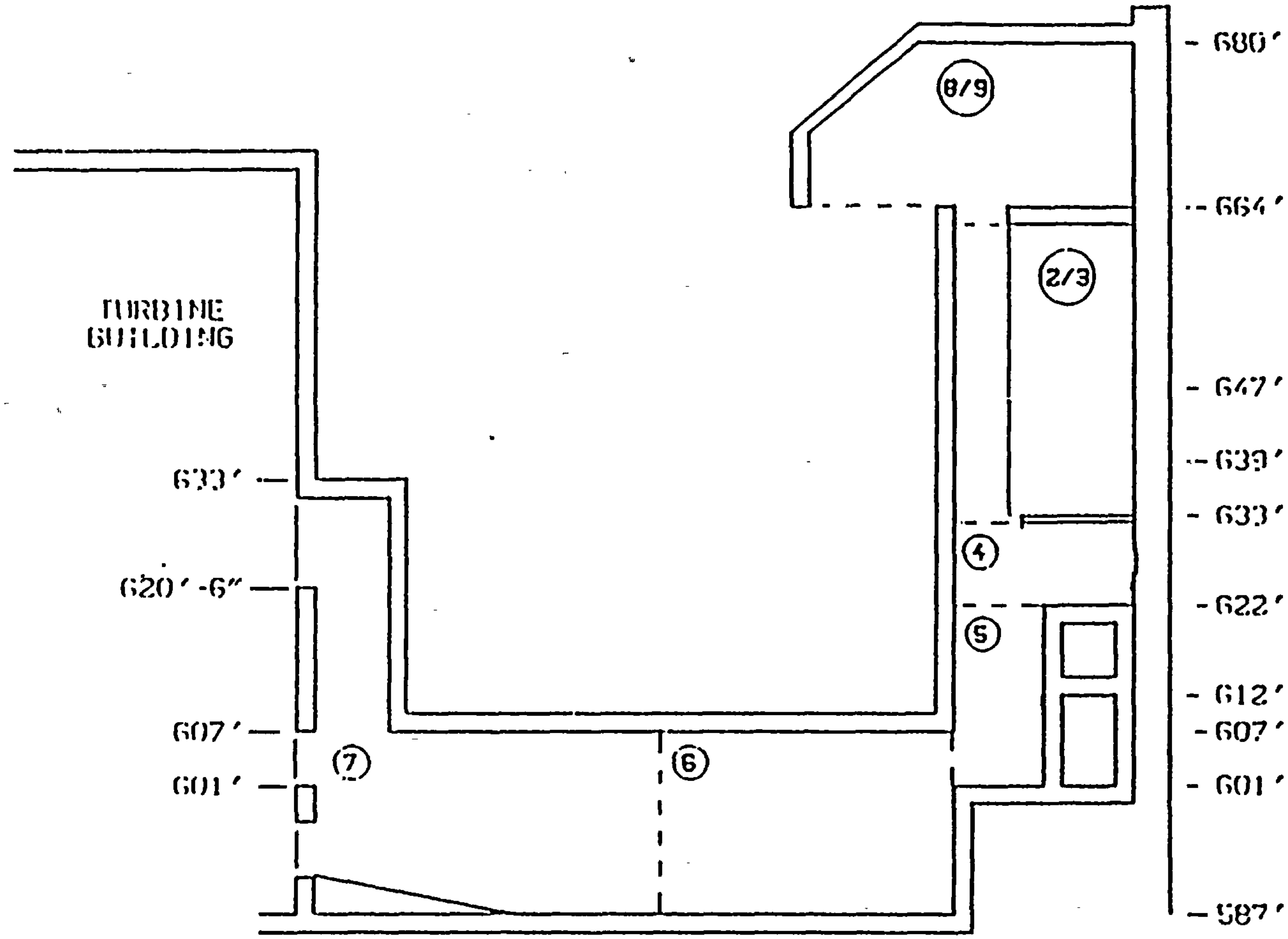


Figure 1: WEST MAIN STEAM ENCLOSURE AND ACCESSWAY

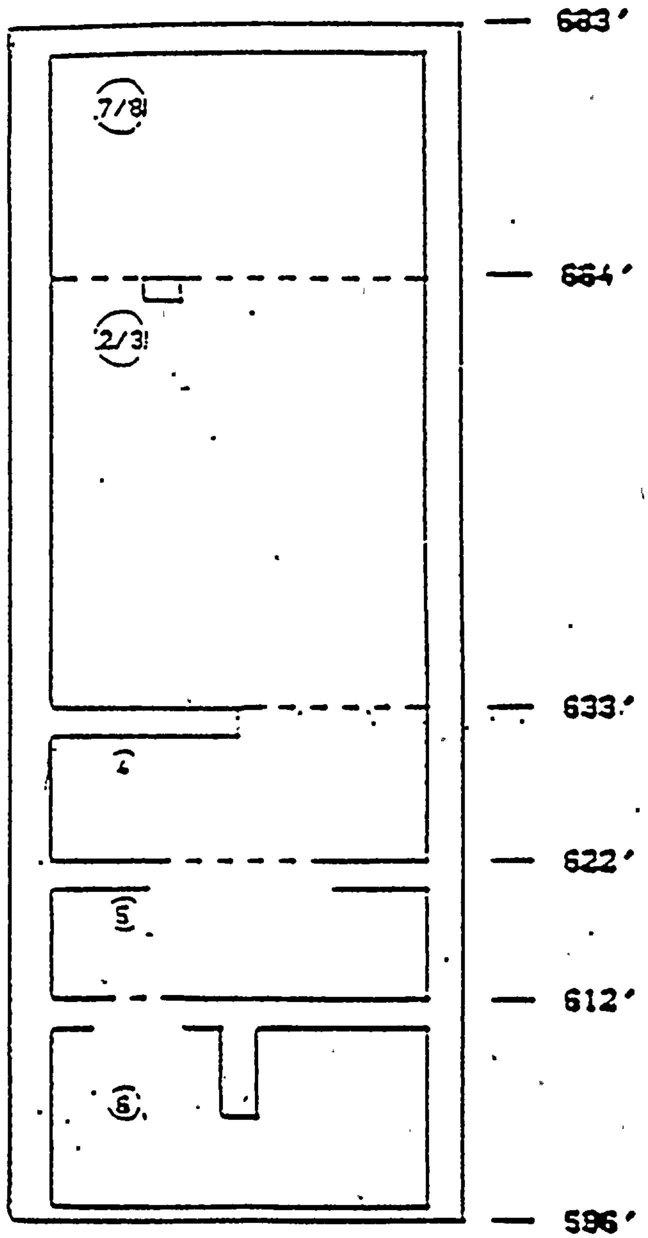


Figure 2: EAST MAIN STEAM ENCLOSURE

TABLE 1

AFFECTED EQUIPMENT

Equipment ID	Node	Qual Temp (°F)	Peak Compartment Temperature Old	Peak Compartment Temperature New	Max Surface Temperature
FFC-210,211	4 E	400	329	414	300
FFC-220,221,230,231	7 W	400	306	388	290
FFI-210,240	4 E	400	329	414	300
FFI-220,230	4 W	400	328	419	300
MPP-210,211,240,241	3 E	400	403	455	350
MPP-220,221,230,231	3 W	400	428	450	370
XSO-293,294,295,296	7 W	346	306	388	310
FMO-211,241	5 E	340	310	394	260
FMO-221,231	7 W	340	309	388	260
FMO-212,242	5 E	315	310	394	260
FMO-222,232	7 W	315	309	388	260

Attachment 4 to AEP:NRC:1140

WCAP-11902, Supplement 1

"Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 and 2 Licensing Report"

WCAP-11902
Supplement 1

RERATED POWER AND REVISED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNITS 1 & 2 LICENSING REPORT

September 1989

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Business Unit
P.O. BOX 355
Pittsburgh, Pennsylvania 15230

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF FIGURES

LIST OF ACRONYMS

ACRS	Advisory Committee for Reactor Safeguards
AEPSC	American Electric Power Service Corporation
AFRP	Auxiliary Feedwater Run Out Protection
AFWS	Auxiliary Feedwater System
ASME	American Society Mechanical Engineers
AVB	Antivibration Bar
BOP	Balance-of-Plant
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DECL	Double-Ended Cold Leg Break
DER	Double-Ended Rupture
DEHL	Double-Ended Hot Leg (Break)
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EFPY	Effective Full Power Years
ESF	Engineered Safety Features
EQ	Environmental Qualification
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
IMPCo	Indiana Michigan Power Company
ITDP	Improved Thermal Design Procedure
LBB	Leak-Before-Break
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
M&E	Mass and Energy
MSIV	Main Steam Isolation Valve
MSS	Main Steam System
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NRC	Nuclear Regulatory Commission

PCT	Peak Clad Temperatures
PCV	Pressure Control Valve
PORV	Power Operated Relief Valve
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RCCA	Rod Cluster Control Assembly
RCCS	Rod Cluster Control System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RPVIN	Reactor Pressure Vessel Inlet Nozzle
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SFPCS	Spent Fuel Pool Cooling System
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection System
VCT	Volume Control Tank
WCAP	Westinghouse Commercial Atomic Power
WDS	Waste Disposal System

EXECUTIVE SUMMARY

WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," was prepared in October, 1988, and was submitted to the Nuclear Regulatory Commission for the purpose of requesting permission to operate Cook Nuclear Plant Unit 1 within a range of primary temperatures from a vessel average temperature of 547°F to the original value of 567.8°F, at either of two values of primary pressure (2100 psia or 2250 psia). Reduced temperature and pressure operation was approved by an NRC Safety Evaluation Report dated June 9, 1989. This supplement to WCAP-11902 describes the results of the analyses and evaluations performed to support a Rerating Program for Cook Nuclear Plant Units 1 and 2. The ultimate goals of the Rerating Program are to support:

1. Operation of Cook Nuclear Plants Unit 1 and 2 within a range of primary temperatures from a vessel average temperature of 547°F to 578.7°F (Unit 1) and 581.3°F (Unit 2) at either of two values of primary pressure (2100 psia or 2250 psia).

2. Uprating the NSSS power levels of Units 1 and 2 to 3425 MWt and 3600 MWt, respectively.

Much of the effort to support these goals is already summarized in WCAP-11902. In the areas where the efforts performed for the first submittal are applicable for both Units 1 and 2, and incorporate power uprating, a brief confirming statement is made to that effect in this supplement to WCAP-11902. Where additional work was performed to support the rerating, but not previously acknowledged in WCAP-11902, the detailed results of that effort are documented in this supplement.

There are additional efforts necessary which were not performed as an integral part of the Rerating Program to support licensing of the uprating for Cook Nuclear Plant Unit 2. These efforts are the fuel-related analyses, which will be submitted to the NRC by AEPSC as part of the Cycle 8 fuel reload effort for

Unit 2, and a containment long term integrity analysis to support operation at 3600 MWt. (The current analysis supports operation at 3425 MWt.) Section S-1.3 describes the scope of the Rerating Program in detail, and lists the power capability assumptions for the NSSS.

The results of the evaluations and analyses performed demonstrate that no unreviewed safety question is involved, and that operation of Cook Nuclear Plant Unit 1 within the conditions analyzed for the Rerating Program remains in compliance with all originally applicable regulatory guides, codes and standards. Reanalysis and evaluation of FSAR accidents has supported these conclusions by demonstrating that the probability of occurrence, possibility of new accidents or margin of safety in any Technical Specification basis has not changed with respect to the original design. The associated changes to the Technical Specifications are provided with this submittal. For the areas analyzed and evaluated to date for Cook Nuclear Plant Unit 2, the conclusions presented above for Unit 1 are valid.

S-1.0 PROGRAM DESCRIPTION

S-1.1 DEFINITION OF GOALS

The goals as stated in Section 1.1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," remain valid for Unit 1. In addition, the efforts performed for the Rerating Program support licensing of a power uprating for Unit 1 to 3425 MWt NSSS over the range of conditions cited in Section S-2.1 (Cases 4 - 6).

For Unit 2, the goals are essentially the same as for Unit 1: All of the efforts performed under the Rerating Program support an "operating window" of primary temperatures described in Section S-2.1 (Cases 7 - 10), the two primary pressure values of 2100 and 2250 psia, and a maximum average steam generator tube plugging level of 10% (peak tube plugging level of 15%). However, with respect to thermal rating, the efforts performed verify the capability of operating if licensed with a power uprating for Unit 2 to 3600 MWt NSSS. In addition to the information contained in the rerating report, submittals will be made to the NRC for the core related accident analyses at 3600 MWt as part of the Cycle 8 reload analysis, and a long term containment integrity analysis verifying compliance with the requirements at the 3600 MWt level will be provided at a future date. The core related analyses can be determined by examining Table S-1.3-1. The analyses, except for the long term containment integrity analysis, labeled with an "N" in the Unit 2 columns are the core related analyses. The goal of the Unit 2 review is to obtain NRC review and approval of rerating analyses at the 3600 MWt level except the two aspects cited above. It is AEPSC's intention to select the desired operating temperature and pressure for Unit 2 within the range of the operating window conditions on a cycle to cycle basis.

S-1.2 APPLICABLE CRITERIA/QA

The quality assurance requirements defined in the Indiana Michigan Power Company (IMPCo) QA specification DCC QA 104 QCN, Quality Assurance Requirements Schedule/Quality Assurance Program Specifications for Safety-related Contracts and Service Orders, apply to this program. Equipment reviews, analyses, and evaluations have been performed in accordance with Westinghouse and industry codes, standards and regulatory requirements applicable to Cook Nuclear Plant Units 1 and 2 per the NSSS contracts and associated change notices as of the date of the Cook Nuclear Plant Rerating Contract (December, 1987).

For a description of the applicable licensing criteria refer to Section S-4.0.

S-1.3 SCOPE

Evaluations and/or analyses have been performed to assess the impact of the rerated power levels and the revised primary temperatures and pressures on the following NSSS systems and components for Donald C. Cook Nuclear Plant Units 1 & 2:

- Reactor vessel
- Reactor internals
- Steam generators
- Pressurizer
- Reactor coolant pumps
- Reactor coolant system piping and primary component supports
- Control rod drive mechanism
- Fluid and auxiliary systems
- Fuel

As discussed in Section 1.1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," several of the safety analyses/evaluations described in WCAP-11902 were performed to support the uprated power levels in addition to the revised temperatures and pressures described in Section S-2.1. Table S-1.3-1 lists the analyses/evaluations described in WCAP-11902 in addition to the analyses/evaluations which are described in this supplement to WCAP-11902 and what conditions they support. A "Y" in the table indicates the condition is supported where as an "N" indicates the condition is not supported.

The discussions of the following analyses/evaluations provided in this Supplement to WCAP-11902 are not based on the same analyses/evaluations as the discussions included in WCAP-11902.

- Steamline Break Containment Long Term Integrity
- Steamline Break Mass/Energy Releases
- Post-LOCA Hot Leg Recirculation

Radiological Consequences
Post-LOCA Hydrogen Production

The affected sections of the Technical Specifications for Cook Nuclear Plant Units 1 & 2 have been identified in Section S-3.13.

TABLE S-1.3-1

CONDITIONS SUPPORTED BY THE RERATING ANALYSES AND EVALUATIONS

Analysis/Evaluation	Section WCAP-11902/ Supplement	Revised Temperature and Pressure Unit 1	Revised Temperature and Pressure Unit 2	Rerated Power (NSSS MWt rating) Unit 1	Rerated Power (NSSS MWt rating) Unit 2
LOCA					
Large Break LOCA	3.1.1/ S-3.1.1	Y[1]	N	3425[1]	N
Small Break LOCA	3.1.2/ S-3.1.2	Y[2]	N	3600[2]	N
LOCA HYDRAULIC FORCES	3.2/ S-3.2	Y	Y[3]	3600	3600[3]
NON-LOCA					
Steamline Break Mass & Energy Releases (Inside Containment)	3.3.4.1/ S-3.3.4.1	Y[4]	Y[4]	3600[4]	3600[4]
Steamline Break Mass & Energy Releases (Outside Containment)	3.3.4.1/ S-3.3.4.1	Y[5]	Y[5]	3600[5]	3600[5]
Startup of an Inactive Reactor Coolant Loop	3.3.4.2/ S-3.3.4.2	Y	N	3425	N
Uncontrolled RCCA Withdrawal From A Subcritical Condition	3.3.4.3/ S-3.3.4.3	Y	N	3425	N
Uncontrolled Control Rod Assembly Bank At Power	3.3.4.4/ S-3.3.4.4	Y	N	3425	N
Rod Cluster Control Assembly Misalignment	3.3.4.5/ S-3.3.4.5	Y	N	3425	N
CVCS Malfunction	3.3.4.6/ S-3.3.4.6	Y	N	3425	N
Loss of Reactor Coolant Flow	3.3.4.7/ S-3.3.4.7	Y	N	3425	N
Locked Rotor	3.3.4.7/ S-3.3.4.7	Y	N	3425	N

TABLE S-1.3-1 (Cont'd)

CONDITIONS SUPPORTED BY THE RERATING ANALYSES AND EVALUATIONS

Analysis/Evaluation	Section WCAP-11902/ Supplement	Revised Temperature and Pressure		Rerated Power (NSSS MWt rating)	
		Unit 1	Unit 2	Unit 1	Unit 2
Loss of External Electrical Load	3.3.4.8/ S-3.3.4.8	Y	N	3425	N
Loss of Normal Feedwater Flow	3.3.4.9/ S-3.3.4.9	Y	N	3425	N
Excessive Heat Removal Malfunctions	3.3.4.10/ S-3.3.4.10	Y	N	3425	N
Excessive Increase In Secondary Steam Flow	3.3.4.11/ S-3.3.4.11	Y	N	3425	N
Loss of All AC Power to the Plant Auxiliaries	3.3.4.12/ S-3.3.4.12	Y	N	3425	N
Rupture of a Steam Pipe	3.3.4.13/ S-3.3.4.13	Y	N	3425	N
RCCA Ejection	3.3.4.14/ S-3.3.4.14	Y	N	3425	N
CONTAINMENT ANALYSIS					
Short-Term Containment Analysis	3.4.1/ S-3.4.1	Y	Y	3600	3600
Long-Term Containment Analysis					
Main Steamline Break Containment Integrity	3.4.2.1/ S-3.4.2.1	Y	Y	3600	3600
LOCA Containment Integrity	3.4.2.2/ S-3.4.2.2	Y[6]	Y[6]	3425[6]	3425[6]
STEAM GENERATOR TUBE RUPTURE	3.5/ S-3.5	Y	Y	3600	3600
POST-LOCA HOT LEG RECIRCULATION TIME	3.6/ S-3.6	Y[7]	N	3600[7]	N

TABLE S-1.3-1 (Cont'd)

CONDITIONS SUPPORTED BY THE RERATING ANALYSES AND EVALUATIONS

Analysis/Evaluation	Section WCAP-11902/ Supplement	Revised Temperature and Pressure		Rerated Power (NSSS MWt rating)	
		Unit 1	Unit 2	Unit 1	Unit 2
REACTOR CAVITY PRESSURE ANALYSIS	3.7/ S-3.7	Y	Y	3600	3600
RADIOLOGICAL ANALYSIS					
Large Break LOCA	3.8.1/ S-3.8.1	Y	Y	3600	3600
Fuel Handling Accident	3.8.2/ S-3.8.2	Y	Y	3600	3600
Steam Generator Tube Rupture	3.8.3/ S-3.8.3	Y	Y	3600	3600
POST-LOCA HYDROGEN PRODUCTION	3.9/ S-3.9	Y	Y	3600	3600
PRIMARY COMPONENTS EVALUATIONS					
Reactor Vessel	3.10.1/ S-3.10.1	Y	Y	3600	3600
Reactor Internals	3.10.2/ S-3.10.2	Y	Y[3]	3600	3600[3]
Steam Generators	3.10.3/ S-3.10.3	Y	Y	3425	3600
Pressurizer	3.10.4/ S-3.10.4	Y	Y	3600	3600
Reactor Coolant Pumps	3.10.5/ S-3.10.5	Y	Y	3600	3600
Reactor Coolant Piping and Supports	3.10.6/ S-3.10.6	Y	Y	3600	3600
Control Rod Drive Mechanism	3.10.7/ S-3.10.7	Y	Y	3600	3600

TABLE S-1.3-1 (Cont'd)

CONDITIONS SUPPORTED BY THE RERATING ANALYSES AND EVALUATIONS

Analysis/Evaluation	Section WCAP-11902/ Supplement	Revised Temperature and Pressure		Rerated Power (NSSS MWt rating)	
		Unit 1	Unit 2	Unit 1	Unit 2
FLUID AND AUXILIARY SYSTEMS EVALUATIONS					
Fluid Systems Evaluation	3.11.2.1/ S-3.11.2.1	Y	Y	3600	3600
Auxiliary Equipment Evaluation	3.11.2.2/ S-3.11.2.2	Y	Y	3600	3600
NSSS/Balance of Plant Interface	3.11.2.3/ S-3.11.2.3	Y	Y	3600	3600
FUEL STRUCTURAL EVALUATION					
Fuel Assembly	3.12/ S-3.12	Y	N	3600	N
Fuel Rod	3.12/ S-3.12	Y	N	3425	N

NOTES

1 With both RHR and HHSI crossties open, 3425 MWt NSSS power is supported. With either RHR or HHSI crossties closed, but not simultaneously, 3262 MWt NSSS power is supported.

2 The SBLOCA analysis supports a power level of 3600 MWt with either RHR or HHSI crossties closed, charging flow imbalance ≤ 10 gpm, and 10% pump degradation. With either RHR or HHSI crosstie closed, charging flow imbalance ≤ 25 gpm, and 10% pump degradation, 3262 MWt NSSS power is supported.

3 The analysis assumes Ẇ 17x17 Standard Fuel. Unit 2 currently (Fuel Cycle 7) contains ANF Fuel. The Cycle 8 and 9 reloads will be W 17x17 Vantage 5 Fuel.

4 The Reduced Temperature and Pressure evaluation (WCAP-11902, Section 3.3.4.1) is based on the FSAR analysis and a new analysis is presented in this supplement, Section S-3.3.4.1 for the Rerating Program.

TABLE S-1.3-1 (Cont'd)

CONDITIONS SUPPORTED BY THE RERATING ANALYSES AND EVALUATIONS

NOTES (Cont'd)

5 The evaluation documented in Section 3.3.4.1 of WCAP-11902 restricts Tavg ≤ 567.8°F and NSSS power ≤ 3262 MWt for Unit 1. The analysis documented in Section S-3.3.4.1 of this supplement supports the full operating window and NSSS power ≤ 3600 MWt.

6 The analysis supports RHR or SI crossties closed, but not both crossties closed.

7 The analysis documented in WCAP-11902 only supports revised temperatures and pressures. The analysis documented in this supplement supports rerating in addition to revised temperatures and pressures.

S-2.0 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

S-2.1 DESIGN POWER CAPABILITY PARAMETERS

S-2.1.1 Discussion of Parameters

Section 2.1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," (Volume 1) describes the parameters which were used as the basis for the evaluations and analyses performed to support reduced temperature and pressure operation for the Donald C. Cook Nuclear Plant Unit 1. The Rerating Program (of which the reduced temperature and pressure program is a part) addressed a broader spectrum of parameters, which included thermal power uprating. As Section S-1.3 describes, many of the analyses and evaluations performed for the rerating incorporate a power level of 3600 MWt NSSS, and the associated primary temperatures and pressures. Certain of the safety analyses performed for the rerating support lower power levels of 3262 MWt or 3425 MWt, as delineated in Section S-1.3. Accordingly, several sets of design power capability parameters are presented in Table S-2.1-1. (All of the cases below with the exception of Case 2 are obtained from Reference 1; Case 2, which comprises the current licensed parameters for Cook Nuclear Plant Unit 2, was obtained using input from Reference 2.) It should be noted here that the philosophy of the rerating program was to perform each evaluation or analysis using the most conservative parameter case(s) necessary to support operation of the Cook units at the highest power level possible. In some areas, (e.g. LOCA hydraulic forces) it was possible to select one parameter case which clearly bounded all of the rerating conditions. In other areas, (e.g. NSSS components structural evaluations), it was necessary to examine the most limiting parameters from several cases in order to support the entire range of rerating parameters.

