Entergy

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Rick J. King
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April 3, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
Additional Information Related to License Amendment
License Amendment Request (LAR) 99-15, Changes to Technical
Specifications for Power Uprate of River Bend Station

File No.: G9.5, G9.4.2

Reference:
1) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45077, dated July 30, 1999
2) U.S. Nuclear Regulatory Commission letter to EOI dated February 3, 2000 (TAC NO. MA6185)
3) U. S. Nuclear Regulatory Commission letter to EOI dated February 25, 2000 (Meeting Minutes of February 10, 2000 Meeting)

RBF1-00-0044
RBG-45293

Ladies and Gentlemen:

In the reference (1) letter, EOI requested a license amendment to NPF-47 and Appendix A – Technical Specifications, of the River Bend Station (RBS). This request is to extend operation of RBS from its current licensed power level of 2894 megawatts thermal (MWt) by five percent to an uprated power level of 3039 MWt. The proposed changes were developed using generic guidelines for boiling water reactors (BWR) power uprates described in General Electric (GE) reports.

A001

In the reference (2) letter, the NRC requested additional information (RAI) in 17 areas concerning the Mechanical & Civil Engineering Branch and the Electrical & Instrumentation Branch. Enclosure 1 contains the commitments resulting from the RAI responses. Enclosure 2 has the RAI responses.

In addition to the formal questions, the NRC staff has requested information on a number of other issues concerning Power Uprate at RBS. The responses to these additional questions are included in Enclosure 3.

EOI has also received two additional questions that were sent as part of Reference (3). These questions will be responded to prior to May 17, 2000.

If you have further questions, contact Mr. Barry M. Burmeister of my staff at 225-381-4148.

Sincerely,

Rick J. King

RJK/bmb
Enclosures

cc:

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ENCLOSURE 1

LAR 99-15

Enclosure 1
Commitment Identification Form
Subject: Additional Information Related to License Amendment (LAR 99-15)
RBF1-00-0044
RBG-45293
Date: April 3, 2000

COMMITMENT	ONE-TIME ACTION	CONTINUING COMPLIANCE
Appropriate controls will be exercised to ensure that "critical clearing times" for the significant breakers (as determined by this study) are maintained. Ref. RAIs # 13 & 14.		X

ENCLOSURE 2

LAR 99-15

RESPONSE TO
REQUEST FOR ADDITIONAL INFORMATION
April 3, 2000

Mechanical & Civil Engineering Branch

1. On page 2-4 of reference 1, you stated that the maximum CRD internal pressure which results in the maximum stress in CRD mechanism indicator tube was caused by an abnormal operating condition. Please briefly describe the abnormal condition and discuss how this abnormal condition will be affected under the proposed uprate condition.

   EOI Response:

   The postulated abnormal operating condition assumed a failure of the system pressure regulating valve that subsequently applies the maximum pump discharge pressure to the CRD mechanism internal components. Since the reactor operating condition does not affect the CRD pump discharge pressure, the abnormal pressure condition assumed in the analysis remains the same regardless of reactor pressure. Therefore, the postulated abnormal operating condition is not affected by the proposed uprate condition.

2. In Section 3.1.1 on page 3-1 of the reference, you stated that the power uprate evaluations are performed using the existing safety relief valve (SRV) setpoint (tolerance) analytical limits as the basis. In item 9(b) of Enclosure 2 of Reference 1, you proposed a change of the present –2/+0 % tolerance on the SRV safety function lift setpoint to +/-3%. Please clarify which tolerance on the SRV safety lift setpoint has been used for the safety analysis for power uprate.

   EOI Response:

   The SRV power uprate evaluations were performed considering the setpoint tolerance limits of +/-3% that were proposed in River Bend License Amendment Request (LAR) 99-15. NEDC-32778P, the River Bend Power Uprate SAR (PUSAR) Section 5.3.3 and Table 5-1 further identify the basis for the new SRV setpoint tolerance limits for the safety function.

   As stated in PUSAR Section 3.1.1, the SRV safety setpoint tolerance is independent of power uprate, and is therefore established separately. The SRV safety setpoint tolerance relaxation (+/-3%) is one of the performance improvement features considered in the power uprate evaluation (Section 1.3.2 and Table 1-2).

3. In Section 3.3.2.1, page 3-4, you stated that the reactor internal component loading is determined by load combinations that include reactor internal pressure difference (RIPD), LOCA, SRV, seismic, and fuel loads. You also stated that power uprate was shown to not increase previously calculated fuel lift loads, and therefore, for operation with power uprate, the reactor internals are evaluated only for the effects of the increased RIPD, seismic and SRV loads. Explain why the seismic and SRV loads increase with power uprate and why the effect of LOCA loads such as reactor cavity asymmetric pressurization loads, jet thrust forces was not considered for the power uprate.

EOI Response:

Seismic and SRV loads increased not due to power uprate but because of the use of GE 11 fuel which affects the dynamic characteristics and thereby the dynamic loading on the reactor internals. Therefore, for the operation using GE 11 fuel with power uprate, the reactor internals are evaluated for the effects of a) increased RIPD and b) increased seismic and SRV loads due to the use of GE 11 fuel.

LOCA loads were identified as contributing to reactor internal component loading but were not discussed in the submitted SAR. The LOCA loads such as reactor cavity asymmetric pressurization loads and jet thrust forces were considered in appropriate load combinations for the evaluations of reactor internal components for power uprate and the governing load combinations were used for detailed component evaluations.

4. In reference to Section 3.5, provide a quantitative evaluation for the reactor coolant pressure boundary (RCPB) piping systems and pipe supports with regard to the stresses and fatigue usage factor at the most critical lines and locations that are affected by the increased pressure, temperature and fluid transients for the power uprate. In light of Table 3-7, the maximum increase in pipe stress and support loads can be as much as 21 percent.

EOI Response:

The maximum stress ratios for the piping systems referenced in Section 3.5 most impacted by power uprate (PU) are provided below. All stresses are less than the applicable ASME Code allowable stress.

System	Location	Condition	PU Stress (psi)	Allowable (psi)	Ratio PU/ Allowable
Main Steam A	032	Upset	14088	31860	0.440
Main Steam A	019	Emergency	15044	40950	0.370
Main Steam A	023	Faulted	18631	53100	0.35
Main Steam B/C	039	Upset	15311	31860	0.48
Main Steam B/C	020	Emergency	16008	40950	0.390
Main Steam B/C	039	Faulted	19105	53100	0.36
Main Steam D	032	Upset	15512	31860	0.49
Main Steam D	032	Emergency	15502	39825	0.390
Main Steam D	018	Faulted	20414	54600	0.37
Recirculation loop A/B	867	Upset	23779	31860	0.75
Recirculation loop A/B	809	Emergency	21260	35266	0.60
Recirculation loop A/B	809	Faulted	22119	39184	0.56

The maximum fatigue usage factors for each of the piping subsystems impacted by power uprate are provided below. All fatigue usage factors satisfy the ASME Code requirements.

Subsystem	Location	Maximum Fatigue Usage Factor
Main Steam A	26	0.061
Main Steam B/C	20	0.061
Main Steam D	26	0.061
Recirculation loop A/B	35	0.0258

The maximum support loads for each of the piping systems impacted by power uprate (PU) are provided below. All support loads are within their allowable.

System	Service Load Case	Support Number.	Location Point	PU Load (Lbs.)	Allowable Load (Lbs.)	Ratio PU/Allowable
Main Steam A	Upset	S-102	13	16927	50000	0.339
	Emergency	S-101	13	10504	66500	0.158
	Faulted	S-102	13	39077	75000	0.522
Main Steam B/C	Upset	S-101	13	18498	50000	0.370
	Emergency	S-101	13	10843	66500	0.163
	Faulted	S-101	13	32866	75000	0.438
Main Steam D	Upset	S-102	12	16871	50000	0.338
	Emergency	S-105	19	13195	66500	0.198
	Faulted	S-102	12	40492	75000	0.540
Recirculation loop A/B	Upset	B-402	834	23950	33000	0.725
	Emergency	B-402	834	24033	43890	0.548
	Faulted	B-402	834	26224	49500	0.529

NOTE:

- For the systems identified above, all piping supports, penetrations, and anchors were evaluated for the impact of 105% power uprate. They are all within allowable limits.
- The Code of Record, Code allowables, and analytical techniques used in the power uprate evaluations are the same as those used in the original and existing design basis piping stress qualifications.

5. In Section 3.5, discuss the methodology and assumptions used for evaluating pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchors. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

EOI Response:

METHODOLOGY:

Existing design basis documents, such as piping stress reports, were reviewed to determine the design and analytical basis for piping systems. The power uprate parameters of piping systems

(Pressure, Temperature & Flow) were compared with the existing analytical basis to determine increases in temperature, pressure, and flow due to power uprate conditions.

The following piping systems were evaluated.

Main Steam piping system (Inside Containment)

Recirculation piping system

ASME B&PV Code, Section III, equations were reviewed to determine the equations impacted by temperature, pressure, and flow increases due to power uprate conditions.

Methodologies as described in LTR1, Section 5.5.2 and Appendix K, and LTR2, Supplement 2, Section 4.8 were used to determine the percent increases in applicable ASME Code stresses, displacements, cumulative usage factors (CUF), and pipe interface component loads (including supports) as a function of percentage increase in pressure, temperature, and flow due to power uprate conditions. The percentage increases were applied to the highest calculated stresses, displacements, and the CUF at applicable piping system node points to conservatively determine the maximum power uprate calculated stresses, displacements and usage factors. This approach is conservative because power uprate does not affect weight and dynamic loads; e.g., seismic loads are not affected by power uprate. The factors were also applied to nozzle load, support loads, penetration loads, valves, pumps, heat exchangers and anchors so that these components could be evaluated for acceptability, where required. The vibration displacement factor is calculated as the square of percentage increase in flow times percentage increase in pressure. No new computer codes were used or new assumptions were introduced for this evaluation.

ASSUMPTIONS:

No new assumptions were introduced for this evaluation.

6. In reference to Section 3.11, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Provide, for the most critical piping systems evaluated, the calculated maximum stresses and fatigue usage factor, and code allowable limits. In reference to the notes of Table 3-7, specify what is meant by ASME "Class 4" piping systems evaluated for the power uprate, and how were they evaluated?

EOI Response:

The feedwater and balance-of-plant (BOP) piping systems significantly effected by power uprate were evaluated. In addition, BOP piping systems, which are not affected by power uprate, also were reviewed. The following is a complete list of balance-of-plant (BOP) systems that were reviewed and evaluated:

System Designator	System	Power Uprate Impact
ADHRS	Alternate Decay Heat Removal System	Impacted
SVV/B21	Automatic Depressurization System (ADS)	Impacted
CNA	Auxiliary Condensate	Impacted
CWS	Circulating Water and Cooling Towers	Impacted
CRS	Cold Reheat	Impacted

System Designator	System	Power Uprate Impact
CPM	Combustible Gas / Hydrogen Mixing	Impacted
CND	Condensate Demineralizers Mixed Bed	Impacted
CNM	Condensate System Up to Demineralizers	Impacted
ARC	Condenser Air Removal	Impacted
DRS	Drywell Cooling	Impacted
ESS	Extraction Steam	Impacted
FWS	Feedwater	Impacted
SVH	Feedwater Heater Relief Vents and Drains	Impacted
FWR	Feedwater Pump Recirculation & Feedwater Pump and Drive Lube Oil	Impacted
HVF	Fuel Building Ventilation	Impacted
SFC	Fuel Pool Cooling and Cleanup	Impacted
HDH	High Pressure Feedwater Heater Drains	Impacted
HRS	Hot Reheat	Impacted
HDL	Low Pressure Feedwater Heater Drains	Impacted
MSS/B21/C85	Main Steam	Impacted
MWS	Make-Up Water	Impacted
DSR	Moisture Separator Reheater Vents & Drains	Impacted
DSM	Moisture Separator Vents and Drains	Impacted
DRMS/D17	Process and Area Radiation Monitoring	Impacted
CCP	Reactor Plant Component Cooling Water	Impacted
HVR	Reactor Plant Ventilation	Impacted
SWP	Service Water (Normal)	Impacted
SWC	Service Water Cooling	Impacted
SWP	Service Water Standby	Impacted
GTS	Standby Gas Treatment	Impacted
CCS	Turbine Plant Component Cooling Water	Impacted
HVT	Turbine Plant Ventilation	Impacted
HVN	Ventilation Chilled Water	Impacted
EGA	Air Start-Up Diesel Generator	Not Impacted
ABD	Auxiliary Boiler Blowdown	Not Impacted
ABF	Auxiliary Boiler Feedwater and Condensate	Not Impacted
HVI	Auxiliary Boiler Room Ventilation	Not Impacted
ABM	Auxiliary Boiler Steam	Not Impacted
HVS	Auxiliary Control Building Air Conditioning	Not Impacted
CNS	Condensate Make-Up and Draw-Off	Not Impacted
CPP	Containment Hydrogen Purge	Not Impacted
HVK	Control Building Chilled Water	Not Impacted
HVC	Control Building Ventilation	Not Impacted

System Designator	System	Power Uprate Impact
HVP	Diesel Generator Building Ventilation	Not Impacted
DWS	Domestic Water	Not Impacted
EGF	Emergency Generator Fuel	Not Impacted
FOF	Engine Driven Fire Pump – Fuel Oil	Not Impacted
FPW	Fire Protection	Not Impacted
DFA	Fuel Building Floor Drains	Not Impacted
IAS	Instrument Air	Not Impacted
DFM	Miscellaneous Building Floor Drains	Not Impacted
MSI/E33	MSIV Leakage Control	Not Impacted
LSV	Penetration Valve Leakage Control System	Not Impacted
DED	Radwaste Building Equipment Drain	Not Impacted
DFW	Radwaste Building Floor Drains	Not Impacted
HVW	Radwaste Building Ventilation	Not Impacted
DFR	Reactor Plant Floor Drains	Not Impacted
SAS	Service Air	Not Impacted
DFD	Standby Diesel Generator Building Floor Drains	Not Impacted
SPC	Suppression Pool Cleanup	Not Impacted
DFT	Turbine Building Floor Drains	Not Impacted
VTP	Turbine Plant Equipment Vents	Not Impacted
WOS	Waste Oil Disposal	Not Impacted
WTW	Waste Water Treating	Not Impacted
WTS	Water Treating	Not Impacted
HVJ	Water Treating Building Ventilation	Not Impacted
HVY	Yard Structures Ventilators	Not Impacted
BCS	Bearing Cooling Water	Insignificant Impact
WTH	Chemical Feed – Hypochlorite	Insignificant Impact
WTA	Chemical Feed Acid	Insignificant Impact
WTL	Clarifier System	Insignificant Impact
CMS	Containment Atmosphere Monitoring	Insignificant Impact
LMS	Containment Leakage Monitoring	Insignificant Impact
LWS	Liquid Radwaste	Insignificant Impact
DER	Reactor Building Equipment Drains	Insignificant Impact
SSR	Reactor Plant Sampling	Insignificant Impact
WSS	Solid Radwaste	Insignificant Impact
DET	Turbine Building Equipment Drains	Insignificant Impact
DTM	Turbine Plant Miscellaneous Drains	Insignificant Impact
SST	Turbine Plant Sampling	Insignificant Impact

The most critical piping systems evaluated for power uprate were those that were impacted by increases in temperature, pressure and fluid transients. They are as follows: Main Steam (MSS), High Pressure Core Spray (CSH), Feedwater (FWS), Main Steam SRV Discharge (SRVDL) and Reactor Core Isolation Cooling (ICS). The calculated maximum stresses, fatigue usage factor and code allowable limits for these systems are provided in the following tables:

MAXIMUM STRESS TABLE FOR THE MAIN STEAM (MSS) PIPING SYSTEM
Piping Material = SA106 Gr B Carbon Steel

Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Sm (psi)					17828		
Eqn. 9 (psi)	142	10678	1.1500	12280	26742	0.459	Acceptable
Eqn. 9E (psi)	142	10713	1.1500	12320	40113	0.307	Acceptable
Eqn. 9F (psi)	142	11277	1.1500	12969	53484	0.242	Acceptable
Functional Capability (psi)	142	11277	1.1500	12969	26742	0.485	Acceptable
PIPING LOCATED IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	125	69751	1.0495	73204	42787	1.711	See Eqns. 12 & 13 *
Eqn. 12 (psi)	125	4506	1.0021	4515	42787	0.106	Acceptable
Eqn. 13 (psi)	125	33099	1.0247	33917	42787	0.793	Acceptable
CUF	125	0.0940	**	0.0831	0.1000	0.831	Acceptable
PIPING NOT LOCATED IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	142	43607	43607	45452	42787	1.062	See Eqns. 12 & 13 *
Eqn. 12 (psi)	142	3291	3291	3298	42787	0.077	Acceptable
Eqn. 13 (psi)	142	21453	21453	24671	42787	0.577	Acceptable
CUF	142	0.0119	0.0119	0.0123	0.1000	0.123	Acceptable

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

MAXIMUM STRESS TABLE FOR THE HIGH PRESSURE CORE SPRAY (CSH) PIPING SYSTEM Piping Material = SA106 Gr B Carbon Steel Inside Drywell Wall							
Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Sm (psi)					18068		
Eqn. 9 (psi)	80	16829	1.2100	20363	27102	0.751	Acceptable
Eqn. 9E (psi)	80	16871	1.2100	20414	40653	0.502	Acceptable
Eqn. 9F (psi)	10	20806	1.2100	25175	54204	0.464	Acceptable
Eqn. 10 (psi)	45	132306	1.0172	134582	54204	2.483	See Eqns. 12 & 13 *
Eqn. 10 (psi)	80	40376	1.0784	43540	43363	1.004	See Eqns. 12 & 13 *
Eqn. 12 (psi)	45	9229	1.0080	9303	43363	0.215	Acceptable
Eqn. 13 for Code, (psi)	45	56702	**	53992	54204	.996	
CUF	45	0.8746	**	0.9586	1.0000	0.959	Acceptable
CUF	10	0.1341	**	0.1622	1.0000	0.162	Acceptable – No Break
CUF	25	0.0701	**	0.0803	0.1000	0.803	Acceptable
CUF	35	0.0530	**	0.0692	0.1000	0.692	Acceptable
EVALUATIONS							
Eqn. 13 (psi)	45	56702	**	53992	54204	0.996	Acceptable

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

MAXIMUM STRESS TABLE FOR THE FEEDWATER (FWS) PIPING SYSTEM Piping Material = SA106 Gr B Carbon Steel							
Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Sm (psi)					19692		
Eqn. 9 (psi)	35	20766	1.0000	20776	29538	0.703	Acceptable
Eqn. 9E (psi)	35	20929	1.0000	20929	44307	0.472	Acceptable
Eqn. 9F (psi)	82	37326	1.0000	37326	59076	0.632	Acceptable
NOT IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	119	76854	1.0229	78614	59076	1.331	See Eqns. 12 & 13 *
Eqn. 12 (psi) SA-106 GR C	100	50802	1.0229	51965	52216	0.995	Acceptable
Eqn. 12 (psi) SA –106 GR B	4	29947	1.0229	30633	42480	0.721	Acceptable
Eqn. 13 psi	119	34085	1.0229	34867	47261	0.738	Acceptable
CUF max > .1	85	0.5376	**	0.7892	1.00	0.789	Acceptable
CUF max <.1	100	0.0715	**	0.0882	0.10	0.862	Acceptable
IN BREAK EXCLUSION AREA							
Eqn. 10 (psi)	205	56872	1.0229	58174	47261	1.231	See Eqns. 12 & 13 *
Eqn. 12 (psi)	150	13578	1.0229	13991	47261	0.295	Acceptable
Eqn. 13 (psi)	205	25557	1.0229	26142	47261	0.553	Acceptable
CUF	205	0.0465	**	0.0668	0.10	0.668	Acceptable

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

MAXIMUM STRESS TABLE FOR THE MAIN STEAM SRV DISCHARGE (SRVDL) PIPING SYSTEM Piping Material = Stainless Steel							
Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Eq. 8 (psi)	70	7435	1.00	7435	14620	0.51	NC
Eq. 9 (psi)	97	14468	1.15	16638	17544	0.95	Acceptable
Eq. 9E (psi)	97	15961	1.15	18355	26316	0.70	Acceptable
Eq. 9F (psi)	97	15972	1.15	18368	35068	0.52	Acceptable
Eq. 10 (psi)	555	21202	1.00	21202	23280	0.91	NC
Eq. 11 (psi)	85	23487	1.00	23487	37900	0.62	NC
Eq. 10 (Bldg. Setl) (psi)							NA

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

MAXIMUM STRESS TABLE FOR THE REACTOR CORE ISOLATION COOLING (ICS) PIPING SYSTEM Piping Material = SA 106 Gr B Carbon Steel Inside Drywell Wall							
Attribute	Node No.	Maximum Levels				Max. Uprate Ration = Calc/Allow	Acceptability
		Existing Value	Uprate Factor	Calculated Uprate Value	Allowable Value		
Sm @ 594°F (psi)					17400		
Eqn. 9 (psi)	330	22718	1.013	23013	26100	0.882	Acceptable
Eqn. 9E (psi)	70 & 75	25772	1.013	26107	39150	0.657	Acceptable
Eqn. 9F (psi)	70	26493	1.013	26637	52200	0.514	Acceptable
Eqn. 10 (psi)	1500	113965	1.011	115219	41760	2.759	See Eqns. 12 & 13 *
Eqn. 12 (psi)	350	30423	1.002	30484	41760	0.730	Acceptable
Eqn. 13 (psi)	8050	42170	1.013	42718	50321	0.849	Acceptable
CUF (High Energy)	1500	0.8610	**	0.9527	1.00	0.953	Acceptable
CUF (High Energy)	8050	0.8654	**	0.9576	1.00	0.958	Acceptable
CUF (Moderate Energy)	850	0.1224	**	0.1354	1.00	0.135	Acceptable

LEGEND:
NA – Not Applicable
NC – No Change due to power uprate
* Per NB-3653.6, if Equation 10 cannot be satisfied, the Eqns. 12 & 13 shall be met.
** A detailed evaluation was performed to qualify the equation.

