NRC-01-043

NMC
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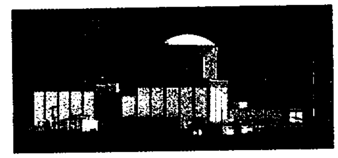
Kewaunee Nuclear Power Plant
N490, State Highway 42
Kewaunee, WI 54216-9511
920-388-2560

Operated by
Nuclear Management Company, LLC



April 20, 2001

10 CFR §50.90

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

Ladies/Gentlemen:

DOCKET 50-305
OPERATING LICENSE DPR-43
KEWAUNEE NUCLEAR POWER PLANT
<u>RESPONSE TO NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION REGARDING KEWAUNEE NUCLEAR POWER PLANT PROPOSED AMENDMENT 172 FOR REACTOR COOLANT MINIMUM FLOW</u>

References: 1) NRC Letter from John G. Lamb to Mark Reddemann, "KEWAUNEE NUCLEAR POWER PLANT – REQUEST FOR ADDITIONAL INFORMATION RELATED TO PROPOSED AMENDMENT 172 TO KEWUANEE NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS (TAC NO. MB1032)," dated April 11, 2001

Attached is the Nuclear Management Company, LLC (NMC), response to a United States Nuclear Regulatory Commission Request for Additional Information (Reference 1) regarding the NMC application for permission to change Kewaunee Technical Specification 3.10.m on Reactor Coolant Minimum Flow. Nothing in this response represents a commitment not previously made in separate correspondence.

If there are questions regarding this response, please contact either Mr. Thomas J. Webb at (920) 388-8537 or me at (920) 755-7627.

Sincerely,

Mark E. Reddemann
Site Vice President

MTVN

Attach.
cc US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

A001

ATTACHMENT 1

Letter from Mark Reddemann (NMC)

To

Document Control Desk (NRC)

Dated

April 20, 2001

<u>Response to NRC Request for Additional Information</u>

On

Kewaunee Nuclear Power Plant Technical Specification 3.10.m

<u>Reactor Coolant Minimum Flow</u>

REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATION 3.10.m AMENDMENT FOR
REACTOR COOLANT MINIMUM FLOW
KEWAUNEE NUCLEAR POWER PLANT

Note:

Nuclear Regulatory Commission (NRC) requests are shown in italics. Nuclear Management Company, LLC (NMC) responses follow in plain text.

NRC REQUEST 1:

It is stated in your submittal that the design transient analyses used as the bases for the Reactor Coolant Minimum Flow value proposed in this amendment request (93,000 gpm) were calculated using RETRAN 3D in the 2D mode. However, the NRC permission for Kewaunee Nuclear Power Plant (KNPP) to use this methodology will be restricted to preclude the use the code (sic) for five transient (sic) and accidents because benchmark analyses were not performed in the Topical Report WPSRSEM-NP, Rev.3. Please confirm that all of the supporting analyses for the proposes (sic) TS changes are performed with approved method

NMC RESPONSE TO REQUEST 1:

The five analyses to which the request refers are:

1. Uncontrolled Rod Withdrawal from Sub-Critical
2. Startup with Inactive Coolant Loop
3. Anticipated Transient Without SCRAM (ATWS)
4. Main Steam Line Break (MSLB)
5. Control Rod Ejection

NMC intends to implement the requested Technical Specification (TS) value for Reactor Coolant Minimum Flow commencing with Kewaunee Cycle 25, which is scheduled to begin in Fall 2001. Accident analyses for the five events enumerated above will be part of the analysis of record for Kewaunee Cycle 25. NMC will perform analyses for these five events using the DYNODE methodology (Reference 1.1) currently approved by the NRC for this purpose.

DYNODE and RETRAN-3D are used for simulation of various NSSS events. Section 3.0 of the Topical Report WPSRSEM-NP, Revision 3 (Reference 1.2) contains seventeen sections, sixteen of which describe accidents or transients. The seventeenth section describes power distribution control. The current use of DYNODE and RETRAN-3D and the planned future use of RETRAN-3D for each of these topical report sections are discussed below. Note that current use refers to the analysis of record for the current Kewaunee Cycle 24. Future use refers to the analysis that will be used with the Reactor Coolant Flow Minimum value (93,000 gpm per loop) proposed by the amendment request. Future use analyses will be used starting with Cycle 25.

Please note that with respect to Cycle 25 in the following discussion, "RETRAN" means the use of RETRAN-3D in the 2D mode, with all applicable conditions met. For RETRAN models that NMC will develop for first use in cycles beyond Cycle 25, "RETRAN" means either:

- The use of RETRAN-3D in 2D mode with all applicable conditions met supported by a submittal; or
- The use of RETRAN-3D in other than 2D mode supported by a submittal.

1. Uncontrolled RCCA Withdrawal from a Sub-Critical Condition: This event is currently analyzed using DYNODE for the NSSS simulation. To support the revised Reactor Coolant Flow Minimum value (93,000 gpm per loop) for Cycle 25, as proposed in the amendment request, NMC will continue to use DYNODE. After Cycle 25, NMC may develop a RETRAN model to replace DYNODE for this event. If such a model is developed, it will meet all applicable RETRAN conditions, notably conditions related to initially sub-critical RETRAN models.
2. Uncontrolled RCCA Withdrawal At Power: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC will use RETRAN for NSSS simulation.
3. Control Rod Misalignment: No NSSS simulation is performed for this event.
4. Control Rod Drop: No NSSS simulation is performed for this event.
5. Chemical and Volume Control System Malfunction: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.
6. Startup of an Inactive Coolant Loop: This event is currently analyzed using DYNODE for NSSS simulation and NMC will continue to use DYNODE for Cycle 25 to support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop. After Cycle 25, NMC may develop a RETRAN model to replace DYNODE for this event. If this model is developed, it will meet all applicable RETRAN conditions, notably conditions related to RETRAN models initially at part power.
7. Excessive Heat Removal Due to Feedwater System Malfunction: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.
8. Excessive Load Increase: This event is currently analyzed using DYNODE for the NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.
9. Loss of External Load: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.
10. Loss of Normal Feedwater Flow: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.

11. Loss of Reactor Coolant Flow – Pump Trip: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.

12. Loss of Reactor Coolant Flow – Locked Rotor: This event is currently analyzed using DYNODE for NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC intends to use RETRAN for NSSS simulation.

13. Fuel Handling Accident: No NSSS simulation is performed for this event. An outside vendor performs a fuel handling accident analysis using its approved methods. NMC then verifies that the vendor analysis applies to each reload. This is currently the case for Cycle 24 and will continue to be the case in the foreseeable future.

14. Main Steam Line Break: This event is currently analyzed using DYNODE for the NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC will continue to use DYNODE. After Cycle 25, NMC may develop a RETRAN model to replace DYNODE for this event. If such a model is developed, it will meet all applicable RETRAN conditions, notably the conditions related to RETRAN models initially at zero power.

15. Control Rod Ejection: This event is currently analyzed using DYNODE for the NSSS simulation. To support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop for Cycle 25, NMC will continue to use DYNODE. After Cycle 25, NMC may develop a RETRAN model to replace DYNODE for this event. If such a model is developed, it will meet all applicable RETRAN conditions, notably the conditions related to RETRAN models initially at less than full power.

16. Loss of Coolant Accident: An outside vendor performs the loss of coolant accident analyses using its approved methods. NMC then verifies that the vendor analysis applies to each reload. This is currently the case for Cycle 24 and is expected to continue in the future.

17. Power Distribution Control Verification: This verification does not involve an NSSS simulation.

As with Section 3.0 of approved topical WPSRSEM-NP-A, Revision 2 (Reference 1.1), Section 3.0 of the Topical Report WPSRSEM-NP, Revision 3 (Reference 1.2) does not contain a section on Anticipated Transient Without SCRAM (ATWS). ATWS is currently analyzed using DYNODE for NSSS simulation. For Cycle 25, NMC will continue to use DYNODE to support the revised Reactor Coolant Flow Minimum value of 93,000 gpm per loop. After Cycle 25, NMC may develop a RETRAN model to replace DYNODE for this event. If such a model is developed, it will meet all applicable RETRAN conditions, notably the conditions related to ATWS models.

REFERENCES:

Reference 1.1: "Wisconsin Public Service Corporation "Reload Safety Evaluation Methods for Application to Kewaunee" (TAC No. 65155), NRC letter from Joseph G. Giitter to D.C. Hintz, dated April 11, 1988. (Docket No. 50-305)

Reference 1.2: "Reload Safety Evaluation Methods for Application to Kewaunee," WPSRSEM-NP, Revision 3, dated September 13, 2000 (prepared) and September 14, 2000 (reviewed).

NRC REQUEST 2:

Provide the results of the re-analyzed non-loss-of-coolant accident (LOCA) design-basis transients and accidents affected by the replacement steam generator (RSG) using assumptions consistent with RSG design and operating characteristics. This information is needed to support the proposed change in TS value of the RCS minimum flow.

NMC RESPONSE TO REQUEST 2:

(See pages attached hereto)

6.8.2 Uncontrolled RCCA Withdrawal From a Subcritical Condition

Results

Figures 6.8.2-1 through 6.8.2-5 show the transient behavior of key parameters for a reactivity insertion rate of 8.2E-4Δk/sec. The accident is terminated by a reactor trip on the high neutron flux (low setting) trip function.

The nuclear power overshoots nominal full power, but only for a very short time. Therefore, the energy release and the fuel temperature increases are small. The heat flux response, of interest for departure from nucleate boiling (DNB) considerations, is shown in Figure 6.8.2-2. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux that is less than the nominal full-power heat flux. There is a large margin to DNB during the transient since the rod surface heat flux remains below the full-power design value. In addition, there is a high degree of subcooling at all times in the core. Figures 6.8.2-3 through 6.8.2-5 show the response of the core average fuel, coolant, and cladding temperature. The average fuel temperature increases to a value that is lower than the nominal full-power value. The average coolant temperature increases to a value that is also less than the full-power nominal value.

The following list shows the comparison of the important calculated safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
Uncontrolled rod withdrawal from subcritical	2.935/1.14	2346/2750	1149/1210

Conclusions

Considering the conservative assumptions used in the accident analysis, it is concluded that in the unlikely event of a control rod withdrawal accident, the core and reactor coolant systems are not adversely affected. The peak heat flux reached remains less than the nominal full-power value. The DNBR is well above its limiting value. The peak average cladding temperature is less than its nominal full-power value. Therefore, there is no possibility of fuel or cladding damage.

6.8.3 Uncontrolled RCCA Withdrawal At Power

Results

Figures 6.8.3-1 through 6.8.3-4 show the response of nuclear power, RCS pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal (8.2E-4Δk/sec) incident starting from full power. This reactivity insertion rate is greater than that for the two highest worth banks, both assumed in their highest incremental worth region, withdrawn at their maximum speed. Reactor trips on OPΔT occur less than 2.5 seconds from the start of the accident. Since this is rapid with respect to the thermal time constants, small changes in T_{avg} and pressure result. A large margin to the minimum DNBR limit is maintained.

The response of nuclear power, RCS pressure, average coolant temperature, and DNBR for a slow RCCA withdrawal (3.0E-5Δk/sec) from full power is shown in Figures 6.8.3-5 through 6.8.3-8. Reactor trips occur on OTΔT and high RCS pressure. The rise in temperature and pressure is larger than for the rapid RCCA withdrawal. The minimum DNBR reached during the transient is greater than the minimum DNBR limit.

The nuclear power, RCS pressure, coolant average temperature, and DNBR responses for an RCCA withdrawal from 60-percent power are shown in Figures 6.8.3-9 through 6.8.3-12 for a rapid withdrawal rate (8.2E-4Δk/sec) and in Figures 6.8.3-13 through 6.8.3-16 for a slow withdrawal rate (1.5E-5Δk/sec). The results demonstrate that the OTΔT, OPΔT, and high RCS pressure trip functions adequately protect the fuel. The minimum DNBR reached is above the minimum DNBR limit.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Uncontrolled Rod Withdrawal	**Minimum DNBR**	**RCS Pressure (psia)**	**MSS Pressure (psia)**
Fast rate full power	1.383/1.14	2327/2750	1034/1210
Slow rate full power	1.368/1.14	2439/2750	1082/1210
Fast rate intermediate power	1.828/1.14	2426/2750	980/1210
Slow rate intermediate power	1.219/1.14	2429/2760	1186/1210

Conclusions

In the unlikely event of an RCCA withdrawal incident during power operation, the core and RCS are not adversely affected since the minimum value of the DNBR reached is greater than the DNBR limit for all RCCA reactivity rates. Protection is provided by the high RCS pressure, OPΔT, and OTΔT trip functions.

6.8.4 RCCA Misalignment

Results

The following list shows the comparison of the important calculated safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Control Rod Drop and Misalignment	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
High T_{avg} $F_{\Delta H}$ = 1.963	1.141/1.14	2280/2750	815/1210
Middle T_{avg} $F_{\Delta H}$ = 2.036	1.141/1.14	2280/2750	763/1210
Low T_{avg} $F_{\Delta H}$ = 2.303	1.142/1.14	2280/2750	595/1210

Conclusions

Dropped or misaligned RCCAs are not deemed to be a hazard to the safe operation of the plant because these events are clearly indicated to the operator, and the analyzed cases of the worst misaligned and dropped rod do not result in a DNBR less than the minimum DNBR limit.

For all cases of dropped banks, the reactor is tripped by the power range negative neutron flux rate trip and, consequently, dropped banks do not cause core damage.

6.8.5 Chemical and Volume Control System Malfunction

Results

Dilution During Refueling

> The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. The high-count rate is alarmed in the reactor containment and the main control room. The count rate increases in proportion to the inverse core multiplication factor. Assuming the reactor is 5-percent shut down at the required refueling boron concentration of 2200 ppm, the time to reach critical conditions is >30 minutes. This is ample time for the operator to recognize the audible high-count rate signal and isolate the reactor makeup water source by closing valves and stopping the reactor makeup water pumps.

Dilution During Startup

An evaluation of the reactor shows that the minimum time required to reduce the reactor coolant boron concentration to a concentration at which the reactor could go critical with all RCCAs in is >15 minutes. This provides adequate time for the operator to respond to the high-count rate signal and terminate dilution flow.

Dilution at Power

With the reactor in automatic control, at full power, the power and temperature increase from the boron dilution results in the insertion of the controlling RCCA bank and a decrease in shutdown margin. A continuation of the dilution and RCCA insertion would cause the rods to reach the lower limit of the maneuvering band. Before reaching this point, however, two alarms would be actuated to warn the operator of the potential accident condition. These two alarms – the low RCCA insertion limit alarm and the low-low RCCA insertion limit alarm – alert the operator to initiate normal boration.

With no boration, the required shutdown margin is maintained for at least 10 minutes during a continuous boron dilution. Therefore, ample time is available following the alarms for the operator to determine the cause, isolate the reactor water makeup source, and initiate re-boration.

If rod control is in manual, and the operator takes no action, the power rises to the high neutron flux trip setpoint and the reactor trips. Figures 6.8.5-1 through 6.8.5-5 show the response of nuclear power, pressure, coolant average temperature, heat flux, and DNBR to a boron dilution event in manual control. The boron dilution in this case is essentially identical to a rod withdrawal accident. The reactivity insertion rate due to the boron dilution is within the range of reactivity insertion rates considered in Section 6.8.3, "Uncontrolled RCCS Withdrawal At Power." Assuming a 1-percent shutdown margin, there is ample time available for the operator to terminate the dilution before the reactor can return to criticality following the trip.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

	Minimum DNBR	**RCS Pressure (psia)**	**MSS Pressure (psia)**
Chemical and volume control system malfunction	1.322/1.14	2501/2750	1056/1210

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered unlikely. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition rate due to the boron dilution is slow enough to allow the operator adequate time to determine the cause of the dilution and take corrective action before required shutdown margin is lost. The dilution event at power is shown to have adequate margin to the minimum DNBR limit.

6.8.6 Startup of an Inactive Reactor Coolant Loop

Results

The results following the startup of an idle loop with the assumptions listed above are shown in Figures 6.8.6-1 through 6.8.6-5. The heat flux response, of interest for DNB considerations, indicates that the peak heat flux reaches a value that is less than the nominal full-power value. This low heat flux, combined with a high degree of subcooling in the core at all times, results in no adverse effects to the core by the transient. No reactor trip occurs.

It is expected that the actual transient effects would be less severe than those shown because of alleviating factors that have not been taken into account. For example, the actual starting time of the reactor coolant pump is likely to be about 20 seconds rather than the 10 seconds assumed in the analysis. This means that the change in core temperature would occur more gradually than shown in the figures. Furthermore, the water entering the core is assumed to exhibit the temperature of the water in the inactive loop, providing the analysis with a high degree of conservatism.

The average temperature of the reactor coolant water increases because of the positive reactivity insertion and power increase brought about by the entry into the core of the cold water in the inactive loop. This leads to an increase in pressurizer pressure. The maximum pressure reached is well below the acceptance criteria of 2750 psia.

The following list shows the comparison of the important calculated safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

	Minimum DNBR	**RCS Pressure (psia)**	**MSS Pressure (psia)**
Startup of inactive loop	5.0/1.14	2307/2750	1064/1210

Conclusions

The results show that for startup of an inactive loop, the power and the temperature excursions are not severe. There is a considerable margin to the limiting minimum DNBR. Therefore, no undue restriction needs to be placed on the plant when starting a reactor coolant pump at power levels up to 12-percent power.

6.8.7 Excessive Heat Removal Due to Feedwater System Malfunctions

Results

Figures 6.8.7-1 through 6.8.7-5 show the transient without automatic reactor control and with a zero moderator reactivity coefficient representing beginning-of-cycle (BOC) conditions. As expected, the average reactor coolant temperature and pressurizer pressure show rapid decreases as the secondary heat extraction remains greater than the core power generation. The core power level increases slowly and eventually comes to equilibrium at a value slightly above the nominal full-power value. There is an increased margin to DNB because of the accompanying reduction in coolant average temperature. The reactor does not trip. There is a small increase in core ΔT as the heat transfer increases through the steam generator.

Figures 6.8.7-6 through 6.8.7-10 illustrate the transient with automatic reactor control. A conservatively large negative moderator coefficient (-4.0E-4 $\frac{\Delta k}{k}$/°F) representing end-of-cycle (EOC) core conditions is assumed. The large negative moderator coefficient increases reactor power, which reduces the decrease in temperature and pressure. Eventually, reactor power comes to equilibrium at a value slightly above the nominal full-power value. The minimum DNBR decreases slightly but is well above the minimum DNBR limit.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Feedwater System Malfunction	**Minimum DNBR**	**RCS Pressure (psia)**	**MSS Pressure (psia)**
BOC manual control	1.585/1.14	2202/2750	815/1210
EOC auto control	1.566/1.14	2201/2750	815/1210

Accidental Opening of Feedwater Regulating Valves

Results

Reactor power increases to slightly above the nominal full-power value due to the reactor cooldown, which is caused by the excessive feedwater flow to both steam generators. As a result, minimum DNBR decreases slightly but is well above the limiting minimum DNBR.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Feedwater System Malfunction	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
Accidental opening of FWRV	1.574/1.14	2394/2750	1154/1210

Conclusions

Feedwater system malfunction transients involving a reduction in feedwater temperature or an increase in feedwater flow rate have been analyzed. The analyses show an increase in reactor power from the reactor temperature reduction due to the excessive heat removal in the steam generators. The results of the most limiting of the feedwater malfunction transients demonstrate that considerable margin to the safety analysis acceptance criteria (minimum DNBR, primary and secondary pressure) exists throughout the transient. Therefore, there is no radioactive release or public hazard in the event of a feedwater malfunction event.

6.8.8 Excessive Load Increase Incident

Results

Figures 6.8.8-1 through 6.8.8-8 illustrate the transient with the reactor in the manual control mode. As expected, the EOC case has a much larger increase in reactor power and ΔT than the BOC case due to the moderator feedback. Both of the manual control cases demonstrate adequate minimum DNBR margin.

Figures 6.8.8-9 through 6.8.8-18 illustrate the transient assuming the reactor is in automatic control. In automatic control, the reactor power transient is greater than for the corresponding case in manual control. The automatic control cases still show adequate margin to the minimum DNBR limit.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Excessive Load Increase	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
BOC manual control	1.585/1.14	2200/2750	815/1210
BOC auto control	1.360/1.14	2202/2750	815/1210
EOC manual control	1.526/1.14	2200/2750	815/1210
EOC auto control	1.431/1.14	2223/2750	815/1210

Conclusions

The four cases analyzed show a considerable margin to the limiting minimum DNBR. It is concluded that reactor integrity is maintained throughout lifetime for the excessive load increase incident.

6.8.9 Loss of Reactor Coolant Flow

Results

A reactor-coolant-flow coastdown curve is shown in Figure 6.8.9-1. Figures 6.8.9-2 and 6.8.9-3 show the nuclear power and the average heat flux response for the two-pump loss of flow. Figure 6.8.9-4 shows the minimum DNBR as a function of time.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Loss of Flow	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
Two-out-of-two pump trip	1.242/1.14	2262/2750	1014/1210

Loss of Coolant Flow – Low Frequency

Results

Figures 6.8.9-5 through 6.8.9-8 show the core flow, nuclear power, average channel heat flux, and minimum DNBR transient responses for the underfrequency event.

Minimum DNBR is always above the minimum DNBR limit. Therefore, fuel rod integrity and safe plant shutdown are ensured.

The following list shows the comparison of the important calculated limiting safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Loss of Flow	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
Underfrequency trip	1.181/1.14	2271/2750	1008/1210

Locked Rotor Accident

Results

The coolant flow through the core is rapidly reduced to <50 percent of its initial value.

The reactor coolant flow, the core heat flux, and the RCS pressure versus time for a locked rotor accident are shown in Figures 6.8.9-9 through 6.8.9-11. The minimum DNBR for a fuel rod having initial $F_{\Delta H}$ values of 1.432, 1.480 and 1.673 for the high, middle and low T_{avg} regimes are shown in Figure 6.8.9-12. The peak $F_{\Delta H}$ rod reaches a minimum DNBR of slightly above the minimum DNBR limit. The minimum DNBR for the 1.70 $F_{\Delta H}$ fuel rod is less than the minimum DNBR limit, and the fuel rod is assumed to fail. Up to 40 percent of the fuel rods in the core can go below the minimum DNBR limit with acceptable radiological consequences. Fuel rod power census curves are generated for each reload to assess the percentage of fuel rods that are expected to go below the minimum DNBR limit of this accident.

Figure 6.8.9-13 shows the cladding temperature transient at the hot spot. Since in the worst case examined, the cladding temperature does not exceed 1800°F, it is not necessary to consider the possibility of a zirconium-steam reaction. The zirconium-steam reaction is only significant above this temperature.

The following list shows the comparison of the important calculated safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

	% Fuel Rods <DNB Limit	Max. Cladding Temp. (°F)	RCS Pressure (psia)	MSS Pressure (psia)
Locked rotor	<40*	1471/2700	2478/2750	1136/1210

*Percentage of fuel rods with $F_{\Delta H} \geq 1.432$, 1.480, or 1.673 for high, middle, or low T_{avg} regimes

Conclusions

In the loss-of-reactor-coolant flow accidents due to pump trips, it has been shown that there is adequate reactor coolant flow to maintain a minimum DNBR greater than the minimum DNBR limit. Since DNB does not occur, there is no cladding damage and no release of fission products into the reactor coolant. Therefore, once the fault is corrected, the plant can be returned to service in the normal manner.

For the locked rotor accident, since the peak pressure reached during the transient is <110 percent of the design, the integrity of the RCS is not endangered. The pressure can be considered as an upper limit because of the following conservative assumptions used in the study:

- Credit is not taken for the negative moderator coefficient.
- It is assumed that the pressurizer relief valves and sprays were inoperative.
- The steam dump is assumed to be inoperative.

The peak cladding temperature calculated for the hot spot can also be considered an upper limit because of the following:

- The hot spot is assumed to be in DNB at the start of the accident.
- A high gap coefficient is used during the transient.
- The nuclear heat released in the fuel at the hot spot is based on a zero moderator coefficient.

6.8.10 Loss of External Electrical Load

Results

Figures 6.8.10-1 through 6.8.10-5 show the transient responses for a total loss of load at BOC with zero moderator temperature coefficient assuming full credit for the pressurizer spray, pressurizer power-operated relief valves, and automatic control rod insertion. No credit is taken for the steam dump system.

Figures 6.8.10-6 through 6.8.10-10 show the responses for the total loss of load at EOC with the most negative moderator temperature coefficient (-4.0E-4Δk/°F). The rest of the plant operating conditions are the same as the case above.

The loss-of-load accident is also analyzed assuming manual RCCA control. In addition for the high-pressure case, no credit is taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump system. Figures 6.8.10-11 through 6.8.10-15 show the manual control beginning of cycle transient with zero moderator coefficient. Figures 6.8.10-16 through 6.8.10-20 show the manual control transient results at end of cycle.

The following list shows the comparison of the important calculated limiting safety parameters to their respective criteria (calculated value/acceptance criterion):

Loss of Load	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
BOC manual control	1.585/1.14	2509/2750	1183/1210
BOC auto control	1.585/1.14	2492/2750	1185/1210
EOC manual control	1.585/1.14	2505/2750	1180/1210
EOC auto control	1.585/1.14	2440/2750	1181/1210

Conclusions

The safety analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within safety analysis limits. The integrity of the core is maintained by the reactor protection system. The minimum DNBR does not fall below its initial value that is above the minimum DNBR limit.

6.8.11 Loss of Normal Feedwater

Results

Figures 6.8.11-1 through 6.8.11-5 show the plant parameters following a loss of normal feedwater accident with the assumptions listed above. Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. The auxiliary feedwater pump is delivering flow 630 seconds following the initiation of the low-low level trip, therefore reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

From Figures 6.8.11-1 through 6.8.11-5, it can been seen that at no time is the tubesheet uncovered in the steam generator receiving auxiliary feedwater flow and at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of one motor-driven pump, the initial reactor power is <102 percent of 1650 MWt. However, if the steam generator water level in one or both steam generators is above the low-low level trip point at the time of trip, then the result is a steam generator minimum water level higher than shown and an increased margin to the point at which reactor coolant water relief occurs.

The following list shows the comparison of the important calculated limiting safety parameters (calculated value/acceptance criterion):

	Minimum DNBR	RCS Pressure (psia)	MSS Pressure (psia)
Loss of feedwater	1.585/1.14	2500/2750	1163/1210

Conclusion

The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves. The loss of normal feedwater also does not result in uncovering the tubesheets of the steam generator being supplied with water.

6.8.12 Anticipated Transients Without Scram

Results

The results of the loss of main feedwater ATWS transient analysis are given below:

Loss of Main Feedwater High T_{avg} Case			
Parameter	**Max. Value**	**Min. Value**	**Assumed Nominal Operating Value**
SG pressure (psig)	1190[1]	808	809
SG wide-range (WR) level (ft)	41.6	5.0	41.6
Pressurizer pressure (psia)	2504	1758	2250
Pressurizer level (%)	63.6	13.8	34.1
T_{avg} (°F)	607	559	578.7
T_{hot} (°F)	632	559	610
Subcooling (°F)	81	59	N/A

Loss of Main Feedwater Low T_{avg} Case			
Parameter	**Max. Value**	**Min. Value**	**Assumed Nominal Operating Value**
SG pressure (psig)	1167[1]	624	626
SG WR level (ft)	41.6	6.1	41.6
Pressurizer pressure (psia)	2502	1722	2250
Pressurizer level (%)	67.5	33.4	33.5
T_{avg} (°F)	583	549	550
T_{hot} (°F)	606	554	582
Subcooling (°F)	91	98	N/A

Note:

1. The loss of main feedwater analysis does not model the steam dump system. Use of steam dumps would stabilize steam generator pressure below the steam generator safety valve setpoint.

The results of the LOMF ATWS analysis were compared to the acceptance criteria for the LOMF transient. A summary of the comparison is given below.

	LOMF ATWS Analysis	**Acceptance Criteria**
RCS pressure (psia)	2504	2750
SG pressure (psia)	1205	1210
Pressurizer level (%)	67.5	<100
Minimum DNBR (MDNBR)	N/A	>1.14

The results of the analysis show that the DSS provides the desired protection. The RCS pressure is within acceptance criteria and subcooling is not lost ensuring that no voiding occurs in the RCS.

In addition to satisfying the safety analysis acceptance criteria, it must also be ensured that the auxiliary feedwater pumps start and operate in the transient. Below are shown the steam generator pressures at the time that the auxiliary feedwater pumps start and at five seconds after they start. These pressures are calculated in the LOMF ATWS analyses. The increase in steam generator pressure after pump start is due to the turbine trip. Steam generator pressure five seconds after the auxiliary feedwater pump start is of interest since the auxiliary feedwater pump discharge trip setpoint must be maintained for five seconds before the low discharge pressure trip will actuate.

	Initial SG Pressure = 809 psig High T_{avg}	**Initial SG Pressure = 626 psig Low T_{avg}**
Pump start time (sec)	50.7	50.7
Reactor trip time (sec)	62.7	62.7
SG pressure at 50 sec. (psig)	852	647
SG pressure at 55 sec. (psig)	1030	805

A steam generator pressure of 640 psig was found to be the point where required NPSH equaled available NPSH and is deemed the lower bound for steam generator pressure at the auxiliary feedwater pump start time. The LOMF ATWS results show that in both high and low T_{avg} cases, the steam generator pressure is greater than 640 psig at the critical times for auxiliary feedwater pump start and auxiliary feedwater operation. Therefore, the auxiliary feedwater pumps can be relied on to start and run throughout the transient.

Conclusions

A re-analysis of the LOMF ATWS event with the DSS, assuming initial plant conditions at both the high and low T_{avg} values of the allowed T_{avg} window and with the RSGs, has been performed. All of the results are within the established acceptance criteria of the LOMF transient event. In addition, AFW pump performance during the LOMF ATWS transient was analyzed. The results of that analysis showed that the AFW pumps can be relied upon to start and run throughout the transient.

Therefore, the time delay inherent in the DSS design does not result in the design basis acceptance for a loss of main feedwater event being exceeded. Since design basis acceptance criteria are satisfied, the DSS is serving its intended preventative function and is providing the desire protection for the plant with RSGs over the operating temperature range allowed by the T_{avg} window.

6.8.13 Loss of AC Power to the Plant Auxiliaries

The average temperature, pressurizer water volume, and steam generator level (assuming the most conservative initial plant conditions and equipment availability) were shown in Figures 6.8.11-1 through 6.8.11-4 for a loss of normal feedwater, including a loss-of-offsite power and RCS natural circulation. It was shown in Section 6.8.11 that a loss of normal feedwater from any cause, including a loss-of-offsite power, does not result in water relief from the pressurizer relief or safety valves.

Conclusion

The loss-of-off-site power to the plant auxiliaries does not cause any adverse condition in the core since it does not result in water relief from the pressurizer relief or safety valves nor does it result in the loss of the steam generator(s) as a heat sink for residual heat removal.

6.8.14 Steam Line Break

Results

The results presented are a conservative indication of the events that would occur assuming a main steam line break, since it is postulated that all of the conditions described above occur simultaneously.

Figures 6.8.14-1 through 6.8.14-5 show the results following a main steam line break (complete severance of the pipe) downstream of the steam generator outlet nozzle flow restrictor, at initial no-load conditions and with outside power available. Core heat flux increases and is stabilized by the negative reactivity feedbacks from rising fuel temperatures and increased enthalpy in the region of the stuck rod.

The analysis assumes the boric acid of the safety injection system is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation, as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

A DNBR analysis is performed for the break downstream of the steam generator outlet nozzle flow restrictor case. The DNBR is calculated for the core conditions that existed at the time of maximum core heat flux. A conservatively high value for hot channel factor ($F_{\Delta H}$) is also assumed. The following list shows the comparison of the important calculated limiting safety parameters to its acceptance criteria (calculated value/acceptance criterion):

MSLB	Minimum DNBR
Downstream of flow restrictor	5.177/1.45

Analysis – Containment Response

Results

Figures 6.8.14-6 and 6.8.14-7 present containment pressure and temperature responses for the limiting containment response steam line break analysis cases. The list below shows, for these limiting cases, the peak calculated containment pressure, temperature, and the corresponding acceptance criteria. All cases analyzed result in a maximum containment pressure that is less than the containment design pressure limit of 60.7 psia. In addition, the limiting containment temperature profile has been evaluated. It was determined it does not create an equipment qualification concern. Although the limiting temperature profile exceeds the containment design temperature of 268°F, containment structural limits are not exceeded. The short duration of the temperature spike and the method of heat transfer to the containment shell preclude the containment shell temperature from exceeding the design temperature.

MSLB	Containment Peak Pressure (psia)	Containment Peak Temperature (°F)
14RYY2	60.5/60.7	267.6/330.0
11NYY0	56.7/60.7	274.7/330.0

Conclusions

The analyses have shown that the main steam line break acceptance criteria are satisfied.

Although DNB and possible cladding perforation are not precluded in the acceptance criteria, the safety analysis has demonstrated that DNB does not occur, provided that core $F_{\Delta H}$ under steam line break conditions is ≤ 5.25.

The peak pressure for the limiting containment response cases does not exceed the containment design pressure. The limiting temperature profile also does not create an environmental qualification concern for equipment in containment.

6.8.15 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

Results

Results of the beginning-of-life and end-of-life full-power and zero-power rod ejection analyses are shown in Figures 6.8.15-1 through 6.8.15-12. These results are also summarized below. The acceptance criteria on average fuel enthalpy (200 cal/g) and average cladding temperature (2700°F) are not exceeded. Therefore, fuel is not expected to be dispersed into the coolant under the most severe conditions of this transient.

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, <15 percent of the rods entered DNB. (This corresponds to 2 percent of the core volume.) Therefore, the position with regard to fission product release is much better than the double-ended coolant pipe break.

A detailed calculation of the pressure surge shows that, assuming an initial pressure of 2250 psia, the peak pressure reached in the transient is well within the criteria of 2750 psia. Therefore, no damage to the RCS will occur.

In the region of the hot spot, there is a large temperature gradient. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the mid-point of the rods toward the hot spot. Physics calculations indicate that the net results of this is a negative reactivity insertion. In practice, no significant bowing is anticipated since the structural rigidity of the core is more than sufficient to withstand the forces produced.

Boiling in the hot-spot region produces a net flow away from that region. However, the fuel heat is released to the water relatively slowly, and it is considered inconceivable that cross-flow would be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot-spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. Therefore, the net effect would therefore be a negative feedback. It is concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback would result. The effect was conservatively ignored in the analyses.

The following list shows a comparison of the important calculated safety parameters to their respective acceptance criteria (calculated value/acceptance criterion):

Control Rod Ejection	Max. Cladding Temp. (°F)	Max. Fuel Centerline Temp. (°F)	Max. Energy Deposition (cal/g)	RCS Pressure (psia)	MSS Pressure (psia)
BOC full power	2040/2700	4420/4700	147/200	2272/2750	930/1210

BOC zero power	2639/2700	4004/4700	148/200	2310/2750	1065/1210
EOC full power	2003/2700	4390/4700	145/200	2286/2750	931/1210
EOC zero power	2564/2700	3935/4700	144/200	2282/2750	1063/1210

Conclusions

Even on the most pessimistic basis, the analyses indicated that the fuel and cladding limits were not exceeded. It was concluded that there was no danger of sudden fuel dispersal into the coolant. The pressure surge was shown to be insufficient to exceed 2750 psia, and it was concluded that there was no danger of consequential damage to the primary coolant system. The amount of fission products released as a result of cladding rupture during DNB is considerably less than in the case of the double-ended main coolant pipe break (the design basis accident). Therefore, this is within the guidelines of 10 CFR 100.

