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Constellation Energy
Generation Group

January 25, 2006

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Response to Requests for Additional Information Regarding Topics Described in Meeting Minutes

By letter dated July 7, 2005, as supplemented by letters dated August 15 and September 30, 2005, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) submitted an application requesting authorization to increase the maximum steady-state thermal power level at the R.E. Ginna Nuclear Power Plant from 1520 megawatts thermal (MWt) to 1775 MWt.

Over the period spanning September 2005 through January 24, 2006, the NRC staff continually engaged the Ginna Extended Power Uprate Project Team with discussions involving the Extended Power Uprate (EPU) Licensing Submittals. Through out the course of these discussions both staff and station personnel have kept meeting minutes. Ginna has reviewed the staff minutes promulgated on the public docket as well as our own records to ensure all information requested by the staff has been provided. Additionally, the NRC performed an audit of Westinghouse Electric Company. During that audit, follow-up informational items were requested.

The purpose of this letter is to provide formal documentation of any outstanding requests received to date as well as our response. Our responses are contained in five attachments. Each attachment represents a specific meeting.

Attachment 1 contains the responses to a September 15, 2005 meeting and can be associated with the NRC's request for additional information dated October 24, 2005.

Attachment 2 contains the responses to a November 17, 2005 meeting and can be associated with an NRC letter dated December 23, 2005.

Attachment 3 contains the responses to a December 21, 2005 phone call in which the NRC requested clarification of information provided by Constellation by letter dated December 7, 2005.

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Attachment 4 contains the response to a December 22, 2005 phone call in which the NRC requested clarification of information provided by Constellation by letter dated November 21, 2005. (Note: This may be considered supplementary information to an earlier license amendment request dated April 29, 2005 regarding Main Feedwater Isolation Valves).

Attachment 5 contains the response to a January 11, 2006 phone call where in the staff requested supplementary information regarding Loss of Coolant Accidents (Note: This information may be considered supplementary information to an earlier license amendment request dated April 29, 2005 regarding the use of Westinghouse Best-Estimate Large Break LOCA Methodology).

The responses do not include any new regulatory commitments.

If you have any questions, please contact George Wrobel at (585) 771-3535 or george.wrobel@constellation.com.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF WAYNE :

I, Mary G. Korsnick, being duly sworn, state that I am Vice President – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 25 day of January, 2006.

WITNESS my Hand and Notarial Seal:

Sharon L Miller
Notary Public

My Commission Expires:
12-21-06

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 20 06

Attachrnents

Cc: S. J. Collins, NRC
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Resident Inspector, NRC

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RESPONSES TO NRC MEETING MINUTES FROM SEPTEMBER 15, 2005 MEETING (See NRC letter dated 10/24/2005)

Question #1

Assumptions and other initial plant conditions used in each non-LOCA transient analysis.

Response

The requested data is currently tabulated in the several tables in License Amendment Request (LAR) Sections 1.1, "Nuclear Steam Supply System Parameters" and Section 2.8.5.0, "Non-LOCA Analyses Introduction" and also in a response to a previous RAI from the November 3, 2005 letter numbered "Rx Sys T&H RAI #3" which contains tabulated non-LOCA analysis assumptions. The following information is provided as clarification to the material previously transmitted.

LR Table 1.1 contains the following nominal conditions,

Table 1-1
NSSS PCWG Parameters for Ginna Station Uprate Program

		EPU(h)			
Thermal Design Parameters	**Current(c)**	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power	100	119.5	119.5	119.5	119.5
MWt	1520	1817	1817	1817	1817
10^6 Btu/hr	5,186	6,200	6,200	6,200	6,200
Reactor Power MWt	1520	1811	1811	1811	1811
10^6 Btu/hr	5,186	6,179	6,179	6,179	6,179
Thermal Design Flow, loop gpm	85,100	85,100	85,100	85,100	85,100
Reactor 10^6lb/hr	64.6	65.8	65.8	64.8	64.8
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.5(a)	6.5(a)	6.5(a)	6.5(a)	6.5(a)
Reactor Coolant Temperature, °F					
Core Outlet	607.8	605.5	605.5	616.2	616.2
Vessel Outlet	603.9	601.0	601.0	611.8	611.8
Core Average	576.9	568.8	568.8	580.3	580.3
Vessel Average	573.5	564.6	564.6	576.0	576.0
Vessel/Core Inlet	543.1	528.3	528.3	540.2	540.2
Steam Generator Outlet	541.3	528.0	528.0	539.9	539.9
Steam Generator					
Steam Outlet Temperature, °F	513.8	506.5	503.1	518.8(b)	515.4
Steam Outlet Pressure, psia	770	722	700	804(b)	781
Steam Outlet Flow, 10^6 lb/hr total	6.60	7.42/7.88	7.41/7.87	7.44/7.9(b)	7.43/7.89
Feed Temperature, °F	425	390/435	390/435	390/435	390/435
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10	0.10
Design FF, hr. sq. ft. °F/Btu	0.00015	0.00015	0.00015	0.00015	0.00015
Tube Plugging Level (%)	0	0	10	0	10
Zero Load Temperature, °F	547	547	547	547	547
Hydraulic Design Parameters					
Pump Design Point, Flow (gpm)/Head (ft.)	90,000/252				
Mechanical Design Flow, gpm	101,200				
Minimum Measured Flow, gpm/total	177,300				

Notes:

a. Core bypass flow includes 2.0% due to TPR.

b. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 855 psia, steam temperature of 525.9°F, and steam flow of 7.92 x 10^6 lb/hr total should be assumed. This envelopes the possibility that the steam generator could perform better than expected.

c. Current parameters obtained from Tables 4.4.1 and 5.4.2 of UFSAR.

LR Tables 2.8.5.0-2 and 2.8.5.0-7 contain initial conditions assumed for the Non-LOCA accident analyses,

Table 2.8.5.0-2
Non-LOCA Plant Initial Condition Assumptions

Parameter	RTDP	Non-RTDP	Notes
NSSS Power (MWt)	1817.0	1817.0	1
Nominal Total Net RCP Heat (MWt)	6.0	6.0	1, 2, 3
Maximum Full-Power Vessel T_{avg} (°F)	576.0	576.0 ± 4.0	1, 4
Minimum Full-Power Vessel T_{avg} (°F)	564.6	564.6 ± 4.0	1, 4
No-Load RCS Temperature (°F)	547.0	547.0	1, 4
Pressurizer Pressure (psia)	2250	2250 ± 60	1
Steam Flow (lbm/hr)	see Note 5	see Note 5	5
Steam Pressure (psia)	see Note 5	see Note 5	5
Feedwater Temperature (°F)	390 to 435	390 to 435	1
Pressurizer Water Level (% span)	see Note 6	see Note 6	6
Steam Generator Water Level (% NRS)	see Note 7	see Note 7	7

Notes:

1. See Table 1-1 in Section 1.0 of Licensing Report.
2. Total RCP heat input minus RCS thermal losses.
3. A maximum net RCP heat of 10 MWt was conservatively assumed in some non-RTDP analyses, e.g., loss-of-normal feedwater.
4. All analyses assumed a programmed no-load T_{avg} of 547°F. For the events initiated from a no-load condition (rod withdrawal from subcritical, steam line break, rod ejection, boron dilution), the use of the no-load temperature as the initial temperature bounded the case of startup operations at Ginna with a temperature less than 547°F.
5. The nominal steam flow rate and steam pressure depended on other nominal conditions. See Table 1-1 of Licensing Report.
6. The nominal/programmed pressurizer water level varied linearly from 20% of span at the no-load T_{avg} of 547°F to either 44.3% of span at the minimum full-power T_{avg} of 564.6°F or 60% of span at the maximum full power T_{avg} of 576°F. The programmed level remained constant at the full-power T_{avg} level for T_{avg} values greater than the full-power T_{avg}. An uncertainty of ±5% of span was applied when conservative.
7. The programmed steam generator water level modeled in the analyses was a constant 52% narrow range span (NRS) for all power levels. An uncertainty of -4% NRS/+8% NRS was applied when conservative.

Attachment 1

Table 2.8.5.0-7 **Summary of Initial Conditions and Computer Codes Used**							
Accident	**Computer Codes Used**	**DNB Correlation**	**RTDP**	**Initial Power, %**	**Vessel Coolant Flow, gpm**	**Vessel Average Coolant Temp, °F**	**RCS Pressure, psia**
Decrease in Feedwater Temperature	Event bounded by the excessive-load-increase event						
Increase in Feedwater Flow	RETRAN VIPRE	WRB-1 (HFP) W-3 (HZP)	Yes (HFP) No (HZP)	100 0 (1817 MWt - NSSS power)	177,300 (HFP) 170,200 (HZP)	576.0 (HFP) 547.0 (HZP)	2250
Excessive Load Increase	N/A	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
Inadvertent Opening of a Steam Generator Relief/Safety Valve	Event bounded by the steam system piping failure event.						
Rupture of a Steam Pipe – Zero Power Core Response	RETRAN VIPRE	W-3	No	0 (1817 MWt - NSSS power)	170,200	547.0	2250
Rupture of a Steam Pipe – Full Power Core Response	RETRAN VIPRE	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
Combined Steam Generator ARV and Feedwater Control Valve Failures	RETRAN	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	Event bounded by the loss-of-external-electrical-load event.						
Loss-of-External-Electrical Load	RETRAN	WRB-1	N/A (pressure) Yes (DNB)	100 (pressure) 100 (DNB) (1817 MWt - NSSS power)	170,200 (pressure) 177,300 (DNB)	580.0 (pressure) 576.0 (DNB)	2190 (pressure) 2250 (DNB)
Turbine Trip	Event bounded by the loss-of-external-electrical-load event.						
Loss-of-Condenser Vacuum	Event bounded by the loss-of-external-electrical load event.						
Loss-of-Offsite-ac-Power to the Station Auxiliaries	RETRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	572.0	2310

