REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8605090013 DOC. DATE: 86/05/02 NOTARIZED: NO DOCKET #
FACIL: 50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH. NAME AUTHOR AFFILIATION
ALEXICH, M. P. Indiana & Michigan Electric Co.
RECIP. NAME RECIPIENT AFFILIATION
DENTON, H. R. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Forwards rept supporting proposed operation of facility at core average temp approx 13 F lower than normal programmed value for 80% rated thermal power. Lower temp will reduce rate of degradation of steam generation.

DISTRIBUTION CODE: A001D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 27
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PWR-A ADTS	1	0	PWR-A EB	1	1
	PWR-A EICSB	2	2	PWR-A FOB	1	1
	PWR-A PD4 PD 01	5	5	WIGGINGTON, D	1	1
	PWR-A PSB	1	1	PWR-A RSB	1	1
INTERNAL:	ADM/LFMB	1	0	ELD/HDS3	1	0
	NRR/DHFT/TSCB	1	1	NRR/ORAS	1	0
	REG FILE 04	1	1	RGN3	1	1
EXTERNAL:	24X	1	1	EG&G BRUSKE, S	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1			

TOTAL NUMBER OF COPIES REQUIRED: LTTR 24 ENCL 20

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

May 2, 1986
AEP:NRC:0916N

Donald C. Cook Nuclear Plant Unit No. 2
Docket No. 50-316
License No. DPR-74
OPERATION AT REDUCED POWER AND TEMPERATURE

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

This letter and attached report are being submitted for your consideration and concurrence. The attached report supports proposed operation of Unit 2 of the Donald C. Cook Nuclear Plant at a core average temperature (Tave) approximately 13°F lower than the normal programmed value for 80% of rated thermal power. The purpose of this lower Tave operation is to reduce the rate of degradation of the Unit 2 steam generators by reducing to approximately 581°F the hot leg temperature experienced by the steam generator tubes. We plan to implement the reduced Tave operation identified in this study as early in Cycle 6 as possible. Cycle 6 operation is currently expected to begin on June 1, 1986. In a telephone conversation on January 27, 1986, with your staff, we agreed to submit this report for NRC review.

By operating Unit 2 at only 80% of its rated thermal power, Indiana & Michigan Electric Company (IMECo) is taking the first step toward reducing steam generator tube failures. This report represents the second step, which is to ascertain what reactor coolant system (RCS) temperature reduction, at 80% rated thermal power, can be supported by existing safety analyses and Technical Specifications. At a later date, we expect to explore further opportunities for RCS temperature reduction through new safety analyses and, as necessary, Technical Specification changes. Although it is IMECo's intent at this time to operate Unit 2 at 80% power, due to steam generator considerations, we want to retain the option to return to the normal Tave program in order to achieve 100% power, should our system load require such output.

The attached report examines the events from Chapter 15 of the NUREG-0800 Standard Review Plan, and for these events, assesses the small change that operating at 80% rated thermal power and at 556°F Tave will have, as compared to operating at full power and at 574°F Tave, the normal analyzed operating point. The "baseline" results of these events, as analyzed for the normal

8605090013 860502
PDR ADOCK 05000316
P PDR

A001
1/1

full power operation, are those reported in our earlier submittals supporting Cycle 6 operation of Cook Unit 2. Although these "baseline" analyses are performed with computer codes and methods still being evaluated for NRC approval, such approval is expected prior to Cycle 6 operation. Therefore, this reduced power and temperature operation review is based on the "analyses of record," now that Cycle 5 has ended.

This report's review of the results of Standard Review Plan events occurring at reduced power and temperature operation shows that in all but one case the results are bounded by the analyses of record from full-power operation. In that one case, Event 15.2.8, Feedwater System Pipe Break, we propose accepting a possible small expulsion of water through pressurizer safety relief valves. As this event is an ANS Condition IV event, highly improbable, and not expected to occur during the life of the plant, we believe that use of a very small portion of the pressurizer safety relief valve capacity should be an acceptable result. Further, please note that Exxon's analysis for feedwater system pipe break makes the extremely conservative assumption of the complete loss of the affected steam generator. Therefore, it is unlikely that the pressurizer safety relief valve would be utilized.

Operation at reduced power and temperature in accordance with the assumptions of the attached report will require certain new setpoint and limit values that remain within that allowed by the Technical Specifications being proposed for Cycle 6 operation. These are:

1. The high trip setpoint on the power range neutron flux (Technical Specification Table 2.2-1) shall change from 109% of rated thermal power to (109% X 80%) of rated thermal power.

2. The Technical Specifications concerning heat flux hot channel factor (3/4.2.2) shall be implemented with the value of P redefined as:

$$P = \frac{\text{Thermal Power}}{\text{80\% of Rated Thermal Power}}$$

3. The overtemperature and overpower ΔT trip setpoints in Technical Specification Table 2.2-1 shall be implemented with ΔT_o, T' and T" being defined by the proposed reduced power and temperature operation. ΔT_o will be indicated ΔT at 80% power. T' and T" will be indicated $\mathring{T}ave$ at 80% power, where the programmed value of Tave is 556°F.