Table 3-7 refers to ASME Class 4 piping. Class 4 piping at River Bend is actually ANSI B31.1 piping. ANSI B31.1 piping was evaluated in the same manner as the ASME Class 2/3 piping, which is outlined in Section 3.11.1 with the exception that the equations and allowables are per ANSI B31.1.

7. In Section 4.1.4 on page 4-7, you stated that a number of the motor-operated-valves within the GL 89-10 program require calculation revisions, actuator adjustments and/or physical changes to ensure their satisfactory performance. Provide a list of affected valves including pressure, differential pressure, temperature, flow rate for both the original and the uprate conditions, their systems, and briefly discuss the proposed adjustments and changes mentioned above.

EOI Response:

There are 22 out of 182 GL 89-10 Program MOVs affected by the planned power uprate. Of these, 12 require calculation revision only: B21-F067A, B, C & D, B21-MOVF016, B21-MOVF019, B21-MOVF085, B21-MOVF086, E51-MOVF022, E51-MOVF076, G33-MOVF039, and G33-MOVF040. Ten MOVs will require either an actuator/motor or gear change-out modification. It is estimated that: 1) four will require a motor replacement (E51-MOVF045, E51-MOVF019, G33-MOVF001 and G33-MOVF004); 2) four will require actuator/motor upgrades (E51-MOVF063, E51-MOVF064, G33-MOVF053 & G33-MOVF054) and 3) two will require a gear change (E51-MOVF013 and E51-MOVF059).

The basis for the suggested changes is included in Attachment One. Our equation to calculate force required to operate is a function of differential pressure, line pressure and packing load; the only impact flow has with the force required to operate is with the water hammer calculation. In general, pressure increases due to water hammer effects are negligible. System flow was not included because it was not a significant contributor in determining the force required to operate the subject valves. Temperature was not provided because for each MOV to be modified (except RWCU MOVs G33-MOVF001, 4, 53 & 54) a relatively conservative conversion factor of 2.31 ft/psig was employed for elevation head corrections. This conversion corresponds to an approximate temperature of 68°F. For RWCU MOVs, a temperature correction of approximately 535°F was employed. When power uprate changes result in the same or higher system temperatures, the existing evaluation temperature employed is conservative.

8. Discuss the functionality of safety-related mechanical components (including air-operated valves, SRVs, pumps and other safety-related valves not covered within GL 89-10) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm all safety-related valves will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed, and provide proposed physical modifications or reanalyzes, if necessary.

EOI Response:

A summary of system capability to support the power uprate is contained in the various sections of the Power Uprate SAR (NEDC-32778P). The supporting system reports evaluate the capability of NSSS and BOP systems to meet their performance specifications under uprated conditions. These reports include evaluations of supporting components, including safety-related valves and other mechanical components, such as pumps and heat exchangers.

Technical specifications were also reviewed, based on the system task evaluations, to ensure that either margins are not reduced by the power uprate or that technical specification modifications are addressed. Mechanical components outside the scope of GL89-10 were evaluated under power uprate conditions and determined to meet performance specifications and technical specification requirements.

The following is a list of systems for which safety related (SR) mechanical BOP components were evaluated for impacts due to power uprate:

- Feedwater
- Standby Service Water
- Reactor Plant Component Cooling Water
- Reactor Plant Sampling (Process Radiation Monitoring)
- Automatic Depressurization
- Alternate Decay Heat Removal
- Fuel Pool Cooling and Clean-Up
- Turbine Plant Miscellaneous Drains
- Reactor Building Equipment Drains
- Combustible Gas / Hydrogen Mixing
- Standby Gas Treatment
- Reactor Plant Ventilation
- Fuel Building Ventilation
- Ventilation Chilled Water
- Ultimate Heat Sink

The functionality of the safety-related mechanical BOP components associated with the above systems is not affected by power uprate. Pressures, temperatures, and flow rates for some of the systems will not change following power uprate. Thus, the safety-related valves not covered within GL 89-10, pumps, and other components in these systems are not impacted. In other systems, pressures, temperatures, and flow rates will increase due to power uprate, but the piping and equipment design conditions bound the maximum uprated conditions. Close and open times for valves not covered within GL 89-10 will not be affected by power uprate. Therefore, all safety-related mechanical BOP components will be capable of performing their intended functions following power uprate without any modifications or reanalysis.

9. In Section 4.1.1.3 on page 4-5, you stated that the maximum drywell pressure values with power uprate are higher than the RBS USAR calculated values but are bounded by the structural design pressure. However, Table 4-1 shows 'NA' for the design structural limit of the peak drywell pressure. Provide the drywell structural design pressure.

EOI Response:

The River Bend USAR reports the drywell pressure design limit in terms of the drywell-to-containment pressure difference instead of a drywell absolute pressure. The statement made in Section 4.1.1.3, page 4-5 was based on a comparison of the calculated peak drywell-to-containment differential pressure of 20.5 psid to the design limit of 25.0 psid reported in the USAR and given in Table 4-1 of the Power Uprate SAR. This information is provided in row 6 of Table 4-1.

10. On page 4-6, you stated that the SRV discharge line (SRVDL) piping loads are discussed in Section 3.11. However, there appears to be no discussion for the SRVDL loads in Section 3.11 under the power uprate condition. Please provide this discussion and results of the evaluation of the effects of increased SRV setpoint pressure on the SRVDL piping and SRVs.

EOI Response:

SRV

The SRV lifted thrust load was originally analyzed as part of the ASME Code calculations based upon an assumed set pressure of 1375 psig [SRV design pressure]. Since the high SRV setpoint (safety function) was increased to 1210 +/- 3% psig, this lifting load therefore envelopes the SRV setpoint pressure increase due to power uprate. The resulting SRVDL dynamically induced moment loading, for the highest SRV loading location, during SRV opening and discharge through the SRVDL was determined to be 450,430 and 232,607 inch-lbs. Moment at the valve inlet and outlet, respectively, for service level D conditions. The power uprate induced moment loading conditions induced into the SRVs remain well within the design allowable flange moment loading permitted for the SRV design. Loading conditions for SRV operability are also within the qualified basis for the SRV design.

The following Table lists the maximum stress for the Main Steam Safety Relief Valve Discharge Piping. All stresses are less than the applicable ASME Code allowable stress.

Equation	Stress (psi)	Allowable (psi)	Ratio
8	9540	15000	0.64
9	17931	18000	0.996
9E	20999	27000	0.78
9F	21773	36000	0.60
11	29304	37500	0.78

11. Provide an evaluation of the potential for flow induced vibration in the main steam and feedwater piping systems, and in the heat exchangers of the condensate and feedwater systems as a result of the proposed power uprate.

EOI Response:

Flow induced vibration for the main steam piping systems (including the piping snubbers, hangers and struts) and piping interfaces with RPV nozzles, penetrations, flanges, and valves were evaluated by extrapolating the data obtained from River Bend startup testing and allowables. It was concluded that the flow induced vibrations are within their allowable limits for power uprate conditions.

Flow induced vibration is not a significant issue for a 5% uprate due to the relatively small changes in main steam, feedwater and condensate flows. There are no modifications being implemented and no equipment being added as a result of uprate that could create the potential for flow induced vibration.

The feedwater heaters were fabricated by Yuba Industries, Heat Transfer Division, in accordance with the closed feedwater specifications, (Reference 4), for guaranteed performance at conditions of approximately 3006 MWt (104 percent) core thermal power (CTP). In addition, the feedwater heaters are specified to be capable of safe continuous operation at feedwater flows of up to 150 percent of the guaranteed performance. Heater conditions for uprated conditions at 3039 MWt CTP, including a 2 percent margin, fall between these specified conditions. On the basis that 150 percent is specified for safe continuous operation, the uprate conditions should also be met for continuous operation.

The heater extraction steam nozzle velocity changes are on the order of 6 to 13 percent and will have negligible effect on nozzle erosion and shell component impingement over the heaters remaining design life.

12. Do you plan to modify piping or equipment supports in conjunction with the proposed power uprate? If there are plans to perform modifications, please provide examples of pipe supports requiring modification and discuss the nature of these changes.

EOI Response:

There are no modifications required due to power uprate in regards to the piping or pipe supports. Piping, piping components and pipe supports are within their applicable code allowables.

Electrical & Instrumentation and Controls Branch

13. It is noted in Section 6.1 of your submittal that an offsite power grid stability analysis review determined that there is no significant effect on grid stability or reliability as a result of increase in electrical output. Please provide a description of what this grid stability power uprate review consisted of. Also, include in this description the major assumptions made for this review and resulting review findings and conclusions. **Note: RAIs # 13 & 14 have combined response.**

14. Please provide a discussion that addresses how the current capability to provide electric power from the transmission network to the RBS will continue to be in full conformance with General Design Criterion 17, "Electric Power Systems", as a result of the power uprate. **Note: RAIs # 13 & 14 have combined response.**

EOI Response:

This study evaluated the RBS upgrade for compliance with the Code of Federal Regulations (CFR), specifically with respect to:

- Grid voltage performance specified in the Branch Technical Position PSB-1 "Adequacy of Station Electrical Distribution System Voltage."
- Off-site power supply reliability based on its susceptibility to Loss of Off-Site Power (LOP) per 10 CFR Part 50, Appendix A General Design Criteria 17.

Off-site voltage and LOP were evaluated under various loading conditions and disturbances including the disturbances in USAR Section 8.2.2 and other critical disturbances identified as a part of this study. The scenarios for this off-site study included all transmission elements in-service with one critical element out-of-service (e.g., unit trip, line or transformer outage). Scenarios simulated both current and upgraded conditions. The relative performance of the upgrade was then determined based upon RBS's current performance.

The 2002 summer peak was the maximum loading condition for load growth and loading level (100%), while the 1999 spring light case was the minimum loading condition for load growth and loading level (40%).

The current RBS model data used all IEEE standard models. The upgraded RBS model only changed the turbine HP stage. The exciter and governor models were unaffected. The increased generator-rating base resulted in increased generator per unit impedance values. The turbine modification combined with the base rating change lowered the inertia constant.

The RBS model included normal BOP auxiliaries plus safety related auxiliaries. The RBS calculation simulated normal loading on the 13.2 kV, 4.16 kV and 480V induction motors in the plant connected to the Fancy Point 230 kV substation. No synchronous motors in the RBS plant were identified.

Motors with ratings greater than or equal to 1000 hp were modeled explicitly on the 13.2 kV and 4.16 kV buses. Other motors were lumped and modeled as equivalents at the appropriate voltage level. Seventy-four (74) induction motors were simulated as negative generators at a fixed lagging power factor in the load flow.

Steady State Analysis:

The following general criteria were applied to the system studies:

Base Case Conditions (N-0).	Loading ≤ Normal rating (RATE1) 0.95 pu ≤ Voltage ≤ 1.05 pu
Contingency Conditions (N-1).	Loading ≤ LTE rating (RATE2) 0.95 pu ≤ Voltage ≤ 1.05 pu

These criteria apply to all 230 kV and 500 kV bus voltages and branch loadings in the Entergy area, which were monitored for criteria deviations. Additional criteria were applied to the 4.16 kV safety related 1E buses SWG*1A, SWG*1B and E22*S004. These criteria address loss of voltage and degraded voltage conditions at safety related buses that could lead to an unnecessary power supply transfer from OPS to the EDGs.

Key study criteria, combining relay settings for loss of voltage and degraded voltage are:

No LOCA	Voltage ≤ 0.71 pu for Time ≤ 3 seconds Voltage ≤ 0.90 pu for Time ≤ 60 seconds

LOCA Voltage ≤ 0.90 pu for Time ≤ 3 seconds

Steady-State Analysis determined the RBS upgrade impact on the Fancy Point 230 kV bus voltage. The analysis examined both load conditions (2002 summer peak and 1999 spring light load) for the pre-upgrade RBS power output (990 MW) and post-upgrade RBS power output (1130 MW). Normal system conditions, as well as contingency conditions, were evaluated.

The contingency list was based on the contingencies listed in the USAR, and revised to incorporate the circuit breaker arrangements at the Fancy Point 230/500 kV and adjacent substations. The steady-state contingency list includes six generator trips, six single line trips, three multiple element trips, one load trip and one LOCA contingency. The LOCA contingency consists of the loss of the RBS generator and switching out loads at busses SWG4A and SWG4B, respectively, and switching in motor loads at SWG*1A (bus number 50034), SWG*1B, and E22*S004. Furthermore, in order to simulate the worst-case LOCA scenario, the largest motors at SWG*1A and SWG*1B were conservatively modeled in the locked-rotor condition.

Contingency List

Contingency Number	Description
1	Loss of River Bend unit
2	Loss of Big Cajun #3
3	Loss of Grand Gulf generator
4	Loss of Arkansas Nuclear generator
5	Loss of Browns Ferry St. generator
6	Loss of Willow Glen generator
7	Loss of Fancy Point – Enjay 230 kV line (L-352, 3Φ)
8	Loss of Fancy Point – Pt. Hudson 230 kV line (L-353, 3Φ)
9	Loss of Fancy Point – Big Cajun1 230 kV line (L-715, 3Φ)
10	Loss of Fancy Point 500/230 transformer (3Φ)
11	Loss of Fancy Point – Big Cajun2 500 kV line (L-746, 3Φ)
12	Loss of Fancy Point – McKnight 500 kV line (L-752, 3Φ)
13	Contingency 1 + Stuck Circuit Breaker (20635, 1Φ) & subsequent trip of RBS Generator (not considered for Steady State Analysis identical to 1)
14	Contingency 7 + Stuck Circuit Breaker (20665, 1Φ); Loss of Fancy Point – Enjay and RSS#2 230 kV lines and RBS generator
15	Contingency 8 + Stuck Circuit Breaker (20650, 1Φ)
16	Contingency 9 + Stuck Circuit Breaker (20740, 1Φ) Loss of Fancy Point – Big Cajun1 230 kV line and Fancy Point 500/230

Contingency Number	Description
	transformer
17	Contingency 11 + Stuck Circuit Breaker (20770, 3Φ fault, normal clearing on two phases backup relay operation on third phase) Loss of Fancy Point – Big Cajun 2 and Fancy Point – McKnight 500 kV lines, and Fancy Point 500/230 kV transformer
18	Loss of load near Enjay 230 kV bus
19	Loss of RBS generator with LOCA motor sequencing (not used for

The results show that the RBS upgrade has a negligible effect on post-contingency steady-state voltage at the Fancy Point 230 kV bus. In all simulations, including the LOCA contingency, the Fancy Point 230 kV bus voltage remained at approximately 1.02 pu, bus voltage at RBS emergency busses remains above 0.90 pu.

Stability Analysis:

Fifteen contingencies were listed in the original USAR and formed the foundation of the contingency list used in this analysis. Revisions and additions to this list were based upon the circuit breaker arrangements at the Fancy Point 230/500 kV and adjacent substations. It should be noted that:

- Transfer trip relaying is employed on all 230 kV (6 cycles) and 500 kV (4.5 cycles) transmission lines connected to the Fancy Point substation.
- Contingencies 1 through 12 are disturbances involving three-phase faults cleared by primary relay operation and line or generator switching.
- Contingencies 13 through 16 are disturbances involving single-phase faults and stuck breakers cleared by backup relay operation and line or generator switching.
- Contingency 14 entails the loss of the RBS auxiliary load served by RSS#2. Therefore, the RBS generator is also tripped at approximately 6 seconds.
- Stuck breaker contingency 17 consists of a three-phase fault on the 500 kV bus, which is cleared by normal relay operation on two phases and backup relay operation on the third phase.
- Contingency 18 is a load trip contingency.

Contingencies 1 through 16 and 18 are normal contingencies, which should be stable and damped. Contingency 17 is an extreme contingency used for testing the robustness of the system.

Single-phase faults were simulated by using a non-zero fault impedance, which was the sum of the negative-sequence and zero-sequence impedance at the faulted bus. The sequence impedances were calculated using the short-circuit feature of the software used for the analyses. The negative and zero sequence impedances calculated by the software were within the appropriate tolerances of the sequence impedances.

Simulation Results

For both 2002 summer peak and 1999 spring light load conditions, the upgrade impact on system performance for contingencies 1 – 16, and 18, is small. System response for these cases is first-swing stable and damped.

Contingency 17 was unstable and therefore additional analysis was performed with the following modifications to software model:

- reduce RBS upgrade power output from 1130 MW to 1100 MW,
- increase induction motor under-voltage tripping time-out from 1 (conservative) to 2 (as-built) seconds
- decrease Fancy Point 230 kV bus backup clearing time for the affected breakers (18.5 to 16 cycles for the 2002 summer peak load conditions & 18.5 cycles to 14 cycles for light load conditions)
- decrease Fancy Point 500 kV bus backup clearing time for the affected breakers (16.5 to 14.5 cycles for the 2002 summer peak load & 16.5 to 12 cycles for the 1999 light load conditions)

The results of the additional analysis show stable performance with no RBS auxiliary motors tripped due to undervoltage protection. The system response to all conditions (considering sensitivity analysis) is stable and well damped.

CONCLUSIONS

The steady-state analysis shows that the up-grade of the RBS plant from 990 MW to 1130 MW has little impact on Fancy Point 230 kV bus voltage.

The stability analysis results show stable performance with an RBS power output level of 990 MW for all contingencies under both 1999 spring light load and 2002 summer peak load conditions. At an RBS power level of 1100 MW, all contingencies under both 1999 spring light load and 2002 summer peak load conditions demonstrate stable performance.

Appropriate controls will be exercised to ensure that "critical clearing times" for the significant breakers (as determined by this study) are maintained.

15. Provide a discussion that addresses the impact of the power uprates on the load, voltage, and short circuit current values for all levels of the station auxiliary electrical distribution system (including ac and dc).

EOI Response:

The On-site power distribution system loads were reviewed under both normal and emergency operating scenarios. In both cases, loads are computed based on equipment nameplate data or brake horsepower (BHP). These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operation at uprated power levels is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running KW or BHP. Therefore, there are no changes to the load, voltage drop or short circuit current values.