A brief description of each set of parameters is provided below:

Case 1: These are the original design power capability parameters for Unit 1 and are shown for comparison with the revised parameters. The NSSS power level of 3250 MWt does not account for reactor coolant pump heat; at the time that Unit 1 was designed, it was the custom to indicate only the core power level value. (These parameters were also presented as Case 1 in Table 2.1-1, Volume 1 of WCAP-11902.)

Case 2: These are the currently licensed design power capability parameters for Unit 2 and are shown for comparison with the revised parameters.

Case 3: These parameters incorporate a core power level of 3250 MWt, an NSSS power level of 3262 MWt (which includes 12 MWt of reactor coolant pump heat), an average steam generator tube plugging level of 15%, primary pressures of either 2250 psia or 2100 psia, and an upper bound vessel average temperature of 576.3°F. This parameter case was used as the basis for the Large Break LOCA analysis, with the RHR crosstie closed.

Case 4: These parameters incorporate a core power level of 3413 MWt, an NSSS power level of 3425 MWt, an average steam generator tube plugging level of 10%, primary pressures of either 2250 psia or 2100 psia, and a lower bound vessel average temperature of 547°F (the lowest vessel average temperature considered for the rerating program). These parameters were used for selected non-LOCA safety analyses (Loss of Normal Feedwater, Loss of Offsite Power to Station Auxiliaries), where low primary temperatures were limiting.

Case 5: These parameters incorporate the same features as Case 4, except that the primary temperatures and resulting secondary parameters incorporate an upper bound vessel average temperature of

578.7°F. This case was used as the basis for selected non-LOCA safety analyses (DNB-related transients plus Locked Rotor, Rod Ejection and Boron Dilution), where high primary temperatures were limiting, as well as the Long-Term Containment Analysis (following LOCA), which is documented in WCAP-11908. (It should be noted that the L/T Containment Analysis also assumed zero steam generator tube plugging, which is conservative since it maximizes heat transfer.)

Case 6: These parameters incorporate the same features as Case 5, except that the average steam generator tube plugging level is 15%. This case was used as the basis for the Large Break LOCA analysis, with the RHR crosstie open.

Case 7: These parameters incorporate a core power level of 3588 MWt, an NSSS power level of 3600 MWt, an average steam generator tube plugging level of 10%, primary pressures of either 2250 psia or 2100 psia, and a lower bound vessel average temperature of 547°F. These parameters were evaluated for short-term containment effects, where low primary temperatures were limiting.

Case 8: These parameters incorporate the same features as Case 7, except that the primary temperatures and resulting secondary parameters incorporate an upper bound vessel average temperature of 581.3°F (the highest vessel average temperature considered for the rerating). These parameters were evaluated for short-term containment effects, where high primary temperatures were limiting. In addition, mass/energy releases for steam line break used these parameters as a basis.

Case 9: These parameters incorporate the same features as Case 7, except that the average steam generator tube plugging level is 15%.

This case was used as the basis for the Small Break LOCA analysis, SGTR (break flow), and the LOCA hydraulic forcing functions evaluation.

Case 10: These parameters incorporate the same features as Case 8, except that the average steam generator tube plugging level is 15%.

S-2.1.2 References

1. Letter AEP-88-232, "RCS Flow Definitions," dated May 2, 1988, by H. C. Walls.

2. Donald C. Cook Nuclear Plant Units 1 & 2 Final Safety Analysis Report, Chapter 14, Table 14.0.2-2 (Unit 2).

TABLE S-2.1-1

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Unit 1, Original) Case 1	(Unit 2, Current) Case 2
NSSS Power, MWt	3250	3423
Core Power, MWt	3250	3411
RCS Flow,(gpm/loop)*	88,500	-***
Minimum Measured Flow, (total gpm)**	366,400	364,960
RCS Temperatures, °F		
Core Outlet	602.0	-
Vessel Outlet	599.3	-
Core Average	570.5	575.5
Vessel Average	567.8	574.1
Vessel/Core Inlet	536.3	-
Steam Generator Outlet	536.3	-
Zero Load	547.0	547.0
RCS Pressure, psia	2250	2250
Steam Pressure, psia	758	794.4
Steam Flow, (10^6 lb/hr.tot.)	14.12	14.6
Feedwater Temperature, °F	434.8	423.4
% SG Tube Plugging	0	10% avg./ 15% peak

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

***Flow values supplied in FSAR for Unit 2 are 141.3 x 10^6 lb/hr for vessel coolant flow, and 134.9 x 10^6 lb/hr for active core flow.

Note: Dashes in Case 2 indicate information which was not contained in the FSAR, and is therefore information which is unavailable to Westinghouse.

TABLE S-2.1-1 (Cont'd)

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Revised) Case 3	(Revised) Case 4	(Revised) Case 5	(Revised) Case 6
NSSS Power, MWt	3262	3425	3425	3425
Core Power, MWt	3250	3413	3413	3413
RCS Flow, (gpm/loop)*	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)**	366,400	366,400	366,400	366,400
RCS Temperatures, °F				
Core Outlet	610.1	583.6	614.0	613.9
Vessel Outlet	607.5	580.7	611.2	611.2
Core Average	579.2	549.7	581.8	581.8
Vessel Average	576.3	547.0	578.7	578.7
Vessel/Core Inlet	545.2	513.3	546.2	546.2
Steam Generator Outlet	545.0	513.1	546.0	546.0
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	807	603	820	806
Steam Flow, (10^6 lb/hr.tot.)	14.20	14.98	15.07	15.06
Feedwater Temperature, °F	434.8	442.0	442.0	442.0
% SG Tube Plugging (average)	15	10	10	15

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

TABLE S-2.1-1 (Cont'd)

COOK NUCLEAR PLANT UNITS 1 AND 2
DESIGN POWER CAPABILITY PARAMETERS FOR RERATING PROGRAM

Parameter	(Revised) Case 7	(Revised) Case 8	(Revised) Case 9	(Revised) Case 10
Power, MWt	3600	3600	3600	3600
Core Power, MWt	3588	3588	3588	3588
RCS Flow, (gpm/loop)*	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)**	366,400	366,400	366,400	366,400
RCS Temperatures, °F				
Core Outlet	585.4	618.0	585.4	618.1
Vessel Outlet	582.3	615.2	582.3	615.2
Core Average	549.9	584.6	549.9	584.7
Vessel Average	547.0	581.3	547.0	581.3
Vessel/Core Inlet	511.7	547.3	511.7	547.4
Steam Generator Outlet	511.4	547.1	511.4	547.2
Zero Load	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250 or 2100	2250 or 2100	2250 or 2100	2250 or 2100
Steam Pressure, psia	587	820	576	806
Steam Flow, (10^6 lb/hr.tot.)	15.90	16.00	15.89	15.99
Feedwater Temperature, °F	449.0	449.0	449.0	449.0
% SG Tube Plugging (average)	10	10	15	15

Flow Definitions:

*RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based on this flow.

**Minimum Measured Flow - The flow specified in the Tech. Specs. which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure.

S-2.2 NSSS DESIGN TRANSIENTS

Revised NSSS design transients were generated as part of the Rerating and Revised Temperature and Pressure Program to bound the revised operating conditions shown in Table S-2.1-1. These revised transients were transmitted to the systems and equipment designers for use in their evaluations of the effects of the revised conditions on their NSSS systems and components.

The most significant revisions to the NSSS design transients are as follows: For those operating conditions where primary temperatures are below the original operating temperatures (corresponding to Cases 7 or 9 in Section S-2.1), cold leg temperature, steam temperature, and pressurizer surge nozzle and spray nozzle temperature, transient swings from zero load to full load are increased. At the upper bound temperature conditions (corresponding to Cases 8 or 10 in Section S-2.1), hot leg temperature swings from zero load to full load are increased. The modifications to the design transients reflect the extremes in primary temperatures and primary pressures. In addition to temperature and pressure extremes, the analyses also considered the number of transients at these conditions in order that the equipment analysts assess the most limiting transients for each component or system. In this way, the systems and component evaluations address the upper and lower bound temperatures and pressures, as well as the range of conditions in between the extremes.

S-2.3 CONTROL SYSTEM SETPOINTS

The control system evaluations performed for the program reflect the objective of optimizing the control parameters, primarily with respect to two aspects of plant behavior: stability of the control systems and operating margins to the various reactor protection system trips.

Since the flexibility exists to adjust the full-load vessel average temperature and primary pressure as necessary on a cycle-to-cycle basis, control system setpoints will be selected for each fuel cycle to limit the amplitude and frequency of oscillation of plant parameters within acceptable values, while maintaining margin to reactor protection system trips.

S-3.0 SAFETY EVALUATIONS/ANALYSES

S-3.1 LOCA (LARGE BREAK AND SMALL BREAK)

S-3.1.1 Large Break LOCA

The analysis of the large break LOCA is discussed in Section 3.1.1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." This analysis supports operation of the Donald C. Cook Nuclear Plant Unit 1 at a reactor power level of 3413 MWt over a range of operating temperatures between 611.2°F and 580.7°F in the hot legs and 546.2°F and 513.3°F in the cold legs (547°F $\leq$ Tavg $\leq$ 578.7°F). Initial RCS pressure was also varied to justify plant operation at 2100 and 2250 psia. Therefore, the conclusions of WCAP-11902 are valid for Cook Nuclear Plant Unit 1 at the rerated core power level of 3413 MWt. The analysis also supports 10% safety injection flow degradation and 15% uniform steam generator tube plugging. The LBLOCA analysis was performed for 15% tube plugging to support the peak plugging level of 15%. In general, the other analyses support only an average plugging level of 10%.

The large break LOCA analysis also considered plant operation at a reduced core power level of 3250 MWt (Case 3 in Table S-2.1-1) with the RHR cross tie valve closed. The reduction in reactor power was necessary to offset the reduction in safety injection flow due to the closed RHR cross-tie valve.

The charging flow imbalance assumption of 25 gpm was included in all large break LOCA cases. This was possible due to the relative insensitivity of the large break LOCA results to changes in charging pump performance.

S-3.1.2 Small Break LOCA

The analysis of the small break LOCA is discussed in Section 3.1.2 of WCAP-11902. This analysis supports operation of the Donald C. Cook Nuclear Plant Unit 1 at a reactor power level of 3588 MWt (This is conservative with respect to the Unit 1 uprated reactor power of 3413 MWt). The analysis was performed assuming a range of operating temperatures in order to justify plant operation between a T_{avg} program setpoint of 547°F and 581.3°F at RCS pressures of 2100 psia and 2250 psia. A spectrum of cold leg breaks were analyzed at the limiting RCS pressure and temperature conditions and the limiting break size was analyzed at other points in the pressure and temperature range. Therefore, the conclusions of WCAP-11902, Section 3.1.2 are valid for Cook Nuclear Plant Unit 1 at the rerated core power level of 3413 MWt.

An evaluation was performed to determine the effects of operation with either RHR or HHSI crossties closed and a charging flow imbalance as high as 25 gpm on the small break LOCA analysis. This evaluation limits the core power to 3250 MWt when either the RHR or HHSI cross-ties are closed. This evaluation was submitted to the NRC June 5, 1989 as an addition to the reduced temperature and pressure submittals and was approved June 9, 1989, with that submittal.

S-3.2 LOCA HYDRAULIC FORCING FUNCTIONS

The analysis of the LOCA Hydraulic Forces is discussed in Section 3.2 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." The purpose of this analysis is to generate the forcing functions that occur on RCS components as a result of a postulated loss of coolant accident. This analysis was performed assuming the most limiting parameters to support operation of the Donald C. Cook Nuclear Plant Units 1 & 2 at the respective rerated reactor power levels of 3413 MWt and 3588 MWt, the revised temperatures listed in Table S-2.1-1, and either an RCS pressure of 2100 psia or 2250 psia. Credit was taken for the "leak-before-break" exemption to limit the break size to the accumulator line. The smaller break size allows the component loads as a result of a LOCA to be bounded by the original structural analysis.

The parameters chosen as most limiting incorporate a conservatively high reactor power level with respect to Cook Nuclear Plant Units 1 & 2 (3588 MWt), as well as the upper bound primary pressure (2250 psia) and the lower bound vessel inlet temperature (511.7°F). The initial primary system pressure assumption maximizes the pressure differential utilized for the blowdown of the system. The lower temperature results in a higher sonic velocity for the decompression wave which propagates through the loop piping towards the vessel internals. The combination of high decompression wave velocity and pressure differential cause a pressure pulse which results in the largest LOCA hydraulic forcing functions for the accumulator line break. These forcing functions are not as sensitive to power level as the parameters discussed above, so the bounding maximum power level was chosen. Therefore, the conclusions of WCAP-11902, Section 3.2 are valid for Cook Nuclear Plant Units 1 & 2 at their respective rerated core power levels of 3413 MWt and 3600 MWt.

S-3.3 NON-LOCA SAFETY EVALUATION

S-3.3.1 Introduction

This section evaluates the effects of the Cook Nuclear Plant Rerating Program on the non-LOCA transients. The non-LOCA safety evaluation provided within is applicable only for Unit 1, with the exception of the steamline break mass/energy releases (inside and outside containment). The effort performed is to support Unit 1 operation with an uprated core power of 3413 MWt in the range of reactor vessel average temperatures between 547°F and 578.7°F at primary pressure values of 2100 psia or 2250 psia. Table S-2.1-1 (Cases 4 and 5) presents the range of conditions possible for the rerating of Unit 1. The steamline break mass/energy release analyses are performed to support the potential future Unit 1 rerating as well as to bound a potential rerating of Unit 2. Table S-2.1-1 (Cases 7 and 8) presents the range of conditions possible for the future rerating of Unit 2. In addition, the evaluation performed is to support a maximum average steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15%.

The following non-LOCA safety evaluation also supports the change and/or relaxation of certain plant parameters to provide Unit 1 with increased operating margin and flexibility. Included in the non-LOCA safety evaluation are:

Increased Most Negative Moderator Temperature Coefficient (MTC) (Tech Spec 3.1.1.4b)
Degraded ECCS Charging Pump Flow (Tech Spec 4.5.2f)
Increased Main Steamline Isolation Valve (MSIV) Closure Time (Tech Spec 4.7.1.5b and Tech Spec Table 3.3-5 items 5h, 6h, & 7c)

The evaluation conservatively assumes 0 ppm boron concentration in the Boron Injection Tank (BIT).

The evaluation also supports a change to the steam generator water level program. The existing level program is a ramp function from 33% narrow range span (NRS) to 44% NRS from 0% power to 20% power and a constant level at 44% NRS between 20% power and 100% power. The proposed steam generator water level program is a constant level at 44% NRS between 0% power and 100% power.

The corresponding updates to the Unit 1 Technical Specifications are presented in Section S-3.13.

The efforts undertaken for the Unit 1 reduced temperature and pressure operation non-LOCA safety evaluation as described in Section 3.3, Volume 1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR THE DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," were performed to support Unit 1 operation with a core power of 3250 MWt over the range of vessel average temperatures from 547.0°F to 576.3°F at primary pressure values of 2100 psia or 2250 psia (See WCAP-11902, Table 3.3-1, Cases 2 and 3). However, the steamline break mass/energy releases outside containment evaluation (Section 3.3.4.1 of WCAP-11902) restricted the full power vessel average temperature to no greater than 567.8°F. The safety evaluation (WCAP-11902) also supported a maximum average steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15%.

The effort to support the reduced temperature and pressure operation for Unit 1 consisted of evaluations and analyses (Section 3.3 of WCAP-11902). For the non-LOCA events which required analyses to support the reduced temperature and pressure operation, the analyses were performed to bound the range of conditions possible for the rerating of Unit 1. These analyses also considered the relaxation of the plant parameters listed above. No additional effort is required to support the rerating conditions of Unit 1 for those non-LOCA events analyzed in WCAP-11902.

The non-LOCA events which were evaluated to support the reduced temperature and pressure operation are the startup of an inactive loop event and the steamline break mass/energy releases (inside and outside containment).

Although the startup of an inactive loop event was only evaluated in Section 3.3.4.2 of WCAP-11902, no additional effort is necessary for this event as discussed in Section S-3.3.4.2.

However, the steamline break mass/energy events do require additional analyses to support the range of conditions possible for the rerating of Unit 1 and the relaxation of the plant parameters listed above. These analyses are presented below. Table S-3.3-1 presents the applicable non-LOCA transients for Unit 1.

In summary, the non-LOCA safety evaluation presented in the following discussion to support the Cook Unit 1 Rerating Program bounds the range of conditions of the Unit 1 rerating (Cases 4 and 5 specified in Table S-2.1-1) and the Unit 1 reduced temperature and pressure operation (Cases 2 and 3 specified in WCAP-11902, Table 3.3-1). The following safety evaluation also supports the relaxation of the plant parameters listed above. In addition, the steamline break mass/energy release (inside and outside containment) analyses address the Unit 1 rerating as well as position Unit 2 for a potential rerating.

S-3.3.2 Reactor Protection System (RPS) and Engineered Safety Features (ESF) Setpoints Assumed in Evaluation

Certain reactor trip and engineered safeguards features setpoints were revised to provide adequate operating margin for the reduced temperature and pressure operation. The revised RPS setpoints for the non-LOCA safety evaluation included only the overtemperature ΔT (OTΔT) and the overpower ΔT (OPΔT) reactor trips. Table 3.3-2 of WCAP-11902 presents the limiting reactor trip setpoints assumed in the analyses performed to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. This table is repeated as Table S-3.3-2 in this supplement.

Section 3.14 of WCAP-11902 provided the OTΔT and OPΔT setpoints to be included in the Technical Specification updates for the reduced temperature and pressure operation. To support the range of conditions possible for the

rerating of Unit 1, the OTΔT and OPΔT setpoints need to be updated to encompass operation up to a full power Tavg of 578.7°F (T' and T"). Although the OTΔT and OPΔT setpoints developed in Section 3.3.2.1 of WCAP-11902 are applicable for the entire range of conditions of the Unit 1 rerating, the full power Tavg value (T' and T") of the equations were limited to 567.8°F. The steamline break mass/energy release outside containment evaluation limited the full power Tavg to 567.8°F. Section S-3.13 provides the revised equations (updated T' and T" values) for the OTΔT and OPΔT setpoints.

The revised ESF setpoint changes for the reduced temperature and pressure operation included only the low steamline pressure value of the high-high steamline flow coincident with low steamline pressure logic. (See Section 3.3.2.2 of WCAP-11902 for a discussion of the change to the low steamline pressure setpoint and Section 3.14 for the revised Technical Specification value.) No change is required to this setpoint to support the Unit 1 rerating. To encompass the higher steam flow associated with the uprated power of the Unit 1 rerating, the high steam flow setpoint of the high-high steamline flow coincident with low steamline pressure logic needs to be updated. Section S-3.13 presents the updated setpoint to be included in the Unit 1 Technical Specifications.

S-3.3.3 Methodology

The Unit 1 non-LOCA safety evaluation for the rerating of Unit 1 was performed using current Westinghouse methodology and computer codes. Table S-3.3-1 presents the applicable non-LOCA transients for Unit 1 which were examined in the evaluation. The majority of analyses are presented in detail in Reference 1 (WCAP-11902) and are summarized below. The steamline break mass/energy releases (inside and outside containment) analyses are presented below.

S-3.3.3.1 Initial Conditions

Section 3.3.3.1 of WCAP-11902 presents the discussion of the initial conditions assumed in the safety evaluation to support the reduced temperature and pressure operation. Table 3.3-4 of WCAP-11902 summarizes the initial conditions and computer codes used in the accident analyses, which were performed to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1.

The steady state errors for the accidents that are not DNB limited presented in Section 3.3.3.1 of WCAP-11902 were employed in the analyses of the steamline break mass/energy releases (inside and outside containment). These maximum steady state errors are as follows:

A.	Core Power	+ 2% calorimetric error allowance
B.	Average RCS Temperature	± 4.5°F controller deadband and measurement error allowance
C.	Pressurizer Pressure	± 35 psi - steady state fluctuations and measurement error allowance (See Note 1 presented after the non-LOCA safety evaluation conclusion, Section S-3.3.5)
D.	Reactor Flow	Thermal Design Flow (354,000 gpm)

Table 3.3-4 summarizes initial conditions and computer codes used in the accident analysis documented in WCAP-11902. This table is repeated as Table S-3.3-3 in this supplement.

S-3.3.3.2 Computer Codes Utilized

The steamline break mass/energy releases were calculated using the LOFTRAN computer code (Reference 2). Section 3.3.3.2 of WCAP-11902 presents summaries of the principal computer codes used in the transient analyses for the reduced temperature and pressure operation.

S-3.3.4 Non-LOCA Safety Evaluation

The following sections contain the descriptions of the impact of the rerating of Unit 1 and the relaxation of the previously mentioned plant parameters on the applicable non-LOCA transients. The steamline break mass/energy release analyses are presented in Section S-3.3.4.1. The remaining sections describe the transients requiring no additional effort beyond the effort documented in WCAP-11902. In all cases the appropriate FSAR acceptance criteria are satisfied. It should be noted that the evaluation supports a steam generator average tube plugging level of 10%, with peak plugging level of 15%, provided the minimum measured flow of 366,400 gpm (plant total) is met and the nominal RCS temperatures do not exceed the range of temperatures presented in Table S-2.1-1, Cases 4 and 5 for the Unit 1 rerating operation or in WCAP-11902, Table 3.3-1, Cases 2 and 3 for the Unit 1 reduced temperature and pressure operation.

S-3.3.4.1 Steamline Break Mass/Energy Releases

This section will discuss the analyses of the steamline break event to determine the mass and energy releases inside containment and the superheated mass and energy releases outside containment for the Cook Rerating Program. The analyses were performed to support the range of conditions possible for the rerating of Unit 1 as well as to position Unit 2 for a potential rerating. The analyses also consider the relaxation of certain plant parameters (Section S-3.3-1).

Steamline Break Mass/Energy Releases Inside Containment

The current mass/energy releases for the inside containment analysis is based on work performed for Unit 2, which is applicable for Unit 1. The calculation of the mass/energy release following a steamline break is described in the Cook Unit 2 FSAR Section 14.1.5. The steamline break mass/energy releases were recalculated to address the rerating of both Units and the relaxation of the plant parameters described in Section S-3.3.1.

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make it difficult to reasonably determine the single "worst case" for both containment pressure and temperature evaluations following a steambreak. The FSAR analysis determined that the limiting scenario of the steambreak cases analyzed for the containment response evaluation were a break size of 0.942 ft^2 occurring at 30% power for the split rupture scenario and a break size of 4.6 ft^2 occurring at full power for the double-ended rupture scenario. (The 30% power split break case was slightly more limiting.) However, it is difficult to conclude if these FSAR cases remain bounding for the range of conditions possible for the reratings of both Units.

Adding to the difficulty in determining the effect of the rerating conditions are the plant parameters changes incorporated into the Cook Rerating Program. The potential changes of certain plant parameters (i.e, relaxed most negative MTC limit, degraded ECCS performance, increased MSIV closure time, and 0 ppm BIT boron concentration requirement) are penalties in the calculation of mass/energy releases. It is not readily apparent as to the total impact of the combination of these changes. As such, a series of steamline breaks, consistent with the cases presented in the FSAR, were analyzed to determine the containment response to a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break sizes.

