Unit 1 Design Power Capability Parameters Used in Non-LOCA Safety Analyses

	(MUR Power Uprate)[1,2]		(Reduced Temperature and Pressure)[1,2]		(Rerating)[2]		(Return to RCS NOP/NOT)[1]	
Parameter	**Case 1**	**Case 2**	**Case 3**	**Case 4**	**Case 5**	**Case 6**	**Case 7**	**Case 8**
NSSS Power, MWt	3,327	3,327	3,262	3,262	3,425	3,425	3,327	3,327
Core Power, MWt	3,315	3,315	3,250	3,250	3,413	3,413	3,315	3,315
RCS Flow, gpm/loop	83,200	83,200	83,200	83,200	88,500	88,500	83,200	83,200
Minimum Measured Flow, gpm/loop	84775	84775	84775	84775	91600	91600	84,775	84,775
RCS Temperatures, ºF								
Core Outlet	593.1	613.6	589.7	611.9	583.6	614.0	593.1	613.6
Vessel Outlet	588.2	609.1	586.8	609.1	580.7	611.2	588.2	609.1

[1] The non-LOCA analyses are based on a Thermal Design Flow (TDF) of 83,200 gpm / loop and a Minimum Measured Flow (MMF) of 84,775 gpm / loop. However, subsequent evaluations were performed to show that the following higher flows are also supported: 88,500 gpm / loop (TDF) and 90,725 gpm / loop (MMF).

[2] Cook Unit 1 is not licensed to operate at the rerated conditions specified by Cases 5 and 6 with 30% steam generator tube plugging (SGTP) levels. However, several events that were previously performed using these conditions were subsequently evaluated to support the 30% SGTP program. Hence, the rerated conditions are also specified in this table for completeness.

Unit 1 Design Power Capability Parameters Used in Non-LOCA Safety Analyses

	(MUR Power Uprate)[1,2]		(Reduced Temperature and Pressure)[1,2]		(Rerating)[2]		(Return to RCS NOP/NOT)[1]	
Parameter	**Case 1**	**Case 2**	**Case 3**	**Case 4**	**Case 5**	**Case 6**	**Case 7**	**Case 8**
Core Average	557.6	579.5	555.8	579.4	549.7	581.8	557.6	579.5
Vessel Average	553.7	575.4	553.0	576.3	547.0	578.7	553.7	575.4
Vessel/Core Inlet	519.2	541.7	519.2	543.5	513.3	546.2	519.2	541.7
Steam Generator Outlet	518.9	541.5	518.9	543.2	513.1	546.0	518.9	541.5
Zero Load	547.0	547.0	547.0	547.0	547.0	547.0	547.0	547.0
RCS Pressure, psia	2,250 or 2,100	2,250 or 2,100	2,250 or 2,100	2,250 or 2,100	2,250 or 2,100	2,250 or 2,100	2,250	2,250
Steam Pressure, psia	618	765	595	749	603	820	618	765
Steam Flow (10^6 lb/hr total)	14.44	14.50	14.12	14.17	14.98	15.07	14.44	14.50
Feedwater Temp., °F	437.4	437.4	434.8	434.8	442	442	437.4	437.4
SG Tube Plugging, %	30	30	30	30	10	10	30	30

REACTOR TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN SAFETY ANALYSES [1]

REACTOR TRIP FUNCTION	LIMITING REACTOR TRIP POINT ASSUMED IN ANALYSIS	TIME DELAY (sec)
Power range high neutron flux, high setting	118 percent	0.5
Power range high neutron flux, low setting	35 percent	0.5
Overtemperature ΔT	Variable, see Figure 14.1-1, -2, -3 & -4	8.0 [2]
Overpower ΔT	Variable, see Figure 14.1-1, -2, -3 & -4	8.0 [3]
High pressurizer pressure	2420 psig	2.0
Low pressurizer pressure	1825 psig [4]	2.0
High pressurizer water level	100% NRS	2.0
Low reactor coolant flow (from loop flow detectors)	87 percent loop flow	1.0
Undervoltage trip	[5]	1.5
Low-low steam generator level	0.0 percent of narrow range level span	2.0
High-High steam generator level [6]: - Turbine Trip - Feedwater Isolation	82 percent of narrow range level span	 2.5 11.0

[1] The control rod scram time to dashpot is 2.4 seconds.

