PROPOSED CHANGES TO THE
COOK NUCLEAR PLANT UNIT NO. 1
TECHNICAL SPECIFICATIONS

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ±3%.#*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.#*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum useable volume of 175,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the useable water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be less than the design limit of 12 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The useable water volume limit reflects the volume of water above the centerline of the discharge pipe. An allowance for water not useable because of tank discharge line location or other physical characteristics is not required.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

With one steam generator stop valve inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the unit hot. The 8 hour completion time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the steam generator stop valves. If the steam generator stop valve cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and the MODES 2 and 3 action statement entered. The completion times are reasonable, based on operating experience, to reach MODE 2 and to close the steam generator stop valves in an orderly manner and without challenging unit systems.

Since the steam generator stop valves are required to be OPERABLE in MODES 2 and 3, the inoperable valves may either be restored to OPERABLE status or closed. When closed, the valves are already in the position required by the assumptions in the safety analysis. The 8 hour completion time is consistent with the MODE 1 action statement requirement. For inoperable steam generator stop valves that cannot be restored to OPERABLE status within the specified completion time, but are closed, the inoperable valves must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day completion time is reasonable, based on engineering judgement, in view of steam generator stop valve status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

If in MODES 2 or 3 the steam generator stop valves cannot be restored to OPERABLE status or are not closed within the associated completion time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

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PROPOSED CHANGES TO THE
COOK NUCLEAR PLANT UNIT NO. 2
TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The **DEFINED TERMS** of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 **THERMAL POWER** shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 **RATED THERMAL POWER** shall be a total reactor core heat transfer rate to the reactor coolant of 3588 MWt.

OPERATIONAL MODE

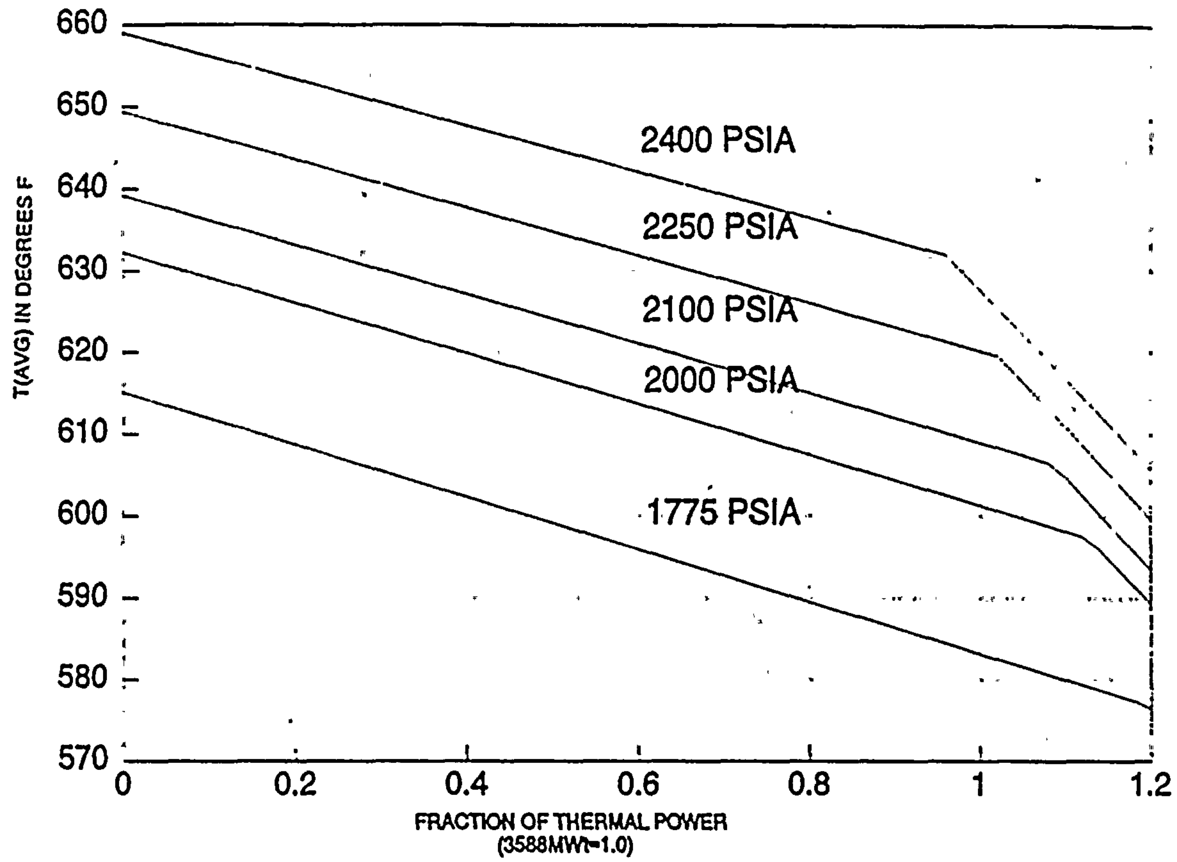
1.4 An **OPERATIONAL MODE** shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 **ACTION** shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be **OPERABLE** or have **OPERABILITY** when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).



Description of Safety Limits

Pressure (psia)	Power (frac)	T_{avg} (°F)	Power (frac)	T_{avg} (°F)	Power (frac)	T_{avg} (°F)	Power (frac)	T_{avg} (°F)
1775	0.00	615.1	1.10	580.0	1.18	577.4	1.2	576.4
2000	0.00	632.2	1.12	597.6	1.14	596.0	1.2	589.4
2100	0.00	639.2	1.08	606.5	1.10	604.8	1.2	593.5
2250	0.00	649.4	1.02	619.5	1.10	610.9	1.2	599.7
2400	0.00	659.0	0.96	631.9	1.1	616.7	1.2	605.7

Flow Rate = 91,600 gpm/loop

Figure 2.1-1
Reactor Core Safety Limits
Four Loops in Operations

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER
		High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level - - High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

* Design flow is 1/4 Reactor Coolant System total flow rate from LCO 3.2.5.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_0 \; [K_1 - K_2 [\frac{1+\tau_1 s}{1+\tau_2 s}] (T-T') + K_3 (P-P') - f_1 (\Delta I)]$

where:	ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER less than or equal to 581.3°F.
	P	=	Pressurizer Pressure, psig
	P'	=	Indicated RCS nominal operating pressure (2235 psig or 2085 psig).
	$\frac{1+\tau_1 s}{1+\tau_2 s}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg}; τ_1 = 22 secs, τ_2 = 4 secs.
	S	=	Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

4 Loops in Operation

$K_1 = 1.17$

$K_2 = 0.0268$

$K_3 = 0.00111$

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -16 percent and +6 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -16 percent, the ΔT trip setpoint shall be automatically reduced by 2.05 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 2.7 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 \; [K_4 - K_5 \, [\frac{\tau_3 S}{1+\tau_3 S}] \, T - K_6 \, (T - T'') - f_2 \, (\Delta I)]$

where:	ΔT_0	=	Indicated ΔT at rated power
	T	=	Average temperature, °F
	T''	=	Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576°F.
	K_4	=	1.08
	K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature
	K_6	=	0.00197 for T greater than T''; K_6=0 for T less than or equal to T''
	$\frac{\tau_3 S}{(1+\tau_3 S)}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in the rate lag controller for T_{avg}; τ_3 = 10 secs.
	S	=	Laplace transform operator
	f_2 (ΔI)	=	0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.75 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.59 percent ΔT span.

DNB AND Tavg OPERATING PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the following operational indicated limits:

1. Reactor Coolant System T_{avg}	Less than or equal to 583.3°F*
2. Pressurizer Pressure	Greater than or equal to 2200 psig (for nominal pressure of 2235 psig)*/** Greater than or equal to 2050 psig (for nominal pressure of 2085 psig)*/**
3. Reactor Coolant System Total Flow Rate	Greater than or equal to 366,400 gpm***

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

*Indicated average of at least three OPERABLE instrument loops.

**Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RTP

***Indicated value

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Manual Initiation	------------------------ See Functional Unit 9 ------------------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure -- High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure -- Low	Greater than or equal to 1815 psig	Greater than or equal to 1805 psig
e. Differential Pressure Between Steam Lines -- High	Less than or equal to 100 psi	Less than or equal to to 112 psi
f. Steam Line Pressure -- Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	------------------------ See Functional Unit 9 ------------------------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure -- High-High	Less than or equal to 2.9 psig	Less than or equal to 3.0 psig
d. Steam Flow in Two Steam Lines -- High Coincident with Tavg -- Low-Low	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.6×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to 4.5×10^6 lbs/hr at full load.	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.75×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to 4.55×10^6 lbs/hr at full load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. Steam Line Pressure -- Low	Greater than or equal to 500 psig steam line pressure	Greater than or equal to 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level -- High-High	Less than or equal to 67% of narrow range instrument span each steam generator	Less than or equal to 68% of narrow range instrument span each steam generator

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ±3%.#*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electrical power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.#*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITIONS FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Determining the seal line resistance at least once per 31 days when the average pressurizer pressure is within 20 psi of its nominal full pressure value. The seal line resistance measured during the surveillance must be greater than or equal to 2.27 E-1 ft/gpm^2. The seal line resistance, R_{SL}, is determined from the following expression:

$$R_{SL} = \frac{2.31(P_{CHP} - P_{SI})}{Q^2}$$

where: P_{CHP} = charging pump header pressure, psig

P_{SI} = 2112 psig (low pressure operation)

= 2262 psig (high pressure operation)

2.31 = conversion factor (12 in/ft)^2/(62.3 lb/ft^3)

Q = the total seal injection flow, gpm

The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

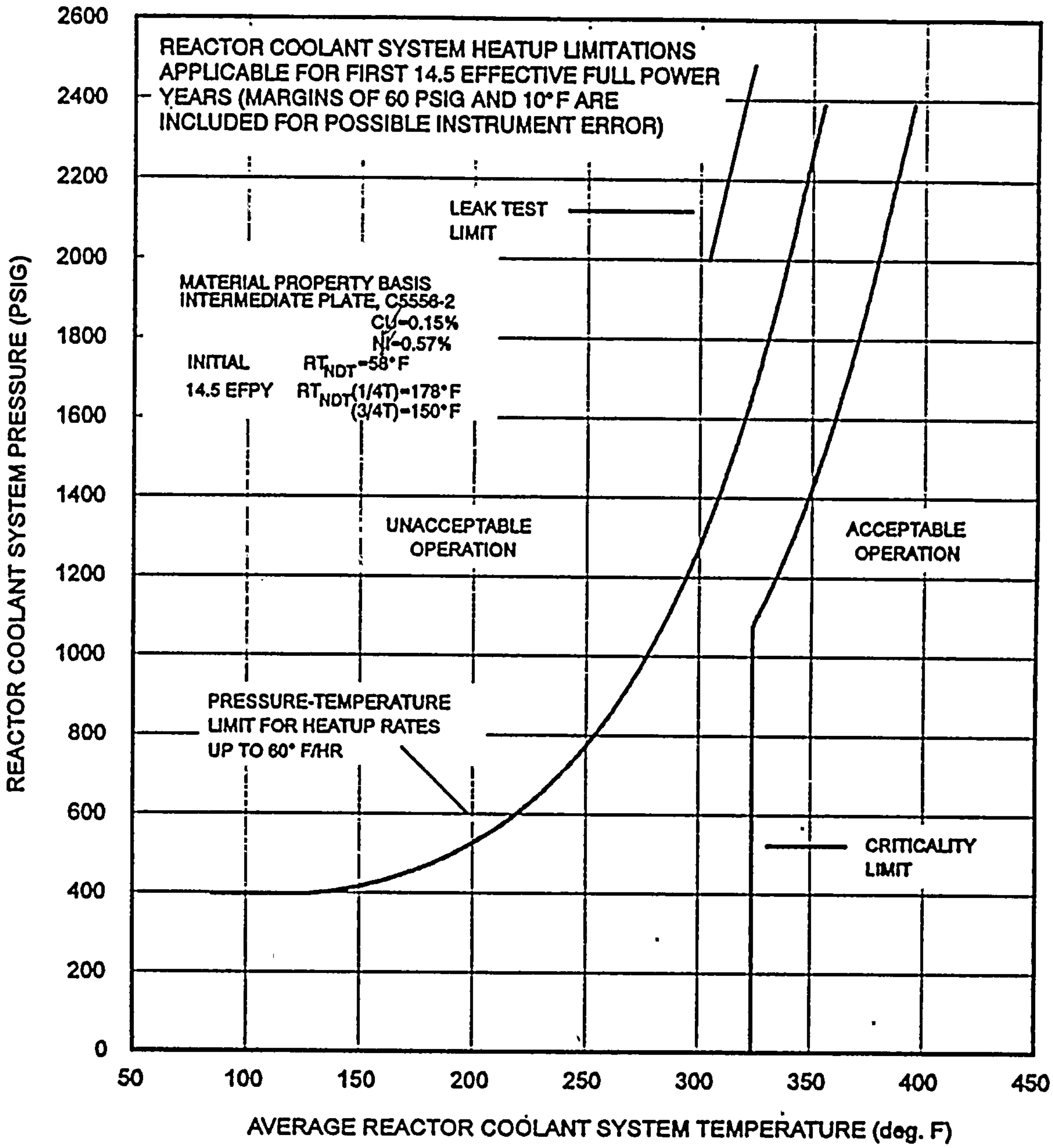


Figure 3.4-2
Reactor Coolant System Pressure - Temperature Limits for 60°F/hr Rate, Criticality Limit and Hydrostatic Test Limit