Method of Analysis

The LOFTRAN computer code (Reference 2) was used to calculate the break flows and enthalpies of the release through the steambreak. Blowdown mass/energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant thick metal heat storage, and reverse steam generator heat transfer.

A bounding analysis was performed to address the range of conditions possible for the potential Unit 1 rerating and the potential Unit 2 rerating. The assumptions on the initial conditions are taken to maximize the mass and total energy released. The higher primary temperatures along with the higher uprated power level associated with the Unit 2 rerating parameters are conservative for the mass/energy release calculations. The upper bound temperature of Table S-2.1-1, Case 8 was used. Since the mass blowdown rate is dependent on steam pressure and the steam pressure is less for the lower bound temperature case, the steam pressure of the upper bound temperature case is limiting for the range of operating conditions possible for the reratings of Unit 1 and Unit 2.

The functions which actuate safety injection and steamline isolation during a steamline rupture event are commonly referred to as the Steamline Break Protection System. A plant's steamline break protection system design can have a large effect on steamline break results. The steamline break protection system designs for Unit 1 and Unit 2 are different. Unit 1's design is referred to as an "OLD" steamline break protection system design. Unit 2's design is referred to as a "HYBRID" steamline break protection system design. The two systems have the following characteristics:

Unit 1 - "OLD" Steamline Break Protection

Safety Injection Signals

1. High-high steam flow coincident with low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of three differential pressure signals between a steam line and the remaining steam lines

4. Two out of three low pressurizer pressure signals

5. Two out of three hi containment pressure signals

Steamline Isolation Signals

1. High-high steam flow coincident with low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of four hi-hi containment pressure signals

Unit 2 - "HYBRID" Steamline Break Protection

Safety Injection Signals

1. Low steamline pressure (two out of four lines)

2. Two out of three differential pressure signals between a steam line and the remaining steam lines

3. Two out of three low pressurizer pressure signals

4. Two out of three hi containment pressure signals

Steamline Isolation Signals

1. Low steamline pressure (two out of four lines)

2. High-high steam flow coincident with low-low Tavg (two out of four lines)

3. Two out of four hi-hi containment pressure signals

The only differences between the Unit 1 and Unit 2 designs is the actuations from a high-high steam flow and low-low Tavg signal and the logic associated with the low steamline pressure signal required to actuate safety injection and steamline isolation. For Unit 1, a high-high steam flow coincident with low-low Tavg signal actuates both safety injection and steamline isolation. For Unit 2, a high-high steam flow coincident with low-low Tavg signal actuates only steamline isolation. However, the difference is not significant for the calculation of the mass/energy releases since the analysis does not take credit for any ESF actuations on a high-high steam flow coincident with low-low Tavg signal.

Unit 1's design requires a coincidence between the low steamline pressure and high-high steam flow for protection actuation. Unit 2's design only requires the low steamline pressure signal for protection actuation; no coincidence with steam flow is required.

The coincidence logic required for safety injection initiation and steamline isolation on high-high steam flow and low steam pressure for Unit 1 is more limiting for the calculation of mass/energy releases inside containment than Unit 2's design. Actuation of safety injection and steamline isolation will limit the mass/energy released to the containment. Delaying the safeguards initiation will result in a conservative calculation of the mass/energy

releases for the containment pressure and temperature evaluation. The coincidence requirement for high-high steam flow with low steam pressure of the Unit 1 design increases the likelihood that safeguards initiation might be delayed compared to Unit 2's design where only a low steam pressure signal is required. In the case where the coincidence logic prohibits safety injection and steamline isolation on high-high steam flow with low steam pressure, one of the other signals must be received before the safeguards are initiated. As such, the Unit 1 steamline break protection system design was assumed in this bounding analysis for the calculation of the mass/energy releases inside containment.

Assumptions

A series of steamline breaks were analyzed to determine the most severe break condition for the containment temperature and pressure response. The following assumptions were used in the analysis:

a. Double-ended pipe breaks were assumed to occur at the nozzle of one steam generator and also downstream of the flow restrictor. Split ruptures were assumed to occur at the nozzle of one steam generator.

b. The blowdown is assumed to be dry saturated steam.

c. As discussed above, the Unit 1 steamline break protection system design is assumed. However, credit was not taken for safeguards actuation on high steam line differential pressure or high-high steam flow coincident with low-low Tavg.

d. Steamline isolation is assumed complete 11 seconds after the setpoint is reached for either high-high steam flow coincident with low steam pressure or hi-hi containment pressure. The isolation time allows 8 seconds for valve closure plus 3 seconds for electronic delays and signal processing. The total delay time for steamline isolation of 11 seconds is assumed to support the relaxation of the main steam isolation valve (MSIV) closure time.

e. 4.6 ft^2 and 1.4 ft^2 double-ended pipe breaks were evaluated at 102, 70, 30, and zero percent power levels.

f. Split pipe ruptures were evaluated at 0.86 ft^2, 102% power; 0.908 ft^2, 70% power; 0.942 ft^2, 30% power; and 0.4 ft^2, hot shutdown.

These split break sizes for each power level were modeled because they reflect the largest breaks for which ESF actuations (i.e., steamline isolation, feedwater isolation, and safety injection) must be generated by high containment pressure trips. The high-high steam flow coincident with low steam pressure is not reached for these break sizes or smaller break sizes. (Reference 5)

g. Failure of a main steam isolation valve, failure of a feedwater isolation valve or main feed pump trip, and failure of auxiliary feedwater runout control were considered. Two cases for each break size and power level scenario were evaluated with one case modeling the MSIV failure and the other case modeling the AFW runout control failure. Each case assumed conservative main feedwater addition to bound the feedwater isolation valve or main feed pump trip failure.

h. The auxiliary feedwater system is manually re-aligned by the operator after 10 minutes.

i. A shutdown margin of 1.3% $\Delta k/k$ is assumed. This assumption includes added conservatism with respect to the Unit 1 end-of-life shutdown margin requirement of 1.6% $\Delta k/k$ at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. The Unit 1 end-of-life shutdown margin requirement was used as the basis for this assumption since it is more limiting than the existing Unit 2 shutdown margin requirement.

j. A moderator density coefficient of 0.54 Δk/gm/cc is assumed to support the relaxation of the most negative moderator temperature coefficient limit.

k. Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of the following systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, 3) the high head (intermediate head) safety injection system, and 4) the charging safety injection system. Only the charging safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

The modeling of the safety injection system in LOFTRAN is described in Reference 2. Figure 3.3-52 of WCAP-11902 presents the safety injection flow rates as a function of RCS pressure assumed in the analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold legs. The safety injection flows assumed in this analysis take into account the degradation of the ECCS charging pump performance. No credit has been taken for any borated water that might exists in the injection lines, which must be swept from the lines downstream of the boron injection tank isolation valves prior to the delivery of boric acid to the reactor coolant loops. For this analysis, a boron concentration of 0 ppm for the boron injection tank is assumed.

After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the safety injection charging pump starts. In 27 seconds, the valves are assumed to be in their final position (VCT charging pump suction valve has closed following opening of RWST charging pump suction valve) and the pump is assumed to be at full speed and to draw suction from the RWST. The volume containing the low concentration borated water is swept into the core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

l. For the at-power cases, reactor trip is available by safety injection signal, overpower protection signal (high neutron flux reactor trip or OPΔT reactor trip), and low pressurizer pressure reactor trip signal.

m. For reactor coolant pump (RCP) operation, offsite power is assumed available. Continued operation of the reactor coolant pumps maximizes the energy transferred from the reactor coolant system to the steam generators.

n. No steam generator tube plugging is assumed to maximize the heat transfer characteristics.

Single Failure Effects

a. Failure of a main steam isolation valve (MSIV) increases the volume of steam piping which is not isolated from the break. When all valves operate, the piping volume capable of blowing down is located between the steam generator and the first isolation valve. If this valve fails, the volume between the break and the isolation valves in the other steamlines, including safety and relief valve headers and other connecting lines, will feed the break. For the cases which modeled a failure of a MSIV, the steamline volumes associated with Unit 2 were assumed since the volume available for blowdown for this scenario is greater than Unit 1. For the cases which did not model a failure of a MSIV, the steamline volumes associated with Unit 1 were assumed since the volume available for blowdown for this scenario is greater than Unit 2.

b. Failure of a diesel generator would result in the loss of one containment safeguards train resulting in minimum heat removal capability.

c. Failure of a feedwater isolation valve would result in additional inventory in the feedwater line which would not be isolated from the steam generator. The mass in this volume can flash into steam and exit through the break. For consistency with the FSAR steamline break mass/energy release analysis, all cases conservatively assumed failure of the feedwater isolation valve, which resulted in the additional inventory available for release through the steambreak and in higher than normal main feedwater flows.

d. Failure of the auxiliary feedwater runout control equipment would result in higher auxiliary feedwater flows entering the steam generator prior to re-alignment of the AFW system. For cases where the runout control operates properly, a bounding constant AFW flow of 670 gpm to the faulted steam generator was assumed. This value was increased to 1325 gpm to simulate a failure of the runout control.

Results

The steamline break mass/energy releases inside containment were calculated to account for the range of conditions possible for the potential reratings of Unit 1 and Unit 2 and for the relaxation of certain plant parameters. One set of mass/energy releases were calculated to bound the reratings for both Units incorporating the limiting steamline break protection design of Unit 1. The analysis assumptions support relaxation of the most negative moderator temperature coefficient limit, degradation of the charging pump performance of the Emergency Core Cooling System, extension of the main steam isolation valve closure time, and relaxation of the minimum BIT boron concentration requirement.

Section S-3.4.2.1 presents the containment integrity evaluation for a main steamline break using the mass/energy releases calculated here. As discussed in Section S-3.4.2.1, the limiting scenarios of the steambreak cases analyzed for the containment response evaluation were a break size of 4.6 ft^2 occurring at 102% power with a main steamline isolation failure for the double-ended rupture scenario and a break size of 0.86 ft^2 occurring at 102% power with an auxiliary feedwater runout protection failure for the split rupture scenario. Table S-3.3-4 presents the mass/energy releases for these limiting steambreak cases of the containment response evaluation.

Steamline Break Mass/Energy Releases Outside Containment

The current mass/energy releases used for outside containment equipment qualification evaluation are documented in Reference 3. The mass/energy releases were calculated in Reference 3 to address concerns over the effect of

superheated steam releases on the Environmental Qualification (EQ) of equipment located outside containment. These superheated mass/energy releases were provided to AEPSC for use in their evaluation of the outside containment equipment qualification issues.

The steamline break mass/energy releases outside containment were re-calculated to determine the effect of the range of conditions possible for the Cook Rerating Program. The re-calculated mass/energy releases are applicable for the full range of temperatures associated with the rerating parameters of Unit 1. Note that the Unit 1 superheated mass/energy releases documented in Reference 3 and discussed in Section 3.3.4.1 of WCAP-11902 are only applicable for a full power T_{avg} of 567.8°F or below. Mass/energy releases were also re-calculated to determine the effect of the range of conditions possible for the potential rerating of Unit 2.

The new mass/energy release calculations also incorporated the plant parameter changes discussed in Section S-3.3.1. The changes of the plant parameters (i.e., relaxed most negative MTC limit, degraded ECCS performance, and increased MSIV closure time) are penalties compared to the assumptions used in the Reference 3 analysis for the calculation of the mass/energy releases. (The existing Reference 3 mass/energy releases included a 0 ppm BIT boron concentration.)

The new superheated mass/energy releases, which incorporate the parameters of the Cook Rerating Program, were provided to AEPSC for use in the outside containment equipment qualification evaluation.

S-3.3.4.2 Startup of an Inactive Loop

The startup of an inactive loop event was evaluated in Section 3.3.4.2 of WCAP-11902 to support the reduced temperature and pressure operation. This accident was not evaluated for the Unit 1 Rerating Program since the event can not occur in Mode 1 or Mode 2 as restricted by the Technical Specifications (Amendment 120). Amendment 120 to the Unit 1 Technical Specifications resulted in the removal of Mode 1 and Mode 2 three loop operating

specifications. Whereas Section 3.3.4.2 of WCAP-11902 contained an evaluation of the startup of an inactive loop event for the reduced temperature and pressure operation, evaluations of this event are not necessary due to the approval of Amendment 120 to the Unit 1 Technical Specifications. Since three loop operation in Mode 1 or Mode 2 is prohibited, the startup of an inactive loop event does not have to be considered to support the rerating of Unit 1.

S-3.3.4.3 Uncontrolled RCCA Withdrawal From A Subcritical Condition

The uncontrolled RCCA withdrawal from a subcritical condition event was analyzed in Section 3.3.4.3 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The plant parameter changes (i.e., degraded ECCS performance, increased MSIV closure time, and revised steam generator water level program) do not impact the safety evaluation since they are not assumed in the uncontrolled RCCA withdrawal from a subcritical condition analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.3 remain applicable for the parameters of the potential Unit 1 rerating.

S-3.3.4.4 Uncontrolled Control Rod Assembly Bank Withdrawal At Power

The uncontrolled control rod assembly RCCA bank withdrawal at power event was analyzed in Section 3.3.4.4 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The moderator density coefficient of 0.54 Δk/gm/cc assumed for the maximum reactivity feedback cases supports a relaxed Technical Specification value for the most negative moderator temperature coefficient (See Section S-3.13).

Although the revised steam generator water level program changes the initial steam generator water level assumed for the 10% power cases, the change does not impact the results of the analysis. The analysis is not sensitive to initial steam generator water level since no fluctuations in steam generator

secondary-side fluid mass are modeled during the analysis. The rod withdrawal at power event is analyzed to show the adequacy of the nuclear overpower and the OTΔT reactor trips to protect the core thermal safety limits. The adequacy of the steam generator water level trips are examined in the loss of normal feedwater (Section S-3.3.4.9), loss of all A.C. power to the station auxiliaries (Section S-3.3.4.12), and the excessive heat removal due to feedwater system malfunction (Section S-3.3.4.10) events.

The revised level program does not impact the 100% and 60% power cases since there is no change to the steam generator water level program above 20% power. The other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the uncontrolled RCCA bank withdrawal at power analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.4 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.5 Rod Cluster Control Assembly Misalignment

The rod cluster control assembly misalignment events were analyzed in Section 3.3.4.5 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The dynamic dropped RCCA cases considered a range of moderator temperature coefficients which encompass a relaxed Technical Specification value for the most negative moderator temperature coefficient. The statically misaligned RCCA analysis does not assume moderator temperature coefficient. The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. The other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the rod cluster control misalignment analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.5 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.6 Chemical and Volume Control System Malfunction

The boron dilution during startup and at power events were analyzed in Section 3.3.4.6 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The plant parameter changes (i.e., relaxed most negative MTC limit, degraded ECCS performance, increased MSIV closure time, and revised steam generator water level program) do not impact the analysis since they are not assumed in the boron dilution event analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.6 remain applicable for the parameters of the Unit 1 rerating.

3.3.4.7 Loss Of Reactor Coolant Flow (Including Locked Rotor Analysis)

The loss of reactor coolant flow (partial and complete) and locked rotor events were analyzed in Section 3.3.4.7 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The conservative direction for the moderator temperature coefficient for these events is to assume the most positive MTC. As such, the relaxation of the most negative MTC limit does not affect the WCAP-11902 analysis. The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. Also, the other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the loss of reactor coolant flow and locked rotor analyses. Thus, the safety analysis and conclusions presented in Section 3.3.4.7 remain applicable for the parameters of the Unit 1 rerating.

3.3.4.8 Loss of External Electrical Load

The complete loss of steam load from full power event was analyzed in Section 3.3.4.8 of WCAP-11902 to support the reduced temperature and pressure

operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The moderator density coefficient of 0.54 Δk/gm/cc assumed for the maximum reactivity feedback cases supports a relaxed Technical Specification value for the most negative moderator temperature coefficient (See Section S-3.13). The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. The other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the loss of external electrical load analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.8 remain applicable for the parameters of the Unit 1 rerating.

3.3.4.9 Loss of Normal Feedwater Flow

The loss of normal feedwater event was analyzed in Section 3.3.4.9 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The conservative direction for the moderator temperature coefficient for this event is to assume the most positive MTC. As such, the relaxation of the most negative MTC limit does not affect the WCAP-11902 analysis.

The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. Also, the other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the loss of normal feedwater analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.9 remain applicable for the parameters of the Unit 1 rerating.

3.3.4.10 Excessive Heat Removal Due to Feedwater System Malfunctions

The excessive heat removal due to feedwater system malfunctions were analyzed in Section 3.3.4.10 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The conservative direction for the moderator temperature coefficient for this event is to assume the most negative MTC. The moderator density coefficient of 0.54 Δk/gm/cc assumed for the analysis supports a relaxed Technical Specification value for the most negative moderator temperature coefficient (See Section S-3.13). The revised steam generator water level program does not impact the analysis. For the full power analysis, there is no change to the steam generator water level program above 20% power due to the proposed level program. The 0% power analysis is performed to determine the maximum reactivity insertion rate caused by the increase in feedwater flow. The 0% power analysis is not sensitive to initial steam generator water level. The other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the excessive heat removal due to feedwater system malfunction analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.10 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.11 Excessive Increase in Secondary Steam Flow

The excessive increase in secondary steam flow event was analyzed in Section 3.3.4.11 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The moderator density coefficient of 0.54 Δk/gm/cc assumed for the maximum reactivity feedback cases supports a relaxed Technical Specification value for the most negative moderator temperature coefficient (See Section S-3.13). The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. The other

plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the excessive increase in secondary steam flow analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.11 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.12 Loss of All AC Power to the Plant Auxiliaries

The loss of all AC power to the plant auxiliaries event was analyzed in Section 3.3.4.12 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The conservative direction for the moderator temperature coefficient for this event is to assume the most positive MTC. As such, the relaxation of the most negative MTC limit does not affect the WCAP-11902 analysis. The revised steam generator water level program does not impact the analysis, which is performed at full power, since there is no change to the steam generator water level program above 20% power. Also, the other plant parameter changes (i.e., degraded ECCS performance and increased MSIV closure time) do not impact the safety evaluation since they are not assumed in the analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.12 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.13 Rupture of a Steam Pipe

The rupture of a steam pipe event was analyzed in Section 3.3.4.13 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis.

The relaxation of the Technical Specification most negative moderator temperature coefficient refers to the core MTC limit in the unrodded configuration. This MTC limit relaxation is incorporated into the steamline

break core response analysis. The WCAP-11902 analysis assumed a negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The reactivity feedback assumption is adjusted to conservatively predict the return to power transient. Verification is performed to show that the reactivity feedback employed in the analysis is conservative.

The analysis conservatively assumed the minimum capability for injection of boric acid solution corresponding to the most limiting single failure in the Emergency Core Cooling System (ECCS). The analysis assumed that the safety injection flow was provided by one charging pump. The analysis assumed degraded performance of the charging pump. Figure 3.3-52 of WCAP-11902 presents the safety injection flow rates as a function of RCS pressure, which takes into account the degraded performance of this ECCS (charging pump) system. The analysis also conservatively assumed a boron concentration of 0 ppm for the boron injection tank (BIT). As such, the analysis supports degradation of the charging pump performance and positions Unit 1 for relaxation of the minimum BIT boron concentration requirement.

The steamline break core response analysis assumed steamline isolation to occur within 11 seconds from receipt of the signal generated by high steam flow coincident with low steam pressure. The 11 second delay is assumed to account for signal processing and electronic delay plus the closure time of the main steamline isolation valves (MSIV). The analysis models only the total delay from the time the setpoint is reached until the time the MSIV is fully closed. Although the WCAP-11902 analysis specified that a MSIV closure time of 7 seconds was assumed, margin is available in the total delay time assumed to support an 8 second MSIV closure time. The 8 second MSIV closure time represents an increase of 3 seconds from the existing Technical Specification limit (5 seconds). As such, the WCAP-11902 steamline break core response analysis supports a relaxation of the MSIV closure time requirement.

The WCAP-11902 steamline break core response analysis is performed at Hot Zero Power, with a corresponding initial steam generator level at 33% NRS. Increasing the initial level to 44% NRS insignificantly impacts the results of

the analysis. Increasing the water level will not have an unacceptable effect on the minimum DNBR for the double-ended rupture (4.6 ft^2, 1.4 ft^2) steamline break core response analysis. This evaluation is based on sensitivity studies presented in WCAP-9227, "Reactor Core Response to Excessive Secondary Steam Releases" (Reference 4). Although this report was not used in support of the WCAP-11902 analysis, the conclusions presented are generic in nature and as such can be applied to Cook Unit 1.

Thus, the safety analysis and conclusions presented in Section 3.3.4.13 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.4.14 Rupture of Control Rod Drive Mechanism Housing (RCCA Ejection)

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The RCCA ejection event was analyzed in Section 3.3.4.14 of WCAP-11902 to support the reduced temperature and pressure operation as well as to bound the range of conditions possible for the rerating of Unit 1. Table S-3.3-3 presents the initial conditions assumed in the WCAP-11902 analysis. The conservative direction for the moderator temperature coefficient for these events is to assume the most positive MTC. As such, the relaxation of the most negative MTC limit does not affect the WCAP-11902 analysis. Also, the other plant parameter changes (i.e., degraded ECCS performance, increased MSIV closure time, and revised steam generator water level program) do not impact the safety evaluation since they are not assumed in the RCCA ejection analysis. Thus, the safety analysis and conclusions presented in Section 3.3.4.14 remain applicable for the parameters of the Unit 1 rerating.

S-3.3.5 Conclusions of Non-LOCA Safety Evaluation

The non-LOCA safety analyses and evaluations presented in Section 3.3 of WCAP-11902 and supplemented by the analyses and evaluations presented in this section support the range of conditions possible for the rerating of Unit 1 (Cases 4 and 5 of Table S-2.1-1), including the reduced temperature and pressure operation of Unit 1 (Cases 2 and 3 of WCAP-11902, Table 3.3-1). The

steamline break mass/energy releases (inside and outside containment) analyses support the range of conditions for the Unit 1 rerating as well as position Unit 2 for a potential rerating. The safety evaluation includes support for the relaxation of certain plant parameters:

Increased Most Negative Moderator Temperature Coefficient (MTC) (Tech Spec 3.1.1.4b)
Degraded ECCS Charging Pump Flow (Tech Spec 4.5.2f)
Increased Main Steamline Isolation Valve (MSIV) Closure Time (Tech Spec 4.7.1.5b and Tech Spec Table 3.3-5 items 5h, 6h, & 7c)

Section S-3.13 of this document presents the updates to the Unit 1 Technical Specifications.

The evaluation also conservatively assumes 0 ppm boron concentration in the Boron Injection Tank (BIT).

The safety evaluation includes support for revising the steam generator water level program to a constant level of 44% NRS from 0% power to full power. The safety evaluation also supports a steam generator average tube plugging level of 10% with a peak tube plugging level of 15% for the range of conditions possible for the Unit 1 rerating and the Unit 1 reduced temperature and pressure operation, provided the minimum measured flow of 366,400 gpm (plant total) is met and the nominal RCS temperatures do not exceed the range of temperatures presented in Table S-2.1-1, Cases 4 and 5, and WCAP-11902, Table 3.3-1, Cases 2 and 3.

NOTE: A safety evaluation independent of the D. C. Cook Rerating Program has been performed to support an increase in the pressurizer pressure uncertainty from ±35 psi to ±58 psi. The evaluation showed that this increase in pressurizer pressure uncertainty does not affect the conclusions of the Unit 1 Reduced Temperature and Pressure safety evaluation (WCAP-11902: Reference 1) and the conclusions of the Unit 1 Rerating safety evaluation presented in this supplement to WCAP-11902. The increase in pressurizer pressure uncertainty from ±35 psi to ±58 psi safety evaluation is documented in Reference 6.