Upon your concurrence with this operation, we propose using administrative controls to implement these values.

We request your review of and concurrence with, this reduced power and temperature operation. We believe that this will further reduce the rate of our steam generator degradation, and we would appreciate your timely consideration.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

RBK
5/2/86

/tld

Attachment

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

Attachment

REVIEW OF STANDARD REVIEW PLAN CHAPTER 15 EVENTS FOR REDUCED POWER AND TEMPERATURE OPERATION AT D. C. COOK UNIT 2, CYCLES 5 AND 6

1.0 INTRODUCTION

In this report are presented the results of a review of Standard Review Plan[1] Chapter 15 events intended to support operation of D.C. Cook Unit 2 through Cycles 5 and 6 on a reduced power and temperature (RPT) program. Instituted to reduce steam generator tube degradation by reducing hot leg coolant temperatures, the proposed operating program is characterized by a maximum core power of 2728 MWt (80% of current rated thermal power) and an average coolant temperature at maximum power of 556°F. The review is structured to address the effect of the reduced power and temperature operating point on the results of the current analysis of record of Standard Review Plan Chapter 15 events. The analysis of record is defined in Table A-1 of the previously submitted plant transient analysis for Cycle 6[2] and expanded on in the Cycle 6 Disposition of Standard Review Plan Chapter 15 Events.[3]

The results of this review are summarized in Section 2.0. In Section 3.0 is given a general discussion of the effect of reduced power and temperature operation on the Chapter 15 event analyses and the results of this review for each event. Event discussions are numbered for convenience in accordance with the Standard Review Plan event numbering scheme. The references cited in this report are listed in Section 4.0.

This document is subject to the restrictions set forth on the first or title page

2.0 SUMMARY

The review of Standard Review Plan Chapter 15 events for D.C. Cook Unit 2 operating on a reduced power and temperature (RPT) program is summarized in this section. Results reported here support operation of the plant at 2728 MWt and an average temperature program defined by 547°F at zero load and 556°F at 2728 MWt. Because plant operating conditions in Modes 3 through 6 are not changed by the proposed RPT operating point, the review is limited to operational modes 1 and 2 (Table 2.1). Nominal plant conditions considered are given in Table 2.2. The results reported here are subject to the restriction of key Limiting Safety System Settings as noted in Table 2.3 and to the reduction of peak operating LHGRs by 20%. The average temperature program high program limit of 556°F is the nominal minimum value supported by the loss of normal feedwater evaluation (Event 15.2.7).

The effect of RPT operation on the results of events limiting with respect to DNBR and fuel centerline temperature is to significantly increase the margin to DNBR and fuel temperature limits. This improvement is due to increased DNBR and decreased fuel temperature at the RPT operating point relative to the rated power operating point, both due largely to the power reduction. Administrative restriction of the overtemperature ΔT, power range neutron flux (high), and overpower ΔT reactor trip setpoints is required to assure that the gain in margin to limits at initial conditions is preserved during transient event evolution.

Peak reactor vessel pressurization occurs during the loss of external load event (15.2.1). Due to reduced power, the vessel pressurization during that event will be reduced by the RPT operating point. The review indicates that the pressurizer liquid swell which occurs during the loss of normal feedwater event (15.2.7) is essentially unchanged by RPT operation relative to the reference analysis; an increased tendency to swell caused by lower initial condition coolant temperature is offset by the 20% reduction in decay heat production. Conservative calculations

This document is subject to the restrictions set forth on the first or title page

indicate that the relief capacity of the safety relief valves will not be exceeded for the main feed line break accident (15.2.8).

Margin to PCT limits for the loss of coolant accident (LOCA; event 15.6.5), is judged to be preserved by the 20% power and peak LHGR reduction. To achieve the peak LHGR reduction, the Technical Specification limits on F_Q should be interpreted with RATED THERMAL POWER assumed to be 80% of 3411 MWt, or 2728 MWt. This will result in the maximum allowed peak LHGR being reduced by 20% consistent with the power level reduction. The slightly negative effect on PCT of a reduced primary system temperature is thereby compensated.

This document is subject to the restrictions set forth on the first or title page

Table 2.1 Operational Modes 1 and 2 for D.C. Cook Unit 2

Mode	Reactivity Condition, k_{eff}	% Rated Thermal Power*	Average Coolant Temperature
1. Power Operation	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. Startup	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F

*Excluding decay heat.

This document is subject to the restrictions set forth on the first or title page

Table 2.2 Nominal Plant Conditions Considered in the Reduced Power and Temperature Event Review

Operating Parameter	Value
Core Thermal Power, MWt	2728
Vessel Average Coolant Temperature*, °F	556
Vessel Coolant Flow, lb/hr**	143.5×10^6 lb/hr
Steam Generator Pressure, psia	748.5

*Value represents the high program limit. T_{avg} program is linear between 547°F at no load and the high program limit at 2728 MWt.