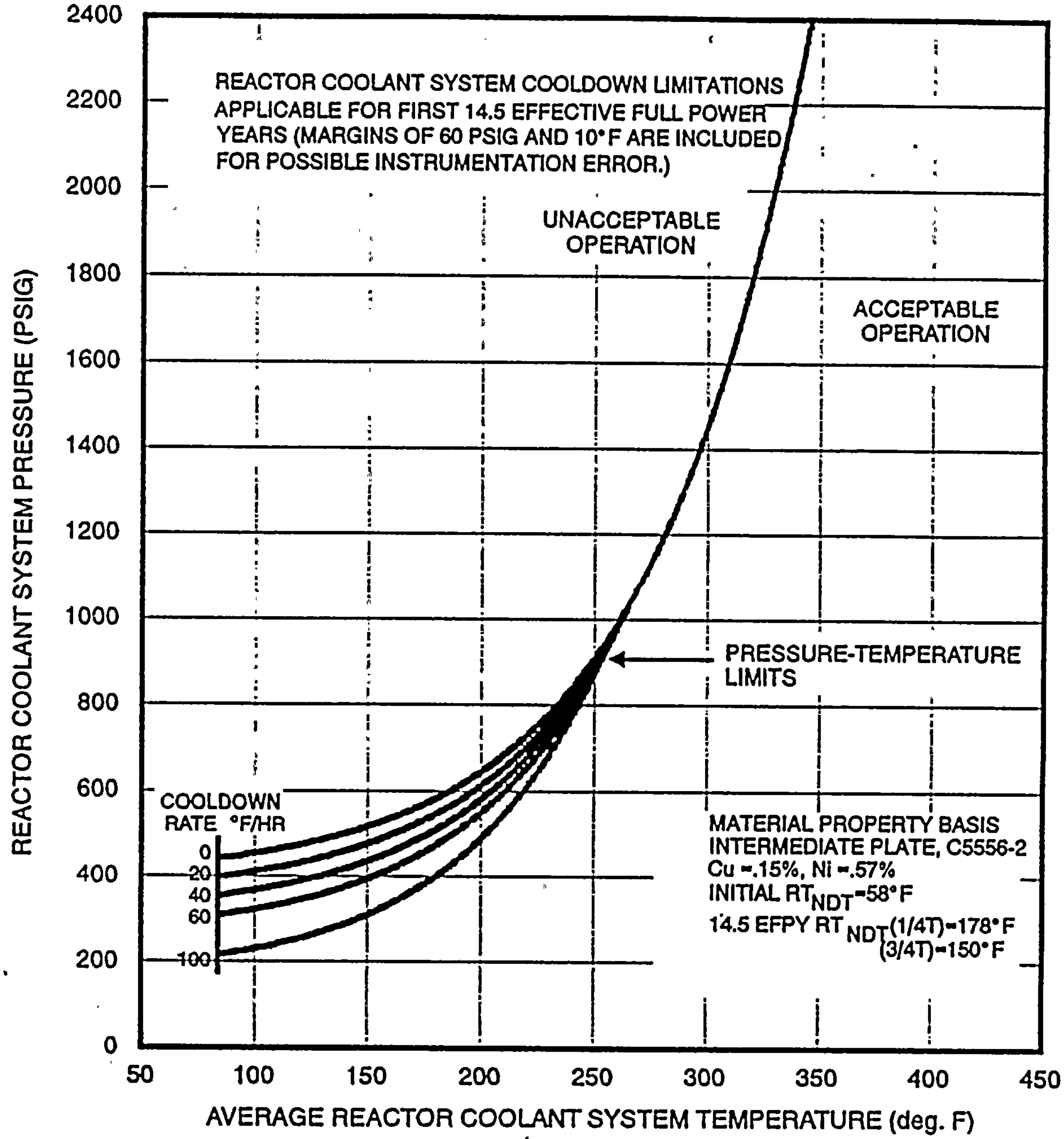


Figure 3.4-3
Reactor Coolant System, Pressure - Temperature, Limits for Various Cooldown Rates

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump,

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	58.1
2	41.2
3	24.5

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum useable volume of 175,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the useable water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-2 correlation and W-3 correlation for conditions outside the range of WRB-2. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-2 correlation for Vantage-5 fuel, and the W-3 correlation for conditions which fall outside the range of applicability of the WRB-2). The correlation DNBR limits are established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for WRB-2 and 1.3 for the W-3).

In meeting the DNB design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are statistically combined with the DNBR correlation statistics such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to a calculated design limit DNBR. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR correlation statistics, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Cook Nuclear Plant Unit 2, the design DNBR values are 1.23 and 1.22 for Vantage-5 fuel typical and thimble cells, respectively. In addition, margin has been maintained by performing safety analyses to a safety analysis limit DNBR. The margin between the design and safety analysis limit DNBR is used to offset known DNBR penalties (i.e., transition core penalties, rod bow, etc.) and provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

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 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.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 2.1% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

Margin between the safety analysis DNBRs and the design limit DNBRs is maintained. (Safety analyses DNBRs: 1.69 and 1.61 for the Vantage 5 typical and thimble cells, respectively. Design limit DNBRs: 1.23 and 1.22 for the Vantage 5 typical and thimble cells, respectively.) A fraction of this margin is utilized to accommodate applicable transition core penalties and the appropriate fuel rod bow DNBR penalty for the Vantage 5 fuel (equal to 1.3% per WCAP-8691, Rev.1). The remainder of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The T_{avg} less than or equal to 583.3°F and pressurizer pressure greater than or equal to 2200 psig (for nominal pressurizer operating pressure of 2235 psig) or greater than or equal to 2050 psig (for nominal pressurizer operating pressure of 2085 psig) are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR thoughout each analyzed transient with allowance for measurement uncertainty. Pressurizer pressure is limited to either of two nominal operating pressures of 2235 psig or 2085 psig, with the corresponding indicated limits set forth in the specifications. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR value for the current fuel type throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of 14.5 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}: The results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 14.5 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 14.5 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515 report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated February 1993.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stem generator less than or equal to 50°F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.5.1 ACCUMULATORS (Continued)

allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be less than the design limit of 12 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The useable water volume limit reflects the volume of water above the centerline of the discharge pipe. An allowance for water not useable because of tank discharge line location or other physical characteristics is not required.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

With one steam generator stop valve inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the unit hot. The 8 hour completion time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the steam generator stop valves. If the steam generator stop valve cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and the MODES 2 and 3 action statement entered. The completion times are reasonable, based on operating experience, to reach MODE 2 and to close the steam generator stop valves in an orderly manner and without challenging unit systems.

ATTACHMENT 3 TO AEP:NRC:1223

EXISTING TECHNICAL SPECIFICATION
PAGES MARKED TO REFLECT PROPOSED CHANGES

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COOK NUCLEAR PLANT UNIT NO. 1
TECHNICAL SPECIFICATIONS

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 1%*#
3%

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

See insert 4-4

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.* #
3%

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional surveillance requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

see insert 4-4

Insert 4-4 for # footnote on tech spec page 3/4 4-4 and 3/4 4-5

\# The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum ~~contained~~ Useable volume of 175,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the Usable water ~~level~~ Volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be less than the design limit of 12 psig, ~~11.89 psig~~, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of $60^{\circ}F$ will limit the peak pressure to ~~11.39 psig which is~~ less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The useable water volume limit reflects the volume of water above the centerline of the discharge pipe. An allowance for water not useable because of tank discharge line location or other physical characteristics is not required.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

With one steam generator stop valve inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the unit hot. The 8 hour completion time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the steam generator stop valves. If the steam generator stop valve cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and the MODES 2 and 3 action statement entered. The completion times are reasonable, based on operating experience, to reach MODE 2 and to close the steam generator stop valves in an orderly manner and without challenging unit systems.

Since the steam generator stop valves are required to be OPERABLE in MODES 2 and 3, the inoperable valves may either be restored to OPERABLE status or closed. When closed, the valves are already in the position required by the assumptions in the safety analysis. The 8 hour completion time is consistent

3/4.7.1.5 STEAM GENERATOR STOP VALVES (continued)

Text moved to pg B 3/4 7-3

with the MODE 1 action statement requirement. For inoperable steam generator stop valves that cannot be restored to OPERABLE status within the specified completion time, but are closed, the inoperable valves must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day completion time is reasonable, based on engineering judgement, in view of steam generator stop valve status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

If in MODES 2 or 3 the steam generator stop valves cannot be restored to OPERABLE status or are not closed within the associated completion time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

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COOK NUCLEAR PLANT UNIT NO. 2
TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~[illegible]~~ MWt.

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OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

DESIGN FLOW = 91,600 GPM/LOOP

DESCRIPTION OF SAFETY LIMITS

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.[illegible]	0.98	583.8	1.02	580.9	1.2	558.1
2000	0.00	631.8	0.86	605.8	0.96	597.5	1.2	568.5
2100	0.00	639.1	0.82	614.0	0.96	601.6	1.2	573.1
2250	0.00	649.2	0.72	628.6	0.98	605.2	1.2	580.4
2400	0.00	659.9	0.62	642.0	1.1	599.0	1.2	588.1

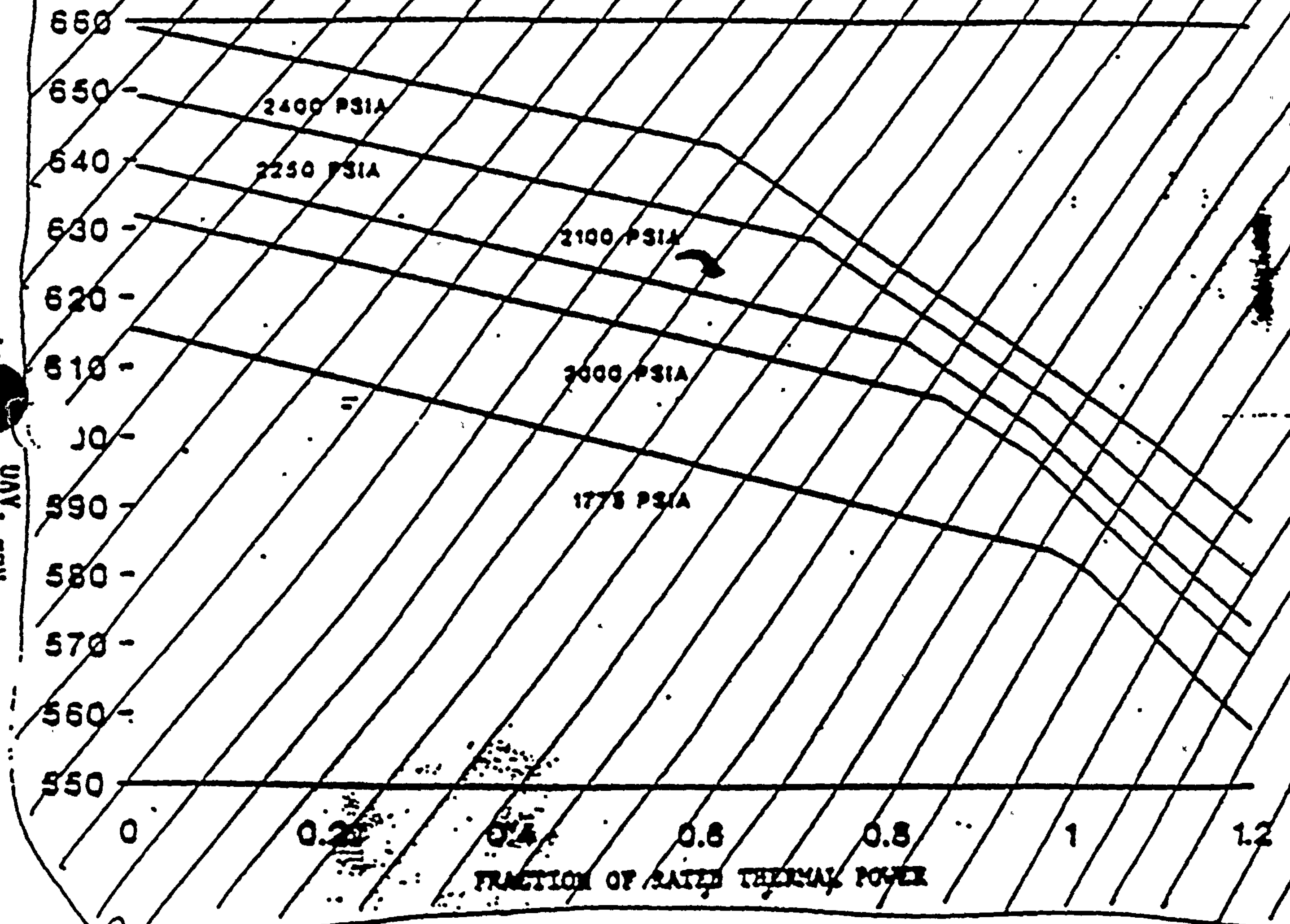

Figure 2.1-1 Reactor Core Safety Limits
Four Loops in Operation

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Flow Rate = 91,600 gpm/loop

Description of Safety Limits

(Full VANTAGE 5 Core)

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.1	1.10	580.0	1.18	577.4	1.2	576.4
2000	0.00	632.2	1.12	597.6	1.14	596.0	1.2	589.4
2100	0.00	639.2	1.08	606.5	1.10	604.8	1.2	593.5
2250	0.00	649.4	1.02	619.5	1.10	610.9	1.2	599.7
2400	0.00	659.0	0.96	631.9	1.1	616.7	1.2	605.7

