RANGE OF PLANT NOMINAL CONDITIONS USED IN SAFETY ANALYSES[1]

Parameter	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
NSSS Power, Mwt	3600	3600	3600	3600	3600	3600
Core Power, Mwt	3588[2]	3588[2]	3588[2]	3588[2]	3588[2]	3588[2]
RCS Flow,(gpm/loop)	88,500	88,500	88,500	88,500	88,500	88,500
Minimum Measured Flow, (total gpm)	366,400	366,400	366,400	366,400	366,400	366,400
RCS Temperatures, °F						
Core Outlet	613.5	585.8	618.4	618.2	585.8	585.7
Vessel Outlet	610.2	582.3	615.2	615.0	582.3	582.2
Core Average	579.5	550.1	584.8	584.9	550.1	550.1
Vessel Average	576.0	547.0	581.3	581.3	547.0	547.0
Vessel/Core Inlet	541.8	511.7	547.3	547.6	511.7	511.8
Steam Generator Outlet	541.6	511.4	547.1	547.4	511.4	511.5
Zero Load	547.0	547.0	547.0	547.0	547.0	547.0
RCS Pressure, psia	2250	2250	2250	2100	2250	2100

[1] A brief description of each case follows Table 14.1.0-1.

[2] The Best Estimate Large Break (LB) LOCA analyses with RHR cross-ties open support plant operation with a core power at 3468 MWt (plus 0.34% uncertainty). The SBLOCA analysis with the High Head SI cross-tie valves open supports plant operation up to a core power of 3600 MWt (plus 0.34% uncertain).. Evaluations have been performed to support the Measurement Uncertainty Recapture (MUR) power uprate, where the sum of the power uprate and the revised, reduced calorimetric power uncertainty remains equal to, or less than, the 2% uncertainty assumed in the safety analyses.

Parameter	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
Steam Pressure, psia	780.4	587.0	820.0	820.0	587.0	587.0
Steam Flow, (106 lb/hr total)	15.98	15.90	16.0	16.0	15.9	15.9
Feedwater Temp.,°F	449.0	449.0	449.0	449.0	449.0	449.0
% SG Tube Plugging	10	10	10	10	10	10

A BRIEF DESCRIPTION OF VARIOUS CASES LISTED

Case 1 and 2: These parameters cases were used to support operation during mixed core cycles (Cycles 8 and 9).

Case 3: These parameters incorporate a core power level of 3588 MWt, an NSSS power level of 3600 MWt (which includes 12 MWt for reactor coolant pump heat), an average steam generator tube plugging level of 10%, RCS pressure of 2250 psia, and an upper bound vessel average temperature of 581.3°F. This parameter case was used to support high RCS temperature and high RCS pressure operation for a full VANTAGE 5 core (Cycle 10 and beyond).

Case 4: These parameters incorporate the same features as case 3, except the RCS pressure is 2100 psia. This parameter case was used to support high RCS temperature and low RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

Case 5: These parameters incorporate the same features as case 3, except the lower bound vessel average temperature is 547°F. This parameter case was used to support low RCS temperature and high RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

Case 6: These parameters incorporate the same features as case 5, except the RCS pressure is 2100 psia. This parameter case was used to support low RCS temperature and low RCS pressure operation for a full VANTAGE 5 core (Cycles 10 and beyond).

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	REACTIVITY COEFFICIENTS ASSUMED Moderator Temperature (pcm/°F)	REACTIVITY COEFFICIENTS ASSUMED Moderator Density (ΔK/gm/cc)	REACTIVITY COEFFICIENTS ASSUMED Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output [1] (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	TWINKLE FACTRAN THINC	See Section 14.1.1.2	N/A[2]	[3]	W-3 ANF WRB-2 and W-3 V-5	No	0	162,840	547.0	2037.0 [4]
RCCA Misalignment	LOFTRAN THINC	N/A	N/A	N/A	W-3 ANF WRB-2 V-5	Yes	3600	366,400	581.3	2100.0 [5]

[1] Includes reactor coolant pump heat, if applicable.
[2] N/A – Not Applicable
[3] Zero Power Doppler Power Defect at BOL assumed to be – 1000 pcm.
[4] Core Pressure
[5] For transition cycles, pressurizer pressure is 2250 psia.