S-3.3.6 REFERENCES

1. Augustine, D. B., and Cecchett, D. L., "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 Licensing Report," WCAP-11902, October 1988.

2. Burnett, T, W..T., et al., "LOFTRAN Code Description," WCAP-7907-A, April 1, 1984.

3. Butler, J. C., and Love, D. S, "Steamline Break Mass/Energy Releases for Equipment Qualification Outside Containment," WCAP-10961, Rev. 1 (proprietary) and WCAP-11184 (nonproprietary), October, 1985.

4. Hollingsworth, S. D., and Wood, D. C., "Reactor Core Response To Excessive Secondary Steam Releases," WCAP-9227, January 1978.

5. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8860, September 1976.

6. "American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1: Safety Evaluation for Including Uncertainty Due to Operator Readability of Pressurizer Pressure Instrumentation," AEP-89-216, Letter from J. C. Hoebel (W) to R. B. Bennett (AEPSC), September 1989.

TABLE S-3.3-1
NON-LOCA ACCIDENTS EVALUATED FOR UNIT 1 RERATING

Events Appearing in WCAP-11902 Supplement

FSAR SECTION/ WCAP-11902 SECTION	ACCIDENT
14.1.5 (Unit 2)/ S-3.3.4.1	Steamline Break Mass/Energy Releases Inside Containment
N/A (Reference 3)/ S-3.3.4.1	Steamline Break Mass/Energy Releases Outside Containment

Events Appearing in WCAP-11902

FSAR SECTION/ WCAP-11902 SECTION	ACCIDENT
14.1.6/3.3.4.2	Startup of an Inactive Loop
14C.3.1/3.3.4.3	Uncontrolled RCCA Withdrawal from a Subcritical Condition
14C.3.2/3.3.4.4	Uncontrolled RCCS Withdrawal at Power
14C.3.3/3.3.4.5	RCCA Misalignment
14C.3.3/3.3.4.5	RCCA Drop
14C.3.4/3.3.4.6	Chemical Volume and Control System Malfunction
14C.3.5/3.3.4.7	Loss of Reactor Coolant Flow (including Locked Rotor Analysis)
14C.3.6/3.3.4.8	Loss of External Load
14C.3.7/3.3.4.9	Loss of Normal Feedwater
14C.3.8/3.3.4.10	Excessive Heat Removal Due to Feedwater System Malfunction
14C.3.9/3.3.4.11	Excessive Load Increase Incident
14C.3.10/3.3.4.12	Loss of All A.C. Power to the Station Auxiliaries
14C.3.11/3.3.4.13	Rupture of a Steam Pipe
14C.3.12/3.3.4.14	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

TABLE S-3.3-2
TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES[c]

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delay (Seconds)
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, see Figure 3.3-1, WCAP-11902	8.0[a]
Overpower ΔT	Variable, see Figure 3.3-1, WCAP-11902	8.0[a,d]
High pressurizer pressure	2420 psig	2.0
Low pressurizer pressure	1825 psig	2.0
High pressurizer water level	100% NRS	2.0
Low reactor coolant flow (From loop flow detectors)	87 percent loop flow	1.0
Undervoltage trip	b	1.5
Low-low steam generator level	0.0 percent of narrow range level span	2.0
High steam generator level	72 percent of narrow range level span	
Turbine Trip		2.5
Feedwater Isolation		11.0

a Total time delay (including RTD bypass loop fluid transport delay effect, bypass loop piping thermal capacity, RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall. The time delay assumed in the analysis supports the 6 second response time of the RTD time response, trip circuit delays, and channel electronics delay presented in the Technical Specifications.

b No explicit value assumed in the analysis. Undervoltage trip setpoint assumed reached at initiation of analysis.

c The control rod scram time to dashpot is 2.4 seconds.

d Overpower ΔT reactor trip was assumed in steamline break mass/energy release outside containment calculations.

TABLE S-3.3-3

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Moderator Density (ΔK/gm/cc)	Reactivity Coefficients Assumed: Doppler	DNB Correlation	Improved Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure(6) (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	Refer to Section 3.3.4.3 (7)		Min (1)	W-3/WRB-1 See Section 3.3.4.3 (7)	No	0	162,840	547	2065 (5)
Uncontrolled Rod Cluster Assembly Bank Withdrawal At Power (2)	LOFTRAN	+5	.54	Min and Max (3)	WRB-1	Yes	3425 2055 343	366,400	578.7 566.02 550.17	2100
Rod Cluster Control Assembly Misalignment	LOFTRAN THINC	NA*	NA	NA	WRB-1	Yes	3425	366,400	578.7	2100
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA	NA	3425 0	NA	NA	NA
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	NA	Max	WRB-1	Yes	3425	366,400	578.7	2100
Locked Rotor (Peak Pressure)	LOFTRAN	+5	NA	Max	NA	NA	3494	354,000	583.2	2285
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	NA	Max	NA	NA	3494	354,000	583.2	2135
Locked Rotor (Rods-in-DNB)	LOFTRAN FACTRAN THINC	+5	NA	Max	WRB-1	Yes	3425	366,400	578.7	2100

* NA - Not Applicable

(1) Minimum Doppler power defect (pcm/%power) = -9.55 + 0.035Q where Q is in % power.

(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.

(3) Maximum Doppler power defect (pcm/% power) = -19.4 + 0.065Q.

(4) Minimum and Maximum reactivity feedback cases were examined.

(5) Core Pressure

(6) See Note 1 in NOTES Section after Section S-3.3.5.

(7) Refers to Sections or Figures of WCAP-11902.

TABLE S-3.3-3 (Cont'd)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (Δk/gm/cc)	Doppler	DNB Correlation	Improved Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure(6) (PSIA)
Loss of Electrical Load and/or Turbine Trip (4)	LOFTRAN	+5	.54	Max and Min	WRB-1	Yes	3494	366,400	583.2	2065
Loss of Normal Feedwater	LOFTRAN	+5	NA	Max	NA	NA	3494	354,000	551.5	2285
Excessive Heat Removal Due to Feedwater System Malfunction	LOFTRAN	NA	.54	Min	WRB-1	Yes	3425 0	366,400	578.7 547	2100
Excess Load Increase Incident	LOFTRAN	NA	0 and .54	Max and Min	WRB-1	Yes	3425	366,400	578.7	2100
Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	+5	NA	Max	NA	NA	3494	354,000	542.5	2285
Rupture of a Steam Pipe	LOFTRAN THINC	See Figure 3.3-51a (7)	NA	See Figure 3.3-51b (7)	W-3	NA	0	354,000	547	2100
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 3.3.4.14 (7)	NA	Min	NA	NA	3494 0	354,000 162,840	583.2 547	2065 (5)
Steamline Break M/E Releases Inside Containment	LOFTRAN	NA	.54	Max	NA	NA	3672 2520 1080 0	354,000	585.8 575.5 561.8 547	2250

* NA - Not Applicable
(1) Minimum Doppler power defect (pcm/%power) = -9.55 + 0.035Q where Q is in % power.
(2) Multiple power levels, Tavg, and reactivity feedback cases were examined.
(3) Maximum Doppler power defect (pcm/% power) = -19.4 + 0.065Q.
(4) Minimum and Maximum reactivity feedback cases were examined.
(5) Core Pressure
(6) See Note 1 in NOTES Section after Section S-3.3.5.
(7) Refers to Sections or Figures of WCAP-11902.

TABLE S-3.3-4

.STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER DER (4.6 FT^2) BREAK
FAILURE - MSIV

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
0.00	0.00	0.0
0.20	10430.00	1.250
3.60	6552.00	7.883
6.60	5612.00	6.748
12.80	4978.00	5.974
13.00	4913.00	5.895
13.20	4847.00	5.816
13.40	4781.00	5.737
13.60	4716.00	5.660
14.00	4587.00	5.504
14.40	4458.00	5.350
14.80	4332.00	5.198
15.00	4269.00	5.123
15.20	4206.00	5.047
15.60	4083.00	4.899
15.80	4022.00	4.826
16.00	3961.00	4.753
16.60	3782.00	4.538
17.20	3606.00	4.328
17.60	3492.00	4.190
17.80	3435.00	4.122
18.40	3268.00	3.921
18.60	3213.00	3.856
18.80	3158.00	3.790
19.20	3050.00	3.660
23.80	1876.00	2.251
28.80	1623.00	1.421
30.40	1575.00	1.883
36.40	1461.00	1.746
39.20	1431.00	1.708
50.70	1369.00	1.634
57.20	1356.00	1.618
106.20	1331.00	1.588
109.20	1331.00	1.587
111.20	1184.00	1.409
118.20	308.70	0.358
125.20	188.10	0.217
136.20	98.97	0.114
602.70	93.24	0.107

TABLE S-3.3-4 (Cont'd)

STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER SPLIT (0.86 FT^2) BREAK
FAILURE - AUXILIARY FEEDWATER RUNOUT PROTECTION

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
0.00	0.00	0.0000
0.20	1394.00	1.6690
1.60	1366.00	1.6370
2.00	1358.00	1.6270
2.40	1350.00	1.6170
2.80	1342.00	1.6080
4.20	1316.00	1.5770
4.40	1312.00	1.5730
8.60	1550.00	1.8540
9.40	1575.00	1.8840
12.00	1632.00	1.9500
12.60	1638.00	1.9570
15.80	1635.00	1.9530
18.00	1618.00	1.9340
21.40	1458.00	1.7460
22.60	1400.00	1.6790
23.60	1357.00	1.6280
23.80	1349.00	1.6180
25.00	1302.00	1.5630
32.00	1103.00	1.3260
32.20	1098.00	1.3210
33.80	1064.00	1.2810
42.00	928.70	1.1180
42.60	920.80	1.1090
43.20	913.10	1.1000
43.80	905.70	1.0910
44.40	898.40	1.0820
55.20	799.10	0.9625
67.20	732.60	0.8823
80.20	691.30	0.8325
82.20	686.60	0.8269
96.20	662.50	0.7977
98.70	659.50	0.7941
118.20	645.70	0.7775
124.20	643.60	0.7749
282.70	633.20	0.7623
285.20	633.10	0.7622
290.20	615.00	0.7402
292.70	579.70	0.6977
297.70	556.60	0.6695
302.70	490.40	0.5896
320.20	304.70	0.3643

TABLE S-3.3-4 (Cont'd)

STEAMLINE BREAK
MASS/ENERGY RELEASES INSIDE CONTAINMENT
102% POWER SPLIT (0.86 FT^2) BREAK
FAILURE - AUXILIARY FEEDWATER RUNOUT PROTECTION

TIME (SEC)	MASS (LBM/SEC)	ENERGY (BTU x 10^6/SEC)
330.20	238.70	0.2845
340.20	206.50	0.2456
352.70	190.20	0.2259
525.20	181.90	0.2160
535.20	182.00	0.2160
600.20	182.10	0.2162
605.20	190.70	0.2258

S-3.4 CONTAINMENT ANALYSIS

S-3.4.1 Short-Term Containment Analysis

The short term containment integrity analysis is used to verify the adequacy of interior structures and walls by demonstrating that calculated differential pressures are less than design limits. The functioning of the ice condenser is demonstrated and containment integrity is also verified. The efforts performed for the short term containment analysis, applicable to the Pressurizer Enclosure, the Fan Accumulator Room, the Loop Compartments and the Steam Generator Enclosure, as described in Section 3.4.1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," support operation of Cook Nuclear Plant Units 1 & 2 over the full range of rerated parameters described in Section S-2.1. There is no direct impact of power level on LOCA short term mass and energy release rate calculations and containment subcompartment response analysis. The major impacts are the resulting effects due to RCS temperature changes whenever power is increased. For the steam generator enclosure, mass and energy releases and the subsequent containment response are performed at zero power, which maximizes effects because steam pressure is maximum. All relevant analyses and evaluations were performed assuming bounding values, for both Units 1 & 2, of the rerated power levels and revised temperatures and pressures described in Section S-2.1. The conclusions of Section 3.4.1.6 of WCAP-11902, therefore, are applicable for Cook Units 1 & 2 at their respective rerated powers and the revised temperatures and pressures.

S-3.4.2 Long-Term Containment Analysis

S-3.4.2.1 Main Steamline Break (MSLB) Containment Integrity

Introduction and Background

An evaluation was performed to determine the impact of reduced temperature and pressure operation on the Donald C. Cook Nuclear Plant Unit 1 Long-Term Main Steamline Break Containment Integrity analysis. This evaluation is documented

in Section 3.4.2 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," and it was concluded that reduced temperature and pressure operation did not have an adverse impact on the analysis results and conclusions. This Section documents the analysis performed for both Donald C. Cook Nuclear Plant Units 1 & 2 to determine the impact of the rerated conditions described in Section S-2.1 on Containment Integrity following a Main Steamline Break.

A series of main steamline split and double-ended breaks were analyzed as a part of the original licensing basis for Donald C. Cook Nuclear Plant Unit 2 to determine the most severe break condition for containment temperature and pressure response for this design basis event. The analysis and evaluation are discussed in Reference 1. These results documented in the FSAR show that the most limiting double-ended break was the 4.6 square foot break, occurring at 102% power with main steam isolation valve failure. The most limiting split break was the 0.942 square foot break, occurring at 30% power with the failure of auxiliary feedwater runout protection. The calculated peak temperatures for these cases were 319.1°F and 328.1°F respectively. Additional generic sensitivities discussed in Reference 2, illustrate that other smaller breaks were not limiting.

Purpose

The purpose of the analysis documented in the following paragraphs is to demonstrate that the peak containment temperature resulting from a design basis main steamline break will not exceed the equipment qualification temperature criterion for Donald C. Cook Nuclear Plants Units 1 and 2, at the rerated conditions. The containment pressure response generated for the LOCA Containment Integrity analysis for the double-ended pump suction RCS break case (Reference 3) bounds the Main Steamline Break containment pressure response, and therefore is not a concern here. This analysis assumes reduced safety injection flow, due to degradation of ECCS performance, closure of the RHR crosstie valves and the current containment heat sink information.

Analytical Assumptions

The analysis performed for the Rerating Program is consistent with the Reference 1 analysis except for assumptions directly related to the rerating parameters. The analytical effort provides bounding system calculations for both Units 1 & 2 at the rerated plant conditions described in Section S-2.1.

A spectrum of split breaks is analyzed at 0.86 ft^2, 102% power; 0.908 ft^2, 70% power; 0.942 ft^2, 30% power and 0.4 ft^2, hot shutdown. Double-ended breaks of 1.4 ft^2 and 4.6 ft^2 are analyzed at power levels of 102%, 70%, 30% and zero power levels.

The break sizes analyzed in the present analysis are based on the current FSAR analysis. As in the FSAR analysis, loss of one containment safeguards train was also assumed for all the cases in addition to the single failure assumed in the mass and energy release calculations.

The following cases were analyzed for containment response:

A. Split break cases

1) 0.86 ft^2, 102% power, MSIV failure
2) 0.86 ft^2, 102% power, AFRP failure
3) 0.908 ft^2, 70% power, MSIV failure
4) 0.908 ft^2, 70% power, AFRP failure
5) 0.942 ft^2, 30% power, MSIV failure
6) 0.942 ft^2, 30% power, AFRP failure
7) 0.40 ft^2, hot shutdown, MSIV failure
8) 0.40 ft^2, hot shutdown, AFRP failure

Note: MSIV - Main Steam Isolation Valve
AFRP - Auxiliary Feedwater Runout Protection

B. Double-ended rupture cases*

1) 4.6 ft^2, 102% power, MSIV failure
2) 4.6 ft^2, 102% power, AFRP failure
3) 4.6 ft^2, 70% power, MSIV failure
4) 4.6 ft^2, 70% power, AFRP failure
5) 4.6 ft^2, 30% power, MSIV failure
6) 4.6 ft^2, hot shutdown, MSIV failure
7) 1.4 ft^2, 102% power, MSIV failure
8) 1.4 ft^2, 102% power, AFRP failure
9) 1.4 ft^2, 70% power, MSIV failure
10) 1.4 ft^2, 30% power, MSIV failure
11) 1.4 ft^2, hot shutdown, MSIV failure

Note: *The limiting 4.6 ft^2 double-ended failure cases (102% and 70% power), with MSIV failure were analyzed with AFRP failure and found to be less limiting than the corresponding MSIV failure cases. Therefore only the most limiting 1.4 ft^2 (102% power) was analyzed with AFRP failure.

The mass and energy releases to the containment as a result of the postulated accident are calculated using the LOFTRAN computer code (Reference 4). The mass and energy releases are calculated using two different failures for each case namely, 1) failure of the auxiliary feedwater runout protection and 2) failure of the main steam isolation valve. As in Reference 1, no credit is taken for entrainment. Section S-3.3.4.1 presents additional details regarding the calculation of the inside containment steamline break mass and energy releases.

The LOTIC-III computer code (Reference 5) is used to calculate the consequence of these releases, in particular the peak containment temperature.

The main steam line break containment integrity calculations are performed with an additional failure of one of the containment safeguards trains, which results in minimum spray flow (this includes a 10% degradation in the spray pump flow). Where applicable, input data consistent with that of the LOCA containment integrity analysis (Reference 3) is used.

The total initial ice mass assumed is 2.11 x 10^6 lbs.

The initial conditions in the containment are a temperature of 120°F in the lower and dead ended compartments, a temperature of 27°F in the ice condenser, and a temperature of 57°F in the upper compartment. All volumes are at a pressure of 0.3 psig and a relative humidity of 15%.

The refueling water storage tank (RWST) temperature is assumed to be 100°F.

A spray pump flow of 1900 gpm to the upper compartment and 900 gpm to the lower compartment is assumed, at a temperature of 100°F.

The spray flow is initiated 45.0 seconds after the containment reaches the hi-hi pressure signal of 3.5 psig. This setpoint includes instrument uncertainties.

Results

The results of the analysis show that the maximum calculated containment temperature is 324.9°F for the 4.6 ft^2 double ended rupture at 102% of the full power. The mass and energy calculations for this case are based on the main steam isolation valve failure.

The maximum containment temperature calculated for the limiting small split break (0.86 ft^2 at 102% of full power) is 324.4°F. The auxiliary feedwater runout protection failure is assumed for this case. Table S-3.4-1 and Figures S-3.4-1 through S-3.4-4 show the results for the two limiting cases.

Comparison of these results to the current FSAR results with respect to the peak containment temperature indicates that the FSAR result was more limiting. This is due to the lower mass and energy releases inside containment, calculated for the present analysis. The peak temperature shown in the FSAR for the limiting split break case (0.86 ft^2 at 102% of full power, with auxiliary feedwater runout protection failure) is higher than the

present case. However, the FSAR results for the limiting double-ended rupture case (4.6 ft^2 at 102% power, with main steam isolation valve failure) is lower than the present double-ended results. A detailed study of the results shows that even though the mass and energy releases within containment are lower in both the present cases, the double-ended break results in a higher temperature due to reduced flows from the lower compartment into the ice-condenser.

The peak occurs very early in the transient (within the first ten seconds). At this early time the only heat removal systems that exist are the containment wall heat sinks and the heat flow between the compartments. In the present case, heat removal by the walls is better (due to more detailed modeling of the walls), but the heat flow from the lower compartment into the ice-condenser is lower (due to the lower initial temperature assumed in the ice-condenser and the upper compartment, which affects the driving force through the ice-condenser).

Conclusions

The main steamline break containment integrity analysis has been performed consistent with the current licensing basis analysis and Donald C. Cook Nuclear Plant Units 1 & 2 rerating program, considering the present plant operating conditions. The results of this analysis are bounded by the current FSAR results. This analysis therefore demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a main steamline break accident.

S-3.4.2.2 LOCA Containment Integrity

The long term peak containment pressure calculation has been recently performed in support of operation with the RHR crosstie valves closed at an NSSS power level of 3425 MWt. This analysis is documented in WCAP-11908. The report contained in this WCAP additionally provides justification for operation at the revised temperature and pressure conditions described in

Section S-2.1 except for Cases 7 - 10 which describe a rerated power level of 3600 MWt NSSS power. The conclusion presented in the report is that operation of Donald C. Cook Nuclear Plant Units 1 & 2 at a maximum NSSS power level of 3425 MWt and at the corresponding revised temperature and pressures described in Section S-2.1 with the RHR crosstie valves closed would result in an acceptable peak containment pressure of 11.89 psig. This is below the design value of 12 psig. Therefore, operation of Cook Units 1 & 2 at a maximum NSSS power level of 3425 MWt and at the revised temperatures and pressures corresponding to this power level is acceptable from a LOCA containment integrity peak pressure standpoint. The LOCA containment integrity analysis presented in WCAP-11908 does not support operation of Cook Unit 2 at the rerated NSSS power of 3600 MWt.

S-3.4.3 References

1. Westinghouse letter # NS-TMA-1946, 9/20/78, " American Electric Power Projects Donald C. Cook Unit 2 (Docket 50-316) Response to Question 022.9".

2. Westinghouse letter #AEP-80-525, 3/10/80, "Response to NRC Question 022.17 - AMP's steamline break analysis".

3. WCAP-11908," Containment Integrity Analysis for Donald C. Cook Nuclear Plant Units 1 and 2", July 1988.

4. WCAP-7907-P-A (Proprietary), "LOFTRAN Code Description", April 1984.

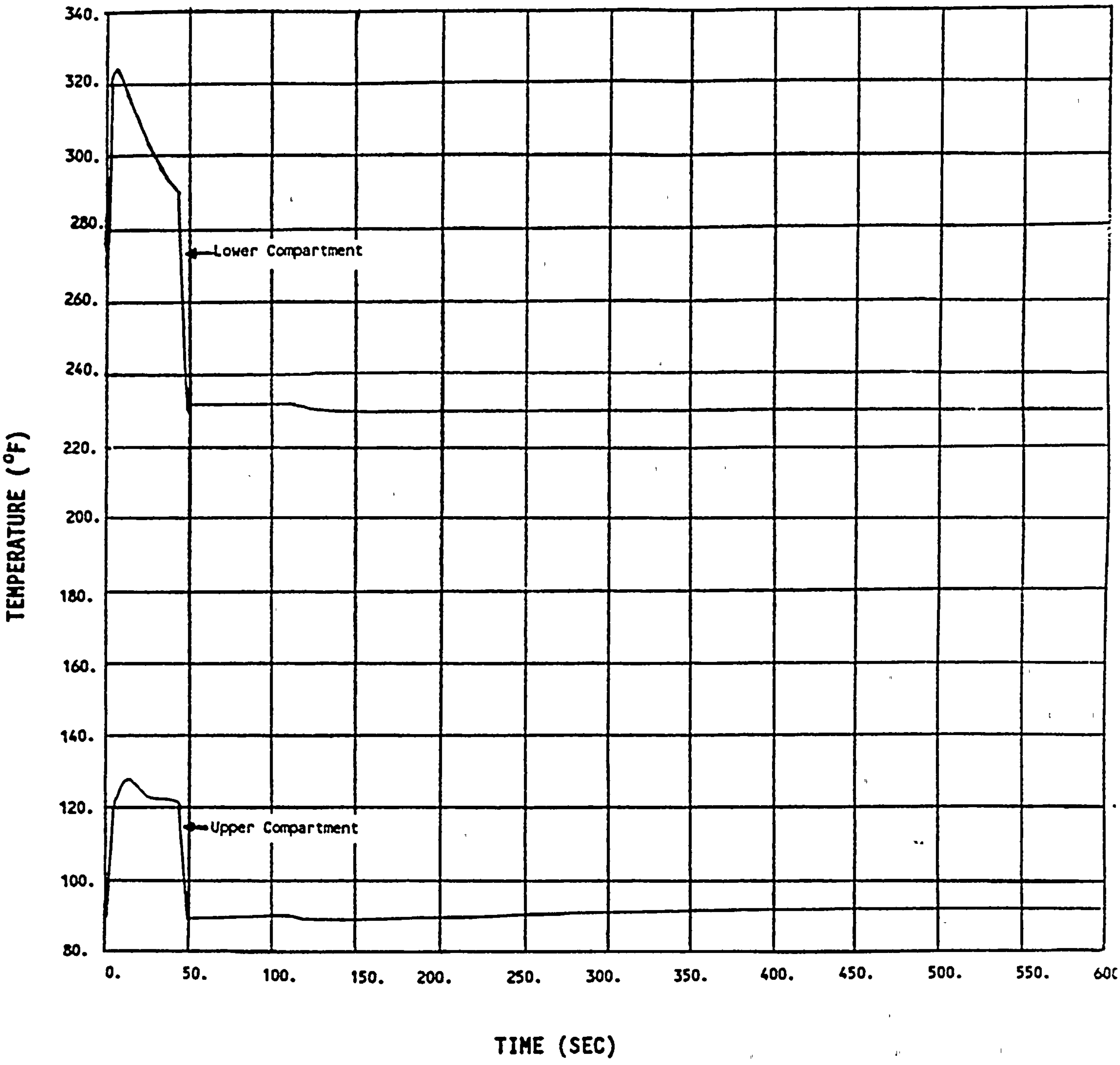
5. WCAP-8354-P-A (Proprietary), Supplement 2, "Long Term Ice Condenser Containment Code - LOTIC-3 Code", February 1979.