[2] Total time delay (including RTD time response, and trip circuit, channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall. The time delay assumed in the analysis supports the response time of the RTD time response, trip circuit delays, and channel electronics delay presented in the UFSAR Table 14.1-2 (Unit 1) or Table 7.2-6 (Unit 2). An evaluation has been performed (Reference 11) that demonstrates that the analyses remains bounding, given that the total 8.0 second time delay in the above table is satisfied.

[3] Overpower ΔT reactor trip was assumed in the steamline break mass/energy release outside containment calculations.

[4] A value of 1845 psig is used in LOCA analyses.

[5] No explicit value assumed in the analysis. Undervoltage trip setpoint assumed reached at initiation of analysis.

[6] Time delay between High-High steam generator level and Turbine Trip is 2.5 seconds. The Turbine Trip subsequently causes a Reactor Trip.

Summary of Initial Conditions and Computer Codes Used

	Reactivity Coefficients Assumed									
Fault Conditions	**Computer Codes Utilized**	**Moderator Temperature (pcm / °F)**	**Moderator Density (ΔK/gm/cc)**	**Doppler**	**DNB Correlation**	**Revised Thermal Design Procedure**	**Initial NSSS Thermal Power Output (MWt)**	**Reactor Vessel Coolant Flow (gpm)[1]**	**Vessel Average Temperature (°F)**	**Pressurizer Pressure (psia)**
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	Refer to Section 14.1.1		Min[2]	W-3/WRB-1 See Section 14.1.1	No	0	146,432	547	2033
Uncontrolled RCCA Bank Withdrawal at Power[3]	LOFTRAN	+5	.54	Min and Max[4]	WRB-1	Yes	3,327 1996 333	339,100	575.4[5] 564.58 549.93	2,100

[1] The non-LOCA analyses are based on a Thermal Design Flow (TDF) of 83,200 gpm / loop and a Minimum Measured Flow (MMF) of 84,775 gpm / loop. However, subsequent evaluations were performed to show that the following higher flows are also supported: 88,500 gpm / loop (TDF) and 90,725 gpm / loop (MMF).

[2] Minimum Doppler power defect (pcm / %power) = -9.55 + 0.00104Q where Q is in MWt.

[3] Multiple power levels, T_{avg}, and reactivity feedback cases were examined.

[4] Maximum Doppler power defect (pcm / %power) = -19.4 + 0.002Q where Q is in MWt.

[5] Value used in the DNB analysis is performed at the MUR power uprate program maximum Tavg of 575.4°F.

Summary of Initial Conditions and Computer Codes Used

		Reactivity Coefficients Assumed									
Fault Conditions	**Computer Codes Utilized**	**Moderator Temperature (pcm / °F)**	**Moderator Density (ΔK/gm/cc)**	**Doppler**	**DNB Correlation**	**Revised Thermal Design Procedure**	**Initial NSSS Thermal Power Output (MWt)**	**Reactor Vessel Coolant Flow (gpm)[1]**	**Vessel Average Temperature (°F)**	**Pressurizer Pressure (psia)**
RCCA Misalignment[6]	LOFTRAN THINC	N/A[7]	NA	NA	WRB-1	Yes	3,250	339,100	576.3[5]	2,100
Uncontrolled Boron Dilution	N/A	N/A	N/A	N/A	N/A	N/A	3,425 0	N/A	N/A	N/A
Loss of Forced Reactor Coolant Flow[6]	LOFTRAN FACTRAN THINC	+5	N/A	Max	WRB-1	Yes	3,270	339,100	576.3[5]	2,100
Locked Rotor (Peak Pressure)	LOFTRAN	+5	N/A	Max	N/A	N/A	3,335	332,800	581.4	2,317
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	N/A	Max	N/A	N/A	3,335	332,800	581.4	2,033
Locked Rotor (Rods-in-DNB)[6]	LOFTRAN FACTRANT HINC	+5	N/A	Max	WRB-1	Yes	3,270	339,100	576.3	2,100

[6] An uprated core power of 3315 MWt (NSSS power of 3327 MWt) is supported via an evaluation that addresses a reduction in power uncertainty from 2% to approximately 0.3%.