The DC power distribution system loads were reviewed in a similar fashion as the On-site power distribution system. In both normal and emergency operating scenarios, loads are computed based on equipment nameplate data or brake horsepower (BHP). These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operation at uprated power levels is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running KW or BHP. Therefore, there are no changes to the load, voltage drop or short circuit current values.

16. In Section 10.3.1.1 of the River Bend Station power uprate submittal, it is stated that the current accident and normal plant conditions for temperature, pressure, and humidity inside the primary containment are "nearly unchanged" for the power uprate conditions. Please provide a detailed discussion to clearly explain how the current accident and normal temperature, pressure, and humidity profiles for inside the primary containment do not change for the power uprate conditions and why these changes have no impact on the environment qualification of electrical equipment. In addition, please provide a similar discussion for the temperature, pressure, and humidity profiles for high energy line break areas outside of the primary containment.

EOI Response:

Normal profiles inside and outside primary containment:
The design basis normal temperature, pressure, and humidity profiles for both inside and outside of the primary containment remain unchanged from the pre-uprate profiles. This is due to the existing margin between actual and design basis conditions and existing margins in the ventilation (cooling) systems as described below for the containment.

The power uprate added approximately 5% heat loads in the containment from piping, fuel pool, etc. For normal operation, the containment unit coolers have design margin of 25% in the cooling capacity of HVR-UC1A, B, &C. The 25% design margin bounds the 5% increase in heat gains as a result of containment pool temperature increase, piping heat gains, etc. The containment unit coolers will continue to maintain design environmental conditions (temperature and relative humidity) stated in USAR and EDC during normal operation. The containment coolers are recirculation type and do not affect the containment pressure. The annulus pressure control system maintains a negative pressure of 3 inches water gauge in the annulus with respect to atmosphere during normal operation. The annulus pressure control system will not be impacted since there is no change in the environmental conditions in the containment.

Accident profiles inside and outside primary containment:
GE calculations show that power uprate conditions will increase the blowdown mass and energy releases (typically by less than 5%).

The peak accident temperature, pressure, and humidity values inside the primary containment remain bounded by the existing profiles. The time histories of mass and energy release rates inside containment continue to be evaluated against the existing temperature, pressure and humidity profiles. This review is not expected to result in major plant hardware changes or substantial plant modifications.

Due to conservatisms in the original design basis analyses, the calculated mass and energy release rates with power uprate for several of the evaluated high energy lines outside containment were determined to be bounded by the original design basis analysis values.

From the evaluation it was determined that:

a) The following high energy line breaks are bounded by the design basis analyses:

   1) 4" RCIC (Double Ended Rupture) DER
   2) 20" feedwater (Single Ended Rupture) SER
   3) 8" RHS DER
   4) All RWCU lines

b) The following high energy line breaks (outside of containment) are not bounded by the original design basis analyses:

   1) 8" RCIC DER and 8" RCIC SER (1% increase in mass blowdown)
   2) 24" Main Steam (MS) SER and 24" MS DER, with water carry over (1% increase in mass blowdown during the steam blowdown period, 35% decrease during the two-phase blowdown period)
   3) 24" MS SER and 24" MS DER, without water carryover (4% increase in mass blowdown).

For those high energy line breaks that are not bounded by the original design basis analysis (see Table 10-1 of Reference 1 for the increased flows, pressures and temperatures) time histories of mass and energy release rates were generated. The temperature and pressures (although increasing slightly) were found to remain within the existing Environmental Design Criteria envelope.

17. In Sections 10.3.1.1 and 10.3.1.2 of the submittal, it is noted that the environmental qualification radiation levels under accident conditions are conservatively evaluated to increase 3% to 8% inside and outside the primary containment. It is also noted that reevaluation of the EQ for the uprated power conditions identified some equipment located inside and outside the containment that is affected by the higher accident radiation level. Please identify this equipment and discuss how this equipment will be requalified for the new radiation values. Also, provide the current, the revised, and the bounding radiation level conditions and provide numerical values for these radiation level conditions.

EOI Response:

EOI is in the process of revising the Equipment Qualification Assessment Reports to reflect the new environmental conditions for equipment affected by power uprate (total: 70 EQAR's being revised). Each EQAR is the qualification document for a particular component TYPE, such as "Rosemount 1153 Transmitters", and includes the data for harsh locations in which that component type is located. The EQAR's also document the qualified life of the component in its various locations. Thirteen of the revised EQAR's document a reduction in qualified life and will require changes to the EQPM replacement frequency once operation at the uprated power starts.

The radiological calculations were reviewed to support of the Power Uprate project. This review included the basis calculations for River Bend's Environmental Design Criteria (EDC). Typically, the pre-Uprate calculations had evaluated a core thermal power level of 3039 MWt that corresponds to 105% of the current licensed core thermal power. For Power Uprate an additional 2% was added in accordance with Regulatory Guide 1.78 recommendations (instrument uncertainty) for an assumed power level of 3100 MWt. Since the source term is proportional to power level one would expect that the increase in doses would simply be 2% if all else were held equal. However, General Electric recommended use of a more recently calculated source term for Power Uprate evaluations. As a result, the relative isotopic concentrations changed which is why different doses increased by a different factor (i.e., airborne may have increased by 3% whereas ECCS piping may increase by a different amount such as 5%). Note that the source term recommended by GE included significantly more isotopes, which accounts for some of the increase. The increase in core thermal power, as well as the change in source term, results is the 3% to 8% increase GE referred to in the Power Uprate SAR.

Other factors also impacted the change to source terms including revisions to calculation methodology to reflect the current methodologies used in dose calculations. Also, the impact of the Hydrogen Water Chemistry (HWC) System was also accounted for even though it is not currently used. Note that HWC has a significant impact to calculated normal operation doses for steam affected areas of the plant since the normal operation N-16 concentration may increase as much as a factor of six.

The Power Uprate/HCW doses were then evaluated for their potential impact to equipment qualification (EQ). Only a few items were identified which required further actions to support equipment qualification. These items, as well as the resolution, are listed below.

- **FCI Flow Switch (LSV-FS 20A & B) located in EQ Zone AB-141-3**
  Upon first review these items did not meet the calculated zone values. LSV-FS20A was removed by MR96-0059. LSV-FS20B was abandoned in place by MR96-0059. Because these components are no longer in service and deleted from the EQ program, no further actions were required.
- **FCI Level Switch (DFR-LSW,X,Y,Z08A & 8B) located in EQ Zone AB-141-3**
  These level switches were originally qualified to the environmental zone general area dose. With the uprated conditions, the general area dose exceeded the qualification dose applied to the equipment. A location specific dose calculation was performed. The calculated location specific dose is bounded by the qualification of the equipment. No further actions are required.

- **DRMS Radiation Monitor (RMS RE15A &B) located in EQ Zone AB-095-6**
  This equipment has permanent shielding (1.5" of steel) installed to shield scatter gamma doses from high-pressure core spray piping following a LOCA. A location specific analysis of this equipment previously existed. A more rigorous analysis was developed to qualify the monitors with the current shielding. No further actions were required.
- **RCIC Terry Turbine Remote Electronics Panel located in EQ Zone AB-095-4**
  This equipment is presently located in the auxiliary building on the 95' elevation. Reactor Water Cleanup piping located in the overhead contributes significantly to the Normal Operation and the Total Integrated Doses (TID). Permanent shielding is installed to reduce these doses. A location specific analysis of this equipment previously existed. After revising the calculation for HWC and Power Uprate the electronics panel exceeded its qualification. Rather than add additional shielding it was determined that moving the panel to a low dose area would be a more reasonable solution. The panel has been moved in RF-9 to the control building in accordance with ER99-0574. The control building is considered a "mild" environment.

  Tables 1 and 2 in Attachment 2 compare Pre HWC/Power Uprate vs. Post HWC/Uprate Normal and Worse Case Accident Radiation doses.

ENCLOSURE 3

LAR 99-15

OTHER NRC INFORMATION REQUESTS SUPPORTING POWER UPRATE

1. Materials Branch needs more detailed information on P-T curves presented in SAR. In a telephone conference, RBS agreed to the following:

   a. Provide tabulated values for P-T limits.

      EOI Response:

      Table of P-T limit values for RBS operation with Power Uprate through 14 and 32 EFPY are given in Attachment 2, Tables 3 and 4.

   b. Provide information on the background/methodology for preparing the P-T limit calculations. Were any unique methods used (e.g., ASME III, Appendix A instead of Appendix G)?

      EOI Response:

      Since two or more credible surveillance data sets are not yet available for RBS, calculation of neutron radiation embrittlement of vessel beltline materials was done in accordance with Regulatory Position C.1 of Regulatory Guide 1.99, Rev. 2. The value of neutron fluence was calculated at a 1/4 depth (1/4T) into the vessel wall from the inside diameter using equation 3, Paragraph 1.1 of Regulatory Guide 1.99, Rev. 2. This 1/4T depth is recommended in ASME BPV Code Section XI, Appendix G, Subarticle G-2120 as the maximum postulated defect depth.

      The adjusted reference temperature (ART) for each beltline material was evaluated using methods consistent with Regulatory Guide 1.99, Revision 2; equations and values found in Paragraph 1.1. Paragraph 3 was used as the criterion to evaluate end of life ART. The decrease in upper shelf energy (USE) for each beltline material was evaluated using methods consistent with Regulatory Guide 1.99, Revision 2; discussion found in Paragraph 1.2. The value of peak 1/4T fluence was used to evaluate USE. Requirements for USE must satisfy 10CFR50 paragraph IV.A.1a.

      The P-T curve plots the minimum RPV temperature required for safe operation of the vessel for a given pressure. The three vessel regions that affect the operating limits are: the closure flange region, the core beltline region, and the non-beltline region. The closure flange region limits are controlling at lower pressures primarily because of 10CFR50, Appendix G requirements. The beltline and non-beltline region limits were evaluated according to the requirements of 10CFR50 Appendix G, ASME BPV Code Section XI Appendix G, and Welding Research Council Bulletin 175. The beltline (core region) portion of the P-T curves accounts for a value of Shift. This Shift is a function of neutron fluence and was calculated using methods consistent with Regulatory Guide 1.99, Revision 2.

      Nuclear Regulatory Commission (NRC) 10CFR50 Appendix G specifies fracture toughness requirements to provide adequate margins of safety during operation to which the pressure-retaining component pressure boundary may be subjected over its service lifetime. The ASME Code Section XI, Appendix G forms the basis for the requirements of 10CFR50 Appendix G. The limits for pressure and temperature are required by 10CFR50 Appendix G for three categories of operation: (a) hydrostatic pressure tests and leak tests, (b) core not critical heatup/cooldown, and (c) core critical operation. The condition that resulted in the highest temperature for the limiting material determined the minimum temperature requirement for the vessel. In all cases, the applicable temperature was the greater of the 10CRF50 minimum temperature requirement and the ASME Code Appendix G limits.

The P-T curves for the non-beltline region were conservatively developed for a large BWR/6 (nominal inside diameter of 251 inches). The analysis is considered appropriate for River Bend as the specific values are bounded by this generic analysis. The generic value was adapted to the conditions at River Bend by using the specific RT_{NDT} values for the reactor pressure vessel (RPV). The presence of nozzles and control rod drive (CRD) penetration holes of the upper vessel and bottom head, respectively, has made the analysis different from a shell analysis such as the beltline. This was the result of the stress concentrations and higher thermal stresses for certain transient conditions, experienced by the upper vessel and the bottom head.

The P-T operating limits for the beltline region were determined according to the ASME Code Appendix G. As the beltline fluence increases with the increase in operating life, the P-T curves shift to a higher temperature.

The stress intensity factors (K_I) were calculated for the beltline region according to ASME Code Appendix G procedures and were based on a combination of pressure and thermal stresses for a 1/4 T flaw in a flat plate. The pressure stresses were calculated using thin-walled cylinder equations. Thermal stresses were calculated assuming the through-wall temperature distribution of a flat plate; values were calculated for 100°F/hr thermal gradient. The shift value of the most limiting ART material was used to adjust the RT_{NDT} values for the P-T limits.

The methods of ASME Code Appendix G were used to calculate the pressure test beltline limits. The vessel shell, with an inside radius (R) to minimum thickness (t_{min}) ratio of 15, was treated as a thin-walled cylinder. The maximum stress is the hoop stress, given as:

$$\sigma_m = PR/t_{min}$$

The stress intensity factor, K_{Im}, was calculated using Figure G-2214-1 of the ASME Code Appendix G accounting for the proper ratio of stress to yield strength. Figure G-2214-1 was taken from WRC Bulletin 175, based on a 1/4 T radial flaw with a six-to-one aspect ratio (length of 1.5T). The flaw is oriented normal to the maximum stress direction, in this case a vertically oriented flaw. This orientation is used even in the case where the circumferential weld is the limiting beltline material, as mandated by the NRC in the past.

The calculated value of K_{Im} for pressure test was multiplied by a safety factor (SF) of 1.5, per ASME Code Appendix G for comparison with K_{Ir}, the material fracture toughness. A safety factor of 2.0 was used for the core not critical and core critical conditions.

The relationship between K_{Ir} and temperature relative to reference temperature ($T - RT_{NDT}$) is shown in Figure G-2210-1 of ASME Code Appendix G, represented by the relationship:

$K_{Im} \bullet SF = K_{Ir} = 1.223 \exp[0.0145 (T - RT_{NDT} + 160)] + 26.78$

This relationship is derived in WRC Bulletin 175 as the lower bound of all dynamic fracture toughness and crack arrest toughness data. This relationship provides values of pressure versus temperature [from K_{Ir} and $(T - RT_{NDT})$, respectively].

For the pressure test curve, a stress intensity factor (K_{it}) was added for a heatup/cooldown rate of 20°F/hr to consider operating conditions. For the core not critical and core critical condition curves, a stress intensity factor was added for a heatup/cooldown rate of 100°F/hr.

c. How was the fluence level post-Power Uprate determined?

EOI Response:

Fast neutron (E > 1 MeV) flux density at a sample capsule near the reactor vessel wall of River Bend Station was determined to be 4.4e9 n/sec-cm^2 at full power 2894 MWt, based on flux wire measurement data taken at end of cycle 1 (EOC 1). For reactor vessel fracture toughness evaluations, the lifetime neutron fluence was projected based on the vessel flux density and an assumed 80% capacity factor, or 32 effective full power years (EFPY) of operation. The peak neutron flux at vessel ID was determined by using a lead factor 0.67, which was calculated for a generic River Bend size plant. The peak value of fast neutron fluence at the vessel ID was estimated to be

$$(4.4e9/0.67) * 32 * 365 * 86400 = 6.6e18 \text{ n/cm}^2$$

As River Bend Station progresses toward power uprate, the shifting of power profile as well as the increase in power density both have certain impacts on the neutron flux level. One of the deciding factors which affect neutron flux level at the reactor vessel is the power density of peripheral bundles near the core edge. In order to find a bounding estimate of the impact of power uprate on the vessel fluence, the cycle-dependent relative power densities of first tier peripheral bundles of the following cycles were selected for comparisons:

Cycle 1 (basis for capsule flux)
Cycle 6 (equilibrium cycle at rated power)
Cycle 8 (last rated cycle)
Cycle 10 (first uprated cycle)
Cycle 14 (equilibrium cycle at uprated power)

The average value of the first tier peripheral bundle relative power densities for each cycle were compared and normalized to the Cycle 1 value. In addition, an adjustment factor was added for uprated cycles to account for the effect of power increase from 2894 MWt to 3039 MWt. Based on the comparison result, it was concluded that prior to power uprate, the average power density of an equilibrium cycle could be as much as 18% higher than that of Cycle 1. It was also evident that the net effect of power uprate on the peripheral bundles is such that power density at the core edge could be 20% higher than that predicted by Cycle 1.

Therefore, the following assumed flux levels at the capsule location were used in calculating the vessel peak surface fluences (E > 1 MeV) for the various EFPY of interest:

- Fluence estimation for periods of rated power (2894 MWt) are based on assumed flux level of 5.2e9 n/sec-cm^2 at the capsule location.
- Fluence estimation for periods of uprated power (3039 MWt) are based on assumed flux level of 5.3e9 n/sec-cm^2 at the capsule location.

2. Received from Reactor Systems. The power uprate amendment was reviewed with consideration given to the recommendations from the Report of the Maine Yankee Lessons Learned Task Group, dated December 1996. This report is documented in SECY-07-042, "Response to OIG Event Inquiry 96-04S regarding Maine Yankee," dated February 18, 1997.
Your submittal implies that NRC approved computer codes and calculation techniques were used to perform the calculations that demonstrate meeting the stipulated criteria.

   a. Identify all codes/methodologies used to obtain safety limits and operating limits and how GE verified that these limits were correct for the appropriate uprate core.
   b. Identify and discuss any limitations associated with these codes/methodologies that may have been imposed by the Staff.
   c. Confirm that EOI has audited GE to assure that GE uses the codes correctly for the RBS power uprate conditions and that GE followed the limitations and restrictions appropriately.

EOI Response:

<u>Response Part a:</u>

The power uprate process uses approved codes and methodologies as documented in the NRC approved License Topical Report 1 (LTR1), NEDC-31897P-A, " Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," dated May 1992.

A list of the codes used by GE to perform power uprate analyses is provided in the attached Table 1, "River Bend Computer Codes for Power Uprate Analysis."

No structural computer codes were used in the power uprate evaluations of reactor internals. The evaluations were the following:

1. Components for which loading did not increase for power uprate when compared with pre-uprate loading. No further evaluation was done for these components and they were qualified as acceptable for power uprate.

2. Components for which loading increased for power uprate. The stresses for these components were scaled up by the ratio of power uprate loading to pre-uprate loading. All scaling calculations were performed without using the structural computer codes. The stresses were shown to be within the allowable limits and the components were qualified for power uprate.

<u>Response Part b:</u>

The application of the codes identified in Table 1 to power uprate complies with any limitations or restrictions specified by the NRC in the approving SER where applicable for each of the codes and in the SERs for the power uprate programs. GE's ECCS-LOCA and transient analysis codes have generic NRC approval, and are docketed under the GESTAR II fuels program.

Table 1 River Bend Computer Codes for Power Uprate Analysis

Evaluation Subject	Computer Code	Version or Revision	NRC Approved	Reference
Reactor Internals Pressure Difference	LAMB	Version 07	Yes	NEDE-20566-P-A, Sept. 1986; see Note 7
	TRACG	Version 01	Yes	NEDE-32178P; see Note 1
	ISCOR	Version 09	Yes	NEDE-24011-P-A
Containment Evaluations	M3CPT	Version 05	Yes	See Note 2
	LAMB	Version 08	Yes	NEDE-20566-P-A, Sept. 1986; see Note 7
	SHEX	Version 04	No	See Note 3
ECCS-LOCA and Appendix R – Fire Protection	SAFER/GESTR-	Version 04	Yes	See Note 4
	LOCA	Version 08	Yes	NEDE-20566-P-A, Sept. 1986
	LAMB	Version 01	Yes	NEDE-20566-P-A, Sept. 1986
	SCAT	Version 03	Yes	NEDE-20566-P-A, Sept. 1986; see Note 8
	TASC			
Radiation Sources and Fission Products	ORIGEN2	Version CCC-371A, 8/6/91	Industry Code	ORNL/TM-7175, A Users' Manual for the ORIGEN2 Computer Code, July 1980
Transients	ODYNV	Version 09	Yes	NEDE-24011-P-A
	TASC	Version 03	Yes	GENE-666-03-0393, March 1993, see Notes 5 and 8
	PANACEA	Version 10	Yes	NEDE-24011-P-A
	ISCOR	Version 09	Yes	NEDE-24011-P-A
ATWS	ODYNV	Version 09	Yes	NEDC 24154P Supplement 1, Vol. 4
	TASC	Version 03	Yes	NEDE-24222, Dec. 1979; see Notes 5 and 8
	STEMP	Version 03	Yes	NEDE-24222, Dec. 1979; see Note 6

Table Notes:

1. TRAC is an industry code, used by many others, including the NRC. TRACG is the GE version of TRAC. The stated reference is for application to SBWR licensing safety analysis. TRACG results have been submitted to the NRC as a best-estimate benchmark in many applications.