Table 2.8.5.0-7 (cont.)
Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Vessel Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Loss-of-Offsite-ac-Power to the Station Auxiliaries	RETRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	572.0	2310
Loss-of-Offsite-ac-Power to the Station Auxiliaries	RETRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	572.0	2310
LONF	RETRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	560.6	2310
Feedwater System Pipe Breaks	RETRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	580.0	2190
Flow Coastdown Accident	RETRAN VIPRE	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
Locked Rotor Accident	RETRAN VIPRE	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	580.0	2310
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	W-3(1) WRB-1(2)	No	0 (1811 MWt - core power)	76,420(3)	547	2190
Uncontrolled RCCA Withdrawal at Power	RETRAN	WRB-1	Yes (DNB) N/A (Pressure)	100 (DNB/MSS Pressure) 60 (DNB/MSS Pressure) 10 (DNB/MSS Pressure) 8 (RCS Pressure) (1817 MWt - NSSS power)	177,300 (DNB/MSS) 170,200 (RCS Press.)	576.0 (100%) 564.4 (60%) 549.9 (10%) 553.9 (8%)	2250 (DNB/MSS) 2190 (RCS Press.)
Startup of an Inactive RCL	See Licensing Report Section 2.8.5.4.4						
CVCS System Malfunction	N/A	N/A	N/A	100 (Mode 1) 5 (Mode 2) 0 (Mode 6)	N/A	580.0 (Mode 1) 547.0 (Mode 2) 140.0 (Mode 6)	2,250 (Modes 1 and 2) 14.7 (Mode 6)
RCCA Ejection	TWINKLE FACTRAN	N/A	N/A	100 (HFP) 0 (HZP) (1811 MWt - core power)	170,200 (HFP) 76,420(3) (HZP)	580.0 (HFP) 547.0 (HZP)	2190
RCCA Drop	LOFTRAN(4) ANC VIPRE	WRB-1	Yes	100	177,300	576.0	2250

Attachment 1

Table 2.8.5.0-7 (cont.) Summary of Initial Conditions and Computer Codes Used							
Accident	**Computer Codes Used**	**DNB Correlation**	**RTDP**	**Initial Power, %**	**Vessel Coolant Flow, gpm**	**Vessel Average Coolant Temp, °F**	**RCS Pressure, psia**
Inadvertent Opening of a Pressurizer Safety or Relief Valve	RETRAN	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
CVCS System Malfunction	N/A	N/A	N/A	100 (Mode 1) 5 (Mode 2) 0 (Mode 6)	N/A	580.0 (Mode 1) 547.0 (Mode 2) 140.0 (Mode 6)	2250 (Modes 1 and 2) 14.7 (Mode 6)
RCCA Ejection	TWINKLE FACTRAN	N/A	N/A	100 (HFP) 0 (HZP) (1811 MWt - core power)	170,200 (HFP) 76,420(3) (HZP)	580.0 (HFP) 547.0 (HZP)	2190
RCCA Drop	LOFTRAN(4) ANC VIPRE	WRB-1	Yes	100	177,300	576.0	2250
Inadvertent Opening of a Pressurizer Safety or Relief Valve	RETRAN	WRB-1	Yes	100 (1817 MWt - NSSS power)	177,300	576.0	2250
ATWS	LOFTRAN	N/A	N/A	100 (1817 MWt - NSSS power)	170,200	574.5(5)	2250

Notes:
1. Below the first mixing vane grid.
2. Above the first mixing vane grid.
3. Single-loop flow = 0.449 * TDF.
4. The LOFTRAN portion of the analysis was generic; the DNB evaluation performed with VIPRE utilized the plant-specific values presented.
5. The ATWS analysis supports a nominal average temperature of 576.0 °F (initial average temperature of 574.5 °F, plus a rod control system deadband of 1.5 °F)

Attachment 1

Tables 2.8.5.0-4 and 2.8.5.0-5 contain trip setpoints and delay times assumed,

Table 2.8.5.0-4
Overtemperature and Overpower ΔT Setpoints

Allowable Full-Power T_{avg} Range	564.6° to 576.0°F
K_1 (safety analysis value)	1.30
K_2	0.00093/psi
K_3	0.0185/°F
K_4 (safety analysis value)	1.15
K_5	0.0014/°F (1)
K_6	0.00/°F
T′	564.6° to 576.0°F (2)
P′	2250 psia
f(ΔI) Deadband (3)	-14% ΔI (4) to +6% ΔI
f(ΔI) Negative Gain (3)	-3.08%/%ΔI (4)
f(ΔI) Positive Gain (3)	+2.27%/%ΔI
High-Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	2425 psia
Low-Pressurizer Pressure Reactor Trip Setpoint (safety analysis value)	1775 psia

Notes:
1. K5 = 0.0014/°F is valid for increasing T_{avg}. For decreasing T_{avg}, K5 = 0.0/°F.
2. Value to be set equal to or less than the full power operating T_{avg} chosen.
3. The f(ΔI) penalty is implicitly assumed in the non-LOCA safety analyses.
4. Value supported by non-LOCA transient analysis. Value will change based on fuel rod design analysis.

Table 2.8.5.0-5Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.1.1	Decrease in Feedwater Temperature	N/A	N/A	N/A
15.1.2	Increase in Feedwater Flow	High-High Steam Generator Water Level Feedwater Regulator Valve Closure	100% NRS	22.0
15.1.3	Excessive Load Increase	N/A	N/A	N/A
15.1.4	Inadvertent Opening of a Steam Generator Relief/Safety Valve	(1)		
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	High-High Steam Flow Setpoint	~155% of nominal	2.0
		Low Steam Pressure Safety Injection (SI) Setpoint	327.7 psia (lead/lag = 12/2)	2.0
		Steam Line Isolation Delay from SI Coincident with High-High Steam Flow	N/A	7.0
		Feedwater Isolation Delay from SI	N/A	32.0
		SI Pumps at Full Flow Following SI Signal (with/without offsite power)	N/A	12.0/22.75
	Steam System Piping Failure – Full Power (Core response only)	OPΔT reactor trip	Table 2.8.5.0-4	10.0(2)
15.1.6	Combined Steam Generator ARV and Feedwater Control Valve Failures	High-High Steam Generator Water Level Feedwater Regulator Valve Closure	100% NRS	22.0
		OPΔT Reactor Trip	Table 2.8.5.0-4	10.0(2)
		Low-Pressurizer Pressure Safety Injection	1715.0 psia	32.0
15.2.1	Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	(3)		
15.2.2	Loss-of-External-Electrical Load	High-Pressurizer Pressure Reactor Trip	2425 psia	2.0
		OTΔT Reactor Trip	Table 2.8.5.0-4	7.0(2)

Table 2.8.5.0-5Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
15.2.3	Turbine Trip	(3)		
15.2.4	Loss-of-Condenser Vacuum	(3)		
15.2.5	Loss-of-Offsite-ac Power to the Station Auxiliaries	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level Auxiliary Feedwater (AFW) Pump Start	0% NRS	60.0
15.2.6	LONF	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level AFW Pump Start	0% NRS	60.0
15.2.7	Feedwater System Pipe Breaks	Low-Low Steam Generator Water Level Reactor Trip	0% NRS	2.0
		Low-Low Steam Generator Water Level AFW Pump Start	0% NRS	60 & 600
15.3.1	Flow Coastdown Accidents	Low RCL Flow Reactor Trip	87%	1.0
		RCP Undervoltage Reactor Trip	N/A	1.5
		RCP Underfrequency Reactor Trip	57 Hz	1.4
15.3.2	Locked Rotor Accident	Low RCL Flow Reactor Trip	87%	1.0
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Power-Range High Neutron Flux Reactor Trip (Low Setting)	35%	0.5
15.4.2	Uncontrolled RCCA Withdrawal at Power	Power-Range High Neutron Flux Reactor Trip (High Setting)	115%	0.5
		OTΔT Reactor Trip	Table 2.8.5.0-4	7.0(2)
		High Pressurizer Pressure Reactor Trip	2425 psia	2.0

Table 2.8.5.0-5Summary of RPS and ESFAS Functions Actuated				
UFSAR Section	**Event Description**	**RPS or ESFAS Signal(s) Actuated**	**Analysis Setpoint**	**Delay (sec)**
15.4.3	Startup of an Inactive RCL	N/A	N/A	N/A
15.4.4	Chemical and Volume Control System Malfunction (Boron Dilution)	OTΔT Reactor Trip	Table 2.8.5.0-4	7.0[(2)]
15.4.5	RCCA Ejection	Power-Range High Neutron Flux Reactor Trip (Low and High Settings)	35% (low setting)	0.5
			118% (high setting)	0.5
15.4.6	RCCA Drop	Low-Pressurizer Pressure Reactor Trip	Note 4	2.0
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	OTΔT Reactor Trip	Table 2.8.5.0-4	10.0[(2)]
15.8	ATWS	ATWS Mitigation System Actuation Circuitry (AMSAC) – Turbine Trip (TT), AFW Pump Start (AFW)	N/A	30 (TT) 60 (AFW)

Notes:

1. Transient bounded by steam system piping failure (UFSAR, Section 15.1.5).
2. Modeling the OTΔT and OPΔT reactor trips included a time constant (first order lag) of 2.0 seconds for the RTDs and a filter (lag) of 3.5 (or 6.0) seconds on the hot-leg temperature measurement. The RTD lag accounted for the response of the RTDs and the RTD electronic filter (if any). In addition, after the overtemperature or overpower setpoint was reached, a delay of 1.5 (or 2.0) seconds was assumed to account for electronic delays, reactor trip breakers opening, and RCCA gripper release.
3. Transient bounded by loss-of-external-electrical load (UFSAR, Section 15.2.2).
4. The generic two-loop dropped RCCA analysis, applicable to Ginna, modeled the low-pressurizer pressure reactor trip setpoint as a "convenience trip." The cases that actuated this function assumed dropped rod and control bank worth combinations that were non-limiting with respect to DNB. The fact that the plant-specific low-pressurizer pressure setpoint (1775 psia) was lower than the value assumed in the generic analysis (1860 psia) did not invalidate the applicability of the generic two-loop statepoints to Ginna. Therefore, the low-pressurizer pressure reactor trip setpoint value that was used in the generic two-loop dropped RCCA analysis (1860 psia) did not represent an analytical limit for this function for Ginna.