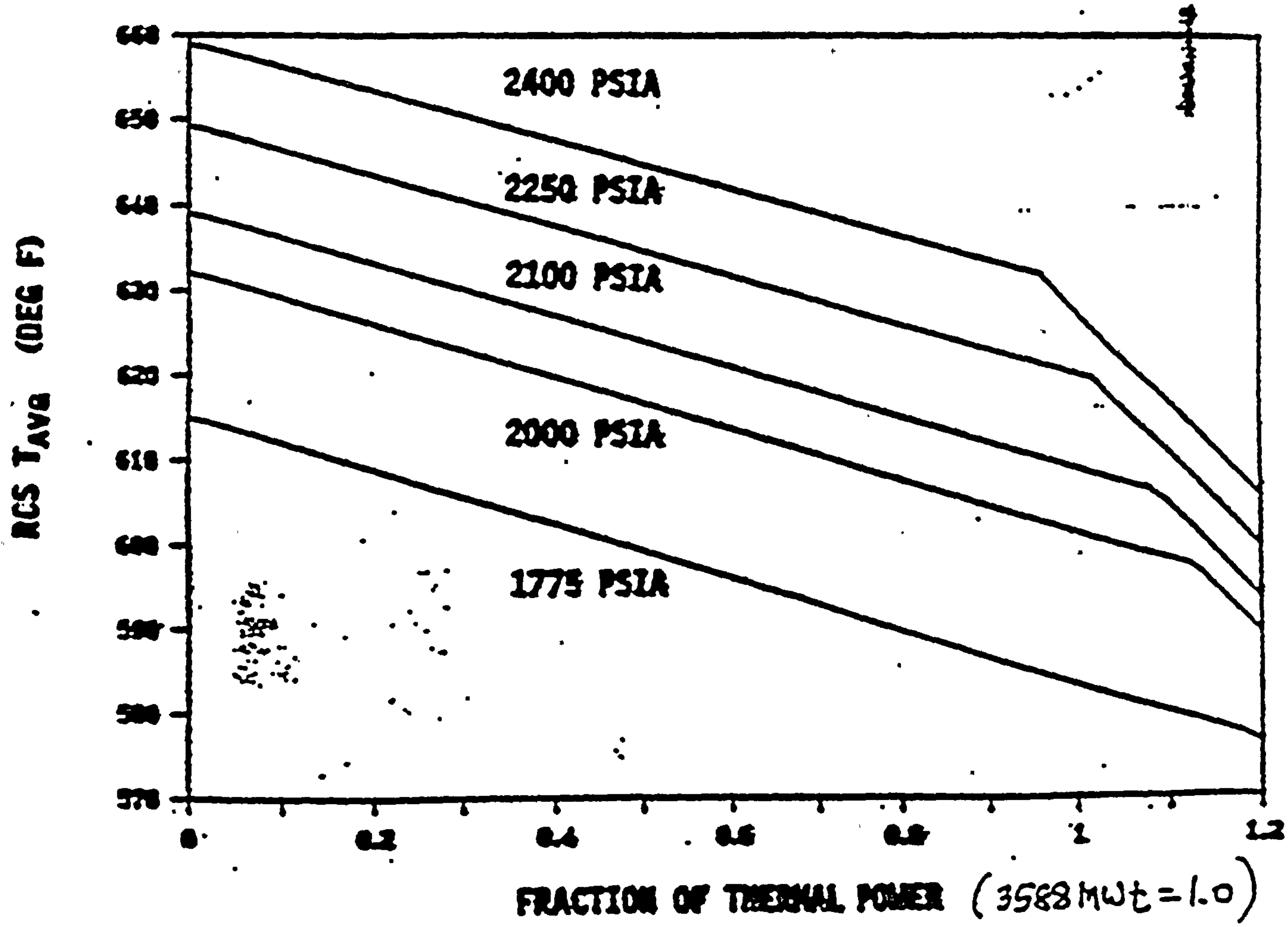


Figure 2.1-1 Reactor Core Safety Limits Four Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3 x 10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

* Design flow is ~~81,400 gpm per loop~~.
1/4 Reactor Coolant System total flow rate from LCO 3.2.5

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1:
Overtemperature $\Delta T \leq \Delta T_o \, [K_1 - K_2[(1 + \tau_1 S)/(1 + \tau_2 S)](T - T') + K_3(P - P') - f_1(\Delta I)]$

Where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER less than or equal to ~~576.0 °F~~ 581.3°F

P = Pressurizer Pressure, psig

P' = ~~2235~~ psig {Indicated RCS nominal operating pressure} (2235 psig or 2085 psig)

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg}: τ_1 = ~~28~~ 22 secs, τ_2 = 4 secs.

S = Laplace transform operator

NOTATION (Continued)

4 Loops in Operation

K1 = ~~1.09~~ 1.17

K2 = ~~0.01331~~ 0.0268

K3 = ~~0.00058~~ 0.00111

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-33~~ -16 percent and +6 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -16 ~~-33~~ percent, the ΔT trip setpoint shall be automatically reduced by ~~3.5~~ 2.05 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.0~~ 2.7 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \; [K_4 - K_5[\tau_3 S/(1+\tau_3 S)]T - K_6[T-T"] - f_2(\Delta I)]$

Where:

ΔT_o	= Indicated ΔT at rated power
T	= Average temperature, ^oF
$T"$	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 ^oF
K_4	= 1.08
K_5	= 0.02/^oF for increasing average temperature and 0 for decreasing average temperature
K_6	= 0.00197 for T greater than T"; K_6 = 0 for T less than or equal to T"
$\tau_3 S/(1+\tau_3 S)$	= The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	= Time constant utilized in the rate lag controller for T_{avg}; τ_3 = 10 secs.
S	= Laplace transform operator
$f_2(\Delta I)$	= 0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.3 percent ΔT span.
3.75

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent ΔT span.
2.59.

DNB AND Tavg OPERATING PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the following operational indicated limits:

a. DNB

Insert 3/4 2-15

1.	Reactor Coolant System T_{avg}	Less than or equal to 578.7°F*
2.	Pressurizer Pressure	Greater than or equal to 2200 psig*/**
3.	Reactor Coolant System Total Flow Rate	Greater than or equal to 366,400 gpm***

b. T_{avg}

1. Reactor Coolant System T_{avg} Greater than or equal to 543.9°F*

Delete

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

* Indicated average of at least three OPERABLE instrument loops.
** Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RTP
*** Indicated value

INSERT 3/4 2-15

1.	Reactor Coolant System T_{avg}	Less than or equal to 583.3°F
2.	Pressurizer Pressure	Greater than or equal to 2200 psig (for nominal pressure of 2235 psig)*/**
		Greater than or equal to 2050 psig(for nominal pressure of 2085 psig)*/**

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Manual Initiation	------------ See Functional Unit 9 -----------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure--Low	Greater than or equal to ~~1900~~ 1815 psig	Greater than or equal to ~~1890~~ 1805 psig
e. Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f. Steam Line Pressure--Low	Greater than or equal to 600 psig steam line pressure 500	Greater than or equal to 585 psig steam line pressure 480

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	------------- See Functional Unit 9 ---------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3.0 psig
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.6×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to 4.5×10^6 lbs/hr at full load.	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.75×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to to 4.55×10^6 lbs/hr at full load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. Steam Line Pressure--Low	Greater than or equal to 600 psig steam line pressure. 500	Greater than or equal to 585 psig steam line pressure. 480
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow range instrument span each steam generator	Less than or equal to 68% of narrow range instrument span each steam generator

REACTOR COOLANT SYSTEM

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± ~~1%~~.* # 3%

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electrical power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

See Insert 4-4

Insert 4-4 for # footnote on tech spec page 3/4 4-4 and 3/4 4-5

The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± *# 3%

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

See Insert 4-4

LIMITING CONDITIONS FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Determining the seal line resistance at least once per 31 days when the average pressurizer pressure is within 20 psi of its nominal full pressure value. The seal line resistance measured during the surveillance must be greater than or equal to 2.27 E-1 ft/gpm². The seal line resistance, R_{SL}, is determined from the following expression:

$$R_{SL} = \frac{2.31\,(P_{CHP} - P_{SI})}{Q^2}$$

where: P_{CHP} = charging pump header pressure, psig

≥2112 psig (low pressure operation)

P_{SI} = 2262 psig (high pressure operation)

2.31 = conversion factor $(12\ in/ft)^2/(62.3\ lb/ft^3)$

Q = the total seal injection flow, gpm .

The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

D.C. COOK-UNIT 2

3/4 4-23

AMENDMENT NO. [illegible]

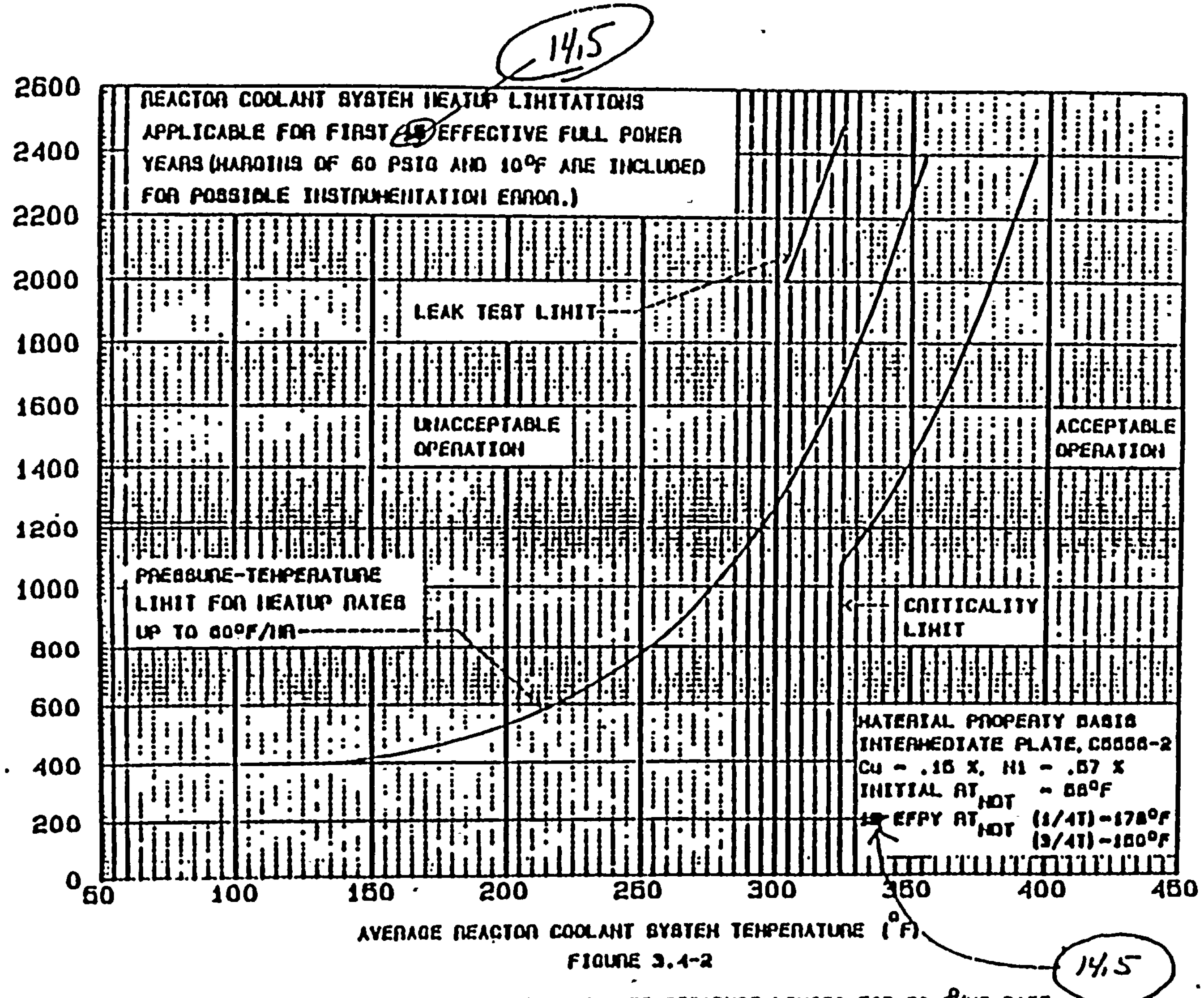
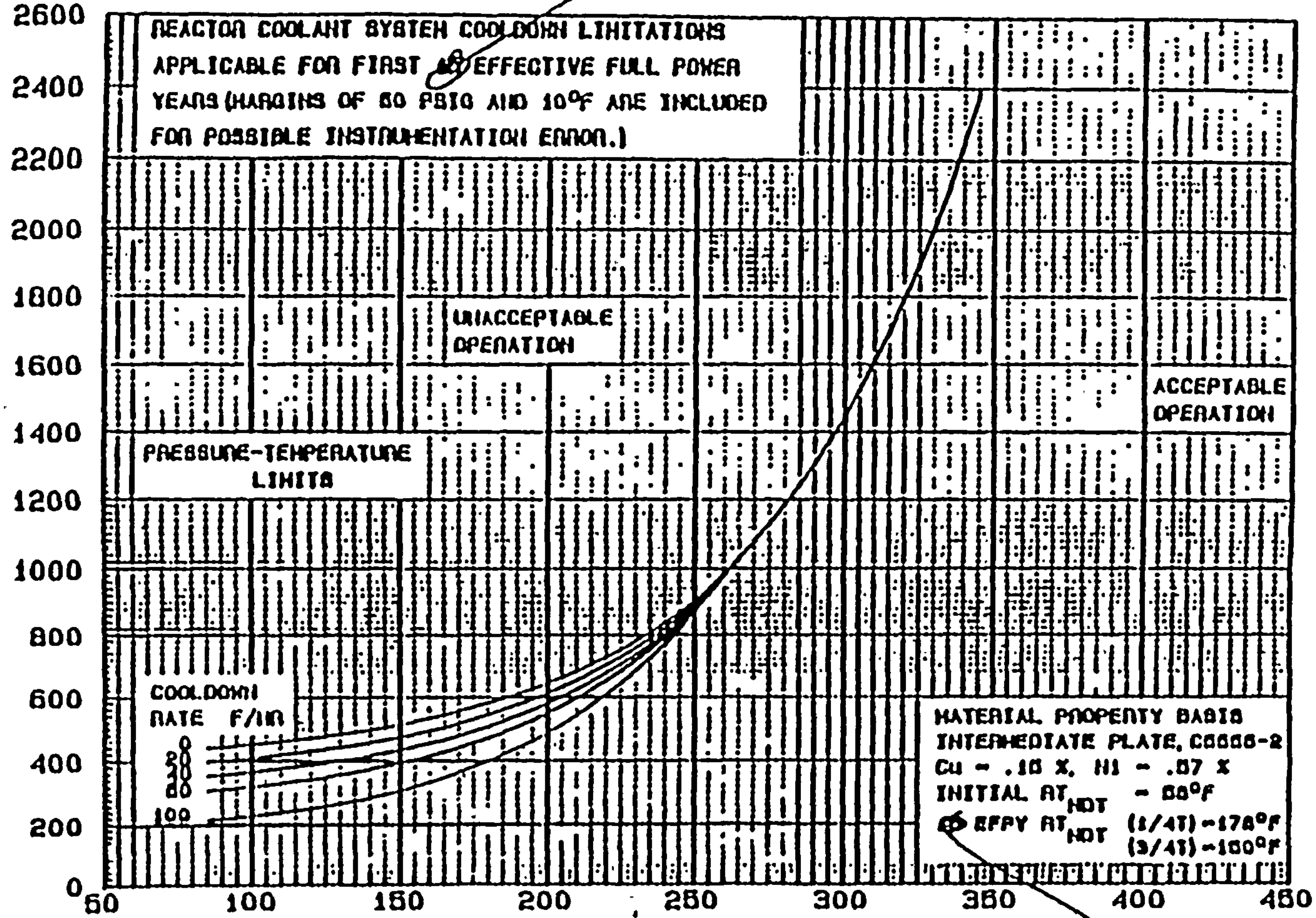


FIGURE 3.4-2

REACTOR COOLANT SYSTEM, PRESSURE - TEMPERATURE, LIMITS FOR 60 °F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT



AVERAGE REACTOR COOLANT SYSTEM TEMPERATURE (°F)

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE – TEMPERATURE LIMITS FOR VARIOUS COOLDOWN RATES

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump,

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

Delete

~~f. All safety injection cross-tie valves open.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

~~b. With a safety injection cross-tie valve closed, restore the cross-tie valve to the open position or reduce the core power level to less than or equal to 3250 MW within one hour. Specification 3.0.4 does not apply.~~

~~c.~~ b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

Delete

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~61.6~~ 58.1
2	~~43.9~~ 41.2
3	~~26.2~~ 24.5

PLANT SYSTEMS

CONDENSATE STORAGE TANK

useable

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of 175,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

useable

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

SA

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-2 correlation and W-3 correlation for conditions outside the range of WRB-2. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-2 correlation for Vantage-5 fuel, and the W-3 correlation for ~~ANF fuel and~~ conditions which fall outside the range of applicability of the WRB-2). The correlation DNBR limits are established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for WRB-2 and 1.3 for the W-3).