TABLE S-3.4-1

MAIN STEAMLINE BREAKS

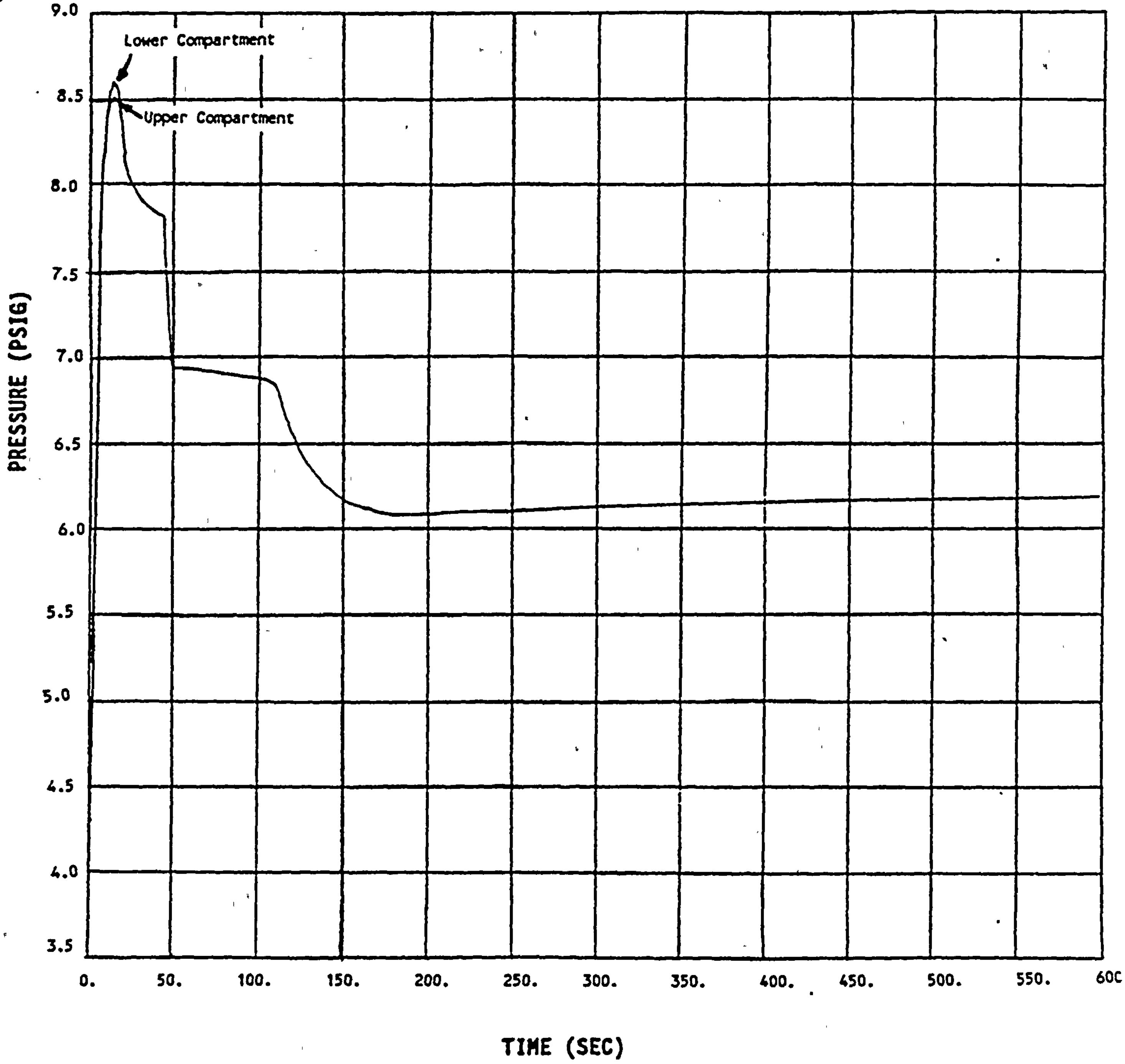
Type of Break	Double-Ended Rupture	Split Break
Break Size (FT^2)	4.6	0.86
Type of Failure	MSIV	AFRP
T_{max} (°F)	324.9	324.4
Time of T_{max} (sec)	6.39	50.72
P_{max} (psig)	8.62	7.24
Time of P_{max} (sec)	14.01	50.72

Note: MSIV - Main Steam Isolation Valve
AFRP - Auxiliary Feedwater Runout Protection



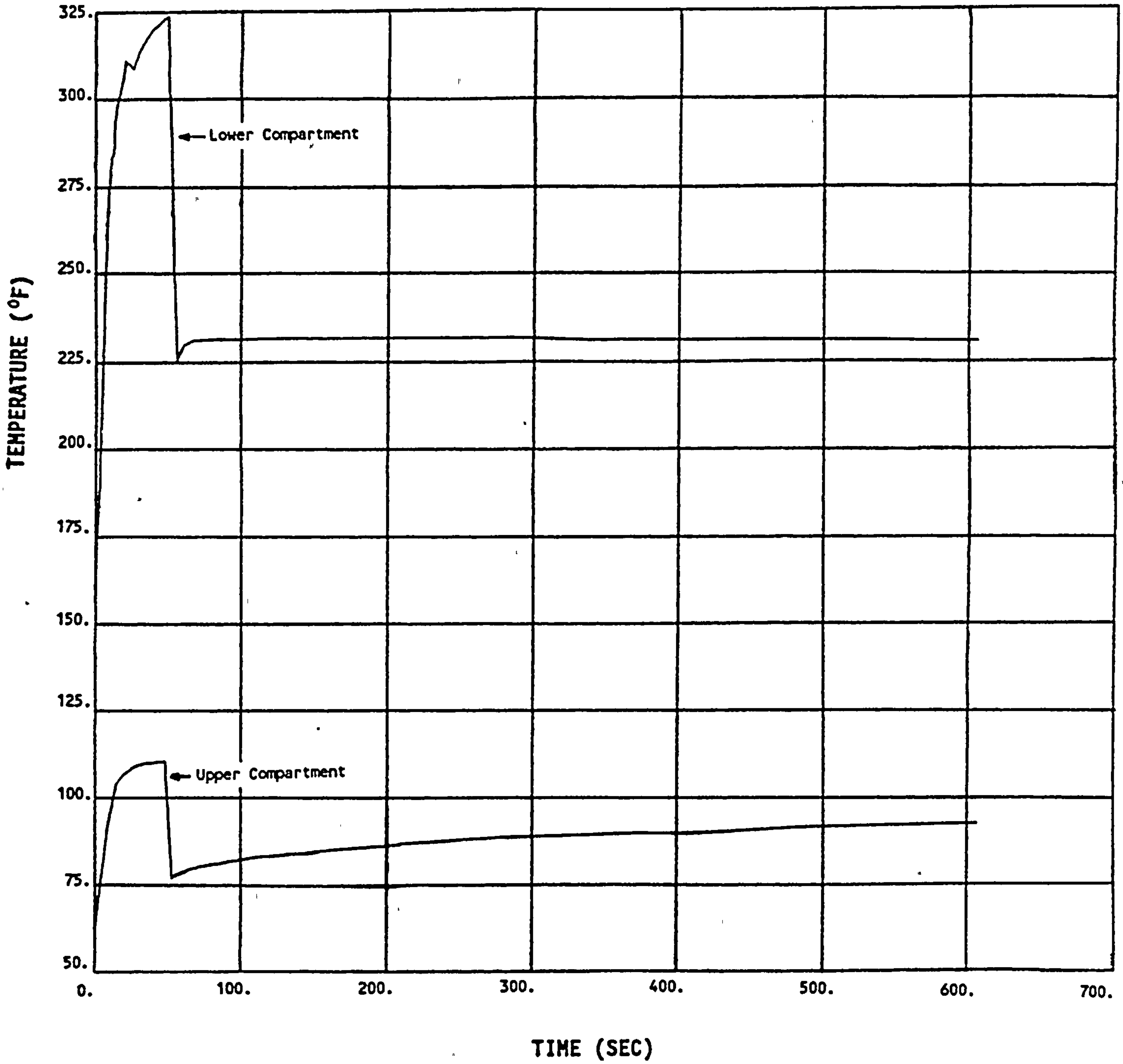
COMPARTMENT TEMPERATURE

Figure S-3.4-1 - 4.6 ft^2 Double-Ended Rupture, 102% Power, MSIV Failure



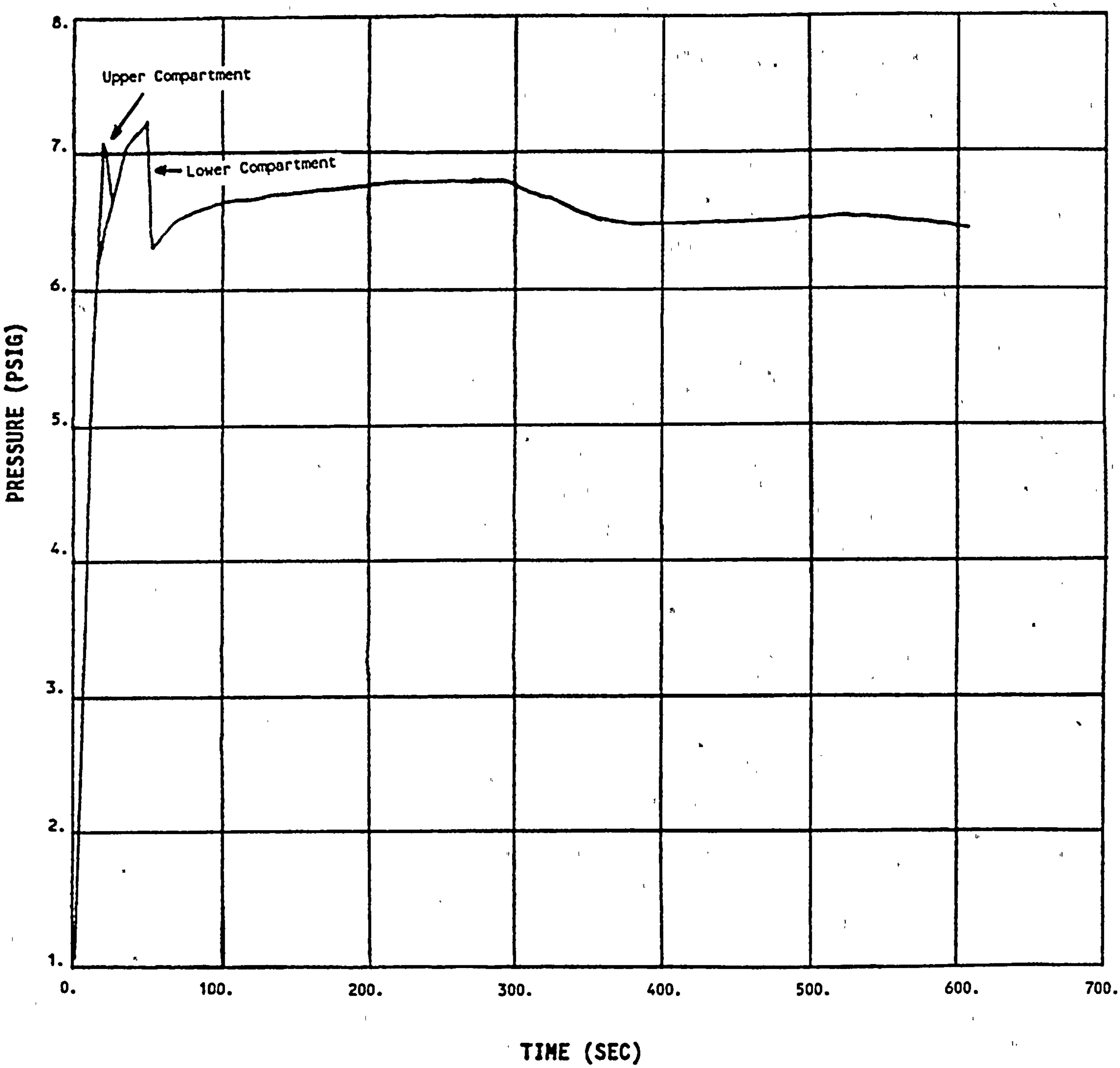
COMPARTMENT PRESSURE

Figure S-3.4-2 - 4.6 ft^2 Double-Ended Rupture, 102% Power, MSIV Failure



COMPARTMENT TEMPERATURE

Figure S-3.4-3 - 0.86 ft^2 Split Break, 102% Power, AFRP Failure



COMPARTMENT PRESSURE

Figure S-3.4-4 - 0.86 ft^2 Split Break, 102% Power, AFRP Failure

S-3.5 STEAM GENERATOR TUBE RUPTURE (SGTR) ACCIDENT

The SGTR accident is discussed in Section 3.5 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." The series of SGTR analyses performed and documented in WCAP-11902 considered the increased or uprated power for both Cook Units 1 and 2. The uprated power has been deemed acceptable because the results of the sensitivity analysis indicate that the FSAR conclusion that the Cook SGTR radiological consequences are within a small fraction of the limits set forth in 10CFR100 remains valid.

The analysis is important to show that the proposed window of operating conditions at Cook Unit 1 and 2, which are different from the conditions considered for the FSAR analysis, do not invalidate the conclusion in the FSAR that the radiological consequences are acceptable. The results of the sensitivity analyses, documented in WCAP-11902 and reiterated in this licensing report, confirm the acceptability of the range of operating conditions.

S-3.6 POST-LOCA HOT LEG RECIRCULATION TIME

S-3.6.1 Introduction

A hot leg recirculation switchover time analysis has been performed for Donald C. Cook Nuclear Plant Unit 1 to determine the time following a LOCA that hot leg recirculation should be initiated. This analysis addresses the concern of boron precipitation in the reactor vessel following a LOCA. The analysis was performed to support the Donald C. Cook Nuclear Plant Unit 1 for the rerated core power level of 3588 MWt and at the corresponding revised temperatures and pressures discussed in Section S-2.1. The Unit 1 rerated core power level is 3413 as discussed in Section S-2.1. The assumption of 3588 MWt for core power level was used to bound both Units 1 and 2.

S-3.6.2 Event Description

During a large break LOCA the plant switches to cold leg recirculation after the RWST switchover setpoint has been reached. If the break is in the cold leg there is a concern that the cold leg injection water will fail to establish flow through the core. Safety injection entering the broken loop will spill out the break, while SI entering the intact cold legs will circulate around the downcomer and out the break. The analysis assumes that the coolant in the core is stagnant. As steam is produced in the core from decay heat the analysis conservatively assumes that the boron associated with the steam remains in the vessel. Thus as steam is boiled off and with no circulation present in the core the boric acid concentration increases in the vessel. The boron concentration in the vessel will increase until the solubility limit of the boric acid solution is reached at which time boron will begin to precipitate. As the boron precipitates, it may plate out on the fuel rods which would adversely affect their heat transfer characteristics. The purpose of the hot leg recirculation switchover time analysis is to provide a time at which hot leg recirculation can be established such that boron precipitation in the core can be prevented.

S-3.6.3 Methodology

The calculation considers the increase in boric acid concentration in the vessel during the long term cooling phase of a LOCA. Since the analysis considers the build-up of boric acid in the vessel, maximum boron concentrations are used. For Cook Unit 1, the maximum boron concentration in the RWST and accumulators was assumed to be 2600 ppm. The initial RCS boron concentration was assumed to be 2400 ppm. The analysis assumes that following a LOCA the steam boil-off from the core does not carry any boron. A constant volume of liquid in the vessel is assumed so that as steam is boiled off and the boron is left behind, the boric acid concentration of the vessel increases. The time when the boric acid solution reaches the solubility limit less 4 weight percent is when hot leg recirculation should be initiated. The solubility limit less 4 weight percent at a solution temperature of 212°F has been established as 23.53%. Thus when the boric acid solution concentration reaches 23.53% hot leg recirculation should be initiated.

S-3.6.4 Results

An analysis has been performed to determine the time following a LOCA that switchover to hot leg recirculation should be initiated to prevent boron precipitation in the reactor vessel. This time has been determined to be consistent with the current value for the Donald C. Cook Nuclear Plant Unit 1 of 12 hours. For operational concerns the switchover time is conservatively truncated to the hour. Although the analysis, which incorporated the higher power, resulted in an earlier calculated time to switchover, the switchover time remained greater than 12 hours.

The analysis considers the increase in boric acid concentration in the reactor vessel during the long term cooling phase of a LOCA, assuming a conservatively small effective vessel volume. This volume includes only the free volumes of the reactor core and upper plenum below the bottom of the hot leg nozzles. This assumption conservatively neglects the mixing of boric acid solution with directly connected volumes, such as the reactor vessel lower plenum. The

calculation of boric acid concentration in the reactor vessel considers a cold leg break of the reactor coolant system in which steam is generated in the core from decay heat while the boron associated with the boric acid solution is completely separated from the steam and remains in the effective vessel volume.

The results of the analysis show that the maximum allowable boric acid concentration of 23.53 weight percent established by the NRC, which is the boric acid solubility limit less 4 weight percent (Reference 1), will not be exceeded in the vessel if hot leg recirculation is initiated 12 hours after the LOCA inception. The operator should reference this switchover time against the reactor trip/SI signal. The typical time interval between the accident inception and the reactor trip/SI actuation signal is negligible when compared to the switchover time.

Procedure philosophy assumes that it would be very difficult for the operator to differentiate between break sizes and locations. Therefore one value for hot leg switchover time, which bounds all break sizes and locations, is used to cover the complete break spectrum.

S-3.6.5 References

1. D. C. Cook Safety Evaluation Report to the ACRS, dated November 4, 1977 (pp. 6-11).

S-3.7 REACTOR CAVITY PRESSURE ANALYSIS

The Reactor Cavity Pressure Analysis is discussed in Section 3.7 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." This analysis was performed to support operation of the Donald C. Cook Nuclear Plant Units 1 and 2 for the range of rerating parameters discussed in Section S-2.1. Since this analysis considers the range of operating parameters (corresponding to a maximum power level of 3600 MWt NSSS) for both units, the conclusions presented in Section 3.7 of WCAP-11902 are valid for operation of Cook Nuclear Plant Units 1 and 2 at rerated conditions.

The rerating parameters affect the Reactor Cavity Pressure Analysis through the mass and energy releases provided as input to the analysis. There is no direct impact of power level on short-term mass and energy release rate calculations and containment subcompartment response analysis. Power level alone has an insignificant effect on short-term mass and energy releases because of the short duration of the event, i.e., less than 3 seconds. The major impact results from changes to RCS temperatures when power is increased. For long-term effects, higher RCS temperatures are conservative because of the higher total RCS energy content. For short-term effects, higher release rates typically result from cooler RCS conditions. The mass and energy releases used as input for the Reactor Cavity Pressure Analysis reflected limiting conditions for all power levels considered for the rerating, up to a maximum power of 3600 MWt NSSS.

S-3.8 RADIOLOGICAL ANALYSIS

S-3.8.1 Introduction

Radiological analyses were performed for the large break LOCA, Fuel Handling Accident and Steam Generator Tube Rupture (SGTR) Accident. A discussion of the radiological analyses is presented in Section 3.8 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." The analyses described in Section 3.8 of WCAP-11902 conclude that the source terms for the large break LOCA and the Fuel Handling Accident are unaffected by the revised temperatures and pressures and, therefore, the analysis presented in the FSAR remains bounding. The following discussions in this section describe the radiological analyses performed to assess the impact of the rerated power levels on the resulting doses for large break LOCA and the Fuel Handling Accident. The conclusion of the analyses is that there is no increase in the consequences of either event due to the rerated power levels. The analyses assume bounding parameters for operation of Cook Nuclear Plant Units 1 & 2 at 3588 MWt core power and the revised temperatures and pressures discussed in Section S-2.1.

The quantity of radioactivity released to the environment for the SGTR event is affected by the revised temperatures and pressures as well as the rerated power levels. The analysis discussed in Sections 3.5 and 3.8 of WCAP-11902 was performed assuming parameters which bound operation of both Cook Nuclear Plant Units 1 & 2 at their respective rerated powers and the revised temperature and pressures discussed in Section S-2.1. Therefore, the conclusion presented in WCAP-11902 are valid for the SGTR radiological analysis for both Cook Nuclear Plant Units 1 & 2 at their respective rerated power levels and the revised temperatures and pressure discussed in Section S-2.1. The discussion of the SGTR radiological analysis presented in this supplement supplies more details of the same analysis discussed in WCAP-11902.

S-3.8.2 Source Terms

An analysis has been performed to determined the effect of the rerated power level on the nuclide inventories contained in the core and reactor coolant. Specifically, inventories were recalculated at the uprated power level of 3588 MWt for the reactor core (which is the LOCA source term), for the lead fuel assembly assumed to be damaged in a fuel handling accident, and for the reactor coolant (which is the source term for the steam generator tube rupture). These values were compared to the corresponding values presented in the FSAR for the current power level.

The analytical methods used today to calculate the inventory of fission products in the core are significantly improved over the methods used for the original inventories. Specifically, the current calculations are performed using the ORIGEN computer code (ref 1) with the ENDF-B/IV based fission product data library (ref 2). The specific Cook Unit 1 fuel cycle that was modeled is described in Table S-3.8-1. This fuel cycle model includes two U-235 enrichments and a variety of specific powers. The fission product inventories currently presented in the FSAR (FSAR Table 14.3.5-2) were calculated assuming a constant U-235 thermal fission rate (derived from the core power level) using cumulative fission yields from Reference 3, for a single region core operation for 650 days (assumed to represent a three region, annual cycle core). The biggest differences result from the effects of the difference in fission yields between U-235 and Pu-239 and the effects of the differences is specific power in a given fuel assembly, rather than temperature and pressure.

The Cook Nuclear Plant Unit 2 fuel cycle is described in Table S-3.8-8. Although the core inventory of radioactive isotopes which are of concern in evaluating the radiological consequences of accidents (i.e., the short half-life noble gases and iodines) was specifically calculated for Cook Unit 1, it is also representative of Unit 2. The increased enrichment and the associated extension of fuel burnup have been shown to have negligible impact on these inventories. This has been documented both by Westinghouse (Reference 4) and by independent review performed for the NRC (Reference 5).

These reviews considered burnups of up to 60,000 MWD/Mtu for the lead rod which bound the fuel designs being provided for both Cook Nuclear Plant Units 1 and 2.