Unit 2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	REACTIVITY COEFFICIENTS ASSUMED: Moderator Temperature (pcm/°F)	Moderator Density (ΔK/gm/cc)	Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output [1] (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Boron Dilution	N/A N/A	N/A N/A	N/A N/A	N/A N/A	N/A N/A	N/A N/A	3600 0	N/A N/A	N/A N/A	N/A N/A
Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	N/A	Max [6]	W-3 ANF WRB-2 V-5	Yes	3608	366,400	581.3 [7]	2100.0 (5)
Locked Rotor (Peak Pressure)	LOFTRAN	+5	N/A	Max (6)	N/A	N/A	3680	354,000	585.4	2312.6
Locked Rotor (Peak Clad Temp)	LOFTRAN FACTRAN	+5	N/A	Max (6)	N/A	N/A	3680	354,000	585.4	2037.4

[6] Maximum Doppler power coefficient (pcm/%power) = -19.4 + 0.002Q, where Q is in MWt (see Figure 14.1.0-1)
[7] For Transition Cycles, Vessel Average Temperature is 576°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 [1] (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Locked Rotor (Rods-in-DNB)	LOFTRAN FACTRAN THINC	+5	N/A	Max [(6)]	WRB-2	Yes	3608	366,400	581.3	2100.0
Loss of Normal Feedwater	LOFTRAN	0	N/A	Max [(6)]	N/A	N/A	3680	354,000	585.4	2312.6
Loss of Offsite Power (LOOP) to the Station Auxiliaries	LOFTRAN	0	N/A	Max [(6)]	N/A	N/A	3680	354,000	541.4	2312.6
Rupture of a Steam Pipe	LOFTRAN THINC	See Figure 14.2.5-1	N/A	See Figure 14.2.5-2	W-3 ANF W-3 V-5	NO	0	354,000	547.0	2100.0

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Fault Conditions	Computer Codes Utilized	REACTIVITY COEFFICIENTS ASSUMED: Moderator Temperature (pcm/°F)	REACTIVITY COEFFICIENTS ASSUMED: Moderator Density (ΔK/gm/cc)	REACTIVITY COEFFICIENTS ASSUMED: Doppler	DNB Correlation	Revised Thermal Design Procedure	Initial NSSS Thermal Power Output [1] (MWt)	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Rupture of a Control Rod Drive Mechanism Housing	TWINKLE FACTRAN	See Section 14.2.6	N/A	[8], [9]	N/A	N/A	3660 [10] 0	354, 000 162, 840	585.4 547.0	2037.4 [(4)]
Rupture of Feedwater Pipe	LOFTRAN	N/A	.54	Max [(6)]	N/A	N/A	3680	354, 000	585.4	2162.6

[8] Full Power Doppler Power defect at BOL and EOL assumed to be -966 pcm and -893 pcm respectively.
[9] Zero Power Doppler only Power defect at BOL and EOL assumed to be -965 pcm and -849 pcm, respective.
[10] Core thermal power.

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SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED: SEPARATE FULL VANTAGE 5 CORE ANALYSES

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)[1]	Reactor Vessel Coolant Flow (GPM)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)
Uncontrolled Rod Cluster Assembly Bank Withdrawal At Power [2]	LOFTRAN	N/A[3] +5	.54 N/A	Max[4] Min[5]	WRB-2	Yes	3608 2165 361	366,400	581.3 567.6 550.4	2100.0
Loss of Electrical Load or Turbine Trip [6]	LOFTRAN	N/A +5	.54 N/A	Max(4) Min(5)	WRB-2	Yes	3600	366,400	581.3	2100.0
Excessive Heat Removal Due to Feedwater System Malfunction	LOFTRAN	N/A N/A	.54 .54	Min(5) Min(5)	WRB-2 WRB-2	Yes Yes	3600 0	366,400 366,400	581.3 547.0	2100.0 2100.0
Excess Load Increase	LOFTRAN	N/A N/A	0 .54	Min(5) Max(4)	WRB-2	Yes	3600	366,400	581.3	2100.0

[1] Includes reactor coolant pump heat, if applicable.
[2] Multiple power levels, T_{avg}, and reactivity feedback cases were examined.
[3] N/A – Not Applicable
[4] Maximum Doppler Power coefficient (pcm/%power) = -19.4 + 0.002Q, where Q is in MWt (see Figure 14.1.0-1).
[5] Minimum Doppler power coefficient (pcm/%power) = -9.55 + 0.00104Q, where Q is in MWt (see Figure 14.1.0-1).
[6] Minimum and maximum reactivity feedback cases were examined.

