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SOUTHERN COMPANY
Energy to Serve Your World℠

May 6, 1998

Docket Nos. 50-321
50-366

HL-5613

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant
Request for Additional Information on
Extended Power Uprate License Amendment Request

Gentlemen:

By letter dated August 8, 1997, Southern Nuclear Operating Company (SNC) submitted a Technical Specifications amendment request for the Edwin I. Hatch Nuclear Plant Units 1 and 2. The proposed amendment increases the authorized maximum power level for both units from the current limit of 2558 MWt to 2763 MWt.

By letter dated February 10, 1998, the NRC requested SNC to provide additional information based on the August 8th submittal. By letter dated March 9, 1998, SNC provided a response to all of the questions associated with the February 10th request.

By letter dated March 27, 1998, the NRC requested SNC to provide additional information that involved review areas not included in the February 10th request. Enclosure 1 is SNC's response to the second NRC request for additional information (RAI) as discussed. Enclosure 1 does not provide responses to NRC Questions 56, 59 and 60, as discussed with the NRR Plant Hatch Project Manager on April 23, 1998. The responses to the remaining RAI questions will be submitted under a separate cover letter no later than May 25, 1998.

Although the enclosure of the subject RAI was issued non-proprietary, SNC as well as all involved parties have reviewed the requests and have determined that all questions are of a non-proprietary nature. Likewise, all responses that are provided in Enclosure 1 of this letter are considered by all parties to be non-proprietary.

If you have any additional questions on this subject, please contact this office.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

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A001

TWL/eb

Enclosure: Request for Additional Information on Extended Power Uprate License Amendment Request

cc: *Southern Nuclear Operating Company*
Mr. P. H. Wells, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

Enclosure

Edwin I. Hatch Nuclear Plant
Request for Additional Information on
Extended Power Uprate License Amendment Request

CONTAINMENT SYSTEMS AND SEVERE ACCIDENTS:

NRC QUESTION 56

Enclosure 6 of NEDC-32749P, Section 4.1, did not include confirmatory calculations with the SHEX code and the HXSIZ code at the extended power level. Please provide the comparative analysis results. Similar decay heat models should be used in SHEX code for both confirmatory and extended power level analyses for results to remain comparable.

SNC RESPONSE

The response to NRC Question 56 will be provided under separate cover letter.

NRC QUESTION 57

Enclosure 6, Section 4.1.2.3, indicated that due to changes in operating conditions with extended power uprate, the actual asymmetrical loads will increase slightly but will remain within the design margins. Please quantify the results.

SNC RESPONSE

Enclosure 6, Section 4.1.2.3, of the Extended Power Uprate Safety Analysis Report indicates that the asymmetrical loads on the reactor vessel, attached piping, and biological shield wall resulting from a postulated pipe break in the annulus will increase slightly due to changes in operating conditions. The increase in liquid subcooling for extended power uprate results in increased critical mass release rates for the liquid line breaks in the annulus (recirculation inlet and outlet, feedwater, and core spray). The increase in loading on structures and components was discussed qualitatively based only on the relative increase in blowdown mass and energy releases. However, a detailed evaluation for extended power uprate identified conservatisms in the subcompartment pressurization analysis methodology, which offset the increase in blowdown mass release rate.

The results for the recirculation outlet line break are representative of all the breaks analyzed for extended power uprate. Table 57-1 presents a comparison of calculated results for extended power uprate with the original FSAR results in section 6A of the Unit 2 FSAR. The extended power uprate results are less than those used in the original analysis of the reactor vessel and supporting structures. Annulus pressurization is not part of the Unit 1 licensing basis, although the design and evaluations are similar.

TABLE 57-1

**RECIRCULATION OUTLET LINE BREAK
COMPARISON OF RESULTS TO ORIGINAL ANALYSIS**

	Original (FSAR)	Extended Power Uprate
Max. Annulus Pressure (psia)	98.78	92.57
Max. Force on RPV (lb_f)	1.56×10^6	1.125×10^6
Max. Moment on RPV (ft-lb_f)	2.23×10^7	1.631×10^7

The differences in results are largely due to using Bechtel's NRC-approved methodology for subcompartment analysis instead of an earlier, more restrictive methodology. The original analyses for the recirculation inlet and feedwater line breaks included additional conservatism in the fractional split of blowdown between the annulus and the drywell. Similar reductions in maximum loads result for these breaks. The core spray line break is not a bounding break for the reactor vessel or the biological shield wall, therefore it was not reanalyzed.

NRC QUESTION 58

Enclosure 6, Section 4.5.2, indicated that the impact of an 8 percent increase in thermal power would cause the Unit 1 CAD System and Unit 2 recombiners to be initiated earlier. Provide the time responses.

SNC RESPONSE

The Unit 1 containment atmosphere dilution (CAD) system and the Unit 2 recombiners are discussed separately as follows:

Unit 1:

The combustible gas control analysis for the original power level of 2436 MWt was based on the guidelines provided by Safety Guide 7 (original issue of Regulatory Guide 1.7). Safety Guide 7 specifies the flammability limit for hydrogen and oxygen as 4 percent and 5 percent by volume respectively. The original design of the CAD system for Plant Hatch was based on maintaining the oxygen level at 4 percent or less by volume. The impact of the original power uprate on the CAD system was evaluated qualitatively, and the impact was insignificant when compared to the design margins.

The CAD system design for extended power uprate is also based on limiting the oxygen concentration inside the drywell and torus to a maximum of 4 vol %. The results of the revised analysis for extended power uprate indicate that an oxygen concentration of 4 vol % will be reached in about 1.55 days inside the torus and in about 2.14 days inside the drywell requiring the use of the CAD system. The CAD system will be started when the oxygen concentration reaches 4 vol %. As shown in Figure 58-1, with a constant CAD system nitrogen flow rate of 30 scfm each to the drywell and the torus, the oxygen concentration will be limited to 4 vol %. The total CAD system flow requirement of 60 scfm is well within the CAD system design capacity of 100 scfm.

Table 58-1 provides the comparison of the results of the analysis for power levels of 2436 MWt and 2763 MWt.

TABLE 58-1

UNIT 1 CAD SYSTEM RESULTS

Power Level	Results of the Analysis
2436 MWt	• Oxygen concentration reaches 5% by volume in 2.5 days by taking credit for dilution of oxygen from either the hydrogen generated or the steam in the containment. • Oxygen concentration reaches 5% by volume in 14 hours for the bounding case analysis by taking no credit for dilution of oxygen from either the hydrogen generated or the steam in the containment.
2763 MWt	• Oxygen concentration reaches 5% by volume in 3.5 days by taking credit for dilution of oxygen from either the hydrogen generated or the steam in the containment (Note). • Oxygen concentration reaches 5% by volume in 9 hours for the bounding case analysis by taking no credit for dilution of oxygen from either the hydrogen generated or the steam in the containment).

NOTE: The difference in time of 3.5 days at power level of 2763 MWt as compared to 2.5 days at a power level of 2436 MWt is mainly due to the dilution effect of the greater metal-water reaction hydrogen generation, based on the latest fuel data. Also, the results of the analysis indicate that an oxygen concentration of 5% by volume is reached earlier in torus. Hence, the time reported is for the torus.

Unit 2:

The combustible gas control analysis for Unit 2 extended power uprate is conservatively based on limiting the hydrogen concentration inside the containment to 3.5 vol % by use of hydrogen recombiners and a metal-water reaction rate of 1% by weight. The analysis assumes a non-inerted containment and a 90 minute heatup time for the hydrogen recombiner. Considering the 90 minute heatup time, the hydrogen recombiners are required to be operating 1 hour and 50 minutes to limit the hydrogen concentration to 3.5%. The hydrogen concentration is expected to be 3.15% at the time the hydrogen recombiners are started.

Initiation of the hydrogen recombiners is controlled procedurally based on the hydrogen concentration inside the containment and not by an analytically determined time. The actual time available for initiation of the hydrogen recombiners is expected to be greater than the time determined by the analysis due to conservatism in the analysis.

The results of the analysis for extended power uprate with the use of the recombiners are reflected in Figures 58-2 and 58-3. Table 58-2 provides a comparison of parameters used for the existing and the revised analysis. Table 58-3 provides the comparison of the results of the existing analysis at a power level of 2537 MWt (4% added for conservatism) and revised analysis for power levels of 2436, 2558, and 2763 MWt by use of identical assumptions and input data.

TABLE 58-2

Parameter	Existing Analysis	Revised Analysis for Extended Power Uprate
Fuel Array	8x8	9x9 (latest fuel type)
Hydrogen Evolved due to Metal Water Reaction (lb-mol)	12.13	14.88
Initial Conditions:		
Drywell Temperature (°F)	135	150
Torus Temperature (°F)	95	100
Drywell Pressure (psig)	0.75	1.75
Torus Pressure (psig)	0.75 (0.0 psig used by the analysis)	1.75

TABLE 58-3

Power Level	Time at Which Hydrogen Recombiners are Required to be Started	Hydrogen Concentration When Hydrogen Recombiners are Started
Existing analysis at a power level of 2537 MWt	6 hours	3.3 % vol
Revised analysis at a power level of		
2436 MWt	3 hours and 45 minutes (Note)	3.33 % vol
2558 MWt	3 hours (Note)	3.27 % vol
2763 MWt	1 hour and 50 minutes (Note)	3.15 % vol

NOTE: Time indicated is for initiation of the hydrogen recombiners and accounts for the 90 minutes heatup time. Actual recombination starts 90 minutes after the initiation of the recombiners.