Also of concern in evaluating the impact of extending fuel burnup on the radiological consenquences of accidents is the fraction of core activity that is assumed to migrate out of the fuel matrix and into the fuel-clad gap region (i.e., the fuel-clad gap fraction) and thus, is available for release in the event the cladding is breached. The accident analysis for Cook have used both calculated gap fractions and those recommended by Regulatory Guide 1.25 (see Table S-3.8-2). As indicated in Reference 4, these gap fractions remain conservative for extended burnup fuel. Reference 5 agrees that the extended burnup fuel gap fractions for most isotopes of concern do not exceed the gap fractions specified in the Regulatory Guide.

Total core and gap inventories are presented in Table S-3.8-1. Total and gap activity for the highest rated discharged assembly are provided in Table S-3.8-2.

The inventories generated by the ORIGEN code are input to the FIPCO code (ref 6) to calculate the reactor coolant fission product inventories. Once again, parameters specific to the current equilibrium fuel cycle and improved methodology, rather than temperature and pressure, influence the reactor coolant source terms. The parameters used to calculate the reactor coolant fission and corrosion product inventories are presented in Table S-3.8-3. The resulting fission and corrosion product activities are presented in Table S-3.8-4.

The following evaluations were performed to determine the impact of operation at 3588 MWt on the radiological consequences of various accidents. The resulting dose estimates are presented in Table S-3.8-5.

S-3.8.3 Large-Break LOCA

The radiological consequences of the large break LOCA have been reanalyzed. The salient parameters used in the reanalysis are presented in Table S-3.8-6. The resulting offsite doses are presented in Table S-3.8-5.

The resulting doses are appropriately within the 10 CFR 100 guideline. Further, the doses are bounded by those presented in FSAR Table 14.3.5-7 for the Maximum Hypothetical Accident.

S-3.8.4 Fuel Handling Accident

An evaluation was performed to estimate the impact of the uprated power level on the radiological consequences of fuel handling accidents. The evaluation was based on the results presented in FSAR Sections 14.1.1, 14.1.2, and 14.3.5.4 and the nuclide inventories calculated specifically for the uprated power level.

The nuclide activities for the lead assembly assumed to be damaged in the fuel handling accident, based on core average inventory, 193 fuel assemblies, radial peaking factor (F delta h) of 1.65 and 100 hours decay, are presented in Table S-3.8-2.

The core nuclide activities for the uprated power were compared to the FSAR activities (Unit 1 FSAR Table 14.3.5-2). Because of the 100 hour decay period prior to fuel handling, the dominant dose contributors for the fuel handling accident are I-131 and Xe-133, and only these nuclides were considered for this evaluation. Also, actual activity, rather than dose equivalent activity, was utilized. A comparison of the iodine and noble gas activities shows an increase of approximately 21% in I-131 and an 11% increase in Xe-133. Thus, the fuel handling accident offsite thyroid and whole body doses are estimated to increase by 21 and 11 percent, respectively, over the doses presented in the FSAR.

Although doses have increased over those presented in the FSAR, there is no increase in the consequences of the accident. This determination is based on the fact that the resulting doses are appropriately within the 10 CFR 100 guideline.

S-3.8.5 Steam Generator Tube Rupture

The dose equivalent I-131 concentration in the reactor coolant, at the uprated power, is approximately 25% greater than the FSAR concentration. There is no significant difference in equivalent Xe concentration. Thus, the SGTR thyroid dose increase, due only to source term effects, is approximately 25%. Although the doses have increased there is no increase in the consequences of the event. This determination is based on the fact that the overall doses are small, being within a small fraction (30 rem thyroid and 2.5 rem whole body) of the 10 CFR 100 guideline.

S-3.8.6 References

1. "ORIGEN - The ORNL Isotope Generation and Depletion Code," M. J. Bell, Oak Ridge National Laboratory, ORNL-4628, May 1973.

2. "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay from ENDF/B," ORNL RSIC, ORNl DLC-38/ORYX-E, September 1975.

3. "Compilation of Fission Product Yields," Vallecitos Nuclear Center, M. E. Meek and B. F. Rider, NEDO-12154-1, January 1974.

4. WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985.

5. NUREG/CR-5008, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

6. "FIPCO, A Computer Code for Calculating the Distribution of Fission Products in Reactor Systems," WCAP-7949 (proprietary), August 1972.

TABLE S-3.8-1
FUEL PARAMETERS AND CORE AND GAP ACTIVITIES

Fuel Parameters:

Region	w/o U-235	No of Assy.	Specific Power (Mwt/MTU)	Fuel Mass (MTU)	Burnup (Mwd/MTU)	Total Power (Mwt)
16a	3.4	48	49.346	22.0576	20725	1088.45
15a	3.4	48	42.763	22.0576	38685	943.249
14a	3.4	1	3.26	0.4595	40054	1.498
16b	3.8	32	42.145	14.7051	17701	619.746
15b	3.8	32	36.591	14.7051	33069	538.074
14b	3.8	32	26.997	14.7051	44408	396.994
Total	--	193	40.456	88.69	--	3588

Cycle Length - 420 EFPD
Cycle Burnup - 16991 MWD/MTU
Core Average Burnup - 30764 MWD/MTU
Average Discharge Burnup - 40991 MWD/MTU

Core and Gap Activities:

Nuclide	Curies in Core ($x10^8$)	Fraction in gap (Note 1)	Curies in gap ($x10^5$)
I-129	14.7 kg	N/A	--
I-131	1.0	0.023	23.0
I-132	1.46	0.0026	3.8
I-133	2.0	0.0079	15.8
I-134	2.2	0.0016	3.52
I-135	1.9	0.0043	8.17
Xe-131m	0.0071	N/A	--
Xe-133m	0.29	0.0127	3.88
Xe-133	2.0	0.0185	37.0
Xe-135m	0.41	0.00096	0.39
Xe-135	0.42	0.0054	2.27
Xe-137	1.8	N/A	--
Xe-138	1.6	N/A	--
Kr-85m	0.26	0.0029	0.75
Kr-85	0.0083	0.2157	1.79
Kr-87	0.48	0.002	0.96
Kr-88	0.68	0.0029	1.97
Kr-89	0.84	N/A	--

Note 1: Unit 1, Table 14.3.5-2
N/A, Not Available from the FSAR

TABLE S-3.8-2

ACTIVITY IN THE HIGHEST RATED DISCHARGED ASSEMBLY
FOR THE RERATED POWER OF 3588 MWT
100 HOURS FOLLOWING REACTOR SHUTDOWN

Nuclide	(Note 1) Assembly Inventory	(Note 2) Calculated gap fraction	Curies in Gap	(Note 3) Reg Guide gap fraction	Curies in Gap
I-131	$6.14x10^5$	0.0476	$2.92x10^4$	0.1	$6.14x10^4$
I-132	$5.2x10^5$	0.00552	$2.87x10^3$	0.1	$5.2x10^4$
I-133	$6.38x10^4$	0.0168	$1.07x10^3$	0.1	$6.38x10^3$
I-135	43.5	0.00946	$4.1x10^{-1}$	0.1	$4.4x10^0$
Kr-85	$7.04x10^3$	0.447	$3.15x10^3$	0.3	$2.11x10^3$
Xe-131m	$5.82x10^3$	--	--	0.1	$5.82x10^2$
Xe-133M	$1.06x10^5$	0.0263	$2.79x10^3$	0.1	$1.06x10^4$
Xe-133	$1.21x10^6$	0.0383	$4.63x10^4$	0.1	$1.21x10^5$
Xe-135	$2.23x10^3$	0.011	$2.45x10^1$	0.1	$2.23x10^2$

Note 1: Lead assembly inventory = (total core inventory decayed for 100 hours/N) x RPF
where N = 193 assemblies and RPF (radial peaking factor) = 1.65.

For the Unit 1 FHA in the auxiliary building, FSAR Section 14.2.1.1, an RPF of 1.3 is specified. For the FHA in containment, FSAR Section 14.2.1.2, an RPF of 1.65 is specified.

Note 2: FSAR Table 14.2.1-2, Conservative Case, FSAR basis for Unit 1 FHA outside containment.

Note 3: Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors". FSAR basis for Unit 1 FHA inside containment and for Unit 2, inside and outside containment.

TABLE S-3.8-3

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

1.	Core thermal power, max. calculated, MWt	3588
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant mass, grams	2.41×10^{8}
4.	Reactor coolant average temperature in core, °F	590
5.	Purification flow rate (normal), gpm	75
6.	Effective cation dimineralizer flow, gpm	7.5
7.	Volume control tank volumes:	
	a. Vapor, ft^3	267
	b. Liquid, ft^3	133
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, Rb, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo, Tc and Ag isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, Zr, Nb, Ru, Rh, La, Ce, Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs--134, 136, 137, Y-90, 91 and Mo-99	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for Cs-134, 136, 137, Y-90, 91 and Mo-99	10.0

TABLE S-3.8-3 (Cont'd)

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

11. Volume control tank noble gas stripping fraction (closed system):

Nuclide	Stripping Fraction
Kr-85	7.3×10^{-5}
Kr-85m	6.1×10^{-1}
Kr-87	8.4×10^{-1}
Kr-88	7.1×10^{-1}
Xe-133	3.7×10^{-2}
Xe-133m	8.5×10^{-2}
Xe-135	3.5×10^{-1}
Xe-135m	9.5×10^{-1}
Xe-138	9.5×10^{-1}

TABLE S-3.8-4

REACTOR COOLANT EQUILIBRIUM FISSION AND
CORROSION PRODUCT ACTIVITIES

Nuclide	Activity (μCi/gram)	Nuclide	Activity (μCi/gram)
H-3	3.500E+00	I-129	5.078E-08
Cr-51	5.500E-03	I-131	2.850E+00
Mn-54	4.400E-04	I-132	2.748E+00
Mn-56	2.000E-02	I-133	5.370E+00
Fe-55	2.000E-03	I-134	5.937E-01
Fe-59	5.200E-04	I-135	2.396E+00
Co-58	1.500E-02	Xe-131m	1.827E+00
Co-60	1.900E-03	Xe-133	2.286E+02
Br-84	4.218E-02	Xe-133m	1.491E+01
Kr-85	8.546E+00	Xe-135	7.060E+00
Kr-85m	1.753E+00	Xe-135m	5.128E-01
Kr-87	1.092E+00	Xe-137	1.807E-01
Kr-88	3.787E+00	Xe-138	6.331E-01
Rb-88	8.139E+00	Cs-134	2.841E+00
Sr-89	4.293E-03	Cs-136	2.940E+00
Sr-90	1.541E-04	Cs-137	1.743E+00
Sr-91	6.075E-03	Cs-138	1.019E+00
Sr-92	1.217E-03	Ba-137m	1.648E+00
Y-90	2.221E-04	Ba-140	4.235E+03
Y-91	5.275E-04	La-140	9.254E-03
Y-91m	3.801E-03	Ce-141	6.255E-04
Y-92	1.143E-03	Ce-143	4.819E-04
Y-93	3.605E-04	Ce-144	4.199E-04
Zr-95	6.295E-04	Pr-143	5.911E-04
Nb-95	6.332E-04	Pr-144	4.190E-04
Mo-99	7.474E-01		
Tc-99m	7.842E-01		
Ru-103	5.730E-04		
Ru-106	1.645E-04		
Rh-103m	5.795E-04		
Rh-106	1.645E-04		
Ag-110m	1.718E-03		
Te-129	2.885E-02		
Te-129m	1.885E-02		
Te-131	1.226E-02		
Te-131m	2.606E-02		
Te-132	3.105E-01		
Te-134	3.036E-02		

TABLE S-3.8-5

ESTIMATED DOSES FOR 3588 MWT POWER OPERATION

Accident Description	Doses in rem
Fuel Handling Accident in the Auxiliary Building	
0-2 hour thyroid at SB	2.6
0-2 hour whole body at SB	0.6
Fuel Handling Accident in Containment	
0-2 hour thyroid at SB	100
0-2 hour whole body at SB	1.4
Steam Generator Tube Rupture	
0-2 hour site boundary	
thyroid	1.7
whole body	0.2
0-8 hour Low Population Zone (LPZ)	
thyroid	0.4
whole body	0.05
Large-Break LOCA*	
0-2 hour	
thyroid	134
whole body	2.4
0-30 day LPZ	
thyroid	126
whole body gamma	1.8

*Calculated doses

TABLE S-3.8-6

PARAMETERS USED TO EVALUATE THE OFFSITE DOSES DUE TO A LARGE-BREAK LOCA AT 3588 MWT

(page 1 of 4)

General

Core power level, MWt 3588

Source Term

Fifty percent of the core iodine is assumed to be uniformly distributed in the lower containment at time zero (TID-14844/Regulatory Guide 1.4)

I-131	5.0×10^{7} curies
I-132	7.3×10^{7}
I-133	1.0×10^{8}
I-134	1.1×10^{8}
I-135	1.9×10^{8}

Iodine Plate-out Factor 0.5

Iodine species

Elemental	0.91
Organic	0.04
Particulate	0.05

100 Percent of the core noble gas is released to containment.

Kr-85m	2.6×10^{7} curies
Kr-85	8.3×10^{5}
Kr-87	4.8×10^{7}
Kr-88	6.8×10^{7}
Xe-131m	7.1×10^{5}
Xe-133m	2.9×10^{7}
Xe-133	2.0×10^{8}
Xe-135m	4.1×10^{7}
Xe-135	4.2×10^{7}
Xe-138	1.6×10^{8}

TABLE S-3.8-6 (cont'd)

PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT
(page 2 of 4)

Containment Parameters

Volume of upper containment, ft^3	7.74×10^5
Volume of lower containment (Includes dead ended volumes)	3.62×10^5
Volume of ice beds	1.11×10^5
Containment leak rate	
0-24 hr, percent/day	0.25
>24 hr.	0.125
Containment Spray System	
Upper Containment	
Spray flow rate, gpm	1900
Spray fall height, ft	85
Lower Containment	
Spray flow rate, gpm	900
Spray fall height, ft	50
Air Steam Flow Rates, cfm	
0 - 10 min.	416,000 (average flow rate from lower to upper containment)
>10 min.	39,000 (recirculated between lower and upper containment through the ice beds)

TABLE S-3.8-6 (cont'd)

PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT
(page 3 of 4)

Iodine Removal Parameters	
Upper Containment	
Elemental iodine removal by spray, hr^{-1}	
injection spray (0 to 16 min)	10
recirculation spray (>20 min)	2.9
Particulate iodine removal by spray, hr^{-1}	6.8
Lower Containment	
Elemental iodine removal by spray, hr^{-1}	
injection spray	10
recirculation spray	2.5
Particulate iodine removal by spray, hr^{-1}	4.1
Ice Condenser Iodine removal efficiency	
0 - 10 min.	0
10 - 40 min.	0.3
>40 min.	0
Elemental iodine DF (includes the combined effects of sprays and the ice condenser)	100
Particulate iodine DF	100

TABLE S-3.8-6 (cont'd)

PARAMETERS USED TO EVALUATE THE OFFSITE DOSES
DUE TO A LARGE-BREAK LOCA AT 3588 MWT
(page 4 of 4)

Miscellaneous Parameters

Atmospheric dispersion factors at the site boundary and at the outer boundary of the low population zone (LPZ), sec/m^3, and breathing rates, m^3/sec:

	Site Boundary	LPZ
0-2 hr.	3.15×10^{-4}	7.5×10^{-5}
2-24 hr.	0	7.5×10^{-5}
1-5 days	0	2.6×10^{-6}
5-30 days	0	7.9×10^{-7}

	Breathing Rate
0-8 hr.	3.47×10^{-4}
8-24 hr.	1.75×10^{-4}
1-30 days	2.32×10^{-4}

Dose Conversion Factors — See Table S-3.8-7

TABLE S-3.8-7

COOK UNIT 1 FSAR TABLE 14.3.5-9 SHEET 5 OF 6
DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS*

Nuclide	Total Body $\frac{rem\text{-}m^3}{Ci\text{-}s}$	Beta Skin $\frac{rem\text{-}m^3}{Ci\text{-}s}$	Thyroid (rem/Ci)
I-131	NA	NA	1.49E+6
I-132	NA	NA	1.43E+4
I-133	NA	NA	2.69E+5
I-134	NA	NA	3.73E+3
I-135	NA	NA	5.60E+3
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-138	2.80E-1	1.31E-1	NA

*"Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 20 Appendix I," USNRC Regulatory Guide 1.109, Rev. 1, October 1977.

TABLE S-3.8-8

D. C. COOK UNIT 2 CYCLE 12

ASSUMED FUEL CYCLE (POWER LEVEL = 3588 MWT)

Region	w/o U-235	No. of Assemblies	Specific Power (Mwt/MTU)	Fuel Mass (MTU)	EOC Burnup (MWD/MTU)	Total Power (Mwt)
14A	3.69	36	54.169	15.2300	22751	825.0
13A	3.69	36	38.693	15.2300	39002	589.3
12A	3.69	1	32.724	0.423057	52746	13.8
14B	4.20	40	54.233	16.9223	22778	917.8
13B	4.20	40	47.671	16.9223	42800	806.7
12B	4.20	40	25.621	16.9223	53561	433.6
TOTAL	-	193	43.994	81.65	---	3588

Fuel Assembly Type = 17 x 17

Cycle Length = 420 EFPD Cycle Burnup = 18457 MWD/MTU

Core Average Burnup = 36484 MWD/MTU

Average Discharge Burnup = 46845 MWD/MTU

Assumed Discharge: 35 Assemblies (13A), 1 Assembly (12A), 40 Assemblies (12B)

Total Discharged = 76 Assemblies

S-3.9 POST-LOCA HYDROGEN PRODUCTION

S-3.9.1 Evaluation Summary

To support the Donald C. Cook Nuclear Plant Units 1 and 2 Rerating Program, hydrogen accumulation in the containment following a LOCA was investigated by AEPSC. Westinghouse provided input in the form of hydrogen generation rates for core radiolysis hydrogen, sump radiolysis hydrogen, corrosion-generated hydrogen, large break LOCA total zirconium-water reaction hydrogen, and the zirconium-water reaction, hydrogen generation rates as a function of time for the small break LOCA for these analyses (References 1, 2, and 3).

The hydrogen reanalysis was affected primarily by a higher radiolysis rate due to three factors:

1. The uprated power level considered for the rerating of 3600 MWt NSSS,

2. An increase in the assumed mass of zirconium in the core to bound future core designs, and

3. The fact that significant amounts of hydrogen were calculated by Westinghouse to be generated during the small break LOCAs as well as during large break LOCAs.

AEPSC analyzed the effects of these changes on overall containment hydrogen concentrations (in volume percent) and on minimum hydrogen skimmer system flow rates required to maintain hydrogen concentrations below four volume percent in the various containment subcompartments. The net result of these changes did not affect the overall containment analyses; that is, hydrogen concentrations were shown to be below four volume percent (Reference 4).

S-3.9.2 References

1. Letter AEP-88-256, "Post-LOCA Hydrogen Generation Analysis Report", dated May 23, 1988, by H. C. Walls.

2. Letter AEP-88-331, "Source Terms Analysis", dated July 26, 1988, by H. C. Walls.

3. Letter AEP-88-394, "(SBLOCA) Zirconium-Water Reaction and Energy Release to Containment", dated September 16, 1988, by H. C. Walls.

4. AEP:NRC:1067, "Reduced Temperature and Pressure Program Analyses and Technical Specification Changes," Attachment 3, dated October 14, 1988, M. P. Alexich (AEPSC) to T. E. Murley (USNRC).

S-3.10 PRIMARY COMPONENTS EVALUATION

Section 3.10 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," addresses the evaluation of all the primary components to support operation at full rated thermal power and the revised primary side temperatures and pressures considered for the Cook Nuclear Plant Unit 1.

Many of the evaluations contained in Section 3.10 of WCAP-11902 were performed using the most limiting parameters offered for a rerating program for both Donald C. Cook Nuclear Plant Units 1 & 2. Therefore, conservatively higher power levels were assumed.

The most conservative parameters considered for the rerating program incorporate an NSSS power level of 3600 MWt, and the upper and lower bound sets of primary temperatures corresponding to that power rating, which are described in Section S-2.1 of this Supplement to WCAP-11902.

The following sections describe the safety evaluations performed on the primary components for operation of Cook Nuclear Plant Units 1 & 2 at their respective rerated power levels and the revised temperatures and pressures listed in Section S-2.1.

S-3.10.1 Reactor Vessel

S-3.10.1.1 Reactor Vessel Structural Evaluation

Westinghouse has completed stress and fatigue calculations on all of the various governing locations in the Donald C. Cook Nuclear Plant Units 1 and 2 reactor vessels, which were analyzed in the reactor vessel stress reports, (References 4 & 8) for the effects of the rerating parameters listed in Table S-2.1-1, and for the associated NSSS design transients. Based on these calculations and reviews and comparisons of the two different reactor vessel designs, Westinghouse concludes that the operation of the reactor vessels under the most limiting conditions of the rerating is acceptable until the expiration of their 40 year design objective. All of the stress intensity and usage factor limits of the 1965 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through Winter 1966 for the Unit 1 reactor vessel, and the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through the Winter 1968 for the Unit 2 reactor vessel are still satisfied when the rerating is incorporated, with the exception of the 3Sm limit for CRDM housings and outlet nozzle safe end. However, the code permits the exceeding of 3Sm, provided plastic or elastic/plastic analysis criteria are met.

The significant results of the evaluations and reviews which were performed are described in Section 3.10.1, Volume 1 of WCAP-11902 (Reference 5).

S-3.10.1.2 Reactor Vessel Integrity

The efforts performed for Reactor Vessel Integrity for neutron embrittlement issues and described in Section 3.10.1.2 of WCAP-11902 "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," support operation of Cook Nuclear Plant Units 1 and 2 over the full range of rerated parameters described in Section S-2.1. The evaluation was performed assuming bounding values, for both Units 1 and 2, of the rerated power levels and revised temperatures and pressures described in Section S-2.1.

Tables S-3.10.1-1 and S-3.10.1-2 have been expanded with respect to WCAP-11902 to include information on fluence projections and surveillance capsule lead factors for Cook Unit 2.

In summary, rerating of Cook Units 1 and 2 will not have an adverse impact on reactor vessel integrity relative to neutron embrittlement issues. Although neutron flux rate changes will occur as a result of the rerating, the impact on surveillance capsule lead factors, heatup and cooldown curves, and PTS irradiation embrittlement measures is minimal. Related changes in systems parameters will neither adversely impact the safety requirements of 10CFR Part 50 - Appendix G nor the risk of vessel failure from PTS events.

TABLE S-3.10.1-1

FAST NEUTRON (E > 1.0 MeV) FLUENCE
PROJECTIONS FOR COOK NUCLEAR PLANT UNITS 1 AND 2
(n/cm^2)

Unit 1 (22.89 EFPY)	Upper Bound*	Lower Bound**
All plates; Weld 9-442	1.84×10^{19}	1.55×10^{19}
Welds 2-442B, 2-442C, 3-442A, 3-442C	1.19×10^{19}	1.01×10^{19}
Welds 2-442A, 3-442B	5.92×10^{18}	5.01×10^{18}
Unit 2 (21,38 EFPY)	**Upper Bound***	**Lower Bound****
All Plates: C-1	1.76×10^{19}	1.52×10^{19}
Welds IS-1, IS-2	1.08×10^{19}	9.28×10^{18}
Welds LS-1, LS-2	5.68×10^{18}	4.88×10^{18}

*3588 MWt for both units; 547°F Downcomer.
**3250 MWt Unit 1/3411 MWt Unit 2; 512°F Downcomer.