[7] N/A - Not Applicable

Summary of Initial Conditions and Computer Codes Used

	Reactivity Coefficients Assumed									
Fault Conditions	**Computer Codes Utilized**	**Moderator Temperature (pcm / °F)**	**Moderator Density (ΔK/gm/cc)**	**Doppler**	**DNB Correlation**	**Revised Thermal Design Procedure**	**Initial NSSS Thermal Power Output (MWt)**	**Reactor Vessel Coolant Flow (gpm)[1]**	**Vessel Average Temperature (°F)**	**Pressurizer Pressure (psia)**
Loss of Electrical Load and/or Turbine Trip[8]	LOFTRAN	+5	.54	Max and Min	WRB-1	Yes No	3,327 3327	339,100 332,800	576.3[5] 581.4	2,100 2033
Loss of Normal Feedwater[9]	LOFTRAN	+5	N/A	Max	N/A	N/A	3,409	332,800	548.9	2,317
Excessive Heat Removal[9] Due to Feedwater System Malfunction	LOFTRAN	N/A	.54	Min	WRB-1	Yes	3,327 0	339,100	576.3[5] 547	2,100
Excess Load Increase[9] Incident	LOFTRAN	N/A	0 and .54	Max and Min	WRB-1	No	3,425	366,400	578.7	2,100

[8] Minimum and Maximum reactivity feedback cases were examined

[9] Values presented were used in the rerating analysis. Subsequent evaluations support the 30% SGTP parameters presented as Cases 1 and 2 of Table 14.1-1

Summary of Initial Conditions and Computer Codes Used

		Reactivity Coefficients Assumed									
Fault Conditions	**Computer Codes Utilized**	**Moderator Temperature (pcm / °F)**	**Moderator Density (ΔK/gm/cc)**	**Doppler**	**DNB Correlation**	**Revised Thermal Design Procedure**	**Initial NSSS Thermal Power Output (MWt)**	**Reactor Vessel Coolant Flow (gpm)[1]**	**Vessel Average Temperature (°F)**	**Pressurizer Pressure (psia)**
Loss of Offsite Power (LOOP)[9] to the Station Auxiliaries	LOFTRAN	+5	N/A	Max	N/A	N/A	3,409	332,800	548.9	2033
Rupture of a Steam Pipe (at hot zero power)	LOFTRAN THINC	See Figure 14.2.5-1	N/A	See Figure 14.2.5-2	W-3	N/A	0	332,800	547	2,100
Rupture of a Steam Pipe (at full power)	LOFTRAN VIPRE ANC	N/A	0.54	Least Negative[10]	WRB-1	Yes	3,327	339,100	575.4	2,250
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 14.2.6	N/A	Min	N/A	N/A	3,335 0	332,800 146,432	581.4 547	2,033

[10] Least negative power defect (pcm / %power) = 9.55 + 0.0355Q, where Q is in %power.

INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS POWER RANGE NEUTRON FLUX

	SETPOINT AND ERROR ALLOWANCES: (% of rated power)	ESTIMATED INSTRUMENTATION ERRORS: (% of rated power)
Nominal Setpoint	109	-
Calorimetric Error	2[1]	0.31
Axial power distribution effects on total ion chamber current	5	3
Instrumentation channel drift and setpoint reproducibility	2	1.0
Maximum overpower reactor trip point assuming all individual errors are simultaneously in the most adverse direction	119	-

[1] MUR power uprate uses reduced calorimetric error allowance. The sum of the change in Rated Thermal Power defined in the Technical Specifications and the MUR reduced calorimetric error allowance is equal to, or less than, the original +2% value supported by the safety analyses.

DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM INPUT ASSUMPTIONS FOR RCS VOLUMES[1]

INPUT ASSUMPTIONS	INITIAL CONDITIONS	
	0% SGTP	30% SGTP
Reactor Vessel (ft^3)	4643	4643
Steam Generators (ft^3 - Total)	4357[2]	3442 [(2)]
Reactor Coolant Pumps (ft^3 - Total)	324	324
Loop Piping (ft^3 - Total)	1185	1185
Surge Line Piping (ft^3)	43	43
Pressurizer (ft^3)	1834	1834
Total RCS Volume (ft^3) (Ambient Conditions)	12,386	11,471
Total RCS Volume (ft^3) (Hot Conditions includes 3% for thermal expansion)	12,758	11,815

[1] The volumes presented in this table represent the reactor coolant system volumes from the Westinghouse IMP database SEC-LIS-4428-C3. Note that the volumes documented in this table are slightly different than those used in the Westinghouse analysis of record, but have been evaluated for acceptability as documented in letter AEP-99-485.