**Coolant flow reflects 10% average steam generator tube plugging (maximum plugging equal or less than 15%).

This document is subject to the restrictions set forth on the first or title page

Table 2.3 Administrative Restrictions on Key Limiting Safety System Settings

LSSS	Current Safety Analysis Value	Administratively Restricted Safety Analysis Value
Power Range Neutron Flux (High)	$\leq$ 118% of 3411 MWt	$\leq$ 118% of 2728 MWt
Overtemperature ΔT	K1 = 1.391 T' = 574.1°F	K1* = 1.391 T' = 556°F
Overpower ΔT	K4 = 1.152 T" = 574.1°F	K4* = 1.152 T" = 556°F

* ΔT_0 in these setpoints redefined as "Indicated ΔT at RTP conditions."

This document is subject to the restrictions set forth on the first or title page

Table 2.4 Margin to Specified Acceptable Fuel Design Limits at the Reduced Power and Temperature Point

Fuel Design Parameter	SAFDL	Value: Full Power & Temp. Condition	Value: Reduced Power & Temp. Condition
MDNBR	1.17	1.787	2.018
Peak Pellet Centerline Temperature, °F	5080	3700*	3100*

*Based on a linear estimate of fuel centerline temperature as function of peak pellet LHGR.

This document is subject to the restrictions set forth on the first or title page

3.0 EVENT REVIEW FOR REDUCED POWER AND TEMPERATURE OPERATION

The basis and results for the event review are presented in this section. For convenience, the subsection numbers and event nomenclature are in accordance with those used in the Standard Review Plan.

In this review, each Standard Review Plan Chapter 15 event is evaluated to determine whether the margins to acceptance criteria reported for that event in the reference analyses[2,3] are unchanged or improved by the proposed reduced power and temperature operating conditions.

This review is limited to consideration of the at-power operating modes 1 and 2. Operation in Technical Specification modes 3-6 is unaffected by the proposed change in operating point, and is therefore adequately addressed by the reference analysis.[2,3] Technical Specification operating modes 1 and 2 for D.C. Cook Unit 2 are defined in Table 2.1. Nominal conditions considered in this review are tabulated in Table 2.2.

As noted in Table 2.4, a 37% improvement in margin to the DNBR limit and about 600°F improvement in margin to the peak pellet centerline temperature limit characterizes the proposed reduced power and temperature operating point relative to the full power and temperature operating point considered in the reference analysis. Thus, initial condition margin to these Specified Acceptable Fuel Design Limits is substantially improved at the reduced power and temperature operating point.

To maintain this large margin of conservatism through transient event evolution, administrative restrictions on the values of key reactor protection system Limiting Safety System Settings will be imposed. These restrictions are determined to maintain the same margin between reactor operating point and reactor trip as that evaluated in the reference analysis.[2,3] This ensures that the improvement in initial condition margins to DNBR and fuel temperature limits obtained at the reduced power and temperature operating point will be maintained throughout the

This document is subject to the restrictions set forth on the first or title page

evolution of the transient. Administrative restrictions on LSSS assumed in this event review are listed in Table 2.3.

Additionally, the maximum allowed peak LHGR is administratively reduced consistent with the 20% power reduction. This is done to maintain margin to the limits for the loss of coolant accident (Event 15.6.5).

Standard Review Plan Chapter 15 events are reviewed individually below.

This document is subject to the restrictions set forth on the first or title page

15.1.1 Decrease in Feedwater Temperature

The event considered is partial bypass of either the high or low pressure feedwater heaters, discussed in Section 15.1.1 of Reference 3. The event can challenge DNBR limits by an uncontrolled power excursion, the magnitude of which is approximately proportional to feedwater flow rate and feedwater heater heat load. Both feedwater flow rate and feedwater heater load are reduced by a reduction in initial condition power. The magnitude of the power excursion characteristic of the event, and therefore the severity of the DNBR and fuel centerline temperature excursions, are thus lessened by operation at the RPT point. Additionally, reduction of the power range neutron flux (high) reactor trip setpoint will limit the power to values less than the reference analysis initial condition. Vessel pressurization limits are not challenged. The event results for RPT operation are therefore bounded by those considered in the reference.[3]

15.1.2 Increase in Feedwater Flow

The Mode 1 event is failure of the turbine inlet pressure signal to the feedwater control system, resulting in full opening of the four feedwater regulating valves and full feed flow delivery. The resulting cooldown may cause an uncontrolled power excursion, challenging DNBR and fuel centerline temperature limits. Event consequences are mitigated for the RPT operating point by the significantly increased initial condition DNBR. The reduction of the power range neutron flux trip setpoint (high) assures that reactor power will not reach the level considered in the discussion of this event given in Reference 3, Section 15.1.2. Vessel pressurization limits are not challenged in this event. The event results for RPT operation are therefore bounded by those considered in Reference 3.