2. M3CPT was reviewed and approved by NRC as part of the generic containment load definition review.

3. SHEX has not been explicitly reviewed by the NRC. It is a containment heat balance model and has been used to calculate suppression pool temperatures in all recent containment applications by GE. These applications of SHEX have been reviewed and approved by the NRC.

4. SAFER02/03 have been reviewed and approved by the NRC per NEDE-23785-1-PA R. 1, Oct. 1984, and NEDE-30996P-A, Oct. 1987.

5. The NRC reviewed the code for this application in the reference stated.

6. The NRC has not specifically reviewed this code, but they have approved analysis using this code as documented in the stated reference.

7. LAMB07 includes more detailed modeling necessary for RIPD analyses, which have been reviewed and approved previously by the NRC.

8. The TASC code is an improved version of the SCAT code, reviewed and approved by the NRC, with advanced fuel features (partial length rods and new critical power correlation) capability.

Response Part c:

EOI has not performed an audit per se, but rather an engineering review of the results. This review focused mainly on the adequacy of the inputs used, the reasonableness of the results obtained, and the applicability of generic analyses to RBS. The analyses reviewed were the containment analysis, transient analysis, and the ATWS analysis. The ECCS/LOCA analysis is excluded from this list as the current analysis was performed using the GE developed, NRC approved SAFER/GESTR methodology (NEDE-23785-1-PA) and included sufficient margin to bound power uprate.

EOI was involved at every stage of the safety analyses performed to support the power uprate submittal. The involvement started with the development of the calculation plan for the analyses, the development of the inputs, and concluding with the review and approval of the analysis reports.

With respect to the transient analysis, the methods and programs used in the evaluation are the same as those currently used to support plant operations and licensing. Here the review focused on the input values used, and the reasonableness of the results. In the case of the generic loss of all feedwater analysis, the inputs for the reference BWR/6 plant were reviewed to ensure that the analysis was applicable to RBS.

With respect to the containment analysis, the methods and programs used for the power uprate evaluations differ from those supporting current power operations. In these cases, GE was requested to perform benchmark runs at the current power level so that comparisons between the AE methods and GE methods. This benchmarking is discussed in the power uprate safety analysis report. Here the review focused not only the inputs used and the reasonableness of the results but also the benchmark to the current analysis results.

With respect to the anticipated transient without scram (ATWS) analysis, the methods and programs used for the power uprate evaluations differ from the methods used for previous ATWS evaluations. Previous ATWS evaluations have used the approach outlined in GE Report NEDE-24222, "Assessment of BWR Mitigation of ATWS", which include the use of the GE developed transient thermal-hydraulic computer program REDY. Since this earlier work, the GE developed, ODYN computer program has been qualified for use in ATWS analyses (NEDC-24154-P, Supplement 1, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors; MFN-019-98, "Revision to ODYN ATWS Qualification Report, Sections 3.1, 4.2, and 5.1.3", including NRC Safety Evaluation Report TAC No. MA3478. Here the review focused not only the inputs used and the reasonableness of the results but also a check was made to ensure that the limitations identified in the qualification documentation and the associated safety evaluation report were addressed in the analysis.

It should be noted that a number of questions were raised by EOI during the review of each of these analyses. As part of the review process, GE has provided responses to each of the EOI questions.

3. In Section 3.4 Reactor Coolant System (pg. 3.4-10); how did the licensee get the SRV setpoints. What SRV setpoints did he use in his overpressure analysis?

   EOI Response:

   The actual setpoints of the SRVs were established as follows:
   - The need to maintain the same margin between the operating reactor pressure and the minimum setpoint of the lower group of SRV safety setpoints, and
   - The need to maintain the upper group of SRV safety setpoints (with tolerance) below the vessel ASME code limit of 1250 psig.

   Therefore, the lower group of SRV setpoints was increased by 30 psi, and the upper two groups of SRV setpoints were increased by 25 and 20 psi, respectively. The grouping of the valves remains as originally designed by number in each group and location of each group.

   The SRV setpoints used in the vessel overpressure analysis were the safety function settings proposed in LAR 99-15, dated July 30, 1999.

4. The submittal included proposed changes to the Technical Specifications. However, the submittal did not provide any matrix or plan indicating which sections of the Updated Safety Analysis Report (USAR) will be superceded by current power uprate analysis. Provide a list or matrix that identifies which subsections of the USAR will be superceded and identify the corresponding sections of the current submittal. The actual updating of the USAR will be governed by the current regulations, however, a new license at the uprated power will be issued and the effected USAR should be documented.

   EOI Response:

   A copy of the USAR has been marked up with changes and the USAR will be updated to be current with the appropriate changes as the different phases of Power Uprate are implemented through the modification process at RBS. All USAR changes will be submitted as part of our periodic update in accordance with 10CFR 50.71(e).

   Attachment 3 - Power Uprate USAR Changes Matrix is a working tool that is being used to track the changes to the USAR and the source documents affecting the changes. We'll be using the matrix when we submit the LCNs for the USAR changes. This matrix is based on significant changes to the RBS USAR that resulted from the GE SAR.

5. Why did the reactor Steam Dome pressure only go from 1050 psig to 1059 psig while other pressures went up 30 psig?

   EOI Response:

   The normal reactor steam dome pressure increased from 1025 psig to 1055 psig for power uprate, and the evaluation considered dome pressures up to 1059 psig at 102% of the uprated power. The current Technical Specification control rod surveillance scram time requirements is applicable for reactor steam dome pressures from 950-1050 psig. Since the scram performance and the requirements are already known for a dome pressure of 1050 psig, the effects of the incremental change in the dome pressure from 1050 psig to 1059 psig were evaluated.

6. The charging water header pressure went from 1520 psig to 1540 psig. Why did it not go up 30 psig? What is the basis for this change?

   EOI Response:

   The Technical Specification control rod surveillance scram time requirements for 100% of the original rated thermal power applies to reactor steam dome pressures of 950-1050 psig. For power uprate, the reactor steam dome pressure range is increased to 950-1059 psig, which is a 9-psi increase for the current scram performance requirement at 1050 psig. The 20 psig increase in the minimum charging water pressure (and scram accumulator pressure) is sufficient to offset the reactor steam dome pressure increase of 9 psig and to maintain the scram performance margin that existed at 1050 psig, based upon scram performance predictions.

7. The SLC system boron concentration and enrichment was changed. Why was this change required?

   EOI Response:

   The boron [concentration * enrichment] product was changed for power uprate conditions to ensure that the peak suppression pool temperature remains below the allowable value of 185°F. Increasing the [concentration * enrichment] product reduces the time to reactor shutdown and mitigates the suppression pool temperature response by reducing the integrated steam flow to the suppression pool. This change was required for the following reasons:

   a. Power uprate produces an increase in the integrated steam flow to the suppression pool for uprated conditions.
   b. The analysis methodology was changed from REDY to ODYN. ODYN provides a more conservative result than REDY for two reasons:
      (1) Several features used in the previous REDY water level control model (water level control band, water injection systems used) are not consistent with current plant emergency operating procedures. The ODYN code provides more accurate modeling of the reactor water level

control strategy and generally produces a more limiting peak suppression pool temperature response than REDY.

(2) A conservative boron mixing threshold has been specified in ODYN to ensure that the ODYN calculated peak suppression pool temperature is bounding relative to a best-estimate methodology. The boron mixing threshold used in ODYN is more conservative than was previously used in REDY.

c. The [concentration * enrichment] product was increased for power uprate to provide margin in the plant design to the 185°F limit on peak suppression pool temperature.

Prior to power uprate, River Bend did not have significant margin to the 185°F limit on peak suppression pool temperature. Therefore, the changes identified above which increased the calculated suppression pool temperature, could not be accommodated without a plant design change. The model changes and provision for additional margin resulted in an increase in the [concentration * enrichment] product that was larger than would be anticipated due to the power level increase alone.

8. Will the SLC relief valve setpoint change or will the setting stay the same? What is the current setpoint of the SLC relief valve?

EOI Response:

The increase in the maximum pump discharge pressure from 1217 to 1247 psig for the 5% power uprate does not require any change to the nominal setpoint for the SLC pump relief valves (1400-psi). Adequate pressure margin remains between the maximum pump discharge pressure and the nominal setpoint for the SLC pump relief valve (153 psi) to allow the relief valve to remain tightly closed during system operation at maximum pump discharge pressure.

9. Section 9B: Is the information contained in this section all based on GE report NEDC-32778P as it relates to the SRV setpoint tolerance change?

EOI Response:

The information contained in Section 9B is based on requirements from NEDC-31753P, BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report, and the NRC SER on that LTR. Section 9B sub-items 1 and 2 summarize the analyses performed to support NEDC-32778P, Safety Analysis Report for River Bend 5% Power Uprate.

10. Does RBS have an all GE supplied (fuel) core? Will RBS continue to use an all GE supplied core at power uprate conditions?

EOI Response:

All fuel in the current RBS core is supplied by GE. The power uprate submittal addresses the impact of the power uprate on the plant and establishes a new licensing basis which reflects the higher power, different operating conditions, setpoint changes, etc. associated with the uprate. RBS is currently scheduled to transition to Sieman Power Corporation (SPC) fuel starting in Cycle 11. RBS will make the required submittals to the NRC for the transition to Siemens Fuel (e.g. Changes to technical specifications, including 5.6.5, to reflect use of Siemens methodology) on this basis.

11. What BWR is the operating power density range within? Page 2-1, section 2.1, (second sentence)

    EOI Response:

    River Bend is currently operating at a power density of 53.9 kW/l based on 146-inch active fuel length. The power density for the 105% power uprate is 56.6 kW/l for the same active fuel length. Other BWR(s) operating in the same power density range are Plant Hatch Units 1 and 2. They are currently operating at a power density of 57.3 kW/l based on 146 inch active fuel length (113.4% EPU).

12. Does RBS have any additional analysis in addition to the documentation submitted with the power uprate submittal regarding SRV setpoint tolerance?

    EOI Response:

    No. RBS does not have any additional information or documentation that it plans to submit in support of the SRV setpoint tolerance change.

13. What are RBS's normal practices during refueling relative to core off load?

    EOI Response:

    RBS does not make a normal practice of full core off loads during refueling periods. RBS has performed only one full core offload in its history. That was during RF-4 when a major Service Water System modification was implemented which affected both divisions of RHR.

    Core shuffles are much more efficient at RBS than full core offloads. Core shuffles are the normal and preferred method of fuel storage/handling at RBS.

Attachment 1

MOVs Affected by Power Uprate

Valve Number DBR Calc System	B21-MOVF067A G13.18.2.3*107, Rev 1 MSS		B21-MOVF067B G13.18.2.3*107, Rev 1 MSS		B21-MOVF067C G13.18.2.3*107, Rev 1 MSS		B21-MOVF067D G13.18.2.3*107, Rev 1 MSS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
MEDP (C)	1,178.0	1,244.0	1,178.0	1244.0	1,178.0	1,244.0	1,178.0	1,244.0
DP Load (O)	-1,527.8	-1,614.4	-1,527.8	-1,614.4	-1,527.8	-1,614.4	-1,527.8	-1,614.4
DP Load (C)	1,593.3	1,682.6	1,593.3	1,682.6	1,593.3	1,682.6	1,593.3	1,682.6
Pmax (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
Pmax (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0
Piston Effect (O)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Piston Effect (C)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Min. Req'd Thrust (O)	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8
Min. Req'd Thrust (C)	3,215.7	3,305.0	3,215.7	3,305.0	3,215.7	3,305.0	3,215.7	3,305.0
TST Min (O)	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8	1,622.4	1,535.8
TST Min (C)	4,707.8	4,838.5	4,643.5	4,772.4	4,643.5	4,772.4	4,701.4	4,831.9
Thrust Max (O)	12,684.0	N/C	12,726.0	N/C	12,726.0	N/C	12,726.0	N/C
Thrust Max (C)	12,558.0	N/C	12,586.0	N/C	12,586.0	N/C	12,586.0	N/C
Q Min (O)	17.2	16.3	17.2	16.3	17.2	16.3	17.2	16.3
Q Min (C)	34.1	35.0	34.1	35.0	34.1	35.0	34.1	35.0
Q Max (O)	48.0	N/C	79.9	N/C	79.9	N/C	48.0	N/C
Q Max (C)	48.0	N/C	79.9	N/C	79.9	N/C	48.0	N/C
C14 Thrust	5,778.0	N/A	6,641.8	N/A	9,050.0	N/A	7,355.0	N/A
Margin								
Open Thrust	11,061.6	11,148.2	11,103.6	11,190.2	11,103.6	11,190.2	11,103.6	11,190.2
Open Torque	30.8	31.7	62.7	63.6	62.7	63.6	30.8	31.7
Close Thrust	7,850.2	7,719.5	7,942.5	7,813.6	7,942.5	7,813.6	7,884.6	7,754.1
Close Torque	13.9	13.0	45.8	44.9	45.8	44.9	13.9	13.0
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	OK	-	OK

Valve Number DBR Calc System	B21-MOVF016 G13.18.2.3*108, Rev 1 MSS		B21-MOVF019 G13.18.2.3*108, Rev 1 MSS		B21-MOVF085 G13.18.2.3*108, Rev 1 MSS		E51-MOVF013 G13.18.2.3*198, Rev. 2 ICS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	1,178.0	1,244.0	1,178.0	1,244.0	1,178.0	1,244.0	1,306.6	1,336.6
MEDP (C)	1,178.0	1,244.0	1,178.0	1,244.0	1,178.0	1,244.0	0.0	0.0
DP Load (O)	4,462.7	4,712.7	4,462.7	4,712.7	4,462.7	4,712.7	23,196.8	23,729.4
DP Load (C)	4,478.9	4,729.8	4,478.9	4,729.8	4,478.9	4,729.8	0.0	0.0
Pmax (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,366.6	1,396.6
Pmax (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,366.6	1,396.6
Piston Effect (O)	-1,158.0	-1,223.6	-1,158.0	-1,223.6	-1,158.0	-1,223.6	-3,287.1	-3,359.3
Piston Effect (C)	1,158.0	1,223.6	1,158.0	1,223.6	1,158.0	1,223.6	3,287.1	3,359.3
Min. Req'd Thrust (O)	4,310.1	4,494.5	4,310.1	4,494.5	4,404.7	4,589.1	22,019.8	22,480.2
Min. Req'd Thrust (C)	6,642.3	6,958.8	6,642.3	6,958.8	6,736.9	7,053.4	5,397.2	5,469.4
TST Min (O)	4,310.1	4,494.5	4,310.1	4,494.5	4,404.7	4,589.1	22,019.8	22,480.2
TST Min (C)	9,458.6	9,909.4	9,458.6	9,909.4	9,593.3	10,044.1	8,090.4	8,198.6
Thrust Max (O)	8,561.0	N/C	8,561.0	N/C	8,561.0	N/C	27,937.2	N/C
Thrust Max (C)	13,035.5	N/C	13,035.5	N/C	13,035.5	N/C	30,206.4	N/C
Q Min (O)	50.0	52.1	50.0	52.1	51.1	53.2	367.7	375.4
Q Min (C)	77.1	80.8	77.1	80.8	78.1	81.8	90.1	91.3
Q Max (O)	185.0	N/C	155.8	N/C	145.0	N/C	319.8	N/C
Q Max (C)	185.0	N/C	155.8	N/C	145.0	N/C	319.8	N/C
C14 Thrust	11,705.4	N/A	12,526.8	N/A	10,810.5	N/A	15,207.0	N/A
Margin								
Open Thrust	4,250.9	4,066.5	4,250.9	4,066.5	4,156.3	3,971.9	5,917.4	5,457.0
Open Torque	135.0	132.9	105.8	103.7	93.9	91.8	Neg	Neg
Close Thrust	3,576.9	3,126.1	3,576.9	3,126.1	3,442.2	2,991.4	22,116.0	22,007.8
Close Torque	107.9	104.2	78.7	75.0	66.9	63.2	229.7	228.5
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	OK	-	OK

Valve Number	E51-MOVF019		E51-MOVF022		E51-MOVF045		E51-MOVF059	
DBR Calc	G13.18.2.3*199, Rev. 4		G13.18.2.3*200, Rev. 2		G13.18.2.3*202, Rev. 1		G13.18.2.3*204, Rev. 4	
System	ICS		ICS		ICS		ICS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	1,363.0	1,393.0	1,351.0	1,381.0	1,165.0	1,231.0	0.0	0.0
MEDP (C)	1,363.0	1,393.0	1,290.0	1,318.1	1,165.0	1,231.0	1,165.0	1,231.0
DP Load (O)	-2,938.6	-3,003.3	-9,228.1	-9,433.0	-7,611.6	-8,042.8	0.0	0.0
DP Load (C)	2,942.7	3,007.5	8,811.4	9,003.3	7,626.6	8,058.7	9,082.0	9,596.5
Pmax (O)	1,363.0	1,393.0	1,374.5	1,404.5	1,165.0	1,231.0	0.0	0.0
Pmax (C)	1,363.0	1,393.0	1,313.5	1,341.6	1,165.0	1,231.0	1,188.5	1,254.5
Piston Effect (O)	0.0	0.0	-34.9	-35.7	0.0	0.0	0.0	0.0
Piston Effect (C)	0.0	0.0	34.9	35.6	0.0	0.0	1,764.8	1,862.8
Min. Req'd Thrust (O)	2,660.0	2,595.3	1,040.0	834.3	1,497.6	1,066.4	1,300.0	1,300.0
Min. Req'd Thrust (C)	5,602.7	5,667.5	9,886.3	10,079.0	9,124.2	9,556.3	12,146.8	12,759.3
TST Min (O)	2,660.0	2,595.3	1,040.0	834.3	1,497.6	1,066.4	1,300.0	1,300.0
TST Min (C)	8,398.4	8,495.5	14,819.6	15,108.4	13,868.8	14,525.5	14,066.0	14,775.3
Thrust Max (O)	12,726.0	N/C	21,816.0	N/C	21,816.0	N/C	21,816.0	N/C
Thrust Max (C)	12,586.0	N/C	21,576.0	N/C	21,576.0	N/C	21,576.0	N/C
Q Min (O)	38.6	37.7	12.0	9.6	15.6	11.1	10.7	10.7
Q Min (C)	81.2	82.1	113.7	115.9	94.9	99.4	99.6	104.6
Q Max (O)	62.0	N/C	213.4	N/C	141.9	N/C	120.7	N/C
Q Max (C)	62.0	N/C	213.4	N/C	141.9	N/C	120.7	N/C
C14 Thrust	10,086.7	N/A	18,626.2	N/A	14,191.0	N/A	14,575.0	N/A
Margin								
Open Thrust	10,066.0	10,130.7	20,776.0	20,981.7	20,318.4	20,749.6	20,516.0	20,516.0
Open Torque	23.4	24.3	201.4	203.8	126.3	130.8	110.0	110.0
Close Thrust	4,187.6	4,090.5	6,756.4	6,467.6	7,707.2	7,050.5	7,510.0	6,800.7
Close Torque	Neg	Neg	99.7	97.5	47.0	42.5	21.1	16.1
Setup Margin								
TST min < C14 Thrust?	-	OK	-	OK	-	No	-	No