Table 2.8.5.0-6 and Figure 2.8.5.0-6 contain reactivity feedback assumptions.

Table 2.8.5.0-6
Core Kinetics Parameters and Reactivity Feedback Coefficients

Parameter	Beginning of Cycle (Minimum Feedback)	End of Cycle (Maximum Feedback)
MTC, pcm/°F	5.0 (< 70% RTP)[1] 0.0 (≥ 70% RTP)	N/A
Moderator Density Coefficient, Δk/(g/cc)	N/A	0.45
Doppler Temperature Coefficient, pcm/°F	-0.91	-2.90
Doppler-Only Power Coefficient, pcm/%power (Q = power in %)	-12.0 + 0.045Q	-24.0 + 0.100Q
Delayed Neutron Fraction	0.0072 (maximum)	0.0043 (minimum)
Minimum Doppler Power Defect, pcm		
– RCCA Ejection	1000	950
– RCCA Withdrawal from Subcritical	1100	N/A

Note:
1. RTP ≡ Rated Thermal Power

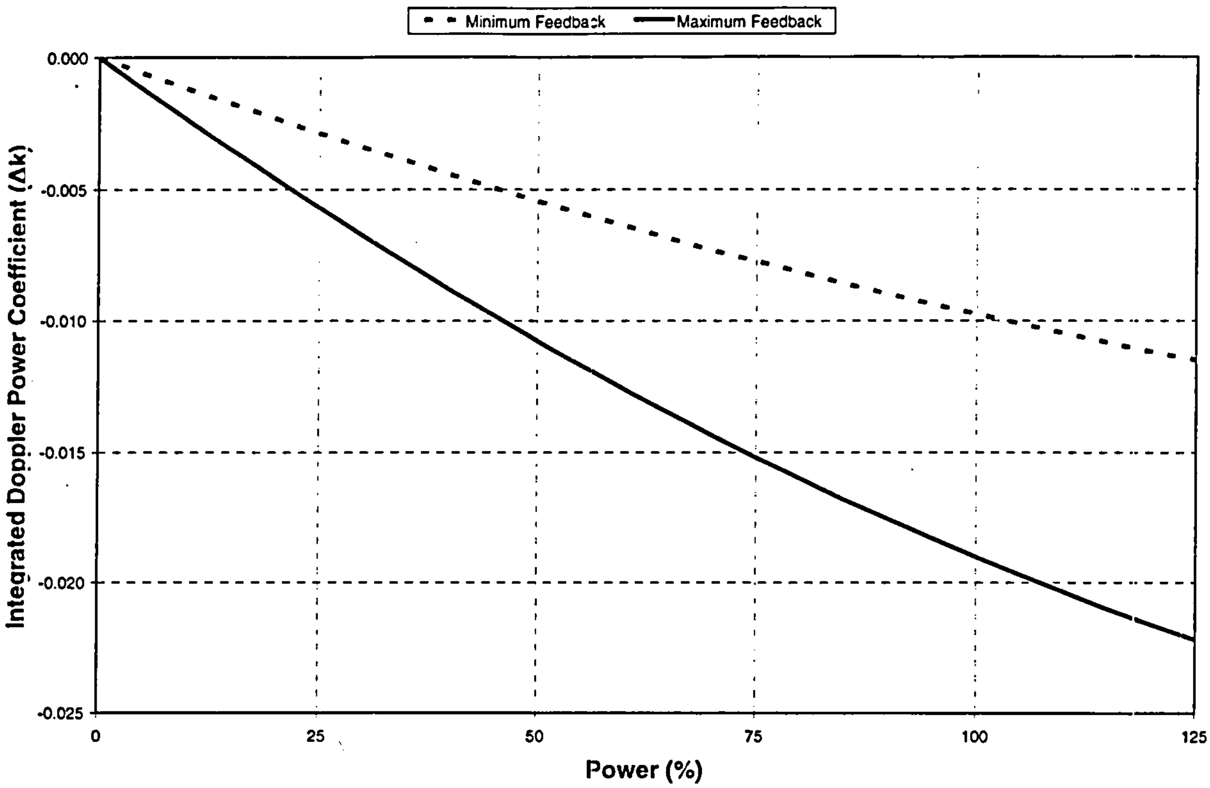

Figure 2.8.5.0-6
Integrated DPC Used in Non-LOCA Transient Analyses

Attachment 1

Analysis results and design limits are tabulated in Table 2.8.5.0-1.

Table 2.8.5.0-1
Non-LOCA Analysis Limits and Analysis Results

UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.1.1	Decrease in Feedwater Temperature	(1)	N/A	N/A
15.1.2	Increase in Feedwater Flow	Minimum DNBR (RTDP, WRB-1) (HFP) Minimum DNBR (STDP, W-3) (HZP)	1.38 (HFP) 1.613 (HZP)	1.60 (HFP) (2) (HZP)
15.1.3	Excessive Load Increase	Minimum DNBR (RTDP, WRB-1)	1.38	> 1.38
15.1.4	Inadvertent Opening of a Steam Generator Relief/Safety Valve	Bounded by Steam Line Break (UFSAR, Section 15.1.5)	N/A	N/A
15.1.5	Steam System Piping Failure – Zero Power (Core response only)	Minimum DNBR (non-RTDP, W-3)	1.566 (OFA)	2.58 (OFA)
	Steam System Piping Failure – Full Power (Core response only)	Minimum DNBR (RTDP, WRB-1 correlation) (typical/thimble)	1.38/1.38 (422V+)	1.392/1.395 (422V+)
		Peak Linear Heat Generation (kW/ft)	22.7(3)	22.67
15.1.6	Combined Steam Generator ARV and Feedwater Control Valve Failures	Minimum DNBR (RTDP, WRB-1)	1.38	1.52
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	Bounded by Loss-of-External-Electrical Load (UFSAR, Section 15.2.2)	N/A	N/A
15.2.2	Loss-of-External-Electrical Load	Minimum DNBR (RTDP, WRB-1)	1.38	1.61
		Peak RCS Pressure, psia	2748.5	2746.8
		Peak MSS Pressure, psia	1208.5	1208.0

Table 2.8.5.0-1 Non-LOCA Analysis Limits and Analysis Results				
UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
15.2.3	Turbine Trip	Bounded by Loss-of-External-Electrical Load (UFSAR, Section 15.2.2)	N/A	N/A
15.2.4	Loss-of-Condenser Vacuum	Bounded by Loss-of-External-Electrical Load (UFSAR, Section 15.2.2)	N/A	N/A
15.2.5	Loss-of-Offsite-AC-Power to the Station Auxiliaries	Maximum pressurizer mixture volume, ft^3	800	635
15.2.6	Loss-of-Normal Feedwater	Maximum Pressurizer Mixture Volume, ft^3	800	537
15.2.7	Feedwater System Pipe Breaks	Margin to Hot Leg Saturation, °F	0.0	2
15.3.1	Flow Coastdown Accident – PLOF(4)	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.601/1.597 (422V+)
	Flow Coastdown Accident – CLOF(5)	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.489/1.491 (422V+)
	Flow Coastdown Accident – UF(6)	Minimum DNBR (RTDP, WRB-1) (typical/thimble)	1.38/1.38 (422V+)	1.385/1.392 (422V+)
15.3.2	Locked Rotor Accident	Peak RCS Pressure, psia	2997	2782
		Peak Cladding Temperature, °F	2700	1924.6 (422V+)
		Maximum Zirc-Water Reaction, %	16	0.53 (422V+)
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Minimum DNBR Below First Mixing Vane Grid (non-RTDP, W-3 correlation) (typical/thimble)	1.447/1.447 (422V+)	1.987/2.238 (422V+)
		Minimum DNBR Above First Mixing Vane Grid (non-RTDP, WRB-1 correlation) (typical/thimble)	1.302/1.302 (422V+)	1.957/1.951 (422V+)
		Maximum Fuel Centerline Temperature, °F	4800(7)	2108 (422V+)
15.4.2	Uncontrolled RCCA Withdrawal at Power	Minimum DNBR (RTDP, WRB-1)	1.38	1.384
		Peak RCS Pressure, psia	2748.5	2748.1
		Peak MSS Pressure, psia	1208.5	1207.7
15.4.3	Startup of an Inactive Reactor Coolant Loop,	No Analysis Performed (See Section Licensing	N/A	N/A

Table 2.8.5.0-1
Non-LOCA Analysis Limits and Analysis Results

UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
	(RCL)	Report 2.8.5.4.4)		
15.4.4	Chemical and Volume Control System (CVCS) Malfunction (Boron Dilution)	Minimum Time to Loss of Shutdown Margin, Minutes	15	30.3 (Mode 1 manual)
			15	33.3 (Mode 1 auto)
			15	25.1 (Mode 2)
			30	32.0 (Mode 6)
15.4.5	Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)	Maximum Fuel Pellet Average Enthalpy, cal/g	200	151.8 (BOC-HZP)(8) 177.9 (BOC-HFP)(9) 155.1 (EOC-HZP)(10) 177.2 (EOC-HFP)(11)
		Maximum Fuel Melt, %	10	0.00 (BOC-HZP)(12) 6.62 (BOC-HFP)(12) 0.00 (EOC-HZP)(13) 9.00 (EOC-HFP)(13)
		Peak RCS Pressure, psia	Generically addressed in Reference 15	
15.4.6	RCCA Drop	Minimum DNBR (RTDP, WRB-1)	1.38	> 1.38
		Peak Linear Heat Generation (kW/ft)	22.7(3)	< 22.7
		Peak Uniform Cladding Strain (%)	1.0	< 1.0
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Minimum DNBR (WRB-1)	1.38	1.49
15.8	ATWS	Peak RCS Pressure, psig	3200	3,193

Table 2.8.5.0-1 Non-LOCA Analysis Limits and Analysis Results				
UFSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case

Notes:

1. Event bounded by the steam system piping failure at full power event. See Section 2.8.5.1.1, Licensing Report.
2. Bounded by zero power steam line break.
3. Corresponds to a UO_2 fuel melting temperature of 4700°F.
4. PLOF ≡ partial loss of flow (one-loop flow coastdown).
5. CLOF ≡ complete loss of flow (two-loop flow coastdown).
6. UF ≡ underfrequency (frequency decay of RCP power supply)
7. UO_2 fuel melting temperature corresponding to a burnup of ~48,276 MWd/MTU.
8. Beginning of cycle HZP.
9. Beginning of cycle HFP.
10. End of cycle HZP.
11. End of cycle HFP.
12. Fuel melting temperature = 4900°F
13. Fuel melting temperature = 4800°F

Furthermore, the following RAI was included in the November 3,2 005 NRC letter and the response to the RAI numbered "Rx Sys T&H RAI #3" contained the following tabulated non-LOCA analysis assumptions.