Initial conditions and event initiator are not significantly changed by RPT operation for the Mode 2 event. The event consequences are therefore essentially unchanged from those considered in the reference analysis.[3]

15.1.3 Increase in Steam Flow

A 14% increase in steam flow from rated power conditions due to opening of the turbine stop and control valves was analyzed and reported in Section 15.1.3 of Reference 2. An uncontrolled power excursion resulting from the increased load demand causes DNBR reduction and increased fuel centerline temperature relative to the initial condition. Initial condition DNBR is significantly increased at the RPT point, and reduction of the power range neutron flux trip setpoint (high) ensures the preservation of this increased margin to DNB limits. The event results are therefore bounded by those reported in Reference 2.

Initial conditions and event initiator are not significantly changed by RPT operation for the Mode 2 event. Event consequences are therefore essentially unchanged from those considered in the reference analysis (Section 15.1.3 of Reference 3).

15.1.4 Inadvertent Opening of Secondary Safety or Relief Valve

The event considered is inadvertent opening of a steam generator safety valve, discussed in Section 15.1.4 of Reference 3. Steam flow through the valve places an additional load demand on the reactor, causing an uncontrolled power excursion with consequent DNBR reduction and fuel centerline temperature increase. Decreased steam density at the RPT operating point due to secondary pressure reduction reduces the steam flow increase relative to the reference case, reducing the power excursion resulting from the event. Challenge to DNBR and fuel centerline temperature limits is significantly reduced at the RPT operating point by the increased initial condition DNBR and reduced fuel temperature and by the smaller power excursion. The power range neutron flux (high) and overtemperature ΔT trip setpoints are reduced for RPT operation to assure preservation of this increased margin to safety limits. Event consequences considered in Reference 3 therefore bound those expected for RPT operation. It is noted that the Mode 2 event for RPT operation is

This document is subject to the restrictions set forth on the first or title page

essentially unchanged with respect to event initiator and initial condition relative to the reference analysis.

15.1.5 Steam System Piping Failures Inside and Outside of Containment

The reference analysis is given in Section 14.2.5 of Reference 4, and is analyzed from the hot zero power state. The proposed RPT operation will not affect hot zero power operating conditions. The event will thus proceed for RPT operation as reported in the reference analysis.

The event is being analyzed for ENC fuel on a delayed submittal schedule. Because the controlling parameters of the event (steam line cross sectional area and EOC moderator temperature coefficient) are unchanged from the reference analysis, event evolution should not differ significantly from the reference analysis.

15.2.1 Loss of External Load

The reference analysis results are given in Section 15.2.1 of Reference 2. The event is chiefly of concern due to the challenge posed to vessel pressurization limits. Primary system pressurization is controlled by the magnitude of the load rejection, proportional to initial condition power level. Because initial condition power level is reduced by 20% for RPT operation relative to the event analyzed in Reference 2, vessel pressurization resulting from loss of external load is bounded for RPT operation by that reported in Reference 2.

It is noted that primary coolant heatup rate for a given load rejection will be slightly increased by a reduced coolant temperature due to reduced coolant specific heat capacity. The effect is small, compensated by the increased coolant density, and outweighed by the power reduction for RPT operation.

The event is also analyzed in Reference 2 to assess DNBR reduction due to coolant heatup. The case simulation resulted in a reactor trip on the

overtemperature ΔT setpoint. The reduction of the ΔT_0 constant and reference temperature setting in that setpoint for RPT operation assure reactor trip at higher DNBRs. MDNBR for the event will thus be significantly improved for RPT operation relative to that calculated in the reference.

15.2.2 Turbine Trip, 15.2.3 Loss of Condenser Vacuum

These events are similar to 15.2.1, Loss of External Load. The discussion of event 15.2.1 above is applicable to these events as well.

15.2.4 Inadvertent Main Steam Isolation Valve (MSIV) Closure

This event is primarily of concern in BWRs and is therefore not addressed here.

15.2.5 Steam Pressure Regulator Failure

The Donald C. Cook Unit 2 plant has no steam pressure regulating devices except those considered in events 15.2.1 through 15.2.3.

15.2.6 Loss of Nonemergency A.C. Power

The event is discussed in Section 15.2.6 of Reference 3 and determined to be bounded in modes 1 and 2 by events 15.3.1, Loss of Forced Reactor Coolant Flow, and 15.2.7, Loss of Normal Feedwater. This determination is unaffected by RPT operation. The effect of RPT operation on this event is as described in Sections 15.2.7 and 15.3.1 of this report.

15.2.7 Loss of Normal Feedwater

The reference analysis is given in Section 15.2.7 of Reference 2. The event is evaluated to assess the adequacy of relief capacity and setpoint of the steam generator safety valves, auxiliary feedwater capacity, and

This document is subject to the restrictions set forth on the first or title page

steam generator inventory to maintain primary system pressure below the 110% pressure vessel design rating and to avoid expulsion of liquid from the primary pressurizer safety valves. These criteria assure long-term cooling capability and the attainment of a safe shutdown condition.