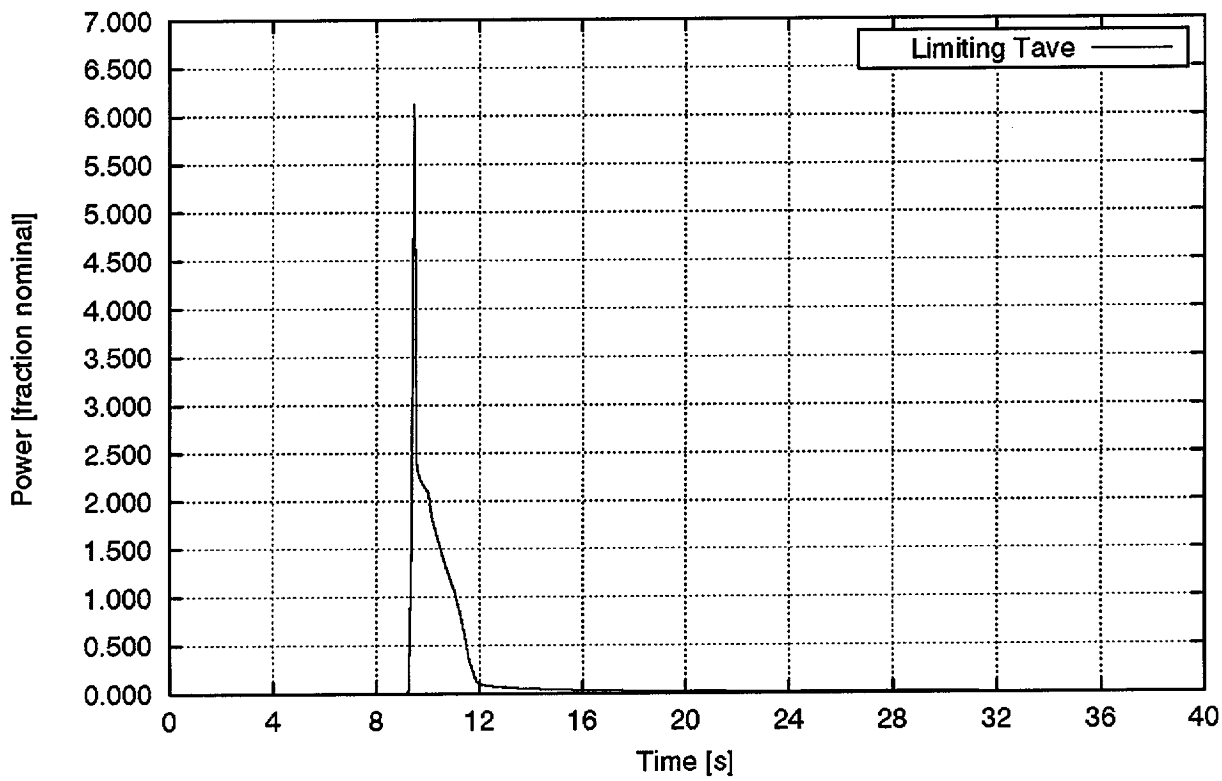


Figure 6.8.2-1
Uncontrolled RCCA Withdrawal from a Subcritical Condition
Reactor Power vs. Time

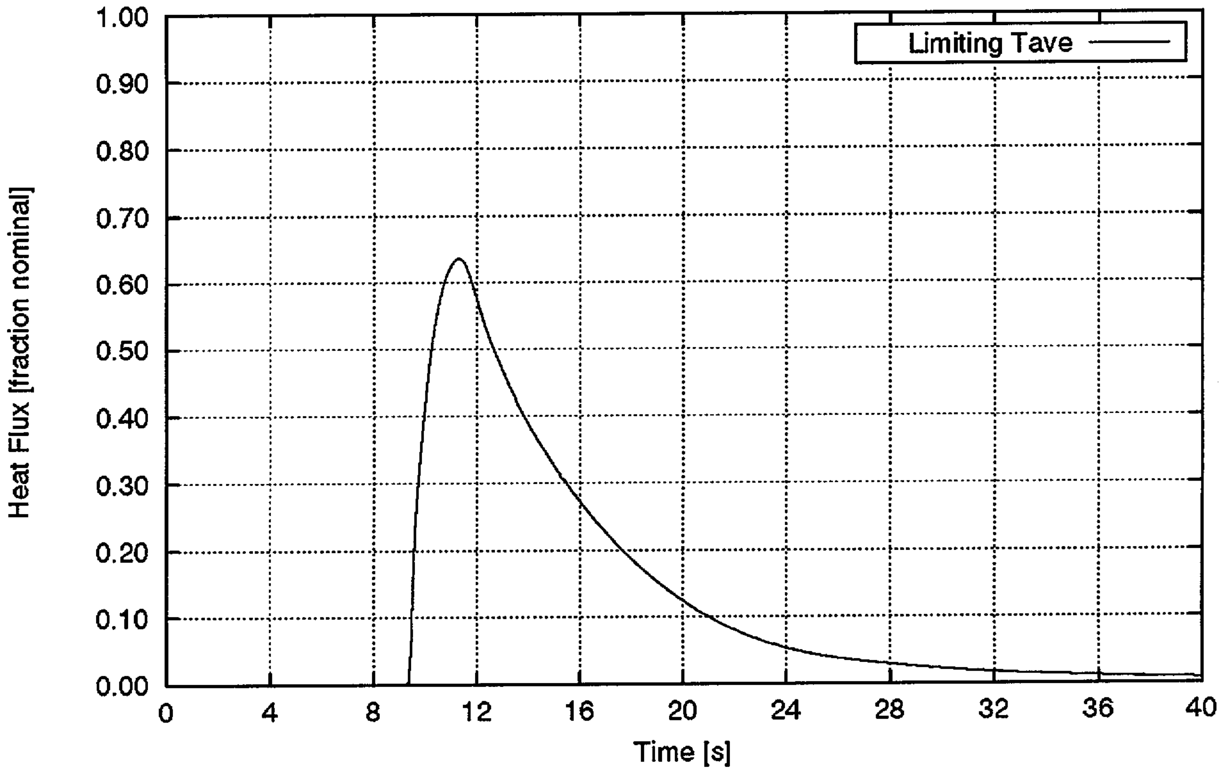


Figure 6.8.2-2
Uncontrolled RCCA Withdrawal from a Subcritical Condition
Heat Flux vs. Time

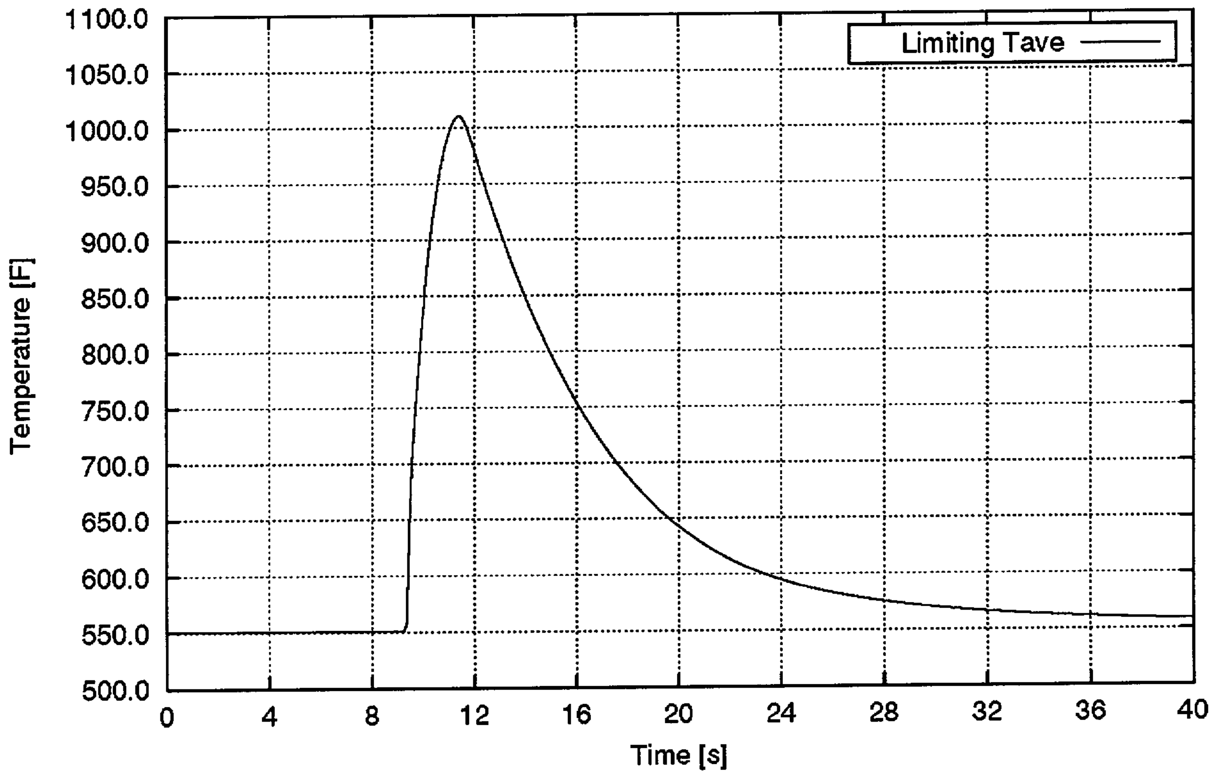


Figure 6.8.2-3
Uncontrolled RCCA Withdrawal from a Subcritical Condition
Fuel Temperature vs. Time

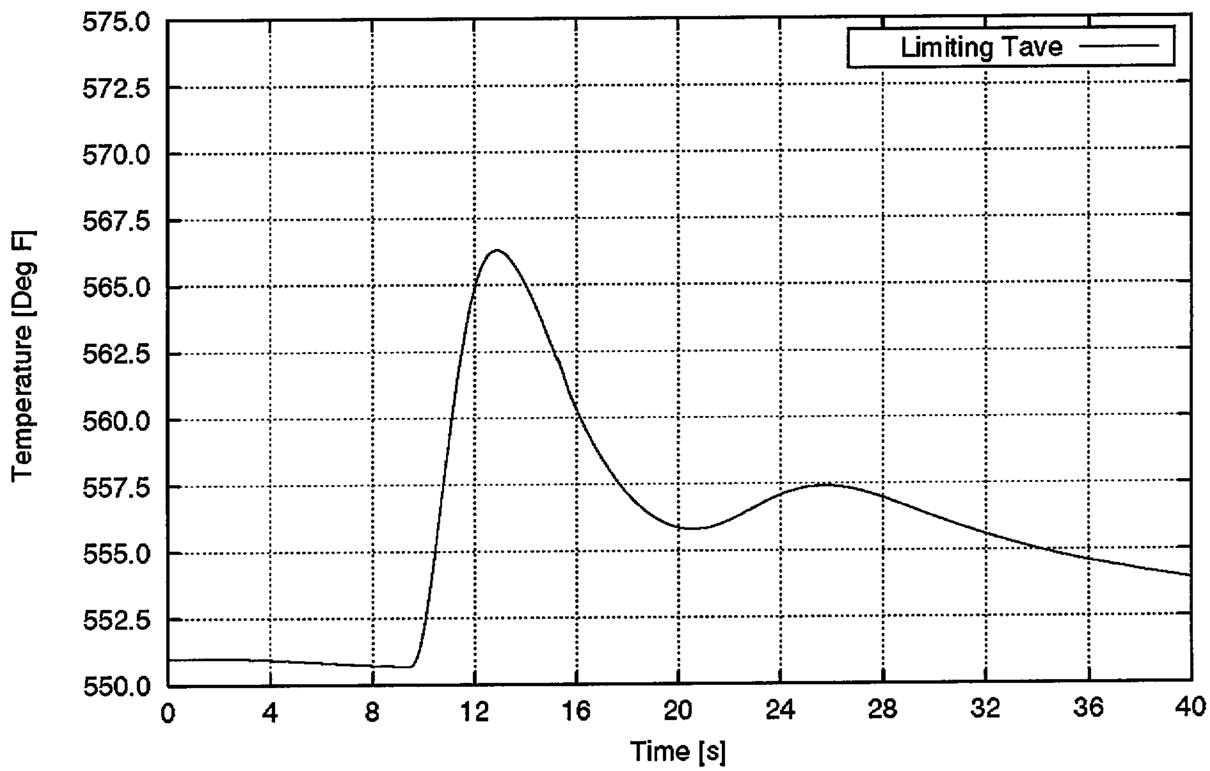


Figure 6.8.2-4
Uncontrolled RCCA Withdrawal from a Subcritical Condition
T_{avg} vs. Time

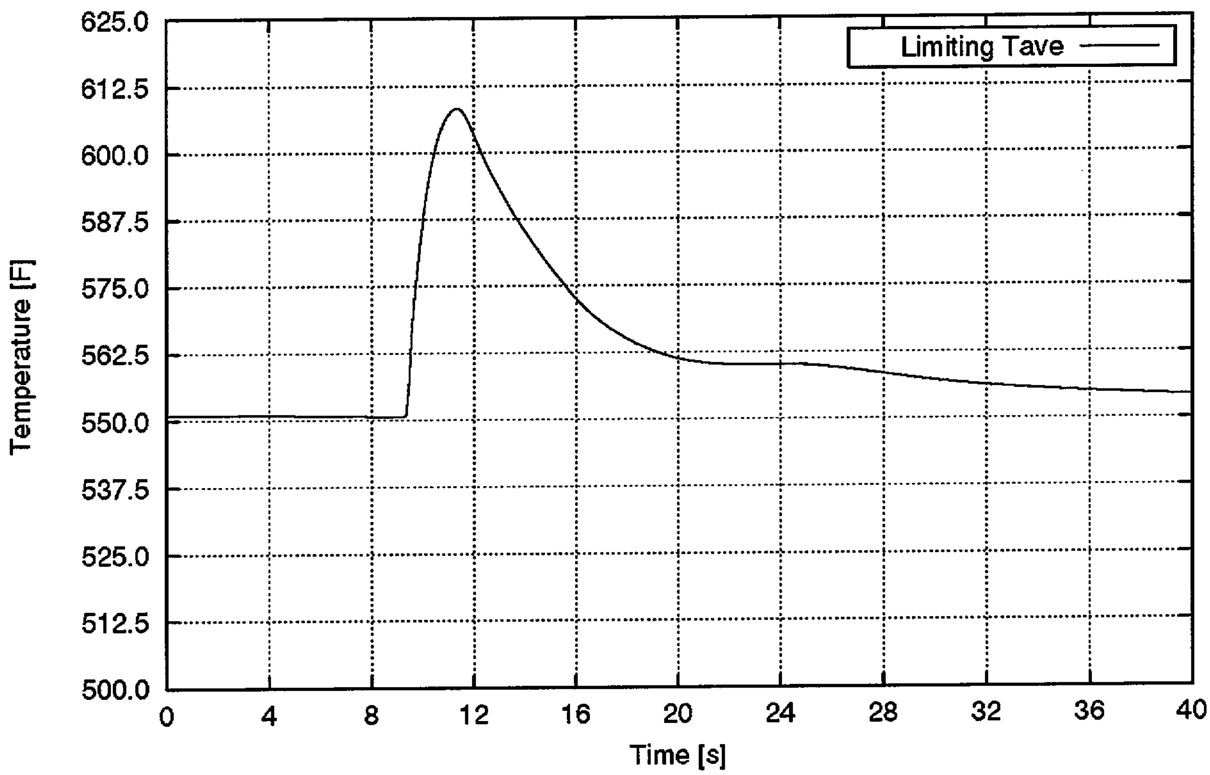


Figure 6.8.2-5
Uncontrolled RCCA Withdrawal from a Subcritical Condition
Hot Spot Cladding Temperature vs. Time

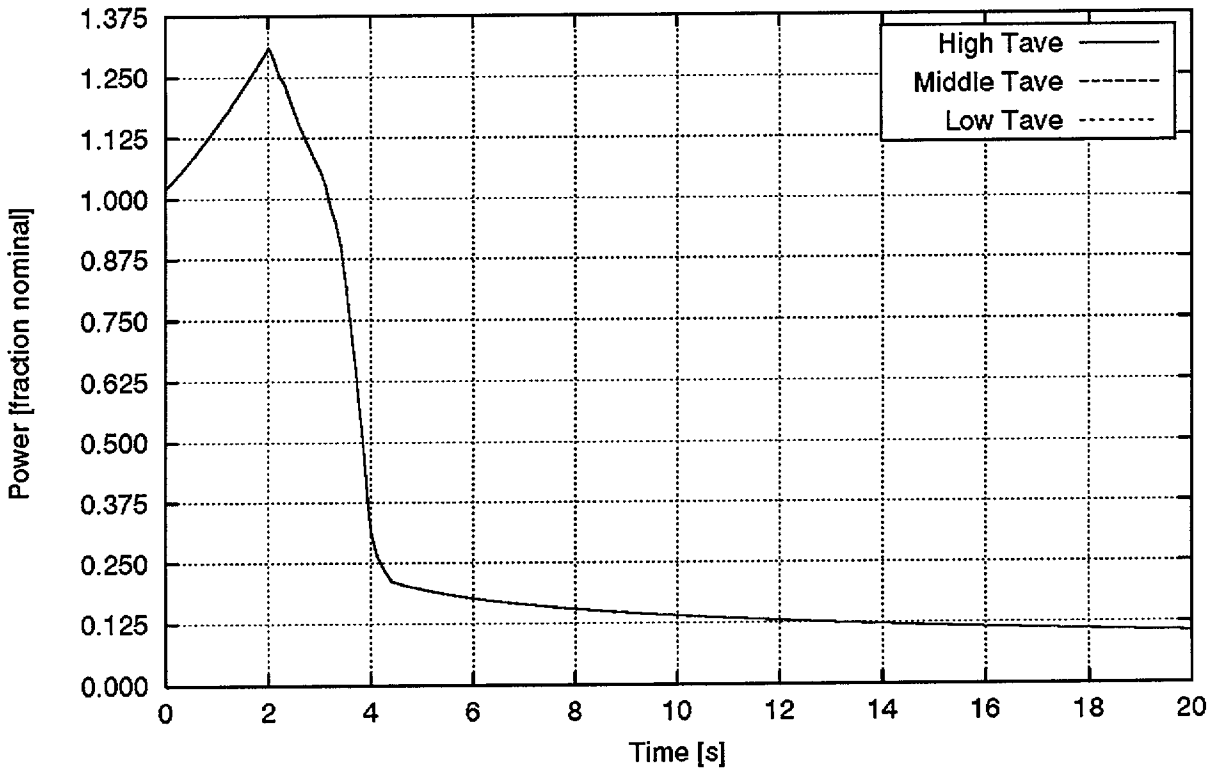


Figure 6.8.3-1
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
Reactor Power vs. Time

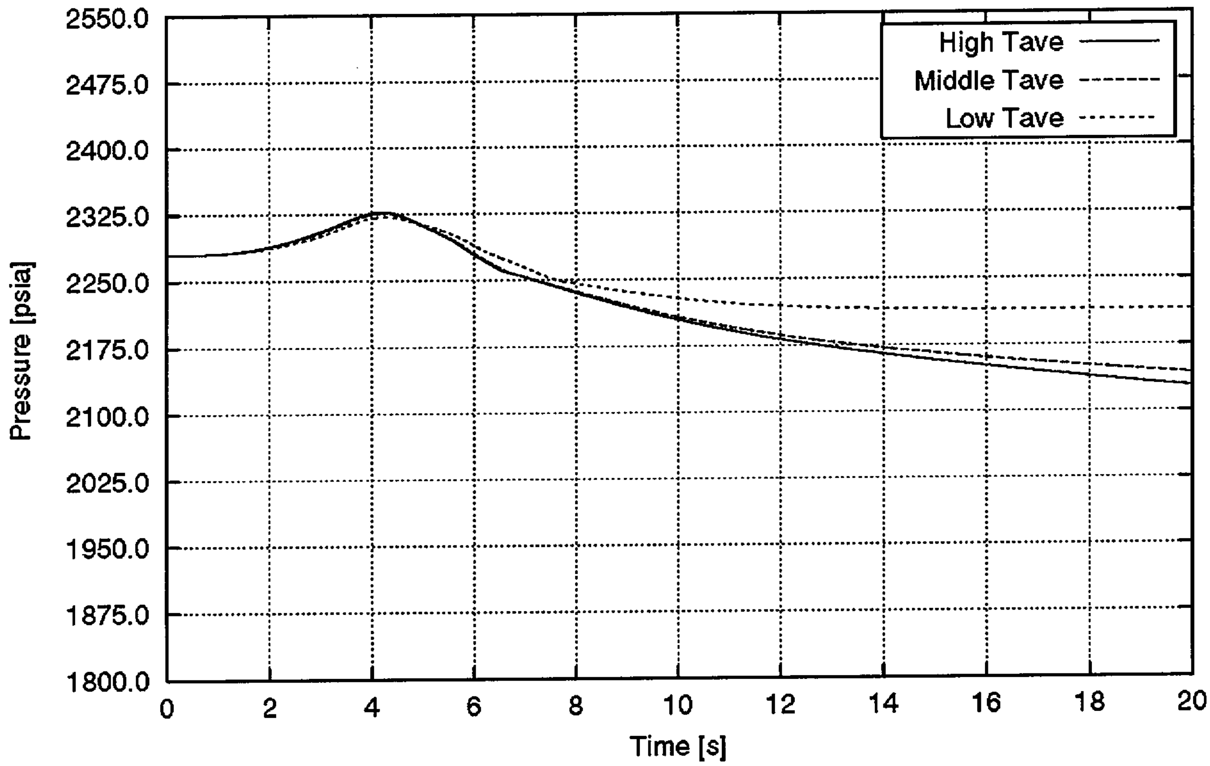


Figure 6.8.3-2
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
Pressurizer Pressure vs. Time

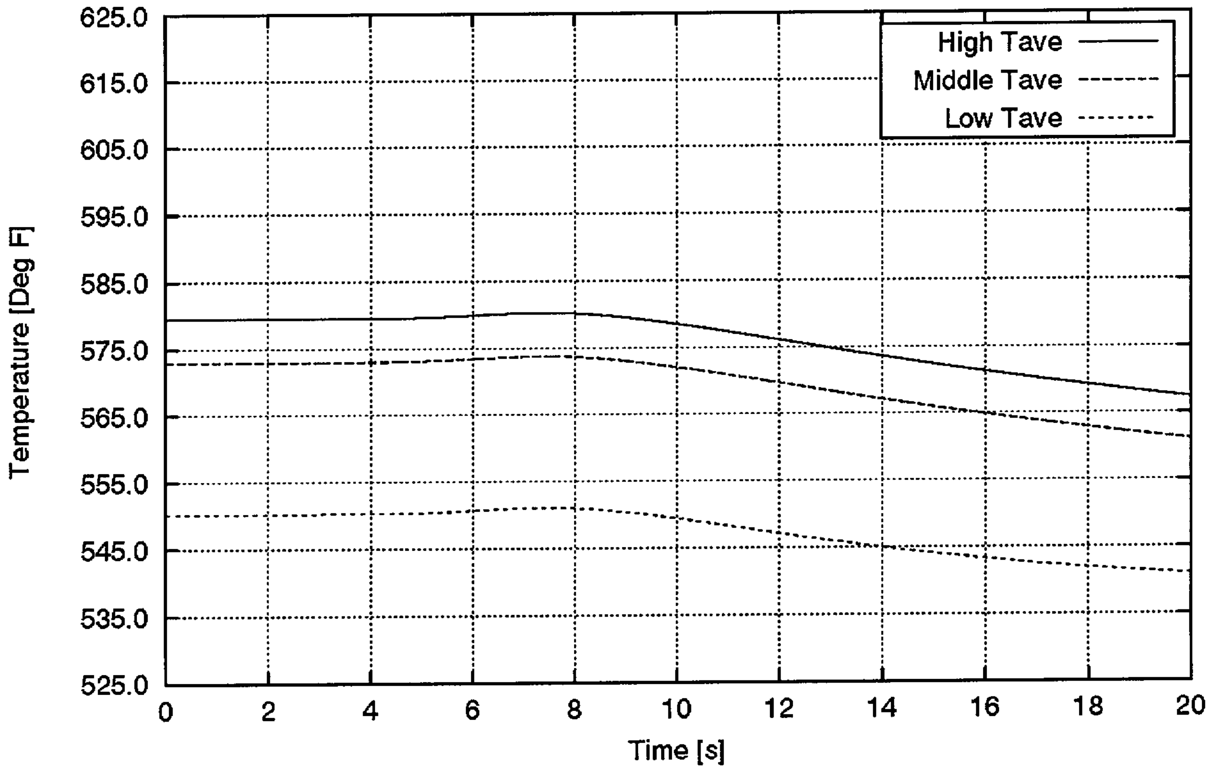


Figure 6.8.3-3
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power - High Pressure
T_{avg} vs. Time

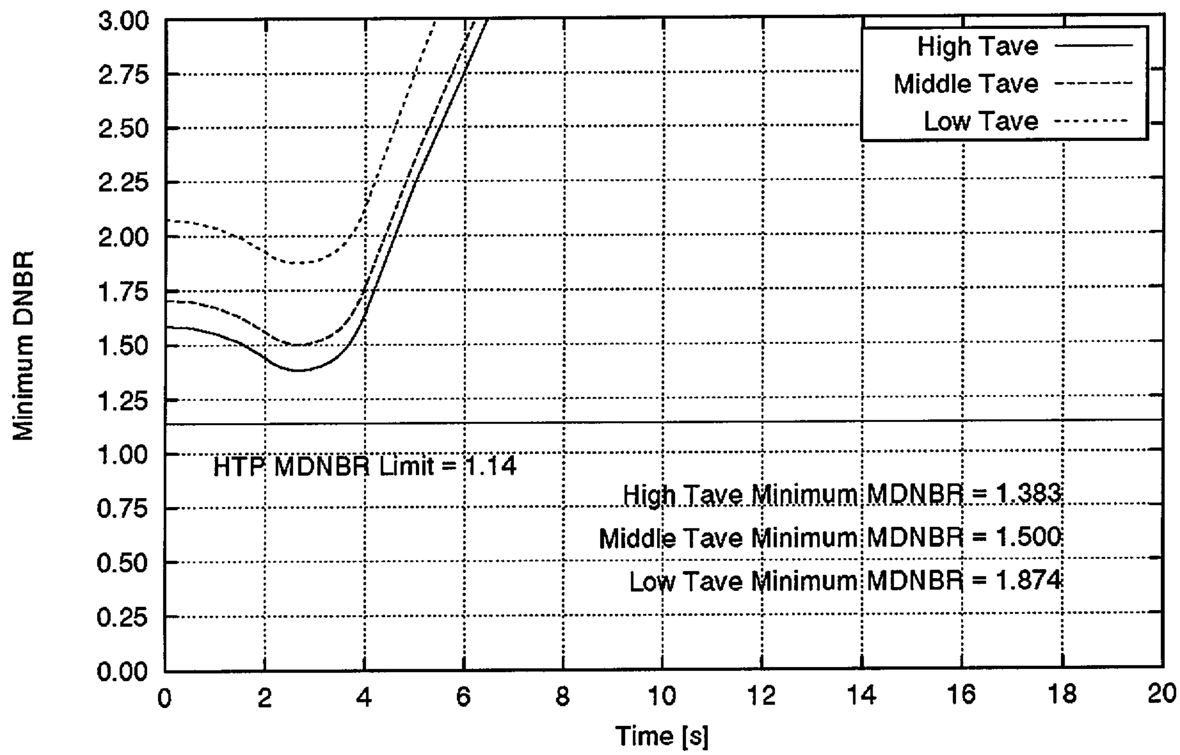


Figure 6.8.3-4
Uncontrolled RCCA Withdrawal - Fast Rate 100% Power
Minimum DNBR vs. Time

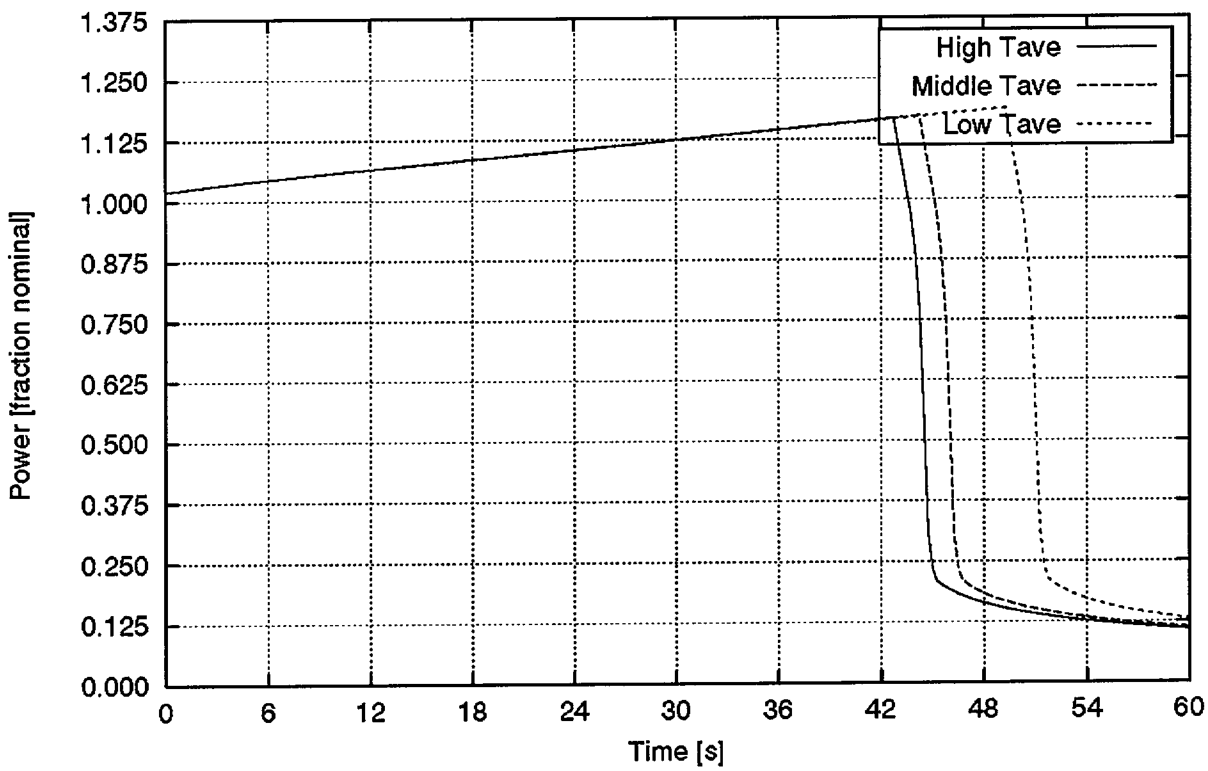


Figure 6.8.3-5
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Reactor Power vs. Time

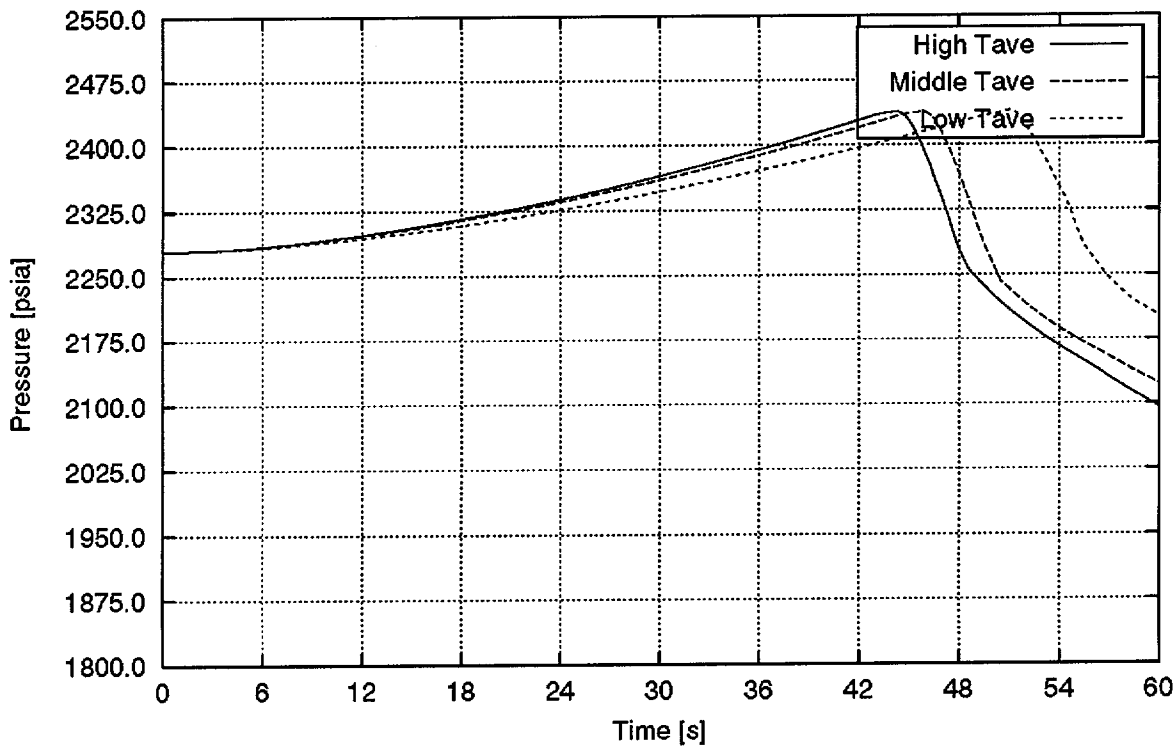


Figure 6.8.3-6
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
Pressurizer Pressure vs. Time

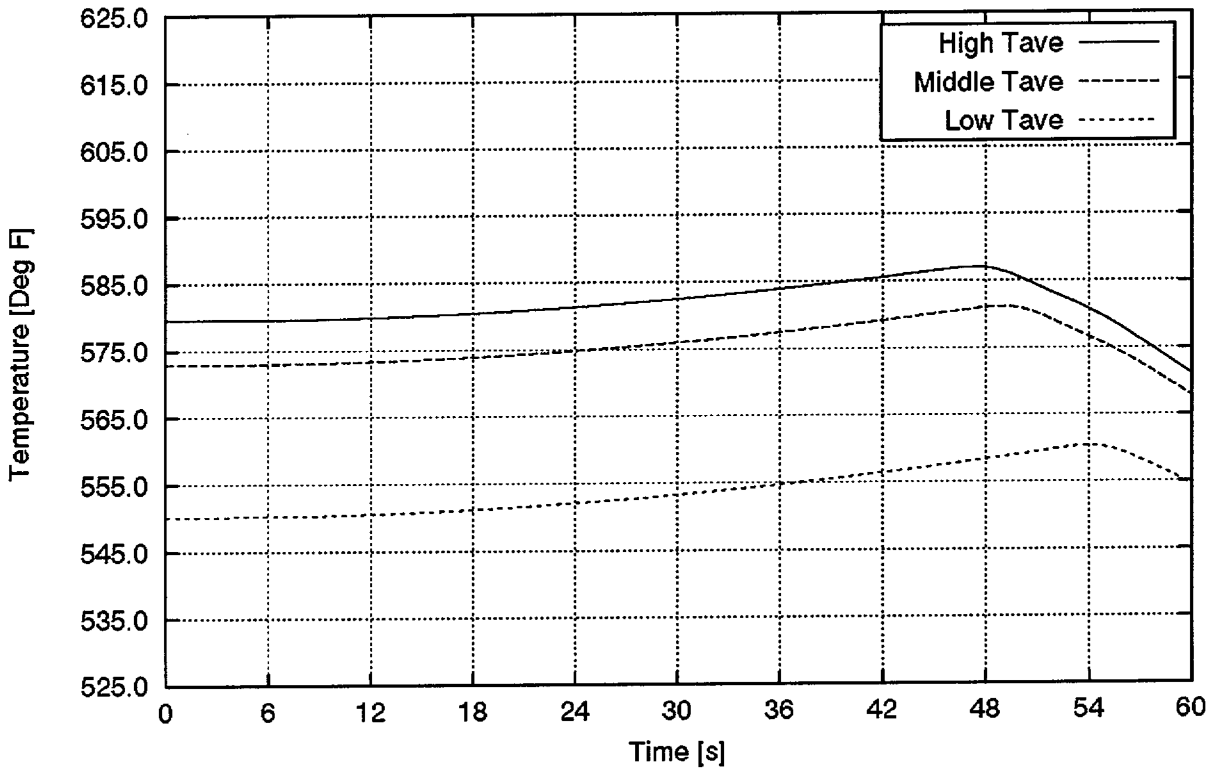


Figure 6.8.3-7
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power - High Pressure
T_{avg} vs. Time

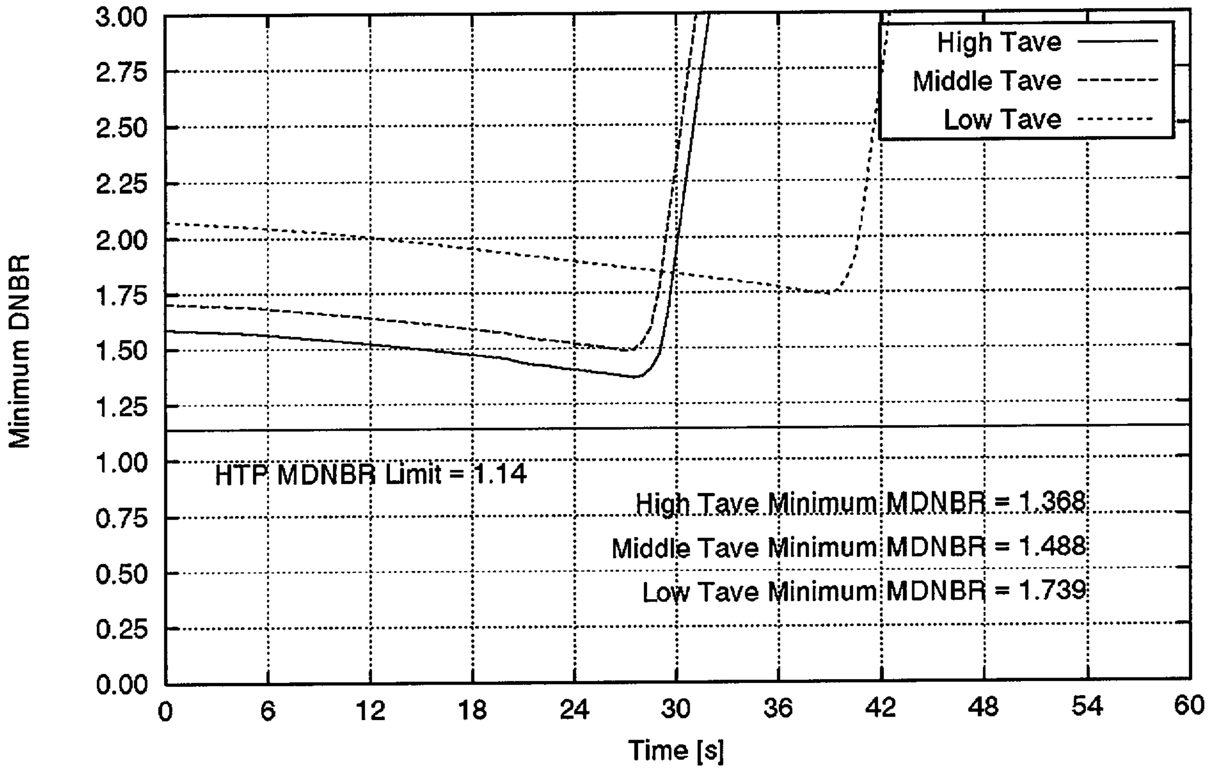


Figure 6.8.3-8
Uncontrolled RCCA Withdrawal - Slow Rate 100% Power
Minimum DNBR vs. Time