FIGURE 58-1

**O2 CONCENTRATION VS TIME
(WITH CAD)**

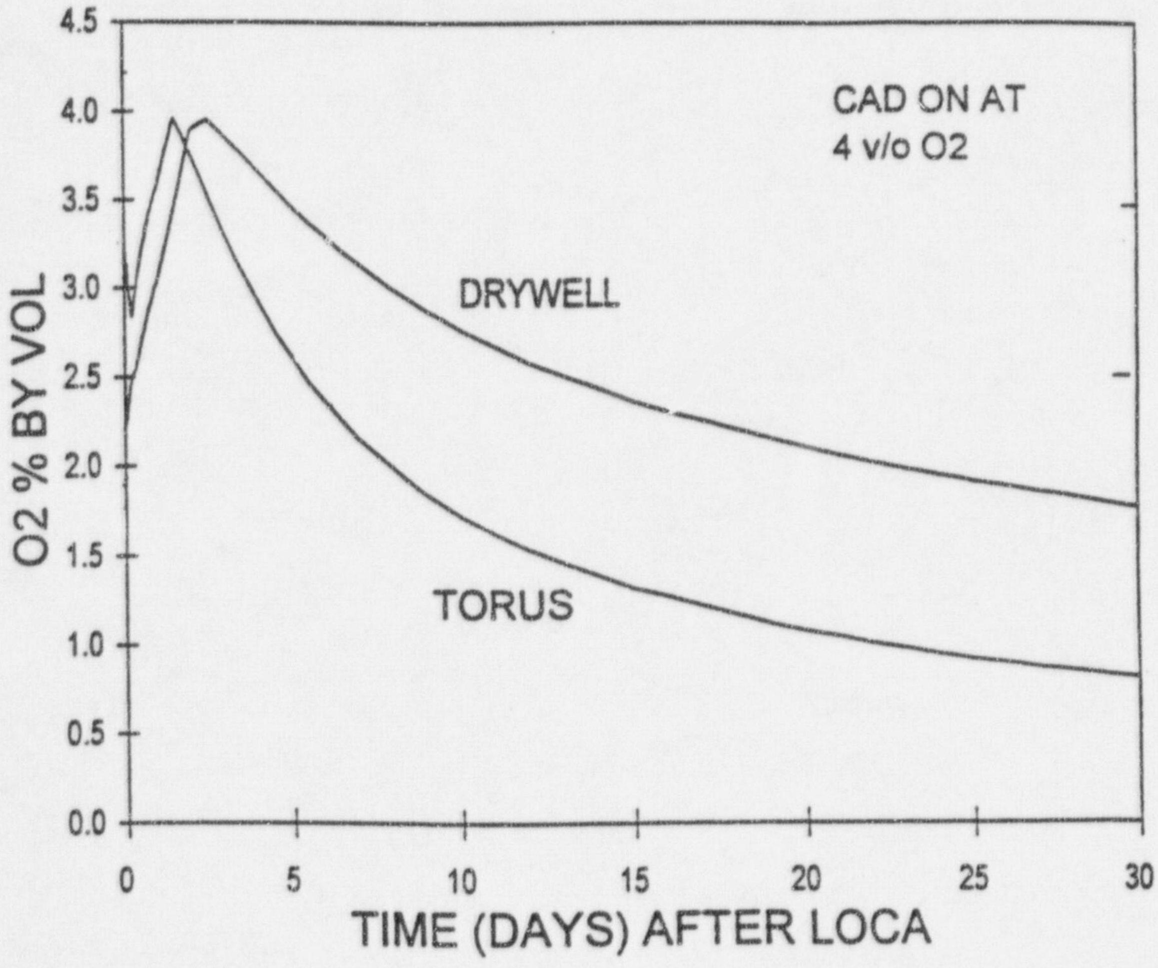


FIGURE 58-2

**H2 CONCENTRATION VS TIME - SHORT TERM
(WITH RECOMBINATION)**

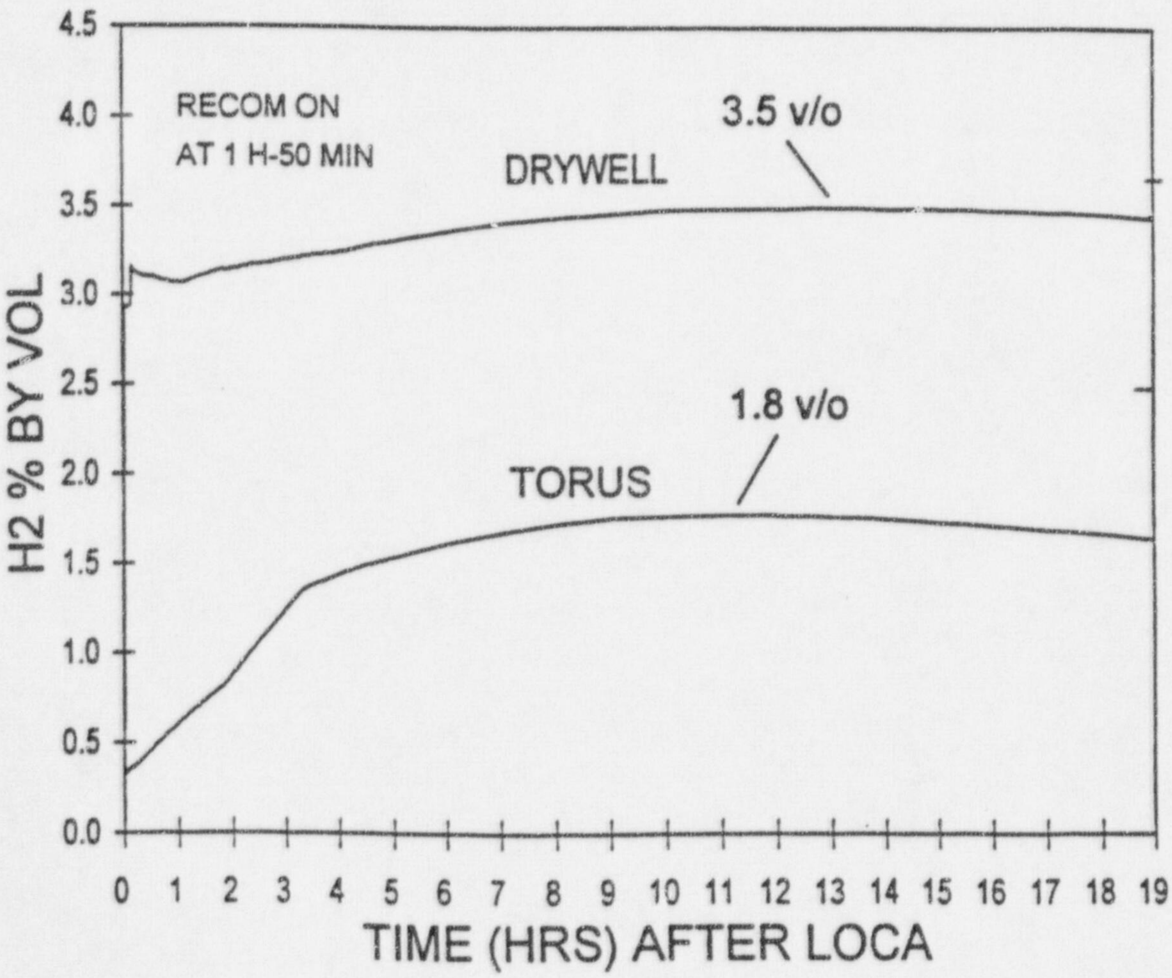
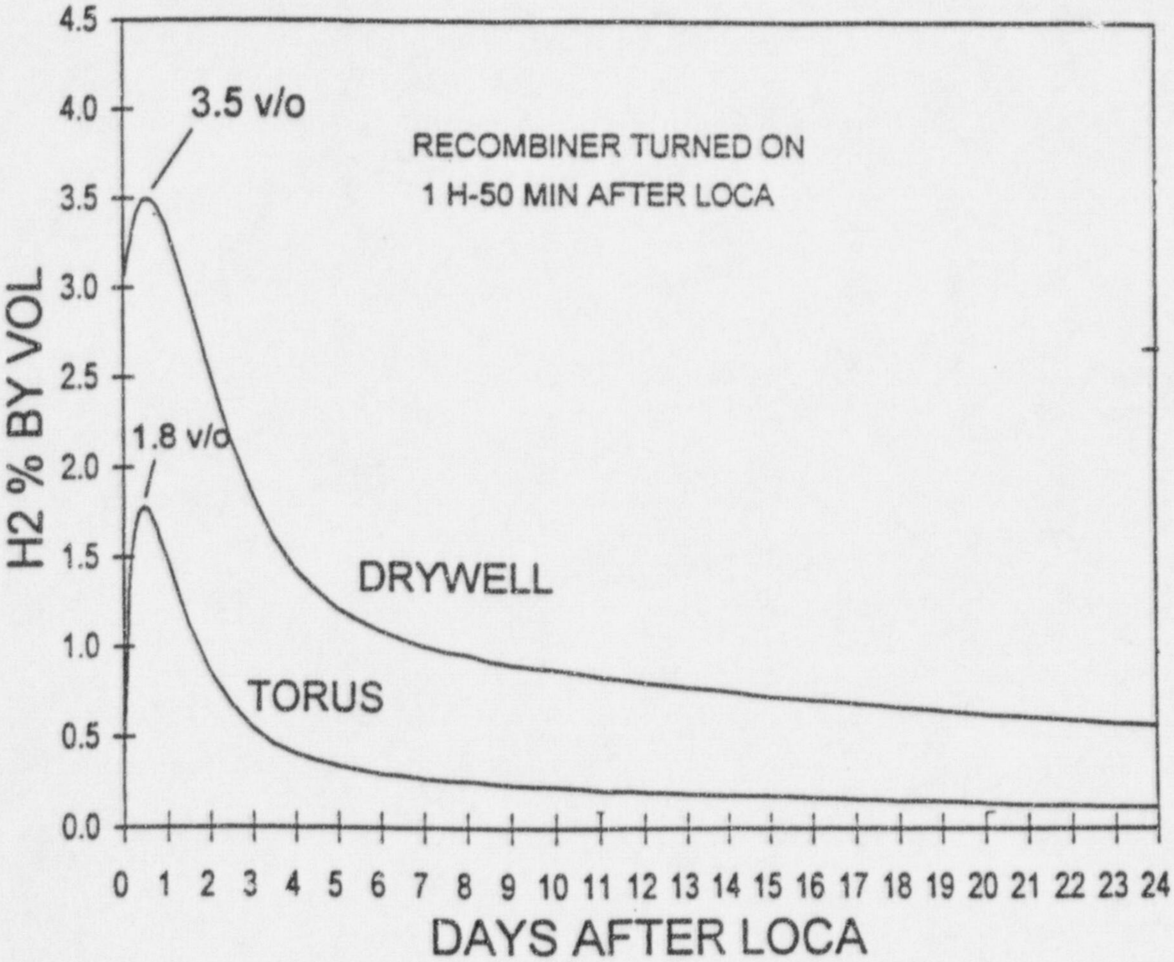


FIGURE 58-3

**H2 CONCENTRATION VS TIME - LONG TERM
(WITH RECOMBINATION)**



NRC QUESTION 59

For review of available containment pressure for the net positive suction head (NPSH), please provide the key assumptions used for the minimum containment pressure analyses. Also provide the updated containment analyses pressure and temperature curves.

SNC RESPONSE

The response to NRC Question 59 will be provided under separate cover letter.

NRC QUESTION 60

For review of used overpressure for Unit 1, please provide the NPSH calculations for residual heat removal and containment [core] spray pumps. The results are tabulated on Page E-5 of your 90-day response to Generic Letter 97-04, dated December 39, 1997.

SNC RESPONSE

The response to NRC Question 60 will be provided under separate cover letter.

PROBABILISTIC SAFETY ASSESSMENT:

NRC QUESTION 61

On page 10-7 (middle of the page), it is stated that "this analysis focused primarily on the evaluation of the considerations: initiating events, level 1 success criteria, and etc...". The GE long term program cites four types of probabilistic safety assessment (PSA) inputs and assumptions that may be affected by the uprate -- which, in addition to the ones on the list, includes a consideration of component failure rates also. A discussion on how component failure rates may be changed by power uprate should be provided. If they are not expected to change, a basis should be provided.

SNC RESPONSE

The component failure rate will not change significantly with extended power uprate because the component monitoring programs that are in place (i.e., environmental qualification, erosion/corrosion, etc.) will be modified to account for the wear as a result of extended power uprate. Extended power uprate may result in components being refurbished or replaced at more frequent intervals; however, the functionality and reliability of components will be maintained to the current standard.

NRC QUESTION 62

On page 10-8, the top paragraph states that "it could be postulated that an increase in certain initiator event frequencies may occur over time due to the operation of the plant with decreased power margins as compared to those which existed at the initial licensed power level." Thus, rather than concluding, without further examination, that the extended power uprate would have no readily discernible adverse effect on initiator frequency, can it be shown through a sensitivity analysis that the impact of a presumed changed in the initiator event frequency may or may not result in a noticeable change in risk?

SNC RESPONSE

A sensitivity study could be performed, but the results, based on input from the engineering analysis would not produce a change in risk significance for any specific initiator that would differ from the present values. PRA initiator frequencies, for the most part, are based on actual plant events. These data are not analyzed or examined any further than to classify them under the proper heading (i.e., turbine trip, reactor scram, etc.). The large values associated with initiator frequencies tend to cause significant change in risk when modified for sensitivity studies, and therefore, tend to easily skew the results. Therefore, a sensitivity study for this issue was not performed.

The complete discussion on page 10-8 of the submittal in reference to initiator frequencies is not a conclusion made without further examination. Engineering evaluations were performed to determine the effects of extended power uprate on the systems and operation of Plant Hatch.

NRC QUESTION 63

On page 10-8, second to the last paragraph, last sentence stated that "success criteria were judged to remain valid." Does this mean that there is no change in the success criteria or that it may change, but still considered "valid"? If it is the latter, what is meant by "valid"?

SNC RESPONSE

The general Level I success criteria were re-evaluated at the postulated extended power uprate conditions. It was concluded that the summary information on Level I success criteria provided in the Plant Hatch individual plant examination (IPE) submittal in Section 3.1.2.1, Subsection B, pages 3.1-9 through 3.1-15, as well as information provide in Tables 3.1-3 through 3.1-6, remain true. There were no significant changes required to the criteria as a result of extended power uprate; therefore, the criteria remain acceptable, or valid, for extended power uprate operation.

NRC QUESTION 64

Please identify dominant sequences and their contribution to the core damage frequency (CDF) increase -- providing a discussion comparing the baseline and the new dominant sequences.

What percentage of the CDF increase (due to power uprate) is due to increase in operator error probabilities? Please provide a discussion on specific operator actions and their contributions to the increase in risk.

SNC RESPONSE

The only operator action affected by extended power uprate is identified as Item 1, Failure To Depressurize With Inadequate High Pressure Injection (non-ATWS), in section 10.5.2.4 of the extended power uprate submittal. Thus, Item 1 represents the bounding case of increased core damage with respect to the impact of operator action for Units 1 and 2. The following discussion is based on a bounding case change in the numerical value for Item 1 due to extended power uprate. This operator failure probability is twice the value used in the original IPE analysis. Please note that in the remainder of this response, Item 1 will be referred to by its probabilistic risk assessment (PRA) model identifier, DE4.

Table 64-1 and 64-2 present the PRA model initiator frequency contributions to CDF for Unit 1 pre-extended power uprate (Table 64-1) and post-extended power uprate (Table 64-2). The impact of DE4 does little to change the significance of initiator contributions to CDF. The most significant CDF increases result from the following three cases:

1. MLOCA (with relation to ATWSM1)
2. DISCH (with relation to LOCV1), and

3. LOSUTD (with relation to ATWS1) ATWST1

Table 64-9, a listing defining each initiator abbreviation, is provided for convenience.

The CDF changes result from the increase in value of DE4, which is most notable in the MLOCA sequences. An ATWS with MSIV closure (ATWSM1) is not affected by DE4 and stays essentially constant resulting in a higher relative frequency value for MLOCA. Case 2 above, related to the isolation of plant service water (DISCH) and a loss of condenser vacuum on Unit 1 (LOCV1) reflects the impact of low frequency sequences (in the range of 1E-8 to 1E-9) which include a DE4 component. In these cases, the increased value of DE4 contributes more to CDF for DISCH than that for LOCV1. An effect similar to that noted in Case 2 applies also to Case 3, the loss of startup transformer D (LOSUTD) compared to an ATWS with turbine trip (ATWST1). In this case, ATWST1 is not affected by DE4, and a higher relative frequency value for LOSUTD results.

Operator action DE4 is not the only operator depressurization action in the PRA models. It is only used for non-ATWS situations requiring depressurization and situations in which the operator would potentially have to correct reactor water level (due to elevated drywell temperatures or overheated main control room instrumentation). This is the reason DE4 impacts the LOMCHV and MLOCA cases. Tables 64-5 and 64-6 are provided to further describe the effects the change in DE4 has on the top 100 individual sequences. Table 64-5 includes the values prior to extended power uprate, and Table 64-6 includes the extended power uprate values for Unit 1.

Before and after extended power uprate, the top five sequences consist of the same set of initiating events. The relative positions of sequences 6 through 12 are shifted due to the effect DE4 has on LOMCHV and MLOCA. The remainder of the differences in relative position noted in the two tables are due to the movement of those sequences which include the DE4 component with its increased value. With the exception of LOMCHV and MLOCA, the increase in the CDF of each sequence tends to be small. Further, as noted in Table 64-5 and 64-6, there are no new sequences created as a result of the increase in value of DE4.

Table 64-3 and 64-4 present the PRA model initiator frequency contributions to CDF for Unit 2 pre-extended power uprate (Table 64-3) and post-extended power uprate (Table 64-4). The results are similar to those previously discussed for Unit 1, the most notable of which are the increase in LOMCHV and MLOCA frequency contributions. These increases are again attributed to the significant contribution made by the increase in value of DE4. Other minor changes are the result of low frequency sequences (in the range of 1E-8 to IE-9) which include a DE4 component.

Table 64-7 and 64-8 are provided to further describe the effects of the change to DE4 on the top 100 individual sequences. Table 64-7 includes the values prior to extended power uprate, and Table 64-8 includes the extended power uprate values for Unit 2. In a manner similar to Unit 1, the top five sequences do not change order. The remaining sequences shift order due to the small effects of their DE4 components. No new sequences result from the extended power uprate.

In conclusion, for each unit, the change in CDF attributable to the proposed extended power uprate is due to the increase in the value of the operator error probability for the DE4 depressurization action. The percentage of this change in CDF is reported in Section 10.5.4 of the extended power uprate submittal.

TABLE 64-1

UNIT 1
INITIATOR CONTRIBUTIONS TO END STATE GROUP (MELT)
(PRE-EXTEND POWER UPRATE)

Initiator	Frequency	Initiator	Frequency	Initiator	Frequency
LOSP	5.2548E-06	INTAKE	2.1749E-07	ULRWCU	1.7906E-08
BUSC	2.9181E-06	MLOCA	2.0152E-07	ULHPCI	1.7747E-08
LODC	2.3897E-06	ALOCA	1.6585E-07	FLOOD4	1.4747E-08
LOFW1	2.1675E-06	LOBUSF	1.0997E-07	ULFWA	1.2125E-08
TTRIP1	1.3902E-06	ATWSF1	1.0395E-07	ULFWB	1.1812E-08
MSIVC1	1.1071E-06	VSEQ	8.6200E-08	SLOCA	1.1440E-08
DCPAN	1.1058E-06	LOSPDC	7.1750E-08	LOBUSG	9.1019E-09
LOPSW	1.0716E-06	ATWST1	4.9659E-08	LOSPML	8.8707E-09
SCRAM1	9.5451E-07	LOSUTD	4.8918E-08	LODWC1	4.0086E-09
LOMCHV	5.6473E-07	LOBUSE	4.3231E-08	ULMSL	2.8293E-09
IORV	4.9969E-07	LOSPPS	3.9344E-08	LOSPVM	1.5439E-09
BUSD	4.5159E-07	LOSPAC	3.6490E-08	LOSPSL	3.6687E-10
ATWSM1	2.4809E-07	LLOCA	2.4378E-08	LOSPLL	2.8534E-11
LOCV1	2.3538E-07	ULRCIC	1.8795E-08		
DISCH	2.2927E-07	FLD24	1.8518E-08		

TABLE 64-2

UNIT 1
INITIATOR CONTRIBUTIONS TO END STATE GROUP (MELT)
(POST-EXTENDED POWER UPRATE)

Initiator	Frequency	Initiator	Frequency	Initiator	Frequency
LOSP	5.3268E-06	INTAKE	2.4894E-07	ULRWCU	1.7906E-08
BUSC	2.9260E-06	ATWSM1	2.4711E-07	ULHPCI	1.7747E-08
LODC	2.3991E-06	ALOCA	1.6585E-07	FLOOD4	1.4747E-08
LOFW1	2.2833E-06	LOBUSF	1.1843E-07	ULFWA	1.2125E-08
TTRIP1	1.6073E-06	ATWSF1	1.0363E-07	ULFWB	1.1812E-08
DCPAN	1.2728E-06	VSEQ	8.6200E-08	SLOCA	1.1268E-08
MSIVC1	1.2571E-06	LOSPDC	7.2035E-08	LOBUSG	9.7045E-09
LOPSW	1.1100E-06	LOSUTD	5.2917E-08	LOSPML	9.5994E-09
SCRAM1	1.0228E-06	ATWST1	4.9508E-08	LODWC1	4.3109E-09
LOMCHV	8.8516E-07	LOBUSE	4.6837E-08	ULMSL	2.8293E-09
IORV	6.6738E-07	LOSPPS	3.9435E-08	LOSPVM	1.6633E-09
BUSD	4.5211E-07	LOSPAC	3.6895E-08	LOSPSL	3.6661E-10
MLOCA	4.0056E-07	LLOCA	2.4378E-08	LOSPLL	2.8534E-11
DISCH	2.7652E-07	ULRCIC	1.8795E-08		
LOCV1	2.6447E-07	FLD24	1.8518E-08		

TABLE 64-3

UNIT 2
INITIATOR CONTRIBUTIONS TO END STATE GROUP (MELT)
(PRE-EXTENDED POWER UPRATE)