Reduced reactor power will mitigate the primary coolant volumetric swell due to reduced decay energy production. Reduced initial condition average coolant temperature will increase the primary coolant temperature rise, resulting in increased volumetric swell. The effect of RPT operation on event consequences is evaluated by a hand calculation, key assumptions of which are described below. Peak pressure for the event will be bounded by that reported for the loss of external load (Event 15.2.1) in Reference 3.

The hand calculation of coolant temperature increase assumes that two motor driven auxiliary feedwater pumps (MDAFP) are available, feeding all four steam generators. During the event, peak average coolant temperature rises above the saturation temperature of the steam generator safety valve pressure setpoint (sink temperature) by an amount proportional to power production; a peak average coolant temperature is evaluated for the RPT case as:

$$T_{max}^{RPT} = 0.8 \times (T_{max}^{Ref} - T_{sink}) + T_{sink}$$

where

T_{max}^{RPT} = Event maximum average coolant temperature for RPT conditions

T_{max}^{Ref} = Event maximum average coolant temperature for reference case (pumps off)

T_{sink} = Saturation temperature at steam generator safety valve setpoint pressure

This document is subject to the restrictions set forth on the first or title page

It is noted that T_{max}^{Ref} is taken at about 200 seconds into the transient depicted in Figure 15.2.7.2 of Reference 2. (The temperature increase depicted after that time is due to dryout of two steam generators not receiving auxiliary feedwater in the reference analysis; that increase will not occur in the case of two available MDAFPs, in which all steam generators receive auxiliary feedwater.)

The reference analysis demonstrated that a primary temperature increase of about 33°F could be tolerated without expulsion of primary liquid from the pressurizer valves due to filling of the pressurizer. This increase will be acceptable from RPT conditions also. A minimum acceptable initial condition temperature, $T_{i.c.}^{RPT}$, is thus estimated for RPT operation as

$$T_{i.c.}^{RPT} = T_{max}^{RPT} - 33°F .$$

The nominal average temperature of 556°F established for RPT operation is obtained from $T_{i.c.}^{RPT}$ by the incorporation of a 4°F temperature uncertainty allowance and a 4°F calculation uncertainty allowance.

Based on this calculation, a nominal average coolant temperature of 556°F will result in no greater coolant volumetric swell at RPT operation than was reported for the reference analysis.

This document is subject to the restrictions set forth on the first or title page

15.2.8 Feedwater System Pipe Breaks

The event considered is the break of one main feedwater pipe between the steam generator and check valve. The event is evaluated to demonstrate the adequacy of the auxiliary feedwater system to prevent overpressurization of the reactor coolant system and to prevent uncovery of the reactor core. The reference analysis for a feedwater system pipe break is presented in Section 15.2.8 of Reference 2, and addresses the heatup branch of the event. The cooldown branch is bounded with respect to acceptance criteria by Event 15.1.5.

Reduced reactor power will mitigate the primary coolant volumetric swell due to reduced decay energy production. Reduced initial condition average coolant temperature will increase the primary coolant temperature rise, resulting in increased volumetric swell. The effect of RPT operation on event consequences is evaluated by a hand calculation, key assumptions of which are described below.

During the event, peak average coolant temperature rises above the saturation temperature of the steam generator safety valve pressure setpoint (sink temperature) by an amount proportional to power production; a peak average coolant temperature is evaluated for the RPT case as:

$$T_{max}^{RPT} = 0.8 \times (T_{max}^{Ref} - T_{sink}) + T_{sink}$$

where

T_{max}^{RPT} = Event maximum average coolant temperature for RPT conditions

T_{max}^{Ref} = Event maximum average coolant temperature for reference case (pumps off)

T_{sink} = Saturation temperature at steam generator safety valve setpoint pressure

It is noted that T_{max}^{Ref} is taken at about 1000 seconds into the transient depicted in Figure 15.2.8.2 of Reference 2.

Mass influx to the pressurizer during this heatup, ΔM_{RPT}, is taken as

$$\Delta M_{RPT} = V \Delta \rho_{RPT}$$

where V is total primary system volume (excluding pressurizer and surge line) and $\Delta\rho$ is the difference in coolant average density between the initial state (556°F T_{ave}, 2290 psia) and the state at maximum temperature (T_{max}^{RPT}, uncertainty adjusted pressurizer relief valve setpoint of 2310 psia). The combination of maximum initial condition pressure and minimum final state pressure maximizes calculated mass influx to the pressurizer. A value of ΔM is also calculated for the reference analysis, ΔM_{Ref}, which differs from ΔM_{RPT} only in the initial and final average coolant temperatures employed.

Pressurizer liquid volume for the RPT event, V_{RPT}, is evaluated from:

$$V_{RPT} = V_{i.c.} + \frac{\Delta M_{RPT}}{\Delta M_{Ref}} (\Delta V_{Ref}^{PZR})$$

where $V_{i.c.}$ is the initial pressurizer liquid volume, and ΔV_{Ref}^{PZR} is the difference between initial pressurizer volume and maximum pressurizer volume for the reference analysis. About 125 cu.ft. of water is expelled as liquid through the primary relief valves, estimated as the difference between V_{RPT} and the pressurizer volume of 1800 cu.ft.