UNIT 2

RPS TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN NON-LOCA SAFETY ANALYSES

Trip Function	Nominal Setpoint	Point Assumed In Analysis	Limiting Trip Time Delay (seconds)
Power range high neutron flux, high setting	109%	118%	0.5
Power range high neutron flux, low setting	25%	35%	0.5
Overtemperature ΔT	See Table 2.2-1	Variable, see Figures 14.1.0-5,6	8.0 [1]
Overpower ΔT	in Tech Spec	Variable, see Figures 14.1.0-5,6	8.0 [(1)]
High pressurizer pressure	2385 psig	2428 psig	2.0
Low pressurizer pressure	1950 psig	1907 psig	2.0
High pressurizer water level	92% of span	100% span	2.0
Low reactor coolant flow (From loop flow detectors)	90% loop flow	87% loop flow	1.0
Undervoltage trip volts each bus	2905 volts each bus	NA [2]	1.5
Underfrequency trip	57.5 Hz	57 Hz	0.6
Low-low steam generator level	21% of narrow range span	0.0% of narrow range span	2.0

[1] Total time delay (Including RTD time response, trip circuit, and 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 the channel electronics delay presented in the UFSAR Table 14.1.0-4. An evaluation has been performed (Reference 9) that demonstrates that the analyses remains bounding given that the total 8.0 second time delay in the above table is satisfied.

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

ESF ACTUATION SETPOINTS AND TIME DELAYS TO ACTUATION ASSUMED IN NON-LOCA SAFETY ANALYSES

ESF Actuation Function	Nominal Setpoint	Limiting Actuation Setpoint Assumed In Analyses	Time Delay (Seconds)
Safety Injection (SI)			
- Low pressurizer pressure	1815 psig	1700 psig	27 w/offsite power [1]
			37 w/o offsite power [2]
- Low steamline pressure	600 psig	344 psig	27 w/offsite power [(1)]
			37 w/o offsite power [(2)]
Auxiliary Feedwater (AFW)			
- Low-low steam generator water level	21% of narrow range span	0.0% of narrow range span	60[3]
High-high steam generator Level Turbine Trip	67% of narrow range span	82% of narrow range span	2.5
Steamline Isolation on low steam line pressure	NA[4]	NA [(4)]	11[5]
Feedwater Line Isolation on high-high steam generator water level	67% of narrow range span	82% of narrow range span	11 [6]
Feedwater Line Isolation on low steam line pressure	NA [(4)]	NA [(4)]	8 [(6)]

[1] Emergency diesel generator starting and sequence loading delays NOT included. Offsite power available. Response time limit includes opening of valves to establish safety injection (SI) path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the volume control tank (VCT) to the refueling water storage tank (RWST) (RWST valves open, then VCT valves close) is included.
[2] Emergency diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valve close) is included.
[3] For Loss of Normal Feedwater and Loss of Offsite Power to Station Auxiliaries occurrences, the delay time assumed is 60 seconds from the initiation of the signals. For Feedwater Line Break event, the delay time assumed is 600 seconds (10 minute operator action delay) from the initiation of the break.
[4] Not Applicable
[5] Steamline isolation total delay time includes valve closure time, and electronics and sensor delay. Technical Specifications require 8.0 second valve closure time.
[6] Feedwater Line isolation total delay time includes valve closure time and electronics and sensor delay time.

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	Fault Conditions	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
14.1.1	Uncontrolled RCCA bank withdrawal from a subcritical condition	Power range high flux (low setpoint)	NA	NA	NA
14.1.2	Uncontrolled RCCA bank withdrawal at power	Power range high flux, overtemperature delta-T, high pressurizer pressure, high pressurizer level	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.3	RCCA misalignment				
14.1.4	(including rod drop)				
14.1.5	Uncontrolled Boron Dilution	Source range high flux power range high flux overtemperature delta-T	NA	Low insertion limit annunciators for boration	NA
14.1.6.1	Partial and complete loss of forced reactor coolant flow	Low flow, undervoltage underfrequency	NA	Steam generator safety valves	NA
14.1.6.2	Reactor coolant pump shaft seizure (locked rotor)	Low flow	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.7	Startup of an inactive reactor coolant loop [1]	-	-	-	-

[1] This cannot occur in Modes 1 and 2 as restricted by the Cook Nuclear Plant Unit 2 Technical Specifications.