Valve Number DBR Calc System	E51-MOVF063 G13.18.2.3*206, Rev. 2 ICS		E51-MOVF064 G13.18.2.3*206, Rev. 2 ICS		E51-MOVF076 G13.18.2.3*208, Rev. 2 ICS		G33-MOVF001 G13.18.2.3*215, Rev 3 WCS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	0.0	0.0	0.0	0.0	1,165.0	1,231.0	0.0	0.0
MEDP (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
DP Load (O)	0.0	0.0	0.0	0.0	-393.2	-415.5	0.0	0.0
DP Load (C)	32,021.8	33,835.9	32,021.8	33,835.9	393.2	415.5	17,559.9	18,544.8
Pmax (O)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
Pmax (C)	1,165.0	1,231.0	1,165.0	1,231.0	1,165.0	1,231.0	1,176.0	1,242.0
Piston Effect (O)	-2,416.1	-2,553.0	-2,416.1	-2,553.0	0.0	0.0	-2,078.2	-2,194.8
Piston Effect (C)	2,416.1	2,553.0	2,416.1	2,553.0	0.0	0.0	2,078.2	2,194.8
Min. Req'd Thrust (O)	3,000.0	2,863.1	3,000.0	2,863.1	741.7	719.4	1,743.3	1,626.7
Min. Req'd Thrust (C)	37,437.9	39,388.9	37,437.9	39,388.9	1,134.9	1,157.2	21,381.4	22,482.8
TST Min (O)	3,000.0	2,863.1	3,000.0	2,863.1	741.7	719.4	1,743.3	1,626.7
TST Min (C)	52,787.4	55,538.3	52,787.4	55,538.3	1,725.0	1,758.9	30,147.8	31,700.8
Thrust Max (O)	62,750.0	N/C	62,750.0	N/C	3,981.4	N/C	35,274.0	N/C
Thrust Max (C)	62,750.0	N/C	62,750.0	N/C	4,611.3	N/C	56,988.0	N/C
Q Min (O)	43.8	41.8	43.8	41.8	5.0	4.8	27.5	25.7
Q Min (C)	546.6	575.1	546.6	575.1	7.7	7.9	337.8	355.2
Q Max (O)	605.5	N/C	491.4	N/C	24.3	N/C	293.4	N/C
Q Max (C)	605.5	N/C	491.4	N/C	24.3	N/C	293.4	N/C
C14 Thrust	8,876.8	N/A	8,633.6	N/A	2,684.9	N/A	13,971.0	N/A
Margin								
Open Thrust	59,750.0	59,886.9	59,750.0	59,886.9	3,239.7	3,262.0	33,530.7	33,647.3
Open Torque	561.7	563.7	447.6	449.6	19.3	19.5	265.9	267.7
Close Thrust	9,962.6	7,211.7	9,962.6	7,211.7	2,886.3	2,852.4	26,840.2	25,287.2
Close Torque	58.9	30.4	Neg	Neg	16.6	16.4	Neg	Neg
Setup Margin								
TST min < C14 Thrust?	-	No	-	No	-	OK	-	No

Valve Number DBR Calc System	G33-MOVF004 G13.18.2.3*216, Rev 3 WCS		G33-MOVF039 G13.18.2.3*220, Rev 2 WCS		G33-MOVF040 G13.18.2.3*220, Rev 2 WCS		G33-MOVF053 G13.18.2.3*223, Rev 1 WCS	
	Current	Uprate	Current	Uprate	Current	Uprate	Current	Uprate
MEDP (O)	0.0	0.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
MEDP (C)	1,176.0	1,242.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
DP Load (O)	0.0	0.0	0.0	0.0	0.0	0.0	9,160.7	9,674.9
DP Load (C)	17,962.4	18,969.8	0.0	0.0	0.0	0.0	9,184.9	9,700.4
Pmax (O)	1,176.0	1,242.0	0.0	0.0	0.0	0.0	1,175.1	1,241.1
Pmax (C)	1,176.0	1,242.0	1,433.8	1,499.8	1,433.8	1499.79	1,175.1	1,241.1
Piston Effect (O)	-2,078.2	-2,194.8	0.0	0.0	0.0	0.0	-1,744.9	-1,842.8
Piston Effect (C)	2,078.2	2,194.8	3,448.7	3,607.4	3,448.7	3,607.4	1,744.9	1,842.8
Min. Req'd Thrust (O)	1,743.0	1,626.4	1,923.9	1,923.9	1,923.9	1,923.9	8,215.8	8,632.0
Min. Req'd Thrust (C)	21,783.6	22,907.6	5,372.6	5,531.3	5,372.6	5,531.3	11,729.8	12,343.3
TST Min (O)	1,743.0	1,626.4	1,923.9	1,923.9	1,923.9	1,923.9	8,215.8	8,632.0
TST Min (C)	30,714.9	32,299.7	7,758.0	7,987.2	7,758.0	7,987.2	16,703.2	17,576.8
Thrust Max (O)	35,274.0	N/C	21,330.6	N/C	21,330.6	N/C	17,554.6	N/C
Thrust Max (C)	56,988.0	N/C	21,095.9	N/C	21,095.9	N/C	21,576.0	N/C
Q Min (O)	27.5	25.7	40.4	40.4	40.4	40.4	123.2	129.4
Q Min (C)	344.2	362.0	112.8	116.1	112.8	116.1	175.9	185.1
Q Max (O)	286.6	N/C	314.0	N/C	445.9	N/C	230.0	N/C
Q Max (C)	286.6	N/C	314.0	N/C	445.9	N/C	230.0	N/C
C14 Thrust	8,191.0	N/A	18,308.0	N/A	17,962.0	N/A	17,233.0	N/A
Margin								
Open Thrust	33,531.0	33,647.6	19,406.7	19,406.7	19,406.7	19,406.7	9,338.8	8,922.6
Open Torque	259.1	260.9	273.6	273.6	405.5	405.5	106.8	100.6
Close Thrust	26,273.1	24,688.3	13,337.9	13,108.7	13,337.9	13,108.7	4,872.8	3,999.2
Close Torque	Neg	Neg	201.2	197.9	333.1	329.8	54.1	44.9
Setup Margin								
TST min < C14 Thrust?	-	No	-	OK	-	OK	-	No

Valve Number DBR Calc System	G33-MOVF054 G13.18.2.3*223, Rev 1 WCS		B21-MOVFO86 G13.18.2.3*109, Rev.1C MSS	
	Current	Uprate	Current	Uprate
MEDP (O)	1,175.1	1,241.1	0.0	0.0
MEDP (C)	1,175.1	1,241.1	0.0	0.0
DP Load (O)	9,160.7	9,674.9	0.0	0.0
DP Load (C)	9,184.9	9,700.4	0.0	0.0
Pmax (O)	1,175.1	1,241.1	0.0	1055.0
Pmax (C)	1,175.1	1,241.1	0.0	1055.0
Piston Effect (O)	-1,744.9	-1,842.8	0.0	-1048.7
Piston Effect (C)	1,744.9	1,842.8	0.0	1048.7
Min. Req'd Thrust (O)	8,215.8	8,632.0	1005.4	500
Min. Req'd Thrust (C)	11,729.8	12,343.3	1005.5	1548.7
TST Min (O)	8,215.8	8,632.0	1005.4	500
TST Min (C)	16,703.2	17,576.8	1005.4	2256.5
Thrust Max (O)	17,554.6	N/C	11779.0	10801.3
Thrust Max (C)	21,576.0	N/C	11779.0	10683.6
Q Min (O)	123.2	129.4	11.7	5.2
Q Min (C)	175.9	185.1	11.7	16
Q Max (O)	224.4	N/C	62	62
Q Max (C)	224.4	N/C	62	62
C14 Thrust	15,862.0	N/A	3587.0	7044.5
Margin				
Open Thrust	9,338.8	8,922.6	10773.6	10301.3
Open Torque	101.2	95.0	50.3	56.8
Close Thrust	4,872.8	3,999.2	10773.6	9134.9
Close Torque	48.5	39.3	50.3	46
Setup Margin				
TST min < C14 Thrust?	-	No	OK	OK

Attachment 2

Pre HWC/Power Uprate vs. Post HWC/Uprate Normal
and Worse Case Accident Radiation Doses.

Table 1 -Comparison of Pre HWC/ Power Uprate VS Post HWC/ Power Uprate Normal Radiation Doses

EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Neutron Fluence (η/cm^2)	
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate
DW-1	7.5E7	8.8E7	5.0E4	No Change	2.2E15	2.2E15
DW-2	1.1E8	1.2E8	5.0E4		8.9E15	9.0E15
DW-3	3.7E7	5E7	5.0E4		1.0E14	1.0E14
DW-4	2.8E7	2.8E7	5.0E4		1.0E15	1.0E15
DW-5	6.4E7	7.8E7	5.0E4		1.8E15	1.9E15
DW-6	2.7E10	2.7E10	5.0E4		6.5E17	6.5E17
CT-1	9.0E2	9.0E2	2.0E3		-	-
CT-2	2.0E3	2.0E3	2.0E3		-	-
CT-3	4.0E4	4.1E4	2.0E3		-	-
CT-4	9.0E2	9.0E2	2.0E3		-	-
CT-5	1.2E8	1.2E8	2.0E3		-	-
CT-5A	8.0E3	8.0E3	2.0E3		-	-
CT-6	1.6E8	8.8E7	2.0E3		1.0E15	2.2E15
CT-7	1.0E6	1.0E6	2.0E3		-	-
CT-7A	9.0E5	9.0E5	2.0E3		-	-
CT-8	3.6E9	3.6E9	2.0E3		-	-
CT-9	2.5E7	2.6E7	2.0E3		2.7E13	2.7E13
CT-10	1.1E6	1.2E6	2.0E3		-	-
CT-11	1.6E8	1.6E8	2.0E3		-	-
CT-SP	2.0E3	2.0E3	2.0E3		-	-
CT-G	9.0E5	9.0E5	2.0E3		-	-
AN-1	3.2E7	8.3E7	0		-	-
AN-2	9.0E5	9.0E5	0		-	-
AN-3	1.1E6	1.2E6	0		-	-
AB-070-1	7.0E3	7.0E3	0		-	-
AB-070-2	2.0E5	2.0E5	0		-	-
AB-070-3	3.0E4	4.0E4	0		-	-
AB-070-4	2.0E3	7.0E3	0		-	-
AB-070-5	2.0E5	2.0E5	0		-	-
AB-070-6	6.0E3	6.0E3	0		-	-
AB-070-7	2.0E5	2.0E5	0		-	-
AB-070-8	2.0E5	2.0E5	0		-	-
AB-070-G	700	800	0		-	-
AB-095-1	700	800	0		-	-
AB-095-2	2.0E5	2.0E5	0		-	-
AB-095-3	3.3E7	3.3E7	0		-	-
AB-095-4	3.3E7	3.3E7	0		-	-
AB-095-5	2.0E5	2.0E5	0		-	-

Table 1 -Comparison of Pre HWC/ Power Uprate VS Post HWC/ Power Uprate Normal Radiation Doses (Continued)

EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Neutron Fluence (η/cm^2)	
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate
AB-095-6	700	800	0	No Change	-	-
AB-095-7	2.0E5	2.0E5	0		-	-
AB-095-8	2.0E5	2.0E5	0		-	-
AB-095-9	9.0E6	9.0E6	0		-	-
AB-095-10	3.2E7	3.3E7	0		-	-
AB-095-G	8.0E2	8.0E2	0		-	-
AB-114-1	700	800	0		-	-
AB-114-2	3.2E7	3.3E7	0		-	-
AB-114-3	700	7.9E3	0		-	-
AB-114-4	700	7.9E3	0		-	-
AB-114-5	2.0E5	2.0E5	0		-	-
AB-114-6	2.0E5	2.0E5	0		-	-
AB-114-7	3.2E7	3.3E7	0		-	-
AB-114-8	2.0E5	2.0E5	0		-	-
AB-114-G	700	800	0		-	-
AB-141-1	700	800	0		-	-
AB-141-2	700	800	0		-	-
AB-141-3	3.1E3	7.9E3	0		-	-
AB-141-4	700	800	0		-	-
AB-141-5	700	800	0		-	-
AB-141-6	700	800	0		-	-
AB-141-G	700	800	0		-	-
AB-170-1	700	800	0		-	-
AB-170-2	7.0E3	7.0E3	0		-	-
AB-170-3	700	800	0		-	-
AB-170-G	700	800	0		-	-
FB-070-1	1.8E6	1.8E6	0		-	-
FB-070-2	9.0E5	9.0E5	0		-	-
FB-070-3	6.3E7	6.3E7	0		-	-
FB-070-4	8.3E7	8.3E7	0		-	-
FB-070-G	700	800	0		-	-
FB-095-1	8.3E7	8.3E7	0		-	-
FB-095-2	1.8E6	1.8E6	0		-	-
FB-095-G	9.0E5	9.0E5	0		-	-
FB-113-1	7.0E4	8.0E4	0		-	-
FB-113-2	700	800	0		-	-
FB-113-3	1.1E6	1.2E6	0		-	-

Table 1 -Comparison of Pre HWC/ Power Uprate VS Post HWC/ Power Uprate Normal Radiation Doses (Continued)

EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Neutron Fluence (η/cm^2)	
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate
FB-113-4	3.6E9	3.6E9	0	0	-	-
FB-113-G	9.0E5	9.0E5	0		-	-
FB-131-1	1.1E6	1.2E6	0		-	-
FB-148-1	700	800	0		-	-
FB-148-2	700	800	0			
FB-148-G	700	800	0		-	-
OG-067-1	2.4E7	6.5E7	0		-	-
OG-095-1	1.1E7	3.1E7	0		-	-
OG-095-G	7.0E3	8.0E3	0		-	-
OG-123-1	2.4E7	2.4E7	0		-	-
OG-123-2	8.0E6	9.1E6	0		-	-
OG-123-3	5.9E8	5.9E8	0		-	-
OG-148-1	5.9E8	5.9E8	0		-	-
OG-148-2	5.8E7	1.8E8	0		-	-
OG-148-3	3.5E8	3.5E8	0		-	-
PT-1	7.0E2	8.0E2	0		-	-
PT-2	9.0E5	9.0E5	0		-	-
PT-3	9.0E5	9.0E5	0		-	-
PT-4	7.0E3	8.0E2	0		-	-
PT-6	7.00E3	8.0E3	0		-	-
PT-7	2.4E7	6.5E7	0		-	-
PT-8	7.00E02	8.00E02	0		-	-
PT-9	7.00E02	8.00E02	0		-	-

Table 2 -Worst-Case Accident Environmental Conditions by EQ Zone

EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Pre HWC & Power Uprate TID (rad γ)	Post HWC & Power Uprate TID(rad γ)
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate		
DW-1	2.3E7	2.4E7	5.3E8	4E8	6.37E8	5.21E8
DW-2	2.3E7	2.4E7	5.3E8	4E8	7.00E8	5.82E8
DW-3	2.5E7	2.6E7	5.3E8	4E8	5.92E8	4.76E8
DW-4	2.3E7	2.4E7	5.3E8	4E8	5.85E8	4.56E8
DW-5	2.5E7	2.6E7	5.3E8	4E8	6.27E8	5.12E8
DW-6	2.3E7	2.4E7	5.3E8	4E8	3.03E10	2.74E10
CT-1	1.2E7	1.3E7	1.4E8	1.5E8	1.52E8	1.63E8
CT-2	1.9E7	2.0E7	1.4E8	1.5E8	1.59E8	1.7E8
CT-3	1.3E7	1.4E7	1.4E8	1.5E8	1.53E8	1.64E8
CT-4	1.2E7	1.3E7	1.4E8	1.5E8	1.52E8	1.63E8
CT-5	9.0E6	8.4E6	1.4E8	1.5E8	2.69E8	2.78E8
CT-5A	1.5E7	1.6E7	1.4E8	1.5E8	1.55E8	1.66E8
CT-6	7.0E6	6.7E6	1.4E8	1.5E8	3.11E8	2.54E8
CT-7	7.0E6	7.4E6	1.4E8	1.5E8	1.48E8	1.58E8
CT-7A	1.5E7	1.6E7	1.4E8	1.5E8	1.56E8	1.67E8
CT-8	7.0E6	6.7E6	1.4E8	1.5E8	1.72E8	3.76E9
CT-9	1.0E7	1.1E7	1.4E8	1.5E8	1.75E8	1.87E8
CT-10	7.0E6	6.7E6	1.4E8	1.5E8	1.48E8	1.58E8
CT-11	9.0E6	8.4E6	1.4E8	1.5E8	3.09E8	3.18E8
CT-SP	2.1E7	2.2E7	1.4E8	1.5E8	1.6E8	1.72E8
CT-G	1.7E7	1.8E7	1.4E8	1.5E8	1.58E8	1.69E8
AN-1	6.0E6	5.9E6	2.0E5	8.20E4	3.82E7	8.9E7
AN-2	5.0E6	5.5E6	2.0E5	8.20E4	6.10E6	6.48E6
AN-3	9.0E5	9.3E5	2.0E5	8.20E4	9.20E6	2.21E6
AB-070-1	8.0E6	8.6E6	500	3.00E3	8.01E6	8.61E6
AB-070-2	1.1E7	1.2E7	500	3.00E3	1.12E7	1.22E7
AB-070-3	5.0E6	4.8E6	500	3.00E3	5.03E6	4.83E6
AB-070-4	8.0E6	8.5E6	500	3.00E3	8.00E6	8.51E6
AB-070-5	1.1E7	1.2E7	500	3.00E3	1.12E7	1.22E7
AB-070-6	9.0E6	9.2E6	500	3.00E3	9.01E6	9.21E6
AB-070-7	1.0E7	1.1E7	500	3.00E3	1.02E7	1.12E7

Table 2 -Worst-Case Accident Environmental Conditions By EQ Zone (Continued)						
EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Pre HWC & Power Uprate TID(rad γ)	Post HWC & Power Uprate TID(rad γ)
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate		
AB-070-8	9.0E6	9.1E6	500	3.00E3	9.20E6	9.3E6
AB-070-G	1.0E7	1.1E7	500	3.00E3	1.00E7	1.1E7
AB-095-1	7.0E6	7.0E6	500	3.00E3	7.00E6	7.00E6
AB-095-2	1.1E7	1.2E7	500	3.00E3	1.12E7	1.22E7
AB-095-3	6.0E5	6.5E5	500	3.00E3	3.36E7	3.37E7
AB-095-4	4.0E6	4.1E6	500	3.00E3	3.70E7	3.71E7
AB-095-5	1.1E7	1.2E7	500	3.00E3	1.12E7	1.22E7
AB-095-6	7.0E6	7.7E6	500	3.00E3	7.00E6	7.7E6
AB-095-7	1.0E7	1.1E7	500	3.00E3	1.02E7	1.12E7
AB-095-8	8.0E6	8.6E6	500	3.00E3	8.20E6	8.8E6
AB-095-9	7.0E6	7.0E6	500	3.00E3	1.60E7	1.6E7
AB-095-10	60	620	500	3.00E3	3.20E7	3.3E7
AB-095-G	7.0E6	7.5E6	500	3.00E3	7.00E6	7.5E6
AB-114-1	5.0E6	5.7E6	500	3.00E3	5.00E6	5.7E6
AB-114-2	4.0E6	4.1E6	500	3.00E3	3.60E7	3.71E7
AB-114-3	3.0E5	1.2E6	500	3.00E3	3.01E5	1.21E6
AB-114-4	8.0E3	8.1E5	500	3.00E3	9.55E3	8.21E5
AB-114-5	7.0E6	7.0E6	500	3.00E3	7.20E6	7.20E6
AB-114-6	6.0E6	5.8E6	500	3.00E3	6.20E6	6.0E6
AB-114-7	30	600	500	3.00E3	3.20E7	3.3E7
AB-114-8A&B	7.0E6	7.0E6	500	3.00E3	7.20E6	7.20E6
AB-114-G	3.0E5	1.2E6	500	3.00E3	3.02E5	1.2E6
AB-141-1	2.0E4	2.6E3	500	3.00E3	3.55E3	6.40E3
AB-141-2	2.0E4	2.9E3	500	3.00E3	3.55E3	6.70E3
AB-141-3	1.0E6	1.4E6	500	3.00E3	1.00E6	1.41E6
AB-141-4	1.0E6	1.5E6	500	3.00E3	1.00E6	1.5E6
AB-141-5	4.4E7	4.4E7	500	3.00E3	4.40E7	4.40E7
AB-141-6	4.4E7	4.4E7	500	3.00E3	4.40E7	4.40E7
AB-141-G	1.0E4	1.1E3	500	3.00E3	1.16E4	4.90E3
AB-170-1	2.0E6	1.4E6	500	3.00E3	2.00E6	1.4E6
AB-170-2	300	800	500	3.00E3	8.15E3	1.08E4
AB-170-3	2.0E6	1.4E6	500	3.00E3	2.00E6	1.40E6
AB-170-G	30	600	500	3.00E3	1.61E3	4.40E3