Initiator	Frequency	Initiator	Frequency	Initiator	Frequency
LOSP	6.2090E-06	MLOCA	1.8801E-07	FLD22	1.8262E-08
LOFW2	2.9873E-06	ALOCA	1.5626E-07	FLD24	1.8262E-08
BUSC	2.8739E-06	ATWSF2	1.4965E-07	ULRWCU	1.8244E-08
LODC	2.3612E-06	LOCV2	1.2053E-07	ULHPCI	1.7522E-08
MSIVC2	1.4682E-06	LOBUSF	9.8539E-08	LOSPML	1.3285E-08
TTRIP2	1.1158E-06	LOSPDC	9.6315E-08	ULFWB	1.2515E-08
LOPSW	1.0582E-06	VSEQ	8.6200E-08	ULFWA	1.2025E-08
DCPAN	1.0144E-06	ATWST2	4.7210E-08	SLOCA	1.1248E-08
SCRAM2	7.3439E-07	LOSPAC	3.7370E-08	LOBUSG	7.7777E-09
LOMCHV	5.2938E-07	LOBUSE	3.7319E-08	ULMSL	3.3670E-09
IORV	4.7383E-07	LOSUTD	3.4358E-08	LOSPVM	1.3057E-09
ATWSM2	3.1196E-07	LODWC2	3.3994E-08	LOSPSL	3.5926E-10
DISCH	2.7619E-07	LLOCA	2.3044E-08	LOSPLL	2.6291E-11
BUSD	2.5446E-07	ULRCIC	1.8795E-08		
INTAKE	2.1376E-07	LOSPPS	1.8677E-08		

TABLE 64-4

**UNIT 2
INITIATOR CONTRIBUTIONS TO END STATE GROUP (MELT)
(POST-EXTENDED POWER UPRATE)**

Initiator	Frequency	Initiator	Frequency	Initiator	Frequency
LOSP	6.3127E-06	INTAKE	2.4652E-07	FLD24	1.8262E-08
LOFW2	3.1465E-06	ALOCA	1.5626E-07	FLD22	1.8262E-08
BUSC	2.8674E-06	ATWSF2	1.4933E-07	ULRWCU	1.8244E-08
LODC	2.3749E-06	LOCV2	1.3095E-07	LOSPML	1.7681E-08
MSIVC2	1.6316E-06	LOBUSF	9.7218E-08	ULHPCI	1.7522E-08
TTRIP2	1.2435E-06	LOSPDC	9.6141E-08	ULFWB	1.2515E-08
DCPAN	1.1340E-06	VSEQ	8.6200E-08	ULFWA	1.2025E-08
LOPSW	1.0982E-06	ATWST2	4.7111E-08	SLOCA	1.0676E-08
LOMCHV	8.5606E-07	LOSUTD	3.8164E-08	LOBUSG	7.7196E-09
SCRAM2	7.3437E-07	LOSPAC	3.7597E-08	ULMSL	3.3670E-09
IORV	6.3601E-07	LOBUSE	3.7346E-08	LOSPVM	1.3830E-09
MLOCA	3.8989E-07	LODWC2	3.3985E-08	LOSPSL	3.5843E-10
DISCH	3.3885E-07	LLOCA	2.3044E-08	LOSPLL	2.6291E-11
ATWSM2	3.1111E-07	ULRCIC	1.8795E-08		
BUSD	2.5414E-07	LOSPPS	1.8697E-08		

TABLE 64-5

UNIT 1
TOP-RANKING SEQUENCES CONTRIBUTING TO GROUP (MELT FREQUENCY) CORE DAMAGE SEQUENCES (PRE-EXTENDED POWER UPRATE)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
1	Transient	U1 Loss of Feedwater	IA015	9.56E-07	4.58
2	Special IE	Loss of Station Battery A Bus	ID006	8.25E-07	3.95
3	Special IE	Loss of 600V Bus C	IA023	7.95E-07	3.81
4	LOSP	Loss of Offsite Power	IB023	6.25E-07	2.99
5	Special Initiator	Loss of PSW Supply Hardware	II0125	5.12E-07	2.45
6	Special IE	Loss of Station Battery A Bus	IA015	4.22E-07	2.02
7	Special Initiator	Loss of MCR Cooling	IA015	2.48E-07	1.19
8	Special IE	Loss of Station Battery A Bus	IA016	2.34E-07	1.12
9	Transient	U1 Loss of Feedwater	IA015	2.28E-07	1.09
10	Special IE	Loss of 600V Bus C	IA016	2.12E-07	1.01
11	LOSP	Loss of Offsite Power	IB015	1.75E-07	.84
12	LOCA	Medium Break Inside the Drywell	IIIB15	1.61E-07	.77
13	Special IE	PSW Disch Valve F303A Transfers Closed	IA015	1.57E-07	.75
14	Special IE	Loss of DC Panel R25-S001	ID006	1.53E-07	.73
15	LOSP	Loss of Offsite Power	IA015	1.52E-07	.73
16	Special IE	Loss of 600V Bus C	ID006	1.49E-07	.71
17	Special IE	Loss of 600V Bus D	IA023	1.36E-07	.65
18	Special Initiator	Loss of PSW Supply Hardware	II0133	1.36E-07	.65
19	Special IE	Loss of Station Battery A Bus	II0121	1.32E-07	.63
20	LOSP	Loss of Offsite Power	II0125	1.30E-07	.62
21	Special IE	Loss of 600V Bus C	II0125	1.24E-07	.60
22	LOSP	Loss of Offsite Power	IB023	1.19E-07	.57
23	Special IE	Loss of DC Panel R25-S001	IA015	1.19E-07	.57
24	Transient	U1 MSIV Closure	IA015	1.17E-07	.56
25	Transient	U1 Loss of Feedwater	IIIB15	1.16E-07	.56
26	Special IE	Loss of 600V Bus C	II0121	1.16E-07	.56
27	Special IE	Loss of 600V Bus C	II0125	1.13E-07	.54
28	LOSP	Loss of Offsite Power	II0125	1.13E-07	.54
29	LOCA	Inadvertently Opened SRV	IIIB15	1.13E-07	.54
30	Transient	U1 Loss of Feedwater	IA015	1.07E-07	.51
31	Transient	U1 Turbine Trip	IA015	1.07E-07	.51
32	Special Initiator	Loss of PSW Supply Hardware	ID014	1.06E-07	.51
33	Special IE	Loss of 600V Bus C	IA015	1.02E-07	.49
34	LOSP	Loss of Offsite Power	IB023	1.01E-07	.48
35	LOSP	Loss of Offsite Power	IB023	9.85E-08	.47
36	Special Initiator	Loss of PSW Supply Hardware	IA016	9.60E-08	.46

TABLE 64-5 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
37	LOSP	Loss of Offsite Power	IA015	9.42E-08	.45
38	Transient	U1 Reactor Scram	IA015	9.11E-08	.44
39	LOSP	Loss of Offsite Power	IB023	8.96E-08	.43
40	LOCA	Interfacing Systems	V	8.62E-08	.41
41	LOSP	Loss of Offsite Power	IB023	8.51E-08	.41
42	Transient	U1 Turbine Trip	IA015	8.48E-08	.41
43	Special IE	Loss of DC Panel R25-S001	IIIB15	8.22E-08	.39
44	Special IE	Loss of DC Panel R25-S001	IA015	7.81E-08	.37
45	Transient	U1 MSIV Closure	IA015	7.68E-08	.37
46	Special IE	Loss of Station Battery A Bus	IA016	7.50E-08	.36
47	Special IE	Intake Structure Plugging	IA015	7.27E-08	.35
48	Transient	U1 Reactor Scram	IA015	7.22E-08	.35
49	LOSP	Loss of Offsite Power	IB023	6.87E-08	.33
50	LOCA	Spur. Elect. SRV Actuation/ Blowdown at PWR	IIIC06	6.77E-08	.32
51	LOSP	Loss of Offsite Power	IB023	6.69E-08	.32
52	LOSP	Loss of Offsite Power	IB006	6.66E-08	.32
53	LOCA	Spur. Elect. SRV Actuation/ Blowdown at PWR	IIIC06	6.61E-08	.32
54	Transient	U1 Turbine Trip	IA015	6.43E-08	.31
55	Special Initiator	Loss of MCR Cooling	ID006	6.32E-08	.30
56	LOCA	Inadvertently Opened SRV	IIIB15	6.18E-08	.30
57	LOSP	Loss of Offsite Power	II0121	6.17E-08	.30
58	Special IE	Loss of Station Battery A Bus	IA015	6.04E-08	.29
59	LOSP	Loss of Offsite Power	II0125	5.95E-08	.28
60	LOSP	Loss of Offsite Power	II0121	5.88E-08	.28
61	LOSP	Loss of Offsite Power	IA015	5.87E-08	.28
62	Special IE	Loss of 600V Bus D	ID006	5.86E-08	.28
63	Special IE	Loss of 600V Bus C	IA023	5.63E-08	.27
64	Transient	U1 Reactor Scram	IA015	5.48E-08	.26
65	Special IE	Loss of Station Battery A Bus	II0121	5.12E-08	.25
66	Special IE	Loss of DC Panel R25-S001	IA016	5.02E-08	.24
67	Special IE	Loss of DC Panel R25-S001	IA015	5.02E-08	.24
68	Transient	U1 Turbine Trip	II0125	4.99E-08	.24
69	Transient	U1 MSIV Closure	IA015	4.96E-08	.24
70	ATWS	U1 MSIV Closure/LOCV	IV0121	4.88E-08	.23
71	Special IE	Loss of 600V Bus C	II0125	4.86E-08	.23
72	Special IE	Loss of Station Battery A Bus	IA015	4.74E-08	.23
73	LOSP	Loss of Offsite Power	IB023	4.65E-08	.22
74	Transient	U1 Turbine Trip	II0121	4.63E-08	.22
75	LOSP	Loss of Offsite Power	IB023	4.59E-08	.22
76	Special IE	Loss of 600V Bus C	ID006	4.53E-08	.22
77	Special IE	Loss of 600V Bus C	II0121	4.53E-08	.22
78	Transient	U1 Turbine Trip	II0125	4.51E-08	.22

TABLE 64-5 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
79	LOSP	Loss of Offsite Power	IB023	4.48E-08	.21
80	Transient	U1 Turbine Trip	IIIB15	4.48E-08	.21
81	Transient	U1 Turbine Trip	IIIB15	4.45E-08	.21
82	Special IE	Loss of 600V Bus C	II0125	4.42E-08	.21
83	Special IE	Loss of 600V Bus D	IA016	4.29E-08	.21
84	Transient	U1 Reactor Scram	II0125	4.25E-08	.20
85	Special IE	Intake Structure Plugging	II0125	4.15E-08	.20
86	LOSP	Loss of Offsite Power	II0125	4.14E-08	.20
87	Special IE	Loss of Station Battery A Bus	ID006	4.08E-08	.20
88	Special IE	Loss of 600V Bus D	IA015	4.02E-08	.19
89	LOCA	Inadvertently Opened SRV	IIIB15	4.00E-08	.19
90	Transient	U1 Reactor Scram	II0121	3.94E-08	.19
91	Transient	U1 MSIV Closure	IA015	3.90E-08	.19
92	Transient	U1 Reactor Scram	II0125	3.84E-08	.18
93	Special IE	PSW Disch Valve F303A Transfers Closed	IIIB15	3.81E-08	.18
94	Special IE	Loss of DC Panel R25-S001	IIIB15	3.69E-08	.18
95	Transient	U1 Loss of Feedwater	IA015	3.68E-08	.18
96	Transient	U1 MSIV Closure	II0125	3.62E-08	.17
97	Special IE	Loss of 600V Bus C	II0125	3.57E-08	.17
98	Special IE	Loss of Station Battery A Bus	II0129	3.50E-08	.17
99	LOSP	Loss of Offsite Power	II0133	3.46E-08	.17
100	Transient	U1 MSIV Closure	II0121	3.36E-08	.16
101	LOSP	Loss of Offsite Power	IB015	3.34E-08	.16

TABLE 64-6

UNIT 1
TOP-RANKING SEQUENCES CONTRIBUTING TO GROUP (MELT FREQUENCY) CORE DAMAGE SEQUENCES (POST-EXTENDED POWER UPRATE)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
1	Transient	U1 Loss of Feedwater	IA015	9.52E-07	4.23
2	Special IE	Loss of Station Battery A Bus	ID006	8.25E-07	3.66
3	Special IE	Loss of 600V Bus C	IA023	7.95E-07	3.53
4	LOSP	Loss of Offsite Power	IB023	6.25E-07	2.78
5	Special Initiator	Loss of PSW Supply Hardware	II0125	5.12E-07	2.27
6	Special Initiator	Loss of MCR Cooling	IA015	4.97E-07	2.21
7	Special IE	Loss of Station Battery A Bus	IA015	4.20E-07	1.87
8	LOCA	Medium Break Inside the Drywell	IIIB15	3.22E-07	1.43
9	Special IE	Loss of Station Battery A Bus	IA016	2.34E-07	1.04
10	Transient	U1 Loss of Feedwater	IA015	2.31E-07	1.03
11	Special IE	Loss of 600V Bus C	IA016	2.12E-07	.94
12	LOSP	Loss of Offsite Power	IB015	1.75E-07	.78
13	Special IE	PSW Disch Valve F303A Transfers Closed	IA015	1.59E-07	.71
14	LOSP	Loss of Offsite Power	IA015	1.53E-07	.68
15	Special IE	Loss of DC Panel R25-S001	ID006	1.53E-07	.68
16	Special IE	Loss of 600V Bus C	ID006	1.49E-07	.66
17	Special IE	Loss of 600V Bus D	IA023	1.36E-07	.61
18	Special Initiator	Loss of PSW Supply Hardware	II0133	1.36E-07	.61
19	Special IE	Loss of Station Battery A Bus	II0121	1.32E-07	.58
20	LOSP	Loss of Offsite Power	II0125	1.30E-07	.58
21	Special IE	Loss of 600V Bus C	II0125	1.24E-07	.55
22	LOSP	Loss of Offsite Power	IB023	1.19E-07	.53
23	Transient	U1 Loss of Feedwater	IIIB15	1.19E-07	.53
24	Special IE	Loss of DC Panel R25-S001	IA015	1.18E-07	.53
25	Transient	U1 MSIV Closure	IA015	1.16E-07	.52
26	Special IE	Loss of 600V Bus C	II0121	1.16E-07	.52
27	LOCA	Inadvertently Opened SRV	IIIB15	1.16E-07	.51
28	Special IE	Loss of 600V Bus C	II0125	1.13E-07	.50
29	LOSP	Loss of Offsite Power	II0125	1.13E-07	.50
30	Transient	U1 Loss of Feedwater	IA015	1.09E-07	.48
31	Transient	U1 Turbine Trip	IA015	1.07E-07	.47
32	Special Initiator	Loss of PSW Supply Hardware	ID014	1.06E-07	.47
33	Special IE	Loss of 600V Bus C	IA015	1.03E-07	.46
34	LOSP	Loss of Offsite Power	IB023	1.01E-07	.45
35	LOSP	Loss of Offsite Power	IB023	9.85E-08	.44
36	Special Initiator	Loss of PSW Supply Hardware	IA016	9.75E-08	.43