Because surge flow rates are minimal during the phase of the event when pressurizer liquid volume is maximized, primary safety relief capacity is not strongly challenged. The three pressurizer safety relief valves will relieve 15 cfs of liquid. The surge flow rate in the time period of interest is less than 0.3 cfs. Thus, only about 1/50 of the available safety relief capacity is utilized. Pressurizer pressure is therefore

This document is subject to the restrictions set forth on the first or title page

expected to reach but not significantly exceed the safety relief valve setpoint of 2500 psia.

15.3.1 Loss of Forced Reactor Coolant System Flow Rate

Coastdown of all four reactor coolant pumps from the rated power condition was analyzed and reported in Section 15.3.1 of Reference 2. The event is of concern because of DNBR reduction caused by coolant flow reduction. The MDNBR for the event depends on the initial condition DNBR and the characteristics of the flow coastdown. The latter factor is not significantly affected by operation at the reduced RPT point, while the initial condition DNBR is significantly increased. The MDNBR for this event is therefore improved by the RPT operating point, and event results are bounded by those reported in Reference 2.

The mode 2 events are considered in Section 15.3.1 of Reference 3. Initial conditions and event initiators for these events are not significantly changed by the proposed RPT operation; event consequences are therefore as considered in Reference 3.

15.3.2 Flow Controller Malfunction - See 15.4.5

15.3.3/.4 Reactor Coolant Pump Rotor Seizure/Shaft Break

The event was analyzed at rated power conditions and reported in Section 15.3.3 of Reference 2. The effects of RPT operation on event consequences are as discussed above for event 15.3.1. Event results for RPT operation are therefore bounded by those reported in Reference 2 and, for mode 2, by those considered in Sections 15.3.3 and 15.3.4 of Reference 3.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

The reference analysis results are given in Section 15.4.1 of Reference

This document is subject to the restrictions set forth on the first or title page

2. The event is analyzed at hot zero power conditions to assess the challenge to DNBR and fuel temperature limits. Because the hot zero power operating point is unchanged by RPT operation, the reference analysis results are applicable to RPT operation.

15.4.2 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

The reference analysis results are given in Section 15.4.2 of Reference 2. The event is analyzed at rated, low, and mid-power levels at BOC and EOC for a spectrum of reactivity insertion rates. The event is characterized by uncontrolled power ascension due to reactivity insertion on rod withdrawal, and by coolant temperature increase due to the consequent reactor power-thermal load mismatch. The event is principally of concern because of DNBR reduction and fuel temperature increase driven by the power and temperature increase.

Event MDNBRs occur shortly after reactor trip at conditions determined by the overtemperature ΔT setpoint and the power range neutron flux (high) setpoint. Reduction of the high flux trip setpoint by 20% of rated power will result in significantly higher MDNBRs due to decreased power at trip for events from RPT conditions relative to the reference analysis (Table 2.3). Reduction of the overtemperature ΔT trip setpoint (Table 2.3) is equivalent to reducing reactor power at trip by 20%, yielding improved DNBRs for the RPT cases relative to the reference analysis. Fuel temperatures for the event are also decreased by the decreased reactor power at trip. The reference analysis results therefore bound those expected at RPT conditions.

15.4.3 Control Rod Misoperation

The control rod misoperation events encompass transient and steady state configurations resulting from different event initiators. The specific events addressed under this event category are:

(1) Dropped control rod assembly bank or group;

This document is subject to the restrictions set forth on the first or title page

(2) Dropped control rod assembly;
(3) Statically misaligned control rod assembly; and
(4) Single control rod assembly withdrawal.

These events significantly challenge the acceptance criteria only in mode 1 operation. The reference analysis is reported in Section 15.4.3 of Reference 2. These events are evaluated to assess the approach to DNBR and fuel temperature limits.

For the first three events listed above, DNBR reduction and fuel temperature increases relative to the initial conditions occur due to increased radial power peaking caused by the event. The RPT operating point is not expected to increase radial power peaking factors for these events significantly, so that DNBR reduction and fuel temperature increase relative to the initial condition will not be significantly larger than for the reference analysis. The large increase in initial condition DNBR and the large reduction in fuel temperature will therefore be preserved through these events, and the reference analysis results are bounding of RPT operation.

For the single control rod assembly withdrawal event, DNBR and fuel temperature transients are determined by radial power peaking increases in the region of the withdrawn rod and by the same power and coolant temperature excursion described in 15.4.2 for an uncontrolled rod withdrawal at power. Per the discussion in 15.4.2, the DNBR reduction and fuel temperature increase for this event at RPT conditions will be improved significantly relative to the reference analysis. The radial power peaking factor will not increase significantly as a result of RPT operation. The reference analysis results thus bound those expected for the single control rod withdrawal event.