Rx Sys T&H Question #3

Provide a tabulation of the thermal design parameters and compare them to values assumed in safety analyses to demonstrate that the safety analyses assumptions are conservative.

Response

The thermal design parameters associated with the Extended Power Uprate are given in Table 1-1 of the LR. A range of operating conditions have been defined which include 0% to 10% steam generator tube plugging, a vessel average temperature range of 564.6°F to 576.0°F and a feedwater temperature range of 390°F to 435°F. The values assumed in the non-LOCA safety analyses are given in Table 2.8.5.0-2 of the LR. The following table lists the thermal design parameters from Table 1.1 along with the analysis values:

Parameter	**Design Value (from Table 1.1)**	**Analysis Value for RTDP Events (from Table 2.8.5.0-2**	**Analysis Value for non-RTDP Events (from Table 2.8.5.0-2)**
NSSS Power (MWt)	1817.0	1817.0	1817.0
Core Power (MWt)	1811.0	1811.0	1811.0
Max. Full Power Vessel Tavg (°F)	576.0	576.0	576.0±4.
Min. Full Power Vessel Tavg (°F)	564.6	564.6	564.6±4.
No-Load RCS Temperature (°F)	547.0	547.0	547.0
Pressurizer Pressure (psia)	2250.0	2250.0	2250.0±60.
Minimum Measured Flow (gpm)	177300.	177300.**	Not Applicable
Thermal Design Flow (gpm)	170200.	Not Applicable	170200. *

* From Section 2.8.5.0.3

** This value is not given in the analysis section of the LR. The TDF is given as 170200 gpm in Section 2.8.5.0.3 and Section 2.8.5.0.3 further notes that "the RCS flow allowance is represented by the difference between TDF and MMF. The flow uncertainty is equivalent to 4%." The MMF value assumed in the analyses is 177300 gpm.

Question #2

Methodology and modeling used for each calculation;

Response

Discussion of general methodology and modeling that is applicable to all the Non-LOCA analyses is provided in LR Sections 2.8.5.0.1 to 2.8.5.0.13. Additional Non-LOCA analysis-specific discussions are contained in the "Regulatory Evaluation" sub-sections of the LR Sections 2.8.5.1 to 2.8.5.7.

Question #3

Verification that the methodology has been approved by the NRC for each application, and provide any differences from the code methodologies;

Response

LR Table A.1-1 presents an overview of the Safety Evaluation Reports (SER) by codes and methods. For each SER, the applicable report subsections and appendix subsections are listed. LR Tables A.2-1, A.3-1, A.4-1 and A.5-1 address compliance with the limitations, restrictions, and conditions specified in the approving safety evaluation of the applicable codes and methods.

Table A.1-1: Safety Evaluation Report Compliance Summary

No.	Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s)	Limitation, Restriction, Condition	Report Section	Appendix Section
1.	Non-LOCA Thermal Transients	WCAP-7908-A (Reference A.1-1) / September 30, 1986	FACTRAN	Yes	2.8.5.4.1 2.8.5.4.6	A.2
2.	Non-LOCA Safety Analysis	WCAP-14882-P-A (Reference A.1-2) / February 11, 1999	RETRAN	Yes	2.8.5.2.4 2.8.5.4.2	A.3
3.	Non-LOCA Safety Analysis	WCAP-7907-P-A (Reference A.1-3) / July 29, 1983	LOFTRAN	Yes	2.8.5.7	A.4
4.	Neutron Kinetics	WCAP-7979-P-A (Reference A.1-4) / July 29, 1974	TWINKLE	None for Non-LOCA Transient Analysis	2.8.5.4.1 2.8.5.4.6	Not Applicable
5.	Multi-dimension al Neutronics	WCAP-10965-P-A (Reference A.1-5) / June 23, 1986	ANC	None for Non-LOCA Transient Analysis	2.8.5.4.3	Not Applicable
6.	Non-LOCA Thermal/H ydraulics	WCAP-14565-P-A (Reference A.1-6) / January 19, 1999	VIPRE	Yes	2.8.5.4.1 2.8.5.4.3	A.5
7.	Steam Generator Tube Rupture	WCAP-14882-P-A (Reference A.1-2) / February 11, 1999	RETRAN	None for Steam Generator Tube Rupture	2.8.5.6.2	A.3

References

A.1-1 WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989.

A.1-2 WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel, et al., April 1999.

A.1-3 WCAP-7907-P-A, "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984.

A.1-4 WCAP-7979-P-A, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, Jr. and R. F. Barry, January 1975.

A.1-5 WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

A.1-6 WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999.

A.2 FACTRAN for Non-LOCA Thermal Transients

Table A.2-1: FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions	
1.	***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."*** Justification The FACTRAN code was used in the analyses of the following transients for Ginna: Uncontrolled RCCA Withdrawal from a Subcritical Condition (Ginna UFSAR Section 15.4.1) and RCCA Ejection (Ginna UFSAR Section 15.4.5). Initial fuel temperatures used as FACTRAN input in the RCCA Ejection analysis were calculated using the NRC-approved PAD 4.0 computer code, as described in WCAP-15063-P-A (Reference A.2-1). As indicated in WCAP-15063-P-A, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures.
2.	***"Table 2 presents the guidelines used to select initial temperatures."*** Justification In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: loss of flow, locked rotor, and rod ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA ejection analysis for Ginna. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (Reference A.2-1), include uncertainties and are conservatively high. FACTRAN was not used in the loss of flow and locked rotor analyses.
3.	***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."*** Justification The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA withdrawal from a subcritical condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the RCCA ejection transient, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.
4.	***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."*** Justification Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Ginna {RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5). Therefore, additional justification is not required.
5.	***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."*** Justification At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Ginna (RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5)).
6.	***"Although time-independent mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-dependent mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."***

Table A.2-1: FACTRAN for Non-LOCA Thermal Transients

Limitations, Restrictions, and Conditions	
	Justification The two transients that were analyzed with FACTRAN for Ginna (RCCA withdrawal from a subcritical condition (Ginna UFSAR Section 15.4.1) and RCCA ejection (Ginna UFSAR Section 15.4.5)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA withdrawal from a subcritical condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.
7.	***"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."*** Justification The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Ginna.

Reference

A.2-1 WCAP-15063-P-A, Revision 1 (with Errata) "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," J. P. Foster and S. Sidener, July 2000.

A.3 RETRAN for Non-LOCA Safety Analysis

Table A.3-1: RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."*** Justification The transients listed in Table 1 of the SER are: • Feedwater system malfunctions • Excessive increase in steam flow • Inadvertent opening of a steam generator relief or safety valve • Steam line break • Loss of external load/turbine trip • Loss of offsite power • Loss of normal feedwater flow • Feedwater line rupture • Loss of forced reactor coolant flow • Locked reactor coolant pump rotor/sheared shaft • Control rod cluster withdrawal at power • Dropped control rod cluster/dropped control bank • Inadvertent increase in coolant inventory • Inadvertent opening of a pressurizer relief or safety valve • Steam generator tube rupture The transients analyzed for Ginna using RETRAN are: • Increase in feedwater flow (Ginna UFSAR Section 15.1.2) • Steam line break (Ginna UFSAR Section 15.1.5) • Combined steam generator atmospheric relief valve and feedwater control valve failures (Ginna UFSAR Section 15.1.6) • Loss of external electrical load (Ginna UFSAR Section 15.2.2) • Loss of offsite alternating current power to the station auxiliaries (Ginna UFSAR Section 15.2.5) • Loss of normal feedwater flow (Ginna UFSAR Section 15.2.6) • Feedwater system pipe breaks (Ginna UFSAR Section 15.2.7) • Flow coastdown accidents (Ginna UFSAR Section 15.3.1) • Locked rotor accident (Ginna UFSAR Section 15.3.2) • Uncontrolled RCCA withdrawal at power (Ginna UFSAR Section 15.4.2) • Inadvertent opening of a pressurizer safety or relief valve (Ginna UFSAR Section 15.6.1) • Steam generator tube rupture (Ginna UFSAR Section 15.6.3) As each transient analyzed for Ginna using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.
2. ***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."*** Justification The Ginna Station consists of a single 2-loop Westinghouse-designed unit that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.