Table 2 -Worst-Case Accident Environmental Conditions By EQ Zone (Continued)						
EQ Zone	Gamma Dose (rad)		Beta Dose (rad)		Pre HWC & Power Uprate TID(rad γ)	Post HWC & Power Uprate TID(rad γ)
	Pre HWC & Power Uprate	Post HWC & Power Uprate	Pre HWC & Power Uprate	Post HWC & Power Uprate		
FB-070-1	40	79	600	1300	1.80E6	1.80E6
FB-070-2	800	870	600	1300	9.02E5	9.02E5
FB-070-3	40	79	600	1300	6.30E7	6.30E7
FB-070-4	40	79	600	1300	8.30E7	8.30E7
FB-070-G	40	79	600	1300	-	2.18E3
FB-095-1	40	79	600	1300	8.30E7	8.30E7
FB-095-2	40	79	600	1300	1.80E6	1.80E6
FB-095-G	800	870	600	1300	9.02E5	9.02E5
FB-113-1	40	79	600	1300	7.10E4	8.14E4
FB-113-2	60	98	600	1300	-	2.20E3
FB-113-3	40	79	600	1300	1.10E6	1.20E6
FB-113-4	40	79	600	1300	3.60E9	3.60E9
FB-113-G	8.0E5	7.7E5	600	1300	1.70E6	1.67E6
FB-131-1	40	79	600	1300	1.10E6	1.20E6
FB-148-1	1.0E5	9.7E4	600	1300	1.02E5	9.91E4
FB-148-2	60	98	600	1300	-	2.20E3
FB-148-G	2.0E3	1.5E3	600	1300	3.70E3	3.60E3
OG-067-1	-	-	-		2.40E7	6.50E7
OG-095-1	-	--	-		1.10E7	3.10E7
OG-095-G	-	-	-		7.00E3	8.00E3
OG-123-1	-	-	-		2.40E7	2.40E7
OG-123-2	-	-	-		8.00E6	9.10E6
OG-123-3	-	-	-		5.90E8	5.90E8
OG-148-1	-	-	-		5.90E8	5.90E8
OG-148-2	-	-	-		5.80E7	1.80E7
OG-148-3	-	-	-		3.5E8	3.5E8
PT-1	200	-	3.0E3	2.70E3	3.90E3	3.50E3
PT-2	200	-	3.0E3	2.70E3	9.03E5	9.03E5
PT-3	200	-	3.0E3	2.70E3	9.03E5	9.03E5
PT-4	200	-	3.0E3	2.70E3	3.90E3	3.40E3
PT-6	-	-	-	2.70E3	7.00E3	1.07E4
PT-7	-	-	-	2.70E3	2.40E7	6.50E7
PT-8	200	-	3.0E03	2.70E3	3.9E03	3.50E03
PT-9	200	-	3.0E03	2.70E3	3.9E03	3.50E03

TABLE 3. River Bend P-T Curve Values for 14 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2a

PRESSURE (PSIG)	14 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	14 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	14 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
0	70.0	70.0	70.0	70.0	70.0	70.0
10	70.0	70.0	70.0	70.0	70.0	70.0
20	70.0	70.0	70.0	70.0	70.0	70.0
30	70.0	70.0	70.0	70.0	70.0	70.0
40	70.0	70.0	70.0	70.0	70.0	70.0
50	70.0	70.0	70.0	70.0	70.0	70.0
60	70.0	70.0	70.0	70.0	70.0	70.0
70	70.0	70.0	70.0	70.0	70.0	71.1
80	70.0	70.0	70.0	70.0	70.0	77.4
90	70.0	70.0	70.0	70.0	70.0	82.7
100	70.0	70.0	70.0	70.0	70.0	87.5
110	70.0	70.0	70.0	70.0	70.0	91.9
120	70.0	70.0	70.0	70.0	70.0	96.1
130	70.0	70.0	70.0	70.0	70.0	100.1
140	70.0	70.0	70.0	70.0	70.0	103.6
150	70.0	70.0	70.0	70.0	70.0	106.8
160	70.0	70.0	70.0	70.0	70.0	109.8
170	70.0	70.0	70.0	72.8	70.0	112.8
180	70.0	70.0	70.0	75.6	70.0	115.6
190	70.0	70.0	70.0	78.2	70.0	118.2
200	70.0	70.0	70.0	80.6	70.0	120.6
210	70.0	70.0	70.0	82.9	70.0	122.9
220	70.0	70.0	70.0	85.2	70.0	125.2
230	70.0	70.0	70.0	87.4	70.0	127.4
240	70.0	70.0	70.0	89.4	70.0	129.4
250	70.0	70.0	70.0	91.4	71.3	131.4
260	70.0	70.0	70.0	93.3	80.9	133.3
270	70.0	70.0	70.0	95.1	89.3	135.1
280	70.0	70.0	70.0	97.0	96.8	137.0
290	70.0	70.0	70.0	98.7	108.3	138.7
300	70.0	70.0	74.2	100.3	114.2	140.3
310	70.0	70.0	79.7	102.0	119.7	142.0
312.5	70.0	70.0	81.0	102.3	121.0	142.3
312.5	70.0	100.0	81.0	130.0	185.5	170.0

TABLE 3. River Bend P-T Curve Values for 14 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2a

PRESSURE (PSIG)	14 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	14 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	14 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
320	70.0	100.0	84.7	130.0	185.5	170.0
330	70.0	100.0	89.4	130.0	185.5	170.0
340	70.0	100.0	93.8	130.0	185.5	170.0
350	70.0	100.0	97.9	130.0	185.5	170.0
360	70.0	100.0	101.8	130.0	185.5	170.0
370	70.0	100.0	105.5	130.0	185.5	170.0
380	70.0	100.0	109.0	130.0	185.5	170.0
390	70.0	100.0	112.4	130.0	185.5	170.0
400	70.0	100.0	115.5	130.0	185.5	170.0
410	70.0	100.0	118.6	130.0	185.5	170.0
420	70.0	100.0	121.5	130.0	185.5	170.0
430	70.0	100.0	124.3	130.0	185.5	170.0
440	70.0	100.0	127.0	130.0	185.5	170.0
450	70.0	100.0	129.6	130.0	185.5	170.0
460	70.0	100.0	132.1	130.0	185.5	170.0
470	70.0	100.0	134.5	130.0	185.5	170.0
480	70.0	100.0	136.8	130.0	185.5	170.0
490	70.0	100.0	139.0	130.0	185.5	170.0
500	71.4	100.0	141.2	130.0	185.5	170.0
510	75.7	100.0	143.3	130.0	185.5	170.0
520	79.7	100.0	145.4	130.0	185.5	170.0
530	83.5	100.0	147.4	130.0	187.4	170.0
540	87.1	100.0	149.3	130.1	189.3	170.1
550	90.6	100.0	151.2	131.1	191.2	171.1
560	93.8	100.0	153.0	132.0	193.0	172.0
570	97.0	100.0	154.8	132.9	194.8	172.9
580	99.9	100.0	156.5	133.8	196.5	173.8
590	102.8	100.0	158.2	134.7	198.2	174.7
600	105.6	100.0	159.9	135.6	199.9	175.6
610	108.2	100.0	161.5	136.5	201.5	176.5
620	110.7	100.0	163.1	137.3	203.1	177.3
630	113.2	100.0	164.6	138.2	204.6	178.2
640	115.6	100.0	166.1	139.0	206.1	179.0
650	117.9	100.0	167.6	139.8	207.6	179.8
660	120.1	100.0	169.0	140.6	209.0	180.6
670	122.2	100.0	170.5	141.4	210.5	181.4
680	124.3	100.0	171.8	141.9	211.8	181.9

TABLE 3. River Bend P-T Curve Values for 14 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2a

PRESSURE (PSIG)	14 EFPY BELTLINE (°F)	NON-BELTLINE (°F)	14 EFPY BELTLINE (°F)	NON-BELTLINE (°F)	14 EFPY BELTLINE (°F)	NON-BELTLINE (°F)
690	126.3	100.0	173.2	142.3	213.2	182.3
700	128.3	100.0	174.5	142.8	214.5	182.8
710	130.2	100.0	175.8	143.2	215.8	183.2
720	132.1	100.0	177.1	143.6	217.1	183.6
730	133.9	100.0	178.4	144.0	218.4	184.0
740	135.7	100.6	179.6	144.4	219.6	184.4
750	137.4	102.0	180.9	144.8	220.9	184.8
760	139.1	103.5	182.0	145.2	222.0	185.2
770	140.7	104.8	183.2	145.6	223.2	185.6
780	142.3	106.2	184.4	146.0	224.4	186.0
790	143.9	107.6	185.5	146.4	225.5	186.4
800	145.4	108.9	186.6	146.7	226.6	186.7
810	146.9	110.2	187.7	147.1	227.7	187.1
820	148.3	111.4	188.8	147.5	228.8	187.5
830	149.8	112.7	189.9	147.9	229.9	187.9
840	151.2	113.9	190.9	148.3	230.9	188.3
850	152.6	115.1	192.0	148.7	232.0	188.7
860	153.9	116.3	193.0	149.0	233.0	189.0
870	155.2	117.5	194.0	149.4	234.0	189.4
880	156.5	118.6	195.0	149.8	235.0	189.8
890	157.8	119.7	196.0	150.1	236.0	190.1
900	159.1	120.8	196.9	150.5	236.9	190.5
910	160.3	121.9	197.9	150.9	237.9	190.9
920	161.5	123.0	198.8	151.2	238.8	191.2
930	162.7	124.0	199.7	151.6	239.7	191.6
940	163.9	125.1	200.6	152.0	240.6	192.0
950	165.0	126.1	201.5	152.5	241.5	192.5
960	166.1	127.1	202.4	153.4	242.4	193.4
970	167.2	128.1	203.3	154.3	243.3	194.3
980	168.3	129.1	204.2	155.2	244.2	195.2
990	169.4	130.0	205.0	156.1	245.0	196.1
1000	170.5	131.0	205.9	157.0	245.9	197.0
1010	171.5	131.9	206.7	157.8	246.7	197.8
1020	172.5	132.9	207.5	158.7	247.5	198.7
1030	173.6	133.8	208.3	159.5	248.3	199.5
1040	174.6	134.7	209.1	160.4	249.1	200.4
1050	175.5	135.6	209.9	161.2	249.9	201.2
1060	176.5	136.5	210.7	162.0	250.7	202.0
1070	177.5	137.3	211.5	162.8	251.5	202.8

TABLE 3. River Bend P-T Curve Values for 14 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2a

PRESSURE (PSIG)	14 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	14 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	14 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
1080	178.4	138.2	212.3	163.6	252.3	203.6
1090	179.3	139.0	213.0	164.4	253.0	204.4
1100	180.3	139.9	213.8	165.1	253.8	205.1
1110	181.2	140.7	214.5	165.9	254.5	205.9
1120	182.1	141.5	215.3	166.7	255.3	206.7
1130	182.9	142.3	216.0	167.4	256.0	207.4
1140	183.8	143.1	216.7	168.1	256.7	208.1
1150	184.7	143.9	217.4	168.9	257.4	208.9
1160	185.5	144.7	218.1	169.6	258.1	209.6
1170	186.4	145.5	218.8	170.3	258.8	210.3
1180	187.2	146.2	219.5	171.0	259.5	211.0
1190	188.0	147.0	220.2	171.7	260.2	211.7
1200	188.8	147.7	220.9	172.4	260.9	212.4
1210	189.6	148.5	221.6	173.1	261.6	213.1
1220	190.4	149.2	222.2	173.8	262.2	213.8
1230	191.2	149.9	222.9	174.5	262.9	214.5
1240	192.0	150.6	223.5	175.1	263.5	215.1
1250	192.7	151.3	224.2	175.8	264.2	215.8
1260	193.5	152.0	224.8	176.5	264.8	216.5
1270	194.2	152.7	225.5	177.1	265.5	217.1
1280	195.0	153.4	226.1	177.8	266.1	217.8
1290	195.7	154.1	226.7	178.4	266.7	218.4
1300	196.4	154.8	227.3	179.0	267.3	219.0
1310	197.2	155.4	227.9	179.6	267.9	219.6
1320	197.9	156.1	228.5	180.3	268.5	220.3
1330	198.6	156.8	229.1	180.9	269.1	220.9
1340	199.3	157.4	229.7	181.5	269.7	221.5
1350	200.0	158.1	230.3	182.1	270.3	222.1
1360	200.6	158.7	230.9	182.7	270.9	222.7
1370	201.3	159.3	231.5	183.3	271.5	223.3
1380	202.0	159.9	232.1	183.9	272.1	223.9
1390	202.6	160.6	232.6	184.5	272.6	224.5
1400	203.3	161.2	233.2	185.0	273.2	225.0

TABLE 4. River Bend P-T Curve Values for 32 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2b

PRESSURE (PSIG)	32 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	32 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	32 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
0	70.0	70.0	70.0	70.0	70.0	70.0
10	70.0	70.0	70.0	70.0	70.0	70.0
20	70.0	70.0	70.0	70.0	70.0	70.0
30	70.0	70.0	70.0	70.0	70.0	70.0
40	70.0	70.0	70.0	70.0	70.0	70.0
50	70.0	70.0	70.0	70.0	70.0	70.0
60	70.0	70.0	70.0	70.0	70.0	70.0
70	70.0	70.0	70.0	70.0	70.0	71.1
80	70.0	70.0	70.0	70.0	70.0	77.4
90	70.0	70.0	70.0	70.0	70.0	82.7
100	70.0	70.0	70.0	70.0	70.0	87.5
110	70.0	70.0	70.0	70.0	70.0	91.9
120	70.0	70.0	70.0	70.0	70.0	96.1
130	70.0	70.0	70.0	70.0	70.0	100.1
140	70.0	70.0	70.0	70.0	70.0	103.6
150	70.0	70.0	70.0	70.0	70.0	106.8
160	70.0	70.0	70.0	70.0	70.0	109.8
170	70.0	70.0	70.0	72.8	70.0	112.8
180	70.0	70.0	70.0	75.6	70.0	115.6
190	70.0	70.0	70.0	78.2	70.0	118.2
200	70.0	70.0	70.0	80.6	70.0	120.6
210	70.0	70.0	70.0	82.9	70.0	122.9
220	70.0	70.0	70.0	85.2	70.0	125.2
230	70.0	70.0	70.0	87.4	73.8	127.4
240	70.0	70.0	70.0	89.4	87.1	129.4
250	70.0	70.0	70.0	91.4	98.3	131.4
260	70.0	70.0	70.0	93.3	107.9	133.3
270	70.0	70.0	76.3	95.1	116.3	135.1
280	70.0	70.0	83.8	97.0	123.8	137.0
290	70.0	70.0	95.3	98.7	135.3	138.7
300	70.0	70.0	101.2	100.3	141.2	140.3
310	70.0	70.0	106.7	102.0	146.7	142.0
312.5	70.0	70.0	108.0	102.3	148.0	142.3
312.5	70.0	100.0	108.0	130.0	212.5	170.0

TABLE 4. River Bend P-T Curve Values for 32 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2b

PRESSURE (PSIG)	32 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	32 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	32 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
320	70.0	100.0	111.7	130.0	212.5	170.0
330	70.0	100.0	116.4	130.0	212.5	170.0
340	70.0	100.0	120.8	130.0	212.5	170.0
350	70.0	100.0	124.9	130.0	212.5	170.0
360	70.0	100.0	128.8	130.0	212.5	170.0
370	70.0	100.0	132.5	130.0	212.5	170.0
380	70.0	100.0	136.0	130.0	212.5	170.0
390	70.0	100.0	139.4	130.0	212.5	170.0
400	70.0	100.0	142.5	130.0	212.5	170.0
410	70.0	100.0	145.6	130.0	212.5	170.0
420	70.0	100.0	148.5	130.0	212.5	170.0
430	70.0	100.0	151.3	130.0	212.5	170.0
440	70.0	100.0	154.0	130.0	212.5	170.0
450	71.7	100.0	156.6	130.0	212.5	170.0
460	77.9	100.0	159.1	130.0	212.5	170.0
470	83.6	100.0	161.5	130.0	212.5	170.0
480	88.9	100.0	163.8	130.0	212.5	170.0
490	93.8	100.0	166.0	130.0	212.5	170.0
500	98.4	100.0	168.2	130.0	212.5	170.0
510	102.7	100.0	170.3	130.0	212.5	170.0
520	106.7	100.0	172.4	130.0	212.5	170.0
530	110.5	100.0	174.4	130.0	214.4	170.0
540	114.1	100.0	176.3	130.1	216.3	170.1
550	117.6	100.0	178.2	131.1	218.2	171.1
560	120.8	100.0	180.0	132.0	220.0	172.0
570	124.0	100.0	181.8	132.9	221.8	172.9
580	126.9	100.0	183.5	133.8	223.5	173.8
590	129.8	100.0	185.2	134.7	225.2	174.7
600	132.6	100.0	186.9	135.6	226.9	175.6
610	135.2	100.0	188.5	136.5	228.5	176.5
620	137.7	100.0	190.1	137.3	230.1	177.3
630	140.2	100.0	191.6	138.2	231.6	178.2
640	142.6	100.0	193.1	139.0	233.1	179.0
650	144.9	100.0	194.6	139.8	234.6	179.8
660	147.1	100.0	196.0	140.6	236.0	180.6

TABLE 4. River Bend P-T Curve Values for 32 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2b

PRESSURE (PSIG)	32 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	32 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	32 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
670	149.2	100.0	197.5	141.4	237.5	181.4
680	151.3	100.0	198.8	141.9	238.8	181.9
690	153.3	100.0	200.2	142.3	240.2	182.3
700	155.3	100.0	201.5	142.8	241.5	182.8
710	157.2	100.0	202.8	143.2	242.8	183.2
720	159.1	100.0	204.1	143.6	244.1	183.6
730	160.9	100.0	205.4	144.0	245.4	184.0
740	162.7	100.6	206.6	144.4	246.6	184.4
750	164.4	102.0	207.9	144.8	247.9	184.8
760	166.1	103.5	209.0	145.2	249.0	185.2
770	167.7	104.8	210.2	145.6	250.2	185.6
780	169.3	106.2	211.4	146.0	251.4	186.0
790	170.9	107.6	212.5	146.4	252.5	186.4
800	172.4	108.9	213.6	146.7	253.6	186.7
810	173.9	110.2	214.7	147.1	254.7	187.1
820	175.3	111.4	215.8	147.5	255.8	187.5
830	176.8	112.7	216.9	147.9	256.9	187.9
840	178.2	113.9	217.9	148.3	257.9	188.3
850	179.6	115.1	219.0	148.7	259.0	188.7
860	180.9	116.3	220.0	149.0	260.0	189.0
870	182.2	117.5	221.0	149.4	261.0	189.4
880	183.5	118.6	222.0	149.8	262.0	189.8
890	184.8	119.7	223.0	150.1	263.0	190.1
900	186.1	120.8	223.9	150.5	263.9	190.5
910	187.3	121.9	224.9	150.9	264.9	190.9
920	188.5	123.0	225.8	151.2	265.8	191.2
930	189.7	124.0	226.7	151.6	266.7	191.6
940	190.9	125.1	227.6	152.0	267.6	192.0
950	192.0	126.1	228.5	152.5	268.5	192.5
960	193.1	127.1	229.4	153.4	269.4	193.4
970	194.2	128.1	230.3	154.3	270.3	194.3
980	195.3	129.1	231.2	155.2	271.2	195.2
990	196.4	130.0	232.0	156.1	272.0	196.1
1000	197.5	131.0	232.9	157.0	272.9	197.0
1010	198.5	131.9	233.7	157.8	273.7	197.8

TABLE 4. River Bend P-T Curve Values for 32 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2b