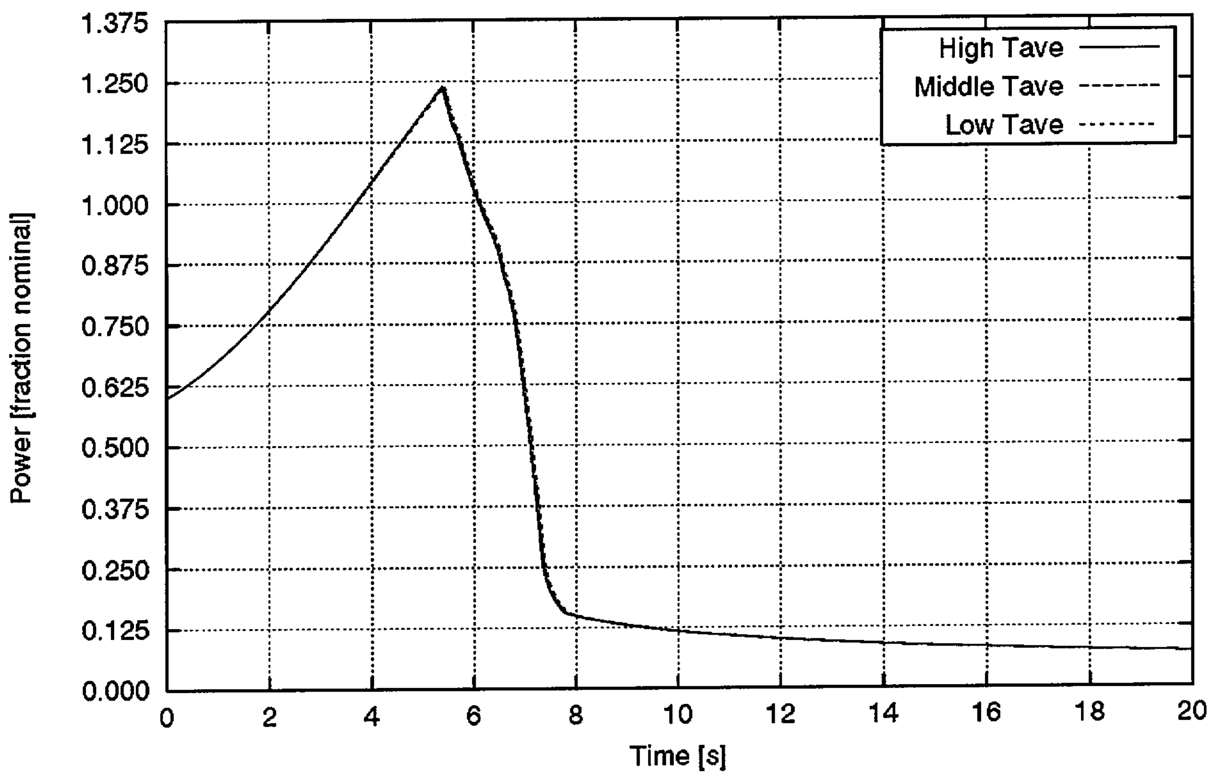


Figure 6.8.3-9
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
Reactor Power vs. Time

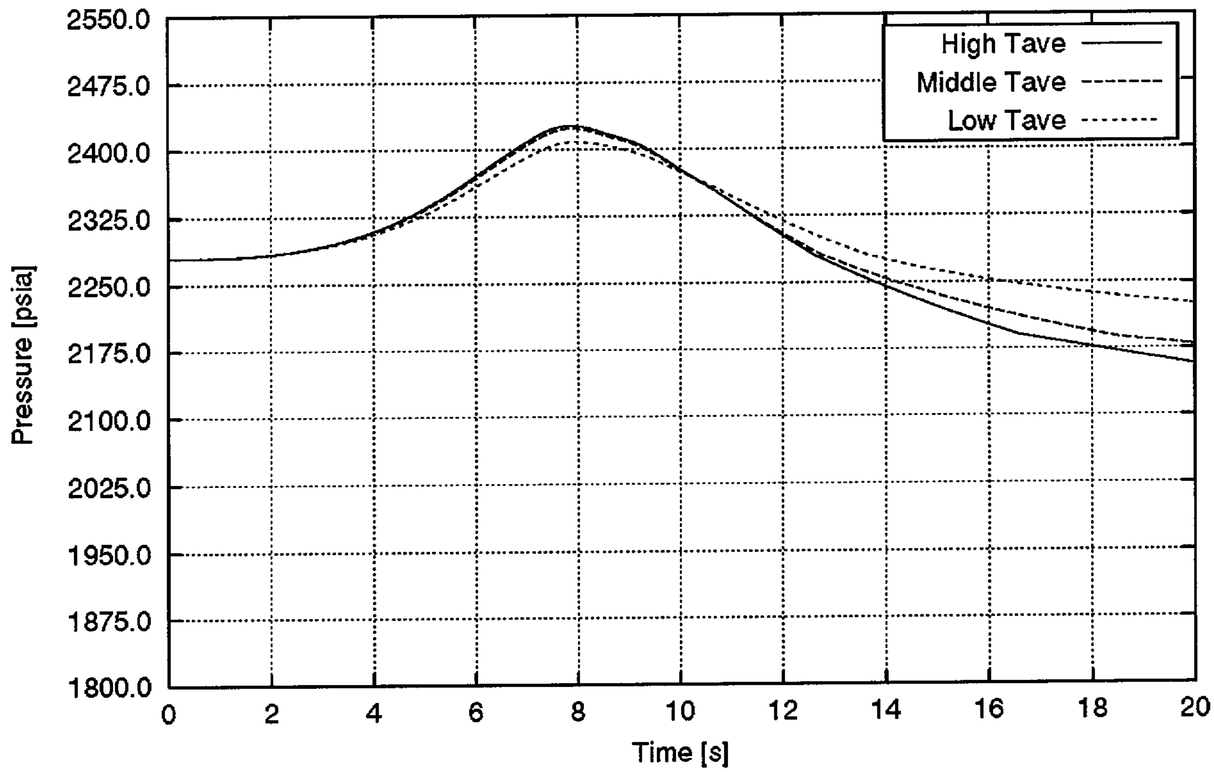


Figure 6.8.3-10
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
Pressurizer Pressure vs. Time

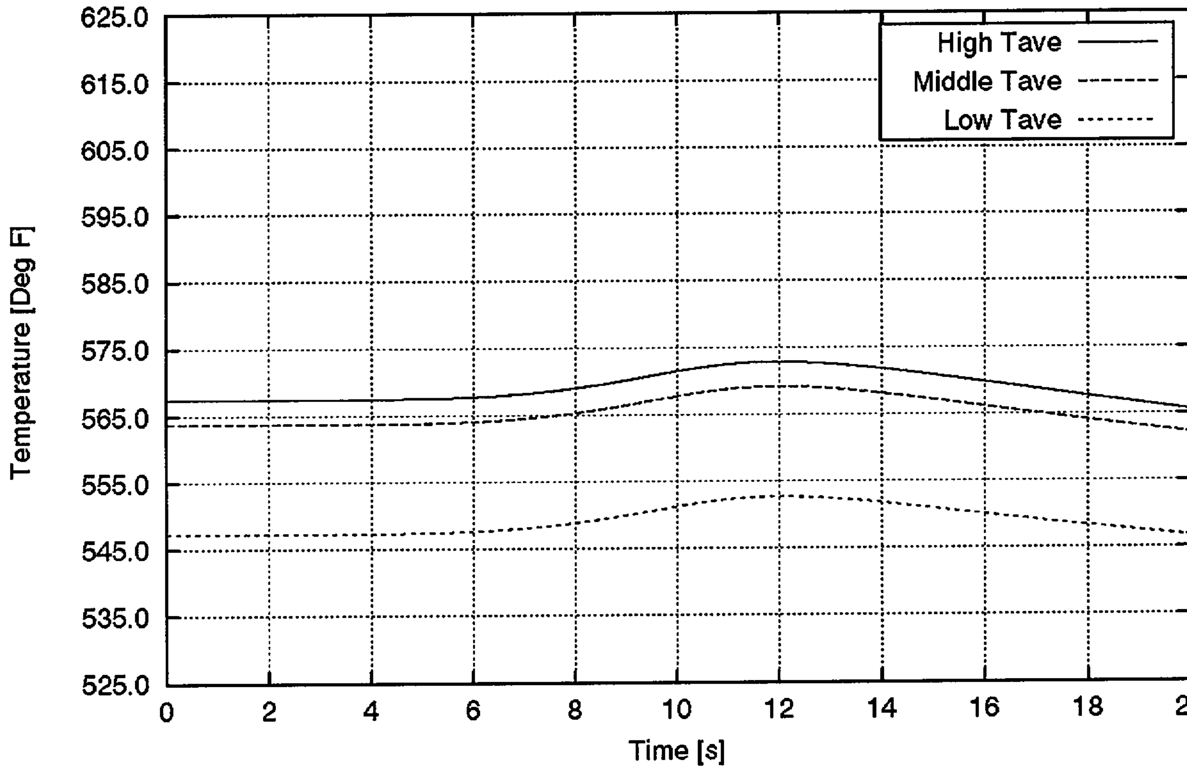


Figure 6.8.3-11
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power - High Pressure
T_{avg} vs. Time

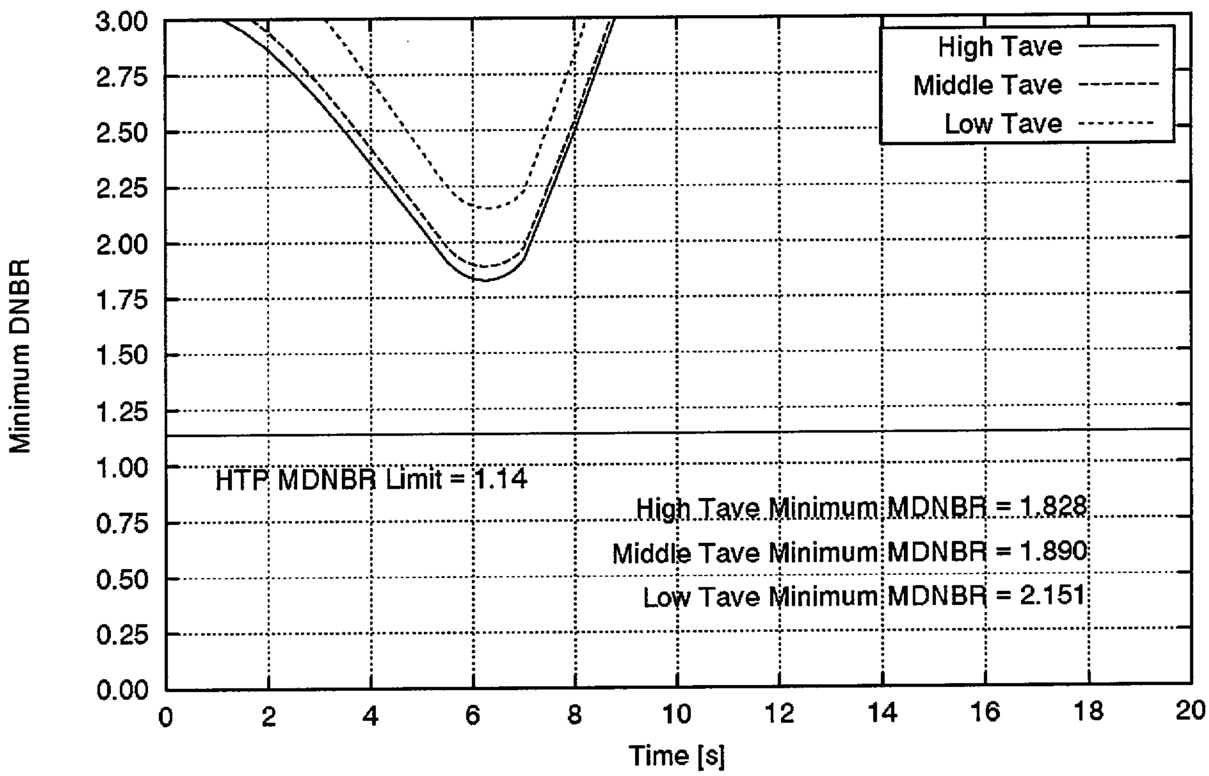


Figure 6.8.3-12
Uncontrolled RCCA Withdrawal - Fast Rate 60% Power
Minimum DNBR vs. Time

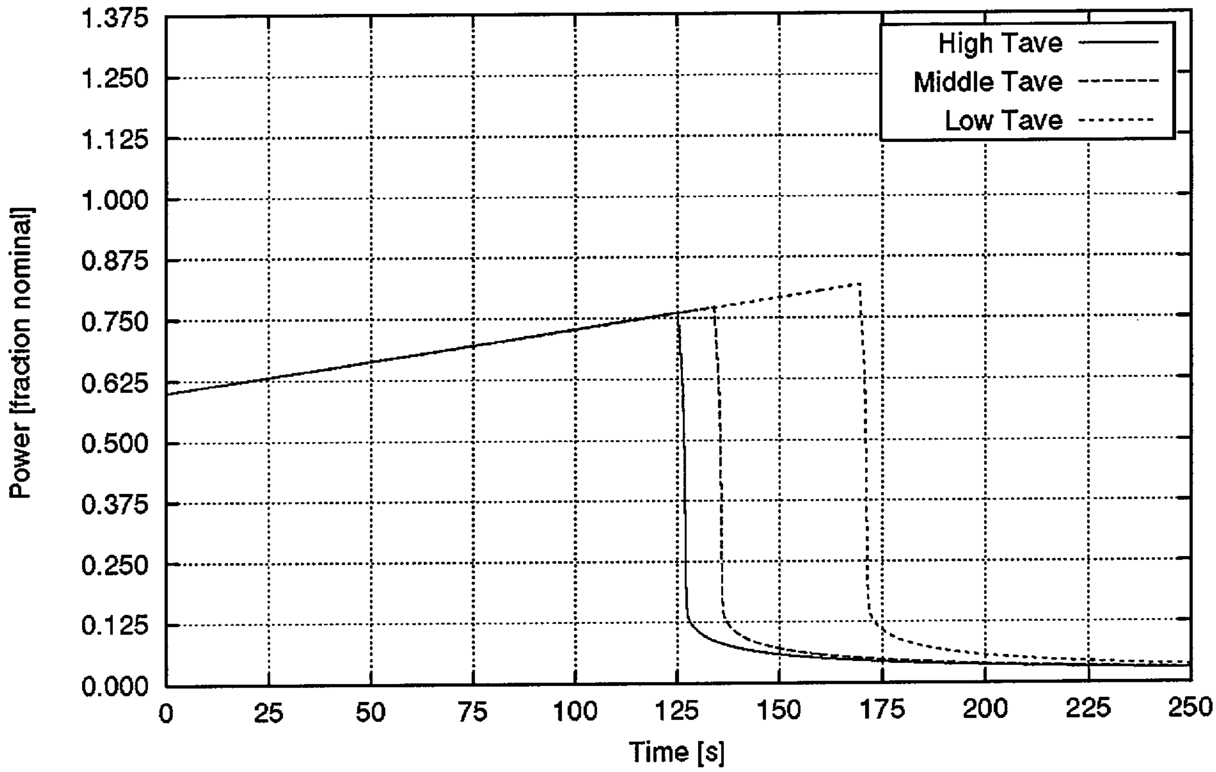


Figure 6.8.3-13
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
Reactor Power vs. Time

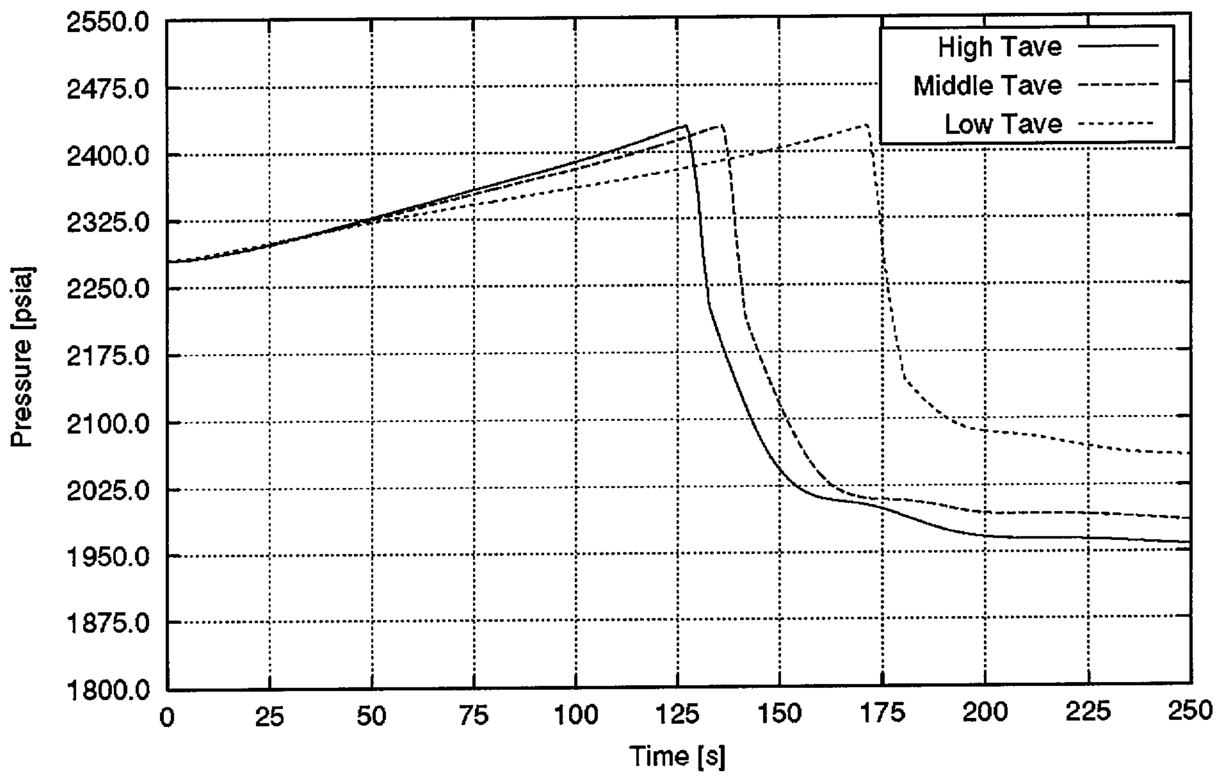


Figure 6.8.3-14
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
Pressurizer Pressure vs. Time

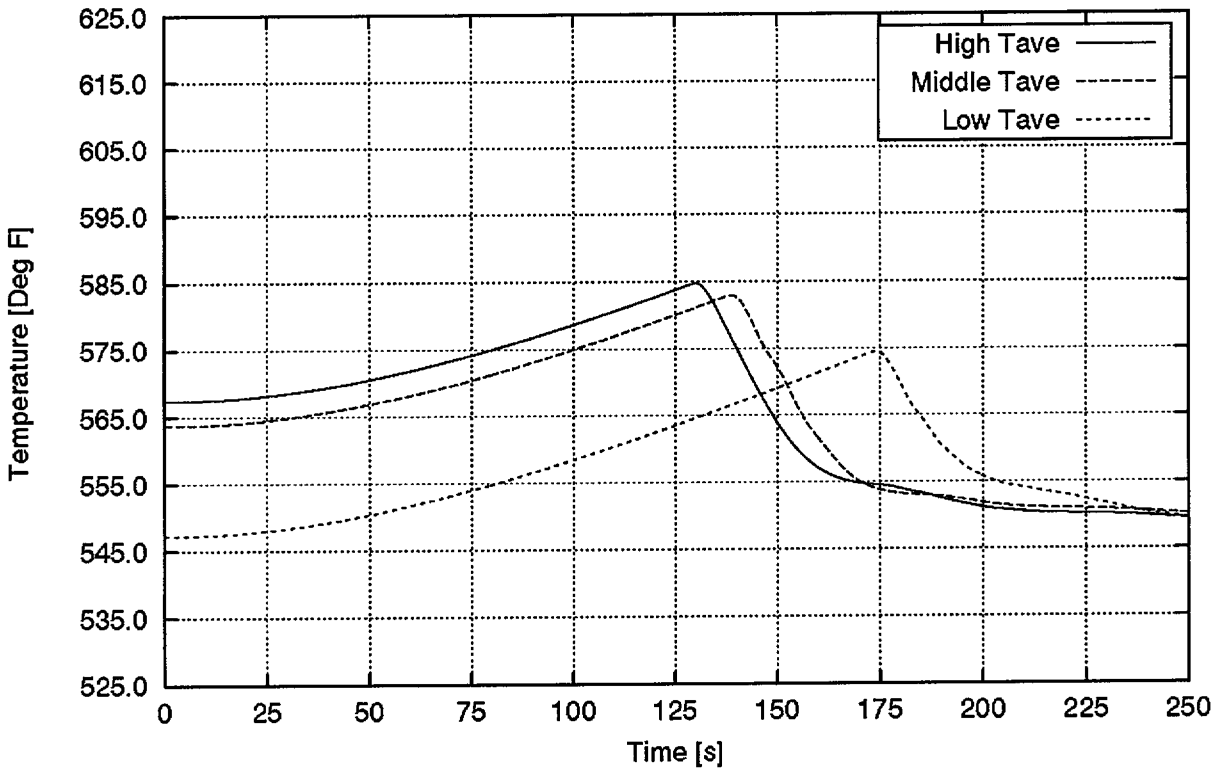


Figure 6.8.3-15
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power - High Pressure
T_{avg} vs. Time

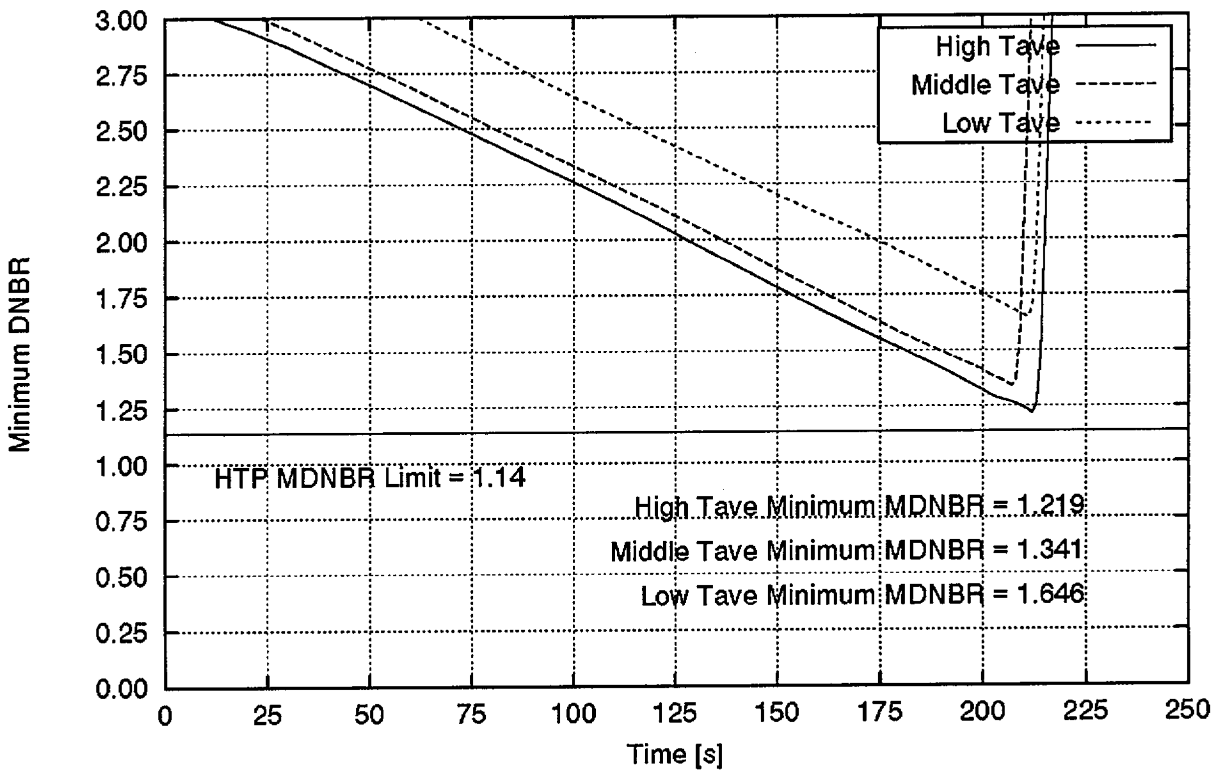


Figure 6.8.3-16
Uncontrolled RCCA Withdrawal - Slow Rate 60% Power
Minimum DNBR vs. Time

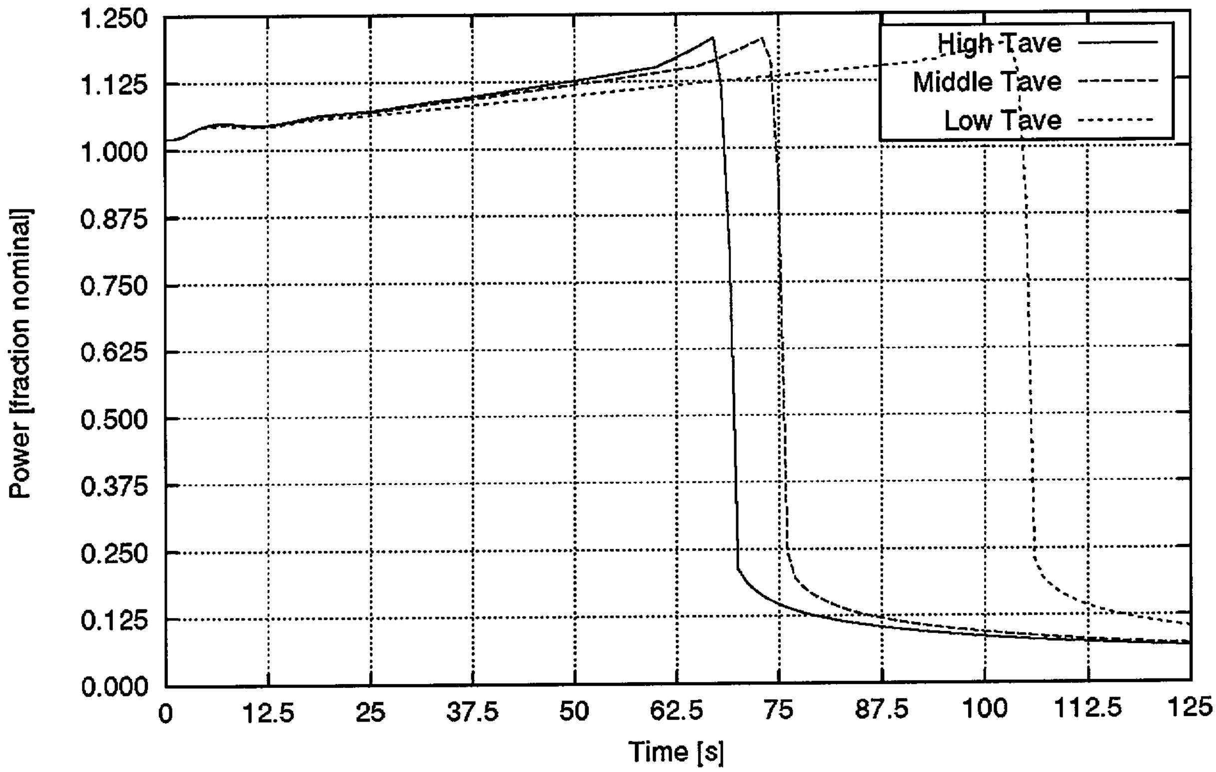


Figure 6.8.5-1
CVCS Malfunction – Dilution at Power - Manual Control - High Pressure
Reactor Power vs. Time

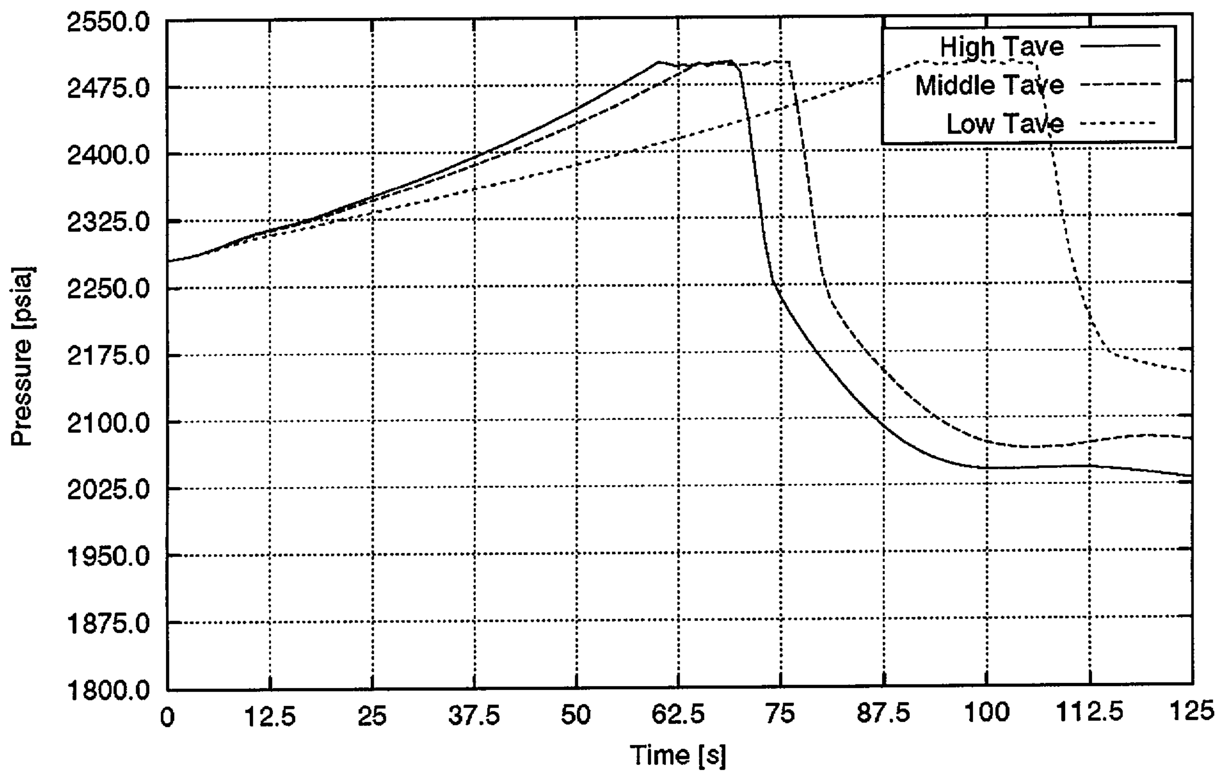


Figure 6.8.5-2
CVCS Malfunction – Dilution at Power - Manual Control - High Pressure
Pressurizer Pressure vs. Time

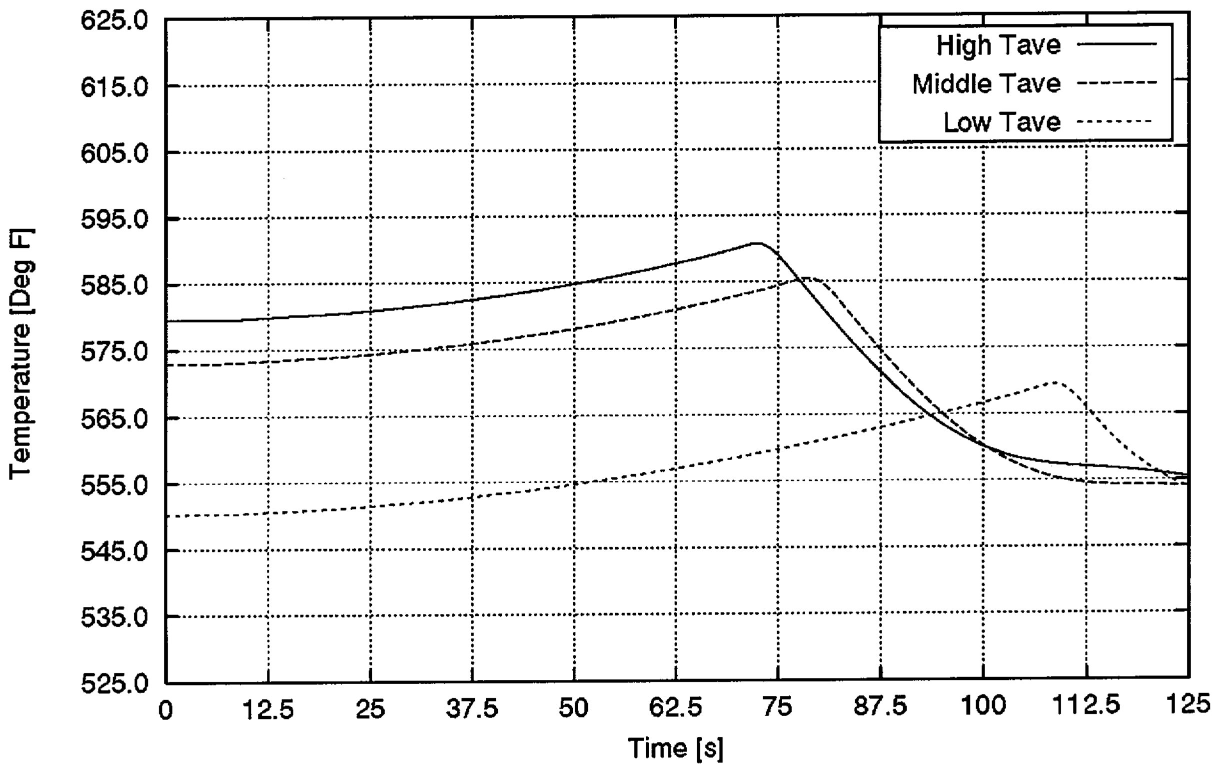


Figure 6.8.5-3
CVCS Malfunction – Dilution at Power - Manual Control - High Pressure
T_{avg} vs. Time

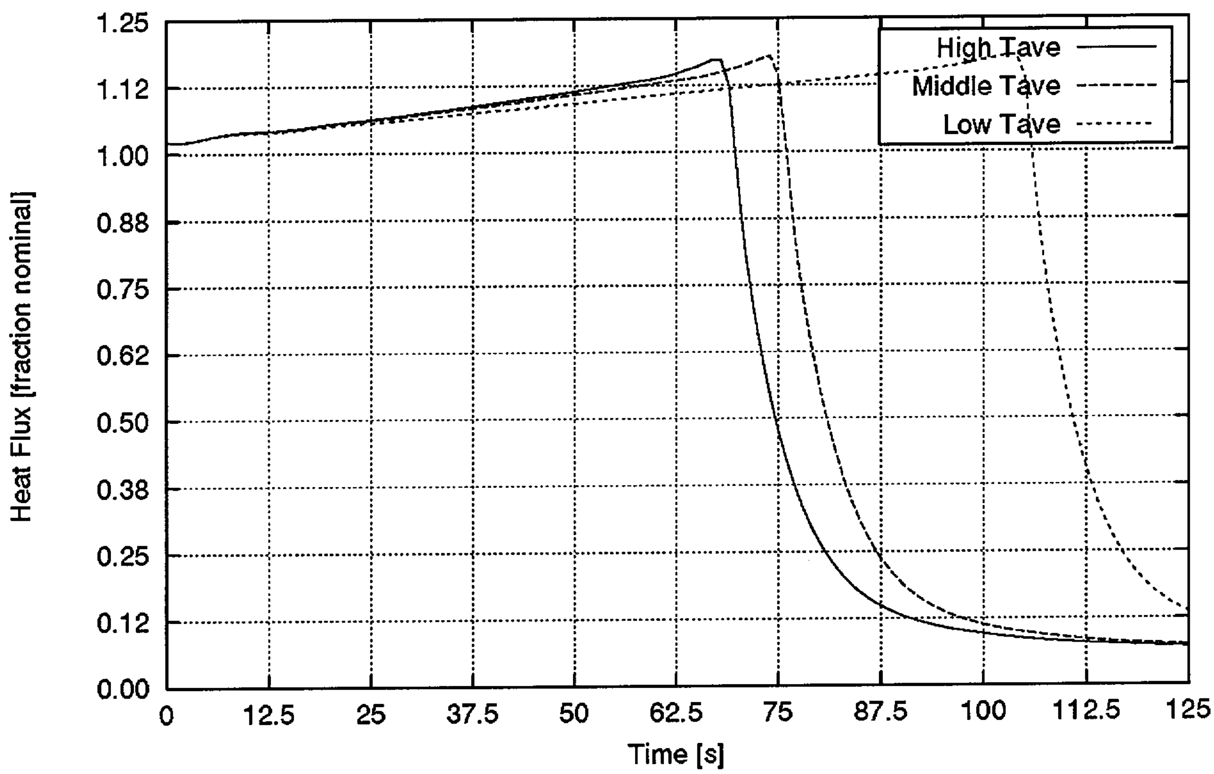


Figure 6.8.5-4
CVCS Malfunction – Dilution at Power - Manual Control - High Pressure
Heat Flux vs. Time