Table A.3-1: RETRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
3. ***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis.."***   Justification   The input data used in the RETRAN analyses performed by Westinghouse came from both Constellation Generation Group and Westinghouse sources. Assurance that the RETRAN input data is conservative for Ginna is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative plant-specific input values were requested and collected from the responsible Constellation Generation Group and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A (Reference A.3-1), the safety analysis input values used in the Ginna analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Reference

A.3-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

A.4 LOFTRAN for Non-LOCA Safety Analysis

Table A.4-1: LOFTRAN for Non-LOCA Safety Analysis

Limitations, Restrictions, and Conditions
1. ***"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:*** • ***Feedwater System Malfunction*** • ***Excessive Increase in Steam Flow*** • ***Inadvertent Opening of a Steam Generator Relief or Safety Valve*** • ***Steamline Break*** • ***Loss of External Load*** • ***Loss of Offsite Power*** • ***Loss of Normal Feedwater*** • ***Feedwater Line Rupture*** • ***Loss of Forced Reactor Coolant Flow*** • ***Locked Pump Rotor*** • ***Rod Withdrawal at Power*** • ***Rod Drop*** • ***Startup of an Inactive Pump*** • ***Inadvertent ECCS Actuation*** • ***Inadvertent Opening of a Pressurizer Relief or Safety Valve*** ***This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."*** Justification For Ginna, the LOFTRAN code was used in the analyses of the dropped rod transient (Ginna UFSAR Section 15.4.6) and ATWS (Ginna UFSAR Section 15.8). As each of these transients matches one of the transients listed in the SER, additional justification is not required.

A.5 VIPRE for Non-LOCA Thermal/Hydraulics

Table A.5-1: VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions, and Conditions
1. ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."*** Justification The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Ginna 14x14 422V+ fuel. The use of the WRB-1 DNB correlation is based on the notification change which introduces the 14x14 422V+ mid-grid design (NPL 97-0538, CAW-97-1166). The basic change is reverting back to the larger OD fuel rod as in standard fuel but with a new Low Pressure Drop mid-grid design. The applicability of WRB-1 to the LPD mid-grid was justified under FCEP (WCAP-12488-A). The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Ginna 422V+ fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse two-loop plants using the same fuel design.
2. ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."*** Justification The core boundary conditions for the VIPRE calculations for the 422V+ fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272-P-A (Reference A.5-1).
3. ***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."*** Justification As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used in the DNB analyses of 422V+ fuel for Ginna. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45, for pressures between 500 psia and 1,000 psia).
4. ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***

Table A.5-1: VIPRE for Non-LOCA Thermal/Hydraulics

Limitations, Restrictions, and Conditions
Justification For application to Ginna safety analysis, the usage of VIPRE in the post-critical heat flux region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions: • DNB was assumed to occur at the beginning of the transient, • Film boiling was calculated using the Bishop-Sandberg-Tong correlation, • The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction. Conservative results were further ensured with the following input: • Fuel rod input based on the maximum fuel temperature at the given power, • The hot spot power factor was equal to or greater than the design linear heat rate, • Uncertainties were applied to the initial operating conditions in the limiting direction.

Reference

A.5-1 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. L. Davidson (Ed.), July 1985.

Question #4

Detailed results of the calculations to determine margins between current and EPU conditions and to show that the current design-basis calculations bound the plant at EPU conditions

Response

A comparison of results for the Non-LOCA EPU analyses to the current analyses of record has been provided in the LR as detailed in the following table:

Section	Title	
2.8.5.1.1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	Bounded by SLB HFP, LR Section 2.8.5.1.1.2.2.4, DNBR and PLHR Limit check Bounded by SLB HZP
2.8.5.1.2	Steam System Piping Failures Inside and Outside Containment SLB at Hot Zero Power (HZP) SLB at Hot Full Power (HFP)	 DNB design limit is met No previous analysis
2.8.5.2.1	Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum	LR Table 2.8.5.2.1-2
2.8.5.2.2	Loss of Non-Emergency AC Power to the Station Auxiliaries	LR Table 2.8.5.2.2-2
2.8.5.2.3	Loss of Normal Feedwater Flow	LR Table 2.8.5.2.3-2
2.8.5.2.4	Feedwater System Pipe Breaks Inside and Outside Containment	LR Table 2.8.5.2.4-2
2.8.5.3.1	Loss of Forced Reactor Coolant Flow	LR Table 2.8.5.3.1-2
2.8.5.3.2	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	LR Table 2.8.5.3.2-2
2.8.5.4.1	Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical or Low-Power Startup Condition	LR Table 2.8.5.4.1-2
2.8.5.4.2	Uncontrolled Rod Cluster Control Assembly Withdrawal at Power	LR Table 2.8.5.4.2-2
2.8.5.4.3	Control Rod Misoperation	LR Section 2.8.5.4.3.3
2.8.5.4.4	Startup of an Inactive Loop at an Incorrect Temperature	LR Section 2.8.5.4.4.3
2.8.5.4.5	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	LR Table 2.8.5.4.5-1
2.8.5.4.6	Spectrum of Rod Ejection Accidents	LR Table 2.8.5.4.6-3
2.8.5.6.1	Inadvertent Pressurizer Pressure Relief Valve Opening	LR Section 2.8.5.6.1.3
2.8.5.7	Anticipated Transients Without Scrams	Generic analysis remains bounding

RESPONSES TO NRC MEETING MINUTES FROM NOVEMBER 17, 2005 MEETING (NRC letter dated 12/23/2005)

Question #1 With respect to the Rod Withdrawal at Power analysis (Section 2.8.5.4.2):

a. Figure 2.8.5.4.2-11, which depicts Pressurizer Pressure vs. Time for the RCS Pressure Case, indicates an uninterrupted rise in pressurizer pressure, to almost 2700 psia, until the reactor is tripped. It appears that the pressurizer safety valves are not modeled, and this seems to be an overly conservative assumption. Please explain.

Response (a)

In the Rod Withdrawal at Power analyses performed to demonstrate that the peak primary pressure limits are met, the Pressurizer Safety Valves (PSVs) are modeled. The PSVs are modeled with an opening setpoint of 2589.5 psia. This setpoint is the nominal setpoint of 2485 psig plus 2.4% tolerance plus a 1% setpoint shift due to the water filled loop seals plus 5 psid for accumulation (1.034 * 2485 + 15 + 5 = 2589.5 psia). A PSV purge delay of 0.8 seconds is also included due to the water filled loop seals. Figure 2.8.5.4.2-11 of the licensing submittal shows the primary pressure transient for the Rod Withdrawal at Power event. In this figure, pressure increases to the PSV setpoint of 2589.5 psia and 0.8 seconds later the PSVs open terminating the pressure transient.

b. Results of this event are presented as an assembly of the minimum DNBR results of many transient analysis cases, mapping the minimum DNBRs as a function of initial power level, core reactivity feedback, and reactivity insertion rate. From this, it is concluded that the automatic Reactor Protection System prevents fuel clad damage for all analyzed cases. In similar fashion, analyze or evaluate the case(s) of operation at low power level with only one loop in operation.

Response (b)

During the audit, questions were asked concerning Rod Withdrawal at Power analyses initiated at less than 8% power with only 1 reactor coolant pump running. Single loop operation with the reactor critical is not procedurally allowed. The plant startup procedure, O-1.2, Plant Startup From Hot Shutdown to Full Load, initial conditions require both RCPs operating prior to reactor startup. The Abnormal Operating Procedure, AP-RCS.2, Loss of Reactor Coolant Flow, directs reactor trip if less than 2 RCPs running. Therefore, a rod withdrawal from low power with only 1 RCP running is prohibited. Therefore, these cases were neither considered in the EPU analyses nor were they considered in the current licensing basis analyses.

In addition, there is the availability of the high flux low setting reactor trip (nominal Tech Spec setpoint is 25% power). In the Rod Withdrawal at Power analyses performed for Ginna, cases for a spectrum of reactivity insertion rates are initiated at 10% power. In these cases, the high flux low setting is not credited and reactor trip does not occur until OTΔT is reached or until the high flux high setting (116%) is reached. In all cases analyzed, the peak heat flux at the time of trip is at least twice the high flux low setting. A 50% reduction in power is a larger benefit than the penalty associated with a 50% reduction in flow. Thus, the cases with one RCP in operation can not be more limiting than the cases already analyzed if credit were to be taken for the high flux low setting reactor trip.

Question #2 With respect to the Steam System Piping Failures analysis (Section 2.8.5.1.2):

Describe how the shutdown margin is determined and used in the single-loop in operation case.

Response

Per the current Ginna Bases [see Section "B 3.4.5" (RCS Loops - Modes 1 = 8.5% RTP, 2, and 3)], a main steam line break was originally analyzed for both the case with one and two RCS loops in operation at hot zero power (HZP) condition. However, with only one RCS loop in operation and offsite power available, additional shutdown margin is required since the reduced flow produces an adverse effect on DNB limits. Note that per current Technical Specifications, Ginna is limited to a power level of up to 8.5% with one inactive loop. To account for this increase in shutdown margin requirements at the N-1 loop conditions, an explicit analysis is made based on part-power rod insertion limits (RIL), adjusted flow (one-loop), and xenon skewing (conservative). The control rods are permitted to be more deeply inserted at 8.5% power than they are at 100% power, per the rod insertion limits. This means that less total rod worth (negative reactivity) is available to trip into the core at this condition. However, the amount of negative reactivity required to reduce the core thermal power level from 8.5% to 0% (i.e., the power defect) is significantly less than the amount required to reduce the power level from 100% to 0%. The net effect of this reactivity trade-off is that there is always more shutdown margin available at the 8.5% power N-1 loop condition than there is at the 100% power N loop condition. The results from the explicit N-1 loop shutdown margin analysis are compared to the required N-1 loop shutdown margin limit, as specified in the Core Operating Limits Report (COLR) per LCO 3.1.1.

Question #3 With respect to the Spurious SI and CVCS Malfunction event (Section 2.8.5.5):

Verify that the operator, following prescribed procedures, can terminate the CVCS Malfunction event before the pressurizer fills (about 13 minutes).

Response

See Ginna response to CVCS Question #1, NRC RAI letter 11/3/2005 provided in Constellation letter dated December 19, 2005.

Question #4 With respect to the Boron Dilution event (Section 2.8.5.4.5):

a. Identify Ginna's licensing basis for the boron dilution event for all modes. Indicate (1) whether the time allotted for operator action begins with operator notification or not, and (2) which operating modes must be analyzed.