TABLE 64-6 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
37	LOSP	Loss of Offsite Power	IA015	9.49E-08	.42
38	Transient	U1 Reactor Scram	IA015	9.07E-08	.40
39	LOSP	Loss of Offsite Power	IB023	8.96E-08	.40
40	Transient	U1 Turbine Trip	IIIB15	8.91E-08	.40
41	LOCA	Interfacing Systems	V	8.62E-08	.38
42	Transient	U1 Turbine Trip	IA015	8.54E-08	.38
43	LOSP	Loss of Offsite Power	IB023	8.51E-08	.38
44	Special IE	Loss of DC Panel R25-S001	IIIB15	8.41E-08	.37
45	Special IE	Loss of DC Panel R25-S001	IA015	7.78E-08	.35
46	Transient	U1 MSIV Closure	IA015	7.74E-08	.34
47	Special IE	PSW Disch Valve F303A Transfers Closed	IIIB15	7.63E-08	.34
48	Special IE	Loss of Station Battery A Bus	IA016	7.50E-08	.33
49	Special IE	Loss of DC Panel R25-S001	IIIB15	7.39E-08	.33
50	Special IE	Intake Structure Plugging	IA015	7.38E-08	.33
51	Transient	U1 Reactor Scram	IA015	7.27E-08	.32
52	LOSP	Loss of Offsite Power	IB023	6.87E-08	.31
53	LOCA	Spur. Elect. SRV Actuation/ Blowdown at Power	IIIC06	6.77E-08	.30
54	LOSP	Loss of Offsite Power	IB023	6.69E-08	.30
55	LOCA	Spur. Elec. SRV Actuation/Blowdown at Power	IIIC06	6.61E-08	.29
56	Transient	Unit 1 Turbine Trip	IA015	6.48E-08	.29
57	LOCA	Inadvertently Opened SRV	IIIB15	6.32E-08	.28
58	LOSP	Loss of Offsite Power	IB006	6.30E-08	.28
59	LOSP	Loss of Offsite Power	II0121	6.17E-08	.27
60	Special IE	Loss of Station Battery A Bus	IA015	6.13E-08	.27
61	Transient	U1 MSIV Closure	IIIB15	6.08E-08	.27
62	Special Initiator	Loss of MCR Cooling	ID006	5.98E-08	.27
63	LOSP	Loss of Offsite Power	II0125	5.95E-08	.26
64	LOSP	Loss of Offsite Power	IA015	5.91E-08	.26
65	LOSP	Loss of Offsite Power	II0121	5.88E-08	.26
66	Special IE	Loss of 600V Bus D	ID006	5.86E-08	.26
67	Special IE	Loss of 600V Bus C	IA023	5.63E-08	.25
68	Transient	U1 Reactor Scram	IA015	5.52E-08	.25
69	Transient	U1 Turbine Trip	IA015	5.15E-08	.23
70	Special IE	Loss of Station Battery A Bus	II0121	5.12E-08	.23
71	Transient	U1 Loss of Feedwater	IIIB15	5.08E-08	.23
72	Special IE	Loss of DC Panel R25-S001	IA016	5.02E-08	.22
73	Special IE	Loss of DC Panel R25-S001	IA015	5.00E-08	.22
74	Transient	U1 Turbine Trip	II0125	4.99E-08	.22
75	Special IE	Loss of DC Panel R25-S001	IIIB15	4.98E-08	.22
76	LOCA	Inadvertently Opened SRV	IIIB15	4.95E-08	.22
77	Transient	U1 MSIV Closure	IA015	4.94E-08	.22
78	ATWS	U1 MSIV Closure/LOCV	IV0121	4.88E-08	.22

TABLE 64-6 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
79	Special IE	Loss of 600V Bus C	II0125	4.86E-08	.22
80	Special IE	Loss of Station Battery A Bus	IA015	4.81E-08	.21
81	Transient	U1 Turbine Trip	IA015	4.76E-08	.21
82	Special Initiator	Loss of PSW Supply Hardware	IIIB16	4.67E-08	.21
83	LOSP	Loss of Offsite Power	IB023	4.65E-08	.21
84	Transient	U1 Turbine Trip	II0121	4.63E-08	.21
85	LOSP	Loss of Offsite Power	IB023	4.59E-08	.20
86	Transient	U1 Turbine Trip	IIIB15	4.58E-08	.20
87	Special IE	Loss of 600V Bus C	ID006	4.53E-08	.20
88	Special IE	Loss of 600V Bus C	II0121	4.53E-08	.20
89	Transient	U1 Turbine Trip	II0125	4.51E-08	.20
90	LOSP	Loss of Offsite Power	IB023	4.48E-08	.20
91	Special IE	Loss of 600V Bus C	II0125	4.42E-08	.20
92	Transient	U1 Reactor Scram	IA015	4.38E-08	.19
93	Special Initiator	Loss of MCR Cooling	IA015	4.32E-08	.19
94	Special IE	Loss of 600V Bus D	IA016	4.29E-08	.19
95	Transient	U1 Reactor Scram	II0125	4.25E-08	.19
96	Special IE	Intake Structure Plugging	II0125	4.15E-08	.18
97	LOSP	Loss of Offsite Power	II0125	4.14E-08	.18
98	Transient	U1 MSIV Closure	IIIB15	4.10E-08	.18
99	LOCA	Inadvertently Opened SRV	IIIB15	4.09E-08	.18
100	Special IE	Loss of Station Battery A Bus	ID006	4.08E-08	.18

TABLE 64-7

UNIT 2
TOP-RANKING SEQUENCES CONTRIBUTING TO GROUP (MELT FREQUENCY) CORE DAMAGE SEQUENCES (PRE-EXTENDED POWER UPRATE)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
1	Transient	U2 Loss of Feedwater	IA015	1.35E-06	5.90
2	Special IE	Loss of Station Battery A Bus	ID006	8.26E-07	3.61
3	Special IE	Loss of 600V Bus C	IA023	7.56E-07	3.30
4	LOSP	Loss of Offsite Power	IB023	6.29E-07	2.75
5	Special Initiator	Loss of PSW Supply Hardware	II0125	5.13E-07	2.24
6	Special IE	Loss of Station Battery A Bus	IA015	4.14E-07	1.81
7	Transient	U2 Loss of Feedwater	IA015	3.13E-07	1.37
8	Special Initiator	Loss of MCR Cooling	IA015	2.38E-07	1.04
9	Special IE	Loss of Station Battery A Bus	IA016	2.12E-07	.93
10	Special IE	Loss of 600V Bus C	IA016	1.97E-07	.86
11	Transient	U2 Loss of Feedwater	IIIB15	1.93E-07	.84
12	LOSP	Loss of Offsite Power	IB015	1.76E-07	.77
13	Transient	U2 MSIV Closure	IA015	1.68E-07	.73
14	Special IE	PSW Disch Valve F303A Transfers Closed	IA015	1.53E-07	.67
15	Special IE	Loss of DC Panel R25-S001	ID006	1.53E-07	.67
16	LOCA	Medium Break Inside the Drywell	IIIB15	1.51E-07	.66
17	Transient	U2 Loss of Feedwater	IA015	1.50E-07	.66
18	LOSP	Loss of Offsite Power	IA015	1.49E-07	.65
19	Special IE	Loss of 600V Bus C	ID006	1.39E-07	.61
20	Special Initiator	Loss of PSW Supply Hardware	II0133	1.37E-07	.60
21	Special IE	Loss of 600V Bus D	IA023	1.35E-07	.59
22	Special IE	Loss of Station Battery A Bus	II0121	1.32E-07	.58
23	LOSP	Loss of Offsite Power	II0125	1.24E-07	.54
24	LOSP	Loss of Offsite Power	IB023	1.20E-07	.52
25	Special IE	Loss of 600V Bus C	II0125	1.17E-07	.51
26	Special IE	Loss of DC Panel R25-S001	IA015	1.17E-07	.51
27	Transient	U2 MSIV Closure	IA015	1.14E-07	.50
28	LOCA	Inadvertently Opened SRV	IIIB15	1.12E-07	.49
29	Special IE	Loss of 600V Bus C	II0125	1.09E-07	.48
30	Special IE	Loss of 600V Bus C	II0125	1.06E-07	.46
31	LOSP	Loss of Offsite Power	IB023	1.02E-07	.44
32	Special IE	Loss of 600V Bus C	IA015	1.00E-07	.44
33	Transient	U2 Turbine Trip Event	IA015	1.00E-07	.44
34	LOSP	Loss of Offsite Power	IB023	9.91E-08	.43
35	Special Initiator	Loss of PSW Supply Hardware	ID014	9.78E-08	.43
36	Special Initiator	Loss of PSW Supply Hardware	IA016	9.57E-08	.42

TABLE 64-7 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
37	LOSP	Loss of Offsite Power	IB023	9.43E-08	.41
38	LOSP	Loss of Offsite Power	IA015	9.20E-08	.40
39	LOSP	Loss of Offsite Power	IB023	9.01E-08	.39
40	Transient	U2 Turbine Trip Event	IA015	8.74E-08	.38
41	LOCA	Interfacing Systems	V	8.62E-08	.38
42	LOSP	Loss of Offsite Power	IB023	8.56E-08	.37
43	Transient	U2 Reactor Scram	IA015	8.41E-08	.37
44	Special IE	Loss of DC Panel R25-S001	IIIB15	8.11E-08	.35
45	Special IE	Loss of DC Panel R25-S001	IA015	7.67E-08	.34
46	Transient	U2 Reactor Scram	IA015	7.33E-08	.32
47	Special IE	Loss of Station Battery A Bus	IA016	7.27E-08	.32
48	Special IE	Intake Structure Plugging	IA015	7.25E-08	.32
49	Transient	U2 MSIV Closure	IA015	7.07E-08	.31
50	LOSP	Loss of Offsite Power	IB023	7.03E-08	.31
51	LOSP	Loss of Offsite Power	IB006	6.86E-08	.30
52	LOCA	Spur Elect. SRV Actuation/Blowdown at PWR	IIIC06	6.79E-08	.30
53	Special Initiator	Loss of MCR Cooling	ID006	6.63E-08	.29
54	Transient	U2 Turbine Trip Event	IA015	6.49E-08	.28
55	Transient	U2 MSIV Closure	IA015	6.12E-08	.27
56	LOCA	Inadvertently Opened SRV	IIIB15	6.11E-08	.27
57	ATWS	U2 MSIV Closure and LOCV	IV0121	6.10E-08	.27
58	LOCA	Spur. Elect. SRV Actuation/Blowdown at PWR	IIIC06	5.83E-08	.25
59	Special IE	Loss of Station Battery A Bus	IA015	5.78E-08	.25
60	LOSP	Loss of Offsite Power	IA015	5.73E-08	.25
61	Transient	U2 Loss of Feedwater	IA015	5.71E-08	.25
62	Transient	U2 Reactor Scram	IA015	5.45E-08	.24
63	Special IE	Loss of 600V Bus C	IA023	5.35E-08	.23
64	Transient	U2 MSIV Closure	II0125	5.27E-08	.23
65	Special IE	Loss of Station Battery A Bus	II0121	5.12E-08	.22
66	Special IE	Loss of DC Panel R25-S001	IA015	4.93E-08	.22
67	Transient	U2 MSIV Closure	II0121	4.89E-08	.21
68	LOSP	Loss of Offsite Power	IA023	4.84E-08	.21
69	Transient	U2 MSIV Closure	II0125	4.77E-08	.21
70	Special IE	Loss of DC Panel R25-S001	IA016	4.76E-08	.21
71	LOSP	Loss of Offsite Power	IB023	4.68E-08	.20
72	Transient	U2 Turbine Trip Event	II0125	4.67E-08	.20
73	Special IE	Loss of Station Battery A Bus	IA015	4.62E-08	.20
74	Special IE	Loss of 600V Bus C	II0125	4.56E-08	.20
75	Transient	U2 MSIV Closure	IA015	4.55E-08	.20
76	LOSP	Loss of Offsite Power	IA015	4.52E-08	.20
77	LOSP	Loss of Offsite Power	IB023	4.44E-08	.19
78	Transient	U2 Turbine Trip Event	II0121	4.33E-08	.19

TABLE 64-7 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
79	Special IE	Loss of 600V Bus C	ID006	4.32E-08	.19
80	Special IE	Loss of 600V Bus C	II0121	4.25E-08	.19
81	Transient	U2 Turbine Trip Event	IIIB15	4.24E-08	.19
82	Transient	U2 Loss of Feedwater	IA015	4.24E-08	.19
83	Transient	U2 Turbine Trip Event	II0125	4.23E-08	.18
84	Transient	U2 MSIV Closure	IIIB15	4.16E-08	.18
85	Special IE	Intake Structure Plugging	II0125	4.15E-08	.18
86	Special IE	Loss of 600V Bus C	II0125	4.15E-08	.18
87	LOSP	Loss of Offsite Power	IA023	4.09E-08	.18
88	Special IE	Loss of Station Battery A Bus	ID0[illegible]	4.08E-08	.18
89	LOCA	Inadvertently Opened SRV	IIB15	3.99E-08	.17
90	Transient	U2 Loss of Feedwater	IIB15	3.96E-08	.17
91	Special IE	Loss of 600V Bus D	IA016	3.96E-08	.17
92	Transient	U2 Turbine Reactor Scram	II0125	3.92E-08	.17
93	Transient	U2 Turbine Trip Event	IIIB15	3.85E-08	.17
94	ATWS	U2 MSIV Closure and LOCV	IC015	3.73E-08	.16
95	Transient	U2 Reactor Scram	II0121	3.64E-08	.16
96	Special IE	PSW Disch Valve F303A Transfers Closed	IIIB15	3.55E-08	.16
97	Transient	U2 Reactor Scram	II0125	3.55E-08	.15
98	Special IE	Loss of DC Panel R25-S001	IIIB15	3.54E-08	.15
99	LOSP	Loss of Offsite Power	II0125	3.54E-08	.15
100	Special IE	Loss of Station Battery A Bus	II0129	3.50E-08	.15
101	ATWS	U2 MSIV Closure and LOCV	IV0121	3.38E-08	.15

TABLE 64-8

UNIT 2
TOP-RANKING SEQUENCES CONTRIBUTING TO GROUP (MELT FREQUENCY) CORE DAMAGE SEQUENCES (POST-EXTENDED POWER UPRATE)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
1	Transient	U2 Loss of Feedwater	IA015	1.37E-06	5.62
2	Special IE	Loss of Station Battery A Bus	ID006	8.26E-07	3.38
3	Special IE	Loss of 600V Bus C	IA023	7.56E-07	3.10
4	LOSP	Loss of Offsite Power	IB023	6.29E-07	2.58
5	Special Initiator	Loss of PSW Supply Hardware	II0125	5.13E-07	2.10
6	Special Initiator	Loss of MCR Cooling	IA015	4.98E-07	2.04
7	Special IE	Loss of Station Battery A Bus	IA015	4.21E-07	1.72
8	Transient	U2 Loss of Feedwater	IA015	3.20E-07	1.31
9	LOCA	Medium Break Inside the Drywell	IIIB15	3.14E-07	1.29
10	Special IE	Loss of Station Battery A Bus	IA016	2.12E-07	.87
11	Transient	U2 Loss of Feedwater	IIIB15	1.98E-07	.81
12	Special IE	Loss of 600V Bus C	IA016	1.97E-07	.81
13	LOSP	Loss of Offsite Power	IB015	1.76E-07	.72
14	Transient	U2 MSIV Closure	IA015	1.71E-07	.70
15	Special IE	PSW Disch Valve F303A Transfers Closed	IA015	1.56E-07	.64
16	Transient	U2 Loss of Feedwater	IA015	1.54E-07	.63
17	Special IE	Loss of DC Panel R25-S001	ID006	1.53E-07	.63
18	LOSP	Loss of Offsite Power	IA015	1.43E-07	.59
19	Special IE	Loss of 600V Bus C	ID006	1.39E-07	.57
20	Special Initiator	Loss of PSW Supply Hardware	II0133	1.37E-07	.56
21	Special IE	Loss of 600V Bus D	IA023	1.35E-07	.55
22	Special IE	Loss of Station Battery A Bus	II0121	1.32E-07	.54
23	LOSP	Loss of Offsite Power	II0125	1.24E-07	.51
24	LOSP	Loss of Offsite Power	IB023	1.20E-07	.49
25	Special IE	Loss of DC Panel R25-S001	IA015	1.18E-07	.49
26	Special IE	Loss of 600V Bus C	II0125	1.17E-07	.48
27	LOCA	Inadvertently Opened SRV	IIIB15	1.15E-07	.47
28	Transient	U2 MSIV Closure	IA015	1.10E-07	.45
29	Special IE	Loss of 600V Bus C	II0121	1.09E-07	.45
30	Special IE	Loss of 600V Bus C	II0125	1.06E-07	.43
31	Transient	U2 Turbine Trip Event	IA015	1.02E-07	.42
32	LOSP	Loss of Offsite Power	IB023	1.02E-07	.42
33	LOSP	Loss of Offsite Power	IB023	9.91E-08	.41
34	Special Initiator	Loss of PSW Supply Hardware	ID014	9.78E-08	.40
35	Special Initiator	Loss of PSW Supply Hardware	IA016	9.77E-08	.40