In meeting the DNB design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are statistically combined with the DNBR correlation statistics such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to a calculated design limit DNBR. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR correlation statistics, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Cook Nuclear Plant Unit 2, the design DNBR values are 1.23 and 1.22 for Vantage-5 fuel typical and thimble cells, respectively~~, and 1.19 and 1.36 for typical and thimble cells for the ANF fuel~~. In addition, margin has been maintained in both fuel types by performing safety analyses to a safety analysis limit DNBR. The margin between the design and safety analysis limit DNBR is used to offset known DNBR penalties (i.e., transition core penalties, rod bow, etc.) and provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

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 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.

Delete

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: (Continued)

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.1. Measurement errors of 2.1% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

Margin between the safety analysis DNBRs and the design limit DNBRs is maintained. (Safety analyses DNBRs: 1.69 and 1.61 for the Vantage 5 typical and thimble cells, respectively~~, and 1.43 and 1.40 for the ANF fuel typical and thimble cells~~. Design limit DNBRs: 1.23 and 1.22 for the Vantage 5 typical and thimble cells, respectively~~, and 1.39 and 1.36 for the ANF fuel typical and thimble cells~~.) A fraction of this margin is utilized to accomodate applicable transition core penalties and the appropriate fuel rod bow DNBR penalty for the Vantage 5 fuel (equal to 1.3% per WCAP-8691, Rev. 1). The remainder of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

R[illegible] w[illegible] Insert A

The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The T_{avg} less than or equal to 578.7°F and pressurizer pressure greater than or equal to 2200 psig are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. The T_{avg} greater than or equal to 543.9°F is conservative to a safety analysis performed to demonstrate that the plant may operate on a linear control program where the analytical limit of T_{avg} at 100% RATED THERMAL POWER may range from 541.4°F to 580.1°F. The limit of 543.9°F contains a margin of 1.1°F. The core may be operated with indicated vessel average temperature at any value between the upper and lower limits. Pressurizer pressure is limited to a single nominal setpoint, with the lower limit of the indicated value setpoint set forth in the specifications. The T/S value was selected for consistency with Unit 1 and contains a margin of 6 psi. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

INSERT A

greater than or equal to

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The T_{avg} less than or equal to 583.3°F and pressurizer pressure greater than or equal to 2200 psig (for nominal pressurizer operating pressure of 2235 psig) or 2050 psig (for nominal pressurizer operating pressure of 2085 psig) are consistent with the USFAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. Pressurizer pressure is limited to either of two nominal operating pressures of 2235 psig or 2085 psig, with the corresponding indicated limits set forth in the specifications. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR value for the current fuel type throughout each analyzed transient.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown, the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY.
14.5

The reactor vessel materials have been tested to determine their initial RT_{NDT}: The results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 15 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments. 14.5

The ~~15~~ EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515 report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated February 1993.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stem generator less than or equal to 50°F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS (Continued)

allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

If a safety injection cross-tie valve is closed, safety injection would be limited to two lines assuming the loss of one safety injection subsystem through a single failure consideration. The resulting lowered flow requires a decrease in THERMAL POWER to limit the peak clad temperature within acceptable limits in the event of a postulated small break LOCA.

Delete

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure ~~expected to be obtained~~ resulting from a LOCA event is ~~9.4 psig. The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 9.7 psig which is less than the design pressure and is consistent with the accident analyses.~~ calculated to be less than the design limit of 12 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to ~~9.4 psig which is~~ less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

PLANT SYSTEMS

BASES

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The ~~contained~~ usable water volume limit ~~includes an~~ reflects the volume of water above the centerline of the discharge pipe. An allowance for water not usable because of tank discharge line location or other physical characteristics is not required.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

With one steam generator stop valve inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the unit hot. The 8 hour completion time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the steam generator stop valves. If the steam generator stop valve cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and the MODES 2 and 3 action statement entered. The completion times are reasonable, based on operating experience, to reach MODE 2 and to close the steam generator stop valves in an orderly manner and without challenging unit systems.

ATTACHMENT 4 TO AEP:NRC:1223

SUMMARY DESCRIPTION OF PROPOSED
UNIT 2 POWER UPRATE
TECHNICAL SPECIFICATIONS

Key for Summary Table

Page	Technical Specification Page
Section	Technical Specification
Group	Related Groups Discussed in Attachment 1, Description of Proposed Changes and 10 CFR 50.92 Significant Hazards Consideration Analysis

Uprate	Group 1, Changes Directly Related to Increased rated thermal power
HHSI	Group 2, Change to Remove Power Restriction for High Head Safety Injection Cross Ties Closed Operation
Margin	Group 3, Changes Proposed to Increase Unit 2 Operating Margin
Transition	Group 4, Changes Related to Transition Core or Transition to Temperature Window/Dual Pressure Technical Specifications
Both*	Group 5, Changes Proposed for both units.
Admin	Group 6, Administrative Change

Description	A Brief Description of Each Proposed Change
Remarks	Brief Comments with a Cross Reference to the Analyses

* Note that all changes are <u>only</u> for unit 2 unless they are included in the "both" group. The changes in this group are proposed for both unit 1 and unit 2 of Cook Nuclear Plant.

Page	Section	Group	Description	Remarks
1-1	1.3	Uprate	Increase rated thermal power to 3588 MWt.	The support for this proposed change consists of analyses that have been performed over a period of years. Including the new analyses, which are described in Attachment 6, WCAP 14489, and the evaluations described in Attachment 7, Balance of Plant Evaluations and Miscellaneous Safety Evaluations, all the necessary analyses and evaluations have been completed to support an uprate of Unit 2 to a core power of 3588 MWt. The new analyses and summaries of earlier analyses and evaluations performed by Westinghouse Electric Corporation for the nuclear steam supply system (NSSS) are described in WCAP 14489. The impact of recent model changes on the new analyses is discussed in Attachment 1 under Group 1 changes as well as in Attachment 6. Attachment 7 describes balance of plant evaluations and miscellaneous safety evaluations. Since the analyses which support the uprated power have been performed over a period of years, Attachment 5 is provided to describe the history of earlier analyses and to identify the submittal of earlier work and the associated SER's. The review status of the analyses supporting the uprated core power is discussed in Attachment 1 under Group 1 changes and in Attachment 5.
2-2	Figure 2.1-1	Margin	Revise Reactor Core Safety Limits	The Safety Limit Figure currently in the Unit 2 Technical Specifications was designed for a mixed core of Westinghouse and Advanced Nuclear Fuel. The Unit 2 core now consists totally of Westinghouse Vantage 5 fuel. The new thermal design is discussed in Section 3.3.2.1 of Attachment 6 WCAP 14489. The proposed Safety Limit Figure is consistent with a rated thermal power of 3588 MWt and an all Vantage 5 core.

Page	Section	Group	Description	Remarks
2-5	Table 2.2-1 Footnote	Admin	Redefine design flow in footnote of Table 2.2-1 to be 1/4 MMF.	Minimum Measured Flow (MMF) is Reactor Coolant System Total Flow Rate of T.S. 3.2.5. MMF is used directly in the DNB analyses as discussed in Section 3.3.2.1 of Attachment 6, WCAP 14489. MMF is 1.035 times thermal design flow (TDF). Therefore, the MMF employed in the DNB analysis is 1.035 times TDF. TDF is specified in Section 3.3.3.1 of WCAP 14489. TDF is generally used in no- DNB analyses. See, for example, WCAP 14489 tables 3.1-7 and 3.1-13. A lower (loop) TDF is indicated in Table 3.5-1 because the containment analysis bounds both units and Unit 1 is analyzed for a 5% flow asymmetry. Design flow in current technical specification Table 2.2-1 is loop MMF or total MMF/4.
2-7	Table 2.2-1	Margin	The upper limit on T′ increased to 581.3°F to reflect analyses.	New OTDT and OPDT setpoints have been calculated to support operation at a rated thermal power of 3588 MWt with all Westinghouse Vantage 5 cores which are currently being used in Unit 2. K1 was increased from its current value of 1.09 to 1.17 thereby significantly increasing load rejection margin. The value 1.17 was selected to allocate some margin to instrumentation, increased allowance for core burnup effects such as changes in hot leg streaming, and an increase in the positive ΔI break point for the f(ΔI) penalty. Increased load rejection margin was also obtained by reducing tau 1 from 28 seconds to 22 seconds. The analysis value of K4 was also evaluated to obtain increased margin for burnup effects. The OTDT and OPDT trips are discussed in Section 3.3.2.1 of WCAP 14489. Details, including T′, T′′, and time constants of the analyzed setpoint, are in Table 3.3-4 of WCAP 14489.
2-7	Table 2.2-1	Margin	Both analyzed, nominal RCS pressures, 2235 psig and 2085 psig, are indicated.	Due to the analysis performed for transition cores of both Westinghouse and Advanced Nuclear Fuel, low pressure operation was not permitted for operation with mixed cores of Westinghouse and Advanced Nuclear Fuel. The basis for this limitation is discussed in Section "Group 4" of Attachment 1. Unit 2 is currently operated with cores of all Westinghouse Vantage 5 fuel. Therefore, low pressure operation is acceptable.
2-7	Table 2.2-1	Margin	Tau 1 reduced from 28 secs to 22 secs.	See remark on increase of T′ upper limit on page 2-7.

Page	Section	Group	Description	Remarks
2-8	Table 2.2-1	Margin	Increase K1 from 1.09 to 1.17.	See remark on increase of T′ upper limit on page 2-7.
2-8	Table 2.2-1	Margin	Increase K2 from 0.01331 to 0.0268	See remark on increase of T′ upper limit on page 2-7.
2-8	Table 2.2-1	Margin	Increase K3 from 0.00058 to 0.00111	See remark on increase of T′ upper limit on page 2-7.
2-8	Table 2.2-1	Margin	Change f(ΔI) penalty.	See remark on increase of T′ upper limit on page 2-7.
2-9	Table 2.2-1	Margin	Maintain the upper limit on T′′ at 576°F to reflect analyses.	This item is not a change. However, due to the fact that 576°F is not the maximum analyzed temperature of the temperature window, it is appropriate to identify the fact that the upper limit on T′′ is being deliberately maintained at its current value. Cook Nuclear Plant Unit 2 is operated a temperature significantly lower than the maximum analyzed temperature. Therefore, the OPDT setpoint was analyzed with a low upper limit on T′′ to convert unused margin to operating margin. The OPDT trip is discussed in Section 3.3.2.1 of WCAP 14489. Details, including T′′, of the analyzed setpoint are in Table 3.3-4 of WCAP 14489.
2-9	Table 2.2-1	Margin	Change the allowable values in notes 3 and 4.	The values indicated in the markups of Attachment 3 and in the proposed technical specifications of Attachment 2 were calculated by our organization using the Westinghouse "stepit" methodology described in WCAP-12741. This is the same methodology used for the calculation of all existing Reactor Trip and Engineered Safety Feature Actuation Setpoints. This methodology is consistant with the requirements of ISA Standard S67.04.

Page	Section	Group	Description	Remarks
3/4 2-15	Section 3.2.5	Transition	Increase DNB temperature limit from 578.7°F to 583.3°F.	Due to the analysis performed for transition cores consisting of both Westinghouse fuel and Advanced Nuclear Fuel, the maximum nominal Tavg was limited to 576°F for operation with mixed cores of Westinghouse and Advanced Nuclear Fuel. Unit 2 is currently operated with cores of all Westinghouse Vantage 5 fuel. Therefore, the full analyzed temperature window analyzed for an all Vantage 5 core is acceptable. The DNB temperature limit is obtained by adding the controller allowance to the high nominal Tavg used in the analysis and then subtracting the readability allowance. The high nominal Tavg for a full Vantage 5 core is 581.3°F and the controller allowance is 4.1°F. These values are found in Table 3.3-1 and Section 3.3.3.1 of WCAP 14489, respectively. The readability allowance, calculated by AEPSC, is 2.1°F. The resulting DNB temperature limit is 583.3°F. The high nominal Tavg for a full Vantage 5 core is identified in the Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant Unit 2 (RTSR), reference 11 of Attachment 5. RTSR was submitted via reference 13 of Attachment 13 of Attachment 5. Operation of Unit 2 with Westinghouse fuel was approved in reference 17 of Attachment 5.