[2] The SG tube volume is assumed to be 762 ft^3/SG (3048 ft^3 total). The increase in SG tube plugging from 0% to 30% results in a total reduction in SG tube volume of approximately 915 ft^3. The reduction between the SG tube volume and SG tube plugging is assumed to be a linear relationship; e.g. at 15% SGTP, total volume reduction is 0.15*(3048 ft^3) = 457.2 ft^3.

DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM INPUT ASSUMPTIONS FOR STEAM GENERATOR SECONDARY MASS

INPUT ASSUMPTIONS	INITIAL CONDITIONS[1]		
	0% SGTP	30% SGTP Cases	
	Original Design Case	Low Temp Case A1	High Temp Case A2
Steam generator secondary side mass (Total lbs/SG)	106,506	106,799[2]	112,192[3]

[1] Initial conditions are presented for SGTP levels of 0% (Original Design) and 30% to bound the range of SGTP levels.

[2] For T_{avg} of 553°F.

[3] For T_{avg} of 576°F.

DONALD C. COOK NUCLEAR PLANT UNIT 1 SGTP PROGRAM INPUT ASSUMPTIONS FOR REACTOR COOLANT SYSTEM PRESSURE DROP[(1)]		
Input Assumptions	**Initial Conditions**	
	0% SGTP Pressure Drop, psi	**30% SGTP Pressure Drop, psi**
Reactor Vessel, including nozzles (psi)	47.21	44.26
Loop Piping (psi)	5.14	4.55
Steam Generator (psi)	43.23	53.58
Total (psi)	95.58[(1)]	102.39[(1)]

ECCS INJECTION TO RECIRCULATION SWITCHOVER MODEL FOR THE CONTAINMENT RESPONSE ANALYSIS TIME AND RWST DELIVERED VOLUME AFTER RWST EARLIEST SWITCHOVER SETPOINT IS REACHED

EVENT	TIME AFTER SWITCHOVER SETPOINT IS REACHED (sec)	RWST DELIVERED VOLUME (gal)
Earliest switchover setpoint is reached	0	280,000
Stop RHR / CTS pumps	0	280,000
Start RHR / CTS pumps	300	285,975
CCP and SI suction realignment to RHR discharge (minimum RWST delivered)	1707	314,000

TIME SEQUENCE OF EVENTS

ACCIDENT	EVENT	TIME (sec)
Uncontrolled RCCA Bank Withdrawal At Full Power		
Case A (high insertion rate, max feedback)	Initiation of uncontrolled RCCA bank withdrawal at a high reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high trip signal initiated	4.9
	Rods begin to fall into core	5.4
	Minimum DNBR occurs	5.8
Case B (small insertion rate, max feedback)	Initiation of uncontrolled RCCA bank withdrawal at a small reactivity insertion rate (4 pcm/sec)	0
	Overtemperature ΔT reactor trip signal initiated	321.2
	Minimum DNBR occurs	322.1
	Rods begin to fall into core	323.2

SEQUENCE OF EVENTS FOR LOSS OF FLOW AND LOCKED ROTOR ACCIDENTS

ACCIDENT	EVENT	TIME (sec)
Complete Loss of Flow	All pumps lose power and begin coasting down, undervoltage trip signal generated	0.0
	Rods begin to drop	1.50
	Minimum DNBR occurs	3.40
Partial Loss of Flow	One operating pump loses power and begins coasting down	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	1.74
	Rods begin to drop	2.74
	Minimum DNBR occurs	3.90
Locked Rotor	One pump rotor seizes	0.0
	Low reactor coolant flow trip setpoint reached in faulted loop	0.04
	Rods begin to drop	1.04
	Time at which minimum DNBR is predicted to occur	2.6
	Maximum RCS pressure occurs	3.20
	Maximum clad temperature occurs	3.49

Parameters Used for the Radiological Consequence Analysis of a Locked Rotor Event