TABLE 64-8 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
36	Special IE	Loss of 600V Bus C	IA015	9.67E-08	.40
37	LOSP	Loss of Offsite Power	IB023	9.43E-08	.39
38	LOSP	Loss of Offsite Power	IB023	9.01E-08	.37
39	LOSP	Loss of Offsite Power	IA015	8.87E-08	.36
40	Transient	U2 MSIV Closure	IIIB15	8.70E-08	.36
41	LOCA	Interfacing Systems	V	8.62E-08	.35
42	LOSP	Loss of Offsite Power	IB023	8.56E-08	.35
43	Transient	U2 Reactor Scram	IA015	8.54E-08	.35
44	Transient	U2 Turbine Trip Event	IA015	8.42E-08	.35
45	Special IE	Loss of DC Panel R25-S001	IIIB15	8.33E-08	.34
46	Transient	U2 Loss of Feedwater	IIIB15	8.28E-08	.34
47	Transient	U2 Turbine Trip Event	IIIB15	8.04E-08	.33
48	Special IE	Loss of DC Panel R25-S001	IA015	7.79E-08	.32
49	Special IE	PSW Disch Valve F303A Transfers Closed	IIIB15	7.41E-08	.30
50	Special IE	Loss of DC Panel R25-S001	IIIB15	7.40E-08	.30
51	Special IE	Intake Structure Plugging	IA015	7.40E-08	.30
52	Special IE	Loss of Station Battery A Bus	IA016	7.27E-08	.30
53	Transient	U2 MSIV Closure	IA015	7.18E-08	.29
54	Transient	U2 Reactor Scram	IA015	7.07E-08	.29
55	LOSP	Loss of Offsite Power	IB023	7.03E-08	.29
56	LOSP	Loss of Offsite Power	IB006	6.86E-08	.28
57	LOCA	Spur. Elect. SRV Actuation/Blowdown at PWR	IIIC06	6.79E-08	.28
58	Transient	U2 Loss of Feedwater	IIIB15	6.37E-08	.26
59	LOCA	Inadvertently Opened SRV	IIIB15	6.28E-08	.26
60	Transient	U2 Turbine Trip Event	IA015	6.26E-08	.26
61	Special Initiator	Loss of MCR Cooling	ID006	6.26E-08	.26
62	ATWS	U2 MSIV Closure and LOCV	IV0121	6.10E-08	.25
63	Special IE	Loss of Station Battery A Bus	IA015	5.90E-08	.24
64	Transient	U2 MSIV Closure	IA015	5.90E-08	.24
65	LOCA	Spur. Elect. SRV Actuation/Blowdown at PWR	IIIC06	5.83E-08	.24
66	Transient	U2 MSIV Closure	IIIB15	5.63E-08	.23
67	LOSP	Loss of Offsite Power	IA015	5.52E-08	.23
68	Transient	U2 Loss of Feedwater	IA015	5.50E-08	.23
69	Special IE	Loss of 600V Bus C	IA023	5.35E-08	.22
70	Transient	U2 MSIV Closure	II0125	5.27E-08	.22
71	Transient	U2 Reactor Scram	IA015	5.25E-08	.22
72	Special IE	Loss of Station Battery A Bus	II012	5.12E-08	.21
73	Special IE	Loss of DC Panel R25-S001	IA015	5.01E-08	.21
74	Transient	U2 MSIV Closure	II0121	4.89E-08	.20
75	LOSP	Loss of Offsite Power	IA023	4.84E-08	.20
76	LOCA	Inadvertently Opened SRV	IIIB15	4.83E-08	.20
77	Special IE	Loss of DC Panel R25-S001	IIIB15	4.79E-08	.20

TABLE 64-8 (Continued)

Rank No.	Sequence Description	Events	End State	Frequency (per year)	Percent
78	Transient	U2 MSIV Closure	II0125	4.77E-08	.20
79	Special IE	Loss of DC Panel R25-S001	IA016	4.76E-08	.20
80	Special IE	Loss of Station Battery A Bus	IA015	4.71E-08	.19
81	LOSP	Loss of Offsite Power	IB023	4.68E-08	.19
82	Transient	U2 Turbine Trip Event	II0125	4.67E-08	.19
83	Special Initiator	Loss of PSW Supply Hardware	IIIB16	4.63E-08	.19
84	Special IE	Loss of 600V Bus C	II0125	4.56E-08	.19
85	LOSP	Loss of Offsite Power	IB023	4.44E-08	.18
86	Transient	U2 MSIV Closure	IA015	4.38E-08	.18
87	Transient	U2 Turbine Trip Event	IIIB15	4.36E-08	.18
88	LOSP	Loss of Offsite Power	IA015	4.35E-08	.18
89	Transient	U2 Turbine Trip Event	II0121	4.33E-08	.18
90	Special Initiator	Loss of MCR Cooling	IA015	4.33E-08	.18
91	Transient	U2 Loss of Feedwater	IIIB15	4.32E-08	.18
92	Special IE	Loss of 600V Bus C	ID006	4.32E-08	.18
93	Special IE	Loss of 600V Bus C	II0121	4.25E-08	.17
94	Transient	U2 Turbine Trip Event	II0125	4.23E-08	.17
95	Special IE	Intake Structure Plugging	II0125	4.15E-08	.17
96	Special IE	Loss of 600V Bus C	II0125	4.15E-08	.17
97	LOCA	Inadvertently Opened SRV	IIIB15	4.10E-08	.17
98	LOSP	Loss of Offsite Power	IA023	4.09E-08	.17
99	Transient	U2 Loss of Feedwater	IA015	4.09E-08	.17
100	Special IE	Loss of station Battery A Bus	ID006	4.08E-08	.17
101	Special IE	Loss of 600V Bus D	IA016	3.96E-08	.16

TABLE 64-9

INITIATING EVENT DESCRIPTION

Initiator	Description
LOSP	Loss of Site Power
BUSC	Loss of 600V AC Essential Bus C
LODC	Loss of Station Service Battery A
LOFW	Loss of Feed Water
TTRIP	Main Turbine Trip
DCPAN	Loss of DC Panel 1R25-S001
MSIVC	MSIV Closure
LOPSW	Loss of Plant Service Water
SCRAM	Scram
LOMCHV	Loss of Main Control Room Cooling
IORV	Inadvertently Opened Safety Relief Valve
BUSD	Loss of 600V AC Essential Bus D
MLOCA	Medium Break LOCA Inside the Drywell
DISCH	Plant Service Water Discharge Valve Closes
LOCV	Loss of Condenser Vacuum
INTAKE	Intake Structure Plugging
ATWSM	ATWS with MSIV Closure
ALOCA	Spurious Electronic Safety Relief Valve Actuation/Blowdown During Operation
LOBUSF	Loss of 4160V AC Emergency Bus F
ATWSF	ATWS with a Loss of Feedwater
VSEQ	Interfacing Systems LOCA
LOSPDC	Loss of Site Power During a Loss of DC Initiator
LOSUTD	Loss of Startup Transformer D
ATWST	ATWS with a Turbine Trip
LOBUSE	Loss of 4160V AC Emergency Bus E
LOSPPS	Loss of Site Power During a Loss of PSW Initiator
LOSPAC	Loss of Site Power During a Loss of AC Bus Initiator
LLOCA	Large Break LOCA
ULRCIC	RCIC Steam Line Break Outside Containment (Failure to Isolate)
FLD	Internal Flooding
ULRWCU	RWCU High Pressure Line Break Outside Containment (Failure to Isolate)
ULHPCI	HPCI Steam Line Break Outside Containment (Failure to Isolate)
FLOOD	Internal Flooding
ULFWA	Feedwater Line A Break Outside Containment (Failure to Isolate)
ULFWB	Feedwater Line B Break Outside Containment (Failure to Isolate)
SLOCA	Small Break LOCA Inside the Drywell
LOBUSG	Loss of 4160V AC Emergency Bus G
LOSPML	Loss of Site Power During a Medium LOCA or IORV
LODWC	Loss of Drywell Cooling
ULMSL	Main Steam Line Break Outside Containment (Failure to Isolate)
LOSPVM	Loss of Site Power During a Loss of MCR Air Conditioning Initiator
LOSPSL	Loss of Site Power During a Small LOCA
LOSPLL	Loss of Site Power During a Large LOCA

NRC QUESTION 65

On page 10-9, the first paragraph lists key operator actions. This list was formed from a review that was conducted to determine which accident scenarios involved changes in event timing that could significantly affect operator responses. What were the specific changes in the available response time for each of the operator actions on the list? And how were these changes in time reflected in the changed human error probabilities assumed for the operator actions on the list? Please list the newly assumed human error probabilities for each of these actions (list also the original base case probabilities). Along with a list of these human error probabilities, provide the bases for the newly assumed human error probabilities. Additionally, what were the bases for increasing the grid recovery probabilities by 0.02 and assuming a probability of 0.1032 for inadequate high pressure injection, which is twice the value employed in the original plant PSA?

SNC RESPONSE

The Hatch PRA models were extensively reviewed to determine the impact of the proposed extended power uprate. Section 10.5 of the extended power uprate submittal provides information directly related to changes in the PRA, PRA-related actions, core damage frequency (CDF), or containment analysis resulting from extended power uprate. Details of the study which did not result in revision of the PRA were not provided.

The following information provides the details of the evaluation for the probability values of Items 1 through 5 (located on page 10-9 of the extended power uprate submittal) which were modified. These probability values were selected to bound the associated changes in CDF and to identify their impact on the CDF. This approach determined whether or not the analysis and calculations associated with these actions required revision to derive a new probability number for extended uprate. The specific points of Question 65 are addressed following this discussion.

Temporary modifications to operator action and grid recovery failure probability values were made only in conjunction with a sensitivity study. The study was conducted to evaluate the integrated core damage effects from the five components of the PRA models affected by extended power uprate discussed above.

The event timing criteria was the only element associated with the five actions that changed as a result of extended power uprate. A sensitivity study was used to determine the relative worth of these events by allowing an option other than considering total failure of the event which would tend to skew the results. Once the relative worth in the form of percent CDF change of the actions was evaluated, the necessity for revising the various analyses used as the basis for the probability numbers quantified in the PRA models was determined. Probability values for Items 1-4 on page 10-9 of the extended uprate submittal were originally based on human error analysis using the Success Likelihood Index Methodology or (SLIM). Original Item 5 values are based on calculations which determine time to core damage and time to reactor vessel failure over a 24 hour mission time where failure to recover electrical power (grid recovery) is modeled for an LOSP.

Item 1 is a conventional depressurization operator action not used with ATWS while Items 2-4 are actions used only in ATWS sequences. Item 5 is not an actual operator action, but instead, a series of grid recovery failure probabilities associated with station blackout and based on diesel generator availability and grid recovery.

The ATWS CDF contribution to the Plant Hatch PRA is discussed in the IPE submittal for Plant Hatch and amounts to approximately 3% per unit.

It was shown that ATWS sequences with the higher probability values used for the sensitivity study contributed only a small amount to CDF increase over the original IPE numbers. This contribution is conservatively integrated with the modified values for Items 1 and 5 as well. The results of the initial sensitivity evaluation indicate Items 2-4 provide a relatively small increase in CDF worth when evaluated against changes to Items 1 and 5 and raised to arbitrarily large values. This would be the case until the probability numbers approached total failure or 1.0 which was not feasible because extended uprate modified the time, it did not totally fail the action by approaching zero time for accomplishment. Items 2-4 were screened from change based primarily on this information and a review of the MAAP analyses which were used to determine the timing changes.

The sensitivity study was repeated with probability numbers for Items 2-4 returned to their original IPE values. This allowed an evaluation of the importance of the grid recovery (GR) values, Item 5, in relation to changes in the probability value for Item 1. For this study, the grid recovery values in question were set to 1.0, or total failure, in order to isolate the effects of Item 1 in the CDF. The results of the study revealed that calculations for the grid recovery values of Item 5 needed to be reevaluated. This reevaluation showed that the small time changes being considered did not alter the probability values to a degree which would cause a change in CDF. Grid recovery values were therefore not modified for extended power uprate.

This left the change to Item 1 which had significant CDF contribution for a singular probability value. A SLIM analysis was performed to determine the conservatism of the probability value (twice the original) used in the sensitivity studies for Item 1. The result was less than the value used (0.0805 versus 0.1032).

The end result of the sensitivity studies was that Item 1, the action where the operator fails to depressurize with inadequate high pressure injection (non-ATWS), was the only item affected by extended power uprate and was the sole contributor to the CDF changes identified in Section 10.5.4 of the submittal. Noting that the previous discussion serves as an explanation of technique the following information will now address the specific parts of Question 65. Each of the five items on page 10-9 of the submittal will be addressed individually and in order.

Item 1: Failure to depressurize with inadequate high pressure injection (non-ATWS)

Original IPE response time: 2.5 minutes
Extended power uprate response time: 1.2 minutes
Original IPE failure probability: 0.0516
Extended power uprate failure probability: 0.1032

Based on the sensitivity analyses, a SLIM analysis was performed to calculate the effects of a shorter time to complete this action. Response time is one of the performance shaping factors for the SLIM analyses, but, in this particular case, it is not the most heavily weighted factor. The SLIM analysis calculated a probability value for this action of 0.0805. This value is less than the bounding value of 0.1032 selected for the sensitivity analysis (that is, twice the original IPE value of 0.0516). The percent CDF increases presented in Section 10.5.4 of the licensing submittal are based on the assumed value of 0.1032 and are considered conservative and bounding.

Item 2: Failure to depressurize with inadequate high pressure injection (ATWS)

Original IPE response time: 7 minutes (based on GE power/level control models)
Extended power uprate response time: 2 minutes (based on MAAP power/level control models)
Original SLIM-based IPE failure probability: 0.04245
Extended power uprate failure probability: 0.297

The MAAP analysis which provided the referenced response time employs a different power/level control model than that used in the Plant Hatch simulator. The Hatch PRA models use the GE power/level control information for their ATWS evaluations. Power/level control is discussed further in the Plant Hatch IPE submittal Volume I, page 3.1-13.

At the top of active fuel, MAAP indicates 22% to 30% reactor power, whereas the GE model used in the simulator indicates approximately 8% to 10% power. The increased calculated power used in MAAP will cause a faster level decrease which results in a shorter time to depressurization. Even considering this calculational difference, the time left for the operator action is two minutes. This is sufficient time for the required action considering the level of attention which would be focused on the problem. The majority of this time is used in securing low pressure injection in preparation for depressurization. This action is proceduralized and the operators receive regular simulator training on this scenario.

The changes in available time for operator action in this extended power uprate scenario were accounted for in the sensitivity analysis by picking an arbitrarily large value for the operator failure probability, in this case 0.297, a value equal to seven times the original value of 0.04245. Time is only one of several weighted performance shaping factors in the SLIM.

Seven times the original operator failure probability results in a value that is much greater than the largest obtainable from increasing the value for the time factor in the SLIM analysis to maximum. The 0.297 value was used to bound the effects that the decreased available operator response time could have on CDF. The contribution to CDF employing this assumption for this scenario in combination with the scenarios of Items 3 and 4 below, (both ATWS actions with the operator failure probability similarly set to the conservative value of seven times the original IPE values) only resulted in a 6% CDF increase on one unit and a 3% CDF increase on the other. The individual contribution to CDF from this individual scenario is lower.

The ATWS action described by Item 2 will not be significantly changed by the extended power uprate based on the following considerations.

1. The increases in CDF calculated in this sensitivity study are relatively small.

2. The majority of the decrease in the available time assumed before and after extended power uprate are the result of differences in the power/level control models.

Thus, the probability of failure for this operator action would not be significantly different than the original IPE value.