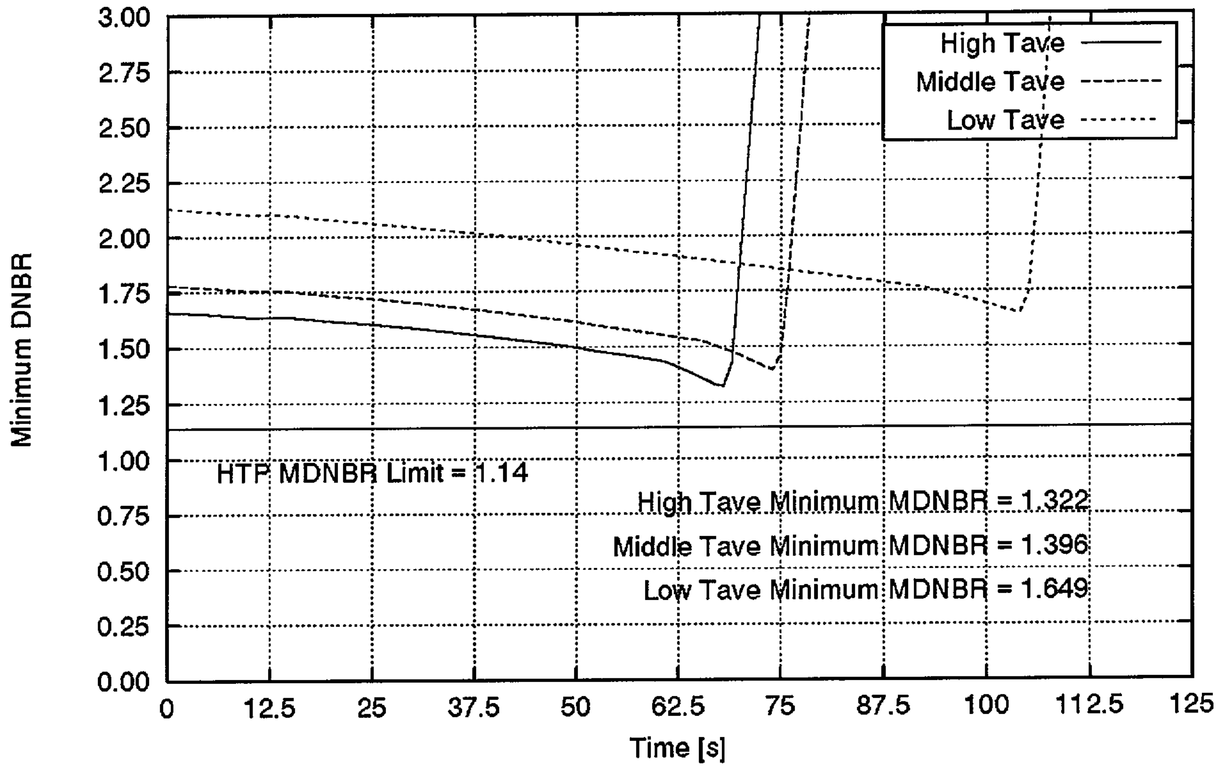


Figure 6.8.5-5
CVCS Malfunction – Dilution at Power - Manual Control - High Pressure
Minimum DNBR vs. Time

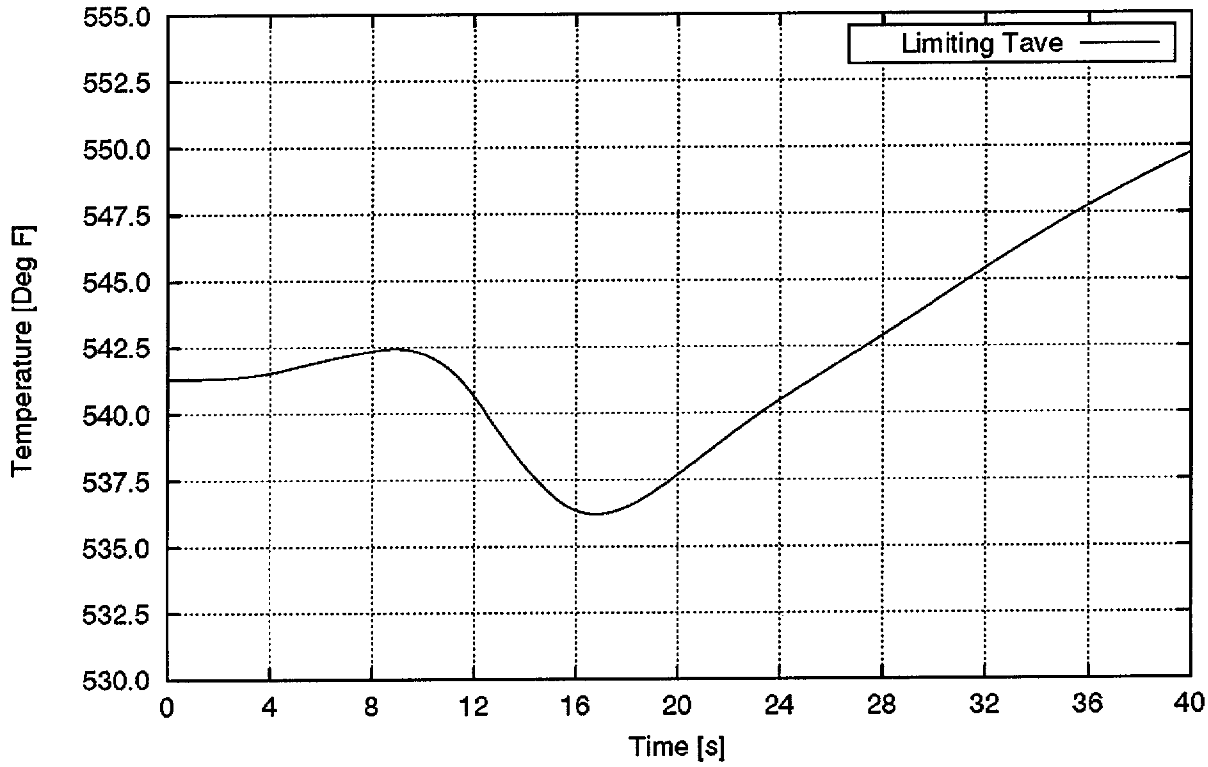


Figure 6.8.6-1
Startup of an Inactive Reactor Coolant Loop
T_{inlet} vs. Time

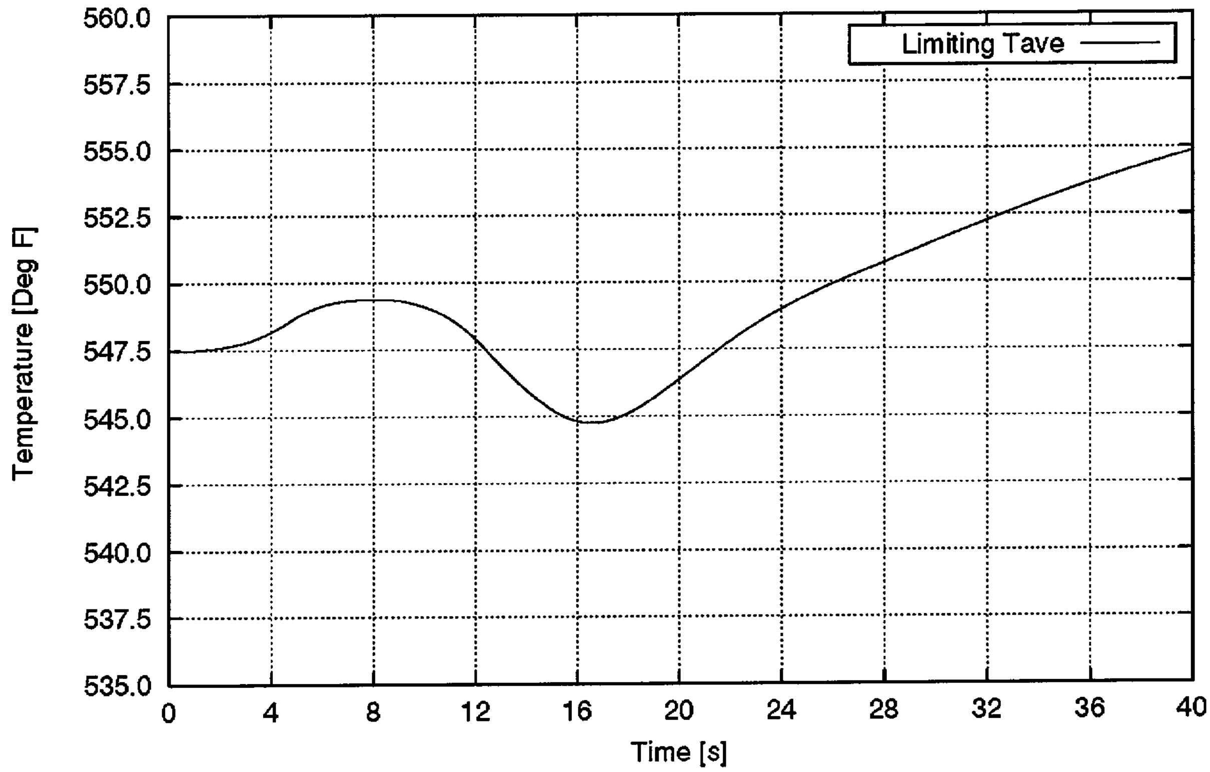


Figure 6.8.6-2
Startup of an Inactive Reactor Coolant Loop
T_{avg} vs. Time

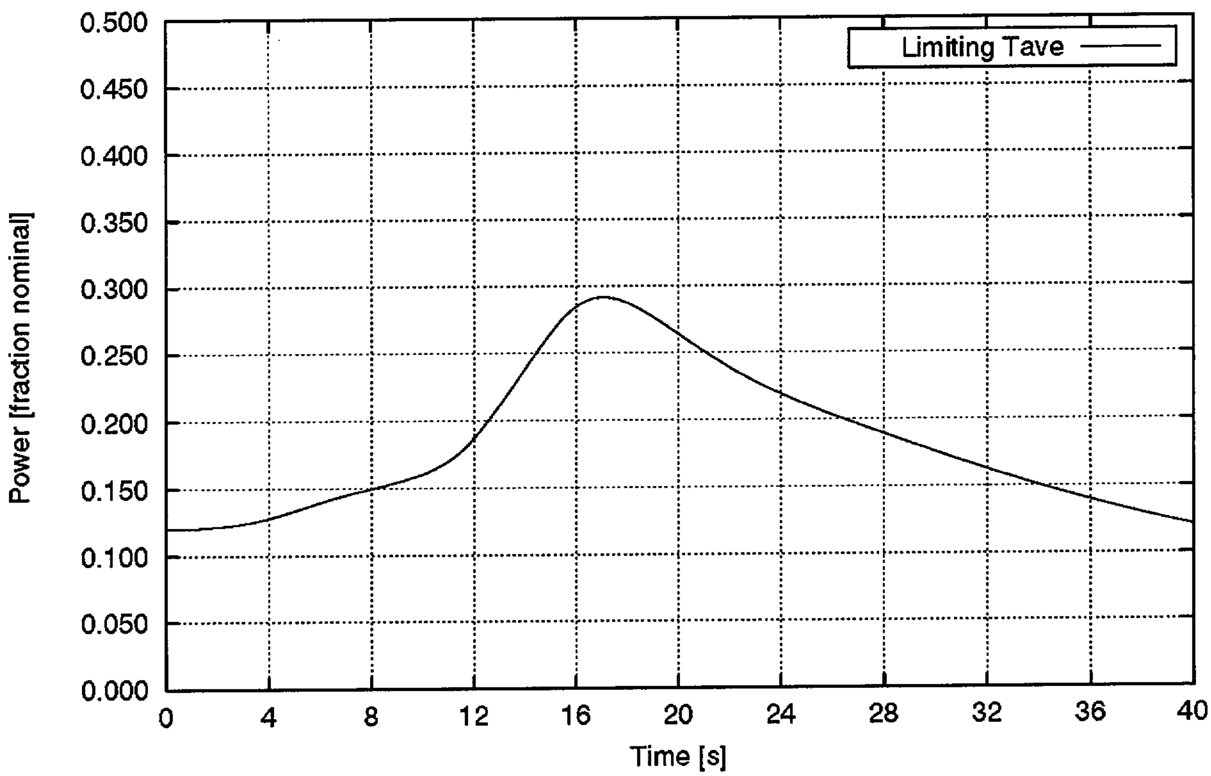


Figure 6.8.6-3
Startup of an Inactive Reactor Coolant Loop
Reactor Power vs. Time

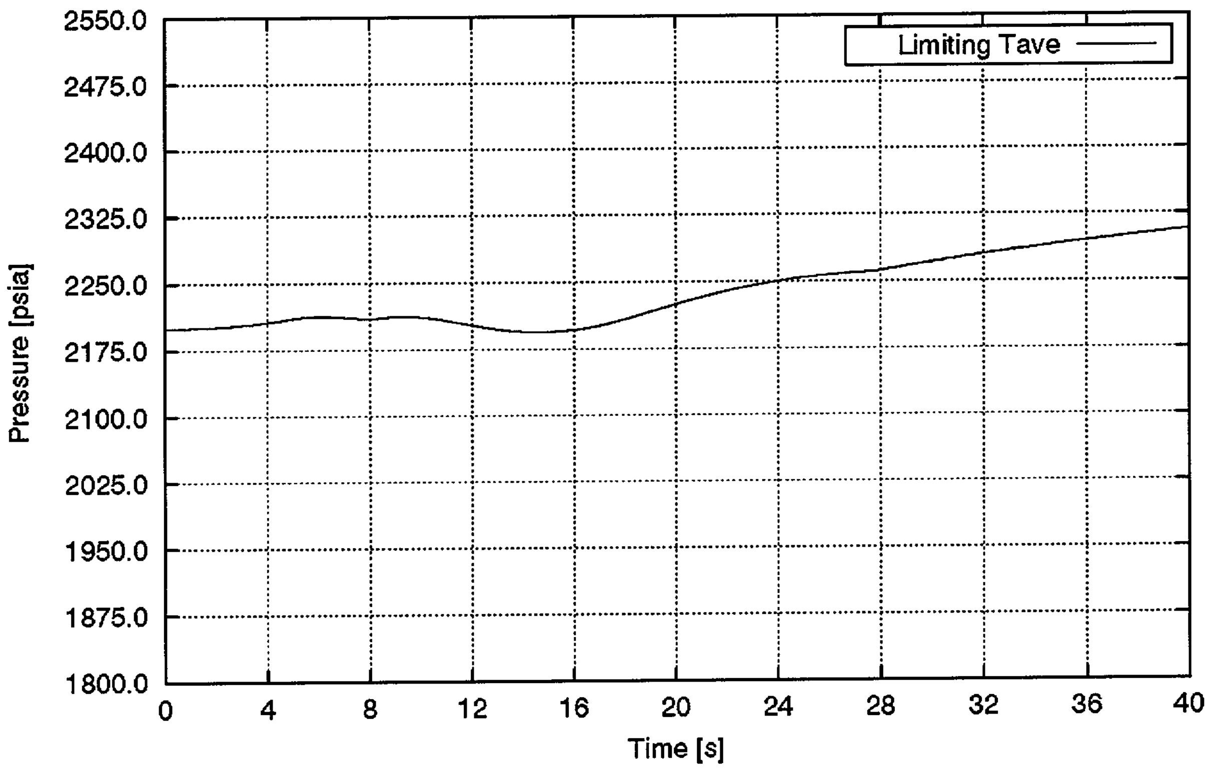


Figure 6.8.6-4
Startup of an Inactive Reactor Coolant Loop
Pressurizer Pressure vs. Time

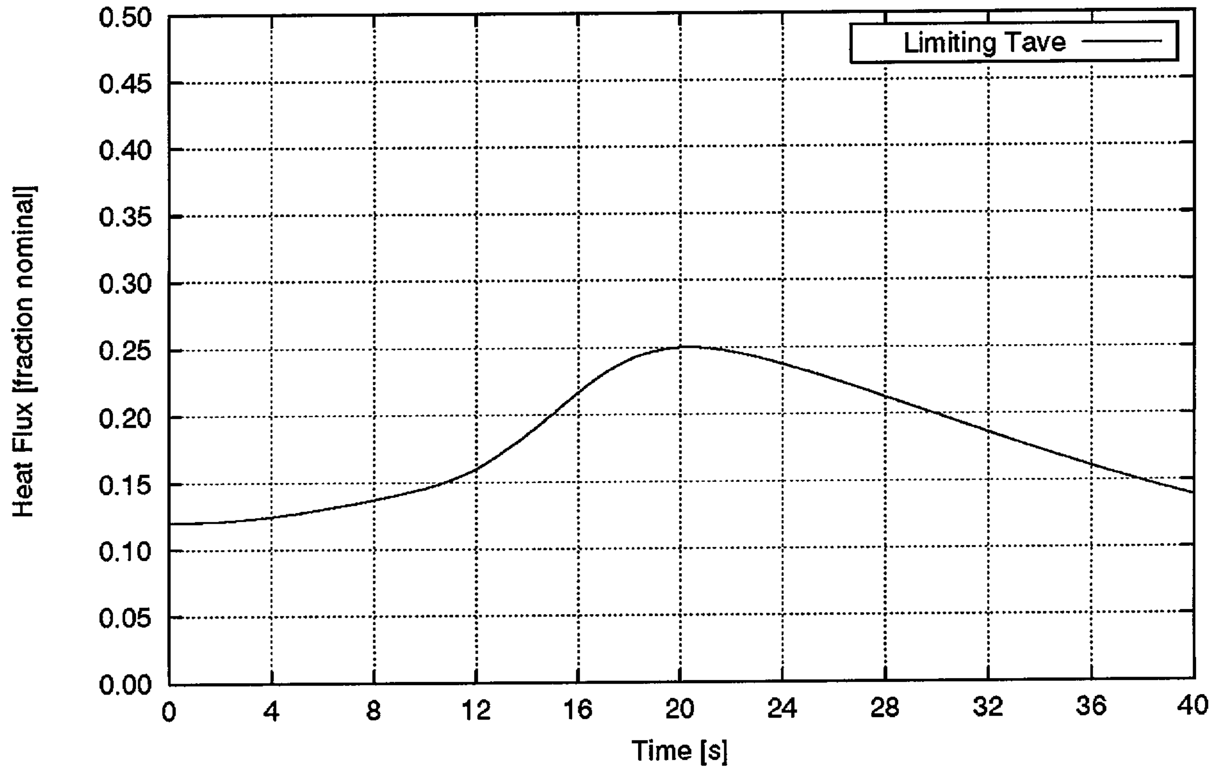


Figure 6.8.6-5
Startup of an Inactive Reactor Coolant Loop
Heat Flux vs. Time

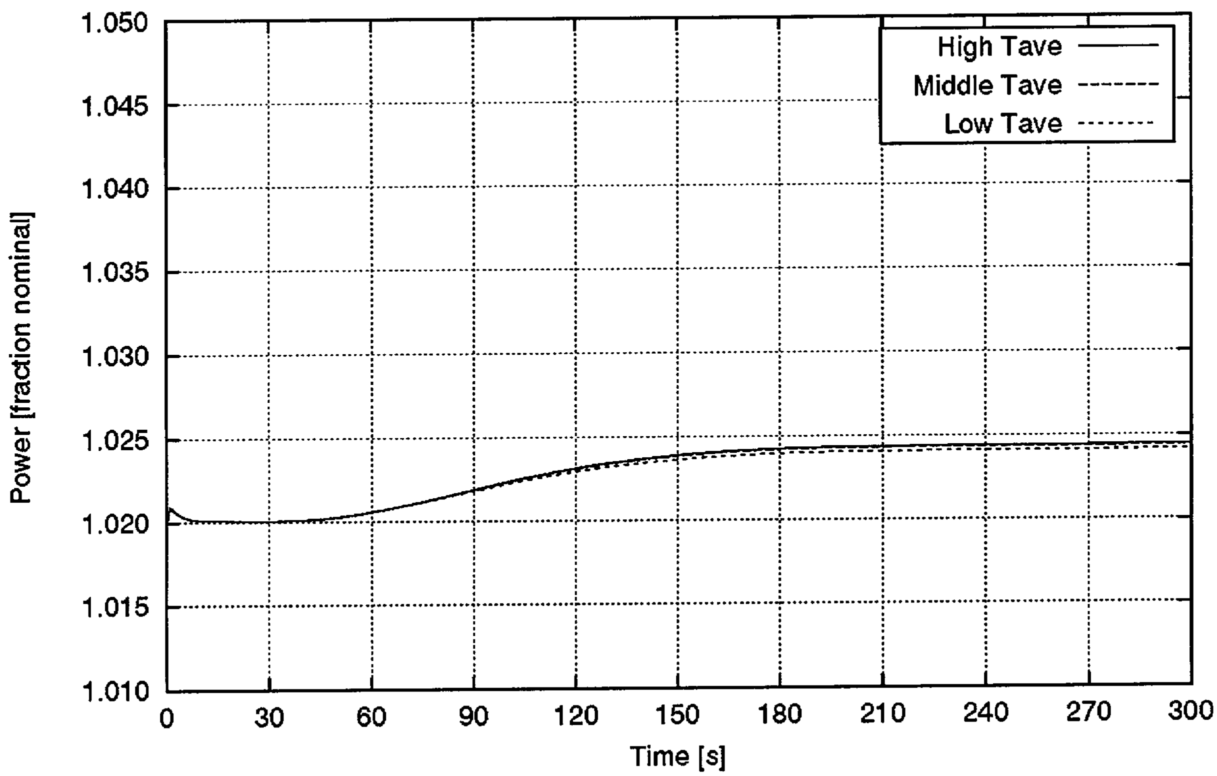


Figure 6.8.7-1
Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control
Reactor Power vs. Time

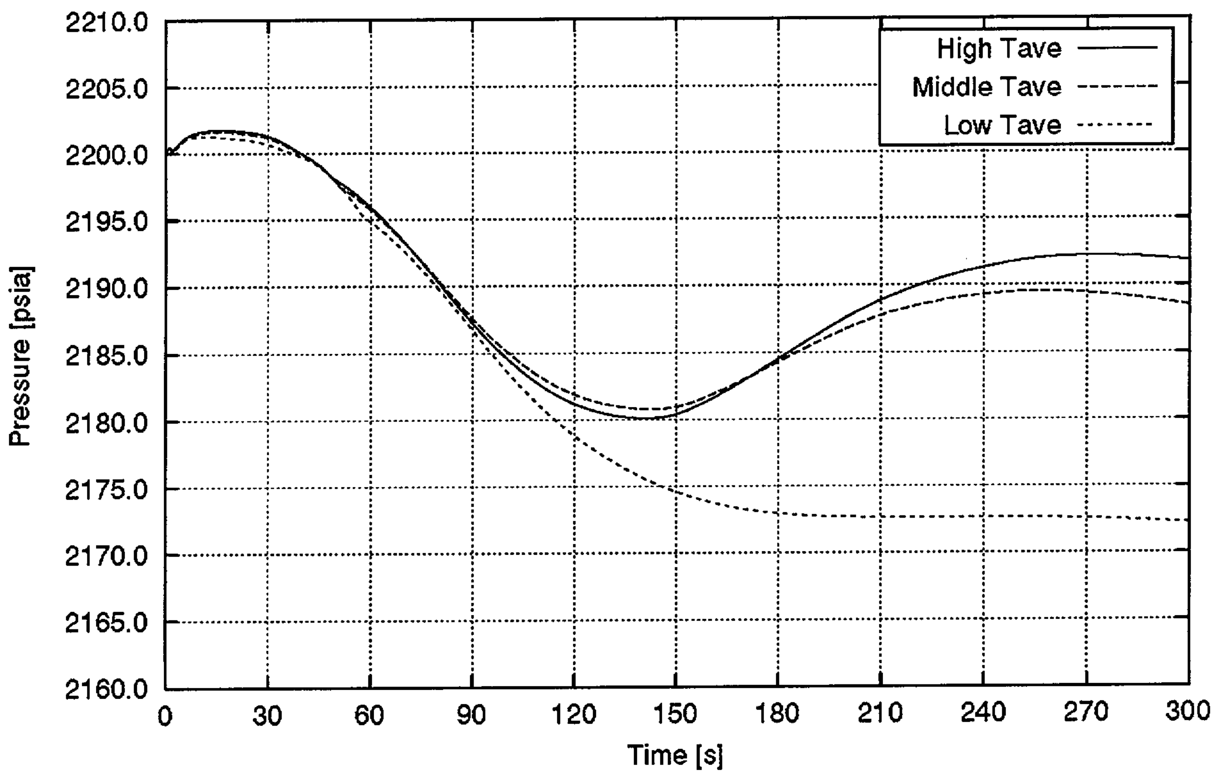


Figure 6.8.7-2
Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control
Pressurizer Pressure vs. Time

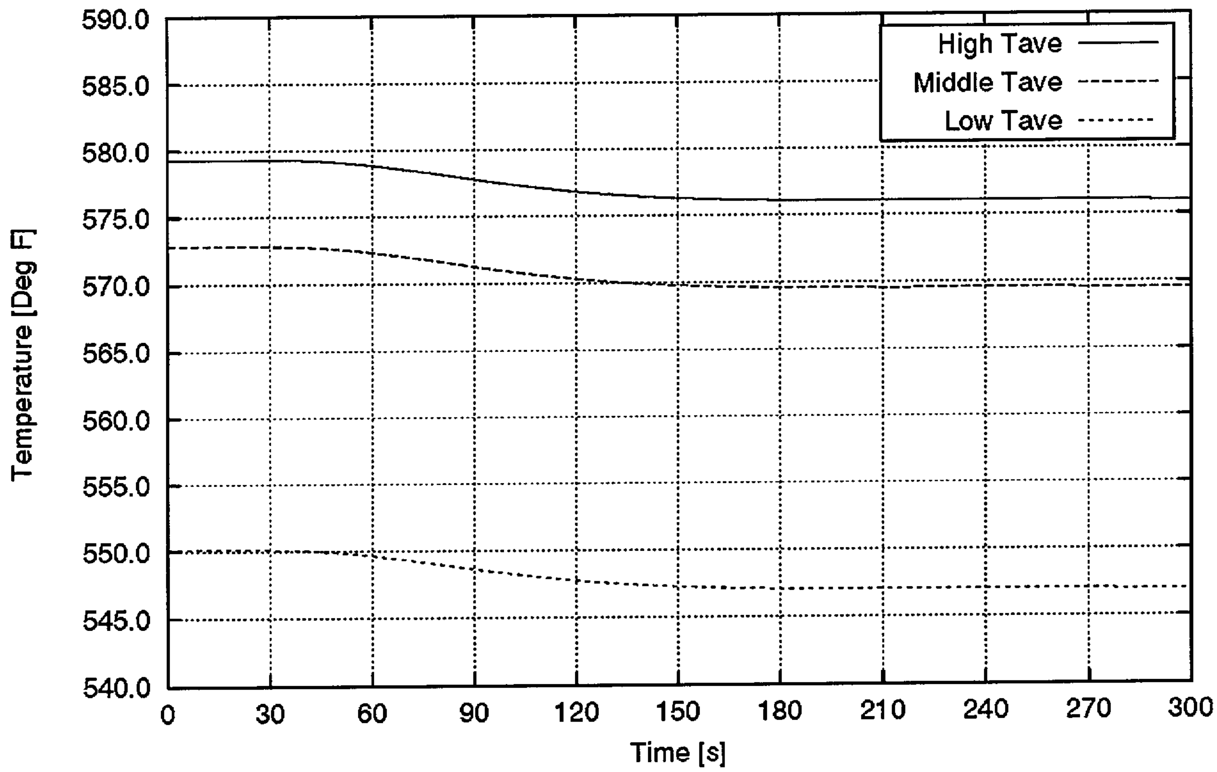


Figure 6.8.7-3
Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control
T_{avg} vs. Time

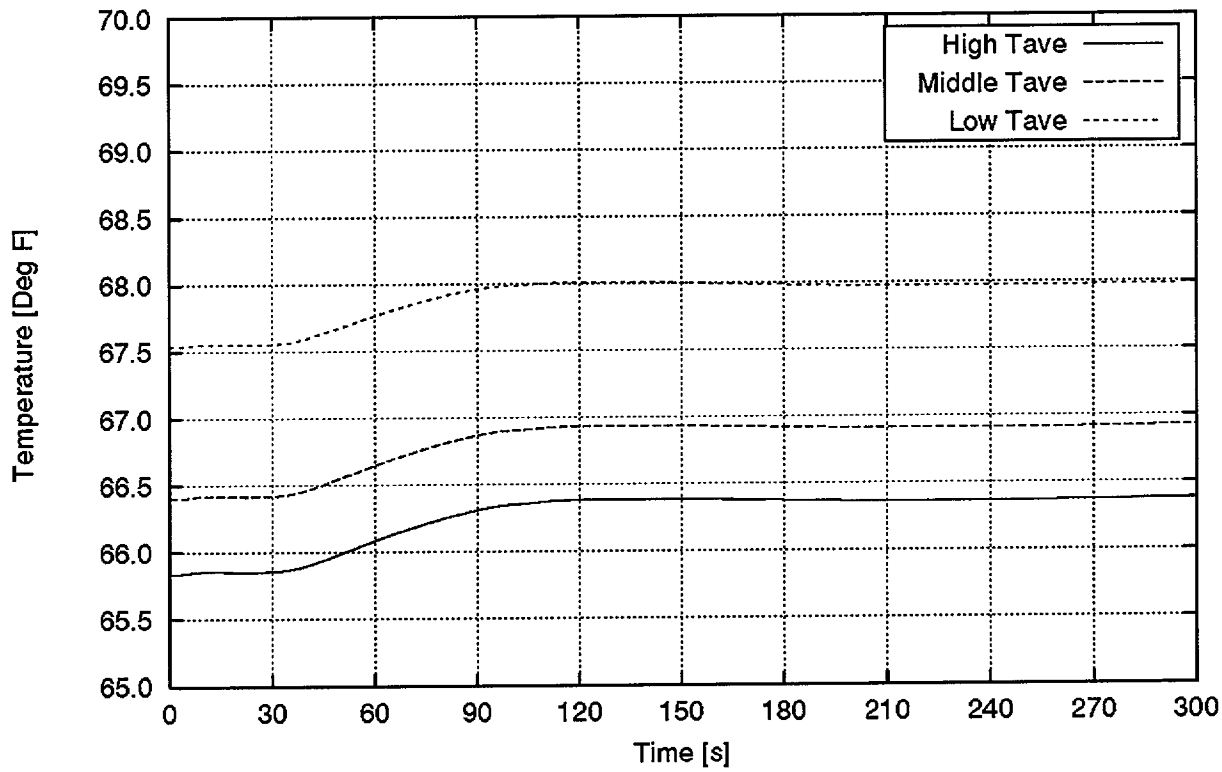


Figure 6.8.7-4
Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control
Delta-T Loop vs. Time

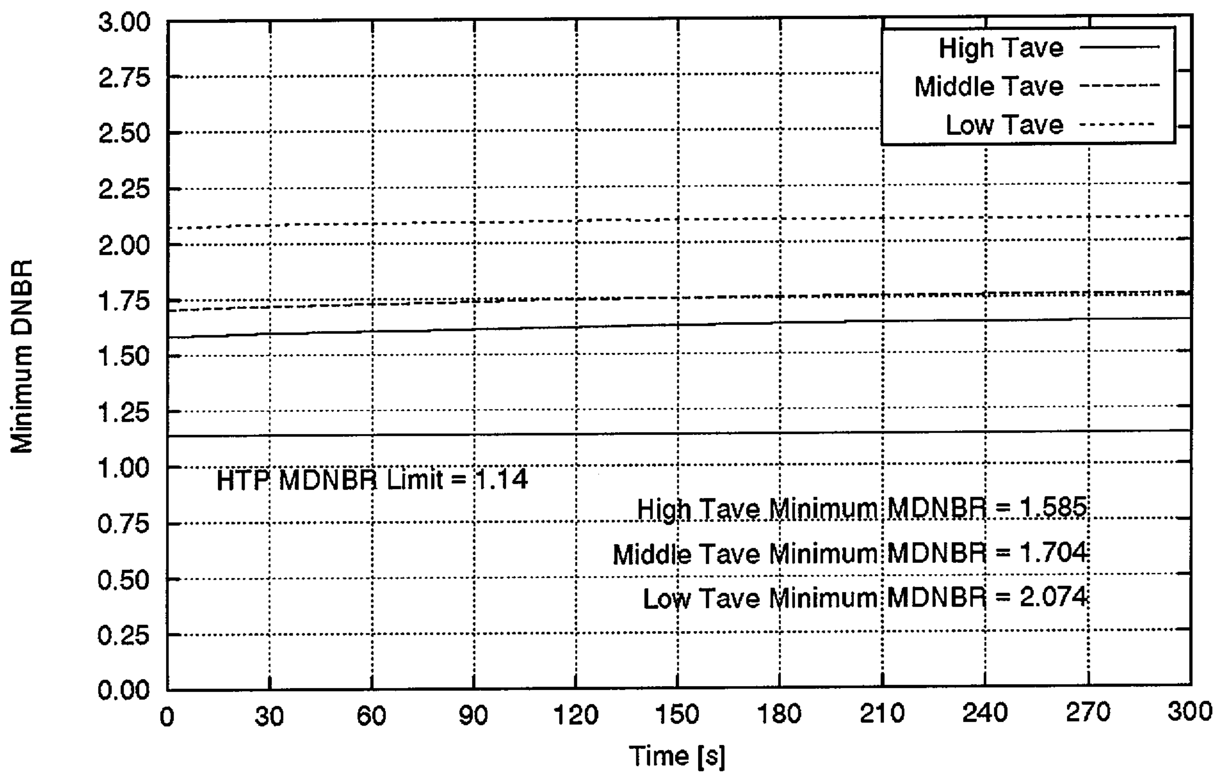


Figure 6.8.7-5
Excessive Heat Removal – Feedwater System Malfunction - BOC Manual Control
Minimum DNBR vs. Time

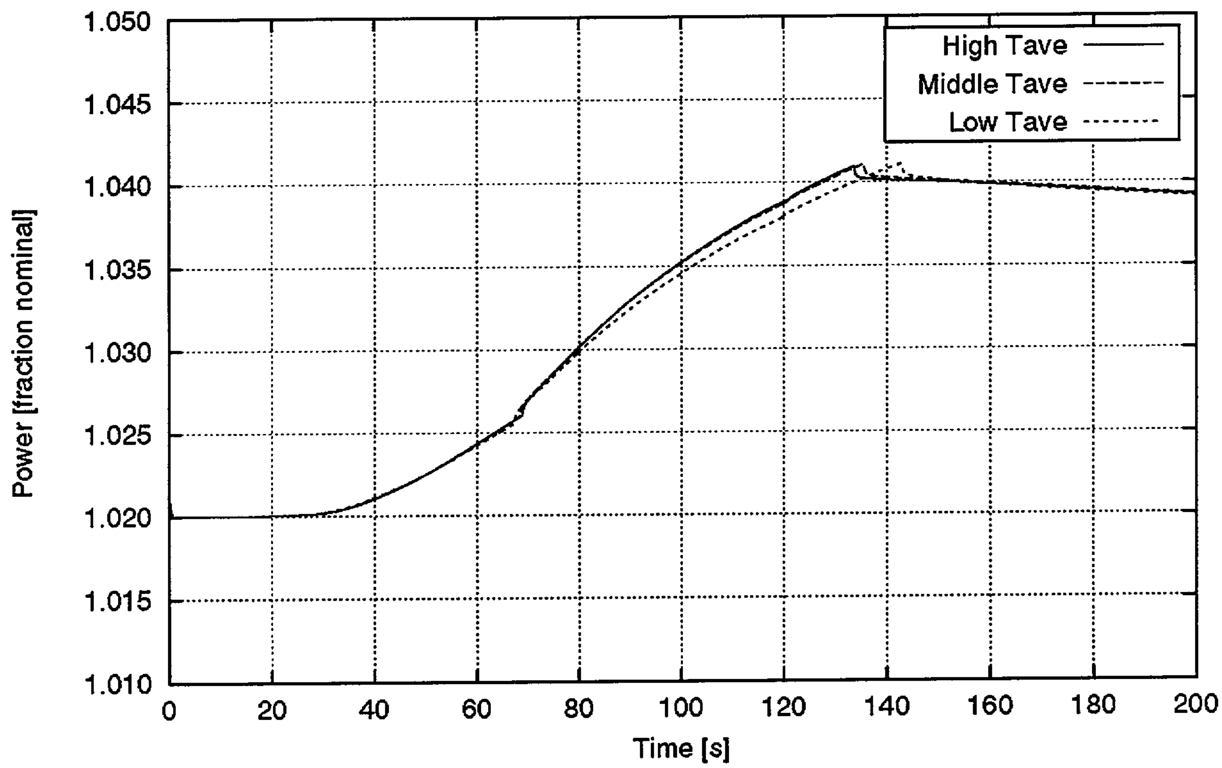


Figure 6.8.7-6
Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control
Reactor Power vs. Time