15.4.4 <u>Startup of An Inactive Loop at an Incorrect Temperature</u>

The D.C. Cook Unit 2 plant is currently prohibited from operating with

This document is subject to the restrictions set forth on the first or title page

less than 4 coolant loops and pumps in service in modes 1 and 2. The event is therefore not credible in these modes.

15.4.5 <u>Flow Controller Malfunction</u>

D.C. Cook Unit 2 has no flow control devices on the primary coolant loops. The event cannot therefore occur at D.C. Cook Unit 2.

15.4.6 <u>CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant</u>

The reference analysis is presented in Section 15.4.6 of Reference 2. The event is evaluated to assess the challenge to DNBR and fuel centerline temperature limits, and to assess the adequacy of available shutdown margin. The challenge to DNBR and fuel centerline temperature limits is as discussed for Event 15.4.2. The effect of RPT operation on the adequacy of shutdown margin is addressed here.

In the boron dilution event, a smaller reactivity insertion rate through the event results in a longer time to loss of shutdown margin. The reactivity insertion rate in this event is determined by the rate of boron concentration reduction, which is proportional to critical boron concentration at the initial state, and the boron worth coefficient. The critical boron concentration at the RPT operating point is smaller than that at hot zero power (HZP) conditions, resulting in a smaller rate of boron concentration reduction at RPT conditions than at HZP conditions. The boron worth coefficient at HZP conditions is more negative than at RPT conditions, resulting in a larger reactivity insertion for a given boron dilution. The reactivity insertion rate at RPT conditions is thus smaller than that at HZP conditions, being proportional to the product of boron worth coefficient and rate of boron concentration reduction. The RPT event will therefore have a longer time to loss of shutdown margin than the HZP event analyzed in the reference analysis. The adequacy of shutdown margin for modes 1 and 2 in RPT operation is thus bounded by the

This document is subject to the restrictions set forth on the first or title page

reference analysis for hot standby, given in Section 15.4.6 of Reference 2.

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

The reference analysis is given in Section 15.4.7 of Reference 2. The event is evaluated to assess the challenge to DNBR and fuel centerline temperature limits. The event is characterized by an increased radial power peaking factor compared to a properly loaded core. Reduction of DNBR and fuel temperature increases relative to the event analysis for a properly loaded core occur only if the radial peaking factor exceeds the Technical Specification $F_{\Delta H}$ limit. DNBR and fuel temperature transients are determined by the magnitude of the difference between $F_{\Delta H}$ for the misloaded core and the Technical Specification limit. That difference is not significantly affected by RPT operation. Therefore, because event consequences for a properly loaded core are significantly improved relative to the reference analyses(as discussed for each event in this review), the reference analysis of 15.4.7 is bounding of RPT operation.

15.4.8 Spectrum of Rod Ejection Accidents

The reference analysis is given in Section 7.3 of Reference 5. The event is evaluated to assess the challenge to the pellet energy deposition limit of 270 cal/g.[6] The controlling parameters of the event are the initial fuel enthalpy and the ejected control rod worth. For the HFP case, initial fuel enthalpy is reduced for RPT operation relative to the reference analysis. This results in a reduced pellet enthalpy deposition for the event at RPT, more than compensating for the small effect of a slight increase in ejected rod worth relative to the HFP case. Event consequences for RPT operation are therefore bounded by those given in the reference analysis for the HFP cases. Event consequences for the HZP cases under RPT operation are unchanged from those reported for the reference analysis because the HZP operating conditions are unchanged.

This document is subject to the restrictions set forth on the first or title page

15.5.1 Inadvertent Operation of the ECCS that Increases Reactor Coolant Inventory

The reference analysis[3] for modes 1 and 2 concludes that no threat to DNBR or fuel centerline temperature limits is expected to result from this event. Continued addition of water to the reactor coolant system via the charging pumps might result in primary system pressurization. Based on a comparison of maximum volumetric charging capacity with pressurizer safety valve relief capacity, the reference analysis concludes that no severe pressurization of the vessel will occur. The proposed RPT operation will not affect the volumetric charging capacity or safety relief capacity. The reference analysis therefore bounds the proposed RPT operation.

15.5.2 Inadvertent Operation of the Chemical and Volume Control System (CVCS) that Increases Reactor Coolant Inventory

The event considered is the inadvertent operation of the charging pumps, discussed in Section 15.5.2 of Reference 3. The potential consequences of the event in modes 1 and 2 are bounded with respect to pressurization of the reactor coolant system by Event 15.5.1, and with respect to possible boron dilution by Event 15.4.6.

15.6.1 Inadvertent Opening of a Pressurizer Relief Valve

The event considered is the inadvertent opening of a pressurizer safety valve. The event is characterized by a rapid depressurization of the primary system caused by steam relief through the valve, of concern because of the DNBR challenge offered by coolant depressurization before reactor scram. The reference analysis is reported in Reference 3.