Parameter	Value
Core Power Level	3480 MWt
Fuel Clad Failure	11%
Core Fractions Released from Damaged Rods	
I-131	0.08
Other Halogens	0.05
Kr-85	0.10
Other Noble Gases	0.05
Alkali Metals	0.12
Fuel Rod Peaking Factor	1.65
Core Release Fraction Multiplier for High Burnup Fuel	1.0104
Secondary Coolant Limit for Normal Operation	0.1 μCi/gm D.E. I-131
Primary Coolant Mass	466,141.5 lbm
Secondary System Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Primary-to-Secondary Leak Rate	1 gpm to all steam generators
Steam Generator Steam Release	
0 - 2 hours	460,000 lbm
2 - 8 hours	1,256,000 lbm
8 - 4 hours	1,347,000 lbm
Partition Coefficients	
Iodines	100
Alkali Metals	500
Nobles Gases	1
Duration of Intact SB Tube Uncovery After Reactor Trip	40 minutes
Intact Tube Leakage Flashing Fraction During Uncovery	0-400 seconds: 8% 400-900 seconds: 6% 900-1700 seconds: 5.5% 1700 seconds-40 min: 4%
Iodine Chemical Form	
Elemental	97%
Organic	3%
Particulate	0%

Parameters Used for the Radiological Consequence Analysis of a Locked Rotor Event

Parameter	Value
Release Location	
Offsite	Unit 1 Main Steam Enclosures
Onsight	Unit 2 PORVs/MSSVs
Offsite Breathing Rates	
0-8 hours	3.5E-04 m^3/sec
8-24 hours	1.8E-04 m^3/sec
24-720 hours	2.3E-04 m^3/sec
Control Room Parameters	
Volume	50,616 ft^3
Normal Ventilation Makeup Flow Rate	880 cfm
Emergency Ventilation Makeup Flow Rate	880 cfm
Emergency Ventilation Recirculation Flow Rate	4520 cfm
Emergency Ventilation Filter Efficiency[1]	
Elemental Iodine	94.05%
Organic Iodide	94.05%
Particulates	98.01%
Delay to Switch to Emergency Mode	20 minutes (manual)
Unfiltered Inleakage	40 cfm
Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.4
Breathing Rate	3.5E-04 m^3/sec

[1] Includes 1% filter bypass leakage

SEQUENCE OF EVENTS FOR LOSS EXTERNAL ELECTRICAL LOAD

CASE	EVENT	TIME (sec)
Minimum Feedback with Pressure Control	Loss of external electrical load	0.0
	OTΔT trip setpoint reached	13.1
	Peak RCS pressure occurs	13.1
	Rods begin to drop	15.1
	Minimum DNBR occurs	16.6
Maximum Feedback with Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	Peak RCS pressure occurs	9.3
	Low-low steam generator level trip setpoint reached	35.2
	Rods begin to drop	37.2
Minimum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	8.6
	Rods begin to drop	10.6
	Peak RCS pressure occurs	11.6
Maximum Feedback without Pressure Control	Loss of external electrical load	0.0
	Minimum DNBR occurs	0.0
	High pressurizer pressure trip setpoint reached	9.2
	Rods begin to drop	11.2
	Peak RCS pressure occurs	12.1

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER

ACCIDENT	EVENT	TIME (sec)
Loss of Normal Feedwater	Main feedwater flow stops	10.0
	Low-low steam generator water level trip signal initiated	35.7
	Rods begin to fall into core	37.7
	Auxiliary Feedwater Pumps Start and Supply the Steam Generators	95.7
	Cold Auxiliary Feedwater is Delivered to the Steam Generators (MFW purged) Steam Generators #1 and #4 Steam Generators #2 and #3	 311.5 1027
	Peak water level in pressurizer occurs	6426
	Core decay heat plus RCP heat decreases to auxiliary feedwater heat removal capacity	~6426