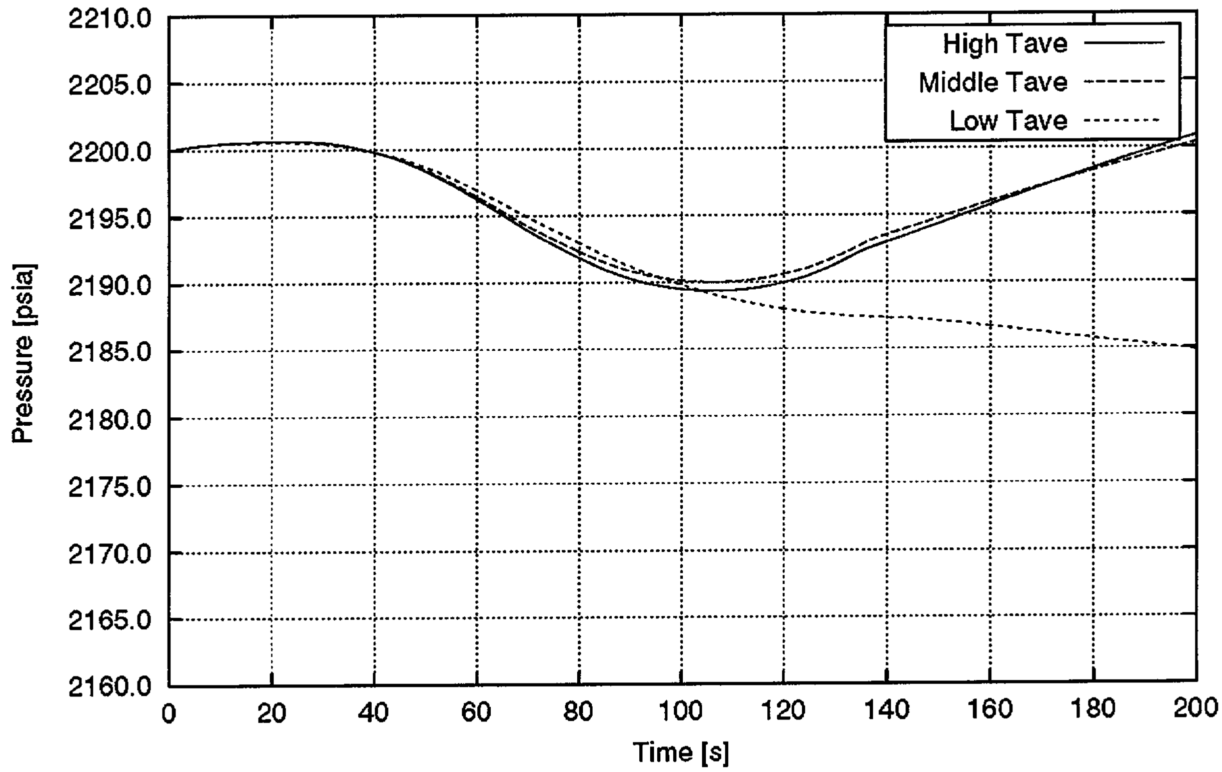


Figure 6.8.7-7
Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control
Pressurizer Pressure vs. Time

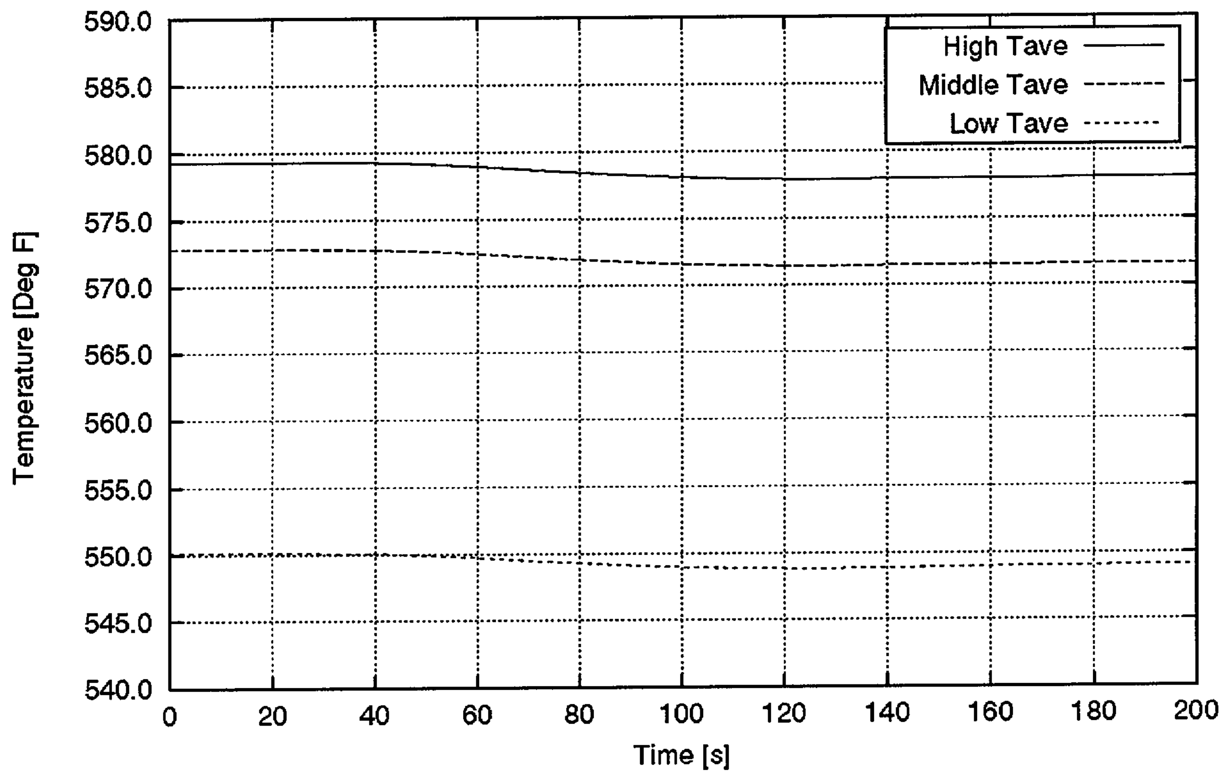


Figure 6.8.7-8
Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control
T_{avg} vs. Time

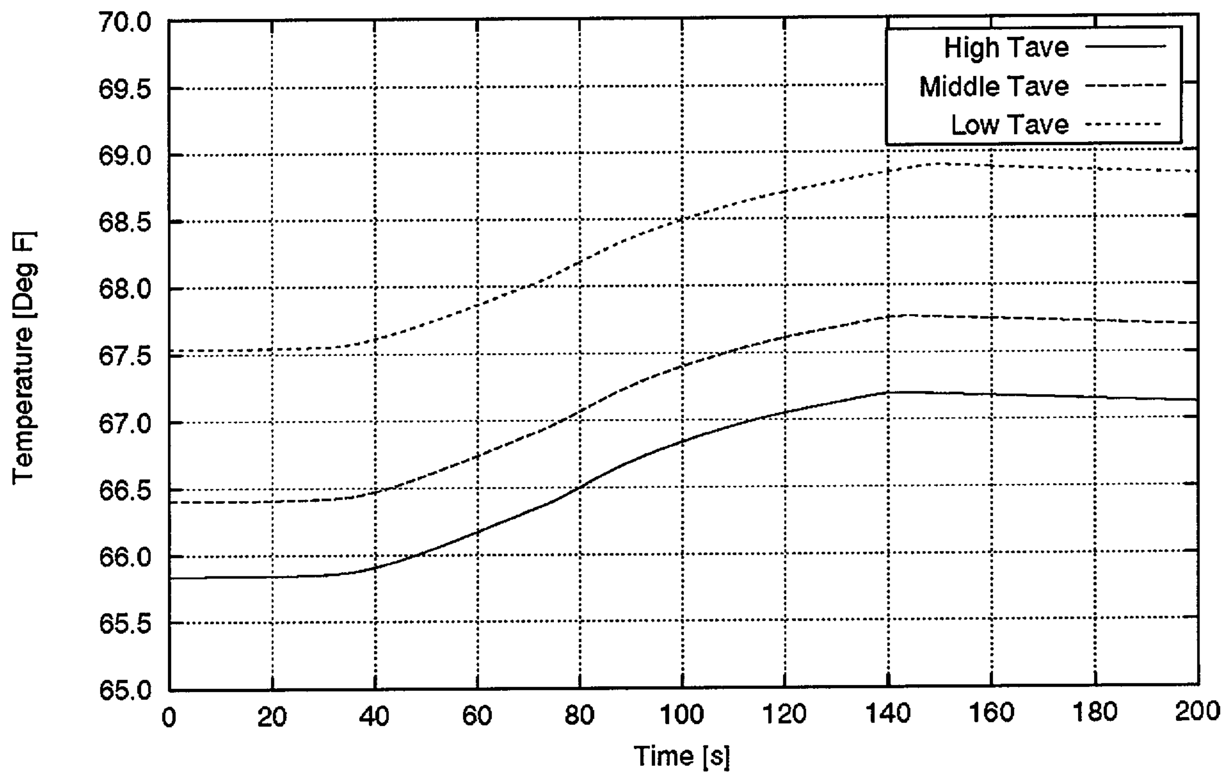


Figure 6.8.7-9
Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control
Delta-T Loop vs. Time

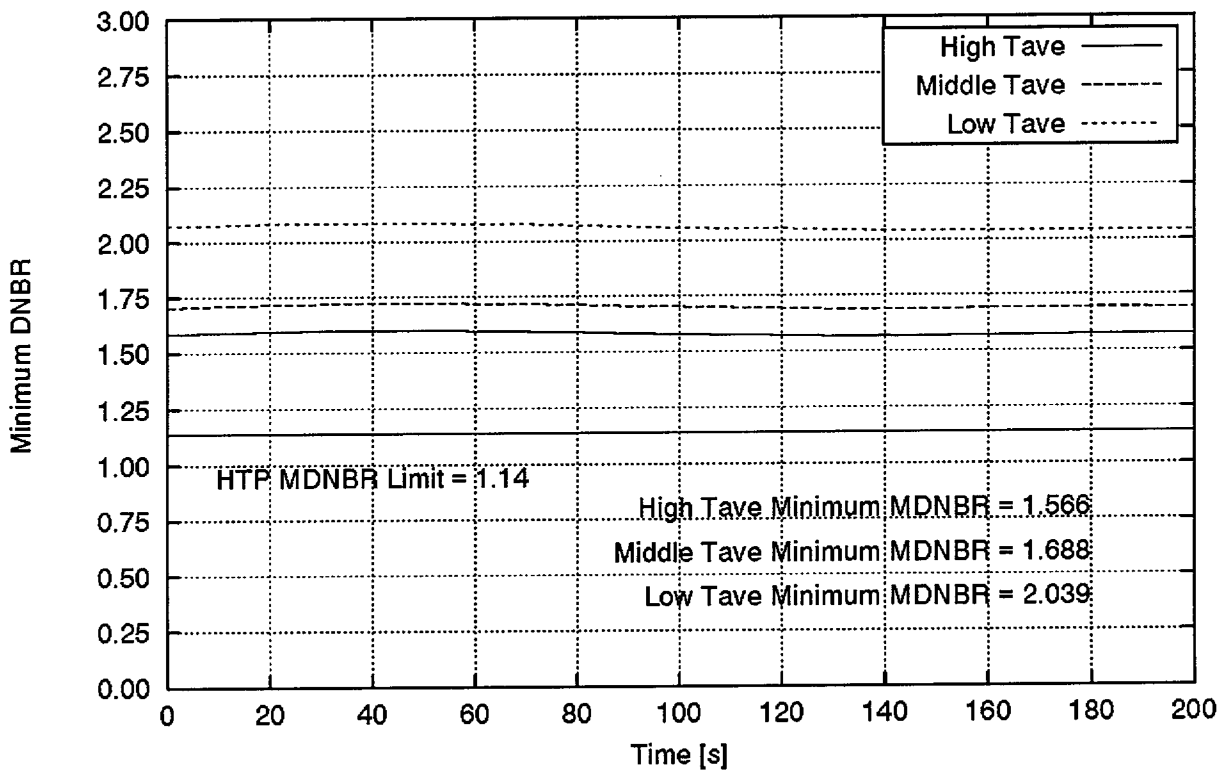


Figure 6.8.7-10
Excessive Heat Removal – Feedwater System Malfunction - EOC Auto Control
Minimum DNBR vs. Time

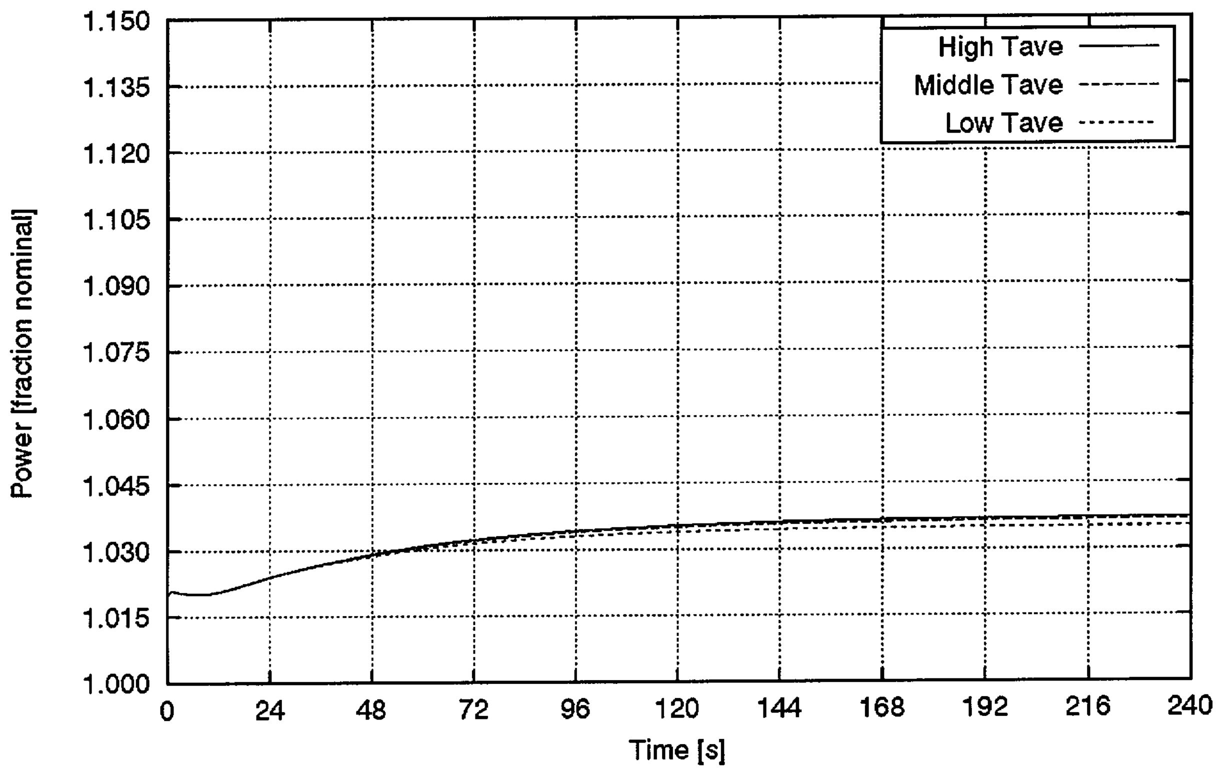


Figure 6.8.8-1
Excessive Load Increase – BOC Manual Control
Reactor Power vs. Time

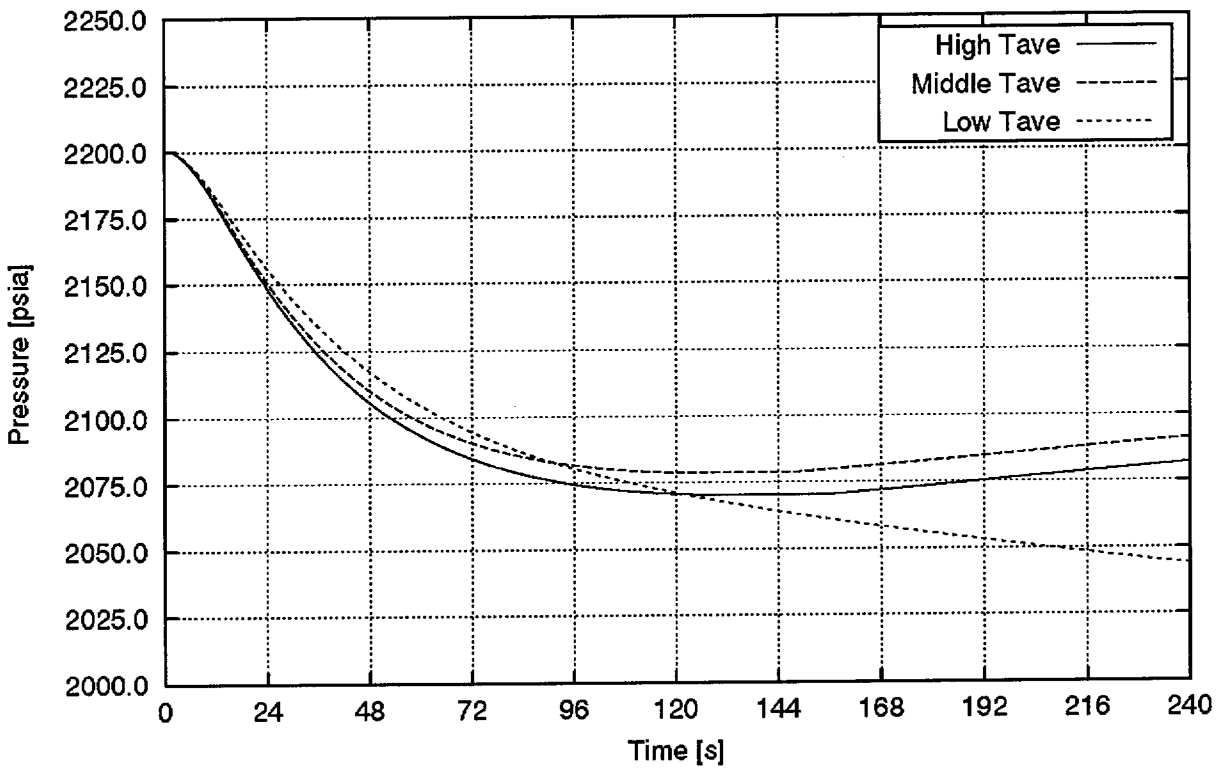


Figure 6.8.8-2
Excessive Load Increase – BOC Manual Control
Pressurizer Pressure vs. Time

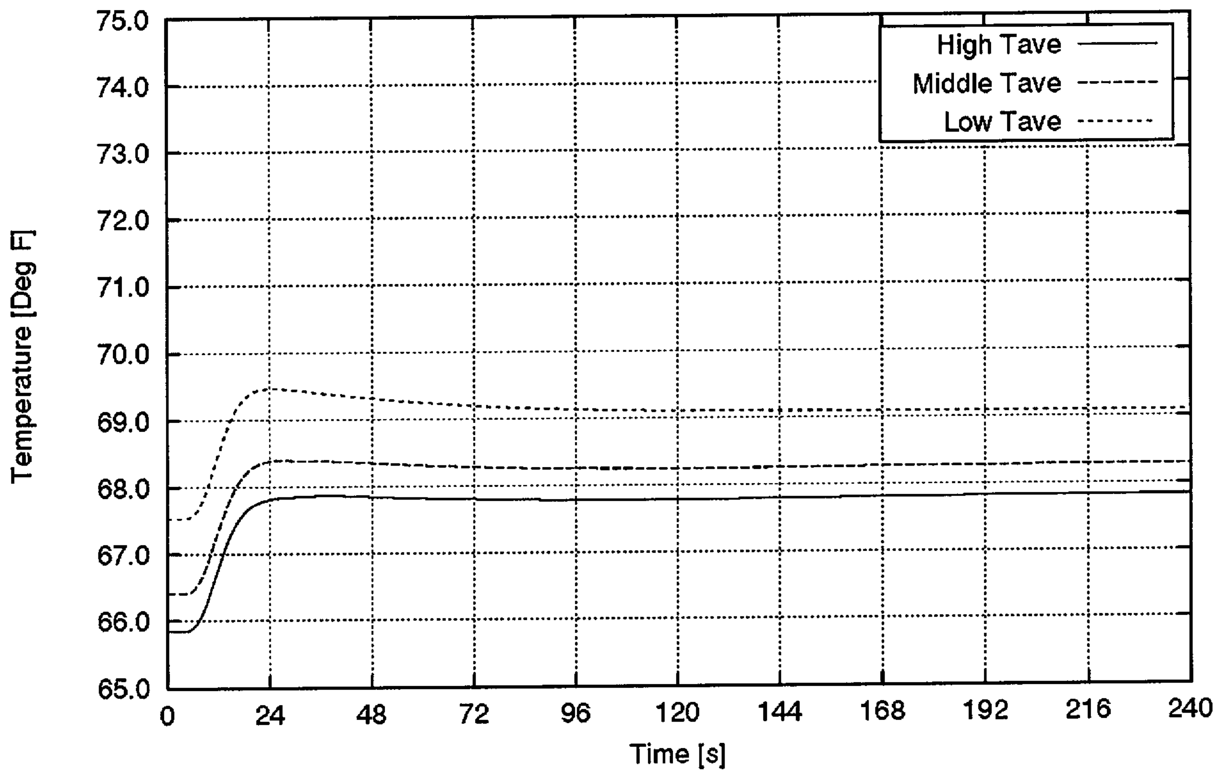


Figure 6.8.8-3
Excessive Load Increase – BOC Manual Control
Delta-T Loop vs. Time

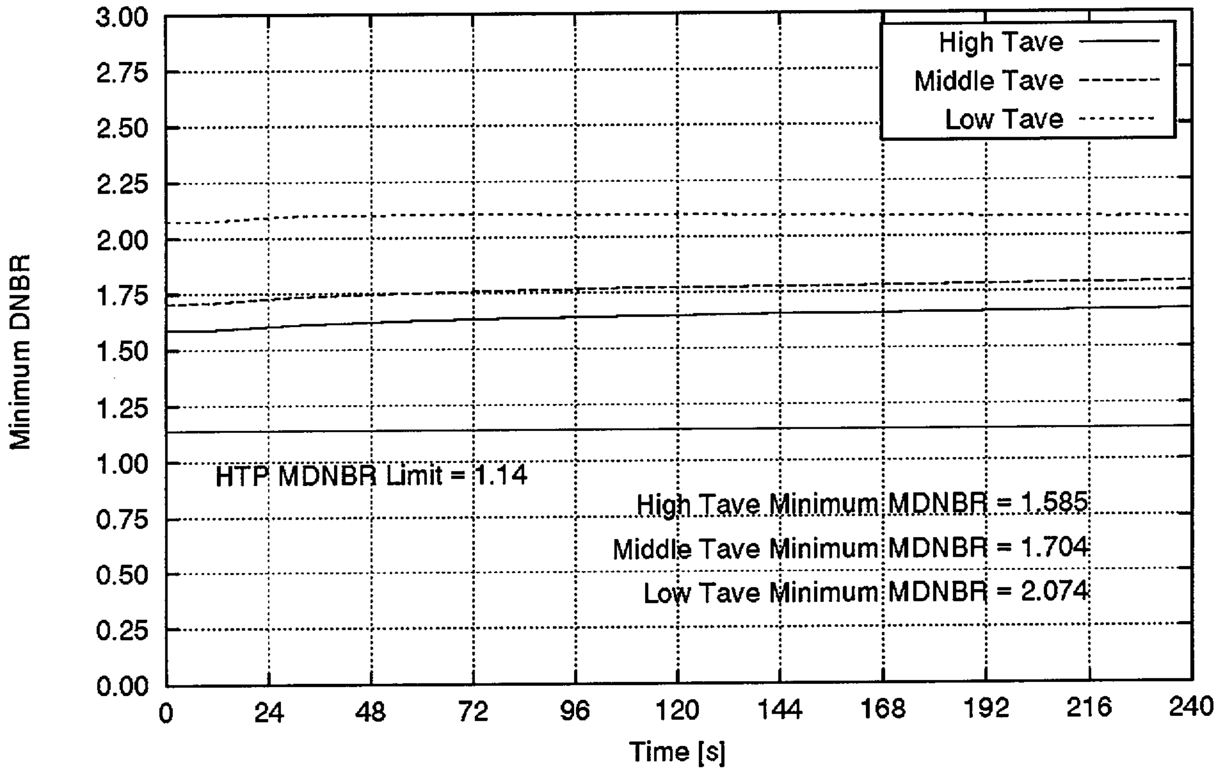


Figure 6.8.8-4
Excessive Load Increase – BOC Manual Control
Minimum DNBR vs. Time

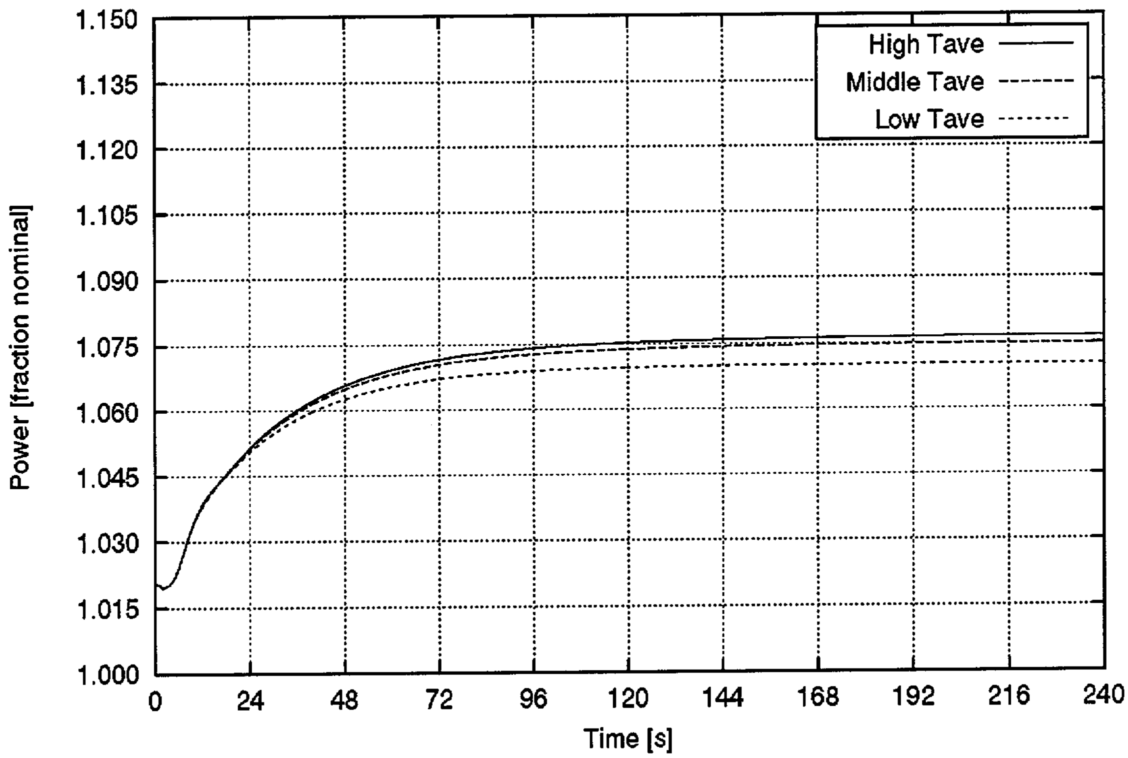


Figure 6.8.8-5
Excessive Load Increase – EOC Manual Control
Reactor Power vs. Time

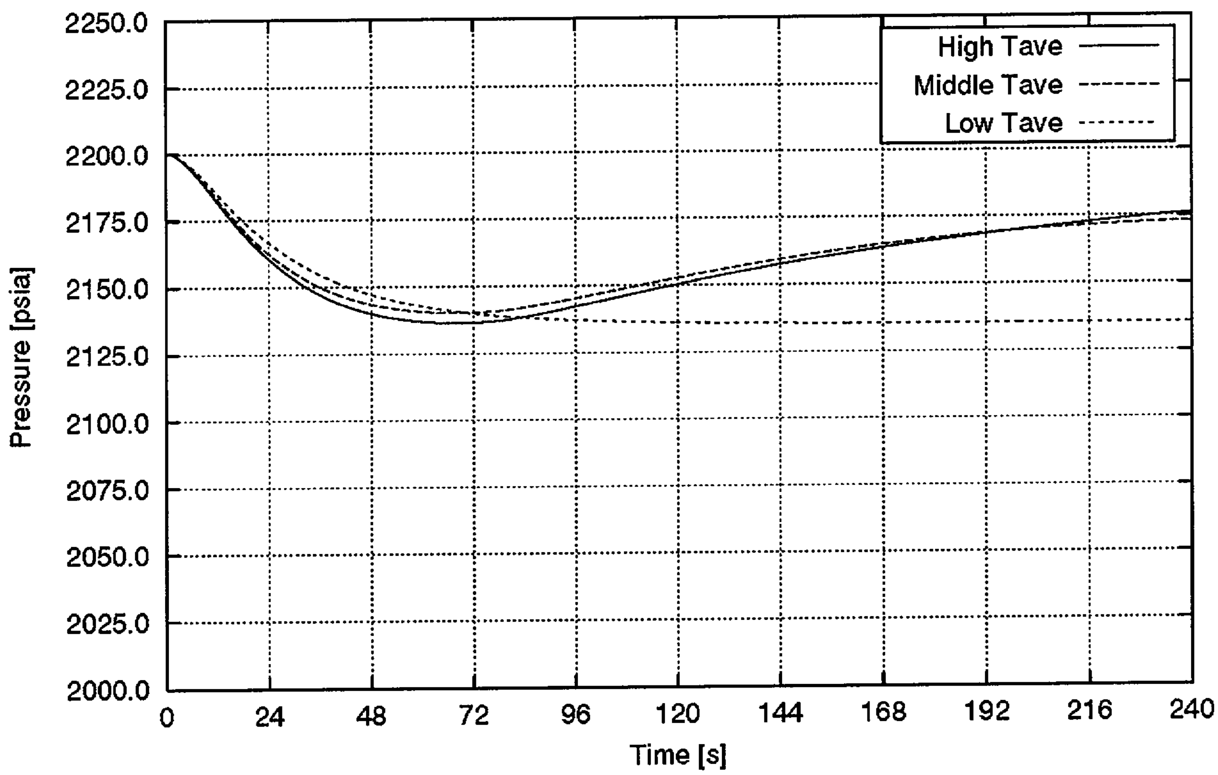


Figure 6.8.8-6
Excessive Load Increase – EOC Manual Control
Pressurizer Pressure vs. Time

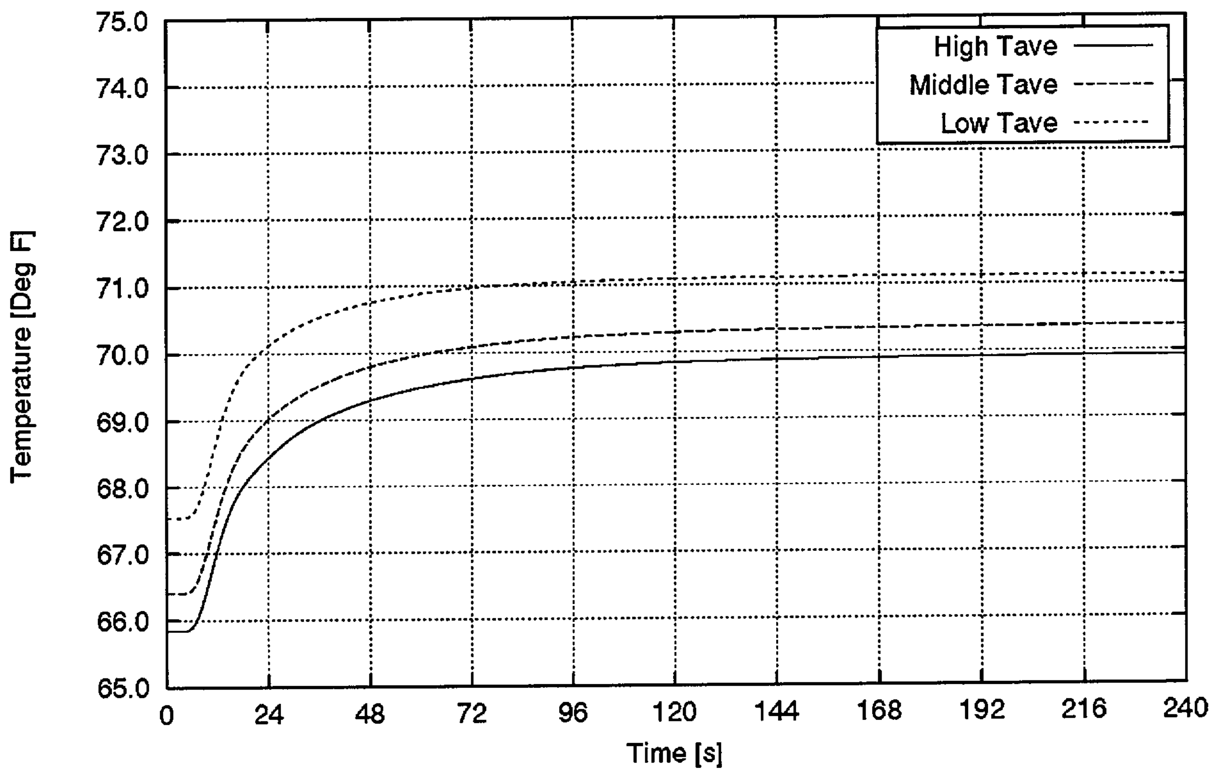


Figure 6.8.8-7
Excessive Load Increase – EOC Manual Control
Delta-T Loop vs. Time

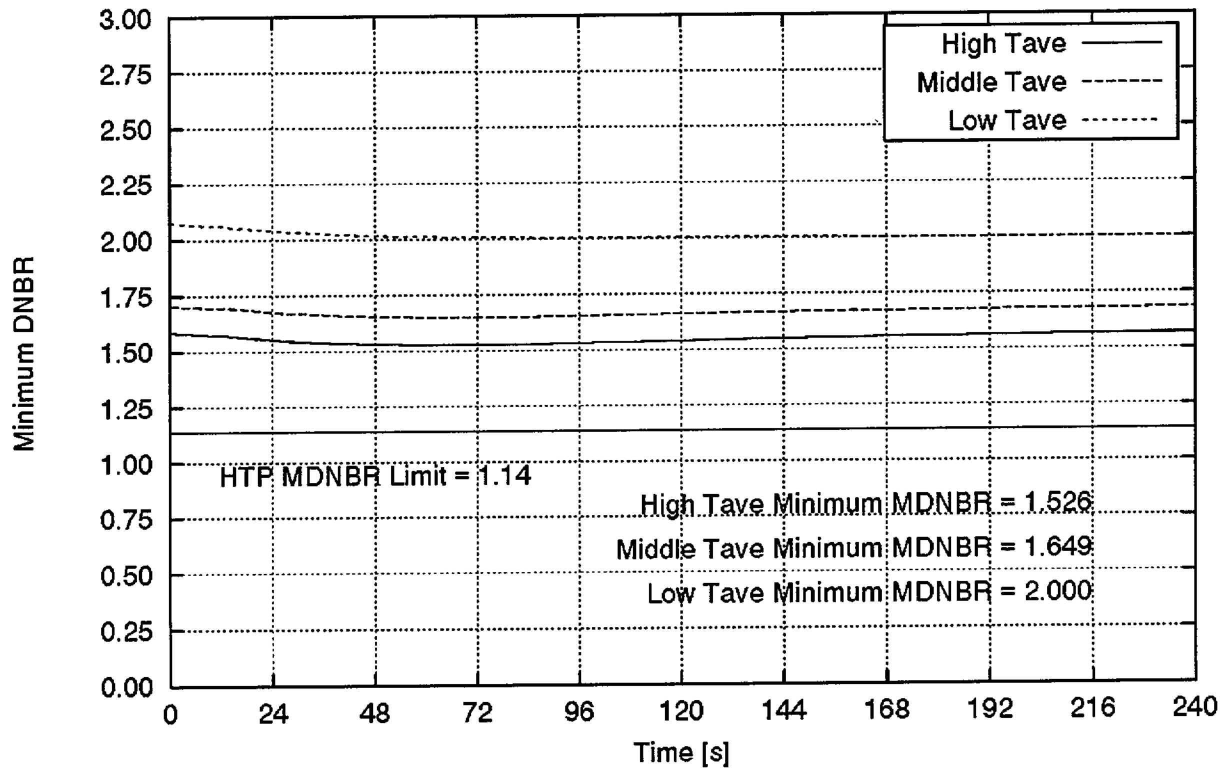


Figure 6.8.8-8
Excessive Load Increase – EOC Manual Control
Minimum DNBR vs. Time