Item 3: Failure to initiate Standby Liquid Control with Turbine Bypass Valve capacity unavailable.

Originally assumed IPE response time: 6 minutes
Assumed extended power uprate response time: 1 minute
Original SLIM-based IPE failure probability: 0.00842
Extended power uprate failure probability: 0.058

The six minute response time assumed in the original IPE analysis was a conservative approximation. The original range of calculated required response time was one to three minutes.

In order to perform a sensitivity study, as discussed in Item 2 above, a response time of one minute was assumed to be available under extended power uprate conditions based on MAAP analysis. The one minute response window calculated by MAAP, with its conservative power/level control model, is judged to be sufficient time to start the standby liquid control (SBLC) system which is designed to be easily started under the conditions of the proposed scenario. The criteria for initiation of SBLC are the occurrence of an ATWS and the MSIVs closed. Operators are trained to immediately consider SBLC injection when presented this situation; therefore, the scenario has a low probability of operator failure. The changes in available time for operator action in this extended power uprate scenario were accounted for in the sensitivity analysis by picking an arbitrarily large value for the operator failure probability, in this case 0.058, a value equal to seven times the original value of 0.00842. As previously noted, time is only one of several weighted performance shaping factors in the SLIM. Seven times the original operator failure probability results in a value that is much greater than the largest obtainable from increasing the value for the time factor in the SLIM analysis to maximum. The 0.058 value was used to bound the effects that the decreased available operator response time could have on CDF.

Again, as was noted in Item 2 above, the contribution to CDF employing this assumption for this scenario in combination with the scenarios of Items 2 and 4, (both ATWS actions with the operator failure probability similarly set to the conservative value of seven times the original IPE values) only resulted in a 6% CDF increase on one unit and a 3% CDF increase on the other. The individual contribution to CDF from this individual scenario is lower.

The ATWS action described by Item 3 will not be significantly changed by the extended power uprate based on the following considerations:

1. The increases in CDF calculated in this sensitivity study are relatively small.

2. The majority of the decrease in the available time assumed before and after extended power uprate are the result of differences in the power/level control models.

Thus, the probability of failure for this operator action would not be significantly different than the original IPE value.

Item 4: Failure to control low pressure injection after depressurization (ATWS)

Original IPE response time: 5 minutes
Assumed extended power uprate response time: 1.2 minutes
Original SLIM-based IPE failure probability: 0.04767
Extended power uprate failure probability: 0.336

In a manner similar to that discussed for Items 2 and 3 above, Item 4 was re-evaluated and included in the sensitivity study previously described. The sensitivity study leads to the conclusion that the ATWS action described by Item 4 will not be significantly changed by the extended power uprate. Thus, the probability of failure for this operator action would not be significantly different than in the original IPE.

Item 5: Grid Recovery (GR) Probability for Station Blackout (SBO) without High Pressure Injection.

Original IPE response time: not applicable
Extended power uprate response time: not applicable
Original IPE failure probability: varied
Extended power uprate failure probability: varied

The GR factors considered were limited to those involved with SBO without injection. They were:

GRC: Failure to recover offsite power with two or three diesel generators failed - injection failed within a 24 hour period

GRD: Failure to recover offsite power with 1 diesel generator failed - injection failed within a 24 hour period

GRE: Failure to recover offsite power with two or three diesel generators failed - injection failed within a 3 hour period

GRF: Failure to recover offsite power with one diesel generator failed - injection failed within a 3 hour period

Grid recovery probabilities for SBO were initially based on MAAP calculations which provided the times to core damage and reactor vessel failure with and without vessel injection from RCIC. These calculations were re-examined for extended power uprate, and it was determined that for SBO without injection both the time to core damage and the time to reactor vessel failure decreased. The time to core damage decreased from 0.85 hours to 0.84 hours, while the time to vessel failure decreased from 2.1 hours to 1.83 hours. The 0.01 hour change in time to core damage was considered negligible, but the impact of the 0.27 hour change in time to vessel failure was evaluated further.

In order to evaluate the worth of these grid recovery values, two sensitivity studies were used. The first study is described in the discussions of Items 2 through 4 above. In that study, the five items were evaluated with revised probability values as previously described. This provided an integrated evaluation of the worth of the GR values. In this study, the affected GR probabilities were increased by 0.02. This number was derived from a conservative calculation used to initially evaluate the effects that extended power uprate would have on the GR numbers due to time changes for core melt and vessel failure.

Because this first study was essentially performed to screen ATWS values, Items 2 through 4 above, a second sensitivity study was performed varying only the inputs associated with Items 1 and 5. Again, the extended power uprate failure probability value of 0.1032 (twice the original value) in Item 1 was used, while the GR values of interest where set to assure failure, that is 1.0. This allowed the GR contributions to be readily distinguished from the dominate independent contribution to CDF of Item 1. The conclusions drawn from the second sensitivity study confirmed the need to reevaluate the detailed calculation for the affected GR values.

The results of re-examining the GR values calculation demonstrated two points. First, the 0.02 increase in the affected GR probabilities used in the sensitivity study did indeed bound the changes to the GR probability values which resulted from the time criteria changes. Secondly, the timing criteria changes did not require modifications to the GRE, GRF, or GRD probability values. The changes to the GRC values, as a result of extended power uprate, were small. Table 65-1 provides a comparison of the current and the extended power uprate grid recovery probabilities.

TABLE 65-1

CURRENT AND EXTENDED POWER UPRATE GRID RECOVERY PROBABILITIES

Grid Recovery Probability	Original IPE Value	Extended Power Uprate Values
GRC1	0.59	0.59
GRC2	0.04	0.03
GRC3	0.37	0.377

These changes were considered negligible since they resulted in no significant change in either CDF or large early release fraction (LERF). The original IPE grid recovery probabilities values were retained in the analysis of the proposed extended power uprate.

NRC QUESTION 66

On page 10-10, in the third paragraph, why was using the original PSA grid recovery probabilities considered more realistic when the list of significant operator actions (previous page) identified recovery of grid as an important operator action whose available response time may be shortened due to power uprate, thus resulting in a higher operator error probability? What is the basis for the change in the assumption?

SNC RESPONSE

Grid recovery probabilities are not actual operator actions. They were included in the section of the submittal which discusses operator actions as a matter of convenience in the organization of the submittal. While it is true that certain timing factors associated with the original grid recovery probability numbers changed as a result of extended power uprate conditions, the effect of these changes on the grid recovery probability values was negligible. Thus, use of the original PRA grid recovery probabilities was considered appropriate and acceptable for evaluating the impact of extended power uprate on the PRA.

The list of operator actions from page 10-9 of the submittal identified actions whose associated timing was shown to be changed by the Plant Hatch MAAP model. However, this identification was only the first step in the evaluation process. The subject grid recovery probabilities deal with Station Blackout (SBO) with a concurrent loss of vessel injection for both a 3 hour and a 24 hour duration. The time to core damage and the time to reactor vessel failure were determined from the MAAP evaluations, and in conjunction with other information, were used in a set of detailed calculations to derive the original and the extended power uprate grid recovery probability values.

The values for the original and the proposed extended power uprate grid recovery probabilities are:

Time to core damage original: 0.85 hours
Time to core damage extended power uprate: 0.84 hours
Time to vessel failure original: 2.1 hours
Time to vessel failure extended power uprate: 1.83 hours

Due to the small magnitude of the changes, the impact on the CDF was not considered significant, therefore a sensitivity study of the impact of changes in the grid recovery probabilities was conducted.

If the contribution of a bounding case probability value to CDF and LERF is small, then a smaller probability value would cause an even smaller change, and a judgment concerning its significance can be made. A value of 0.02 was added to the subject grid recovery probabilities to establish the initial bounding values. This 0.02 increase was chosen based upon a conservative evaluation of portions of the grid recovery calculations using the time changes resulting from the extended power uprate.

The first sensitivity study integrated the changes fo. grid recovery values with the bounding changes assigned to the other four items found on page 10-9 of the submittal. The results of this sensitivity study and associated evaluations were that Items 2 through 4 (ATWS actions) need not be modified for extended power uprate. However, the grid recovery values associated with the bounding value for Item 1, found on page 10-9 of the submittal, resulted in a significant CDF increase in this sensitivity study.

A second study was performed using Item 1, with a modified operator failure probability value, twice that used in the original IPE, and the grid recovery probabilities in question. The difference in this study and the first sensitivity study discussed above was that the operator failure probabilities for Items 2 through 4 (ATWS actions) were returned to their original IPE values. In order to determine the significance of the grid recovery probability values compared with the strong individual contribution of the operator failure probability of Item 1, the specific grid recovery values were set to 1.0 to reflect a total failure. The conclusion drawn from this study was that, even though the time changes in question were small, there was a need to re-evaluate the calculation of the grid recovery values in detail.

There are four grid recovery probabilities in the Plant Hatch PRA model which are involved with SBO without injection. They are:

GRC: Failure to recover offsite power with two or three diesel generators failed - injection failed within a 24 hour period

GRD: Failure to recover offsite power with 1 diesel generator failed - injection failed within a 24 hour period

GRE: Failure to recover offsite power with two or three diesel generators failed - injection failed within a 3 hour period

GRF: Failure to recover offsite power with 1 diesel generator failed - injection failed within a 3 hour period

The evaluation established that the 0.02 increase used to generate a bounding case for these values was appropriate. Further, the detailed evaluation of the calculation of the GRE, GRF, and GRD values determined that these values did not change from the original IPE values.

However, the values for GRC changed to a degree as can be seen in Table 66-1.

TABLE 66-1

GRID RECOVERY PROBABILITY

Grid Recovery Probability	Original IPE Value	Extended Power Uprate Value
GRC1	0.59	0.59
GRC2	0.04	0.03
GRC3	0.37	0.377

These changes were considered negligible since they resulted in no significant change in either CDF or LERF.

The bounding values assumed for sensitivity studies are conservative and are not intended to be the permanent values. The original IPE grid recovery probabilities values were retained in the analysis of the proposed extended power uprate.

NRC QUESTION 67

On page 10-9, middle paragraph, the second sentence states that operator error probabilities were chosen to bound the impact of the 50 percent decrease in event times for a couple of the actions listed in the previous paragraph. On page 10-10, middle paragraph, the third sentence states "in light of small magnitude of the changes in the available time...". Does Hatch consider the 50 percent reduction in response time availability small? If yes, what is the basis?

SNC RESPONSE

A 50% reduction in response time availability is not necessarily considered to be small within the Plant Hatch PRA. In the case of Items 3 and 4 on page 10-9 of the extended uprate submittal, the response times discussed are based on the results of MAAP analyses which use a conservative ATWS power/level control model. The original response times were based on calculations which used input data from the Plant Hatch training simulator, specifically, the power level at top of active fuel during an ATWS. This simulator power level is much lower than the corresponding MAAP analysis value. This difference causes a faster rate of reactor water level decrease due to the increased power level at top of active fuel and accounts for the variance in operator action response times.

The 50% reduction was an approximation derived by comparing the extended power uprate-based MAAP calculated times with the original simulator-based times. In order to compare the changes in time when calculated by the same methodology, MAAP calculations were run for before and after extended power uprate levels (2558 MWt and 2763 MWt respectively). When evaluated by similar methodologies, the change is insignificant for both Item 3 and 4 as shown below.

Item 3: Failure to Initiate Standby Liquid Control with Turbine Bypass Valve Capacity Unavailable (ATWS)

Original IPE time: 6 minutes (realistic estimation) (1-3 minutes calculated)
MAAP calculated original time: 0.947 minutes
MAAP calculated extended power uprate time: 0.843 minutes

Item 4: Failure To Control Low Pressure Injection After Depressurization (ATWS)

Original IPE time: 5 minutes (estimation)
MAAP calculated original time: 1.33 minutes
MAAP calculated extended power uprate time: 1.16 minutes

The implications of small changes in timing derived by the MAAP calculations would likewise apply to the submittals, calculations, and estimations. The difference in the time values results from the differences in the power level input at top of active fuel assumed by the two methodologies. It is not considered likely that a change in overall core thermal power from 2558 MWt to 2763 MWt will make a significant difference in the response times, as supported by the MAAP calculations.

NRC QUESTION 68

Given the results of Level 1 analysis (CDF increase was 6.6 percent for Unit 1 and 4.1 percent for Unit 2), what is the impact of these results on the containment? Please provide quantitative results for Level 2 analysis. As an example, increased operator error probability for vessel depressurization may affect mitigation of fire, resulting in an increased risk due to internal fire.

SNC RESPONSE

An approximate 1% change in CDF was noted for those sequences which were part of the LERF for each unit when evaluated for extended power uprate. This is considered a negligible change and is attributed to Item 1 on page 10-9 of the extended power uprate submittal. The revised probability value for Item 1 introduced no new sequences to be addressed in the Level II analysis. The containment analysis, therefore, remains as described in Section 4.7 of Volume II of the Plant Hatch IPE submittal.

The radiological portion of the Level II analysis for extended power uprate is provided in Table 68-1 of this response. This information is associated with Sections 4.7.3 and 4.7.4 in Volume II of the original IPE submittal. In Table 68-1, there is reference to MAAP Revision 8.01 as well as Revision 10. Revision 8.01 was used in the original IPE submittal, but has since been revised. The original Plant Hatch sequences were repeated using MAAP Revision 10 and the IPE parameter file to develop a revised Level II radiological analysis. The same sequences were then run using MAAP 10 with the modified, or extended power uprate, parameter file. The uprate parameter file included not only the changes due to extended power uprate, but also other parameter changes which have occurred since the development of the original IPE. The results of the original IPE (Revision 8.01), the recalculated IPE (Revision 10), and the results of the extended power uprate (Revision 10) are described in Table 68-1.

The change in fire risk due to changes in operator error probability for Item 1 is listed under Section 10.5.4 of the extended uprate submittal. The change for Unit 1 is negligible, the change for Unit 2 is as noted. In order to compare this change to the individual plant examination for external events (IPEEE) submittal number of 1.0E-7 (see Table 4.6-1 in the IPEEE submittal), the difference in the before and after extended power uprate numbers under Section 10.5.4 for Unit 2, should be added to the Unit 2 fire CDF listed in Table 4.6-1. This results in a revised fire CDF for Unit 2 equal to 5.5E-6.