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	Fault Conditions	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
14.1.8	Loss of external electric load or turbine trip	High pressurizer pressure overtemperature delta-T, lo-lo steam generator level	Steam generator lo-lo level	Pressurizer safety valves, steam generator safety valves	Auxiliary Feedwater System
14.1.9	Loss of normal feedwater	Steam generator lo-lo level, manual	Steam generator lo-lo level	Steam generator safety valves, pressurizer safety valves	Auxiliary Feedwater System
14.1.10	Feedwater system malfunctions that result in an increase in feed water flow	Power range high flux, (low and high setpoints), steam generator lo-lo level (Intact steam generators)	High-high steam generator level-produced feedwater isolation and turbine trip	Feedwater isolation	NA
14.1.11	Excessive load increase	Power range high flux, overtemperature delta-T, overpower delta-T	NA	Pressurizer safety valves, steam generator safety valves	NA
14.1.12	Loss of offsite power to the station Auxiliaries	Steam generator lo-lo level	Steam generator lo-lo level	Steam generator valves, pressurizer safety valves	Auxiliary Feedwater System
14.2.4	Steam generator tube failure	Reactor Trip System	Engineered Safety Features Actuation System	Steam generator safety and/or relief valves, steamline stop valves	Emergency Core Cooling System, Auxiliary Feedwater System, Emergency Power System

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	Fault Conditions	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
14.2.5	Rupture of a Steam Line	SIS, low pressurizer pressure, manual	Low pressurizer pressure low compensated steamline pressure, high containment pressure, manual	Feedwater isolation steamline stop valves	Auxiliary Feedwater System, Safety Injection System
	Inadvertent opening of a steam generator relief or safety valve	SIS	Low pressurizer pressure, low compensated steamline pressure	Feedwater isolation steamline stop valves	Auxiliary Feedwater System, Safety Injection System
14.2.6	Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate	NA	NA	NA
14.2.8	Feedwater system pipe break	Steam generator lo-lo level, high pressurizer pressure, SIS	High containment pressure, steam generator lo-lo water level, low compensated steamline pressure	Steamline isolation valves, feedline isolation, pressurizer self-actuated safety valves, steam generator safety valves	Auxiliary Feedwater System, Safety Injection System

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR FAULT CONDITIONS

	Fault Conditions	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
14.3	Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System Component Cooling Water System steam generator safety and/or relief valves	Emergency Core Cooling System, Auxiliary Feedwater System, Containment Heat Removal System, Emergency Power System

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Uncontrolled RCCA Withdrawal From A Subcritical Condition		
	Initiation of uncontrolled RCCA withdrawal (63 pcm/sec)	0.0
	High Neutron Flux Reactor Trip Setpoint (low setting) reached	12.2
	Rods begin to fall into core	12.7
	Minimum DNBR occurs	14.8
	Peak Clad Average Temperature occurs	15.3
	Peak Fuel Average Temperature occurs	15.6
	Peak Fuel Centerline Temperature Occurs	16.0

TIME SEQUENCE OF EVENTS (FULL VANTAGE 5 CORE)		
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	5.8
	Rods begin to fall into core	6.3
	Minimum DNBR occurs	6.4
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	314.5
	Minimum DNBR occurs	316.2
	Rods begin to fall into core	316.5

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Uncontrolled Boron Dilution		
1. Dilution during Refueling	Dilution begins	0
	Shutdown margin lost	1848
2. Dilution during startup	Dilution begins	0
	Shutdown margin lost	2100
3. Dilution during full power operation		
a. Automatic reactor control	Dilution begins	0
	Shutdown margin lost	2760
b. Manual reactor control	Dilution begins	0
	Overtemperature ΔT reactor trip	90
	Shutdown margin lost	2760

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Loss of Forced Reactor Coolant Flow		
Four loops in operation, four pumps coasting down		
	All operating pumps lose power and begin coasting down	0.0
	Reactor coolant pump under-voltage trip point reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	3.7
Four loops in operation, one pump coasting down		
	Coastdown begins	0.0
	Low flow reactor trip	1.28
	Rods begin to drop	2.28
	Minimum DNBR occurs	3.40

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Single Reactor Coolant Pump Locked Rotor		
Four loops in operation, one locked rotor		
	Rotor in one pump locks	0.00
	Low reactor coolant flow trip setpoint reached	0.02
	Rods begin to drop	1.02
	Time at which minimum DNBR is predicted to occur	2.2
	Maximum RCS pressure occurs	3.10
	Maximum clad temperature occurs	3.60

TIME SEQUENCE OF EVENTS

(FULL VANTAGE 5 CORE)