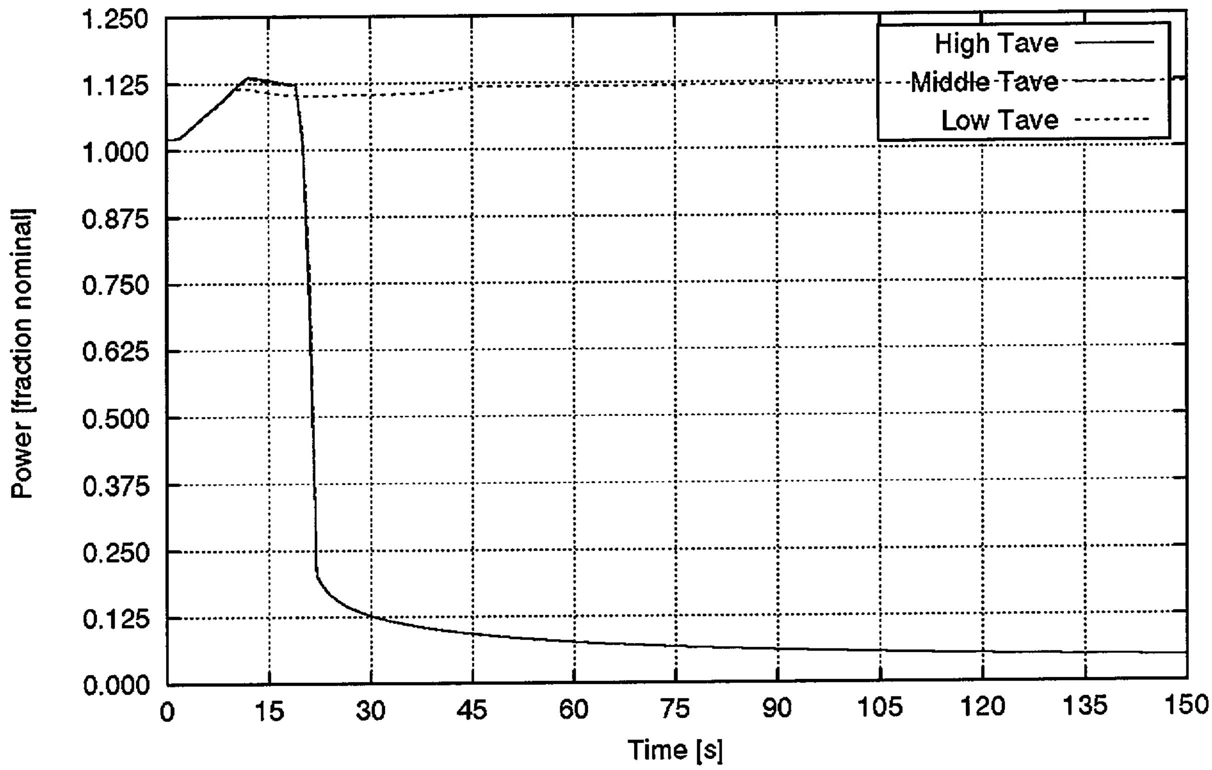


Figure 6.8.8-9
Excessive Load Increase – BOC Auto Control
Reactor Power vs. Time

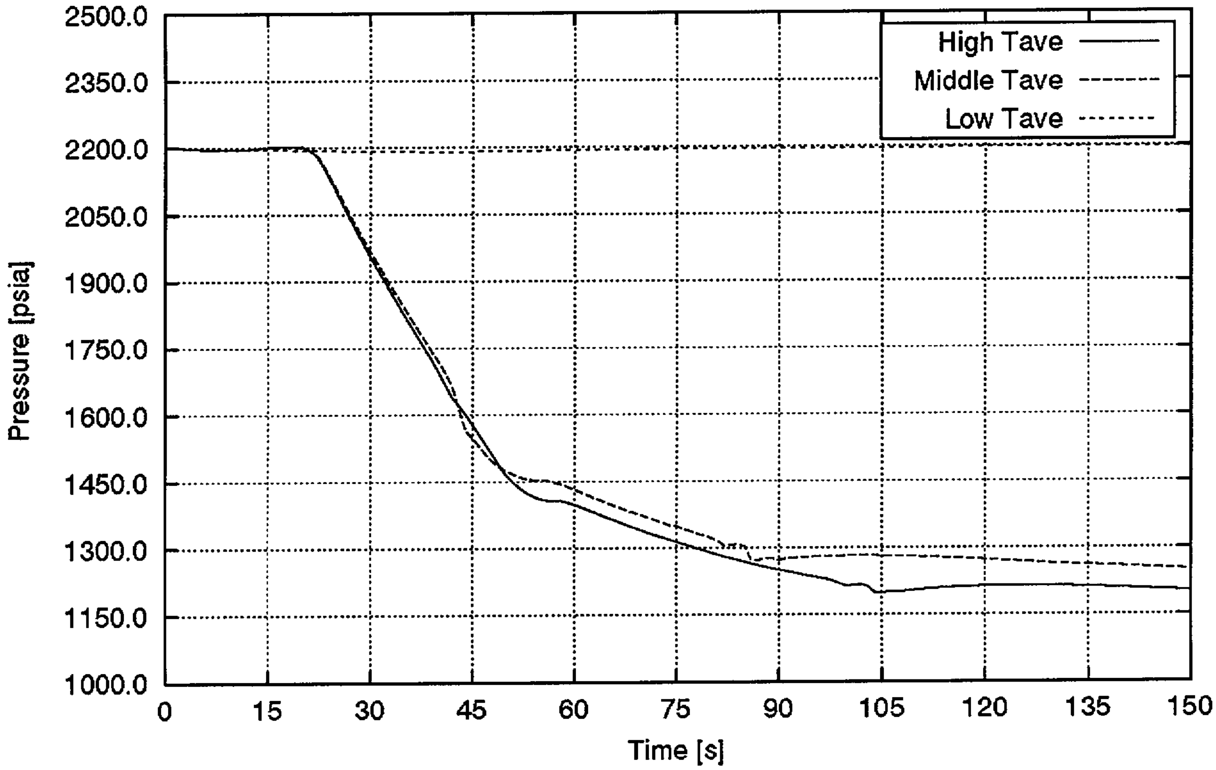


Figure 6.8.8-10
Excessive Load Increase – BOC Auto Control
Pressurizer Pressure vs. Time

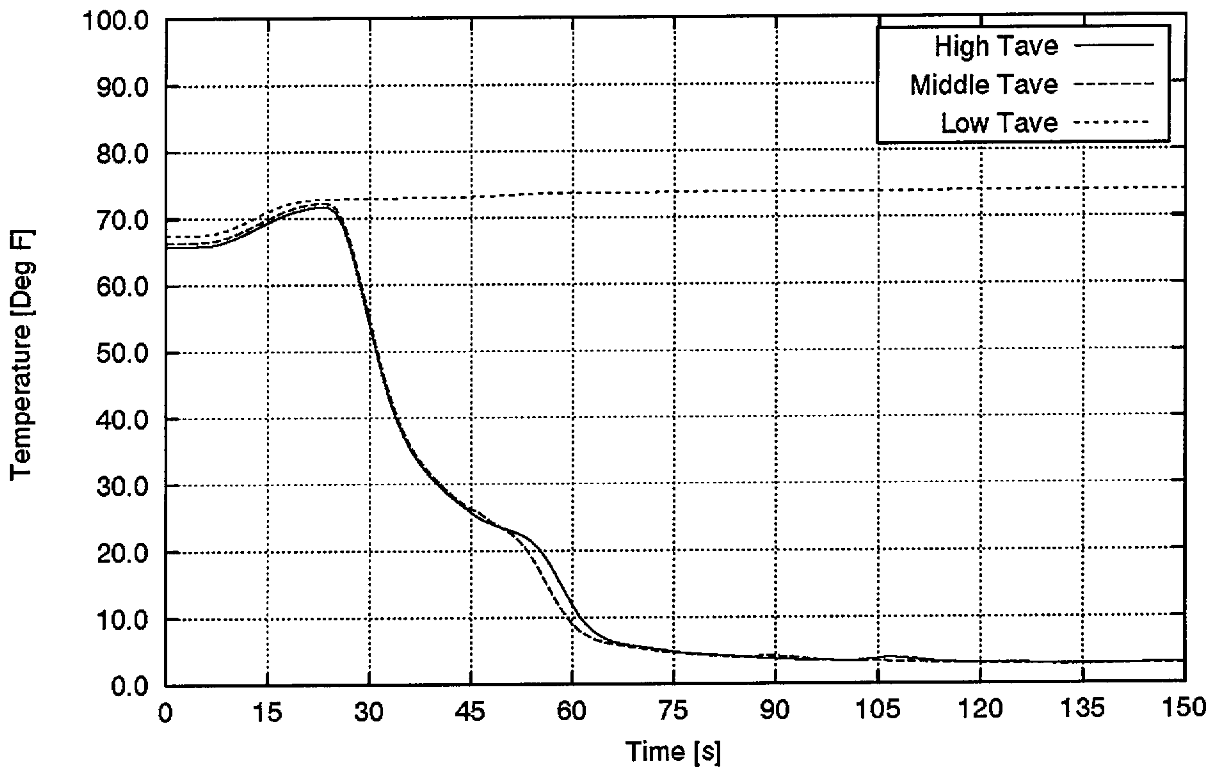


Figure 6.8.8-11
Excessive Load Increase – BOC Auto Control
Delta-T Loop vs. Time

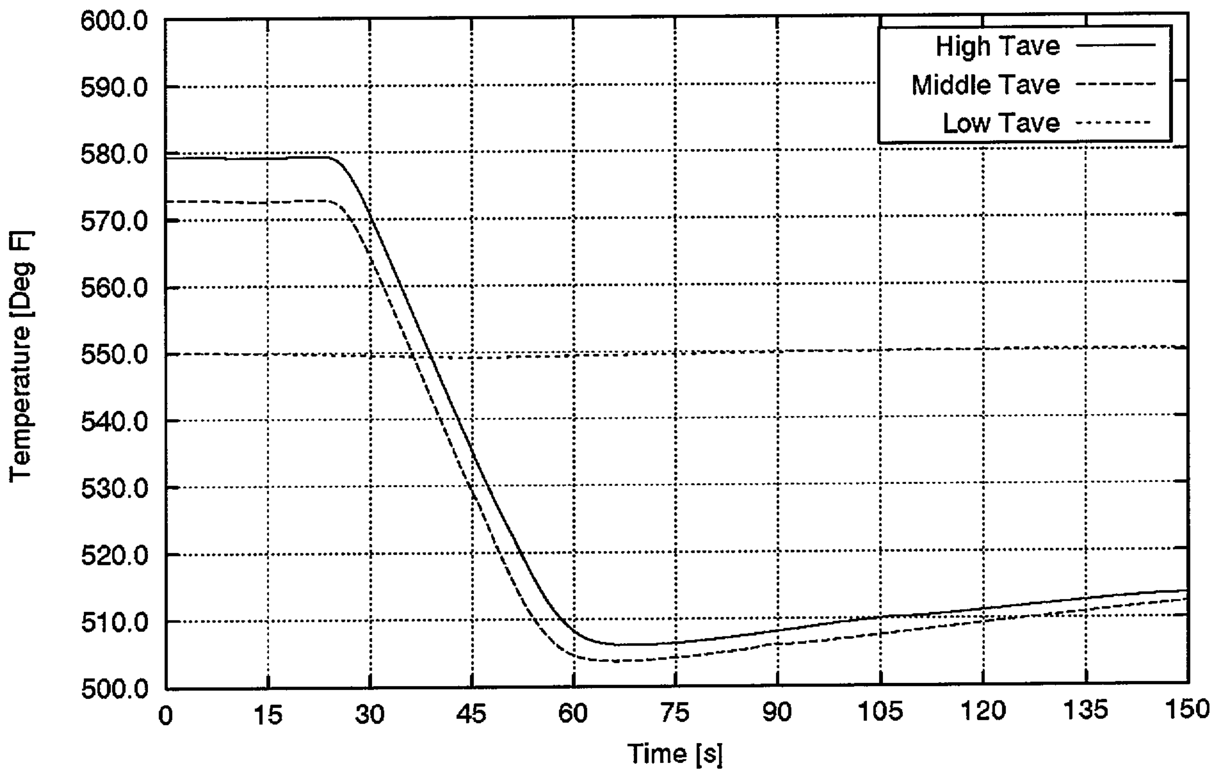


Figure 6.8.8-12
Excessive Load Increase – BOC Auto Control
T_{avg} vs. Time

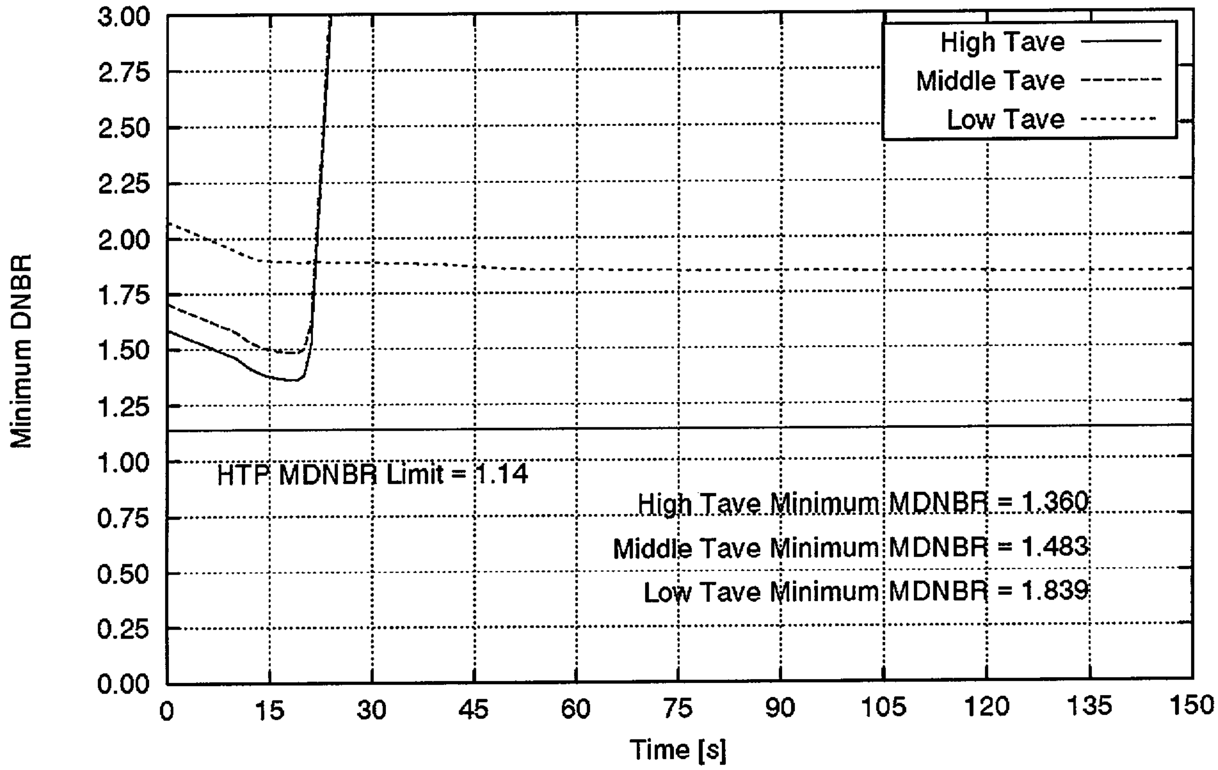


Figure 6.8.8-13
Excessive Load Increase – BOC Auto Control
Minimum DNBR vs. Time

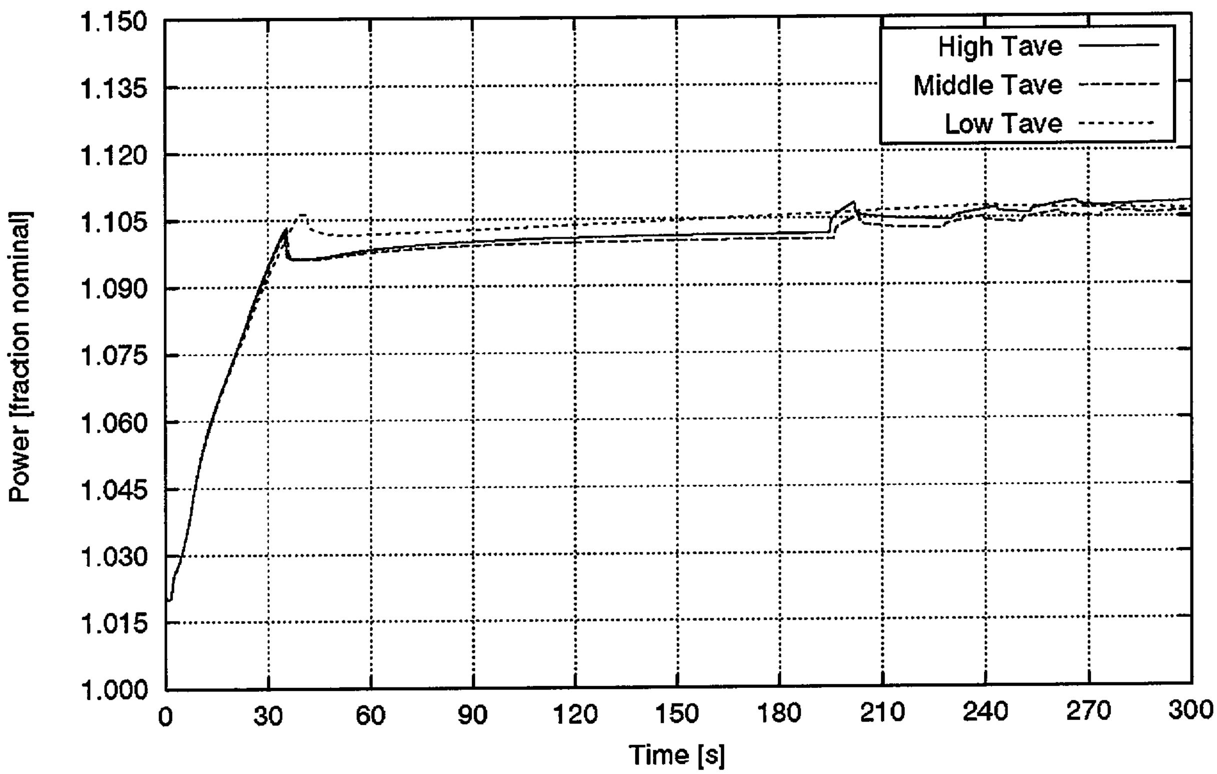


Figure 6.8.8-14
Excessive Load Increase – EOC Auto Control
Reactor Power vs. Time

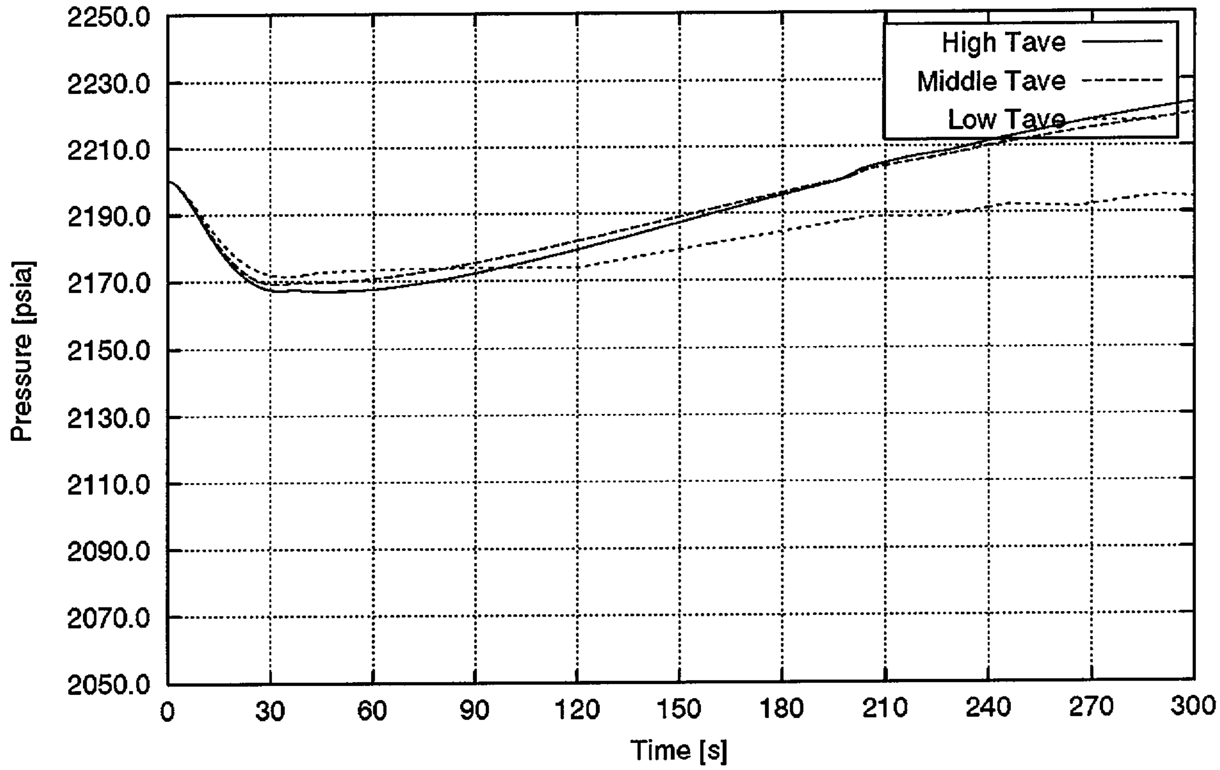


Figure 6.8.8-15
Excessive Load Increase – EOC Auto Control
Pressurizer Pressure vs. Time

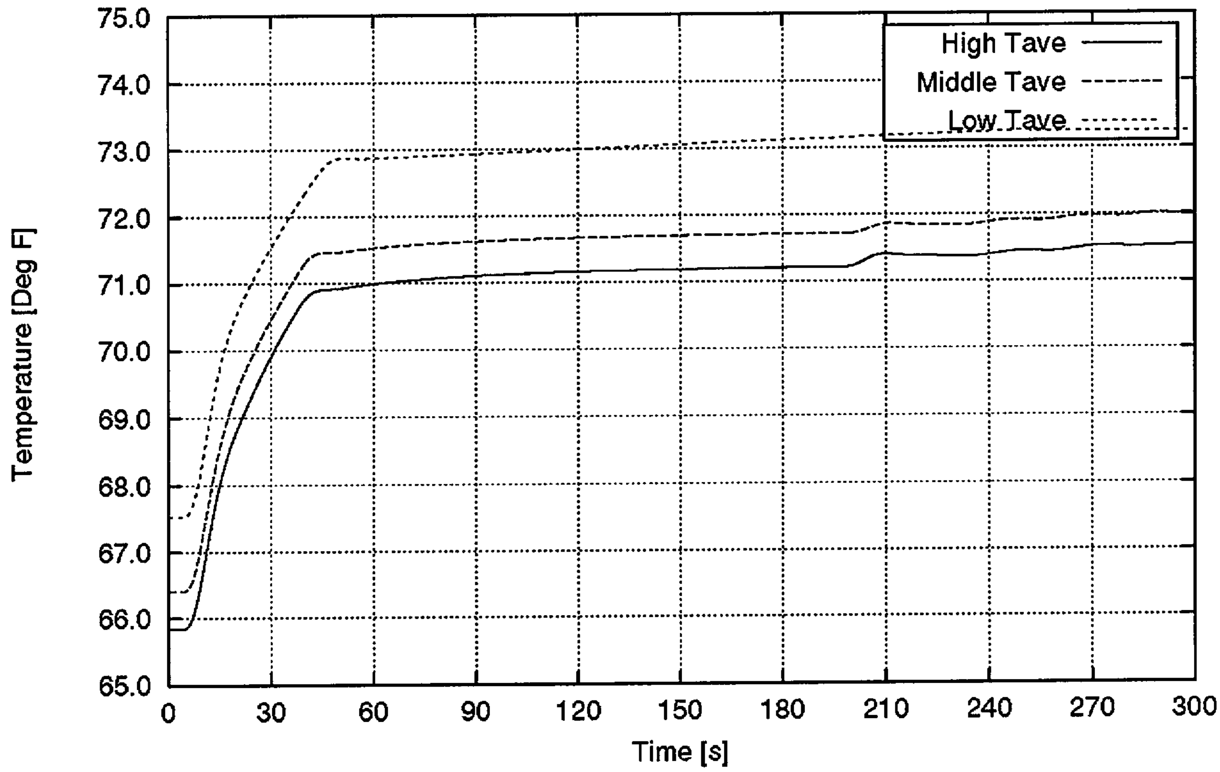


Figure 6.8.8-16
Excessive Load Increase – EOC Auto Control
Delta-T Loop vs. Time

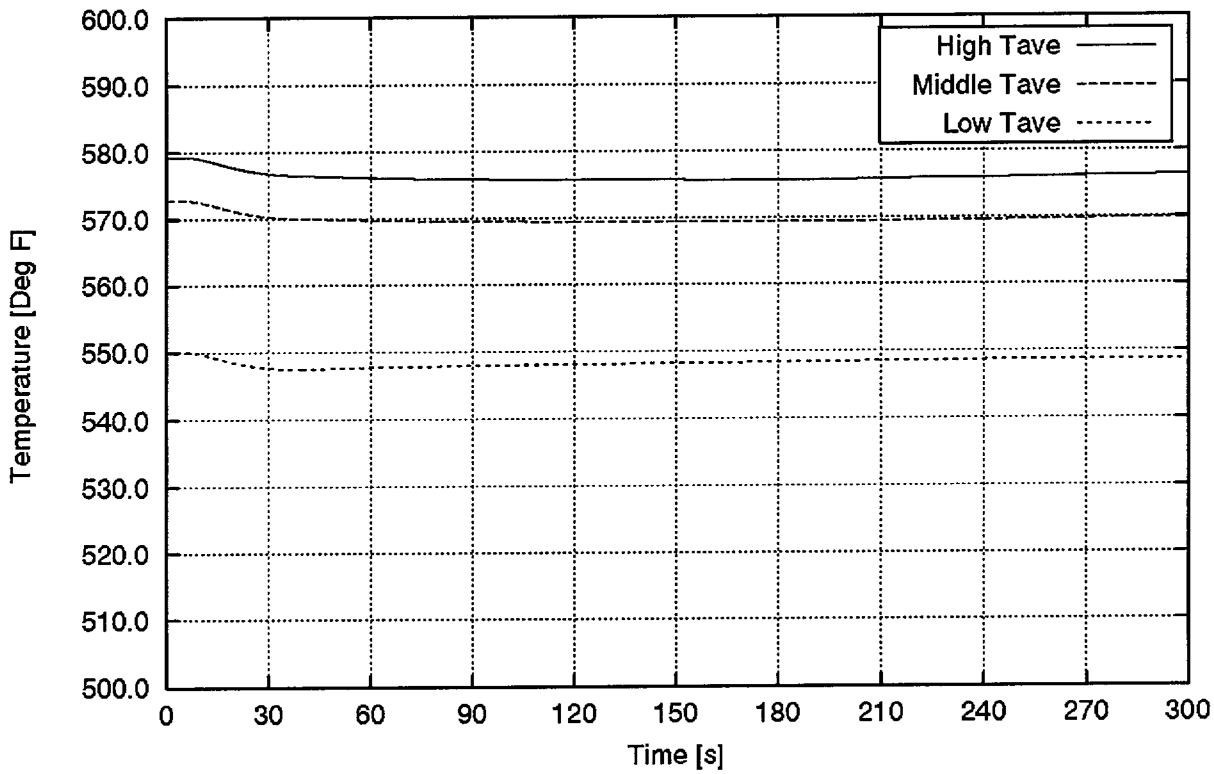


Figure 6.8.8-17
Excessive Load Increase – EOC Auto Control
T_{avg} vs. Time

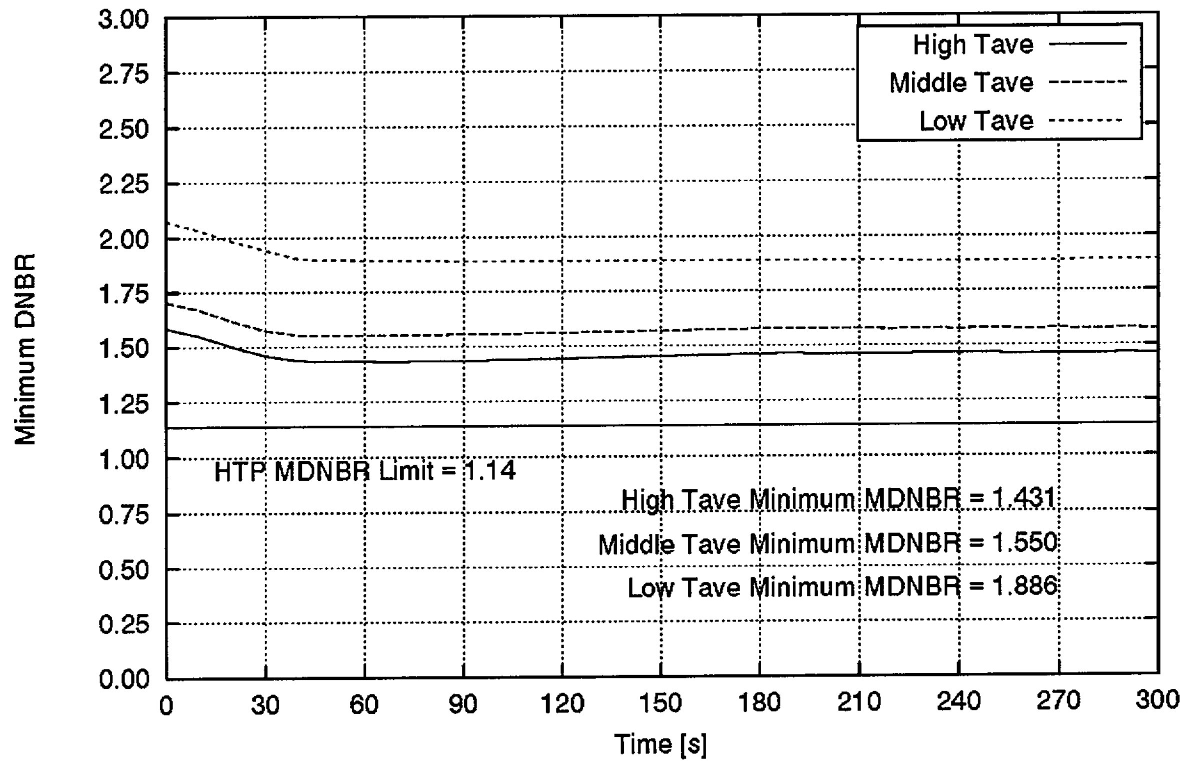


Figure 6.8.8-18
Excessive Load Increase – EOC Auto Control
Minimum DNBR vs. Time

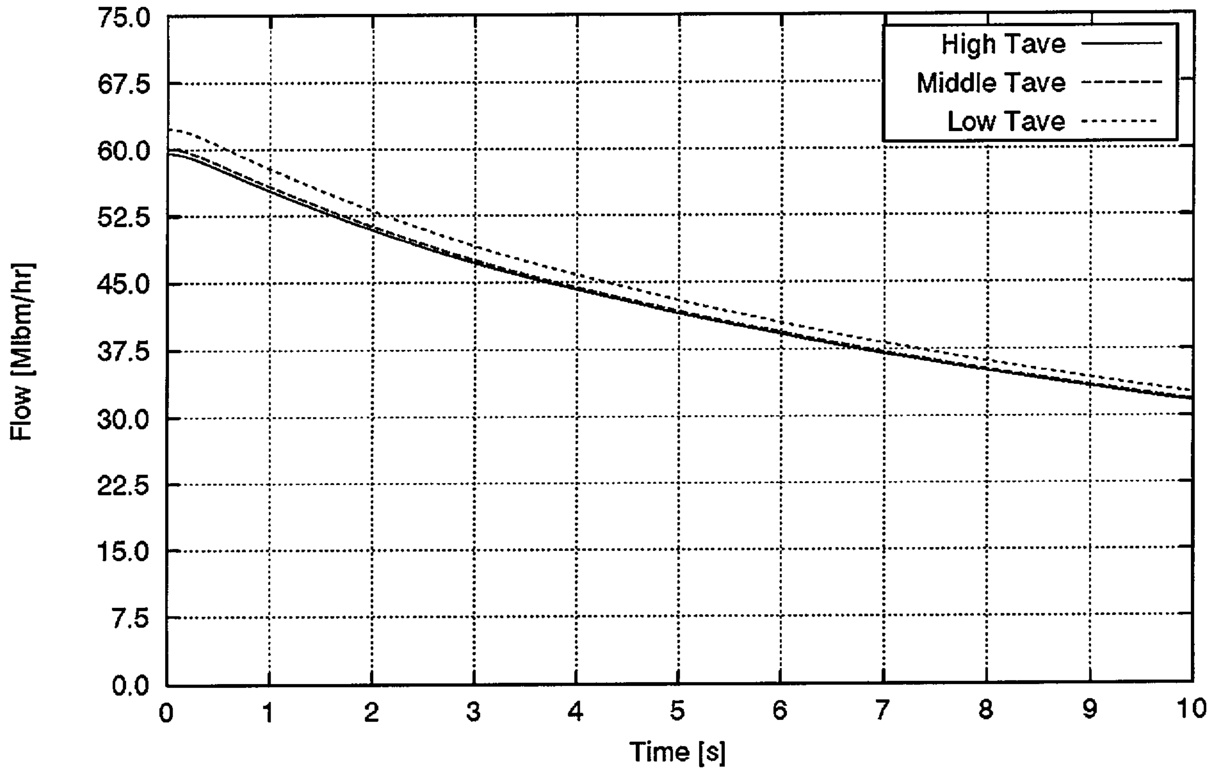


Figure 6.8.9-1
Loss of Reactor Coolant Flow - Two Pump Trip
Core Flow vs. Time

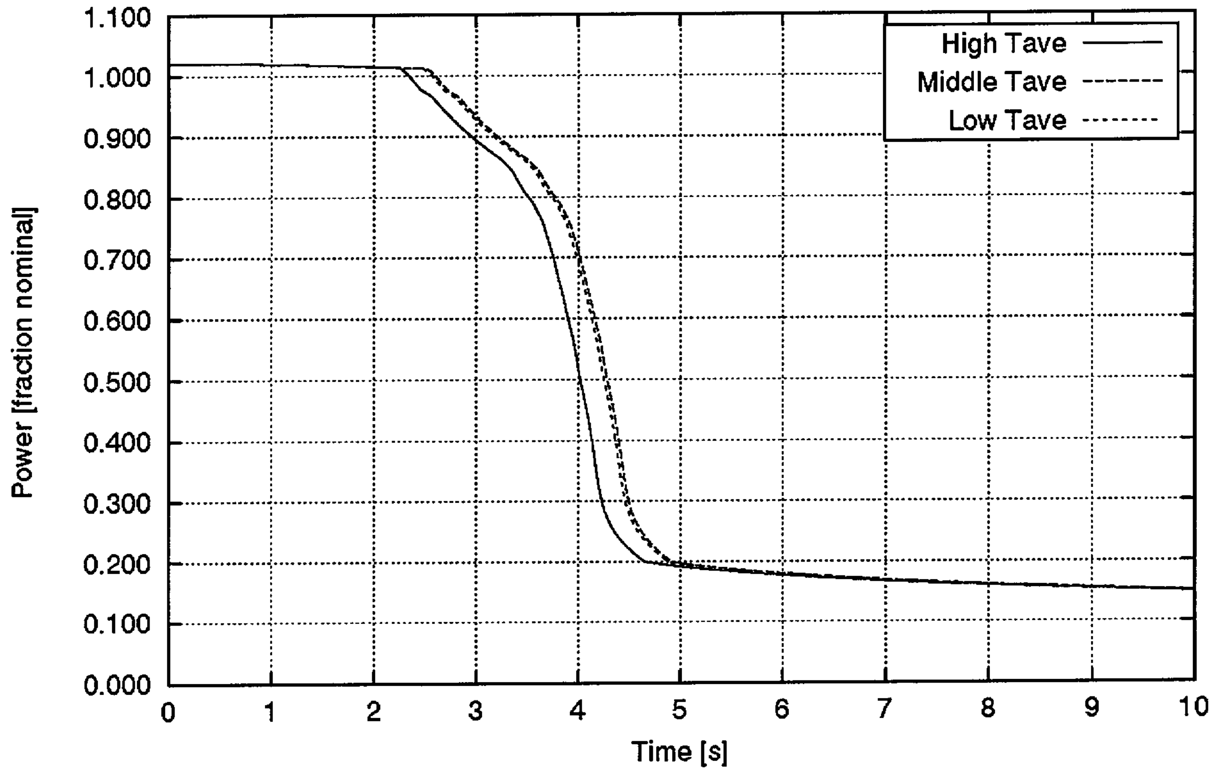


Figure 6.8.9-2
Loss of Reactor Coolant Flow - Two Pump Trip
Reactor Power vs. Time