Response (a)

See Ginna response to CVCS Questions #2 and #3, NRC RAI letter 11/3/2005 provided in Constellation letter dated December 19, 2005. The time allotted for operator action begins with the initiation of the event. This is consistent with the current licensing basis for Ginna.

b. Explain why the parameters used in the boron dilution calculations are conservative

Response (b)

Explicit analyses are performed in Modes 1, 2 and 6. The Mode 1 analysis assumes a 127 gpm dilution rate, an active RCS volume of 5123 ft^3, an initial boron concentration of 2100 ppm and a critical boron concentration of 1800 ppm. The Mode 2 analysis assumes a 120 gpm dilution rate, an active RCS volume of 5123 ft^3, an initial boron concentration of 2000 ppm and a critical boron concentration of 1800 ppm. The Mode 6 analysis assumes a 120 gpm dilution rate, an active RCS volume of 2042 ft^3, and an initial boron concentration to critical boron concentration ratio of 1.2914. The assumed boron concentrations are confirmed to be conservative during the reload evaluation performed prior to each cycle start-up. The active RCS volumes assumed are conservative in that no credit is taken for thermal expansion, maximum (10%) steam generator tube plugging is assumed, the Modes 1 and 2 volumes do not include the pressurizer or surge line, the Mode 6 volume includes only the reactor vessel (without the upper head) and the RHR volume.

The results of the analysis demonstrate that there is over 30 minutes in Mode 1 between the initiation of the event and a loss of shutdown margin, over 25 minutes in Mode 2 and 32 minutes in Mode 6. The acceptance criteria applied (15 minutes in Modes 1 and 2 and 30 minutes in Mode 6) is consistent with other plants licensed prior to the issuance of Reg. Guide 1.70, Revision 2 (~1980).

c. Since some sections of the boron dilution submittal are written in past tense, verify that the acceptance criteria are currently satisfied.

Response (c)

The acceptance criteria are currently satisfied.

Question #5 With respect to the Functional Design of the Control Rod System (Section 2.8.4.1):

a. Licensee states, "Changes to the rod position indication systems ... are currently being assessed to ensure that correct individual rod position indications are available to the operator." Please provide the results of that assessment.

Response (a)

Assessment of data from the Microprocessor Rod Position Indication (MRPI) system from prior plant startups indicated that the first transition coil would still be actuated in locations where control rods were located in 422V+ fuel. This assessment took into account the increased adapter plate height (approximately 3 inches) of the 422V+ fuel design including irradiation-induced growth of the assembly skeleton. Based on this assessment it was decided to not install spacers below the indicating coil stacks that surround the twenty-nine (29) control rod pressure housings. The MRPI firmware will be modified to adjust the position indication provided to the operator due to the change in top nozzle adapter plate height. System alarm setpoints and operating procedures will also be adjusted to account for the height difference.

b. Please provide the rod drop time analysis.

Response (b)

LR Section 2.2.3.2.3, "Description of Analyses and Evaluations", sub-section "RCCA Scram Performance Evaluation", presents the results of the rod drop time analysis and confirms that at the rod drop time at EPU conditions continues to meet the Tech Spec limit. In addition, Westinghouse calculation note CN-RCDA-04-125, Revision 1 could be made available to the NRC for review at the Westinghouse Twinbrook office.

Question #6 With respect to the Fuel System Design (Section 2.8.1) and Nuclear Design (Section 2.8.2):

a. The FCEP for the 9 grid 422V+ fuel does not provide the overall assembly flow loss coefficient, nor an explanation of how it is determined. Please provide that information, either in the FCEP or in the Ginna submittal.

Response (a)

The FCEP was provided to the NRC for notification of a revision to the existing generic (7-grid) Westinghouse 14x14 422 V+ Design. The Ginna-specific (9-grid) application of the generic 14x14 422 V+ is discussed in LR Section 2.8.1.2.1. The assembly pressure drops (total, not by component) for the Ginna 9-grid 14x14 422 V+ fuel design appear in LR Table 2.8.3.1. The "losses", per se, are not listed within the LR.

Section 2.8.3 states that full-scale hydraulic tests were performed on the 14x14 Westinghouse 422V+ fuel assembly design to confirm the pressure loss compatibility with the 14x14 OFA fuel design. Based on comparison of the overall core pressure loss, the 14x14 Westinghouse 422V+ fuel assembly design pressure drop is slightly lower than the 14x14 OFA design due principally to differences in the grid designs (see Section 2.8.1 Fuel System Design).

b. The analysis for the transition cycles indicates that the current fuel (OFA) could possibly exceed 60GWD/MTU during the first transition cycle. Since this does not represent an actual core loading, it is not necessary, at this time, to submit any information to support exceeding 60GWD/MTU. The licensee agreed to provide this information in the future, in the event that an actual core loading is predicted to exceed 60GWD/MTU.

Response (b)

The Ginna Cycle 32 reload is predicted to exceed 60,000 MWD/MTU and has already gone through the process. The Ginna Cycle 32 Reload Safety Evaluation (QLP-05-24) contains the following notification: The NRC-approved Fuel Criteria Evaluation Process (FCEP) (References 35 and 37) was used to justify additional lead rod average burnup exposure of up to 2,000 MWD/MTU beyond the currently licensed limit of 60,000 MWD/MTU. This Reload Safety Evaluation concludes that all the fuel design criteria continue to be satisfied for Cycle 32. The rod burnup extension is licensable under 10 CFR 50.59 and requires no prior NRC approval.

c. The issue and commitment in item (b), above, also apply to the 422V+ fuel assemblies in the first equilibrium cycle.

Response (c)

Those processes used for the current fuel (OFA) will be used when applicable for the 422V+ fuel assemblies.

RESPONSES TO NRC VERBAL QUESTIONS FROM DECEMBER 21, 2005 PHONE CALL

With respect to operational experience used in developing the Ginna EPU test plan, this response is provided as supplementary information to RAI question #1 in NRC letter dated October 25, 2005 (initial response provided in Constellation letter dated December 6, 2005):

Question #1

Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," specifies in Part III.C, the guidance and acceptance criteria the licensee should use to provide justification for a test program that does not include all of the power-ascension testing that would normally be considered for inclusion in the EPU test program. Previous operating experience should be considered, as applicable, when justifying elimination of power-ascension tests.

In Section 2.12, "Power Ascension and Testing Plan," of the licensing report, the licensee stated that "operating experience has been incorporated into the proposed test plan."

However, the licensee has not provided information of specific operating experience incorporated into their proposed test plan. Provide additional information regarding specific examples of operating experiences incorporated into the proposed test plan.

Response

Ginna has done extensive reviews of industry operating experience associated with plant uprates at other facilities and will be incorporating lessons learned from this experience in a number of ways to facilitate uprate implementation: e.g. modification scope and design, operator training, procedure development and testing. One aspect of the operating experience review that will permeate the test program is incorporation of applicable operating experience into pre-job briefs for all test procedures that will be performed. In addition, below is a list of specific examples where operating experience will be used to enhance test procedures. Since the detailed test procedures have not been written, and since our operating experience review continues, we expect that there will be additional examples of how operating experience is used to enhance the test program when the procedures are fully developed.

The vibration monitoring plan will be enhanced based on industry failures of small bore piping and other components due to vibration induced fatigue. We have reviewed the industry events caused by vibration post power uprate, including such events as main steam relief valve pilot valve vent line failure and lift setpoint drift, turbine control valve hydraulic fluid accumulator tubing failure, main steam low point drain pipe failure, turbine hydraulic fluid pump skid piping and pressure switch failures, feedwater regulating valve solenoid valve failure and feedwater heater level control valve and positioner failures. As a result of this industry experience, the vibration monitoring program will be enhanced to include monitoring of all of these components. In addition, all branch lines attached to lines that will see an increase in process fluid flow have been included in the scope of lines to be closely monitored.

The power escalation test plan will include a turbine valve stroke test in part as a result of industry experience including load instability during valve testing and load swings when transferring control from automatic to manual.

The power escalation test plan will also include turbine vibration monitoring and detailed operator guidance to address turbine rubs as a result of industry experience with turbine rubs associated with mono-block rotors.

The post-modification test plan and power escalation test plan will be enhanced to include performance of Iso-Phase Bus Duct air flow testing and temperature monitoring as a direct result of industry experience with failures resulting from increased bus duct air flow and higher bus duct temperatures.

Additional walkdown monitoring points will be incorporated into the power escalation test procedure to verify that condenser hotwell and feedwater heater level control systems are operating properly throughout the power escalation process based on plant operating experience with these control systems prior to uprate.

As described in the Licensing Report Section 1.0, Introduction to the Ginna Station Extended Power Uprate Licensing Report, use of industry operating experience, Ginna carefully considered the lessons learned accrued from industry experience with uprates.

NRC provided information such as event reports (e.g.: Event numbers 38916) and announced special inspections (e.g.: NRC news No. III-06-002), as well as information notices have provided valuable insights which are continuously applied to project engineering activities.

In addition to the data supplied by the NRC, other OE, not available to the general public, provided inputs. These included direct communication with other utilities who've under gone uprate as well as data supplied by the Institute of Nuclear Power Operations (INPO).

Specific additional details regarding the operating experience utilized in developing the Ginna EPU test plan can be found in the following INPO documents:

Event #s:

265-020402-1	main steam low point drain pipe failure
278-951023-1	turbine hydraulic fluid pump skid pipe failure
237-020626-1	turbine hydraulic fluid pressure switch failure
333-981028-1	feedwater heater control valve and positioner failure

OE #s:

17530	main steam relief valve pilot valve vent line failure
20915	main steam relief valve lift setpoint drift
14149	turbine control valve accumulator tubing failure
9684	turbine load instability during valve testing
12280	turbine load swings when transferring control from automatic to manual
20891	turbine rubs associated with monoblock rotors
18874	iso-phase bus duct failure due to increased air flow (see also SER 4-04)

RESPONSES TO NRC VERBAL QUESTIONS FROM DECEMBER 22, 2005 PHONE CALL

With respect to the feed isolation valve modification, these responses are provided as supplementary information to RAI questions #3, 4 and 5 in NRC letter dated October 19, 2005 (initial response provided in Constellation letter dated November 21, 2005):

Question #3

Discuss the calculation of the thrust necessary to operate the main feedwater isolation valves (MFIVs) under the pressure and flow conditions for their new safety function.