TABLE S-3.10.1-2

SURVEILLANCE CAPSULE LEAD FACTORS FOR
COOK NUCLEAR PLANT UNITS 1 AND 2

	4° Capsules	40° Capsules
Unit 1/Unit 2 Base Case (3250 MWt, 536°F Downcomer)	1.3	4.2
Unit 1/Unit 2 At Uprated Power Upper Bound (3588 MWt for both units; 547°F Downcomer)	1.3	4.2
Unit 1/Unit 2 Current Licensed Power Lower Bound (3250 MWt Unit 1/3411 MWt Unit 2; 512°F Downcomer)	1.3	4.4

S-3.10.2 Reactor Internals

The efforts performed for Reactor Internals integrity and described in Section 3.10.2 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," support operation of Cook Nuclear Plant Units 1 and 2 over the full range of rerated parameters described in Section S-2.1. The evaluation was performed assuming bounding values, for both Units 1 and 2, of the rerated power levels and revised temperatures and pressures described in Section S-2.1.

Additional evaluations of the reactor internals were performed for Unit 2, assuming Westinghouse 17x17 Standard Fuel, and are summarized below. Unit 2 currently (Fuel Cycle 7) contains ANF Fuel. The Cycle 8 and 9 reloads will be Westinghouse 17x17 Vantage 5 Fuel. Evaluations of the Unit 2 reactor internals will be performed as part of the Cycle 8 reload.

1. The calculated core bypass flow is within the limit of 4.5% which was specified for the rerating parameters for Unit 2, assuming Westinghouse 17x17 Standard Fuel.

2. The hydraulic uplift forces acting on the lower internals used to evaluate the contact forces between the vessel ledge and lower internals have increased for the rerating parameters compared to the original lifting forces. The contact force is maintained during the rerating steady state conditions for both Units 1 and 2. However during a hot pump overspeed condition (which is short term) this contact force is marginal for Unit 2, meaning there is potential for the lower internals to lift off during this condition. Fuel specific evaluations will be performed as a part of the Unit 2, Cycle 8 reload effort to address this issue. For Unit 1, there is adequate contact margin even for the hot pump overspeed condition to eliminate a liftoff concern.

3. Since Cook Unit 2 employs a non-Westinghouse fuel type, it is not possible to define the limiting RCCA drop time for use in the accident reanalyses for Unit 2. However, the limiting RCCA drop time-to-dashpot entry that is applicable to units with Westinghouse 17x17 standard fuel is 2.2 seconds. This drop time would remain limiting (for Westinghouse 17x17 standard fuel units) for those design operating conditions characteristic of the rerating program with or without fuel assembly thimble plugging devices.

4. The discussion presented in Section 3.10.2 (Item 4) in WCAP-11902 on the potential for flow induced vibration is applicable for Cook Nuclear Plant Units 1 and 2 at the rerated conditions.

5. Stresses and fatigue usage for the limiting components of the upper and the lower internals were evaluated for the changes in design transients under the rerating conditions and are within acceptable limits.

The analysis of the reactor internals was primarily affected by the revised temperatures and pressures, with the uprated power being a minor contributor.

S-3.10.3 Steam Generators

S-3.10.3.1 Thermal-Hydraulic Performance Evaluation

Thermal hydraulic characteristics of the steam generators, the circulation ratio, thermal-hydraulic stability and secondary mass were calculated for the steam generators at each of the rerating conditions and projected steam generator tube plugging levels.

Circulation ratio is primarily a function of power. For the uprated conditions for Unit 1 the circulation ratio is slightly smaller but is acceptable. The circulation ratio in the replacement steam generators in Unit 2 have a lower circulation ratio than the original steam generators. A review of flow studies find the circulation ratio for all projected conditions acceptable based on favorable flow velocities and flow distributions.

The damping factor characterizes the thermal-hydraulic stability of the units at the various operating conditions. A negative damping factor indicates a stable unit. That is, small perturbations of pressure or circulation ratio will die out rather than grow in amplitude. For Unit 1, all damping factors are negative, with approximately the same level as the current operating conditions. The damping values for Unit 2 are more negative than the original steam generators. All operating conditions under consideration are, therefore, stable.

The values of secondary mass for the various operating conditions were calculated. At the lower primary temperatures which correspond to lower steam pressure, the secondary mass is reduced by 5-10% below current values in Unit 1. The secondary mass for the replacement steam generators in Unit 2 is slightly lower than Unit 1, for the same conditions. The primary influence of secondary mass on the safe operation of the steam generator is due to the water level in the steam generator during design transients and postulated accident conditions. The safety analysis of Cook Nuclear Plant Units 1 and 2 (presented in Sections 3.3 and S-3.3) have accounted for the range of secondary masses possible for the various operating conditions.

S-3.10.3.2 U-Bend Tube Fatigue Evaluation

The discussion of the evaluation of flow in the U-bend region of Donald C. Cook Nuclear Plant Unit 1 in Volume 1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," was based on a study which included flow conditions representing rerated power conditions for Unit 1. Since the uprated power level of 3425 MWt was included in the original evaluation, no additional discussion is required in this supplement. Some conditions evaluated had induced flow conditions slightly worse than the original design conditions. Any evaluation of the effect of the rerating on the information submitted by AEPSC in response to USNRC Bulletin 88-02, to verify that operation of the Cook Nuclear Plant Unit 1 steam generators under the new operating conditions is in compliance with recent NRC requirements relative to U-bend tube fatigue, is separate from this report.

An evaluation of the potential for tube fatigue due to fluid elastic vibration is not required for Unit 2. The replacement steam generators in Unit 2 have stainless steel support plates. The NRC required these evaluations only for units with carbon steel support plates, reference USNRC Generic Letter 88-02.

S-3.10.3.3 Corrosion

The discussion in WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," of the estimate of the steam generator corrosion propensity calculated for the rerating of Donald C. Cook Nuclear Plant Unit 1 includes conditions corresponding to the current power level and some conditions at the uprated power level of 3425 MWt. For this supplement, the corrosion propensity for Unit 1 has been determined for additional sets of conditions at the uprated power level.

As discussed in WCAP 11902, the computation of corrosion propensity is based on an algorithm which considers plant features, operating conditions, and operating chemistry. For consideration of the effect of rerating on corrosion propensity the plant features remain constant and the operating chemistry is

assumed to be at the upper level of the Steam Generator Owners Group (SGOG II)/Westinghouse guidelines. Consideration of changes in operating conditions due to rerating includes the effect of temperature changes and stress effects due to primary to secondary pressure differences.

The evaluation for this supplement considered pressure and temperature conditions at a uprated NSSS power level of 3425 MWt for Unit 1 in addition to those used for WCAP 11902. The evaluation determined that, as in the previous evaluation for the current Unit 1 power level of 3250 MWt, the propensity for corrosion at the lower bound temperature conditions and rerated power level is judged to be within an acceptable range. At the rerated power level and upper bound temperature conditions for Unit 1, the corrosion propensity is judged to be within the unacceptable range. For both power levels the corrosion propensity for the reduced temperature conditions is approximately half that for the current operating conditions used as a baseline.

An evaluation using the method described in WCAP 11902 has been made of the corrosion propensity for the replacement steam generators in Unit 2. This evaluation was made at NSSS power levels of 3425 MWt and 3600 MWt at the various pressures and temperatures considered for the rerating assuming 10% tube plugging. For these cases, the corrosion propensity was judged to be acceptable.

S-3.10.3.4 Structural

The structural evaluation described in WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," was done based on enveloping conditions which bounded the rerated conditions and the revised temperature and pressure operation. The structural analysis, which also bounds the uprated power conditions, was primarily affected by the revised temperature and pressure conditions. Any effects of uprated power independent of pressure and temperature are minor contributors to the analysis. No additional structural evaluation is required to support operation of the Unit 1 steam generators at the rerated power conditions.

The structural analyses for the Unit 2 replacement steam generators was performed in two portions. The portions of the steam generator to be replaced were analyzed in a stress report as required by the ASME Boiler and Pressure Vessel Code. The portions of the steam generator replaced are the primary chamber, tubesheet, lower shell, transition cone with a portion of the upper shell, wrapper and tube bundle. New feedrings were also provided. These sections were analyzed using transients and conditions consistent with the nominal and uprated power conditions. The portions of the steam generator which were reused with the new sections were reanalyzed using revised parameters and transients which bound the rerating operation.

The stress report for the new portion of the steam generators includes a compilation of stress calculations which verify the structural integrity of the replacement steam generators. The calculations demonstrate that the design meets the requirements of the ASME Code.

The evaluation of the upper shell for the rerating conditions considered the main feedwater nozzle, the secondary manway, and the steam nozzle components. These components (per the previous analysis) are the limiting structural members for the upper shell region. In each case, the analyses were performed utilizing the existing analyses of these components, scaling stresses and updating the fatigue calculations as necessary. Primary stresses and maximum stress ranges are not impacted by the rerating conditions, and these calculations were not repeated in the reanalysis. In general, the impact of rerating on the upper shell is in terms of the variation in steam pressure and temperature.

In reviewing the transient parameters for the rerating conditions, the four sets of parameters reduce to just two sets of conditions for the upper shell components, which are unaffected by variations in T_{hot}, T_{cold}, and primary side pressure. The two sets of conditions correspond to high and low temperature, respectively. The high temperature conditions are very close to the transient conditions used in the reference analyses, and the resulting fatigue usages show only slight variations from the reference conditions. However, the low temperature conditions can result in increased fatigue

usages. This is the result of the lower steam pressure during normal operation. The lower pressure causes a reduction in the corresponding stresses at normal operation, which results in a higher stress range when cycling between hot shutdown, because hot shutdown pressure is the same for both reference and rerate operations.

A comparison has been made between the fatigue values for the reference (100% power) analyses and the values corresponding to rerating conditions. These results show the ASME Code usage factor limit of 1.0 to be satisfied in all cases. The results for the secondary manway shell penetration show a reduction in the fatigue usage from the reference value. This is because the reference analysis used a more conservative grouping of the transient cycles than is used for this analysis. The steam nozzle appears to be unaffected by rerating. This is due to the conservative manner in which the transient cycles were umbrellaed in the reference analysis. All transient conditions are bounded by the highest stress condition, such that lowering of the stress at normal operation does not affect the maximum stress range.

In conclusion, the previously used and replacement components of the steam generators in the Cook Nuclear Plant Unit 2 are found to satisfy the requirements of the ASME B&PV Code, Section III, for this program. Therefore the structural adequacy of the steam generators has been demonstrated for the rerated conditions for Unit 2.

S-3.10.4 Pressurizer

The pressurizer analysis/evaluation was performed with the input assumptions which are bounding for both Cook Units 1 and 2 at their uprated power levels and at the revised temperatures and pressures described in Section S-2.1. The efforts performed for the pressurizer are described in Section 3.10.4 , Volume 1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," and are applicable for both units. No impact to the satisfaction of applicable ASME Code criteria has resulted from the modifications associated with the rerating program.

S-3.10.5 Reactor Coolant Pumps (RCP)

The efforts performed for the Reactor Coolant Pumps and described in Section 3.10.5 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," support operation of Cook Nuclear Plant Units 1 & 2 over the full range of rerated parameters described in Section S-2.1. The analyses and evaluations were performed assuming bounding values, for both Units 1 & 2, of the rerated power levels and revised temperatures and pressures described in Section S-2.1. The evaluation of the reactor coolant pumps was primarily affected by the revised reactor coolant temperatures, with the uprated power being a minor contributor. The conclusions of Section 3.10.5 of WCAP-11902, therefore, are applicable for Cook Units 1 & 2 at their respective rerated powers and the revised temperatures and pressures.

S-3.10.6 Reactor Coolant Piping and Supports

The efforts performed for the reactor coolant piping and supports described in Section 3.10-6 of WCAP-11902 are bounding for Donald C. Cook Nuclear Plant Units 1 and 2 reduced temperature and pressure operation.

The intent of this evaluation is to show that the temperature changes associated with the Cook Units 1 and 2 rerating have only an insignificant impact on the existing design basis analyses. The design parameters were used as the basic input into the evaluation for the Reactor Coolant Loop. The effects of a change in the current design temperatures in the primary loop piping was the focus of this evaluation. The power reratings, as such, do not impact the evaluation except as they affect temperature. For the hot leg, the normal operating temperature range considered was 579.1°F to 615.2°F compared to the original values of 599.3°F for Unit 1 and 606.4°F for Unit 2. For the cold leg, the normal operating temperature range considered was 511.7°F to 547.4°F compared to the original values of 536.3°F for Unit 1 and 541.3°F for Unit 2.

The impact of the rerated conditions on the Cook Nuclear Plant has been reviewed and evaluated to determine if any changes are required in the analyses for the components considered. The reactor coolant loop analysis was reconciled to the new rerated conditions by considering the conservatisms and assumptions used in the original analyses as well as criteria changes that have added margin using the original design basis (notably Leak Before Break). For those rerated conditions where the primary loop temperatures exceed the original design basis temperatures by more than 5°F, an inspection of the support gaps is strongly recommended. The temperatures which affect the analysis are the hot and cold leg temperatures. The original design hot and cold leg temperatures are: Unit 1: T_{hot} = 599.3°F, T_{cold} = 536.3°F; Unit 2: T_{hot} = 606.4°F, T_{cold} = 541.3°F. If an interference is found in the lateral supports, the support gaps should be reshimmed to eliminate it. Since the reactor coolant loop analysis results have been reconciled to the rerated conditions, no new analysis is necessary. If the loop analysis does

not change, then the primary loop piping, the primary equipment supports, and the primary equipment nozzles are acceptable, and the existing analysis results envelope the rerated conditions.

S-3.10.7 Control Rod Drive Mechanism

The efforts performed for the control rod drive mechanism evaluation, and described in Section 3.10.7, Volume 1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," were performed to support operation of the Donald C. Cook Nuclear Plant Units 1 and 2 over the full range of rerating parameters described in Section S-2.1. The evaluation of the control rod drive mechanism was primarily affected by the revised primary temperatures and pressures, with the uprated power being a minor contributor. Therefore, the conclusions presented in WCAP-11902 for the Control Rod Drive Mechanism are valid for Cook Nuclear Plant Units 1 & 2 at their respective rerated core power levels.

S-3.10.8 Conclusions

This safety evaluation was limited to an evaluation of all the primary components to support operation of Donald C. Cook Nuclear Plant Units 1 & 2 at their respective rerated power levels and at the revised temperatures and pressures described in Section S-2.1.

The Program constitutes a change to the FSAR of the plant. As required by 10CFR50.59 an evaluation of this change was conducted and constitutes the balance of this evaluation. It has been determined that:

1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;

2) the probability for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created; and

3) the margin of safety as defined in the basis for any technical specification is not reduced.

These determinations are reconciled by reviews of the revised design power capability parameters and NSSS design transients associated with the rerating program and its impact on the existing design basis for the components of the Reactor Coolant System.

In conclusion, this safety evaluation of the mechanical effects on the primary components of the Reactor Coolant System provides assurance that the Cook Nuclear Plant Units 1 and 2 can operate safely at their uprated power levels and at the revised temperatures and pressures described in Section S-2.1.

S-3.10.9 References

1. Letter AEP-88-232, "RCS Flow Definitions," dated May 2, 1988, by H. C. Walls.

2. "Consolidated Finalized NSSS Component Design Transients for Phase-1 of D. C. Cook Units 1 and 2 Reduced T-hot/Rerating Program," Westinghouse letter dated February 25, 1988.

3. System Standard Design Criteria (SSDC) 1.3, Rev. 1, dated April, 1971.

4. Combustion Engineering, Inc. Report No. CENC-1162, "Analytical Report for Indiana & Michigan Electric Company Donald Cook Nuclear Power Plant Unit No. 1 Reactor Vessel," dated July 1971.

5. WCAP-11902, "Reduced Temperature and Pressure Operation for the D. C. Cook Nuclear Plant Unit 1 Licensing Report".

6. WCAP-11967, "Thot Reduction/Rerating Reactor Vessel Evaluation...Cook Unit No. 1," dated August, 1988.

7. WCAP-11968, "Thot Reduction/Rerating Reactor Vessel Evaluation...Cook Unit No. 2," dated August, 1988.

8. Chicago Bridge and Iron Company Stress Report for the American Electric Power, Donald C. Cook #2 Nuclear Station, 1734. Pressurized Water Reactor Vessel, CBI Contract Number 68-3262, dated 12/75.

S-3.11 FLUID AND AUXILIARY SYSTEMS EVALUATIONS

S-3.11.1 Introduction

This Section provides a discussion of the evaluation of the fluid and auxiliary systems for the Donald C. Cook Nuclear Plant Units 1 & 2 performed for the Rerating program. The rerated parameters and the current operating parameters are described in Section S-2.1.

In order to provide licensing support for operation of the Cook Nuclear Plant Units 1 & 2 at the rerated power levels and revised temperatures and pressures, it is necessary to evaluate the following systems at the revised conditions: 1) Chemical and Volume Control System, 2) Safety Injection System, 3) Residual Heat Removal System, 4) Spent Fuel Pool Cooling System, 5) Waste Disposal System, 6) Auxiliary Feedwater System and 7) Main Steam System.

The brief description of each system and its intended function(s) provided in WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," is applicable to both Cook Nuclear Plant Units 1 and 2 at the rerated conditions.

S-3.11.2 Discussion of Evaluations Performed

This evaluation addresses the impact of operating the Cook Nuclear Plant Units 1 and 2 at their respective rerated power levels and the revised primary temperatures and pressures. The NSSS power level is increased from 3262 to 3600 MWt, the upper and lower bound vessel average temperature values are 581.3°F and 547.0°F, respectively, and the primary pressure can be either 2100 or 2250 psia.

The system and component aspects of the fluid and NSSS/balance-of-plant interface systems listed above, were evaluated to ensure that operation of the units at the rerated power levels and revised primary temperatures and pressures, remains within the licensing basis of Cook Nuclear Plant Units 1 and 2.

S-3.11.2.1 Fluid Systems Evaluation

The impact of the proposed rerated power levels and revised primary temperatures and pressures, on the ability of the Westinghouse-designed fluid systems to perform their intended functions has been evaluated for the CVCS, SIS, RHRS, SFPCS and WDS.

Chemical and Volume Control System (CVCS)

The evaluation and conclusions presented in WCAP-11902 for the Cook Nuclear Plant Unit 1 Reduced Primary Temperature and Pressure Program remain valid for both units at the rerated conditions, with one change to the evaluation. The previous Unit 1 evaluation considered variations in T-cold from 514.9°F to 545.2°F. However, this evaluation for Cook Units 1 and 2 considers variations in T-cold from 511.7°F to 547.3°F.

The CVCS provides for boron injection, chemical additions for corrosion control, reactor coolant clean-up and degassification, reactor coolant make-up, reprocessing of water letdown from the Reactor Coolant System (RCS), and Reactor Coolant Pump (RCP) seal water injection.

Reactor coolant is discharged to the CVCS from a reactor coolant loop cold leg. The coolant flows through the shell side of the regenerative heat exchanger. The coolant temperature is reduced due to the transfer of heat loads to the charging flow which passes through the tubes. The coolant experiences a large pressure reduction as it passes through the letdown orifice. The cooled, low pressure water then leaves the reactor containment and enters the auxiliary building. A second temperature reduction occurs in the tube side of the letdown heat exchanger followed by a second pressure reduction due to the low pressure letdown valve.

The regenerative and letdown heat exchangers are designed to cool letdown flow from T_{cold} to 115°F. This reduction in temperature is required to ensure that the normal Rector Coolant Pump (RCP) seal injection temperature value of 130°F will be maintained, including an allowance for a 15°F temperature rise across the centrifugal pump. The variations in T_{cold} considered for this rerating program, from 511.7 to 547°F, are bounded by the design inlet temperature of 547°F for the regenerative heat exchanger. Therefore, the cooling temperature of the letdown line is met with the revised operating parameters considered for this program.

The letdown line is designed to reduce the static pressure of the reactor letdown stream from the RCP suction pressure to the Volume Control Tank (VCT) operating pressure, such that the design pressure of intervening piping and components is not exceeded and fluid is maintained in a subcooled condition throughout the system. The primary pressure reduction is taken across the letdown orifices. A pressure control valve, PCV-131, ensures that adequate back pressure is maintained on the letdown orifices to ensure subcooled fluid conditions. The pressurizer pressures considered for this program (2100 psia or 2250 psia) are bounded by the design pressurizer operating pressure. In addition, it has been verified that PCV-131 is capable of maintaining sufficient backpressure on the letdown orifices to ensure subcooled fluid conditions when the pressurizer pressure is reduced to 2100 psia. Therefore, the pressure reduction function of the letdown line is verified with the revised operating parameters proposed by this program.

Safety Injection System (SIS)

The Safety Injection System provides emergency core cooling in the event of a break in either the Reactor Coolant System (RCS) or the Main Steam System (MSS). The Unit 1 analyses from WCAP-11902 demonstrated the acceptability of the SIS for its accident function. These analyses bound the uprated conditions. Similar analyses for Unit 2 will be performed as part of the fuel reload.

Residual Heat Removal System (RHRS)

The RHRS is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the Reactor Coolant System (RCS) are approximately 400 psig and less than 350°F, respectively. Under normal operating conditions, the RHRS is designed to reduce the temperature of the reactor coolant to 140°F within 20 hours following reactor shutdown. The 20 hour cooldown is the original design basis of the RHR system, which is based on two RHR heat exchangers and pumps in service and that the heat exchangers are being supplied with 5000 gpm of component cooling water at 95°F. It has been verified that the uprated power does not adversely affect the capability of the RHRS to reduce the reactor coolant temperature to 140°F within the 20 hour limit with both trains operating. In the event of a train failure, the RHRS is designed to reduce the reactor coolant temperature to 200°F within 36 hours after reactor shutdown. It has been verified that the RHRS is also capable of meeting the original cooldown design basis.

Spent Fuel Pool Cooling System (SFPCS)

The primary function of the SFPCS is to remove decay heat which is generated by the spent fuel pool elements stored in the pool. Decay heat generation is proportional to plant power level. The plant NSSS power level has increased, thus, the demands on the SFPCS have also increased.

During normal operation, with up to forty percent of the core from each unit stored in the pool, the SFPCS is expected to maintain the pool temperature below 120°F. It was determined that the pool temperature, under these conditions with 100°F component cooling water supplied to the spent fuel pit heat exchanger, would be 117°F. With only one cooling train in operation, the pool temperature is expected to be maintained below 140°F. It was determined that the pool temperature under these conditions with 100°F component cooling water supplied to the spent fuel pit heat exchanger, would be 133°F. Under the maximum anticipated heat loading forty percent of a core from each unit plus a full core from Unit 2 and only one cooling train available, the temperature is expected to be maintained below the point where the pool would boil (212°F). It was determined that under these conditions with 100°F component cooling water supplied to the spent fuel pit heat exchanger the pool temperature would be 167°F, which is well below boiling. Therefore, it has been verified for the increased power levels, that the pool temperature can be maintained within all of the values described.

If all cooling is lost and forty percent of a core from each unit is stored in the pool, the time it would take the spent fuel pool to heat up from 120°F to 180°F is approximately 16 hours. With one complete core added, the time is reduced to approximately 8 hours. It has been verified that the spent fuel pool heat up rates for a complete loss of cooling are acceptable.

Waste Disposal System (WDS)

The evaluation and conclusions provided in WCAP-11902 for the Unit 1 Reduced Temperature and Pressure Program remain valid for both Cook Nuclear Plant Units 1 and 2 at the rerated conditions.

S-3.11.2.2 Auxiliary Equipment Evaluation

The evaluation and conclusions provided in WCAP-11902 for the Unit 1 Reduced Temperature and Pressure Program remain valid for both Cook Nuclear Plant Units 1 and 2 at the rerated conditions.

The Cook design transients were modified to reflect the proposed power capability parameters and reviewed to demonstrate that the revised power level, primary temperatures and pressures, have no adverse effect on the integrity, operability or qualification of the auxiliary systems' components.