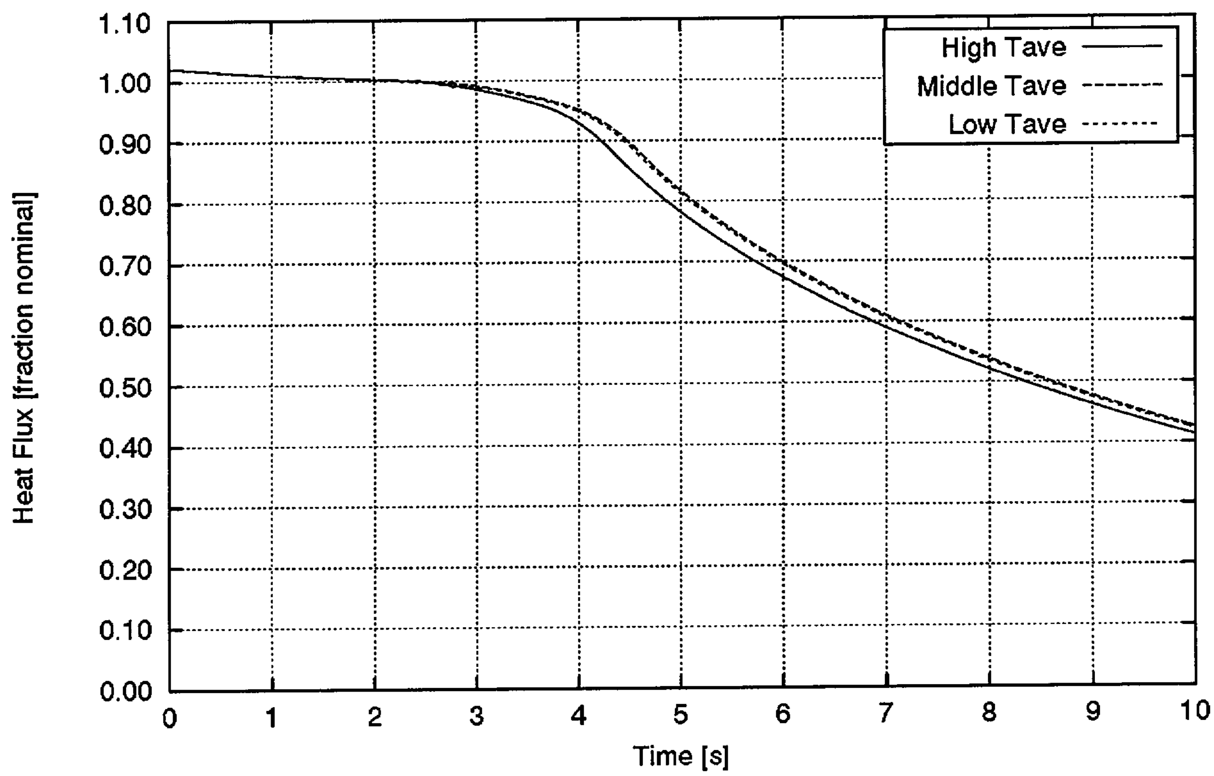


Figure 6.8.9-3
Loss of Reactor Coolant Flow - Two Pump Trip
Heat Flux vs. Time

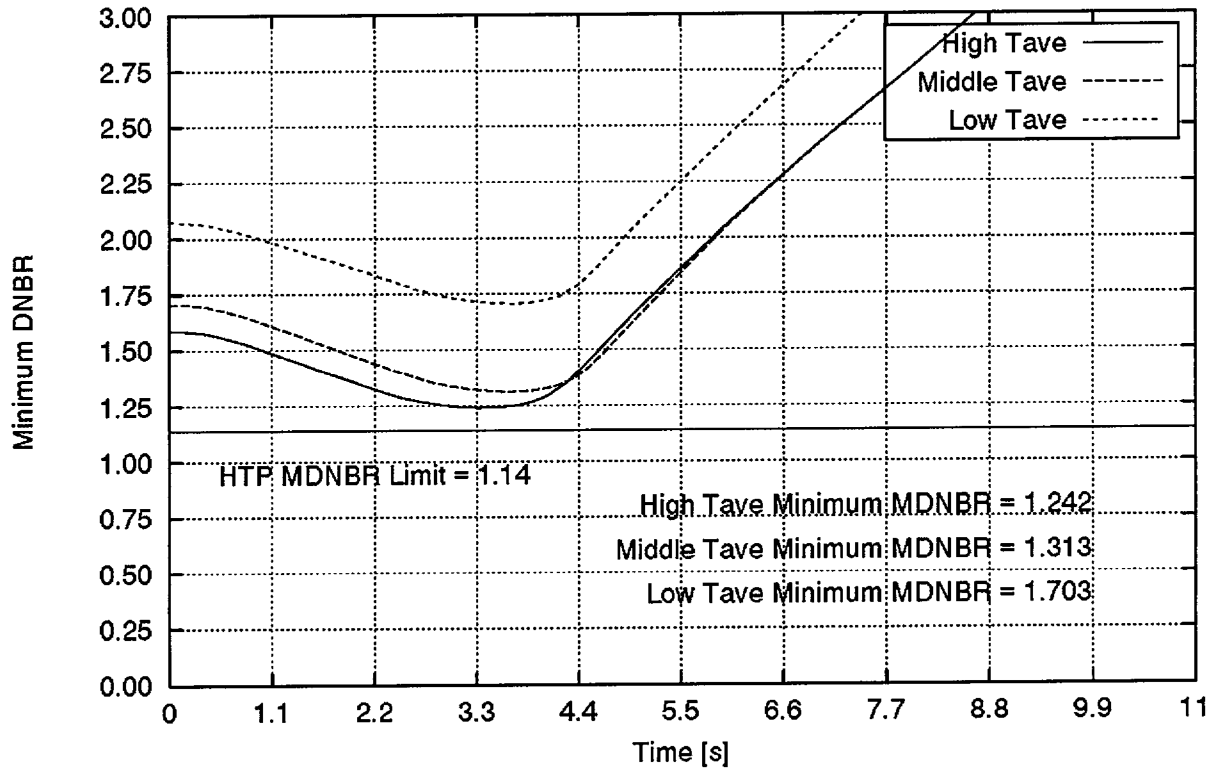


Figure 6.8.9-4
Loss of Reactor Coolant Flow - Two Pump Trip
Minimum DNBR vs. Time

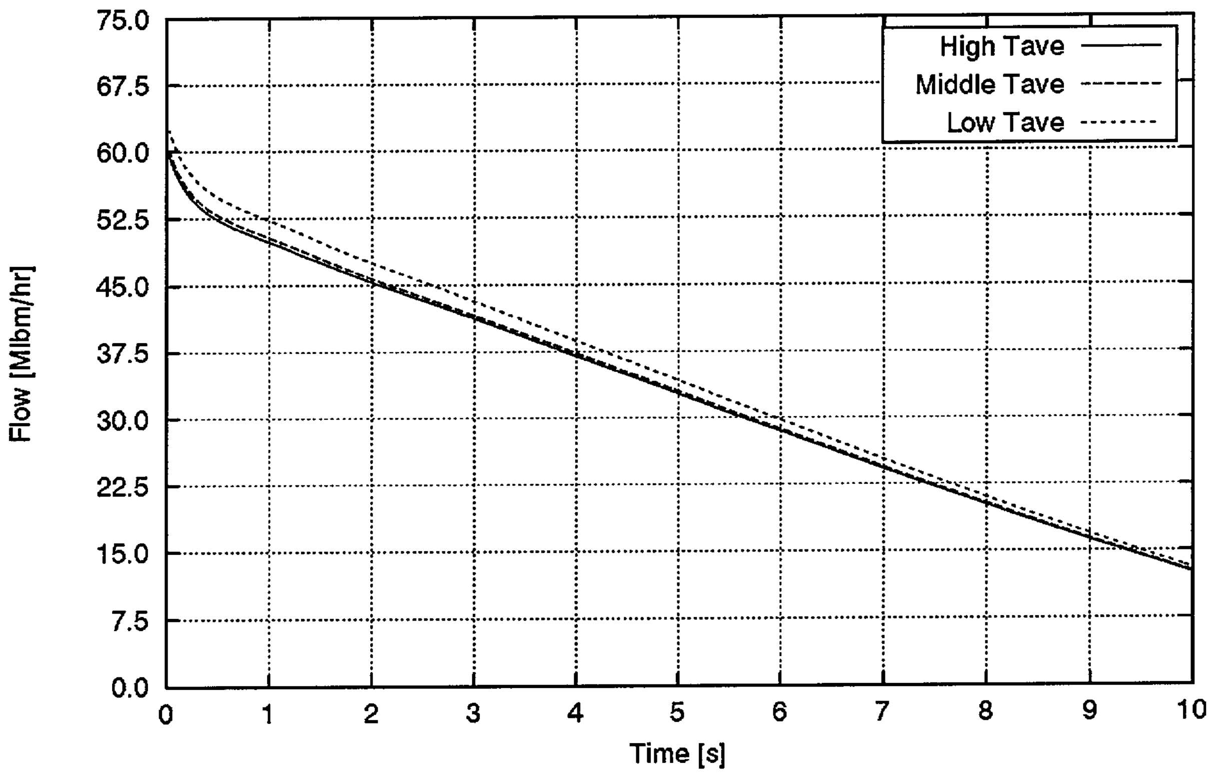


Figure 6.8.9-5
Loss of Reactor Coolant Flow – Underfrequency Trip
Core Flow vs. Time

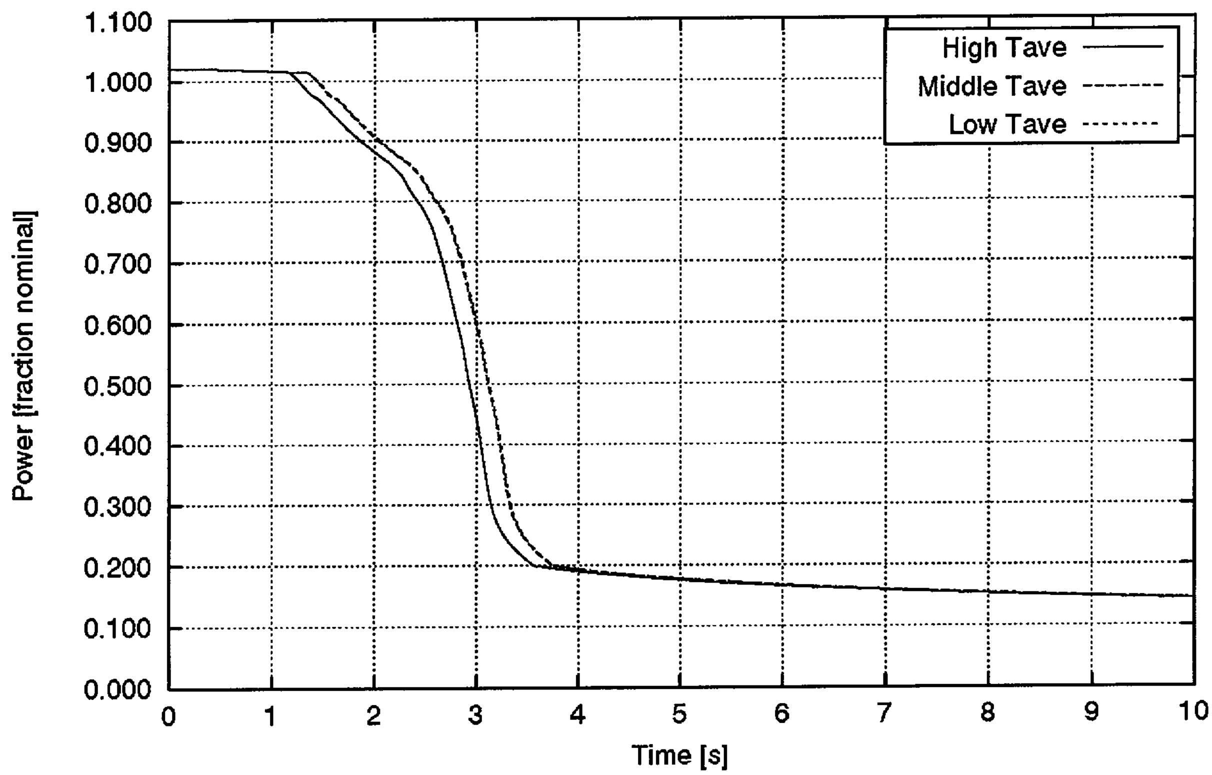


Figure 6.8.9-6
Loss of Reactor Coolant Flow – Underfrequency Trip
Reactor Power vs. Time

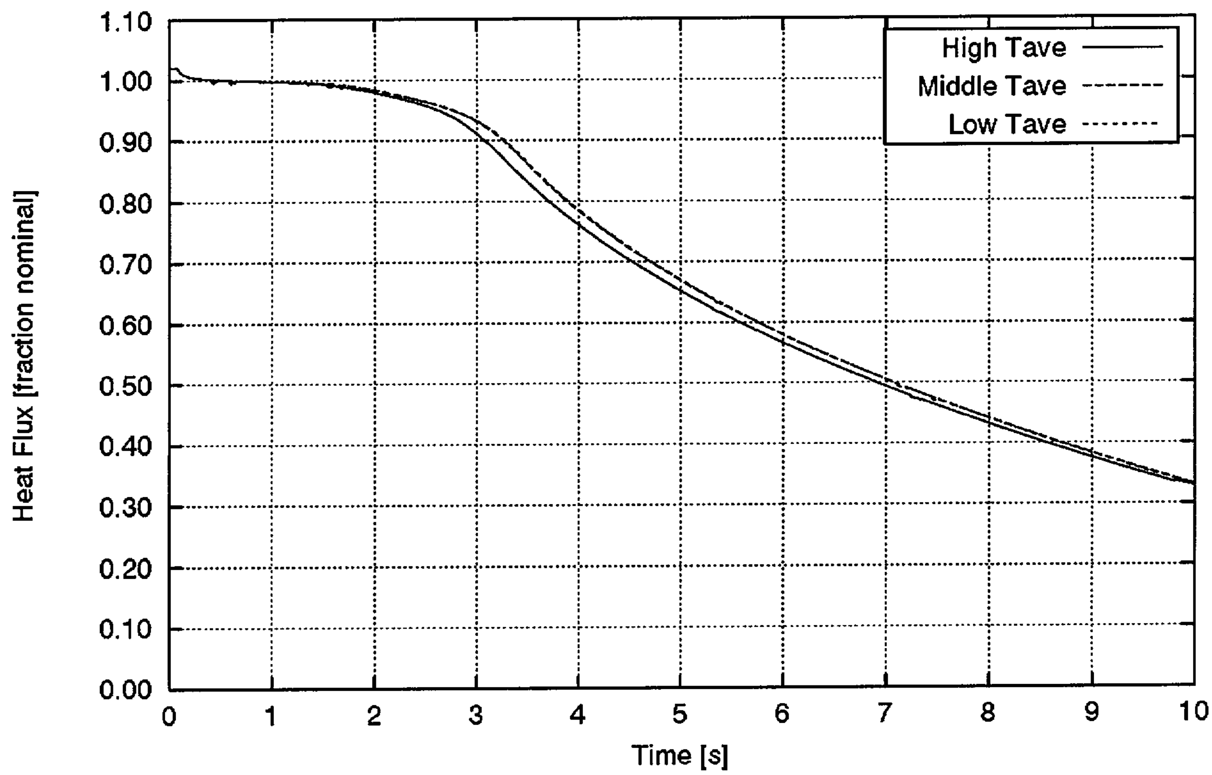


Figure 6.8.9-7
Loss of Reactor Coolant Flow – Underfrequency Trip
Heat Flux vs. Time

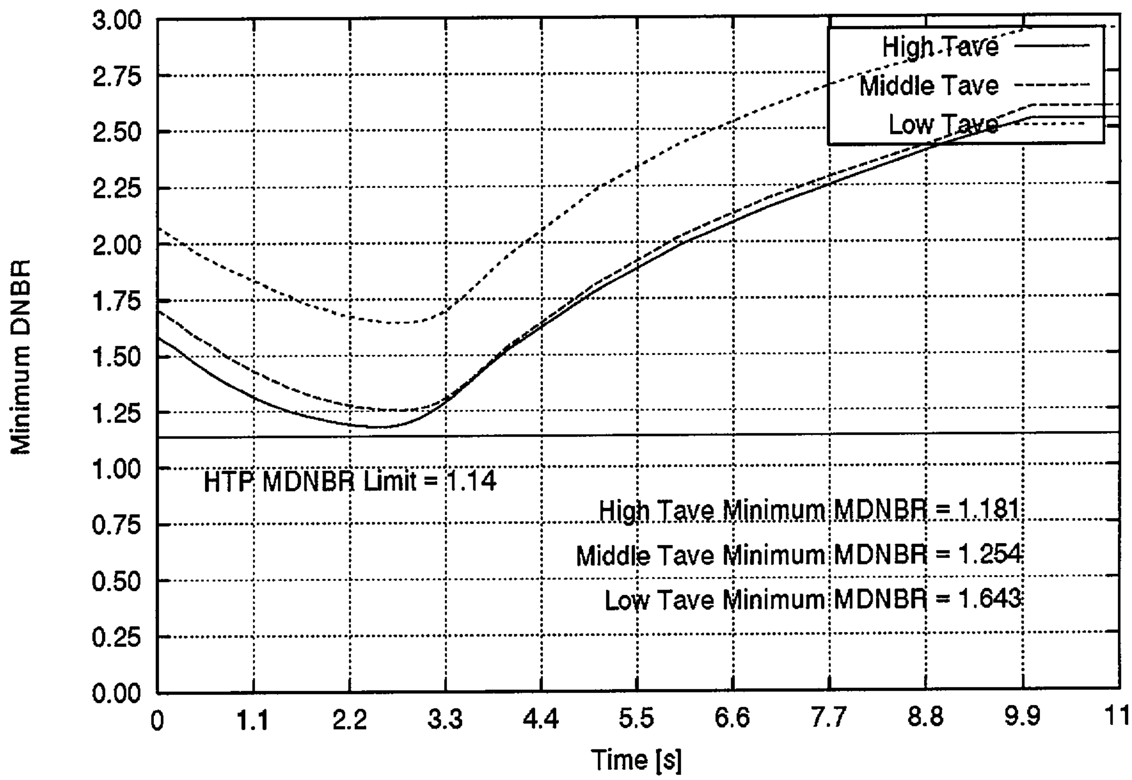


Figure 6.8.9-8
Loss of Reactor Coolant Flow – Underfrequency Trip
Minimum DNBR vs. Time

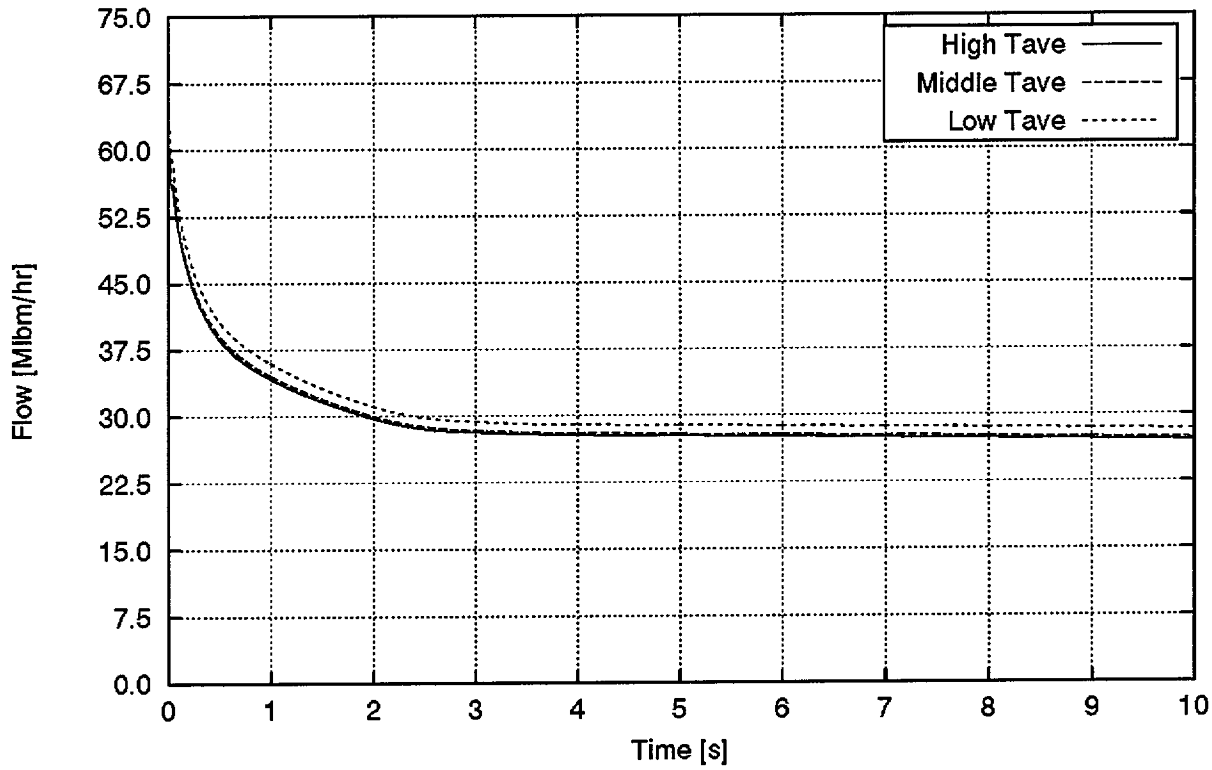


Figure 6.8.9-9
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Core Flow vs. Time

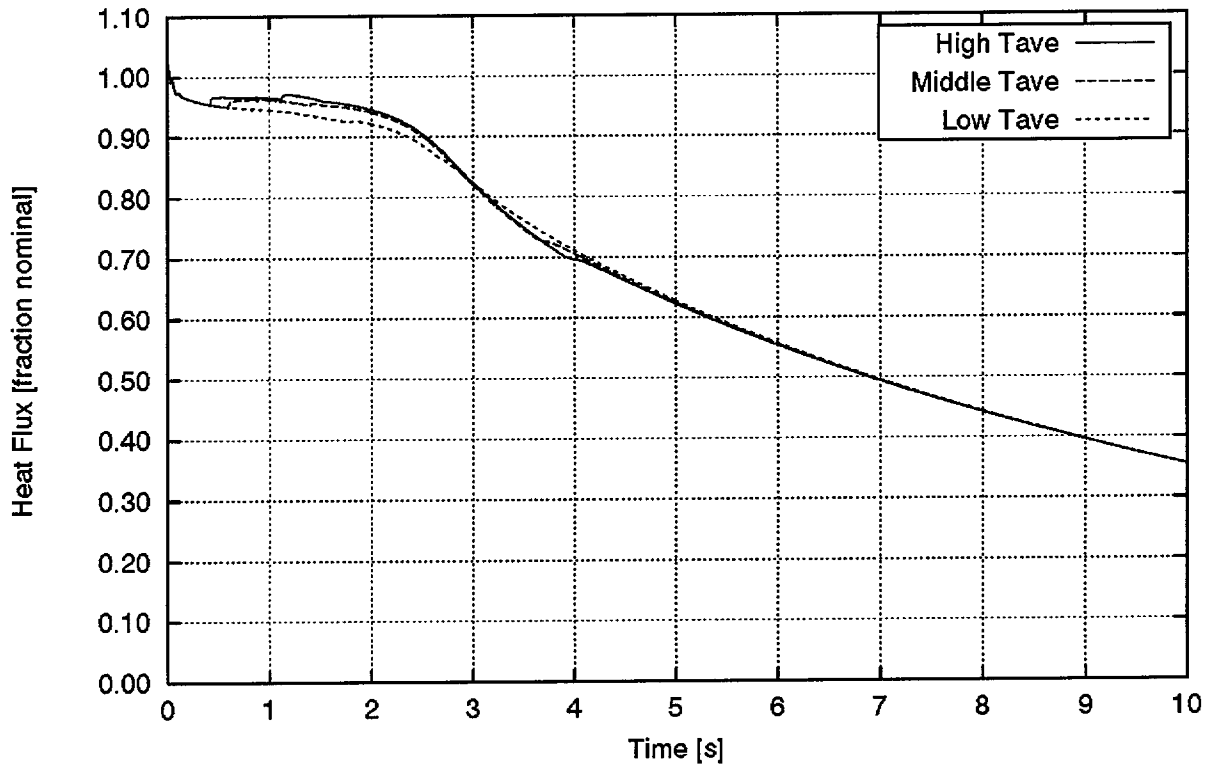


Figure 6.8.9-10
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Heat Flux vs. Time

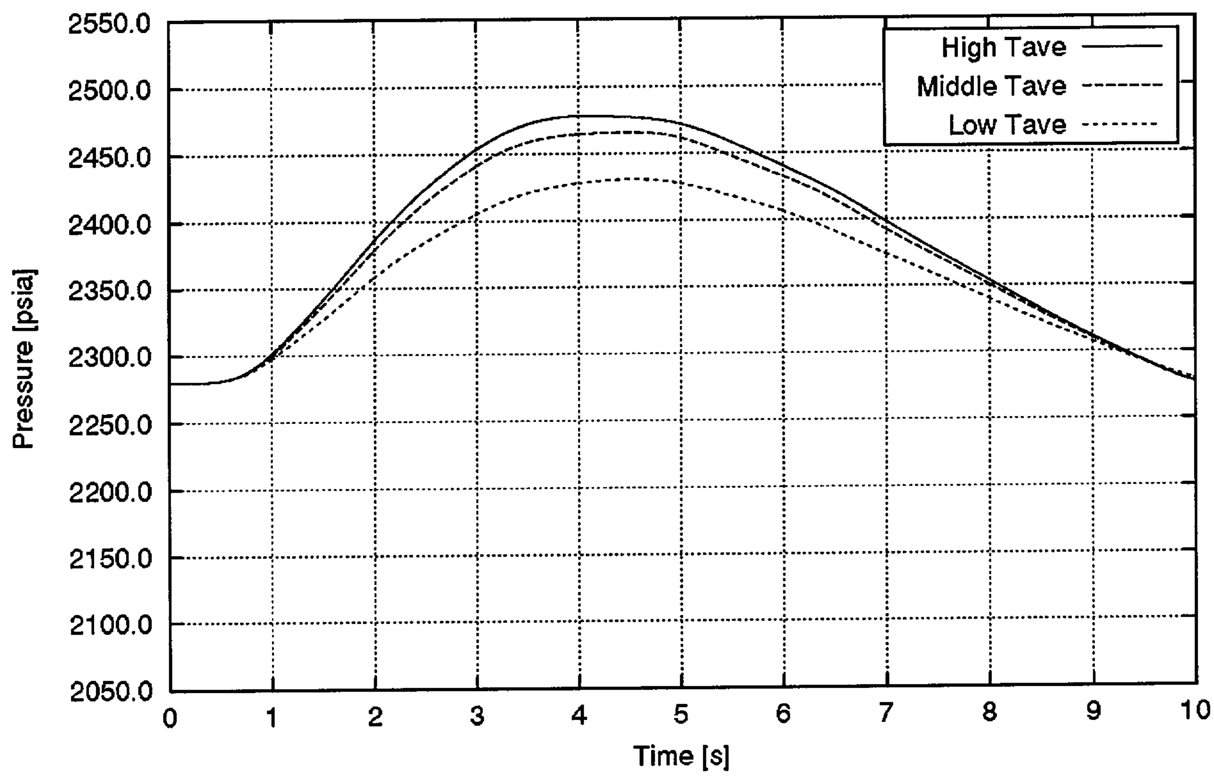


Figure 6.8.9-11
Loss of Reactor Coolant Flow - Locked Rotor - High Pressure
Pressurizer Pressure vs. Time

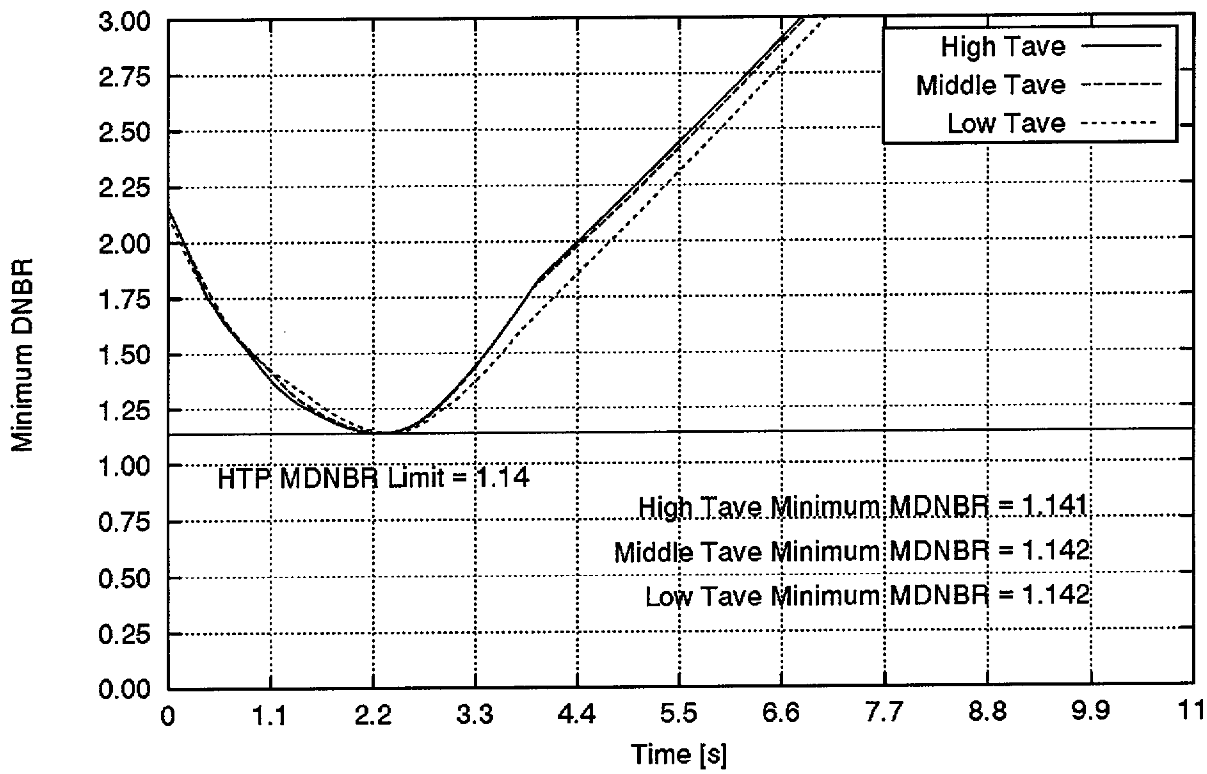


Figure 6.8.9-12
Loss of Reactor Coolant Flow - Locked Rotor
Minimum DNBR vs. Time

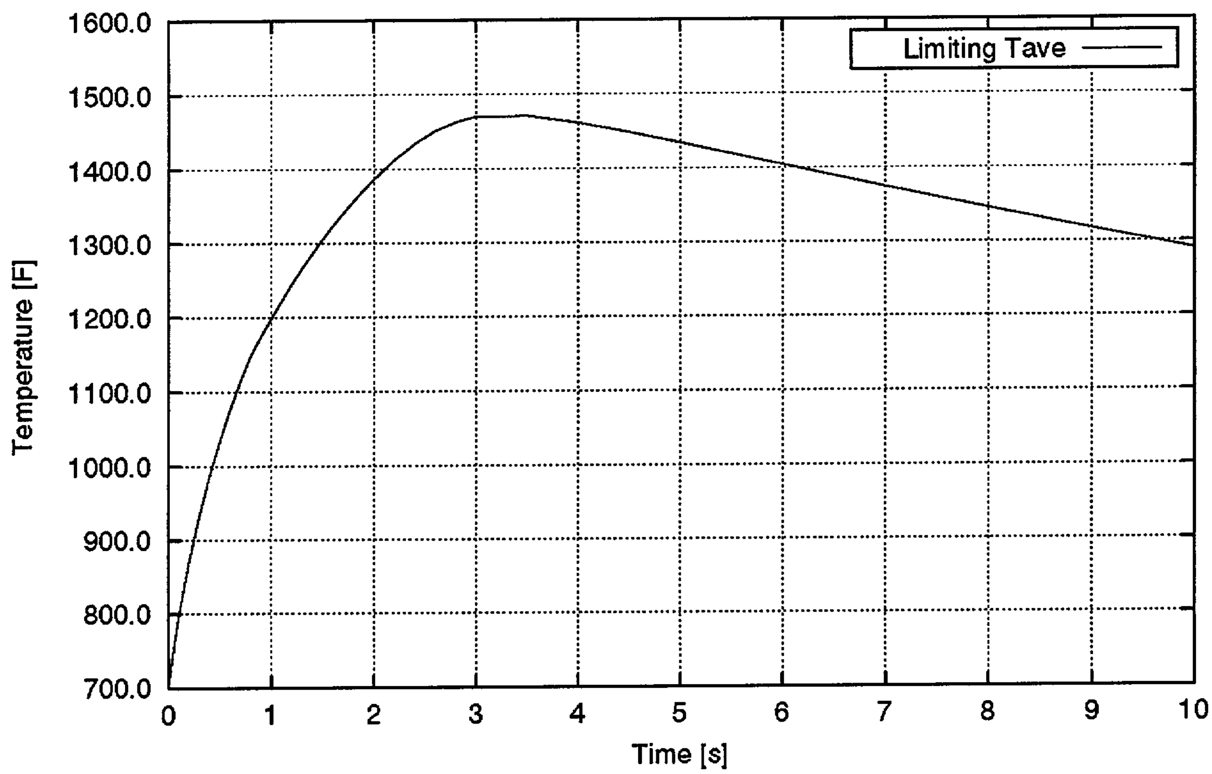


Figure 6.8.9-13
Loss of Reactor Coolant Flow - Locked Rotor
Hot Spot Cladding Temperature vs. Time

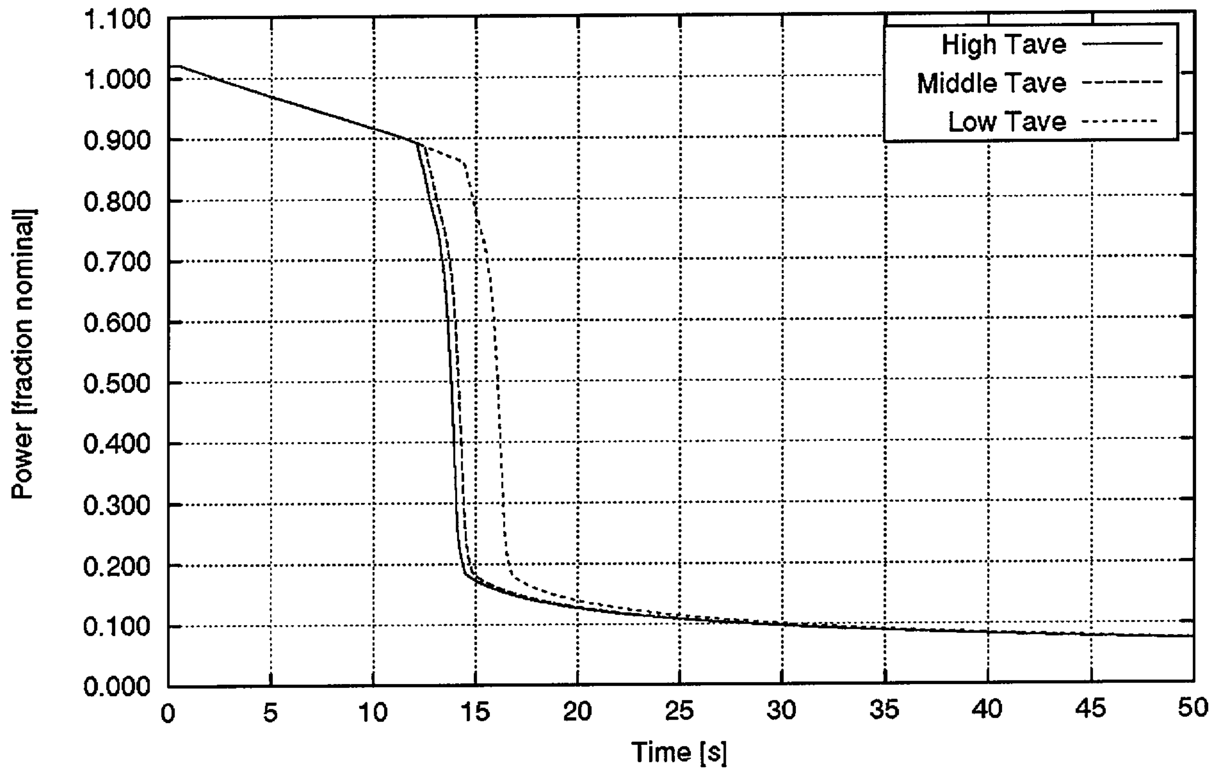


Figure 6.8.10-1
Loss of External Electrical Load - BOC Auto Control
Reactor Power vs. Time

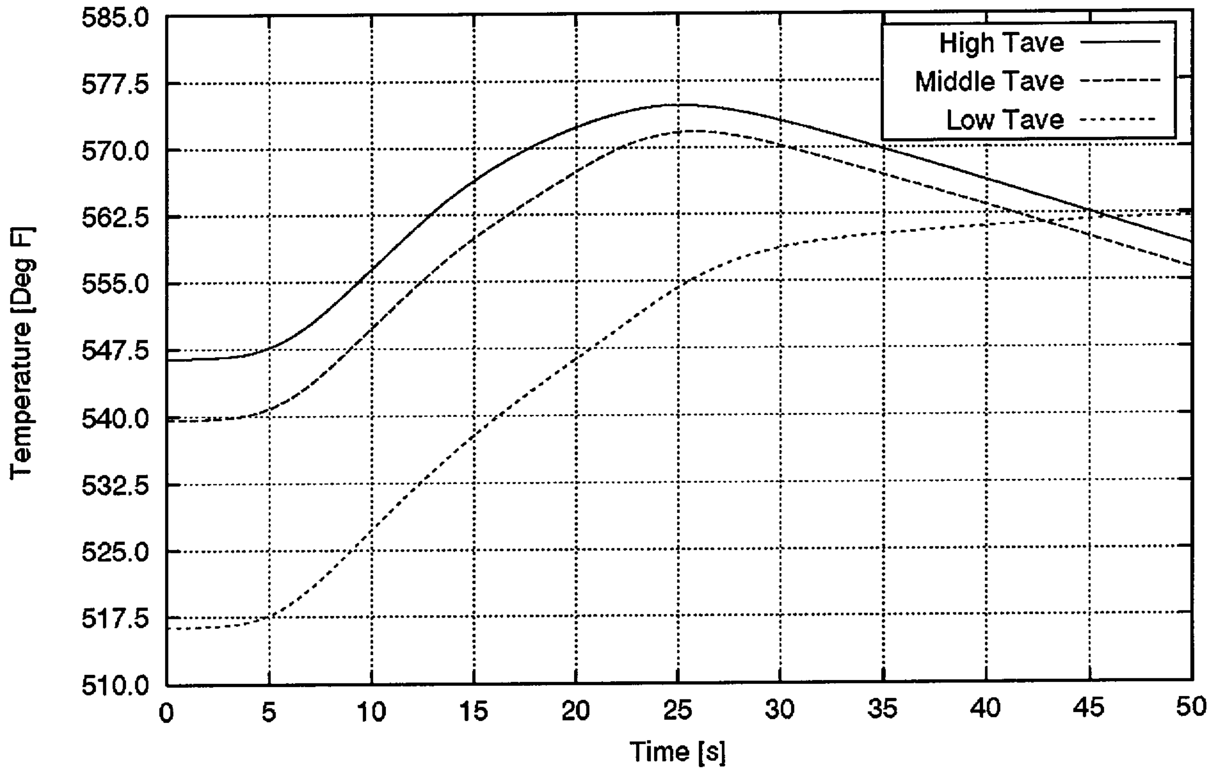


Figure 6.8.10-2
Loss of External Electrical Load - BOC Auto Control
T_{inlet} vs. Time