TABLE 68-1

RELEASES FOR ANALYZED SEQUENCES

Sequence No. 1 **Sequence Type : Medium LOCA**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.226	0.226	0.201
Time of Vessel Failure (hr)	2.177	2.146	1.903
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.2161	0.2192	0.2523
UO_2 in Pedestal (lbm)	90700.0	90742.2	97567.4
UO_2 in Drywell (lbm)	138000.0	138391.1	144527.0
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	2.5400	2.5718	2.5355
Volatile FP Release (%)	0.0054	0.0055	0.0036
Non-Volatile FP Release (%)	0.0005	0.0005	0.0005
Release Category	A	A	A

Sequence No. 2 **Sequence Type : SBO**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	4.935	5.039	5.326
Time of Vessel Failure (hr)	8.528	8.244	8.118
Time of Containment Failure (hr)	9.689	12.491	13.099
Fraction of Zr Reacted in Vessel	0.1934	0.1787	0.2273
UO_2 in Pedestal (lbm)	38200	28739.4	34372.1
UO_2 in Drywell (lbm)	191000	168798.3	180151.3
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	9.8800	6.8329	7.9651
Non-Volatile FP Release (%)	0.3023	0.0409	0.0979
Release Category	D	C	C

TABLE 68-1 (Continued)

Sequence No. 3 **Sequence Type : Loss of CHR/ Torus Failure**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	31.392	20.386	20.058
Time of Vessel Failure (hr)	37.053	24.891	23.912
Time of Containment Failure (hr)	29.271	24.894	23.915
Fraction of Zr Reacted in Vessel	0.2024	0.1802	0.2242
UO_2 in Pedestal (lbm)	61300	44351.9	46960.8
UO_2 in Drywell (lbm)	73900	115739.1	117591.2
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	1.0480	0.5501	0.2865
Non-Volatile FP Release (%)	0.0100	0.0037	0.0020
Release Category	C	B	B

Sequence No. 4 **Sequence Type : Loss of CHR/ Drywell Failure**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	31.379	20.386	20.058
Time of Vessel Failure (hr)	36.891	24.891	23.912
Time of Containment Failure (hr)	29.149	24.894	23.915
Fraction of Zr Reacted in Vessel	0.2007	0.1802	0.2242
UO_2 in Pedestal (lbm)	65400	45060.7	46405.7
UO_2 in Drywell (lbm)	68200	114384.8	117417.4
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	27.8260	29.7464	32.1777
Non-Volatile FP Release (%)	0.0455	0.0985	0.2138
Release Category	D	D	D

TABLE 68-1 (Continued)

Sequence No. 5 Sequence Type : LOCA Outside Containment			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.051	0.072	0.070
Time of Vessel Failure (hr)	1.411	1.626	1.241
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.0923	0.109	0.1449
UO_2 in Pedestal (lbm)	31700	31525.1	35418.4
UO_2 in Drywell (lbm)	197000	197595	206647.5
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	76.7690	75.7257	75.7777
Non-Volatile FP Release (%)	2.9780	6.0352	3.3441
Release Category	D	D	D

Sequence No. 6 Sequence Type : High Pressure Transient w/Venting			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.629	0.667	0.648
Time of Vessel Failure (hr)	2.949	2.864	2.491
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.1737	0.1632	0.2067
UO_2 in Pedestal (lbm)	94200	93414	101949.7
UO_2 in Drywell (lbm)	135000	135718.7	140146.0
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.0049	0.0033	0.0081
Non-Volatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

TABLE 68-1 (Continued)

Sequence No. 7 **Sequence Type : ATWS w/Injection Failure**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.078	0.079	0.069
Time of Vessel Failure (hr)	1.657	1.573	1.259
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.1398	0.1391	0.1795
UO_2 in Pedestal (lbm)	92700	92554	99400.8
UO_2 in Drywell (lbm)	136000	136576.1	142693.0
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	2.4000	2.3801	2.3934
Volatile FP Release (%)	0.0002	0.0002	0.0002
Non-Volatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

Sequence No. 8 **Sequence Type : Large LOCA**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.021	0.021	0.021
Time of Vessel Failure (hr)	0.984	0.997	0.839
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.0733	0.0755	0.0957
UO_2 in Pedestal (lbm)	93700	92856	98654.7
UO_2 in Drywell (lbm)	135000	136274.7	143438.6
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	2.0600	2.1967	2.2636
Volatile FP Release (%)	0.0019	0.0019	0.0017
Non-Volatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

TABLE 68-1 (Continued)

Sequence No. 9 **Sequence Type : Low Pressure Transient**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.110	0.131	0.130
Time of Vessel Failure (hr)	1.502	1.619	1.247
Time of Containment Failure (hr)	14.794	13.912	18.323
Fraction of Zr Reacted in Vessel	0.0321	0.032	0.0401
UO_2 in Pedestal (lbm)	28400	28358.7	30419.5
UO_2 in Drywell (lbm)	201000	200774.9	211673.1
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	90.3484
Volatile FP Release (%)	2.0450	1.5420	0.8605
Non-Volatile FP Release (%)	0.0012	0.0028	0.0006
Release Category	C	C	B

Sequence No. 10 **Sequence Type : ATWS Torus Failure**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.930	0.706	0.618
Time of Vessel Failure (hr)	3.167	2.347	1.978
Time of Containment Failure (hr)	0.760	0.706	0.718
Fraction of Zr Reacted in Vessel	0.0972	0.0466	0.0742
UO_2 in Pedestal (lbm)	31700	31712.3	35127.7
UO_2 in Drywell (lbm)	197000	197396.8	206913.4
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	1.7550	2.4362	1.0652
Non-Volatile FP Release (%)	0.0613	0.0151	0.0164
Release Category	C	C	C

TABLE 68-1 (Continued)

Sequence No. 11 Sequence Type : ATWS Drywell Failure			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.940	0.706	0.618
Time of Vessel Failure (hr)	3.216	2.330	1.972
Time of Containment Failure (hr)	0.763	0.712	0.723
Fraction of Zr Reacted in Vessel	0.1035	0.0474	0.0701
UO_2 in Pedestal (lbm)	31100	32096.1	35089.3
UO_2 in Drywell (lbm)	198000	197003.5	206946.5
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	11. 4700	23.9213	12.6186
Non-Volatile FP Release (%)	0.2290	0.3461	0.3803
Release Category	D	D	D

Sequence No. 12 Sequence Type : High Press. Transient w/Loss of CHR			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.629	0.667	0.648
Time of Vessel Failure (hr)	2.949	2.864	2.491
Time of Containment Failure (hr)	15.061	15.092	14.484
Fraction of Zr Reacted in Vessel	0.1737	0.1632	0.2067
UO_2 in Pedestal (lbm)	45000	44799.2	42773.6
UO_2 in Drywell (lbm)	184000	184329.5	199317.7
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	5.3780	5.4914	6.6014
Non-Volatile FP Release (%)	0.1970	0.1966	0.2170
Release Category	C	C	C

TABLE 68-1 (Continued)

Sequence No. 13 **Sequence Type : Medium LOCA w/Venting**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power (EPU) Uprate:	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	0.3[illegible]	0.380	0.377
Time of Vessel Failure (hr)	1.937	1.933	1.708
Time of Containment Failure (hr)	---	---	---
Fraction of Zr Reacted in Vessel	0.1713	0.1749	0.2328
UO_2 in Pedestal (lbm)	76800	76937.9	99848.8
UO_2 in Drywell (lbm)	103000	102746.9	142247.4
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.0097	0.0106	0.0033
Non-Volatile FP Release (%)	0.0000	0.0000	0.0001
Release Category	A	A	A

Sequence No. 14 **Sequence Type : SBO w/CI**			
Code Version	**8.01**	**10**	**10**
IPE or Extended Power Uprate (EPU):	**IPE**	**IPE**	**EPU**
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (hr)	4.970	4.519	5.332
Time of Vessel Failure (hr)	8.563	7.670	8.134
Time of Containment Failure (hr)	---	---	39.773
Fraction of Zr Reacted in Vessel	0.194	0.181	0.2258
UO_2 in Pedestal (lbm)	39000	29754.8	34903.4
UO_2 in Drywell (lbm)	190000	167780.9	179416.5
ENVIRONMENTAL RELEASE @40 Hr			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.7130	1.0805	1.3932
Non-Volatile FP Release (%)	0.0070	0.0057	0.0033
Release Category	B	C	C

NRC QUESTION 69

Please provide a discussion as to how uncertainty in human error probabilities as well as modeling uncertainty is addressed in the overall analysis -- both qualitatively and quantitatively.

SNC RESPONSE

Extended power uprate affects only one human error probability, the failure to depressurize with inadequate high pressure injection (non-ATWS). A point source value was used to address this change with regard to the Plant Hatch PRA. This was considered appropriate for evaluating the potential effects of extended power uprate on CDF. The point source value is not associated with a value distribution and does not allow for an uncertainty analysis during model quantification. A complete discussion on the Plant Hatch PRA human error probability, both quantitative and qualitative, is presented in Volume I, Section 3.3.3 of the Plant Hatch IPE submittal.

Modeling uncertainty is evaluated in the discussion regarding the construction of the Plant Hatch PRA model in Section 3.1 of Volume I of the IPE submittal. Extended power uprate does not cause a change or modify any information presented in this section.

PLANT SYSTEMS:

NRC QUESTION 70

In Section 4.4, what effect does extended power uprate have on the total post-loss-of-coolant accident (LOCA) halogen loading and the total integrated dose calculated for the Standby Gas Treatment System?

SNC RESPONSE

The standby gas treatment system (SGTS) review for the initial power uprate assured compliance with the guidelines of Regulatory Guide 1.52, Revision 2. The initial power uprate analysis, based on the TID-14844 source term, concluded that the iodine (halogen) loading is within the regulatory limit of 2.5 mg/g of charcoal based on the estimated amount of charcoal in the filter trains.

To be consistent with the extended power uprate accident dose analyses, the halogen loading analysis was based on the GE generic source term. The iodine (halogen) loading for the extended power uprate analysis was found to be within the requirements of Regulatory Guide 1.52, Revision 2.

Since the SGTS review for the original power uprate used a different source term than the analysis for extended power uprate, a reanalysis of the current power level (2558 MWt) was performed using the GE generic source term. The effect of extended power uprate on the iodine (halogen) loading is therefore the difference in the current power charcoal requirement and the extended power uprate charcoal requirement analyses utilizing the GE generic source term. Table 70-1 provides a summary of the results of the analyses.

TABLE 70-1

UNIT 1 Estimated Amount of Charcoal 1257 Lbs	Minimum Amount of Charcoal Required (Lbs)	Iodine Loading (mg/g of Charcoal)	Regulatory Limit (mg/g of Charcoal)
2558 MWt (TID-14844)	1130	2.25	2.5
2558 MWt (GE Generic Source Term)	926	1.84	2.5
2818 MWt (GE Generic Source Term) [Note]	1020	2.04	2.5
UNIT 2 Estimated Amount of Charcoal 1305 Lbs	**Minimum Amount of Charcoal Required (Lbs)**	**Iodine Loading (mg/g of Charcoal)**	**Regulatory Limit (mg/g of Charcoal)**
2558 MWt (TID-14844)	1130	2.16	2.5
2558 MWt (GE Generic Source Term)	936	1.77	2.5
2818 MWt (GE Generic Source Term) [Note]	1020	1.96	2.5

NOTE: 2818 MWt (2763 MWt + 2% margin).

As was done with the original power uprate, the total integrated dose (TID) calculations for safety-related equipment associated with or located near the SGTS filters were revised to address the impact of extended power uprate. The revisions were based on the revised GE generic source term. The results indicate that adequate equipment dose margin is available following extended power uprate implementation. Table 70-2 provides the impact on the total integrated dose as a result of extended power uprate.

TABLE 70-2

SGTS UNSHIELDED CONTACT TIDS COMPARISON

Unit	Current Power	EPU
1	3.52E + 07 Rads	4.0E + 07 Rads
2	2.58E + 07 Rads	3.0E + 07 Rads

NRC QUESTION 71

Section 4.5.2 states that additional margin has been provided by designing the post-LOCA Combustible Gas Control System to control oxygen within 4 percent volume. Does the margin encompass the increase in oxygen due to the extended power uprate?

SNC RESPONSE

The Unit 1 post-LOCA combustible gas control system evaluation was reanalyzed for extended power uprate, and the results demonstrate the maximum oxygen concentration can be maintained at less than 4 vol % with the existing CAD system. The oxygen concentration for Unit 1 is required to be maintained within 5 vol %. The statement in Section 4.5.2 merely refers to the margin of 1 vol % between 4 and 5 vol %.

NRC QUESTION 72

Section 4.5.5 states that the increase in the radioactivity levels caused by operating at the higher power level would result in an increase in the control room operator dose under post-LOCA conditions. Explain how much the dose would increase, and how the increase corresponds to the existing allowable dose to the control room operators.

SNC RESPONSE

The dose analysis for extended power uprate is based on the GE generic source term and revised atmospheric dispersion factors (χ/Qs). The revised χ/Qs are based on the guidelines provided by NUREG/CR-6331. The dose conversion factors are based on ICRP 30. Since the extended power uprate dose analysis is based on revised methodology and inputs, main control room (MCR) doses at the existing power level of 2558 MWt were recalculated using the revised methodology and input, are included in Table 72-1 for comparison. The detailed post-LOCA dose methodology and analyses results for extended power uprate were provided to the NRC by SNC letter dated April 17, 1997, "Edwin I. Hatch Nuclear Plant Revised LOCA Doses."

TABLE 72-1

SUMMARY OF TOTAL MAIN CONTROL ROOM DOSES

Power Level	MCR Thyroid (rem)	MCR Beta Skin (rem)	MCR Whole Body (rem)
Existing Power Level (2558 MWt) (As reported in the FSARs)	15.13	2.56	0.177
Existing Power Level 2609 MWt (2558 MWt + 2% Margin) [Note]	25.0	4.4	0.6
Extended Power Level 2818 MWt (2763 MWt + 2% Margin)	27.0	4.7	0.7
Regulatory Limits	30.0	30.0	5.0

NOTE: The MCR doses for a power level of 2609 MWt by use of the methodology and inputs for extended power uprate were calculated to estimate the change in MCR doses from original power uprate to extended power uprate.

NRC QUESTION 73

In Table 6-2 on fuel pool cooling, explain what is meant by, "normal condition" for each area in the table. Why do the normal condition temperatures specified under the bulk fuel pool temperature area, exceed the maximum (core offload) condition temperatures?

SNC RESPONSE

Normal condition refers to a heat load for the spent fuel pool that is assumed to be filled to the maximum capacity based on 18 month fuel cycles with the last bundle allowed to decay for 30 days and a single train of the fuel pool cooling system used to remove the decay heat. The normal condition temperatures are higher than that for the maximum (core off-load) condition because in the maximum (core off-load) condition the RHR heat exchanger (RHR Fuel Pool Cooling Assist Mode) is used for decay heat removal instead of a single train of the fuel pool cooling system. The heat removal capability of the RHR heat exchanger is much greater than the fuel pool cooling heat exchanger.

NRC QUESTION 74

Section 6.3 states that the spent fuel pool (SFP) heat loads will slightly increase resulting from plant operations at the proposed power level. Provide the following information:

a. Provide/compare the heat loads and corresponding peak calculated SFP temperatures (for plant operations at the current power level and at the proposed extended power uprate level) during planned refueling and unplanned full core offload. Single failure of the SFP cooling system does not need to be assumed for the unplanned full core offload.

b. Is full core offload the general practice for planned refuelings?

c. How many SFP cooling system trains will be available/operable prior to a planned refueling outage or an unplanned full core offload?

SNC RESPONSE

Response to NRC Question 74.a

The SFP heat load is constantly changing as the fuel decays and as additional bundles are added because of refueling activities; therefore, for design purposes, three cases are evaluated: normal, refueling, and full core offload. Table 74.a-1 includes a summary of the conditions associated with the three cases, and Table 74.a-2 includes the heat loads and pool temperatures for the three cases.

The design conditions that were postulated in sizing the heat removal capacity of the fuel pool cooling system are described in the Unit 1 and 2 FSARs, Sections 10.4 and 9.1 respectfully. It should also be noted that the fuel pool cooling systems for Plant Hatch were not designed as safety related, and are therefore not capable of withstanding a single failure. However, the total loss of all fuel pool cooling, which results in pool boiling is evaluated in the above referenced FSAR sections. (Also, see response to Question 76.)

TABLE 74.a-1

Condition	Heat Load Assumptions	Decay Time Assumptions	Decay Heat Removal Method
Normal	The spent fuel pool is filled to the maximum capacity based on 18 month fuel cycles.	The last bundle is assumed to decay for 30 days.	One train of the fuel pool cooling system
Refueling	The spent fuel pool is filled to the maximum capacity based on 18 month fuel cycles.	The last bundle has decayed for 150 hours.	Two trains of the fuel pool cooling system
Full Core Offload	The spent fuel pool contains a full core which is the last off load.	The full core has decayed for 150 hours.	RHR in fuel pool cooling assist mode

TABLE 74.a-2

	Current Power	Extended Power Uprate
Heat Load (MBTU/Hr)		
Normal Condition	7.93	8.55
Refueling Condition	11.71	12.83
Full Core Off Load Condition	31.09	33.31
Bulk Fuel Pool Temperature (°F)		
Normal Condition	146	149
Refueling Condition	135	137
Full Core Off Load Condition	141	145

Response to NRC Question 74.b

A normal refueling results in either a partial-core (fuel shuffle) offload into the fuel pool or a discharge of the entire core into the pool.

Response to NRC Question 74.c

Two trains of fuel pool cooling, as well as the DHR system, are available for planned refueling outages. For full core offloads, the RHR fuel pool cooling assist mode, two trains of fuel pool cooling, and the DHR system are available.