In some cases, the procurement of auxiliary equipment for early vintage plants, such as Cook, predates the ASME Code. Pumps, valves, tanks, and heat exchangers were procured according to the design conditions specified in data sheets or equipment specifications. The temperature and pressure that the components were expected to experience, enveloped the specified design conditions. Note, the design and operating parameters are not necessarily dependent on full-power primary system temperature, pressure, or NSSS power level.

The NSSS design transients for Cook Units 1 and 2 are similar to the transients shown in the System Standard Design Criteria (SSDC) 1.3, Revision 1. The auxiliary tanks typically have insignificant transients and, therefore, thermal fatigue is not a concern. In addition, thermal fatigue failures are uncommon on auxiliary equipment. The most important parameters that were considered to affect the integrity of auxiliary components for the Cook units are the temperatures and pressures during normal operating conditions. Detailed evaluations were performed and it was determined that the conditions originally specified for procurement of the auxiliary equipment, either remain unchanged or are bounding for the proposed revised conditions.

S-3.11.2.3 NSSS/Balance-of-Plant Interface Systems Evaluation

The evaluation and conclusions provided in WCAP-11902 for the Unit 1 Reduced Temperature and Pressure Program remain valid for both Donald C. Cook Nuclear Plant Units 1 and 2 at the rerated conditions.

The potential impact of an increased power rating and reduced RCS temperature and pressure on the ability of Westinghouse designed NSSS/balance-of-plant interface systems to perform their required functions has been evaluated for the Main Steam and Auxiliary Feedwater Systems.

Main Steam System (MSS)

The minimum relieving capacity of the steam generator safety relief valves requires that the combined ASME rated capacity of the valves be sufficient to relieve the amount of steam generated in the worst case loss of heat sink, without causing the MSS pressure to exceed 110 percent of the system design pressure. The combined ASME rated capacity is equivalent to 16.380×10^6 lbs/hr at an inlet pressure of 1133 psia. For the new conditions, 15.088×10^6 lbs/hr of steam is generated at a system pressure of 1133 psia in the worst case loss of heat sink. Note that the inlet pressure of 1133 psia is conservative with respect to 110 percent of the system design pressure (1.10 x 1100 psia = 1210 psia).

Since the combined ASME rated capacity exceeds the required relieving capacity, it is determined that the Cook steam generator relief valve minimum relieving capacity requirement is met with the new conditions proposed by this program.

The maximum actual relieving capacity of a single steam generator safety relief valve is limited to 890,000 lbs/hr at an inlet pressure equivalent to the system design pressure. The limit is imposed to prevent unacceptable high rates of heat removal due to a stuck-open relief valve. The actual capacity for each Cook relief valve is equivalent to 883,513 lbs/hr at the system design pressure of 1100 psia.

Since the actual capacity of the relief valves does not exceed the 890,000 lbs/hr capacity limit, it is determined that the maximum actual relieving capacity of the Cook steam generator safety relief valves is acceptable for the new conditions of the program.

Auxiliary Feedwater System (AFWS)

The AFW pumps draw suction from a 500,000 gallon Condensate Storage Tank (CST). The current Cook Technical Specifications indicate that 175,000 gallons of the total CST volume are to be reserved for AFWS functions. The 175,000 gallons of AFW storage volume is to ensure that sufficient AFW is available to keep the plant in hot standby conditions for nine hours following a reactor trip. A minimum AFW volume of 146,000 gallons is required by the new conditions to maintain the plant at hot standby for nine hours with one Reactor Coolant Pump (RCP) operating. Therefore, with the new conditions of the program, the 175,000 gallon minimum AFW volume remains adequate to satisfy the original requirement for nine hours of hot standby operation, with or without the operation of one RCP.

S-3.11.3 Conclusions

The potential safety impact of the Rerated Power and Revised Primary Temperature and Pressure Program on the fluid and NSSS/BOP interface systems and components has been addressed by this safety evaluation. It is determined that the new range of primary temperatures and pressures shown in Table S-2.1-1 has no adverse affect on the ability of the fluid and NSSS/BOP interface systems to perform their required functions. It has also been demonstrated that the new design transients, revised to reflect the new conditions, have no adverse affect on the integrity, operability or qualification of the auxiliary system components.

Based on the results of the Fluid System, Auxiliary Equipment and NSSS/BOP interface system evaluations, it has been determined that:

- the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;

- the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created; and

- the margin of safety as defined in the basis for any technical specification is not reduced.

Thus, implementation of the Rerated Power and Revised Primary Temperature and Pressure Program does not impair the design and operation of the auxiliary and NSSS/BOP interface systems and their components, is not expected to lead to any event that would exceed the bounds of prior analyses and, therefore, does not produce an unreviewed safety question.

S-3.12 FUEL STRUCTURAL EVALUATION

S-3.12.1 Fuel Assembly Structural Evaluation

The efforts performed for the fuel assembly structural evaluation, and described in Section 3.12, Volume 1 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," were performed to support operation of the Cook Nuclear Plant Units 1 and 2 over the full range of rerating parameters described in Section S-2.1. The conclusions presented in that submittal for Unit 1 thus remain valid at uprated conditions. It should be noted that the structural evaluation for the fuel assembly is primarily affected by primary temperature reduction, with increased power being a minor contributor.

The fuel assembly structural evaluation for Cook Unit 2 will be provided as part of the Vantage 5 Reload Submittal.

S-3.12.2 Fuel Rod Structural Evaluation

The efforts performed for the fuel rod structural evaluation, and described in Section 3.12.2, Volume 1 of WCAP-11902, were performed to support operation of Cook Nuclear Plant Unit 1 over the full range of rerating parameters corresponding to operation at a maximum power level of 3425 MWt NSSS. These analyses have demonstrated that fuel rod design bases can be supported for operation at power levels up through 3425 MWt NSSS.

For Cook Nuclear Plant Unit 2, the fuel rod structural integrity will be addressed as part of a later submittal, incorporating Westinghouse Vantage 5 fuel.

S-3.13 TECHNICAL SPECIFICATION/REACTOR TRIP & ESF IMPACT

As a result of the analysis and evaluation efforts performed for the Donald C. Cook Nuclear Plant Rerating program, certain changes to the plant Technical Specifications are appropriate. The purpose of this report section is to document the basis for the Technical Specification changes which are recommended as a result of the rerating effort for Cook Nuclear Plant Unit 1. As discussed in the Executive Summary section of this Supplement to WCAP-11902, there are additional efforts necessary which were not performed as an integral part of the Rerating Program to support licensing of the uprating for Cook Nuclear Plant Unit 2. The Technical Specification changes and their bases for Unit 2 will be presented with this additional effort when it is performed.

The basis for the Technical Specification changes which are recommended as a result of the Reduced Temperature and Pressure program are discussed in Section 3.14 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT." The basis for reactor protection system/ESF changes necessary for operation of Cook Nuclear Plant Unit 1 at the revised temperatures and pressures discussed in Section 2.1 of WCAP-11902 is provided in Table 3.14-1 of WCAP-11902. The basis for reactor protection system/ESF changes necessary for operation of Cook Nuclear Plant Unit 1 at the rerated NSSS power level of 3425 MWt is provided in Table S-3.13-1 of this section. Additional Technical Specification changes not directly related to the reactor protection system/ESF are tabulated, and the basis provided, in Table 3.14-2 of WCAP-11902 and in Table S-3.13-2 of this section.

TABLE S-3.13-1
REACTOR PROTECTION SYSTEM/ESF SETPOINTS

Function	Nominal Setpoint	Allowable Value	Limiting Analysis
Table 2.2-1 pages 2-7 through 2-9			
a. Overtemperature delta-T reactor trip[1]	*	**	Protection of core thermal safety limits, non-LOCA-related DNB events.
b. Overpower delta-T reactor trip[2]	***	****	non-LOCA Protection of thermal safety limits, steamline break M/E release outside containment.

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[1]Overtemperature delta-T $\leq$ delta-T_o $[K1-K2 [\frac{1+t_1S}{1+t_2S}] (T-T')+K3(P-P')-f_1(\text{delta-I})]$

delta-T_o = indicated delta-T at Rated Thermal Power

[2]Overpower delta-T $\leq$ delta-T_o $[K4-K5 [\frac{t_3S}{1+t_3S}] T - K6(T-T'')-f_2(\text{delta-I})]$

delta-T_o = indicated delta-T at Rated Thermal Power

*K1 = 1.32
K2 = 0.0230
K3 = 0.00110
T = average temperature, °F
T' = indicated Tavg at Rated Thermal Power ($\leq$ 578.7 °F)
P = pressurizer pressure, psig
P' = 2235 psig (for 2235 psig operation)
= 2085 psig (for 2085 psig operation)
t_1 = 22 seconds
t_2 = 4 seconds
f_1(delta-I) = 0 for delta-I between (-37, +2)
Positive slope of f_1(delta-I) = 2.17%/% for delta-I > 2%
Negative slope of f_1(delta-I) = 0.33%/% for delta-I < -37%

**Max. trip setpoint shall not exceed its computed trip setpoint by more than 3.2% delta-T span.

***K4 = 1.083
K5 = 0.0177/°F for increasing Tavg
= 0 for decreasing Tavg
K6 = 0.0015 for T > T''
= 0 for T $\leq$ T''
T = average temperature, °F
T'' = indicated Tavg at Rated Thermal Power ($\leq$ 578.7 °F)
t_3 = 10 seconds
f_2(delta-I) = 0

****Max. trip setpoint shall not exceed its computed trip setpoint by more than 2.1% delta-T span.

TABLE S-3.13-1 (Cont'd)
REACTOR PROTECTION SYSTEM/ESF SETPOINTS

Function	Nominal Setpoint	Limiting Analysis
Table 3.4-4		
c. Steam Flow in Two Steam Lines -- High, SLI pages 3/4 3-24 3/4 3-26	≤ A function defined as follows: A delta-P corresponding to 40% of full steam flow between 0% and 20% load and then increasing linearly to a Δ-P corresponding to 110% of full steam flow at full power.	Non-LOCA Steamline break events core response & M/E releases. See Section S-3.3.2.
	Allowable Value	
	≤ A function defined as follows: A delta-P corresponding to 44.2% of full steam flow between 0% and 20% load and then increasing linearly to a Δ-P corresponding to 111.4% of full steam flow at full power.	Note, the scaling of this trip setpoint has been revised to reflect Standard Tech Spec format and also matches the parameters which are actually measured by the instrumentation providing the protection function. The generic nature of the setpoint description is such that it would apply both to the current rated power level and to the rerating power level. The safety analysis modelling in the Non-LOCA steamline break events (core response & mass/energy releases) provide setpoints which are limiting for this function.

TABLE S-3.13-1 (Cont'd)
REACTOR PROTECTION SYSTEM/ESF SETPOINTS

Function	Nominal Setpoint	Limiting Analysis
d. Bases Limiting Safety System Settings page B 2-5 (Reactor Trip Instrumentation OPΔT)	Reference item a.	Non-LOCA analysis (steamline break M/E releases outside containment). Reflects credit taken for OPΔT in accident analyses.

TABLE S-3.13-2

OTHER TECHNICAL SPECIFICATION CHANGES

Tech. Spec. Item/ Page	Description of Change	Basis for Change
1. DNBR & Tavg Limit/ 3.2.5 & (Table 3.2-1) page 3/4 2-13	Full power Tavg from 543.9°F to 581.8°F indicated Tavg.	Westinghouse analyses support temperature span. Instrument error and readability are included within the analysis boundaries
	Full power pressure 2200 psig or 2050 psig indicated pressures.	RTP analyses support two pressures. Instrument error and readability are included within the analysis boundaries. Reference letter AEP-89-216, from J. C. Hoebel to R. B. Bennett.
2. MTC limit/ 3.1.1.4 page 3/4 1-5	Less negative than -4.4×10^{-4} Δ-k/K/°F at Rated Thermal Power.	Non-LOCA analysis supports MTC change.
3. ESF response times Table 3.3-5 pages 3/4 3-27 through 3/4 3-30	Containment pressure high, safety injection (ECCS) $\leq 27.0^{@@}/27.0^{++}$ seconds.	Refer to Notes (1) and (2) for table notation and explanations of new response times. The calculation of the

TABLE S-3.13-2

OTHER TECHNICAL SPECIFICATION CHANGES

Tech. Spec. Item/ Page	Description of Change	Basis for Change
3. ESF response times cont'd.		steamline break M/E releases inside containment takes credit for this ESF function.
	Pressurizer pressure low, safety injection (ECCS) $\leq 27.0^{@@}/27.0^{++}$ seconds.	Refer to Notes (1) and (2) for table notation explanations of new response times. The steamline break core response analysis of a credible break took credit for this ESF function.
	Differential pressure between steam lines high, safety injection (ECCS) $\leq 27.0^{@@}/37.0^{@}$ seconds.	Refer to Notes (1) and (2) for table notations and explanations of new response times. The non-LOCA analysis did not take credit for this ESF function.
	Steam flow in two steam lines high, coincident with Tavg low-low, safety injection (ECCS) $\leq 29.0^{@@}/39.0^{@}$ seconds steam line isolation ≤ 13.0 seconds.	Refer to Notes (1) and (2) for table notations and explanation of new response times. The non-LOCA analysis did not take credit for this ESF function.

TABLE S-3.13-2 (Continued)
OTHER TECHNICAL SPECIFICATION CHANGES

Tech. Spec. Item/ Page	Description of Change	Basis for Change
3. ESF response times (Continued)	Steam flow in two steam lines high, coincident with steam line pressure low, safety injection(ECCS) $\leq$27.0@@/37.0@ seconds, steam line isolation $\leq$11.0 seconds.	Refer to Notes (1) and (2) for table notations and explanations of new response times. The steamline break analyses (core response and M/E releases) took credit for this ESF function.
	Containment pressure hi-hi, steam line isolation $\leq$11.0 seconds.	The steamline break M/E releases inside containment took credit for steamline isolation 11 seconds following containment pressure hi-hi signal.
4. Steam generator stop valves 4.7.1.5.1 page 3/4 7-10	Surveillance requirements, total closure time of 8 seconds.	Steamline break core response and M/E release analysis supports increased closure time requirements.
5. RHR and HHSI Cross-tie closure, charging flow imbalance 3.5.2 page 3/4 5-3	Limitations on operations of RHR or the HHSI cross-ties.	LB LOCA limiting analysis, SB LOCA limiting analysis.

TABLE S-3.13-2 (Continued)

OTHER TECHNICAL SPECIFICATION CHANGES

Tech. Spec. Item/ Page	Description of Change	Basis for Change
6. Definitions 1-0 page 1-1	Rated Thermal Power (RTP).	Reflects rerating.
7. Rod Group Insertion Limits Figure 3.1-1 page 3/4 1-24	Delete Thermal Power value for RTP.	Allows definition value for RTP.
8. Emergency Core Cooling System Section 3.5.2 page 3/4 5-3	Cross tie closure.	Limiting condition for rerating.
9. Emergency Core Cooling System Section 3.5.2 page 3/4 5-5	RHR : 160 psid SI : 1385 psid Charging: 2290 psid	These values represent 10% pump degradation and are consistent with the Westinghouse analyses. These values supersede those presented in WCAP-11902.
10. Emergency Core Cooling System Section 3/4.5.4 Boron Injection Tank pages 3/4 5-9&10, and Bases page B 3/4 5-2	Modified to replace or remove the BIT minimum boron concentration requirement. Changes reduces or eliminate surveillance requirements.	Westinghouse analyses support zero ppm boron concentration.

TABLE S-3.13-2 (Continued)

OTHER TECHNICAL SPECIFICATION CHANGES

Tech. Spec. Item/ Page	Description of Change	Basis for Change
11. Bases Moderator Temperature Coefficient (MTC) page B 3/4 1-2	Revised MTC limits	Relaxation of limits utilizing some safety margin for operational margin.
12. Bases (DNB) page B 3/4 2-6	Revised values to allow operation over a continuous temperature range and with one of two discrete pressures.	Reflects reduced temperature range and with one of two discrete pressures.
13. Bases Emergency Core Cooling System (ECCS) 3/4.5.2 and 3/4.5.3 page B 3/4 5-1	Explanation of RTP limits for 3413 MWT operation.	Cross tie operation limitations.
14. Bases Limiting Safety System Setting (LSSS) page B 2-5	Explanation of accidents utilizing Overpower Delta T trip function.	Credit for this trip was taken in steamline M/E release outside of containment.
15. Reactor Trip System Instrumentation Response Times page 3/4 3-10 (Table 3.3-2)	Added Overpower Delta T Response time of $\leq$ 6.0 seconds.	Credit for this trip was taken in steamline M/E release outside of containment.

Note (1)

Table 3.3-5 notation to be revised as follows:

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

++ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is NOT included.

@ Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

@@ Diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

Note (2)

In support of the relaxed BIT minimum boron concentration requirement, the non-LOCA analyses (specifically steamline break core response and M/E releases) are sensitive to the sequential transfer of charging pump suction valves from the VCT to the RWST. An SI path (availability of borated water from the RWST) is only established once the charging pump suction valves from the VCT close following opening of the charging pump suction valves from RWST. As such, the ESF response times are increased to account for the sequential transfer.

S-4.0 LICENSING CRITERIA REVIEW

S-4.1 10 CFR 50.59 (Unreviewed Safety Question)

Changes made to an existing nuclear power plant must be evaluated pursuant to this regulation for impact on technical specifications, involvement of an unreviewed safety question and impact on procedures and analyses as described in the Final Safety Analysis Report (FSAR).

The evaluations presented in Section 3.0 of WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," and Section S-3.0 of this Supplement to WCAP-11902 demonstrate that no unreviewed safety question is involved. Reanalysis and evaluation of affected FSAR accidents support this conclusion by showing that the probability of occurrence, possibility of new accidents or margin of safety in any Technical Specifications basis has not changed from the original design. Plant operation at the rerated power levels and revised temperatures and pressures are not a test or experiment. However, as a result of the incorporation of rerated power, and reduced temperature and pressure into the safety analyses and evaluations described in this document, Technical Specification changes are necessary.

S-4.2 10 CFR 50.92 (Significant Hazard Determination for Issuance of Amendment)

The criteria in this regulation must be considered when a proposed change to an existing plant also involves an amendment to the operating license. An evaluation of significant hazard considerations accompanies the application for amendment submittal.

S-4.3 10 CFR 50.36 (Technical Specifications)

This regulation defines the type of information to be included in Technical Specifications at the time of application for an operating license. The necessary amendments to Technical Specifications for this program are done in accordance with 10 CFR 50.90.

S-4.4 10 CFR 50.71 (FSAR Update)

Paragraph (e) of this regulation provides guidelines for periodic updates of the FSAR. Donald C. Cook Nuclear Plant Units 1 & 2 share a common FSAR and periodic amendments will continue to be routinely provided as modifications accumulate.

5.0 CONCLUSIONS

The results of the preceding evaluations and analyses performed for this program demonstrate that the safety of the Donald C. Cook Nuclear Power Plant Unit 1 is not compromised by full power operation within the range of parameters described in Section S-2.1 corresponding to a maximum power level of 3425 MWt NSSS. Each of the individual conclusions delineated in this report individually supports this conclusion for the event, component, or system that was reviewed.

The evaluations and analyses described in this Supplement and listed in Section S-1.3 as applicable for Cook Unit 2 demonstrate that the safety of the unit is not compromised by full power operation within the range of parameters described in Section S-2.1 corresponding to a maximum power level of 3600 MWt NSSS. However, additional efforts are necessary which were not performed as an integral part of the Rerating Program to support licensing of the uprating for Cook Nuclear Plant Unit 2. These efforts are the fuel-related analyses, which will be submitted to the NRC by AEPSC as part of the Cycle 8 fuel reload effort for Unit 2, and a 3600 MWt long term containment integrity analysis, which currently supports a maximum NSSS power level of 3425 MWt and will be provided to the NRC at a future date to be determined.

The conclusions from the LOCA analyses (Large Break and Small Break) presented in WCAP-11902, "REDUCED TEMPERATURE AND PRESSURE OPERATION FOR DONALD C. COOK NUCLEAR PLANT UNIT 1 LICENSING REPORT," Volume 1, remain valid for the rerating program for Cook Nuclear Plant Unit 1. As noted above, LOCA analyses have not yet been performed for Unit 2 for the rerated conditions.

The conclusions of the LOCA Hydraulic forcing functions analysis presented in WCAP-11902, Volume 1, are valid for Cook Nuclear Plant Units 1 and 2, assuming a maximum power level of 3600 MWt NSSS.

The conclusions of the non-LOCA transients evaluations and analyses presented in WCAP-11902, Volume 1 are valid for operation of Unit 1 under the rerating conditions corresponding to a maximum power level of 3425 MWt NSSS.

Additional analyses were performed and are presented in this Supplement in the area of Main Steam Line Break mass and energy releases. Mass and energy releases (inside and outside containment) were calculated which bound the rerating conditions for both Cook Units 1 and 2 at 3600 MWt NSSS. A containment analysis was performed for the rerating program, using the revised mass and energy releases; the results of this analysis are bounded by the current FSAR results. This analysis therefore demonstrates the capability of the containment heat removal systems to rapidly reduce containment temperature and pressure in the event of a main steamline break accident, for the rerating conditions. The results of analyses performed by AEPSC, considering the revised mass and energy releases outside containment, are provided elsewhere in this submittal.

The conclusions presented in WCAP-11902, Volume 1, for Short Term Containment Integrity are valid for Cook Units 1 and 2 over the range of rerating conditions corresponding to a maximum power level of 3600 MWt NSSS. The conclusions presented in WCAP-11902, Volume 1, for Long Term Containment Integrity following a LOCA, are valid for Cook Units 1 and 2 over the range of rerating conditions corresponding to a maximum power level of 3425 MWt NSSS.

The conclusions presented in WCAP-11902, Volume 1, for Steam Generator Tube Rupture remain valid for Cook Units 1 and 2 for the rerating conditions corresponding to a maximum power level of 3600 MWt NSSS.

The conclusions presented in WCAP-11902, Volume 1, for post-LOCA hot leg recirculation time, remain valid for Cook Unit 1 for the rerating conditions described in Section S-2.1.

The conclusions presented in WCAP-11902, Volume 1, are valid for Cook Units 1 and 2 for the rerating conditions described in Section S-2.1.

A radiological analysis was performed to assess the impact of the rerating on the resulting doses from Large Break LOCA and the Fuel Handling Accident, in addition to the doses from the Steam Generator Tube Rupture accident (which was mentioned above). The conclusions of the analysis is that there is no

increase in the consequences of either event due to the rerating. These conclusions are valid for Cook Units 1 and 2, up to a maximum power level of 3600 MWt NSSS.

Post-LOCA hydrogen generation rates for the rerating which bound both Cook Units at a maximum power level of 3600 MWt NSSS were provided to AEPSC by Westinghouse. AEPSC analyzed the effects on containment hydrogen concentrations and determined that there was no effect on the overall containment analyses as a result of the rerating.

The conclusions presented in WCAP-11902, Volume 1, for the NSSS fluid and mechanical systems and components, and NSSS/Balance-of-Plant interface systems, remain valid for Cook Nuclear Plant Units 1 and 2 at the rerated conditions corresponding to a maximum NSSS power level of 3600 MWt.

The fuel assembly structural integrity is demonstrated for Cook Unit 1 at the rerated conditions corresponding to a maximum NSSS power level of 3600 MWt NSSS. The fuel assembly structural evaluation for Cook Unit 2 will be provided as part of the Vantage 5 Reload Submittal. The conclusions presented in WCAP-11902, Volume 1, for the fuel rod structural integrity are valid for Cook Unit 1 for the rerated conditions corresponding to a maximum NSSS power level of 3425 MWt.