Page	Section	Group	Description	Remarks
3/4 2-15	Section 3.2.5	Transition	DNB pressure limits for both analyzed nominal RCS pressures, 2200 psig and 2050 psig, are indicated.	Due to the analysis performed for transition cores consisting of both Westinghouse fuel and Advanced Nuclear Fuel, low pressure operation was not permitted for operation with mixed cores of Westinghouse and Advanced Nuclear Fuel. The basis for this limitation is discussed in Section "Group 4" of Attachment 1. Unit 2 is currently operated with cores of all Westinghouse Vantage 5 fuel. Therefore, low pressure operation is acceptable. The DNB pressure limit is obtained by subtracting the total pressure allowance used in the analysis from the nominal operating pressure used in the analysis and then adding the readability allowance. The nominal pressures and the total allowance are found in Section 3.3.1 and 3.3.3.1 of WCAP 14489, respectively. The readability allowance, calculated by AEPSC, is 18.9 psi. The pressure limit currently in the T/Ss for high pressure operation is conservatively higher than the calculated value of 2191 psig. The proposed limit of 2050 psig for low pressure operation is an addition. It is conservatively higher than the calculated value of 2041 psig. The proposed value for the low pressure limit is the same as the unit 1 limit.
3/4 2-15	Section 3.2.5	Transition	Remove reference to low temperature limit in order to return the specification to a purely DNB specification.	The proposed change converts the DNB specification back to a purely DNB specification. This proposal is discussed further in Attachment 1.
3/4 3-23	Table 3.3-4	Transition	Reduce safety injection actuation setpoint on low pressurizer pressure to 1815 psig.	Additional margin to trip for safety injection on low pressurizer pressure was needed as discussed in the description of changes for Group 4 of Attachment 1. The required evaluations to lower the safety analysis value for the setpoint have been performed as discussed in Sections 3.3.4.5 and 3.3.4.6 of WCAP 14489. The nominal Technical Specification setpoint proposed is the same as the corresponding setpoint for Unit 1. The instrument allowances calculated by our organization support this nominal setpoint.

Page	Section	Group	Description	Remarks
3/4 3-23	Table 3.3-4	Transition	Change allowable value for safety injection actuation setpoint on low pressurizer pressure to 1805 psig.	The value 1805 psig, indicated in the markups of Attachment 3 and in the proposed technical specifications of Attachment 2, is conservative to the value that was calculated by our organization. It is consistent with value for unit 1 and is proposed in order to make the Technical Specifications of the two units more similar.
3/4 3-23	Table 3.3-4	Transition	Reduce safety injection actuation setpoint on low steam line pressure to 500 psig.	At the time of the transition from Advanced Nuclear Fuel to Westinghouse fuel, the reanalysis of mass and energy release (M&E) outside containment for rerating and reduced temperature/reduced pressure operation was not complete. The evaluation of the then applicable analysis assumed an NSSS power of 3425 MWt and a nominal setpoint for low steam line pressure no less than 520 psig. The AEPSC analysis of the impact of the steam line mass and energy release (SM&E) outside containment on the operability of equipment in the main steam enclosures, at 3588 MWt core power, was described and submitted in reference 24 of Attachment 5. Reference 24 was our proposal to operate both units with 0 ppm boron concentration in the boron injection tank (BIT). The SM&E was analyzed consistently with the proposed safety injection setpoint on low steam line pressure. The mass and energy release portion of this analysis is discussed in Section 3.3.4.7 of WCAP 14489. The core response steamline and feedwater line breaks submitted with the RTSR support the proposed setpoint. The references for the RTSR, its submittal, and approval are 11, 13, and 17 of Attachment 5. Evaluations of these analyses are discussed in Sections 3.3.4.6 and 3.3.4.7 of WCAP 14489. The revised SM&E release analysis to containment also supports the proposed setpoint. This analysis is discussed in WCAP 14285, reference 29 of Attachment 5. WCAP 14285 was submitted via reference 30 of Attachment 5. It is not yet approved. This analysis is summarized in Section 3.5.4 of WCAP 14489.

Page	Section	Group	Description	Remarks
3/4 3-23	Table 3.3-4	Transition	Change allowable value for safety injection actuation setpoint on low steam line pressure to 480 psig.	The value 480 psig, indicated in the markups of Attachment 3 and in the proposed technical specifications of Attachment 2, is conservative to the value that was calculated by our organization. It is consistent with value for unit 1 and is proposed in order to make the Technical Specifications of the two units more similar.
3/4 3-25	Table 3.3-4	Transition	Reduce steam line isolation actuation setpoint on low steam line pressure to 500 psig.	See remark on reduction of safety injection actuation on low steam line pressure on page 3/4 3-23.
3/4 3-25	Table 3.3-4	Transition	Change allowable value for steam line isolation actuation setpoint on low steam line pressure to 480 psig.	See remark on the change of allowable value for safety injection actuation setpoint on low steam line pressure on page 3/4 3-23.
3/4 4-4	Section 3.4.2	Margin	Increase Pressurizer Valve Setpoint Tolerance to 3%.	The Non-LOCA accidents were reanalyzed or reevaluated based on a pressurizer valve setpoint tolerance of 3%. This is noted in section 1.1 and 3.3.2.2 of WCAP 14489. The analyses affected are discussed in Sections 3.3.4.3, 3.3.4.4, 3.3.4.6, 3.3.5.1, and 3.3.5.2 of WCAP 14489.
3/4 4-4	Section 3.4.2	Both	Add footnote requiring an as left tolerance of 1%.	This requirement is consistent with a similar requirement which was approved for the main steam safety valves. It is being submitted for both units because it was inadvertently omitted in our submittal AEP:NRC:1207, dated May 26, 1995, which included the analytical justification for an increase in pressurizer safety valve setpoint tolerance for Unit 1.
3/4 4-5	Section 3.4.3	Margin	Increase Pressurizer Valve Setpoint Tolerance to 3%.	See remark on setpoint tolerance magnitude on page 3/4 4-4.
3/4 4-5	Section 3.4.3	Both	Add footnote requiring an as left tolerance of 1%.	See remark on as left setpoint tolerance on page 3/4 4-4.

Page	Section	Group	Description	Remarks
3/4 4-16	Section 4.4.6.2.1	Transition	Pressure criteria for both analyzed nominal RCS pressures are indicated.	Due to the analysis performed for transition cores of both Westinghouse and Advanced Nuclear Fuel, low pressure operation was not permitted for operation with mixed cores of Westinghouse and Advanced Nuclear Fuel. The basis for this limitation is discussed in Section "Group 4" of Attachment 1. Unit 2 is currently operated with cores of all Westinghouse Vantage 5 fuel. Therefore, low pressure operation is acceptable.
3/4 4-25	Figure 3.4-2	Uprate	Reduce the applicability of the heatup curve from 15 to 14.5 EFPY's.	The applicability of the current heatup and cooldown curves is discussed in Section 3.11.2.1 of Attachment 6, WCAP 14489.
3/4 4-26	Figure 3.4-3	Uprate	Reduce the applicability of the cooldown curve from 15 to 14.5 EFPY's.	See the remark on the applicability of the heatup and cooldown curves on page 3/4 4-25
3/4 5-3	Section 3.5.2	HHSI	Remove power reduction currently required for operation with HHSI cross tie valves closed.	When the SBLOCA analysis performed to support the main steam safety valve tolerance relaxation to 3% was carried out, it was found that a power reduction was required to obtain satisfactory results with the high head safety injection (HHSI) cross ties closed. Since then, improvements have been made to the Westinghouse NOTRUMP SBLOCA model. As indicated in the cover letter to this submittal, the SBLOCA analysis performed for this submittal was performed using the new model. The results of this analysis show that an acceptable PCT results with the HHSI cross tie valves closed at a core power of 3588 MWt. The SBLOCA analyses are described in Section 3.1.2.4 of WCAP 14489.
3/4 7-2	Table 3.7-1	Uprate	Lower the maximum Allowable Power Range Setpoint.	These setpoints are calculated in accordance with the prescription in Westinghouse Nuclear Safety Advisory Letter 94-001. In this prescription, nominal NSSS power appears in the denominator of the equation for the setpoint. Therefore, the proposed setpoints were lowered accordingly.

Page	Section	Group	Description	Remarks
UNIT 1 3/4 7-7	UNIT 1 Sections 3.7.1.3 and 4.7.1.3.1	Both	Change contained volume to useable volume.	The analysis of 3.10.2.5 of WCAP-14489 shows that 174,500 gallons of water are required to maintain the RCS at hot standby for 9 hours. The Technical Specifications currently require a minimum contained volume of 175,000 gallons of water. However, due to the fact that the zero of the level instrumentation is located at the centerline of the discharge pipe, above the level for required NPSH, all the indicated volume is useable. Therefore, the proposed Technical Specifications are revised to address useable volume.
UNIT 2 3/4 7-7	UNIT 2 Sections 3.7.1.3 and 4.7.1.3.1			
B 2-1	Bases Section 2.1.1	Transit ion	Remove references to Advanced Nuclear Fuel.	The Unit 2 core now consists totally of Westinghouse Vantage 5 fuel.
B 2-5	Bases Section 2.2.1	Margin	Remove detail from the discussion of the OPDT protection trip.	The discussion of the proper normalization of T'' is being removed. This information is documented in Section 3.3.2.1 of WCAP 14489 and will be controlled administratively.
B3/4 2-4a	Bases Section 3/4.2.2 and 3/4.2.3	Transit ion	Remove references to Advanced Nuclear Fuel.	The Unit 2 core now consists totally of Westinghouse Vantage 5 fuel.
B 3/4 2-5	Bases Section 3/4.2.5	Transit ion	Remove reference to low temperature limit in order to return the specification to a purely DNB specification.	See remark on low temperature limit on page 3/4 2-15.
B 3/4 4-6	Bases Section 3/4.4.9	Uprate	Reduce the applicability of the heatup and cooldown curves from 15 to 14.5 EFPY's.	See remark on the applicability of the heatup curve on page 3/4 4-25.
B 3/4 4-10	Bases Section 3/4.4.9	Uprate	Reduce the applicability of the heatup and cooldown curves from 15 to 14.5 EFPY's.	See remark on the applicability of the heatup curve on page 3/4 4-25.

Page	Section	Group	Description	Remarks
B 3/4 5-1a	Bases Sections 3/4.5.2 and 3/4.5.3	HHSI	Remove power reduction currently required for operation with HHSI cross tie valves closed.	See remark on power reduction required by SBLOCA on page 3/4 5-3.
UNIT 1 B 3/4 6-2	UNIT 1 Bases Sections 3/4.6.1.4 3/4.6.1.5	Both	Change peak containment pressure to reflect analysis result.	Discussion of maximum calculated containment pressure is given in Sections 1.2 and 3.5.2 of WCAP 14489. It is noted that the analysis bounds both Unit 1 and 2 in Section 3.5.2.1 and the peak containment pressure is documented in Section 3.5.3.6.
UNIT 2 B 3/4 6-2	UNIT 2 Bases Sections 3/4.6.1.4 3/4.6.1.5			
UNIT 1 B 3/4 7-3	UNIT 1 Bases Section 3/4.7.1.3	Admin	Change contained volume to useable volume.	See remark on changing contained volume to useable volume page 3/4 7-7.
UNIT 1 B 3/4 7-3	UNIT 2 Bases Section 3/4.7.1.3			

ATTACHMENT 5 TO AEP:NRC:1223

DISCUSSION OF PREVIOUS RELATED SUBMISSIONS

Introduction

The analyses that support the proposed uprating of Donald C. Cook Nuclear Plant unit 2 have been performed over a period of years in several contexts. The analysis of the nuclear steam supply system (NSSS) for an NSSS power of 3600 MWt was performed in conjunction with analyses to operate unit 1 at reduced temperature and pressure (the "Rerating Program"). Most of the core response analyses were performed at an uprated core thermal power of 3588 MWt as a part of the transition from Advanced Nuclear Fuel to Westinghouse Vantage 5 fuel. The recently submitted analyses, AEP:NRC:1223, to support an increase in the permitted level of steam generator tube plugging (SGTP) for unit 1 includes a steam mass and energy release (SM&E) analysis to the containment which bounds both units at 3600 MWt. For this submittal, previous NSSS analyses and core response analyses have been reviewed, new analyses have been performed where necessary, and the balance of plant (BOP) evaluated, as described within this submittal, to support the proposal to increase the core rated thermal power to 3588 MWt.

Attachment 6 to this submittal is WCAP 14489. It describes the analyses, evaluations, and reviews performed by Westinghouse Electric Corporation and summarizes earlier work performed by Westinghouse Electric Corporation to support an increased core rated thermal power for unit 2. WCAP 14489 also describes analyses and evaluations performed simultaneously to support certain increases in operating margin such as increased setpoint tolerance for the pressurizer safety valves. Attachment 7 discusses balance of plant evaluations that have been performed by AEPSC to assess the impact of increased core power.

Section 2.0 of WCAP 14489 discusses the previous work performed by Westinghouse Electric Corporation to support the uprated core power for unit 2. The evaluations described in WCAP 14489 are based on these earlier analyses. The earlier analyses are described in Rerating Program WCAP's 11902 and 11902 Supplement 1, references 3 and 10, and in the Vantage 5 Reload Transition Safety Report for Donald C. Cook Nuclear Plant Unit 2, Revision 1, March 1990 (RTSR), reference 11. The SGTP SM&E analysis is described in WCAP 14285, reference 29.

WCAP 11902 and its supplement are referred to as the "Rerating Program" in WCAP 14489. The reload transition safety report is referred to as "RTSR" in WCAP 14489. The increase in the permitted level of steam generator tube plugging program is referred to as "SGTP Program" in WCAP 14489. The rerating Program, RTSR, SGTP Program, WCAP 14489 (the unit 2 Uprating Program), and the BOP evaluations provide the support for this submittal.

Purpose of Attachment 5

The purposes of this attachment are to:

1. indicate those aspects of earlier analyses which have been submitted for NRC review and approved,
2. indicate those aspects of earlier analyses which have been submitted for NRC review but are not yet approved (This category is comprised of the SGTP Program.),

3. briefly describe the earlier analyses, and
4. provide references for previous submittals and NRC SER's for the convenience of the reviewer.

The discussion of this attachment describes submittals for both units because much of the supporting analysis for unit 2 was performed to bound both units. Submittals primarily for unit 1 are sometimes supported by analyses bounding both units and/or include Technical Specifications modifications for unit 2.

The following list summarizes the status of analysis features of earlier analyses:

<u>Principal Features of the Earlier Analyses Which Have Been Reviewed and Approved by NRC</u>

1. Reduced temperature operation for unit 2.
2. 10% degradation for the RHR and HHSI pumps for both units.
3. Increased MSIV response time for both units.
4. BIT 0 ppm boric acid concentration for both units.
5. Reduced temperature and pressure operation for unit 1.
6. Reduced minimum measured flow for unit 1.

<u>Principal Features of the Earlier Analyses Which Have Been Submitted for Review But Have Not Been Approved</u>

(These features are proposed in the unit 1 increased steam generator tube plugging limit submittal, reference 30.)