PRESSURE (PSIG)	32 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	32 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	32 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
1020	199.5	132.9	234.5	158.7	274.5	198.7
1030	200.6	133.8	235.3	159.5	275.3	199.5
1040	201.6	134.7	236.1	160.4	276.1	200.4
1050	202.5	135.6	236.9	161.2	276.9	201.2
1060	203.5	136.5	237.7	162.0	277.7	202.0
1070	204.5	137.3	238.5	162.8	278.5	202.8
1080	205.4	138.2	239.3	163.6	279.3	203.6
1090	206.3	139.0	240.0	164.4	280.0	204.4
1100	207.3	139.9	240.8	165.1	280.8	205.1
1110	208.2	140.7	241.5	165.9	281.5	205.9
1120	209.1	141.5	242.3	166.7	282.3	206.7
1130	209.9	142.3	243.0	167.4	283.0	207.4
1140	210.8	143.1	243.7	168.1	283.7	208.1
1150	211.7	143.9	244.4	168.9	284.4	208.9
1160	212.5	144.7	245.1	169.6	285.1	209.6
1170	213.4	145.5	245.8	170.3	285.8	210.3
1180	214.2	146.2	246.5	171.0	286.5	211.0
1190	215.0	147.0	247.2	171.7	287.2	211.7
1200	215.8	147.7	247.9	172.4	287.9	212.4
1210	216.6	148.5	248.6	173.1	288.6	213.1
1220	217.4	149.2	249.2	173.8	289.2	213.8
1230	218.2	149.9	249.9	174.5	289.9	214.5
1240	219.0	150.6	250.5	175.1	290.5	215.1
1250	219.7	151.3	251.2	175.8	291.2	215.8
1260	220.5	152.0	251.8	176.5	291.8	216.5
1270	221.2	152.7	252.5	177.1	292.5	217.1
1280	222.0	153.4	253.1	177.8	293.1	217.8
1290	222.7	154.1	253.7	178.4	293.7	218.4
1300	223.4	154.8	254.3	179.0	294.3	219.0
1310	224.2	155.4	254.9	179.6	294.9	219.6
1320	224.9	156.1	255.5	180.3	295.5	220.3
1330	225.6	156.8	256.1	180.9	296.1	220.9
1340	226.3	157.4	256.7	181.5	296.7	221.5
1350	227.0	158.1	257.3	182.1	297.3	222.1
1360	227.6	158.7	257.9	182.7	297.9	222.7

TABLE 4. River Bend P-T Curve Values for 32 EFPY

Required Temperatures at 100 °F/hr for Curves B & C and 20 °F/hr for Curve A

FOR FIGURE 3-2b

PRESSURE (PSIG)	32 EFPY BELTLINE CURVE A' (°F)	NON-BELTLINE CURVE A (°F)	32 EFPY BELTLINE CURVE B' (°F)	NON-BELTLINE CURVE B (°F)	32 EFPY BELTLINE CURVE C' (°F)	NON-BELTLINE CURVE C (°F)
1370	228.3	159.3	258.5	183.3	298.5	223.3
1380	229.0	159.9	259.1	183.9	299.1	223.9
1390	229.6	160.6	259.6	184.5	299.6	224.5
1400	230.3	161.2	260.2	185.0	300.2	225.0

Attachment 3

Power Uprate USAR Changes Matrix

POWER UPRATE USAR CHANGES

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(Sorted by USAR Section)

USAR SECTION	USAR DESCRIPTION	PU SAR SECTION	POWER UPRATE SAFETY ANALYSIS REPORT DESCRIPTION / NOTES
1.1	Introduction – Changed 2894 MWt to 3039 MWt. Figure 1.1-1 updated to new heat balance.	1.1 Table 1-2	This evaluation justifies a power uprate to 3039 MWt, which corresponds to 105% of the current rated thermal power for River Bend Station.
3.9.5.2.2B	Pressure Differential During Rapid Depressurization – Added statement describing additional analysis performed using LAMB thermal-hydraulic computer code. Discusses methodology used in calculating Reactor Internal Pressure Differentials (RIPDs).	4.1 3.3.2.1	Containment System Performance - Describes the more detailed RPV model used in those containment analyses which were used to evaluate hydrodynamic loads for power uprate (LAMB). Reactor Internals and Pressure Differentials - The reactor internal component loading is determined by load combinations that include reactor internal pressure difference (RIPD), LOCA, SRV, seismic, and fuel lift loads.
3.9.5.2.3.2B	Effects of Initial Reactor Power and Core Flow – Added description of analysis performed at low power, high recirc flow condition (cavitation interlock).	3.4	Reactor Recirculation System
Table 3.9B-2a	Vessel Support Skirt – Changed maximum cumulative usage factor from "0.4905 at RPV bottom head" to "0.999 at RPV support skirt bottom head junction".	Table 3-2	Primary Plus Secondary Stress Intensities Range and Cumulative Fatigue Usage Factors of Limiting Components – Lists changes to usage factors due to uprate.
Table 3.9B-2a	Shroud Support – Changed maximum calculated stress (psi) for Normal, Upset, Emergency and Faulted Conditions.	Table 3-3	Component Stresses Adjusted for Power Uprate – Lists adjusted stresses for various RPV internal components. ???
Table 3.9B-2a	RPV Feedwater Nozzle – Changed maximum cumulative usage factors for nozzle safe ends.	Table 3-2	Primary Plus Secondary Stress Intensities Range and Cumulative Fatigue Usage Factors of Limiting Components – Contains uprated fatigue usage factors of feedwater nozzles.
Table 3.9B-2b	Grid – Highest Stressed Beam – Changed maximum calculated stress for Grid under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-2b	Core Plate (Ligament in Top Plate) – Changed maximum calculated stress for Core Plate under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-2v	Jet Pumps – Changed calculated stress for Jet Pumps under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-2w	Highest Stressed Region on the LPCI Coupling (Strut to Weld) – Changed calculated stress for LPCI coupling under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	

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Table 3.9B-2x	Control Rod Guide Tube – Changed calculated stress for Control Rod Guide Tube under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-2y	Incore Housing – Changed calculated stress for Incore Housing under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-2z	Reactor Vessel Support Equipment (CRD Housing Support) – Changed calculated stress for CRD Housing under Normal, Upset, Emergency and Faulted Conditions.	GE Design Record File	
Table 3.9B-5	Pressure Differentials across Reactor Vessel Internals – Changed maximum pressure differences occurring during a steam line break.	Table 3-6 GE Design Record File	RIPDs for Faulted Conditions – Lists uprated pressure differences. Numbers from GE Design Record File Section 40.07, Volume 4.
Figure 3.9B-5a	Transient Pressure Differentials following a Steamline Break – Replaced figure.	GE Design Record File	
Figure 3.9B-5b	Transient Pressure Differentials following a Steamline Break – Replaced figure.	GE Design Record File	
Figure 3.9B-11	Pressure Nodes used for Depressurization Analysis	GE Design Record File	
4.3.2.5.1	Rod Control and Information System – Changed low power setpoint based on turbine monitoring.	5.3.12	Rod Control and Information System – New turbine first-stage pressure setpoints will be determined.
4.3.2.5.2	Bank Position Mode – Changed low power setpoint based on turbine monitoring.	5.3.12	Rod Control and Information System – New turbine first-stage pressure setpoints will be determined.
4.3.2.5.3	Rod Withdrawal Limited Mode – Changed low power setpoint based on turbine monitoring.	5.3.12	Rod Control and Information System – New turbine first-stage pressure setpoints will be determined.
4.3.2.8	Vessel Irradiations – Added 10.43 EFPY as the beginning of power uprate to be used to calculate flux and fluence.	3.3.1.1 Fig. 3-2a	Reactor Vessel Fracture Toughness Minimum RPV Metal Temperature vs. Reactor Vessel Pressure For 14 EFPY.
Table 4.3-5	Dosimeter and Vessel Peak Fluxes and Fluences – Changed EOC1 Peak I.D. and 32 EFPY Peak I.D. to new uprate values.	3.3.1.1 Fig. 3-2a Fig. 3-2b	Reactor Vessel Fracture Toughness; Minimum RPV Metal Temperature vs. Reactor Vessel Pressure For 14 EFPY; Minimum RPV Metal Temperature vs. Reactor Vessel Pressure For 32 EFPY.
4.4.3.3.1.2	Maximum Extended Load Line – Changed description of rod line to "...passes through the 100% power / 81% core flow point	Fig. 2-1	Power/Flow Operating Map for Power Uprate.

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	(approximately 115% rod line)...".		
4.4.3.3.3	Design Features for Power Flow Control – Changed low feedwater flow interlock setpoint from 25% of rated to 24%.	Fig. 2-1	Power/Flow Operating Map for Power Uprate.
Figure 4.4-5	Updated Power-Flow Operating including MELLL / ICF – Replaced power to flow map.	Fig. 2-1	Power/Flow Operating Map for Power Uprate.
4.6.1.2.3	CRD Housing Description – Changed reactor bottom head operating pressure from 1086 to 1100 psig ; changed the highest force resulting from a postulated failure from 32,000 to 32,500 lbs.	2.5.1	Control Rod Drive System.
4.6.2.3.2.2.1	Drive Housing Fails at Attachment Weld – Added statement that calculated values may increase slightly when operating at power uprate reactor pressure.	2.5.1 3	Control Rod Drive System
5.2.2	Overpressure Protection – Described re-analysis of MSIV event with flux scram at 3100 MWt.	3.2	Reactor Overpressure Protection
5.2.2.1	Design Basis – Updated description of overpressure analysis (thermal power, pressure).	2.5.1 3.2	Control Rod Drive System Reactor Overpressure Protection
5.2.2.2.2.1	Operating Conditions – Changed operating power, vessel dome pressure and steam flow.	3.2 7.1	Reactor Overpressure Protection Turbine-Generator
5.2.2.2.2.4	Safety/Relief Valve Transient Analysis Specification – Changed pressure setpoint for relief mode from "1125 to 1155 psig" to "1163 to 1183 psig".	Table 5-1	Analytical Limits for Setpoints
5.2.2.2.3.1	Safety / Relief Valve Capacity – Changed maximum vessel dome pressure and relief setpoint range of the SRVs.	3.2 Table 5-1	Reactor Overpressure Protection Analytical Limits for Setpoints
Table 5.2-2	Nuclear System Safety / Relief Setpoints – Changed spring set pressure, relief pressure, low-low set relief pressure and rated capacity of each SRV.	Table 5-1 GE Design Record File	Analytical Limits for Setpoints
Table 5.2-9	Sequence of Events for Fig. 5.2-1 – Updated sequence of events.	Fig. 3-1 GE Design Record File	Response to MSIV Closure with Flux Scram. New Sequence numbers are from ODYN computer run which are in the GE Design Record Files.
Fig. 5.2-1	MSIV Closure with Flux Scram and Installed Safety/Relief Valve Capacity – Replaced figures.	Fig. 3-1	Response to MSIV Closure with Flux Scram.
5.3.1.6.2	Neutron Flux and Fluence Calculations – Changed peak fluence.	3.3.1.1	Reactor Vessel Fracture Toughness

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		Table 3-1	Beltline Art Values for River Bend Power Uprate
5.3.2.1	Limit Curves – Referenced 10CFR50 Appendix H and ASTM E185.	3.3.1.1	Reactor Vessel Fracture Toughness
5.3.2.1.2	Temperature Limits for ISI Hydrostatic or Leak Pressure Tests – Adjusted belt line curve.	Fig. 3-2a	Minimum RPV Metal Temperature vs. Reactor Vessel Pressure for 14 EFPY.
Table 5.3-1	RBS Reactor Vessel Charpy Test Results – Changed belt line RTD temperatures.	Fig. 3-2b	Minimum RPV Metal Temperature vs. Reactor Vessel Pressure for 32 EFPY.
Fig. 5.3-4a	Minimum Temperatures required vs. Reactor Pressure (Valid up to 32 EFPY) – Updated figure.	Fig. 3-2b	Minimum RPV Metal Temperature vs. Reactor Vessel Pressure for 32 EFPY.
Fig. 5.3-4b	Minimum Temperatures required vs. Reactor Pressure (Valid up to 12 EFPY) – Updated figure.		Reactor Vessel Fracture Toughness. 14 EFPY figure replaces 12 EFPY.
Fig. 5.3-4c	RPV Minimum Temperature Required vs. Pressure (Valid up to 14 EFPY) – Added new figure.	Fig. 3-2a	Minimum RPV Metal Temperature vs. Reactor Vessel Pressure for 14 EFPY.
Fig. 5.3-5	Predicted Adjusted Reference Temperature as a function of Effective Full Power Years of Operation – Deleted figure.		Per GE, this is an outdated figure that is no longer used in the industry. It has no historical value or useful information and is being deleted.
5.4.4.2	Adjusted steam line flow rate.	Table 1-2 Fig. 1-1a & b	Original and Uprated Plant Operating Conditions Uprated Heat Balance
5.4.5.2	Description - Adjusted steam line flow rate and description of MSIV.	Table 1-2 Fig. 1-1a & b	Original and Uprated Plant Operating Conditions Uprated Heat Balance
5.4.6.1.1.1	Residual Heat – Changed to “maximum steady state steam flow at 1246 psia”.	3.8 5.3.15 Table 5-1	Reactor Core Isolation Cooling RCIC Steam Line and RCIC/RHR Steam Line High Flow Isolations Analytical Limits for Setpoints
5.4.6.2.2.2	Design Parameters – Updated RCIC operating parameters.	3.8 Table 5-1	Reactor Core Isolation Cooling Analytical Limits for Setpoints Reactor Core Isolation Cooling System Performance Comparison
5.4.6.2.4	System Reliability Considerations – Corrected most limiting operating condition for RCIC.	3.8	Reactor Core Isolation Cooling
Fig. 5.4-10	Vessel Coolant Temperature vs. Time (Two heat exchangers available) – Updated figure.	Appendix A	Changes to figures resulted from an analysis reported in Appendix A.
Fig. 5.4-11	Vessel Coolant Temperature vs. Time (One heat exchanger available) – Updated figure.	Appendix A	Changes to figures resulted from an analysis reported in Appendix A.
Table 5.4-1	Revised Recirc Pump and Jet Pump design parameters.		BILBO Computer Runs

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		GE Design Record File	A22-00-81, Section 40.06
Table 5.4-2	RWCU System Equipment Design and Operating Data – Added RWCU operating data.		Correcting existing discrepancy with NRHX design pressure and temperature.
6.2	Revised drywell and containment differential pressure. Changed description of containment internal pressure. Revised peak containment temperature.	4.1.1.3 Table 4-1	Short-Term Containment Pressure Response Containment Performance Results
6.2.1.1.2	Changed amount of water on the drywell floor prior to reaching top of weir wall.		Containment Analysis Key inputs used in Containment Analyses
6.2.1.1.3.1	Accident Response Analyses - Updated computer programs used to determine containment response following a pipe break in the drywell.	4.1	Containment System Performance
6.2.1.1.3.1.2	Initial Conditions - Changed peak differential pressure.	4.1.1 4.1.1.3 Table 4-1	Containment Temperature and Pressure Response Short-Term Containment Pressure Response Containment Performance Results
6.2.1.1.3.1.4.1	Break Area and Mass Energy Release Assumptions - Revised description of break area and mass energy released.	4.1 GE Design Record File	Containment System Response
6.2.1.1.3.1.4.2	Containment System Response – Added computer code references and revised containment response.	4.1 Table 4-1 GE Design Record File	Containment System Response Containment Performance Results
6.2.1.1.3.1.5.1	Changed stored energy of fuel at higher temperatures.	GE Design Record File	
6.2.1.1.3.1.5.2	Changed description of figures to first 30 seconds of transient response.		Containment Analysis
6.2.1.1.3.7.1	LOCA Containment Response Model - Changed computer program codes.	4.1	Containment System Response
6.2.1.1.3.7.1	Added statement that "sensitivity analyses were performed using the original USAR methods and with an initial reactor power of 2952 MWt".	3.0 4.0 5.0	Assumptions

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6.2.1.2.3	Expanded description of blowdown release and eliminated the RCIC head spray line break.	GE Design Record File 3.7.2	Drywell Head Pressurization
6.2.1.3	Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents – Changed computer codes.	4.1	Containment System Response
6.2.2.3	Design Evaluation - Changed MWt and computer code.	3.0 4.0 5.0 4.1	Assumptions Assumptions Assumptions Containment System Response
Table 6.2-1	Containment Design Parameters – Updated table.	GE Design Record File	
Table 6.2-3	Initial Conditions for Containment Response Analyses – Updated table with new uprate values for Reactor Coolant System, Drywell and Containment.	3.1.1.2 GE Design Record File	Drywell Temperature
Table 6.2-4	Blowdown Data Main Steam Line Break (Long Term, Minimum ESF, with Feedwater) – Updated table.	GE Design Record File	
Table 6.2-4a	Blowdown Data Main Steam Line Break (Short Term, Minimum ESF, No Feedwater) – Updated table.	GE Design Record File	
Table 6.2-5	Blowdown Data Recirculation Line Break (Long Term, Minimum ESF, with Feedwater) – Updated table.	GE Design Record File	
Table 6.2-5a	Blowdown Data Recirculation Line Break (Short Term, Minimum ESF, No Feedwater) – Updated table.	GE Design Record File	
Table 6.2-6	Passive Heat Sinks – Replaced table.		Replaced with Tables A-3 Drywell Heat Sinks, A-4 Containment Heat Sinks and A-5 Thermophysical Properties of Passive Heat Sink Materials.
Table 6.2-7	Results of Containment Response Analyses – Replaced table.	GE Design Record File	
Table 6.2-8	Energy Balance for Main Steam Line Break – Replaced table.	GE Design Record File	

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USAR SECTION	USAR DESCRIPTION	PU SAR SECTION	POWER UPRATE SAFETY ANALYSIS REPORT DESCRIPTION / NOTES
Table 6.2-9	Energy Balance for Recirculation Line Break – Replaced table.	GE Design Record File	
Table 6.2-10	Time Sequence of Events for Main Steam Line Break – Replaced table.	GE Design Record File	
Table 6.2-11	Time Sequence of Events for Recirculation Line Break – Replaced table.	GE Design Record File	
Fig. 6.2-2	Main Steam Line Break Flow Area – Replaced figure.	GE Design Record File	
Fig. 6.2-3	Recirculation Line Break Schematic – Replaced figure.	GE Design Record File	
Fig. 6.2-4	Containment Pressure Response for Main Steam Line Break – Replaced figures.	0	Replaced with Figures: B-3 DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V). C-9 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V). C-11 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, t >1800 sec, (SHEX-04V).
Fig. 6.2-5	Containment Pressure Response for Recirculation Line Break – Replaced figures.		Replaced with Figures: B-11 DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V). D-5 DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V).
Fig. 6.2-6	Containment Temperature Response for Main Steam Line Break – Replaced figures.		Replaced with Figures: B-4 DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V).