The MDNBR for the event will be improved by operation at reduced power and temperature. The improvement is a result of increased DNBR margin at the RPT operating point, due to decreased reactor power.

This document is subject to the restrictions set forth on the first or title page

The reference analysis addresses the concern of core uncovery due to coolant inventory depletion. The reference analysis results are not affected by the proposed RPT operating point.

The results of this event reported in the reference analysis are therefore either improved or unchanged by the proposed RPT operation.

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

The event considered is the failure outside containment of a small line carrying primary coolant. The event is discussed in Section 5.4 of Reference 4. Charging flow and SI delivery are expected to avert core uncovery, so no fuel damage is expected to result. Radiological release therefore depends on primary coolant activity and the coolant leakage rate; neither would be changed by the proposed RPT operation.

15.6.3 Radiological Consequences of Steam Generator Tube Failure

The event considered is the complete severence of a single steam generator tube, resulting in leakage of primary coolant to the secondary system. Secondary relief valve action may thus result in the release of primary coolant-borne activity to the atmosphere. The reference system analysis is discussed in Reference 3 and reported in Section 14.2.4 of Reference 4.

The potential radiological release is controlled by the initial primary and secondary radioactivity levels, total primary to secondary leakage through the ruptured tube, and the amount of steam released through the secondary safety valves. Radioactivity levels are set by Technical Specification limits, unchanged by operation at the RPT point. The primary to secondary flow rate will not be significantly increased by the RPT operation. Steam release will be reduced for RPT operation due to the reduced decay heat load. The radiological consequences are thus less severe than those considered in Reference 3.

This document is subject to the restrictions set forth on the first or title page

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

D.C. Cook Unit 2 is not a BWR. The consequences of this event for a PWR are addressed under SRP Event 15.1.5.

15.6.5 Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (LOCA)

The reference analysis for large break LOCA is given in Reference 7. The event results are strongly dependent on initial power level, peak LHGR, and slightly dependent on initial condition coolant temperature. The power reduction results in less fuel stored energy and strongly reduces calculated clad PCT. The coolant temperature reduction retards reflood slightly and therefore tends to worsen calculated results. The 20% power and peak LHGR reduction is judged sufficient to offset the effect of the primary coolant temperature reduction of 18.1°F with sufficient margin.

The reference analysis for the small break LOCA is given in Reference 8. The peak clad temperature which occurs during the event is strongly dependent on decay power production in the hot rod, which depends on the initial condition hot rod power. At the reduced power and peak LHGR initial condition, hot rod power is reduced by 20% from that which characterizes full power operation; consequently, the driving force for peak rod heatup is substantially reduced at the proposed RPT operating point. The RPT operating point is therefore expected to improve event results relative to the reference analysis.

15.7 Radioactive Releases from a Subsystem or Component

This category of events deals with potential release paths for radioactive discharges from various plant subsystems and components. A number of different release paths are considered as described by the following specific events:

This document is subject to the restrictions set forth on the first or title page

15.7.3 Postulated Radioactive Release Due to Liquid-Containing Tank Failures

15.7.4 Radiological Consequences of Fuel Handling Accidents

15.7.5 Spent Fuel Cask Drop Accidents

Events 15.7.1 and 15.7.2 have been deleted from the Standard Review Plan.

None of these events are adversely affected by a reduction in reactor power and coolant average temperature. The reference analyses cited in Appendix A of the Cycle 6 Plant Transient Analysis report[2] for these events remain applicable to the proposed operating point.

This document is subject to the restrictions set forth on the first or title page

4.0 REFERENCES

(1) "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, July 1981.

(2) "Plant Transient Analysis for D.C. Cook Unit 2 with 10% Steam Generator Tube Plugging," XN-NF-85-64(P), Exxon Nuclear Company, Inc., Richland, WA, November 1985.

(3) "D.C. Cook Unit 2, Cycle 6 Safety Analysis Report: Disposition of Standard Review Plan Chapter 15 Events," XN-NF-85-28(P), Supp. 1, Exxon Nuclear Company, Inc., Richland, WA, October 1985.

(4) "Donald C. Cook Nuclear Plant Final Safety Analysis Report," as amended to July 1982, Indiana and Michigan Electric Company.

(5) "D.C. Cook Unit 2, Cycle 6 Safety Analysis Report," XN-NF-85-28, Exxon Nuclear Company, Inc., Richland, WA, July 1985.

(6) U.S. Nuclear Regulatory Commission Regulatory Guide 1.77.

(7) "Donald C. Cook Unit 2 Limiting Break LOCA-ECCS Analysis for 10% Steam Generator Tube Plugging," XN-NF-85-68, Exxon Nuclear Company, Inc., Richland, WA, August 1985.

(8) Indiana and Michigan Electric Company letters to U.S. NRC transmitting submittal requesting modification to piping geometry of the D.C. Cook Unit 2 Safety Injection Pump miniflow line, dated March 1 and March 15, 1984.

This document is subject to the restrictions set forth on the first or title page