Response

The thrust necessary to operate the MFIVs has been calculated by the valve manufacturer based upon the worst case differential pressure and flow conditions provided in the valve specification (450 psid and 8,000,000 lbs/hr). The manufacturer determines required thrusts to overcome friction based upon industry experience. The frictional forces are small relative to the forces required to overcome the pressures due to flow acting on the valve plug and stem. An analysis of the forces required to overcome the pressure forces acting on the valve plug and stem has been performed consistent with the methods recommended in the EPRI Air Operated User's Guide, TR-107322 and TR-107321. The calculated thrust required to close and seat the disc is specified to be 90,000 lb_f.

Question #4

Discuss the qualification of the actuators to be installed on the MFIVs to perform the new safety function.

Response

The valve actuators will be certified in accordance with the criteria described in 10CFR50 Appendix B by the actuator manufacturer to be capable to perform their specified function. The manufacturer, Hiller, has many years of experience providing Safety Related Air Actuators for use in the US nuclear industry. The actuators themselves and the associated pilot valves and solenoid valves, etc. will be procured safety related and are selected based on historical reliable performance. The actuators will be installed using safety related work control documents by qualified individuals, including vendor representatives from both the valve and actuator manufacturers, as needed. The Intermediate Building where these valves are located is not a harsh environment during the steam line break inside containment for which these valves must operate and these valves are not required to operate during events which may create a harsh environment in the Intermediate Building. Appropriate preventive maintenance will be applied to the valves to ensure reliable operation. The valves will be periodically tested in accordance with Technical Specifications and corresponding ASME Inservice Testing Program.

The Hiller methodology used to design and procure materials and subcomponents for safety related equipment is defined in Hiller Procedures and in the Hiller QA Manual which have been provided to Ginna for review. These procedures include design control, commercial grade dedication, a corrective action program, inventory control, safety classification, Control of Assembly Processes, and testing control, etc. as necessary to meet the criteria of 10CFR50, Appendix B.

The tests required to meet the requirements specified in the design specification will be controlled in accordance with the Testing Controls defined in the Hiller QA Manual. These tests include a pneumatic pressure test of the unit and a breakaway pressure test which quantifies actuator internal frictional forces. The thrust capability of the actuator will be determined by analysis of the thrust available based on the actuator minimum air pressure and area, less the actuator internal frictional forces. The results of these analyses and tests will be used to definitively ascertain the output thrust capacity of the actuator and assure that it meets the design specification.

The valves and actuators will be functionally tested by Ginna after final assembly prior to being put into service.

Question #5

Discuss the monitoring and surveillance of the performance of the MFIVs as part of the Inservice Testing Program at Ginna

Response

The valves will be stroke time tested and position indication verified during Cold Shutdown in accordance with the IST Program. The valves will be included in the Air Operated Valve Program and base-lined and monitored for degradation to ensure reliable performance.

The Ginna AOV Program is based on the Joint Owners Group on Air Operated Valves (JOG AOV) document which was distributed to utilities in 1999. The JOG AOV program provided guidance for verification of valve functionality at design conditions and long-term periodic verification. The Ginna AOV Program contains all nine AOV Program Elements of the JOG Document including Baseline Testing, Periodic Testing, and Post Maintenance Testing.

The JOG AOV program included lessons learned from nuclear plant Motor Operated Valve Programs citing similarities in valve designs between AOVs and MOVs indicate the potential for GL 89-10 issues, such as initial valve setup assumptions being non-conservative. GL 89-10 and Regulatory Issue Summary 2000-003 attributes of a Successful Power-Operated Valve Design Capability and Long-Term Periodic Verification Program were reviewed and adopted in the Ginna AOV Program.

Furthermore, a review of the MPR-2524 "Joint Owners' Group (JOG) Motor Operated Valve Periodic Verification Program Summary" did not identify any additional activities that would warrant inclusion above those already contained in the JOG Document. The JOG MOV PV Program concluded that for disc globe valves there is no age-related degradation in required thrust. The JOG PV Summary also concluded that there is no service-related degradation in required thrust for disc globe valves in both water and steam service.

With respect to overpressure protection during power operation, this response is provided as supplementary information to RAI question #1 in NRC letter dated November 3, 2005 (initial response provided in Constellation letter dated December 19, 2005):

Question #1

In the Licensing Report, the licensee included descriptions of provisions to address overpressure protection for Ginna operating at the uprated power. This information addresses only the change in the pressurizer safety valve upper lift setting; not the adequacy of the safety valve capacity. Although UFSAR Table 5.2-1 refers to the ASME Code, Section III, Nuclear Vessels, 1965, it does not detail the analyses that were performed assuming the uprated power to demonstrate the adequacy of the safety valve capacity and to quantify the sufficiency of the design margin of the safety valve(s).

Note that WCAP-7769, Revision 1, provides a demonstration of compliance for Ginna, based upon ASME Code, Section III, Articles NB-7300 and NC-7300, "Protection Against Overpressure," 1971. However, this demonstration was for Ginna operating at 1518.5 MWt.

Provide the results of analyses based upon methods consistent with those of WCAP-7769 (including credit for the second (or later) safety grade trip from the reactor protection system) to show continued sufficiency of margin of design capacity for the Ginna pressurizer and steam line safety valves, with Ginna operating at the uprated power of 1775 MWt.

Response

The Ginna EPU overpressure analyses are consistent with the requirements of SRP 5.2.2. SRP 5.2.2 requires that the second safety grade reactor trip signal be credited for safety valve sizing calculations. This is consistent with the safety valve sizing procedure discussed in Section 2 of WCAP7769. WCAP7769 states, "For the sizing, main feedwater flow is maintained and no credit for reactor trip is taken." This analysis is typically performed prior to construction of the plant to provide a basis for the capacity requirements for the safety valves and the requirement of SRP 5.2.2 provides a conservative basis for the number and design of the valves.

However, WCAP7769 goes on to say, "After determining the required safety valve relief capacities, as described above, the loss of load transient is again analyzed for the case where main feedwater flow is lost when steam flow to the turbine is lost... For this case, the bases for analysis are the same as described above except that credit is taken for Doppler feedback and appropriate reactor trip, other than direct reactor trip on turbine trip." This describes the analysis performed in Chapter 15 of the UFSAR which verifies that the overpressure limits are satisfied with the current/latest design.

The analyses performed in support of the Ginna EPU Program are not safety valve sizing calculations - no changes are being made to the safety valves as a result of this uprating. The Loss of External Electrical Load / Turbine Trip analysis performed for the EPU Program, presented in Section 2.8.5.2.1, demonstrates that the safety valves have adequate capacity to maintain peak primary pressure below 110% of design which satisfies the requirements of GDC-15. GDC15 applies to "any condition of normal operation, including anticipated operational occurrences" which does not include a common mode failure of the first safety grade reactor trip signal.

The Loss of External Load / Turbine Trip RCS overpressure analysis is performed to demonstrate that, in the event of a sudden loss of the secondary heat sink, the associated increase in reactor coolant system temperature does not result in over-pressurization of the RCS system.

However, in response to the NRC RAI, a one-time calculation was performed using the Ginna EPU RETRAN model developed for the Loss of External Electrical Load / Turbine Trip analysis. The calculation was performed consistent with the licensing basis Loss of External Electrical Load / Turbine Trip analysis assumptions but with no credit taken for the first trip function, which in this case is the High Pressurizer Pressure trip function. Conservatism was removed from the modeling of the pressurizer safety valves. The calculated peak primary pressure continues to meet the analysis acceptance criteria that the primary pressure be maintained below 110% of the design pressure.

With respect to fire protection, this response is provided as supplementary information to RAI question #2 in NRC letter dated October 25, 2005 (initial response provided in Constellation letter dated December 6, 2005):

Question #2

Attachment 1 to Matrix 5 of NRR RS-001 states that "...where licensees rely on less than full capability systems for fire events..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping."

The NRC staff notes that Section 2.5.1.4 of the licensing report does not address these two items. Provide a discussion that addresses these two items.

Response

The licensing report was written with the presumption that, since the CLB already addressed these issues, only the effects of EPU on the previous evaluations need to be described. To clarify our submittal with respect to these two items, please review the section heading "Alternative Shutdown Capability" beginning on page 2.5.1.4-9. Thus, while we explain what we need to do to preserve these two items we did not explicitly state that is why we're doing them. I regret any confusion this may have caused.

Fuel integrity is maintained during all fire scenarios for the Ginna EPU as the fuel remains covered and no fuel design limits are exceeded. Similarly, there are no adverse consequences on the reactor pressure vessel integrity or the attached piping as RCS pressure remains less than 110% of design pressure for all fire scenarios.

RESPONSES TO NRC VERBAL QUESTIONS FROM JANUARY 11, 2006 PHONE CALL

With respect to additional best estimate LOCA analysis results, Ginna provides the following supplementary information:

Question #1

Provide the Peak Clad Temperature (PCT) and Local Maximum Oxidation (LMO) results for the already burned, resident OFA fuel at EPU conditions for Ginna. Specific quantified results should be provided, as well as a brief summary of the methodology used to determine these values.

Response

The Westinghouse 422V+ fuel design was analyzed and shown to meet the 10 CFR 50.46 acceptance criteria for the R. E. Ginna LBLOCA ASTRUM analysis to support an Extended Power Uprate (EPU). For R. E. Ginna, a selected transient calculation was performed with a mixed core of 422V+ and the resident OFA fuel, and the OFA fuel was shown to be bounded by the 422V+ fuel design. The mixed core assessment was performed at the uprated power level of 1811 MWt and with the same peaking factors analyzed in the ASTRUM analysis. The results are presented in Table 1, and the PCT and oxidation assessments are discussed in more detail below.