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			C-10 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V). C-12 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. MSLB 102% Uprated Power/ 100% Rated Core Flow, t > 1800 sec, (SHEX-04V).
Fig. 6.2-7	Containment Temperature Response for Recirculation Line Break – Replaced figures.	0	Replaced with Figures: B-12 DBA-LOCA Short-Term Drywell and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V). D-6 DBA-LOCA Long-Term Drywell, Suppression Pool and Containment Airspace Temperature. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V).
Fig. 6.2-8	Vent Flow Rate for Main Steam Line Break – Replaced figure.	GE Design Record File	
Fig. 6.2-9	Blowdown Mass Flow Rate for Main Steam Line Break -- Replaced figure.	GE Design Record File	
Fig. 6.2-9a	Blowdown Comparison Between RBS & CPS Main Steam Line Break Mass Flow Rate vs. Time – Replaced figure with Blowdown Mass Flow Rate for Main Steam Line Break, SHEX 0-1800 Seconds.	GE Design Record File	
Fig. 6.2-9b	Blowdown Comparison Between RBS & CPS Main Steam Line Break Energy Flow Rate vs. Time – Replaced figure with Blowdown Mass Flow Rate for Main Steam Line Break, SHEX Long-term.	GE Design Record File	
Fig. 6.2-10	Blowdown Mass Flow Rate for Recirculation Suction Line Break – Replaced figure with Blowdown Mass Flow Rate for Recirculation Suction Line Break (M3CPT 0-30 seconds).	GE Design Record File	
Fig. 6.2-10a	Blowdown Comparison Between RBS & CPS Recirculation Line Break Mass Flow Rate vs. Time – Replaced figure with Blowdown Mass Flow Rate for Recirculation Suction Line Break, SHEX Long Term.	GE Design Record File	

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Fig. 6.2-10b	Blowdown Comparison Between RBS & CPS Recirculation Line Break Energy Flow Rate vs. Time – Deleted this figure.	GE Design Record File	
Fig. 6.2-11	Fission Product and Heavy Element Decay Heat for Main Steam Line Break – Replaced with two new figures: Fig. 6.2-11 Normalized Core Power for Short Term Main Steam Line Break (M3CPT 0-30 sec); Fig. 6.2-11a Normalized Core Power for Long Term Main Steam Line Break (SHEX Long Term).	GE Design Record File	
Fig. 6.2-12	Fission Power Coastdown Energy for Main Steam Line Break – Deleted figure.	GE Design Record File	Per GE, this is an outdated figure that is no longer used in the industry. It has no historical value or useful information and is being deleted.
Fig. 6.2-13	Reactor System Metal and Core Stored Energy for Main Steam Line Break – Replaced figure with Main Steam Line Break Integrated Core and Vessel Metal Energy SHEX Long Term.	GE Design Record File	
Fig. 6.2-14	Metal Water Reaction Heat for Main Steam Line Break – Deleted figure.	GE Design Record File	Per GE, this is an outdated figure that is no longer used in the industry. It has no historical value or useful information and is being deleted. Combined with Figures 6.2-11 and 6.2-11a.
Fig. 6.2-15	ECCS Pump Heat for Main Steam Line Break – Replaced with new figure.	GE Design Record File	
Fig. 6.2-16	Suppression Pool Water Level for Short Term (0-600 sec) – Replaced figure with 3 new ones: Fig. 6.2-16 Suppression Pool Water Level Main Steam Line Break M3CPT (0-30 sec); Fig. 6.2-16a Suppression Pool Water Level Main Steam Line Break, SHEX (0-1800 sec); Fig. 6.2-16b Suppression Pool Water Level Main Steam Line Break, SHEX Long Term.	GE Design Record File	
Fig. 6.2-17	Suppression Pool Water Level for Long Term (600-10^6 sec) – Replaced figure with 2 new ones: Fig. 6.2-17 Suppression Pool Water Level Recirculation Line Break M3CPT (0-30 sec); Fig. 6.2-17a Suppression Pool Water Level Recirculation Line Break, SHEX Long Term.	GE Design Record File	
Fig. 6.2-18	RHR Heat Exchanger Duty for Main Steam Line Break – Replaced figure with RHR Heat Exchanger Duty for Main Steam Line Break	GE Design Record File	

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	SHEX Long Term.		
Fig. 6.2-19	RHR Heat Exchanger Duty for Recirculation Suction Line Break – Replaced with new figure.	GE Design Record File	
Fig. 6.2-20	Containment Unit Cooler Duty for Main Steam Line Break – Replaced with new figure: Containment Unit Cooler Duty for Main Steam Line Break, SHEX Long Term.	GE Design Record File	
Fig. 6.2-21	Containment Unit Cooler Duty for Recirculation Suction Line Break – Replaced figure.	GE Design Record File	
Fig. 6.2-22	Passive Sink Heat Removal Rate for Main Steam Line Break – Replaced figure with two new ones: Fig. 6.2-22 Passive Drywell Heat Sink Heat Removal Rate for Main Steam Line Break, SHEX Long Term; Fig. 6.2-22a Passive Containment Heat Sink Heat Removal Rate for Main Steam Line Break, SHEX Long Term.	GE Design Record File	
Fig. 6.2-23	Passive Sink Heat Removal Rate for Recirculation Line Break – Replaced figure with two new ones: Fig. 6.2-23 Passive Drywell Heat Sink Heat Removal Rate for Recirculation Line Break, SHEX Long Term; Fig. 6.2-23a Passive Containment Heat Sink Heat Removal Rate for Recirculation Line Break, SHEX Long Term.	GE Design Record File	
Fig. 6.2-24	Passive Sink Surface Heat Transfer Coefficient for Main Steam Line Break – Deleted this figure.	GE Design Record File	Per GE, this is an outdated figure that is no longer used in the industry. It has no historical value or useful information and is being deleted.
Fig. 6.2-25	Passive Sink Surface Heat Transfer Coefficient for Recirculation Line Break – Deleted this figure.	GE Design Record File	Per GE, this is an outdated figure that is no longer used in the industry. It has no historical value or useful information and is being deleted.
Fig. 6.2-27a	Drywell and Containment Pressures for a Break Area of 0.1 Ft^2 with Bypass Leakage of 1.15 Ft^2 – Replaced with Table Summary of Results for Steam Bypass Analysis.		Replaced with Figure G-3 Steam Bypass - 0.1 ft^2 Steam Break, 1.15 ft^2 Bypass Leakage Area, Long-Term Drywell and Containment Airspace Pressure. 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V).
6A.1	Introduction – Added description of hydrodynamic loads and methodology used.		3.8 – DBA LOCA Hydrodynamic Containment Loads

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Table 6A.1-1	Summary of Specified and Realistic Design Values – Updated peak drywell and wetwell pressures.	Table 4-1	Containment Performance Results
6A.2.1	Design Basis Accident (DBA) – Changed MWt, computer codes, peak drywell pressure and modified description of containment dynamic loads.	Table 4-1	Containment Performance Results 4.1 – Containment System Performance
6A.4.1.2	Drywell Pressure – Changed peak drywell psid and computer code.	4.1 Table 4-1	Containment System Performance Containment Performance Results
6A.4.5	Drywell Environmental Envelope – Added description of drywell computer code analysis.	4.1.1.2	Short Term Gas Temperature Response
6A.6.1.11	Long Term Transient – Added new computer codes.	4.1	Containment System Performance
6A.6.1.12	Containment Environmental Envelope – Added new computer codes.	4.1	Containment System Performance
6A.12	Loads on Structures at and above the HCU Floor Elevation – Added new computer code and new pressure differential.	4.1	Containment System Performance
O6A.4	Assumptions Used in the Analysis – Added to analysis description.	4.1.1.1(b) 4.1.2.2	Local Pool Temperature with SRV Discharge Safety Relief Valve Loads
O6A.4.1	General Assumptions – Added specific criteria used in analysis.		2.2 Inputs 2.3 Methodology Appendix A Table A-2 Feedwater Mass and Energy Table 3-4 Results for Stuck Open Relief Valve Event
O6A.5	Results of Transient Analyses – Describes changes in response curve.		Table E-1 Results for Case 2 SORV Event
Table O.6A-1	Suppression Pool Temperature Transcript Summary – Changed to reflect new numbers from new computer code.		Table E-2 SORV Event Chronology
Table O.6A-2	Feedwater System Inventory – Added basis description.		Table A-2 Feedwater Mass and Energy
Table O.6A-2a	Feedwater System Inventory – Added new table with power uprate values.	GE Design Record File	Table A-2 Feedwater Mass and Energy is the initial starting values used for inputs to calculate new values. Nodes are combined in new table. See GE DESIGN RECORD FILE.
Figure 6A4-4	Added new Figure 6A4-4a Drywell Pressure Envelope M3CPT & SHEX (3100 MWt).	GE Design Record File	New envelopes obtained with power uprate analyses.

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Figure 6A4-4	Added new Figure 6A4-4b Drywell Temperature Envelope M3CPT & SHEX (3100 MWt).	GE Design Record File	New envelopes obtained with power uprate analyses.
Figure 6A6-4	Added new Figure 6A6-4a Containment Pressure Envelope M3CPT & SHEX (3100 MWt).	GE Design Record File	New envelopes obtained with power uprate analyses.
Figure 6A6-4	Added new Figure 6A6-4b Containment Temperature Envelope M3CPT & SHEX (3100 MWt).	GE Design Record File	New envelopes obtained with power uprate analyses.
Figure 0.6A-3	Added new Figure 6A-3a Reactor Vessel Pressure, Case 2a, Stuck Open Relief Valve (3100 MWt, SHEX).		Containment Analysis, Figure E-3.
Figure 0.6A-4	Added new Figure 6A-4a Suppression Pool Temperature, Case 2a, Stuck Open Relief Valve (3100 MWt, SHEX).		Containment Analysis, Figure E-4.
9.3.5.2	System Description – Revised information on SLC System.	6.5	Standby Liquid Control System Standby Liquid Control System 2.2 Inputs 2.3.3 Neutron Absorber Weight
9.3.5.3	Safety Evaluation – Revised information on SLC System.	6.5	Standby Liquid Control System Standby Liquid Control System 2.2 Inputs Table 3 List of Key Input Parameters for Pre-Power Uprate Table 4 List of Key Input Parameter for Power Uprate
Table 10.1-1	Steam and Power Conversion Systems Principal Design and Performance Characteristics – Revised turbine rating.		
Table 12.3-9	Technical Support Center Post-LOCA Continuous 30-Day Occupancy Doses – Corrected 30 day dose for uprate conditions.	8.5.3	Post-Accident
15.0	General – Added description of uprate baseline transient analysis.	9.1 9.2 Table 9-2	Reactor Transients Design Basis Accidents Transient Analysis Results for Power Uprate
15.0.2	Analytical Categories – Added description of uprate's impact on Table 15.0-1.	Table 9-2	Transient Analysis Results for Power Uprate
15.0.3.3.2	Input Parameters and Initial Conditions for Analyzed Events – Added new table reference. Split Table 15.0-2 into two tables.	Table 9-2	Transient Analysis Results for Power Uprate
15.0.3.3.4	Results – Added new table references. Added new tables 15.0-1A	Table 9-2	Transient Analysis Results for Power Uprate

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	and 15.0-1B.		
15.0.5	Conformance to Regulatory Guide 1.49 – Revised rated thermal power and described analysis method.	11.4.2.1	Uprate Analysis Basis
15.0.8.1	Description – Revised description of rod line.	Figure 2-1	Power/Flow Operating Map for Power Uprate
15.0.8.2	Maximum Extended Load Line Limit Analysis (MELLLA) – Changed rated thermal power.	Table 1-2	Original and Uprated Plant Operating Conditions
15.0.9.1	Initial Analyses – Changed rated thermal power.	9.1	Reactor Transients
15.0.9.2	Cycle Specific Analyses and Results – Changed rated thermal power.	9.1	Reactor Transients
Table 15.0-1	Results Summary of Transients Events Applicable to BWRs – Divided table into 3 tables.	Table 9-2	Transient Analysis Results for Power Uprate
15.1.1	Loss of Feedwater Heating – Added uprated description of feedwater heating analysis.	Table 9-2	Transient Analysis Results for Power Uprate
15.1.2	Feedwater Controller Failure – Maximum Demand – Added new detail on feedwater controller failure analysis.	Table 9-2	Transient Analysis Results for Power Uprate
15.1.2.3.2	Input Parameters and Initial Conditions – Lowered percent feedwater flow.		
15.1.2.3.3	Results – Raised time to pump trip and steam line peak pressure.		
Table 15.1-3	Sequence of Events for Feedwater Controller Failure, Maximum Demand (Fig. 15.1-3) – Adjusted timing of sequence of events.		
Fig. 15.1-3	Feedwater Controller Failure, Maximum Demand – Replaced figure.	GE Design Record File	
15.2.1	Pressure Regulator Failure – Closed – Added description of uprate reanalysis.	Table 9-2	Transient Analysis Results for Power Uprate
15.2.1.3.3.2	Pressure Regulation Downscale Failure – Changed percent of parameters.	GE Design Record File	
15.2.1.4.2	Pressure Regulation Downscale Failure – Increased peak reactor pressure.	Table 9-2 GE DESIGN RECORD	Transient Analysis Results for Power Uprate

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15.2.2.1.2.2	Generator Load Rejection with Bypass Failure – Added description of reanalyzed load rejection.	Table 9-2	Transient Analysis Results for Power Uprate
15.2.2.3.3.2	Generator Load Rejection with Failure of Bypass – Changed percentage of parameters.	Table 9-2	Transient Analysis Results for Power Uprate
15.2.2.4.2	Generator Load Rejection with Failure of Bypass – Changed peak pressure.	Table 9-2 GE Design Record File	Transient Analysis Results for Power Uprate
15.2.3.1.2.2	Turbine Trip with Failure of the Bypass – Added explanation of basis for analysis.	Table 9-2	Transient Analysis Results for Power Uprate
15.2.3.3.3.2	Turbine Trip with Failure of Bypass – Changed percentage of parameters.	Table 9-2	Transient Analysis Results for Power Uprate
15.2.3.4.2	Turbine Trip with Failure of the Bypass – Changed peak pressure.	Table 9-2 GE Design Record File	Transient Analysis Results for Power Uprate
15.2.9.3.4.2	Added discussion of Suppression Pool peak temperatures.	4.1.1.1(a)	Bulk Pool Temperature 3.4 Alternated Shutdown Event
Table 15.2-1	Sequence of Events for Pressure Regulation Downscale Failure (Fig. 15.2-1) – Adjusted sequence times.	GE Design Record File	
Table 15.2-3	Sequence of Events for Generator Load Rejection with Failure of Bypass (Fig. 15.2-3) – Adjusted sequence times.	GE Design Record File	
Table 15.2-5	Sequence of Events for Turbine Trip with Failure of Bypass (Fig. 15.2-5) – Adjusted sequence times.	GE Design Record File	
Table 15.2-14	Sequence of Events for Failure of RHR Shutdown Cooling (Configuration for Activity CI (a)) – Adjusted sequence times and maximum Suppression Pool temperature.	GE Design Record File	
Table 15.2-14a	Sequence of Events for Failure of RHR Shutdown Cooling (Configuration for Activity CI (b)) – Adjusted sequence times and maximum Suppression Pool temperature.	GE Design Record File	
Table 15.2-15	Input Parameters for Evaluation of Failure of RHR Shutdown	GE Design	

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	Cooling – Modified table parameters for uprate conditions.	Record File	
Fig. 15.2-1	Results of a Pressure Regulation Downscale Failure – Replaced figure.		Transient Analysis, Figure 9-4 Pressure Regulator Downscale Failure – Transient Response.
Fig. 15.2-3	Generator Load Rejection, Trip Scram, Bypass – Off – Replaced figure.		Transient Analysis, Figure 9-1 Load Rejection With Bypass Failure – Transient Response.
Fig. 15.2-5	Turbine Trip, Trip Scram, Bypass – Off, RPT – On – Replaced figure.		Transient Analysis, Figure 9-2 Turbine Trip With Bypass Failure – Transient Response.
Fig. 15.2-12	Activity C1 Alternate Shutdown Cooling Path Utilizing RHR Loops B and C – Revised figure with uprate parameters.		Containment Analysis, Section 3.4 Alternate Shutdown Event Table 3-8
Fig. 15.2-13	RPV Pressure Response – Failure of RHR Shutdown Cooling – Replaced with new figures.		Containment Analysis, Figure F-3 RPV Pressure Response – Failure of RHR Shutdown Cooling – Activity C1(a); Figure F-7 RPV Pressure Response - Failure of RHR Shutdown Cooling – Activity C1(b).
Fig. 15.2-14	Temperature Response – Failure of RHR Shutdown Cooling – Replaced with new figures.		Containment Analysis, Figure F-4 Temperature Response Failure of Shutdown Cooling – Activity C1(a); Figure F-8 Temperature Response Failure of Shutdown Cooling – Activity C1(b).
15.4.1	Rod Withdrawal Error – Low Power – Revised description of rod withdrawal accident / error analysis as impacted by uprate.	Table 9-2	Transient Analysis Results for Power Uprate 1.3.1 Classification of Transient Events
15.4.2	Rod Withdrawal Error at Power - Revised description of rod withdrawal accident / error analysis as impacted by uprate.	Table 9-2	Transient Analysis Results for Power Uprate 1.3.1 Classification of Transient Events
15.4.9.3.3	Results – Corrected the number of fuel rods assumed to fail and MWt.		Radiological Consequence Analysis of Design Basis Accidents Table 1 - Control Rod Drop Accident Inputs and Assumptions
15.4.9.5.1	Fission Product Release from Fuel - Corrected the number of fuel rods assumed to fail and MWt.		Radiological Consequence Analysis of Design Basis Accidents Table 1 - Control Rod Drop Accident Inputs and Assumptions
Table 15.4-11	Control Rod Drop Accident Evaluation Parameters – Revised table with new uprate parameters.		Radiological Consequence Analysis of Design Basis Accidents Table 1 - Control Rod Drop Accident Inputs and Assumptions
Table 15.4-12	Control Rod Drop Accident (Design Basis Analysis) Activity Release to Environment (Curies) – Revised table with new uprate values.		Radiological Consequence Analysis of Design Basis Accidents Table 4 - Activity Released to the Environment, in Curies (CRDA Offsite Dose Evaluation)
Table 15.4-13	Control Rod Drop Accident (Design Basis Analysis) Radiological Effects – Revised table with new uprate values.		Radiological Consequence Analysis of Design Basis Accidents Table 6 – CRDA Offsite and Control Room Doses
15.6.5.5.1	Fission Product Release from Fuel – Updated discussion to uprate	8.3	Radiation Sources in the Reactor Core

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	limits.	11.4.2.10	Combined Effects Radiological Source Terms Radiological Consequence Analysis of Design Basis Accidents
15.6.5.5.2	Fission Product Transport to the Environment – Updated discussion to uprate limits.		Radiological Consequence Analysis of Design Basis Accidents, 3.5.2 – Airborne Releases
Table 15.6-5	Loss-of-Coolant Accident Parameters Tabulated for Postulated Accident Analysis – Revised table with uprate conditions.		Radiological Consequence Analysis of Design Basis Accidents, Table 24 – Loss of Coolant Accident Inputs and Assumptions
Table 15.6-6	Loss-of-Coolant Accident (Design Basis Analysis) Activity Release To Environment (Curies). Deleted table. Uprate analysis has no equivalent.	Deleted	Per GE, this is an outdated figure. It has no historical value or useful information and is being deleted.
Table 15.6-7	Loss-of Coolant Accident (Design Basis Analysis) Radiological Effects – Replaced table with new uprated values.		Radiological Consequence Analysis of Design Basis Accidents, Table 25 – Loss of Coolant accident Offsite and Control Room Doses
Fig. 15.6.5-1,2,3	Added 3 new figures: Containment Release Paths, 0-24 Seconds Post-LOCA; Containment Release Paths, 24-724 Seconds Post-LOCA; Containment Release Paths, 724 Seconds – 720 Hours Post-LOCA.		Radiological Consequence Analysis of Design Basis Accidents, Figure 1 – Containment Release Paths, 0 – 24 Seconds Post-LOCA; Figure 2 – Containment Release Paths, 24 – 724 Seconds Post-LOCA; Figure 3 – Containment Release Paths, 724 Seconds – 720 Hours Post-LOCA.
15.8.3.1	Analysis Method – Updated computer codes used for ATWS simulation.	9.4 Table 9-2	References Transient Analysis Results for Power Uprate
15.8.3.3	Primary Analysis Inputs – Changed the number of SRVs required to meet the peak vessel pressure acceptance criteria from 15 to 11. Also updated MWt and core flow assumptions. Removed two minute delay on operator actions for tripping recirc pumps, reducing reactor water level, and initiating SLC after Suppression Pool temperature reaches 110 deg. F. Update core flow conditions used in ATWS analysis.	3.2	Reactor Overpressure Protection Transient Analysis, Table 9-1 Parameters Used for Transient Analysis
Table 15.8-1	Initial Conditions for ATWS Analysis – Updated table.	GE Design Record File	
Table 15.8-2	ATWS – Equipment Performance Characteristics – Updated table.	Table 5-1 GE Design Record File	Analytical Limits for Setpoints

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Table 15.8-3	Typical Sequence of Events of ATWS Main Steamline Isolation Valve Closure Event – Updated table with new sequence times.	GE Design Record File	
Table 15.8-4	Summary of Peak Results – Updated with new uprate values.	GE Design Record File	