SEQUENCE OF EVENTS FOR EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

CASE	EVENT	TIME (sec)
Feedwater System Malfunctions: Excessive feedwater flow at full power to a single steam generator (Automatic Rod Control)	One main feedwater control valve fails fully open	0.0
	Hi-hi steam generator water level signal generated	40.4
	Turbine trip occurs due to hi-hi steam generator water level	42.9
	Minimum DNBR occurs	43.0
	Reactor trip occurs due to turbine trip	44.9
	Feedwater isolation achieved	51.4
Feedwater System Malfunctions: Excessive feedwater flow at full power to a single steam generator (Manual Rod Control)	One main feedwater control valve fails fully open	0.0
	Hi-hi steam generator water level signal generated	40.5
	Turbine trip occurs due to hi-hi steam generator water level	43.5
	Minimum DNBR occurs	43.5
	Reactor trip occurs due to turbine trip	45.0
	Feedwater isolation achieved	51.5

SEQUENCE OF EVENTS FOR EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

CASE	EVENT	TIME (sec)
Feedwater System Malfunctions: Excessive feedwater flow at full power to all four steam generators (Automatic Rod Control)	One main feedwater control valve fails fully open	0.0
	Hi-hi steam generator water level signal generated	42.7
	Minimum DNBR occurs	45.0
	Turbine trip occurs due to hi-hi steam generator water level	45.2
	Reactor trip occurs due to turbine trip	47.2
	Feedwater isolation achieved	53.7
Feedwater System Malfunctions: Excessive feedwater flow at full power to all four steam generators (Manual Rod Control)	One main feedwater control valve fails fully open	0.0
	Hi-hi steam generator water level signal generated	42.4
	Turbine trip occurs due to hi-hi steam generator water level	44.9
	Minimum DNBR occurs	42.5
	Reactor trip occurs due to turbine trip	46.9
	Feedwater isolation achieved	53.4

TIME SEQUENCE OF EVENTS FOR LOSS OF ALL AC POWER TO STATION AUXILIARIES

ACCIDENT	EVENT	TIME (sec)
Loss of All AC Power to Station Auxiliaries	AC power is lost	10.0
	Main feedwater flow stops	10.0
	Low-low steam generator water level trip signal initiated	35.7
	Rods begin to fall into core	37.7
	Reactor coolant pumps begin to coastdown	39.7
	Auxiliary Feedwater Pumps Start and Supply the Steam Generators	115.7
	Cold Auxiliary Feedwater is Delivered to the Steam Generators (MFW purged) Steam Generators #1 and #4 Steam Generators #2 and #3	 331.5 1047
	Peak water level in pressurizer occurs	3980
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~3980

Potential Turbine-Generator Missiles

Unit 1 (GE) [1]

Part (Name)	Weight (Pounds)	Failure Speed (Rpm) (and % of 1800)		Cross Sectional Area Average			Velocity & Kinetic Energy after Leaving Turbine Casing	
				Min.	(Sq. Ft.)	Max.	(Ft/Sec)	(Ft-lbs)
Vane of Last Stage Bucket	54	3,130	(174%)	0.033		1.6	1,170	1×10^6
120ºF Segment of Last Stage Wheel	8,264	3,190	(177%)	5.16	8.43	11.70	409	21.5×10^6

Unit 2 (BB) [2]

Part (Name)	Weight (Pounds)	Failure Speed (Rpm) (and % of 1800)		Cross Sectional Area Average			Velocity & Kinetic Energy after Leaving Turbine Casing	
				Min.	(Sq. Ft.)	Max.	(Ft/Sec)	(Ft-lbs)
Vane of Last Stage Bucket	168	3,040	(169%)	0.106		3.64	1,135	3.4×10^6
120º Segment of Next-to-Last Disc (Disc 3)	8,360	3,170	(176%)	5.8	13.2	20.6	551	39.4×10^6
120º Segment of Disc 1	13,350	3,170	(176%)	9.55	15.08	20.60	634	83.3×10^6
120º Segment of Disc 2	12,100	3,170	(176%)	7.24	13.92	20.60	574	61.8×10^6
120º Segment of Disc 4	16,600	3,170	(176%)	11.70	15.70	19.70	595	91.3×10^6

[1] The postulated turbine missile information in this table is for the removed General Electric low pressure turbines as this analysis bounds other rotating elements. The current missile analysis for the Unit 1 low pressure turbines is based on the missile probability analysis discussed in Section 1.4. 7.

[2] The postulated turbine missile information in this table is for the removed Brown Boveri low pressure turbine as these analyses are also included in structural design criteria shown in Table 5.1-1. The current missile analysis for the Unit 2 low pressure turbines is based on missile probability analysis discussed in Section 1.4.7.