The mixed core calculation modeled the hot rod/assembly as fresh OFA fuel, surrounded by Westinghouse 422V+ fuel. The resident OFA fuel was analyzed in the WCOBRA/TRAC calculation as though it would operate at the same peaking factors as the Westinghouse 422V+ fuel. That calculation showed that the fresh OFA fuel was non-limiting with respect to PCT by 56°F.

The difference in Peak Cladding Temperature (PCT) between the full core of 422V+ transient and the mixed core transient are applied to the results from the Ginna ASTRUM analysis to conservatively estimate the PCT for the resident OFA fuel. The conservative aspect to this approach is that the reference and mixed core transients have PCTs which occur earlier, and are somewhat lower than the limiting ASTRUM case. The zirc-water heat source increases with temperature, therefore, the PCT sensitivity tends to become more pronounced as the base case PCT increases. The delta PCT for the resident OFA fuel is based on a lower PCT case which minimizes the delta PCT between the OFA fuel and the 422V+ fuel. Since the resident OFA fuel was determined to be less limiting than the 422V+ fuel, the OFA fuel estimated PCT is maximized since the difference between the 422V+ and OFA fuel is minimized.

There is a clear relationship between the Local Maximum Oxidation (LMO) and the hot rod PCT for the R. E. Ginna ASTRUM analysis. The results of the ASTRUM analysis were used to determine an estimated change in LMO based on the change in hot rod PCT (Figure 1). Applying this relationship yields a 0.9% reduction in LMO based on the reduced PCT for the resident OFA fuel. Applying the delta PCT and oxidation values from the mixed core evaluation yields a PCT result of 1,814°F and an LMO result of 2.5% for the resident OFA fuel.

Table 1 contains the results of the ASTRUM analysis for the full core of 422V+ fuel, as well as the evaluation results for the resident OFA fuel (estimated values). The maximum expected total of the normal operation (pre-transient) and LOCA transient oxidation, for any time in life, was considered as part of this evaluation for both the 422V+ and the resident OFA fuel. The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for both of the fuel designs that will be included in the EPU cores is <16%. The actual upper bound pre-transient values are expected to be below this value. The transient oxidation decreases from the near BOL value of 3.4% for the 422V+ fuel (2.5% for the resident OFA fuel) to a negligible value at EOL. The sum of the pre-transient plus transient oxidation remains below 17% at all times in life, for all fuel types in the core at uprated conditions.

Figure 2 shows the PCT transients for a full core of 422V+ fuel compared to a transition core with mixed fuel. Figure 3 compares the hot assembly collapsed liquid level between the two cases. It can be seen that the fuel design has little effect on the re-flooding behavior of the hot assembly.

Table 1 – LBLOCA Fuel Cladding Results

10 CFR 50.46 Requirement	Value	Criteria
Maximum PCT (°F)	1,870 (422V+ fuel) 1,814 (resident OFA fuel)	< 2,200
Maximum Cladding Oxidation during LOCA Transient (%)[1]	3.4 (422V+ fuel) 2.5 (resident OFA fuel)	< 17
Maximum Core Wide Oxidation (%)	0.30 (all fuel)	< 1

1. The maximum total oxidation of any fuel in the core, including pre-transient oxidation, is less than 17% throughout the life of the fuel.

Attachment 5

Figure 1 – Local Maximum Oxidation (LMO) versus PCT

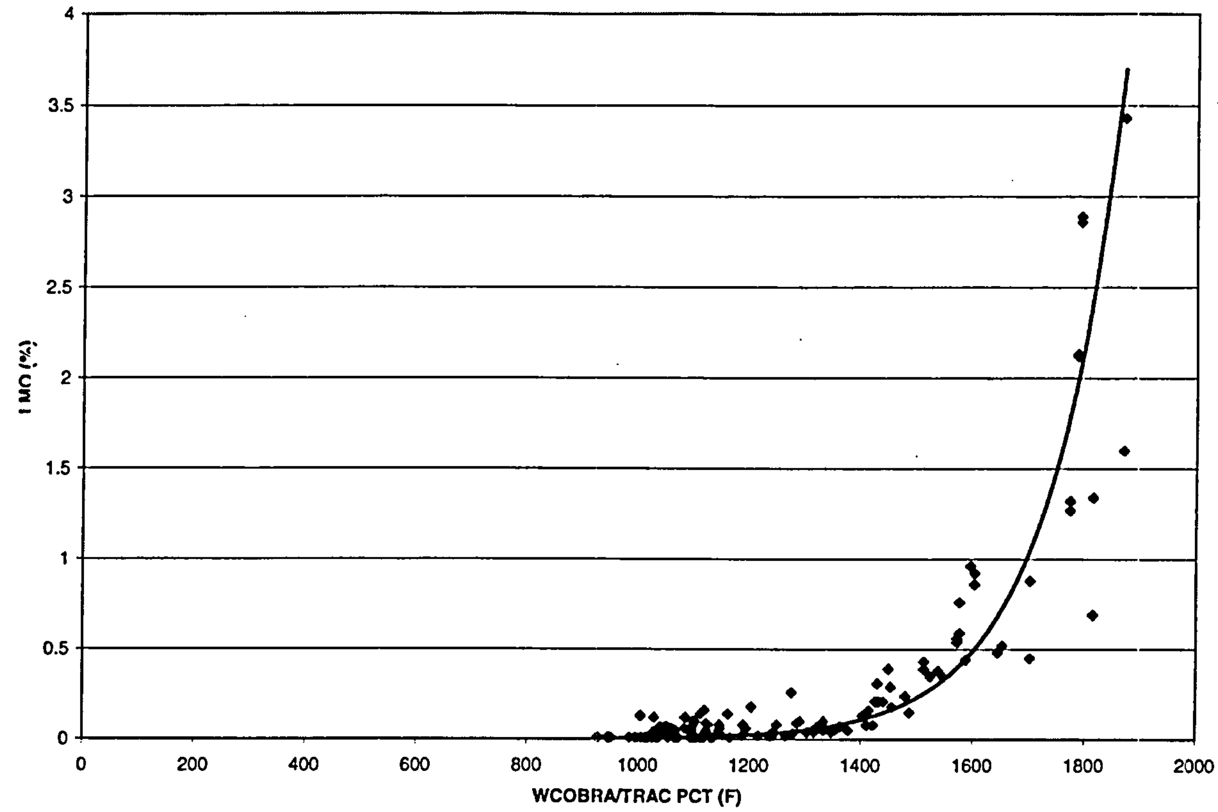

Figure 2 – Hot Rod Peak Cladding Temperature

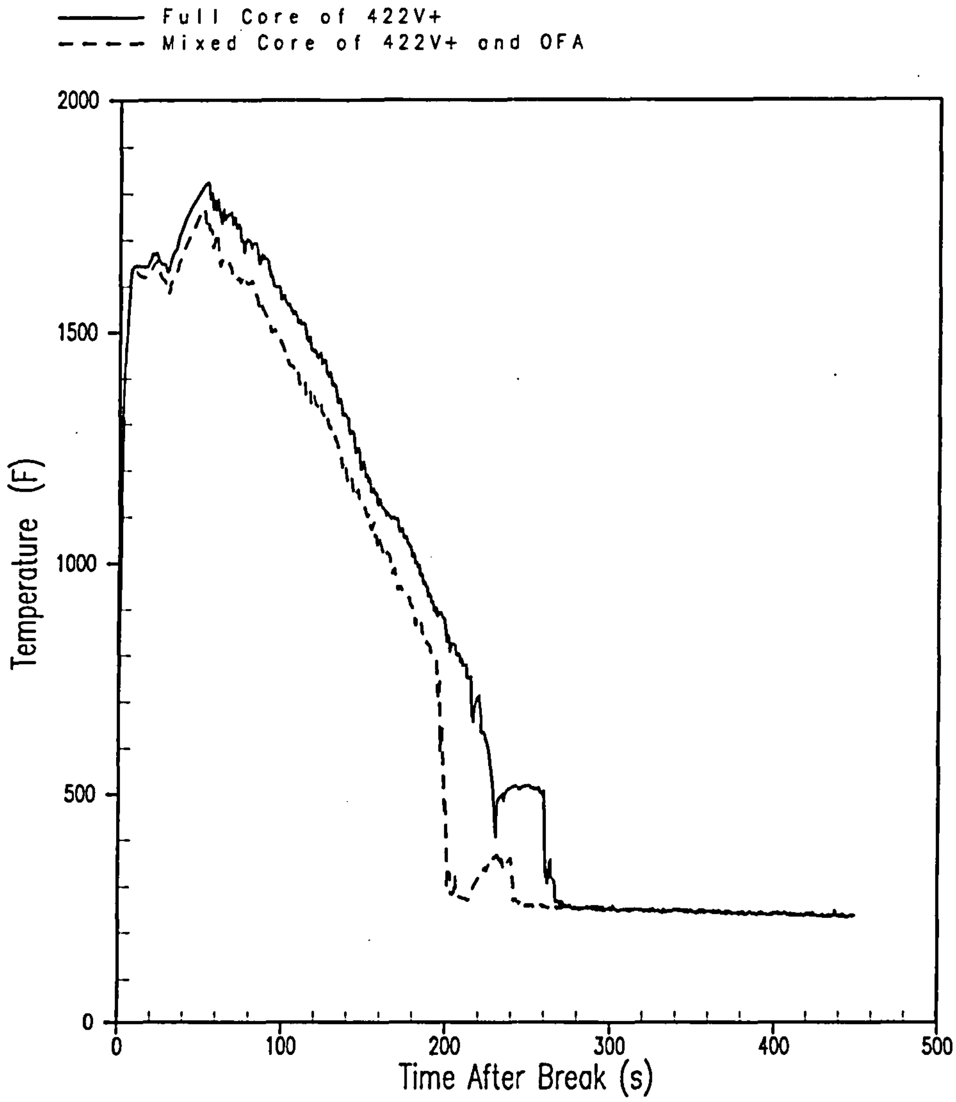

Figure 3 – Hot Assembly Collapsed Liquid Level