NRC QUESTION 75

In Section 6.3, what is the maximum temperature that the fuel racks are designed for, and what temperature would the racks actually experience with the extended power uprate?

SNC RESPONSE

The fuel racks were designed for a maximum temperature of 212°F. By design, the racks could experience 150°F during normal operation and 212°F in case of the loss of all cooling.

NRC QUESTION 76

In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling) and the boil-off rate (based on the heat load for the unplanned full core offload scenario). Also, discuss sources and capacity of makeup water and the methods/systems (indicating system seismic design Category) used to provide the makeup water.

SNC RESPONSE

The fuel pool high temperature alarms annunciate at 125°F, however, the pool boiling calculation is initiated from an initial pool temperature of 138°F. The resulting time to boil is 13.56 hours from 138°F, with a boil off rate of 27.55 gpm. The safety related seismic Category I (seismic design) makeup source is the plant service water (PSW) system. The PSW system makeup rate to the fuel pool for both Unit 1 and Unit 2 is ≥ 300 gpm.

The normal makeup source is the seismic Category II (non-seismic) condensate storage and transfer (CST) system. The CST system makeup rate for Unit 1 is 390 gpm and for Unit 2 is 500 gpm.

NRC QUESTION 77

In Section 6.4, what effect does extended power uprate have on the reactor building chilled water system and the service water that supplies the emergency diesel generators?

SNC RESPONSE

No equipment or components requiring additional cooling water from the reactor building chilled water (RBCW) system are to be installed as a result of extended power uprate. Hence, the evaluation of the RBCW system for extended power uprate is based on the small increase in heat loads due to increased pump motor horsepower requirements and increased feedwater piping temperature. The evaluation indicates that the heat loads established in the original design calculations were determined using conservative design inputs. These design heat loads do not change for extended power uprate since the NSSS equipment will operate within present nameplate ratings. There is no significant increase in feedwater piping temperatures as a result of the extended power uprate (< 1°F for Unit 1 and < 2°F for Unit 2), and only a portion of the feedwater piping is located in the reactor building. The increase in feedwater piping temperature is within the original design temperature used for the heat transfer rate from the piping. There is no significant increase in area heat loads, and the heat loads are within the design capacity of the reactor building cooling units. Hence, any additional cooling requirements imposed on the RBCW system remain within the original system design capacity.

There are no additional cooling water requirements for the emergency diesel generators (EDGs) because extended power uprate does not affect the EDG loading. During normal plant operation, there is a small increase in the drywell heat loads for Unit 1 due to the increase in reactor recirculation pump winding losses. This increased heat load is within the design cooling capacity of the drywell coolers and does not impact the PSW flow requirements. There is a slight increase in normal operation PSW flow requirements due to the increase in heat loads for stator winding cooling and isophase bus duct cooling systems associated with the turbine generators. The increased flow requirements are listed in Table 77-1. These increases were evaluated and are within the capability of the existing PSW system design. The PSW supply to the turbine building components is isolated following LOSP and/or LOCA events. PSW flow to the EDGs is needed only after an LOSP or during a LOCA. Hence, the existing PSW system design is not adversely impacted as a result of extended power uprate, and the PSW system is adequate to meet the cooling water flow requirements of the emergency diesel generators at the extended power level of 2763 MWt.

The turbine building components at Plant Hatch are served by the plant service water (PSW) system, not a turbine building closed cooling water system. The effect on the PSW system due to increased heat load from the main generator as a result of extended power uprate was evaluated. The increased heat loads from the stator winding cooling system are within the design capacity of the PSW system. The isophase bus duct coolers for Unit 2 will be replaced to accommodate the additional heat load. The piping modifications required to accommodate the increased flow include increasing the supply and return PSW piping as well as removing a restrictive orifice. The increased heat load for Unit 1 isophase bus duct coolers is within the design capacity of the existing system, therefore no PSW system design change is necessary.

NRC QUESTION 79

Section 7.1 states that the planned modifications to the high pressure turbine and moisture separator reheaters should allow Unit 1 to operate at or near the new licensed power level with adequate turbine pressure control. Explain what is meant by "should" for both Units 1 and 2.

SNC RESPONSE

The intention of "should" within the licensing submittal is in reference to the fact that the BOP modifications are based on assumed design margins which will have to be validated during startup testing. The proposed extended power uprate will be achieved by increasing steam flow. The additional steam flow requires modifications to the high pressure turbine and moisture separator reheaters (MSR). These modifications will allow the additional steam produced in the reactor to be available for power conversion. Adequate turbine pressure control will be maintained regardless of the success of the modifications to the HP turbine and MSR.

NRC QUESTION 80

Section 7.4 states that the feedwater regulating valves were originally designed for greater than rated flow conditions, have been evaluated for extended power uprate conditions, and are considered adequate. What are the original design flows of the feedwater regulating valves, and how do they compare with the flow conditions that result from extended power uprate?

SNC RESPONSE

Section 7.4 states the feedwater heaters and associated regulating valves were originally designed for greater than rated flow conditions. The heater regulating valves are all Fisher air-operated level control valves. The maximum flow and valve flow coefficients (CVs) from the original Fisher control valve data sheets were compared to the design conditions at extended power uprate. The valves were considered adequate for the extended power uprate conditions, with the exception of the 5th stage heater emergency dump valve. The control valve in the line from the 5th stage heater to the condenser will be replaced with a valve of higher capacity prior to implementation of extended power uprate.

Table 80-1 provides a detailed summary comparing the original control valve design conditions with those at extended power uprate conditions.

TABLE 80-1

FEEDWATER HEATER DRAINS

		Design		Extended Power Uprate	
Drains	Control Valve	Flow LB/HR	Valve CV	Flow LB/HR	Valve CV Req'd
UNIT 1					
5th Stg Htr to 7th Stg Htr	1N22-F201 A&B	1491966	320	1051806	212
5th Stg Htr to Cond	1N22-F202 A&B	1153442	235	1051806	*403
7th Stg Htr to 8th Stg Htr	1N22-F203 A&B	4406919	1810	1813192	763
7th Stg Htr to Cond	1N22-F204 A&B	4087917	1810	1813192	783
8th Stg Htr to 10th Stg Htr	1N22-F205 A&B	3069564	1640	2096831	1003
8th Stg Htr to Cond	1N22-F206 A&B	4289029	1640	2096831	820
10th Stg Htr to 12th Stg Htr	1N22-F207 A&B	3389829	2650	2346720	1534
10th Htr to Cond	1N22-F208 A&B	2767784	2650	2346720	1734
12th Stg DC to Cond	1N22-F209 A&B	4050788	2790	2522699	1464
12th Stg Htr to Cond	1N22-F210 A&B	3825345	4130	2522699	2308
UNIT 2					
4th Stg Htr to 6th Stg Htr	2N22-F235 A&B	1653616	320	955362	194
4th Stg Htr to Cond	2N22-F243 A&B	1707576	320	955362	187
6th Stg Htr to 7th Stg Htr	2N22-F237 A&B	1596629	433	1220566	332
6th Stg Htr to Cond	2N22-F138 A&B	1568267	433	1220566	337
7th Stg Htr to 8th Stg Htr	2N22-F241 A&B	4283908	2070	1960495	757
7th Stg Htr to Cond	2N22-F141 A&B	2444193	1340	1960495	890
7th Stg Htr to 10th Stg Htr	2N22-F131 A&B	4421338	3160	1960495	993
8th Stg Htr to 10th Stg Htr	2N22-F133 A&B	3995976	3160	2242871	1305
8th Stg Htr to Cond	2N22-F147 A&B	2424780	1340	2242871	1122
10th Stg Htr to 12th Stg Htr	2N22-F135 A&B	4166071	3160	2473068	1617
10th Stg Htr to Cond	2N22-F144 A&B	5202966	3160	2473068	1400
12th Stg DC to Cond	2N22-F153 A&B	3620113	7170	2669031	3182
12th Stg Htr to Cond	2N22-F150 A&B	3978988	4620	2669031	2060

*1N22-F202 will be replaced prior to operation at extended power uprate

NRC QUESTION 81

Section 7.4 states that recent transient analyses have indicated that the system need only have the capacity to provide at least 105 percent of the extended power uprate feedwater flow to assure that the plant remains available during water level transients, avoids scrams, and minimizes challenges to plant safety systems. Explain what transient analysis is being referred to in this statement.

SNC RESPONSE

An evaluation of the feedwater flow required for reactor water level control was performed for the following operational maneuvers and transient events:

- Power increase along the maximum flow control\rod line.
- Turbine valve test (e.g., bypass valve or control valve).
- Abrupt transfer of control to the backup pressure regulator. *
- Turbine/generator trip. **
- MSIV closure. **

* This is the only event in which the original design utilized more than 105% feedwater flow. The event response at extended power uprate conditions was fully satisfactory with the 105% capability limit.

** Events which can involve a rapid water level reduction due to pressurization and scram. Avoidance of the low level setpoint for initiation of recirculation pump trip and RCIC/HPCI (as appropriate) was not affected by the system capacity because of the rapid level decrease and recovery before a maximum demand is received by the control system.

NRC QUESTION 82

In Section 7.4, explain the impact of extended power uprate on the feedwater heater drains as a result of the higher flow rates.

SNC RESPONSE

The operating conditions at extended power uprate were used to calculate new volumetric flow rates and velocities for the piping, and the required CVs for the control valves. These values were then compared to original design values. Velocities and valve CVs were found to be adequate for extended uprate, with the exception of the 5th stage heater emergency dump valve. The control valve in the line from the 5th stage heater to the condenser will be replaced with a valve of higher capacity prior to extended power uprate implementation.

The higher extended uprate conditions, specifically temperature and velocity, will result in increased flow assisted corrosion (FAC). Monitoring and predictive evaluations will be performed using the Plant Hatch FAC Program, which is based on EPRI guidelines and computer models.

NRC QUESTION 83

In Section 10.1, how do the changes in feedwater, condensate, and reactor water cleanup temperatures impact the mass and energy release rates following high energy line breaks (HELBs)? Explain how the HELB effects, for outside the primary containment, were evaluated for the impact of extended power uprate? What is the impact of a steam jet air ejector steam line break as a result of extended power uprate?

SNC RESPONSE

No change in operating pressure is proposed for extended power uprate. Pressure dependent setpoints (e.g., high pressure scram, ATWS RPT, SRV setpoints) are not being increased from their current values. The stroke times of the MSIV, HPCI, and RCIC steam supply isolation valves, and the RWCU isolation valves are the same. Therefore, the stroke times on key valves which mitigate the postulated high energy line breaks (HELBs) have not been changed with extended power uprate. The feedwater check valves, which terminate the backflow from the reactor for the feedwater line break outside containment will respond in the same manner at extended power uprate conditions as at current rated power. The increase in feedwater temperature for extended power uprate is very small and will not significantly affect the percentage of mass flow from the feedwater piping flashing to steam. The volume of water in the main condenser also does not change significantly for extended power uprate. The RWCU and feedwater line break mass and energy release are actually not the limiting HELB events.

The temperature and enthalpy conditions shown in Figures 1-1 and 1-2 (Reactor Heat Balance, Units 1 & 2) in Enclosure 6 of the licensing submittal for extended power uprate reflect a small increase in the amount of subcooling for the liquid lines connected to the reactor pressure vessel from that previously evaluated for a power level of 2558 MWt. The original mass and energy release rates were based on saturated liquid conditions corresponding to the local internal vessel pressure, established by assuming a 1060 psia steam dome pressure. The change in critical mass flux assuming a 1050 psia steam dome pressure for extended power uprate and using the reduced liquid enthalpy was determined to be insignificant.

The original mass and energy release rates for high energy line breaks outside primary containment were based on saturated fluid conditions and frictionless critical mass fluxes determined at the local reactor pressure vessel pressure, assuming a 1060 psia steam dome pressure. Figures 1-1 and 1-2 in Enclosure 6 of the licensing submittal for extended power uprate indicate a reduced steam dome pressure of 1050 psia. The existing high energy line break analyses were determined to be bounding for extended power uprate conditions for both steam and liquid lines. Additional conservatism was incorporated in the original analyses by neglecting all piping losses and by assuming continuous blowdown at the maximum rate (i.e., no vessel depressurization) until isolation is complete.

A break in the steam jet air ejector steam line was determined to be bounded by a large main steam line break in the turbine building for all environmental effects. No changes to this system will be necessary as a result of extended power uprate.

NRC QUESTION 84

In Section 10.1.1, explain how the structural/equipment HELB analysis for the original 5 percent power uprate for all systems evaluated in the Final Safety Analysis Report remains bounding for extended power uprate.

SNC RESPONSE

As shown in Figures 1-1 and 1-2 (Reactor Heat Balance, Units 1 and 2) in Enclosure 6 of the licensing submittal for extended power uprate the reactor pressure vessel (RPV) dome pressure and the steam conditions (pressure, temperature, and moisture content) remain the same as for the current power level of 2558 MWt, and no change in operating pressure is proposed for extended power uprate. The stroke times of the MSIV, HPCI, and RCIC steam supply isolation valves, and the RWCU isolation valves are the same. Therefore, the pressure dependent setpoints (e.g., high pressure scram, ATWS RPT, SRV setpoints) are not increased from the current values. Stroke times on key valves which mitigate the postulated high energy line breaks (HELBs) are not changed with extended power uprate. The feedwater check valves, which terminate the backflow from the reactor for the feedwater line break outside containment will respond in the same manner at extended power uprate conditions as at current rated power. The increase in feedwater temperature for extended power uprate is very small and will not significantly affect the percentage of mass flow from the feedwater piping flashing to steam. The volume of water in the main condenser also does not change significantly for extended power uprate. The RWCU and feedwater line break mass and energy release are actually not the limiting HELB events.

Therefore, the mass and energy releases considered for the power level of 2558 MWt adequately represent the releases for the extended power uprate conditions, and the results of the existing pressure/temperature analyses remain bounding for extended power uprate.

NRC QUESTION 85

In Section 10.2.2, for the slightly increased extended power uprate temperature and pressure conditions, do the nonmetallic parts of nonelectrical equipment/components (pumps, heat exchangers, etc.) continue to meet the following design and qualification requirements:

a. Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

b. Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

c. Design control measures shall be established for verifying the adequacy of design.

d. Equipment qualification records shall be maintained and shall include the results of tests and material analyses.

SNC RESPONSE

The design and qualification requirements above will continue to be met at extended power uprate conditions. The Quality Assurance (QA) program is provided in section 17 of the Unit 2 FSAR and is implemented by documented administrative controls. Structures, systems and components that could be impacted by changes associated with extended power uprate have been identified. Engineering Evaluation Reports were prepared for impacted equipment areas. The engineering evaluations identified impacts to structures, systems or components and determined adequacy or the need for additional actions. Evaluations included consideration of service conditions such as changes in system pressures, temperatures, and flow rates. In most cases, the component was shown to be within its original design capabilities, and no additional actions were required.

If components, including non-metallic parts of non-electrical components, were impacted in such a manner that a design change is necessary to assure compatibility with normal or accident conditions, these changes will be performed under the QA program. These changes are processed under the requirements of the design and modification processes. The QA program, as implemented in the design modification process, provides controls to ensure that the applicable design and qualification requirements are met.

NRC QUESTION 86

In Section 10.4, what testing will be performed on the feedwater and condensate systems prior to the implementation of extended power uprate?

SNC RESPONSE

Testing prior to implementation of extended power upate will be included in the start-up test plan. The condensate and feedwater testing will include the following:

- Verification that the condensate and feedwater system pressures and flows are within acceptable ranges with respect to the predicted values.

- Verification of acceptable margin for the condensate and condensate booster pumps suction pressures and temperatures.

- Verification of acceptable pump motor currents at extended power uprate conditions.

- Verification of acceptable margin for the reactor feed pump suction pressures and temperatures.

- Dynamic testing of the feedwater controls system.

- Reactor feed pump capability test.

- Vibration testing of feedwater lines.