1. 10% degradation for the centrifugal charging pumps for both units.
2. Minimum RWST temperature of 70°F for both units.
3. Shutdown Margin requirement of 1.3% for both units.
4. Proposals to increase operating margin for unit 1.
5. Analysis to support operation of the spent fuel pool with one unit operating at 3588 MWt

<u>Principal Features of This unit 2 Uprate Submittal</u>

1. Unit 2 rerate to 3588 MWt.
2. Increase the tolerance of pressurizer safety valve setpoint to 3% for unit 2.
3. Increase OTΔT/OPΔT operating margin.
4. Remove mixed core penalties.
5. 15% degradation for safety injection and RHR pumps.

<u>Purpose of the Earlier Analyses (Rerating Program)</u>

The earlier analyses were performed to accomplish a number of goals. The first of these was to permit operation of unit 1 at reduced primary temperature and pressure. The benefit of operating in a reduced primary temperature and pressure mode was to slow the degradation of the unit 1 steam generators. In addition, since essentially all of the analytic basis of the Cook Nuclear Plant units had to be reviewed or revised, analyses were performed to position unit 1 for subsequent uprating to 3413 MWt core power and unit 2 to 3588 MWt core power. The margin formerly intended to be allocated to a potential unit 1 uprate was subsequently allocated to an increased steam generator tube

plugging limit in the unit 1 increased steam generator tube plugging limit submittal, reference 30. The earlier analyses also supported increased operating margins in selected areas. Among these were increased allowable ECCS pump degradation, reduction of required shutdown margin (SDM), a reduction in the minimum temperature of the refueling water storage tanks (RWST), reduction to zero of the boron concentration in the boron injection tanks (BIT), and slower response times for certain components and systems which applied to unit 2.

Description and Review History of Prior Submittals

This section describes the prior analyses for the Cook units in essentially chronological order with an emphasis on unit 2.

The first of the earlier analyses is described in reference 1, WCAP-11908, "Containment Integrity Analysis for Cook Nuclear Plant units 1 and 2." It was submitted for NRC review by reference 2. Reference 1 presented a long term containment analysis which bounded both units at a core power of 3413 MWt, operation at a reduced temperature and pressure, and operation of the ECCS with residual heat removal (RHR) crossties closed.

Since the analysis described in reference 1 was performed, two new long term containment integrity analyses have been performed for the Cook Nuclear Plant units. One was performed in conjunction with the proposal to increase the unit 1 steam generator tube plugging limit. It was performed at an NSSS power of 3425 MWt and is described in reference 29. Therefore, a new analysis was performed at an NSSS power of 3600 MWt. It is described in WCAP-14489 which is Attachment 6 to this submittal and reference 31. Neither of these new analyses has been reviewed at this time by the NRC.

The next group of analyses is described in reference 3, WCAP-11902, "Reduced Temperature and Pressure Operation for Cook Nuclear Plant Unit 1 Licensing Report." Reference 3 presented the remainder of the analyses and evaluations necessary to support operation of unit 1 at reduced temperature and pressure. The NSSS systems and components analysis was performed for an NSSS power level of 3600 MWt. Reference 3 was submitted for NRC review by reference 4.

The letters of references 5, 6, 7, and 8 provided supplementary information to the staff related to the request for approval (references 2 and 4) to operate unit 1 at reduced temperature and pressure. The request to operate unit 1 in this manner was approved by reference 9.

Reference 10, WCAP 11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Cook Nuclear Plant Units 1 & 2 Licensing Report", describes the balance of the analyses which were performed by Westinghouse Electric Corporation to support the operation of unit 1 at uprated power and reduced temperature and pressure. This report describes analyses and evaluations which were performed to bound both units at an uprated NSSS power of 3600 MWt. In particular, NSSS systems and components were evaluated for an uprated NSSS power of 3600 MWt for both units. The report also describes an analysis of the steam mass and energy release (SM&E) to containment, the associated containment analysis, and the SM&E

release outside containment. These two analyses assumed a shutdown margin of 1.3%, an increased time response for main steam isolation, 0 ppm boron concentration in the boron injection tank (BIT), and were performed to bound both units at the unit 2 uprated core power of 3588 MWt.

Since the SM&E to containment analysis described in reference 10 was performed, the SM&E to containment has been reanalyzed. The new analysis also bounds both units at an NSSS power of 3600 MWt. It was performed in conjunction with the proposal to increase the unit 1 steam generator tube plugging limit. It is described in reference 29 and was submitted with reference 30.

Reference 11, "Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant Unit 2 (RTSR)", together with reference 1, "Containment Integrity Analysis for Cook Nuclear Plant units 1 and 2", reference 3, WCAP 11902, Reduced Temperature and Pressure Operation, and reference 10, WCAP 11902, Supplement 1, Rerated Power and Revised Temperature and Pressure Operation, support reduced temperature and pressure operation for unit 2 at an uprated core power of 3588 MWt. However, reference 1 and the RHR and HHSI crosstie closed LOCA cases of reference 11 only support a unit 2 core power of 3413 MWt. The analyses reported in references 1, 10, and 11 support operation with Westinghouse fuel, 10% degradation of the CCP's, HHSI pumps, and RHR pumps, an increase of 3 seconds in MSIV response time for unit 2, 0 ppm boric acid concentration in the BIT for unit 2, a minimum RWST temperature of 70°F for unit 2, and a SDM of 1.3% for unit 2.

The letter of reference 13 submitted reference 11, RTSR, and the portions of reference 10, WCAP 11902, Supplement 1, which addressed the SM&E to the containment.

The letters of references 14, 15, and 16 provided supplementary information to the staff related to reference 13. Operation of unit 2 with Westinghouse fuel at reduced temperature, with 10% degradation of the RHR and HHSI pumps, was approved by reference 17. Some changes to both the unit 1 and unit 2 Technical Specifications which returned certain activities to administrative control were also made.

The letters of references 18 and 19 for unit 1 and unit 2 respectively proposed technical specifications that implemented an increase of 3 seconds in the MSIV response times. These proposals were supported by reference 3, WCAP-11902, for both units, reference 10, WCAP-11902, Supplement 1, for both units, reference 11, RTSR, for unit 2, and evaluations performed by us. The letters in references 18 and 19 submitted the portions of reference 10, WCAP 11902, Supplement 1, which addressed the SM&E to the containment. The proposals to increase the MSIV response times by 3 seconds were approved by references 20 and 21.

The letter of reference 22 proposed to reduce the primary system minimum measured flow (MMF) for unit 1. This proposal was approved by reference 23.

The letter of reference 24 proposed to reduce the boron concentration in the BIT's of both units to 0 ppm. This proposal was supported by reference 3, WCAP-11902, reference 10, WCAP-11902, Supplement 1, reference 11, RTSR, and analyses performed by us. The AEPSC analysis of the impact of the steam line mass and energy release (SM&E) outside containment on the operability

of equipment in the main steam enclosures was described in reference 24. Reference 10, WCAP 11902, Supplement 1, was submitted in its entirety in support of this proposal. The proposal was approved by reference 25.

The letters of references 26 and 27 proposed to relax the tolerance of the main steam safety valve (MSSV) setpoints for both Cook Nuclear Plant units. The proposal was based on new analyses and on evaluations performed by Westinghouse Electric Corporation. The evaluations were based on the analyses described in reference 1, WCAP-11908, "Containment Integrity Analysis for Cook Nuclear Plant units 1 and 2", reference 3, WCAP-11902, reference 10, WCAP-11902, Supplement 1, and reference 11, RTSR. The descriptions of the new analyses and evaluations were included as attachments to these letters, references 26 and 27.

The unit 2 SBLOCA analysis described in the MSSV submittal was performed assuming that the high head safety injection (HHSI) crossties were closed. It was performed at a core power of 3250 MWt. As a result provisions were included in the Technical Specifications which required a power reduction when the HHSI crossties are closed. In this submittal, a proposal to remove this provision is made based on a new SBLOCA analysis using new models. The new analysis is described in Attachment 6 of this submittal which is reference 31.

The MSSV setpoint relaxation proposal was approved by reference 28.

A recent submittal which impacts the proposal to uprate unit 2 is the proposal to increase the limit of unit 1 steam generator tube plugging. This submittal is reference 30. It has not yet been approved by the NRC. Reference 30 includes a revised steam mass and energy release to containment analysis which bounds both units at an NSSS power of 3600 MWt and proposals to increase operational margin for unit 2. The increases in operating margin include 10% head degradation for the centrifugal charging pumps, a reduction in the minimum refueling water storage tank temperature to 70°F, and a reduction in the required shutdown margin to 1.3%. Reference 30 also proposes to increase the pressurizer safety valve tolerance from 1% to 3% for unit 1 only. Two of the proposed changes in this submittal, AEP:NRC: 1223, is the addition of footnotes which requires that the as left magnitude of the pressurizer safety valves be 1%. This proposal was inadvertently omitted from Reference 30. Attachment 6 to reference 30 is reference 29, WCAP 14285, Revision 1, "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report".

In addition, this submittal requires the approval of previous submittals (references 32 and 33) in order to be implemented. They have not yet been approved by the NRC. These submittals contain a "Refueling Operations Decay Time Technical Specification Amendment Request" permitting core offload 100 hours after subcriticality. The analyses for this proposal support the operation of one unit, Donald C. Cook Nuclear Plant unit 2, at a core power of 3588 MWt from a spent fuel pool thermal hydraulic point of view.

The final submittal for unit 2 is this submittal. It proposes to uprate the core rated thermal power to 3588 MWt. It also proposes increases in unit 2 operating margin by increasing as

found pressurizer safety valve setpoint tolerance to 3%, increasing the OTΔT/OPΔT operating margin, and increasing the analyzed safety injection and residual heat removal pump head degradation to 15%. Attachment 6 to this submittal is reference 31, WCAP 14489, Revision 1, "Donald C. Cook Nuclear Plant Unit 2, 3600 MWt Uprating Program Licensing Report".

References

1. WCAP-11908, Containment Integrity Analysis for Cook Nuclear Plant Units 1 and 2, M. E. Wills, July 1988.

2. Letter AEP:NRC:1024D, Containment Long Term Pressure Analysis to Support RHR Crosstie Closure, from M. P. Alexich to T. E. Murley, August 22, 1988.

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

4. Letter AEP:NRC:1067, Reduced Temperature and Pressure Program Analyses and Technical Specification Changes, from M. P. Alexich to T. E. Murley, October 14, 1988.

5. Letter AEP:NRC:1067A, Supplemental Technical Specification Changes for Reduced Temperature and Pressure Program, from M. P. Alexich to T. E. Murley, December 30, 1988.

6. Letter AEP:NRC:1067B, Additional Information on Reduced Temperature and Pressure Submittal: Boron Dilution Accident, from M. P. Alexich to T. E. Murley, February 6, 1989.

7. Letter AEP:NRC:1067C, Unit 1 RTP Program: Additional Information on Containment Structural Analysis, from M. P. Alexich to T. E. Murley, March 14, 1989.

8. Letter AEP:NRC:1067D, Modification of Reduced Temperature and Pressure Program Technical Specification Changes, from M. P. Alexich to T. E. Murley, June 5,1989.

9. Amendment No. 126 to Facility Operating License No. DPR-58.

10. WCAP 11902, Supplement 1, Rerated Power and Revised Temperature and Pressure Operation for Cook Nuclear Plant Units 1 & 2 Licensing Report, September 1989.

11. Vantage 5 Reload Transition Safety Report for Cook Nuclear Plant Unit 2, B. W. Gergos, Editor, January 1990.

12. No reference 12.

13. Letter AEP:NRC:1071E, Unit 2 Cycle 8 Reload Licensing, Proposed Technical Specifications for Unit 2 Cycle 8, and Related Unit 1 Proposals, from M. P. Alexich to T. E. Murley, February 6, 1990.

14. Letter AEP:NRC:1071H, Modification to Our Previous Submittal AEP:NRC:1071E; Revised Figures for the Loss of Load Event, from M. P. Alexich to T. E. Murley, April 6,

1990.

15. Letter AEP:NRC:1071I, Information to Supplement Our Previous Submittals AEP:NRC:1071E and 1071H, from M. P. Alexich to T. E. Murley, May 29, 1990.

16. Letter AEP:NRC:1071K, Offsite Dose Calculation for the Reactor Coolant Pump Locked Rotor Event for Unit 2 Cycle 8, from M. P. Alexich to T. E. Murley, July 23, 1990.

17. Amendment No. 148 to Facility Operating License No. DPR-58 and Amendment No. 134 to Facility Operating License No. DPR-74.

18. Letter AEP:NRC:1120, Expedited Technical Specification Change Request Steam Generator Stop Valves, from M. P. Alexich to T. E. Murley, January 31, 1990.

19. Letter AEP:NRC:1123, Technical Specification Change Request, Steam Generator Stop Valves, from M. P. Alexich to T. E. Murley, May 14, 1990.

20. Amendment No. 147 to Facility Operating License No. DPR-58.

21. Amendment No. 135 to Facility Operating License No. DPR-74.

22. Letter AEP:NRC:1130, Technical Specification Change for Unit 1 Cycle 11, from M. P. Alexich to T. E. Murley, July 23,1990.

23. Amendment No. 152 to Facility Operating License No. DPR-58.

24 Letter AEP:NRC:1140, Technical Specification Change Request, BIT Boron Concentration Reduction, from M. P. Alexich to T. E. Murley, March 26, 1991.

25. Amendment No. 158 to Facility Operating License No. DPR-58 and Amendment No. 142 to Facility Operating License No. DPR-74.

26 Letter AEP:NRC:1169, Technical Specifications Change to Increase the Allowable Tolerance for Main Steam Safety Valves, from E. E. Fitzpatrick to T. E. Murley, November 11, 1992.

27. Letter AEP:NRC:1169A, Update for Technical Specification Change to Increase the Allowable Tolerances for Main Steam Safety Valves, from E. E. Fitzpatrick to T. E. Murley, December 17, 1993.

28. Amendment No. 182 to Facility Operating License No. DPR-58 and Amendment No. 167 to Facility Operating License No. DPR-74.

29. WCAP 14285, Revision 1, Donald C. Cook Nuclear Plant Unit 1, Steam Generator Tube Plugging Program Licensing Report, May 1995.

30. Letter AEP:NRC:1207, Technical Specification Changes Supported by Analyses to Increase Unit 1 Steam Generator Tube Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Related Analyses, from E. E. Fitzpatrick to

USNRC Document Control Desk, May 26, 1995.