Accident	Event	Time (sec)
Loss of External Electric Load or Turbine Trip		
1. With pressurizer control (min fdbk)	Loss of electrical load	0.0
	Overtemperature ΔT reactor trip point reached	12.2
	Peak pressurizer pressure occurs	14.2
	Rods begin to drop	12.5
	Minimum DNBR occurs	16.0
2. With pressurizer control (max fdbk)	Loss of electrical load	0.0
	Peak pressurizer pressure occurs	8.5
	Low-low steam generator water level reactor trip point reached	53.7
	Rods begin to drop	55.7
	Minimum DNBR occurs	[1]
3. Without pressurizer control (min fdbk)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	7.3
	Peak pressurizer pressure occurs	9.3
	Rods begin to drop	9.0
	Minimum DNBR occurs	(1)
4. Without pressurizer control (max fdbk)	Loss of electrical load	0.0
	High pressurizer pressure reactor trip point reached	7.4
	Rods begin to drop	9.4
	Peak pressurizer pressure occurs	9.5
	Minimum DNBR occurs	(1)

[1] DNBR never decreases below its initial value.

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Loss of Normal Feedwater		
	Main feedwater flow stops	10.0
	Low-low steam generator water level trip signal initiated	55.7
	Rods begin to fall into core	57.7
	Two Motor-Driven Auxiliary Feedwater Pumps Start and Supply the Steam Generators	115.7
	Cold Auxiliary Feedwater is Delivered to the Steam Generators	515.0
	Peak water level in pressurizer occurs	4672
	Core decay heat plus RCP heat decreases to auxiliary feedwater heat removal capacity	4800

TIME SEQUENCE OF EVENTS
(FULL V-5 CORE)

Accident	Event	Time (sec)
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	30.2
	Turbine trip occurs due to hi-hi steam generator water level	32.7
	Minimum DNBR occurs	34.0
	Reactor trip occurs due to turbine trip	34.7
	Feedwater isolation achieved	41.2

TIME SEQUENCE OF EVENTS

(FULL V-5 CORE)

Accident	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	30.1
	Turbine trip occurs due to hi-hi steam generator water level	32.6
	Minimum DNBR occurs	33.0
	Reactor trip occurs due to turbine trip	34.6
	Feedwater isolation achieved	41.1

TIME SEQUENCE OF EVENTS

(FULL V-5 CORE)

Accident	Event	Time (sec)
Feedwater System Malfunctions: Excessive feedwater flow at full power to all four steam generators (Manual Rod Control)		
	All four main feedwater control valves fail fully open	0.0
	Hi-hi steam generator water level signal generated	31.5
	Turbine trip occurs due to hi-hi steam generator water level	34.0
	Minimum DNBR occurs	34.5
	Reactor trip occurs due to turbine trip	36.0
	Feedwater isolation achieved	42.5

TIME SEQUENCE OF EVENTS

(FULL V-5 CORE)

Accident	Event	Time (sec)
Feedwater System Malfunctions: Excessive feedwater flow at full power to all four steam generators (Automatic Rod Control)	All four main feedwater control valves fail fully open	0.0
	Hi-hi steam generator water level signal generated	31.8
	Turbine trip occurs due to hi-hi steam generator water level	34.3
	Minimum DNBR occurs	35.5
	Reactor trip occurs due to turbine trip	36.3
	Feedwater isolation achieved	42.8

TIME SEQUENCE OF EVENTS

(FULL V-5 CORE)

Accident	Event	Time (sec)
Excessive Load Increase		
1. Manual reactor control (Min fdbk)	10% step load increase	0.0
	Equilibrium conditions reached	160.0
2. Manual reactor control (Max fdbk)	10% step load increase	0.0
	Equilibrium conditions reached	40.0
3. Automatic reactor control (Min fdbk)	10% step load increase	0.0
	Equilibrium conditions reached	160.0
4. Automatic reactor control (Max fdbk)	10% step load increase	0.0
	Equilibrium conditions reached	70.0

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
Loss of Offsite Power to the Station Auxiliaries		
	AC power is lost	10.0
	Main feedwater flow stops	10.0
	Low-low steam generator water level trip signal initiated	56.0
	Rods begin to fall into core	58.0
	Reactor coolant pumps begin to coastdown	58.0
	Two Motor-Driven Auxiliary Feedwater Pumps Start and Supply the Steam Generators	117.0
	Cold Auxiliary Feedwater is Delivered to the Steam Generators	534.0
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~800.0
	Peak water level in pressurizer occurs	1406.0