31. WCAP 14489, Revision 1, Donald C. Cook Nuclear Plant Unit 2, 3600 MWt Uprating Program Licensing Report, May 31, 1996.

32. Letter AEP:NRC:1202, "Refueling Operations Decay Time Technical Specification Amendment Request", from E. E. Fitzpatrick to W. T. Russell, November 16, 1994.

33. Letter AEP:NRC:1202A, "Refueling Operations Decay Time Technical Updated Analysis and Response to Request for Additional Information", from E. E. Fitzpatrick to Document Control Desk, February 1, 1996

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of [redacted] Mwt.

3588

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

DESIGN FLOW - 91,600 GPM/LOOP

DESCRIPTION OF SAFETY LIMITS

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.4	0.98	583.8	1.02	580.9	1.2	558.[illegible]
2000	0.00	631.8	0.86	605.8	0.96	597.5	1.2	568.5
2100	0.00	639.1	0.82	614.0	0.96	601.6	1.2	573.[illegible]
2250	0.00	6[illegible].2	0.72	628.6	0.98	605.2	1.2	580.[illegible]
2400	0.00	659.0	0.62	642.0	1.1	599.0	1.2	588.[illegible]

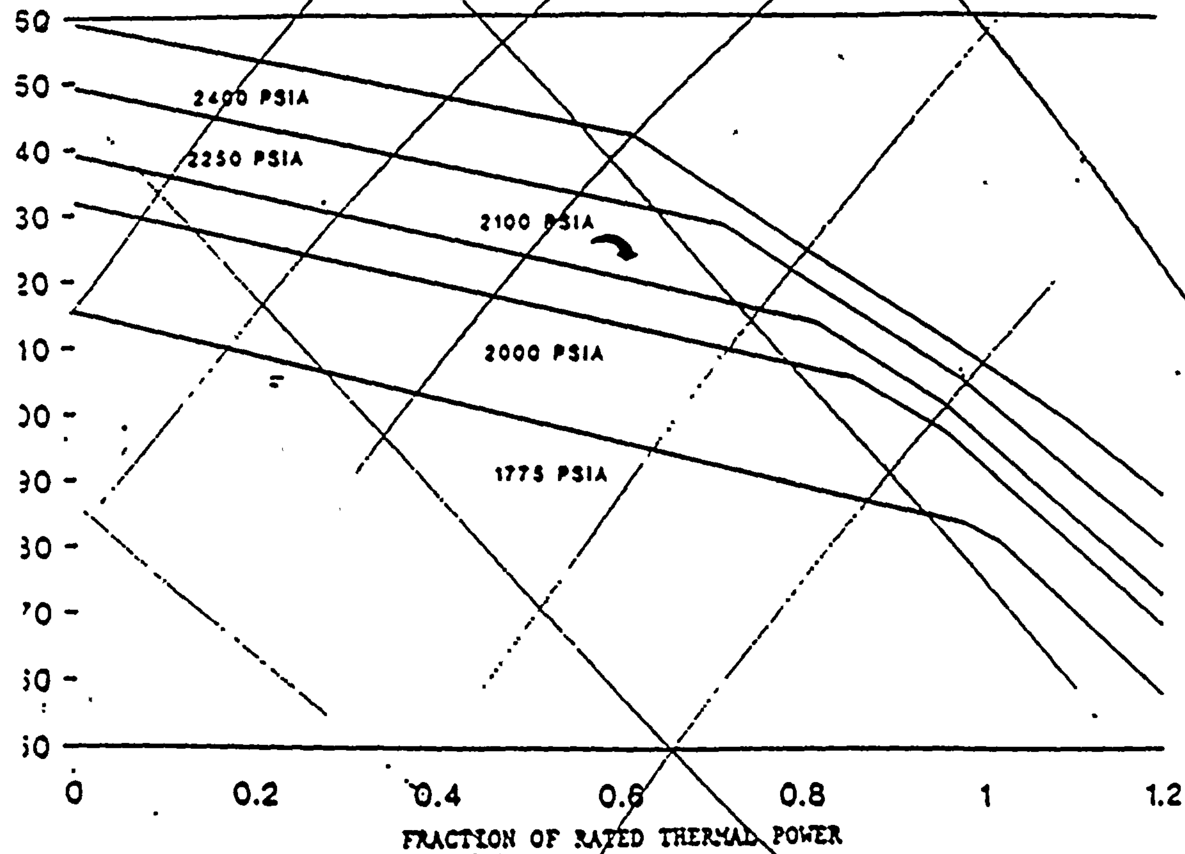


Figure 2.1-1 Reactor Core Safety Limits Four Loops in Operation

Replace with Insert 2.1-1

Insert 2.1-1, pg 2-2

Design Flow Rate = 91,600 gpm/loop

Description of Safety Limits

~~(Full VANTAGE 5 Core)~~

Pressure (psia)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)	Power (frac)	Tavg (°F)
1775	0.00	615.1	1.10	580.0	1.18	577.4	1.2	576.4
2000	0.00	632.2	1.12	597.6	1.14	596.0	1.2	589.4
2100	0.00	639.2	1.08	606.5	1.10	604.8	1.2	593.5
2250	0.00	649.4	1.02	619.5	1.10	610.9	1.2	599.7
2400	0.00	659.0	0.96	631.9	1.1	616.7	1.2	605.7

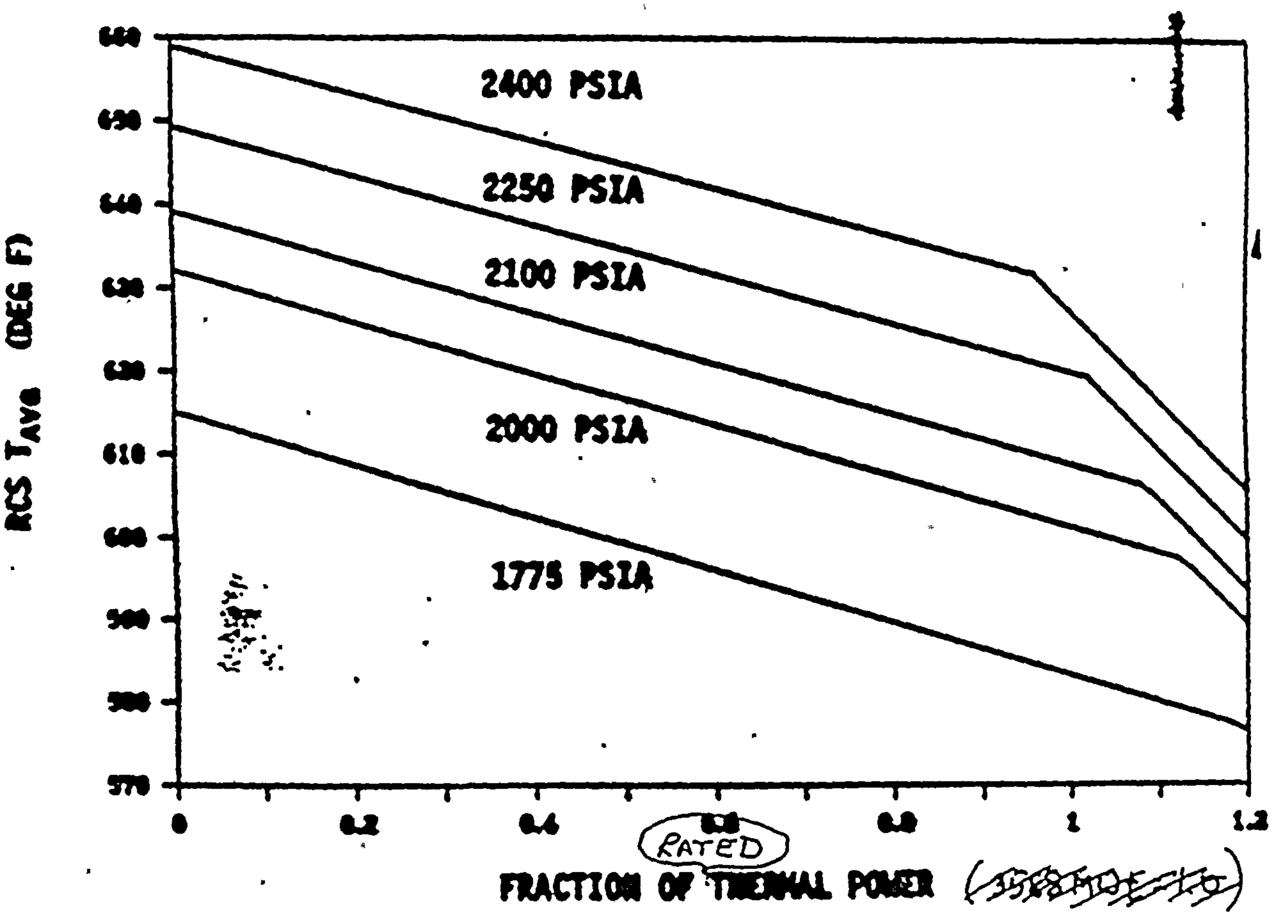


Figure 2.1-1 Reactor Core Safety Limits
Four Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
5. Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3 x 10^5 counts per second
7. Overtemperature Delta T	See Note 1	See Note 3
8. Overpower Delta T	See Note 2	See Note 4
9. Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10. Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11. Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12. Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

* Design flow is ~~[illegible] gpm per loop~~. 1/4 Reactor Coolant System total flow rate from LCO 3.2.5

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-2 correlation and W-3 correlation for conditions outside the range of WRB-2. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-2 correlation for Vantage-5 fuel, and the W-3 correlation for ~~[illegible]~~ conditions which fall outside the range of applicability of the WRB-2). The correlation DNBR limits are established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for WRB-2 and 1.3 for the W-3).

Delete

In meeting the DNB design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are statistically combined with the DNBR correlation statistics such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to a calculated design limit DNBR. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR correlation statistics, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Cook Nuclear Plant Unit 2, the design DNBR values are 1.23 and 1.22 for Vantage-5 fuel typical and thimble cells, respectively, ~~[illegible]~~. In addition, margin has been maintained in both ~~fuel types by~~ performing safety analyses to a safety analysis limit DNBR. The margin between the design and safety analysis limit DNBR is used to offset known DNBR penalties (i.e., transition core penalties, rod bow, etc.) and provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature below which the calculated DNBR is no less than the design DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1:

Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2[(1 + \tau_1 S)/(1 + \tau_2 S)](T - T') + K_3(P - P') - f_1(\Delta I)]$

Where:	ΔT_o	= Indicated ΔT at RATED THERMAL POWER
	T	= Average temperature, ^{o}F
	T'	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to ~~[illegible]~~ 581.3 ^{o}F
	P	= Pressurizer Pressure, psig
	P'	= ~~[illegible]~~ psig ~~{~~Indicated RCS nominal operating pressure~~}~~ (2235 psig or 2085 psig)
	$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	= The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	= Time constants utilized in the lead-lag controller for T_{avg}; τ_1 = ~~[illegible]~~ 22 secs, τ_2 = 4 secs.
	S	= Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

K1 = ~~[illegible]~~ 1.17

K2 = ~~[illegible]~~ 0.0268

K3 = ~~[illegible]~~ 0.00111

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-35~~ -16 percent and +6 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER);

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds ~~-35~~ -16 percent, the ΔT trip setpoint shall be automatically reduced by ~~[illegible]~~ 2.05 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by ~~[illegible]~~ 2.7 percent of its value at RATED THERMAL POWER.

Note a - See Insert 2-8

Insert 2-8 for Note a for tech spec page 2-8

Note a T' shall be set to a value equal to or less than the Indicated T_{avg} at RATED THERMAL POWER. Indicated T_{avg} and T' can be set to any value within the range of 547 to 581.3 deg. F.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \{K_4 - K_5[\tau_3 S/(1+\tau_3 S)]T - K_6[T-T'] - f_2(\Delta I)\}$

Where:

ΔT_o	= Indicated ΔT at rated power
T	= Average temperature, ^oF
T'	= Indicated T_{avg} at RATED THERMAL POWER less than or equal to 576.0 ^oF
K_4	= 1.08
K_5	= $0.02/^oF$ for increasing average temperature and 0 for decreasing average temperature
K_6	= 0.00197 for T greater than T'; K_6 = 0 for T less than or equal to T'
$\tau_3 S/(1+\tau_3 S)$	= The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	= Time constant utilized in the rate lag controller for T_{avg}; τ_3 = 10 secs.
S	= Laplace transform operator
$f_2(\Delta I)$	= 0.0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than ~~[illegible]~~ percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than ~~[illegible]~~ percent ΔT span.

Limits Provided by AEPSC

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 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.

Delete

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: (Continued)

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 2.1% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

Margin between the safety analysis DNBRs and the design limit DNBRs is maintained. (Safety analyses DNBRs: 1.69 and 1.61 for the Vantage 5 typical and thimble cells, respectively~~, and 1.[illegible] and [illegible] for the ANF fuel typical and thimble cells~~. Design limit DNBRs: 1.23 and 1.22 for the Vantage 5 typical and thimble cells, respectively~~, and [illegible] and [illegible] for the ANF fuel typical and [illegible]~~.) A fraction of this margin is utilized to accomodate applicable transition core penalties and the appropriate fuel rod bow DNBR penalty for the Vantage 5 fuel (equal to 1.3% per WCAP-8691, Rev. 1). The remainder of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

Delete

Delete

POWER DISTRIBUTION LIMITS

DNB AND T_{avg} OPERATING PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the following operational indicated limits:

Insert 1

~~a. DNB~~

~~1. Reactor Coolant System T_{avg}~~	~~Less than or equal to 578.7°F*~~
~~2. Pressurizer Pressure~~	~~Greater than or equal to 2200 psig*/**~~
~~3. Reactor Coolant System Total Flow Rate~~	~~Greater than or equal to 366,400 gpm***~~

~~b. T_{avg}~~

~~1. Reactor Coolant System T_{avg}~~	~~Greater than or equal to 563.9°F*~~

Delete

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

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* Indicated average of at least three OPERABLE instrument loops.

** Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RTP

*** Indicated value

insert 1 to page 3/4 2-15

1. Reactor Coolant System T_{avg}
   $T_{avg} \leq (581.3+5.1-$ Indication error$^{(1)}) °F$*

2. Pressurizer Pressure - for normal Pressure Operation,
   Przr Pres $\geq$ (2235-63+ Indication error$^{(1)}$) psig*/**

   at Reduced Pressure Operation,
   Przr Pres $\geq$ (2085-63+ Indication error$^{(1)}$) psig*/**

3. Reactor Coolant System $\geq$ 366,400 gpm*** Total Flow Rate

(1) Indication error to be provided by AEPSC for these limits

note that once AEPSC determines the indication errors for these DNB limits the absolute limit can be calculated and inserted into the LCO

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Manual Initiation	------------ See Functional Unit 9	----------
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	Less than or equal to 1.1 psig	Less than or equal to 1.2 psig
d. Pressurizer Pressure--Low	Greater than or equal to ~~[illegible]~~ psig 1815	Greater than or equal to ~~[illegible] psig~~
e. Differential Pressure Between Steam Lines--High	Less than or equal to 100 psi	Less than or equal to 112 psi
f. Steam Line Pressure--Low	Greater than or equal to 600 psig steam line pressure 500	Greater than or equal to ~~[illegible]~~ psig steam line pressure

AEPSC SCOPE

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINTS	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	------------- See Functional Unit 9 ----------	
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	Less than or equal to 2.9 psig	Less than or equal to 3.0 psig
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.6×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to 4.5×10^6 lbs/hr at full load.	Less than or equal to a function defined as follows: A Delta-p corresponding to 1.75×10^6 lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to to 4.55×10^6 lbs/hr at full load.
	T_{avg} greater than or equal to 541°F	T_{avg} greater than or equal to 539°F
e. Steam Line Pressure--Low	Greater than or equal to 600 psig steam line pressure 500	Greater than or equal to [illegible] psig steam line pressure HEPSC Scope
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	Less than or equal to 67% of narrow range instrument span each steam generator	Less than or equal to 68% of narrow range instrument span each steam generator

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± ~~1%~~.* # 3%

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes** and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electrical power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

See Insert 4-4

Insert 4-4 for # footnote on tech spec page 3/4 4-4 and 3/4 4-5

\# The pressurizer code safety valve shall be reset to the nominal value ±1% whenever found outside the ±1% tolerance.

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± ~~[illegible]~~.*+ 3%

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

+ See Insert 4-4

REACTOR COOLANT SYSTEM

LIMITING CONDITIONS FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be wi nin each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Determining the seal line resistance at least once per 31 days when the average pressurizer pressure is within 20 psi of its nominal full pressure value. The seal line resistance measured during the surveillance must be greater than or equal to 2.27 E-1 ft/gpm^2. The seal line resistance, R_{SL}, is determined from the following expression:

$$R_{SL} = \frac{2.31\,(P_{CHP} - P_{SI})}{Q^2}$$

where: P_{CHP} = charging pump header pressure, psig

P_{SI} = 2262 psig (high pressure operation)

P_{SI} = 712 psig (low pressure operation)

2.31 = conversion factor (12 in/ft)^2/(62.3 lb/ft^3)

Q = the total seal injection flow, gpm

The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2. Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

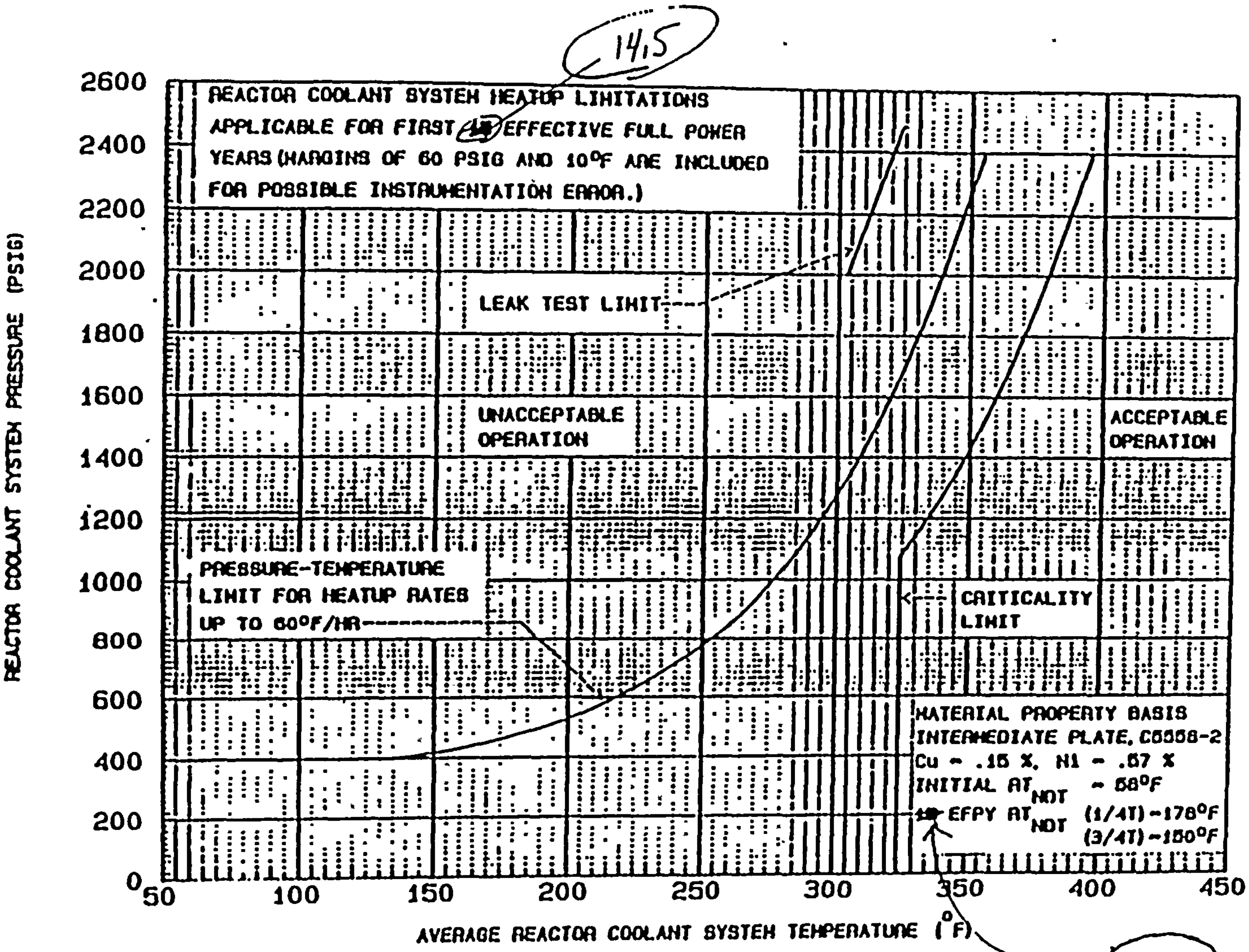

FIGURE 3.4-2

REACTOR COOLANT SYSTEM, PRESSURE — TEMPERATURE, LIMITS FOR 60 °F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

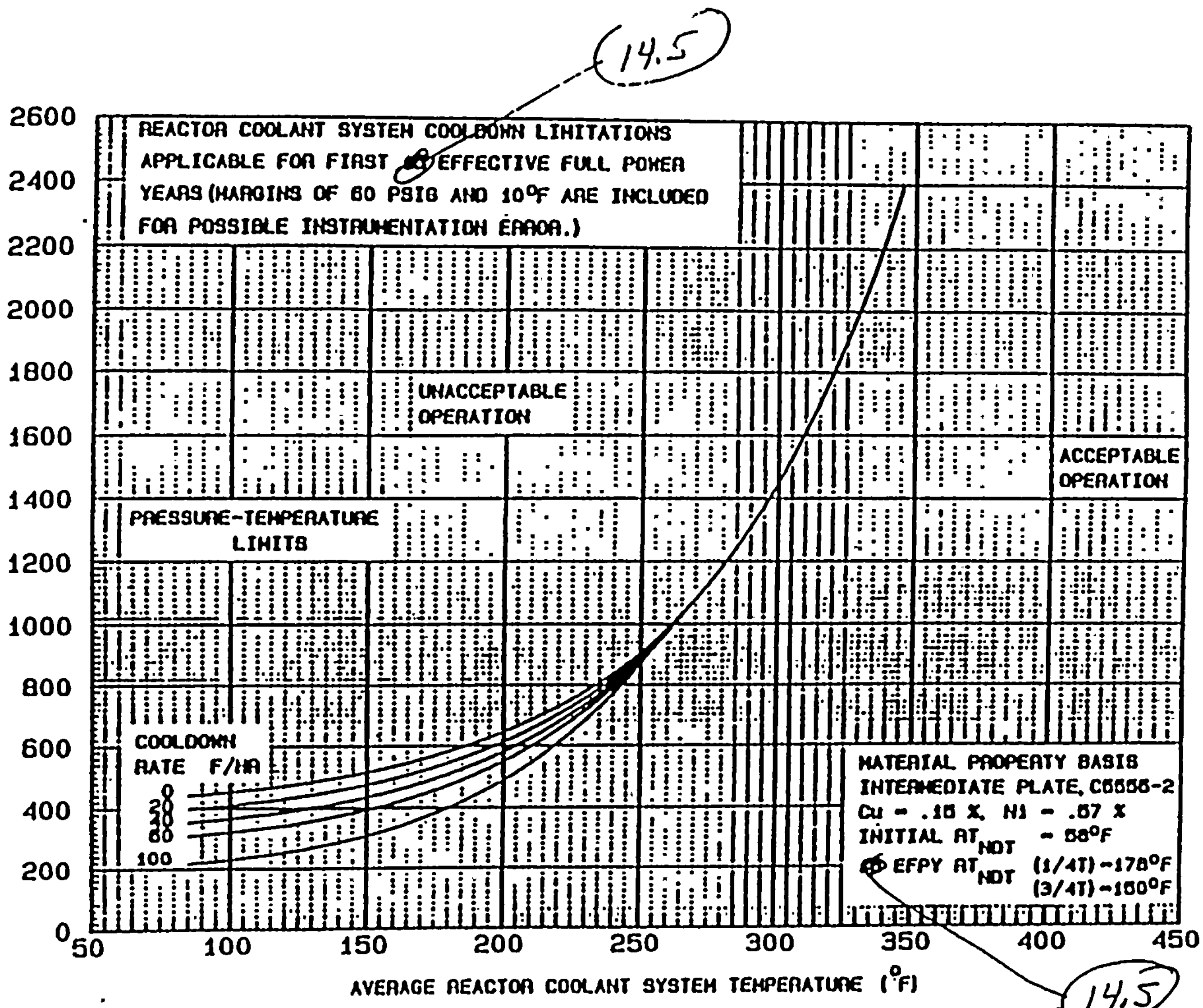

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE – TEMPERATURE LIMITS FOR VARIOUS COOLDOWN RATES

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump,

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

Delete

~~f. All safety injection cross-tie valves open.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

~~b. With a safety injection cross-tie valve closed, restore the cross-tie valve to the open position or reduce the core power level to less than or equal to 3250 MW within one hour. Specification 3.0.4 does not apply.~~

~~c.~~ b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

Delete

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of ~~125~~ 200,000 gallons of water.

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

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

3/4.2 POWER DISTIBUTION LIMITS
BASES

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

Replace with Insert 1

3/4.2.5 DNB PARAMETERS

~~The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The T_{avg} less than or equal to 578.7°F and pressurizer pressure greater than or equal to 2200 psig are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. The T_{avg} greater than or equal to 543.9°F is conservative to a safety analysis performed to demonstrate that the plant may operate on a linear control program where the analytical limit of T_{avg} at 100% RATED THERMAL POWER may range from 541.4°F to 580.1°F. The limit of 543.9°F contains a margin of 1.1°F. The core may be operated with indicated vessel average temperature at any value between the upper and lower limits. Pressurizer pressure is limited to a single nominal setpoint with the lower limit of the indicated value setpoint set forth in the specifications. The T/S value was selected for consistency with Unit 1 and contains a margin of 6 psi. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.~~

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

Insert 1 to pg B 3/4 2-5

~~INSERT A~~

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The T_{avg} less than or equal to 581.0°F and pressurizer pressure greater than or equal to ~~[illegible]~~ psig (for nominal pressurizer operating pressure of 2235 psig) or ~~[illegible]~~ psig (for nominal pressurizer operating pressure of 2085 psig) are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. Pressurizer pressure is limited to either of two nominal operating pressures of 2235 psig or 2085 psig, with the corresponding indicated limits set forth in the specifications. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR value for the current fuel type throughout each analyzed transient.

(581.3 - 5.1 - Indication Error)°F

(2235 - 63 + Indication Error)

(2085 - 63 + Indication Error)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown, the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of ~~[illegible]~~ EFPY. 14.5

The reactor vessel materials have been tested to determine their initial RT_{NDT}: The results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

EMERGENCY CORE COOLING SYSTEMS

ACCUMULATORS (Continued)

allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

If a safety injection cross-tie valve is closed, safety injection would be limited to two lines assuming the loss of one safety injection subsystem throu a single failure consideration. The resulting lowered flow requires a decrease in THERMAL POWER to limit the peak clad temperature within acceptable limits in the event of a postulated small break LOCA.

Delete

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is ~~[illegible]~~ 11.66 psig, ~~The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 9.7 psig which is less than the design pressure and is consistent with the accident analyses.~~ which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to ~~[illegible]~~ 11.66 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum ~~contained~~ usable volume of 175,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water System as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the ~~contained~~ usable water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The Essential Service Water System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the Essential Service Water System is in operation whenever the Essential Service Water System is the supply source for the auxiliary feedwater pumps.

reflects the volume of water above the centerline of the discharge pipe.

PLANT SYSTEMS

BASES

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The ~~contained~~ water volume limit ~~includes an~~ An allowance for water not usable because of tank discharge line location or other physical characteristics is not required.

usable

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

With one steam generator stop valve inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the unit hot. The 8 hour completion time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the steam generator stop valves. If the steam generator stop valve cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the [illegible] placed in MODE 2 within 6 hou[illegible] 2 and 3 action statement entered. The completion times are reasonable, based on operating experience, to reach MODE 2 and to close the steam generator stop valves in an orderly manner and without challenging unit systems.

ATTACHMENT 6 TO AEP:NRC:1223

DESCRIPTION OF ANALYSES PERFORMED BY WESTINGHOUSE ELECTRIC CORPORATION FOR COOK NUCLEAR PLANT UNIT 2

WCAP 14489