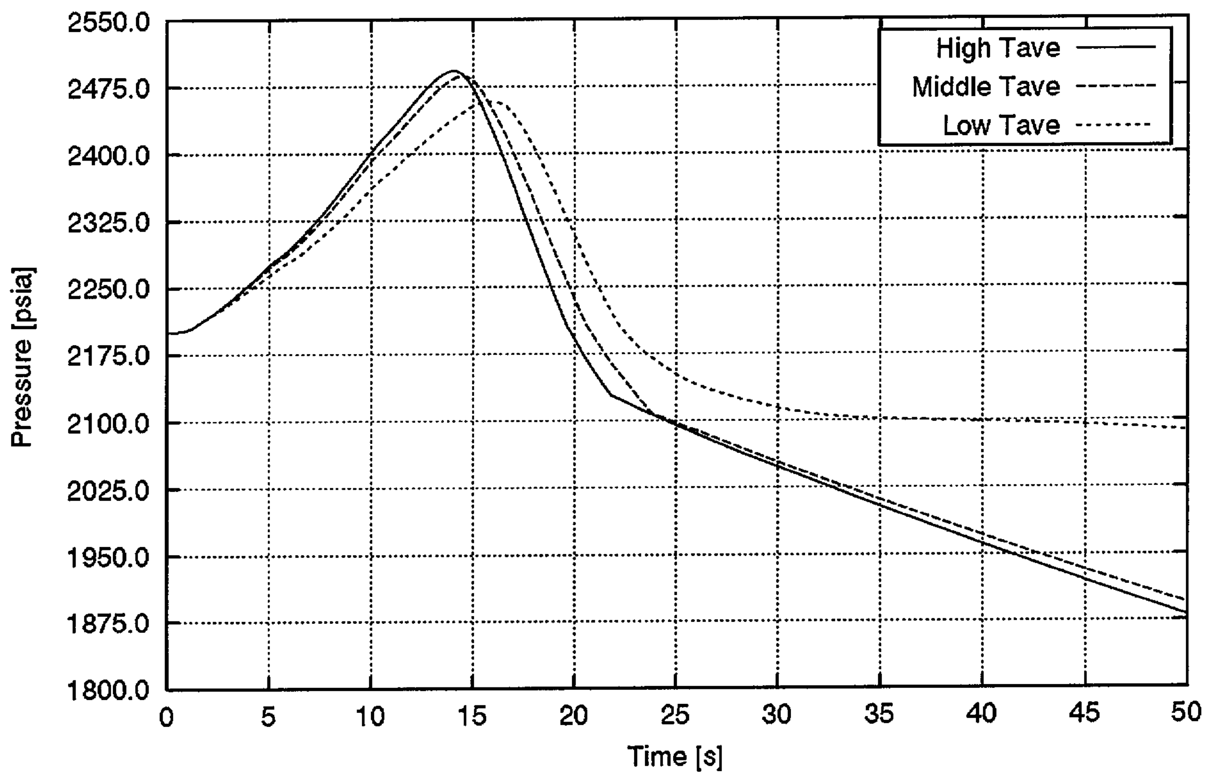


Figure 6.8.10-3
Loss of External Electrical Load - BOC Auto Control
Pressurizer Pressure vs. Time

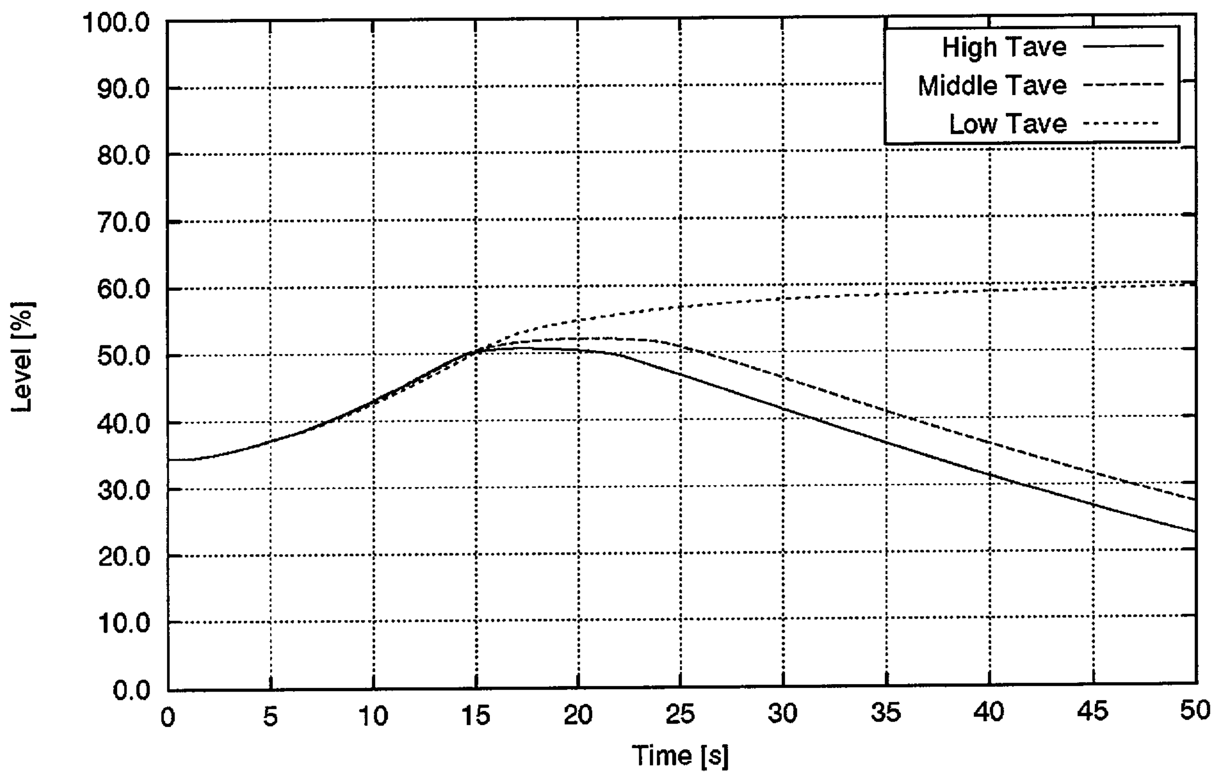


Figure 6.8.10-4
Loss of External Electrical Load - BOC Auto Control
Pressurizer Water Level vs. Time

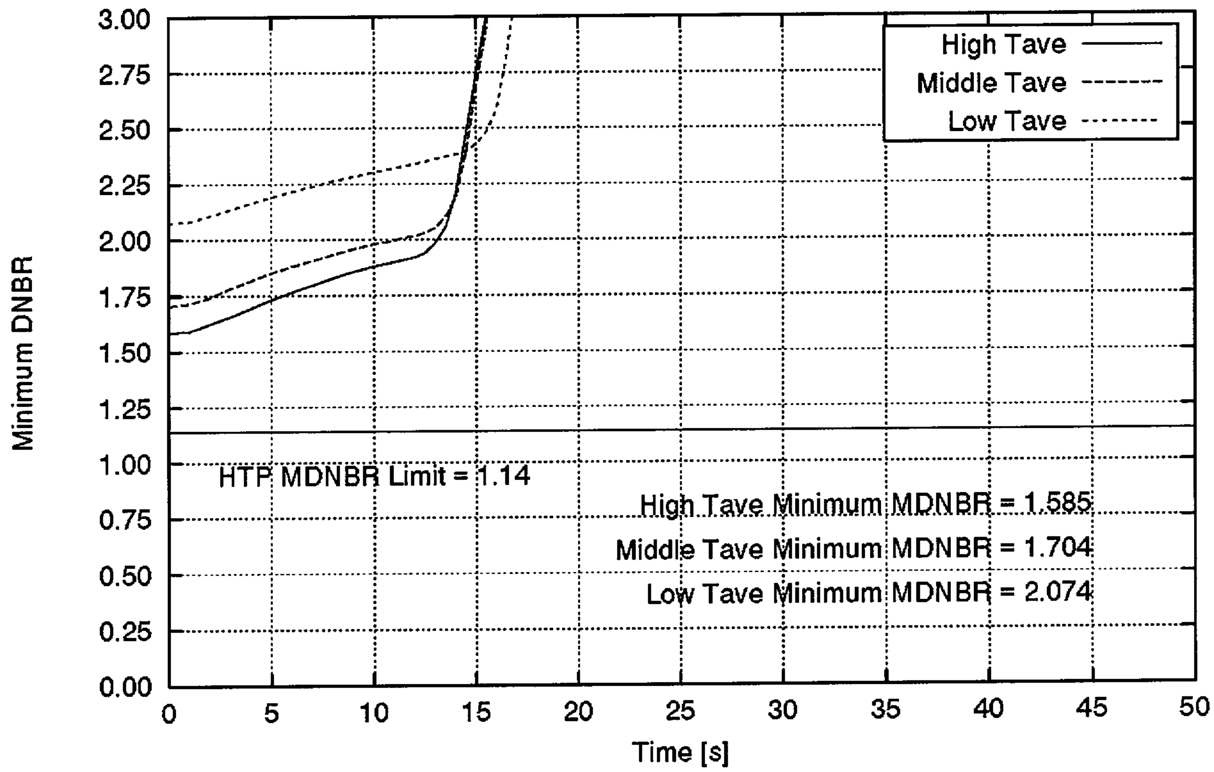


Figure 6.8.10-5
Loss of External Electrical Load - BOC Auto Control
Minimum DNBR vs. Time

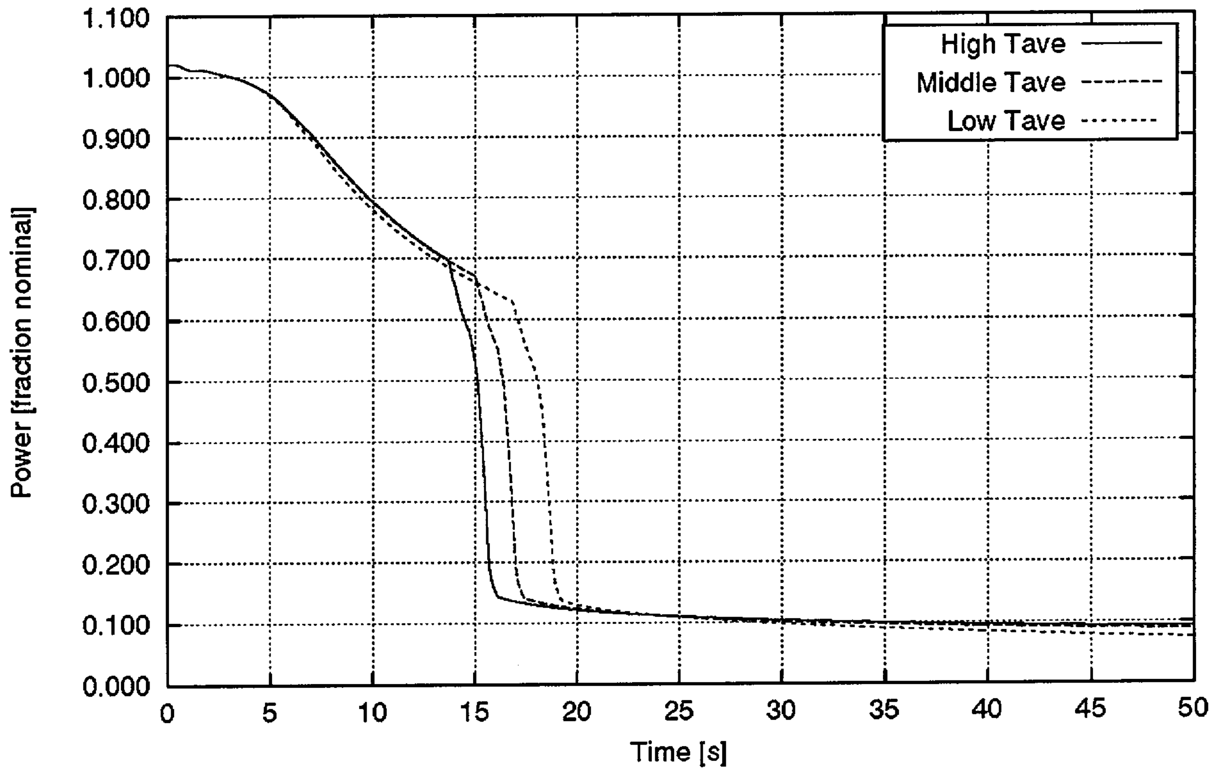


Figure 6.8.10-6
Loss of External Electrical Load - EOC Auto Control
Reactor Power vs. Time

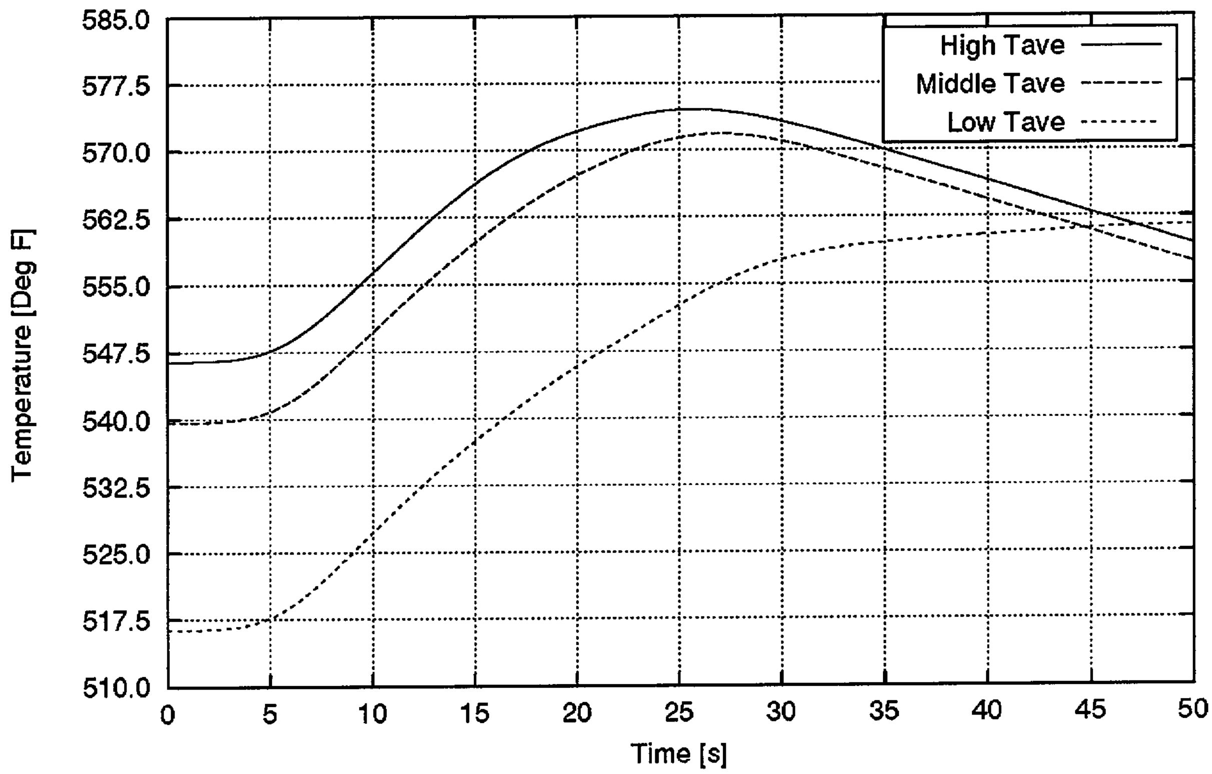


Figure 6.8.10-7
Loss of External Electrical Load - EOC Auto Control
T_{inlet} vs. Time

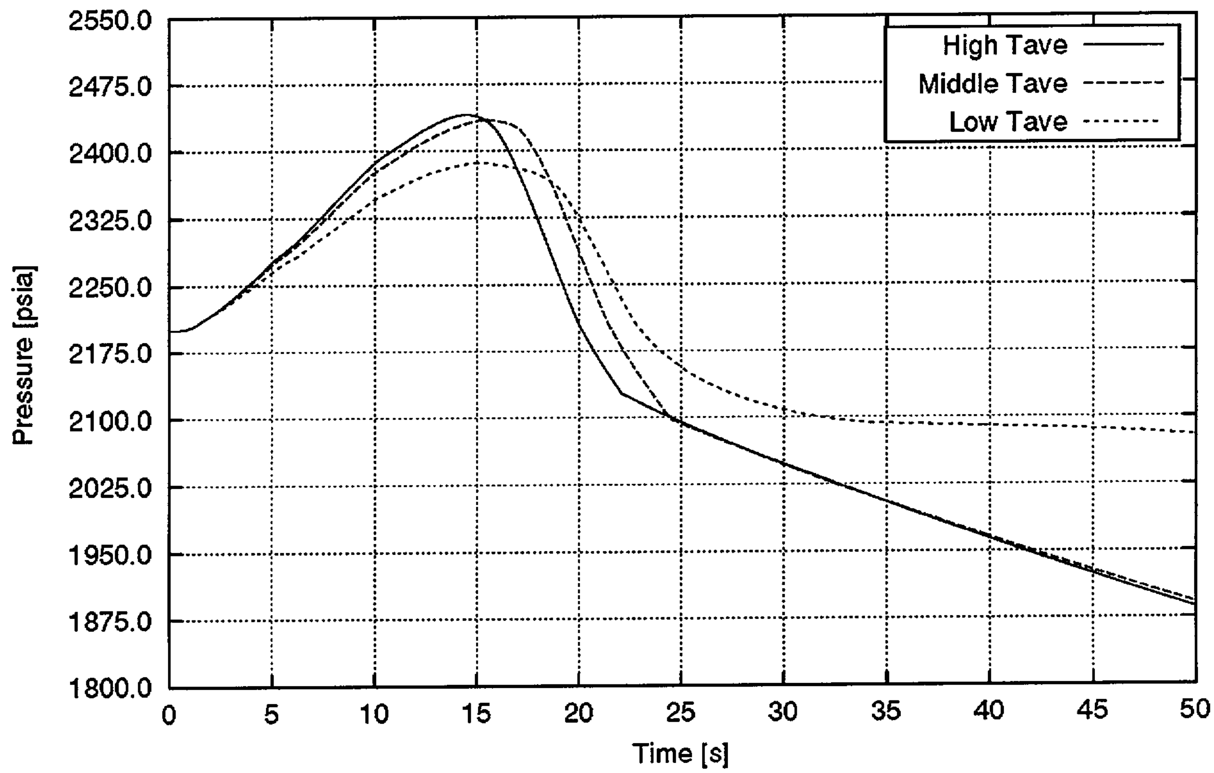


Figure 6.8.10-8
Loss of External Electrical Load - EOC Auto Control
Pressurizer Pressure vs. Time

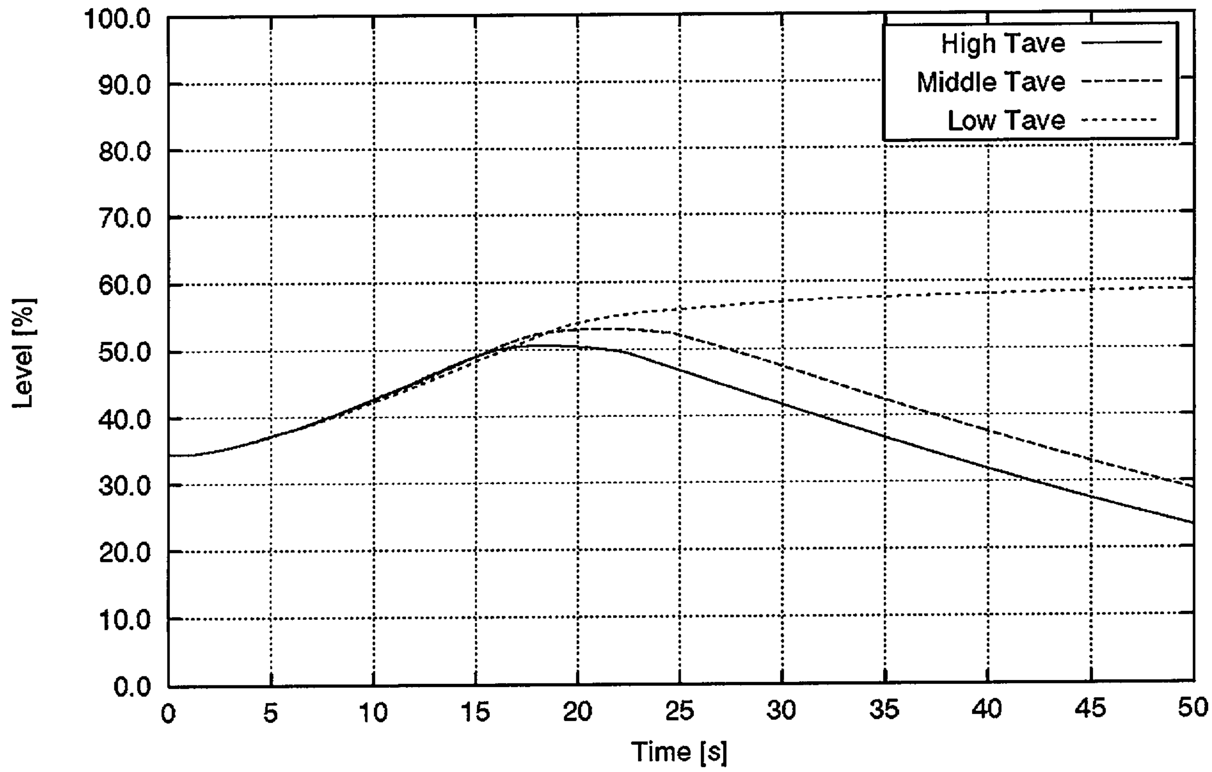


Figure 6.8.10-9
Loss of External Electrical Load - EOC Auto Control
Pressurizer Water Level vs. Time

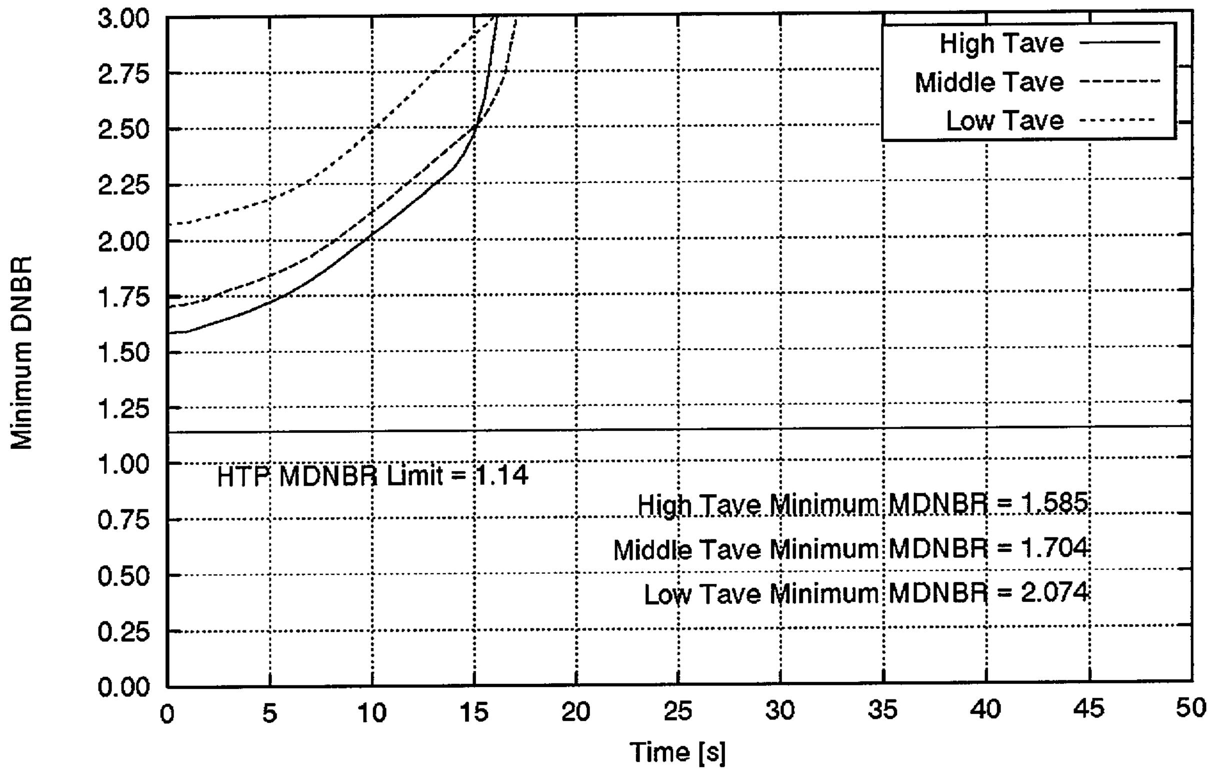


Figure 6.8.10-10
Loss of External Electrical Load - EOC Auto Control
Minimum DNBR vs. Time

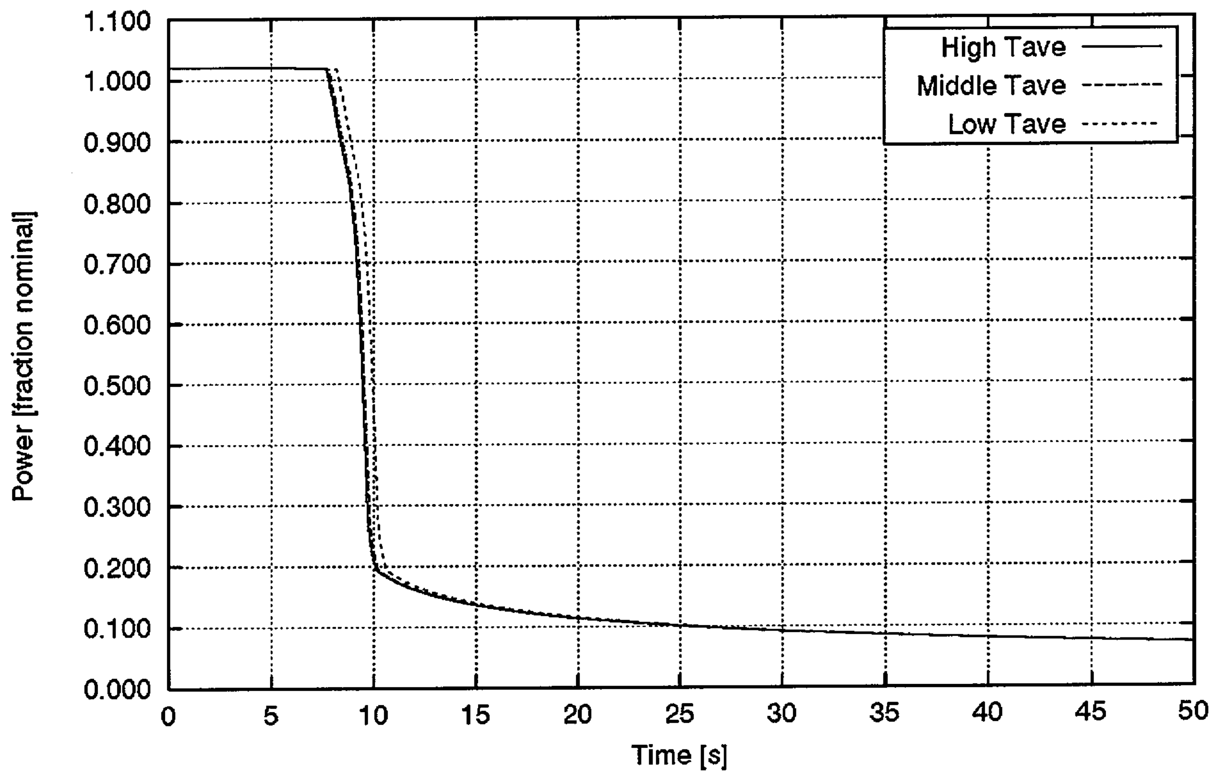


Figure 6.8.10-11
Loss of External Electrical Load - BOC Manual Control - High Pressure
Reactor Power vs. Time

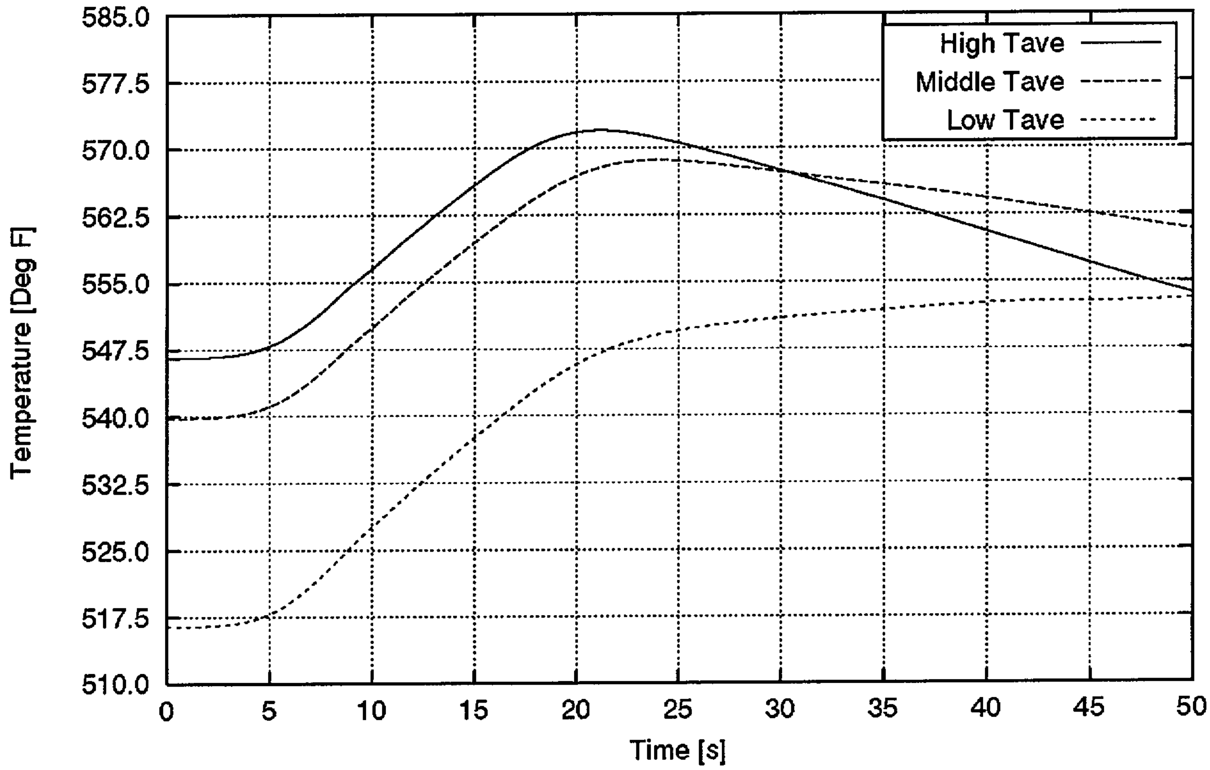


Figure 6.8.10-12
Loss of External Electrical Load - BOC Manual Control - High Pressure
T_{inlet} vs. Time

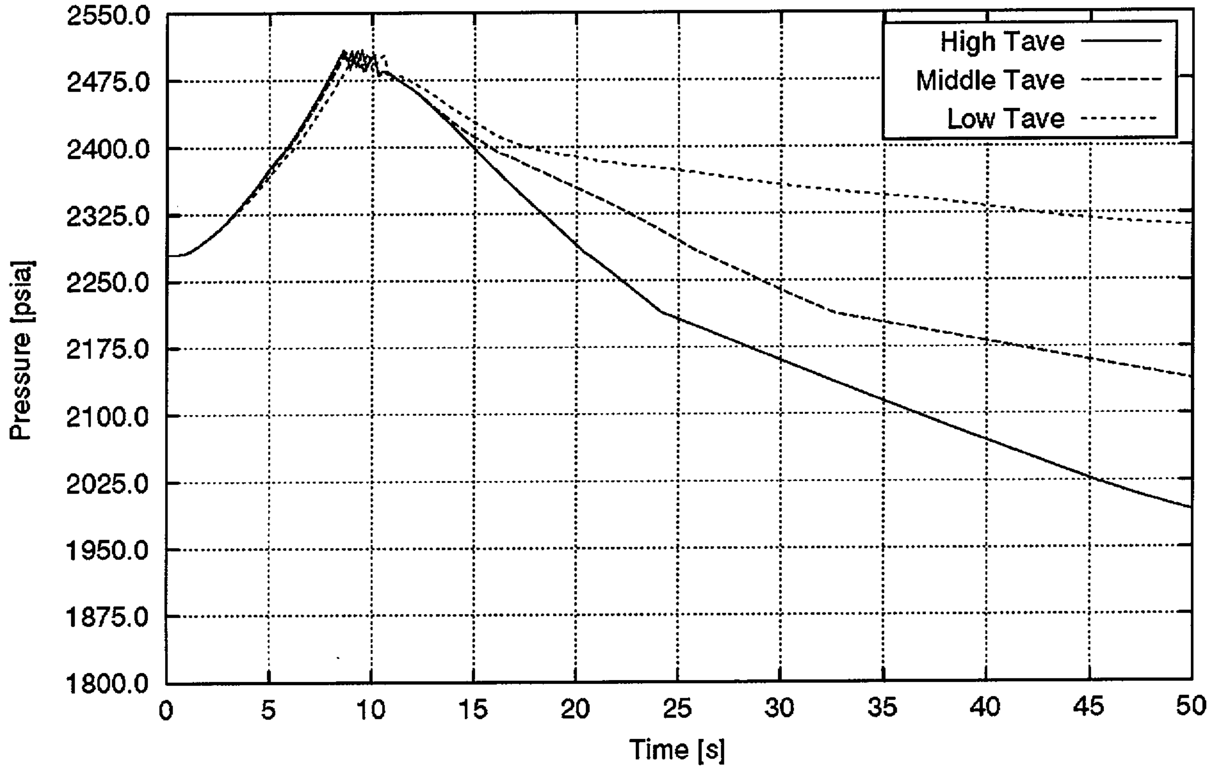


Figure 6.8.10-13
Loss of External Electrical Load - BOC Manual Control - High Pressure
Pressurizer Pressure vs. Time

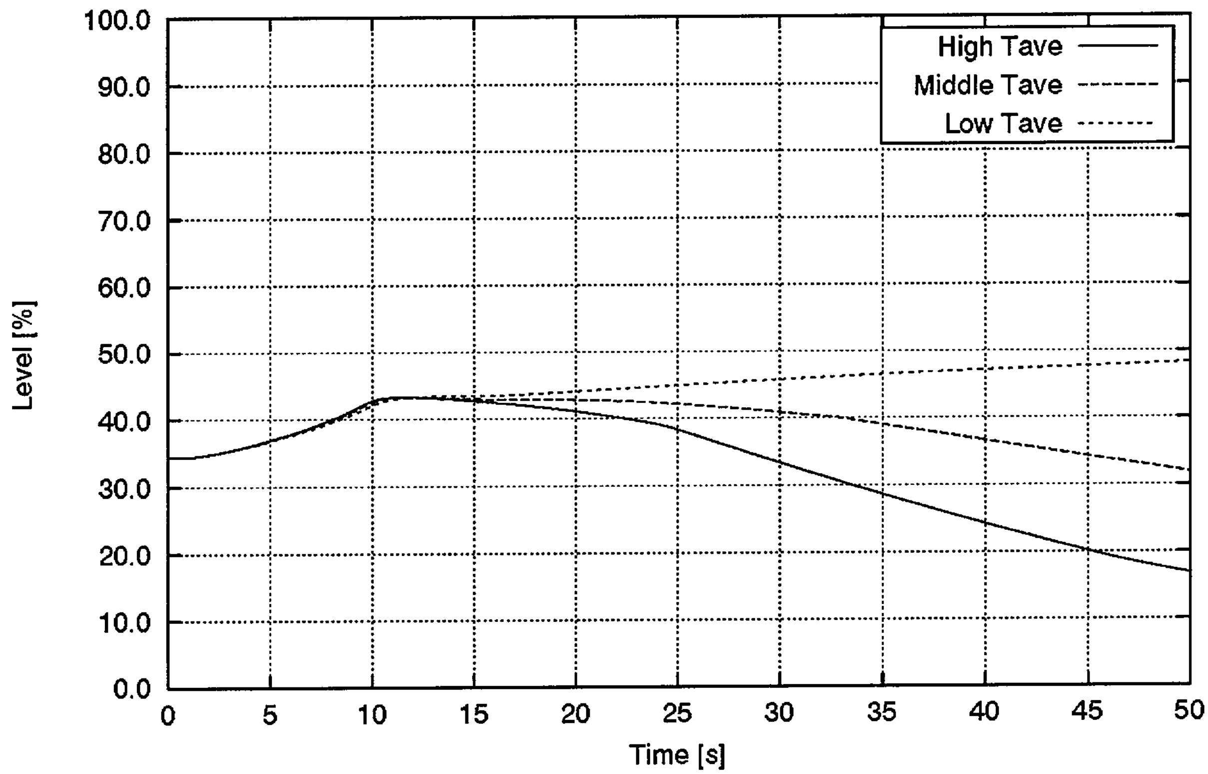


Figure 6.8.10-14
Loss of External Electrical Load - BOC Manual Control - High Pressure
Pressurizer Water Level vs. Time

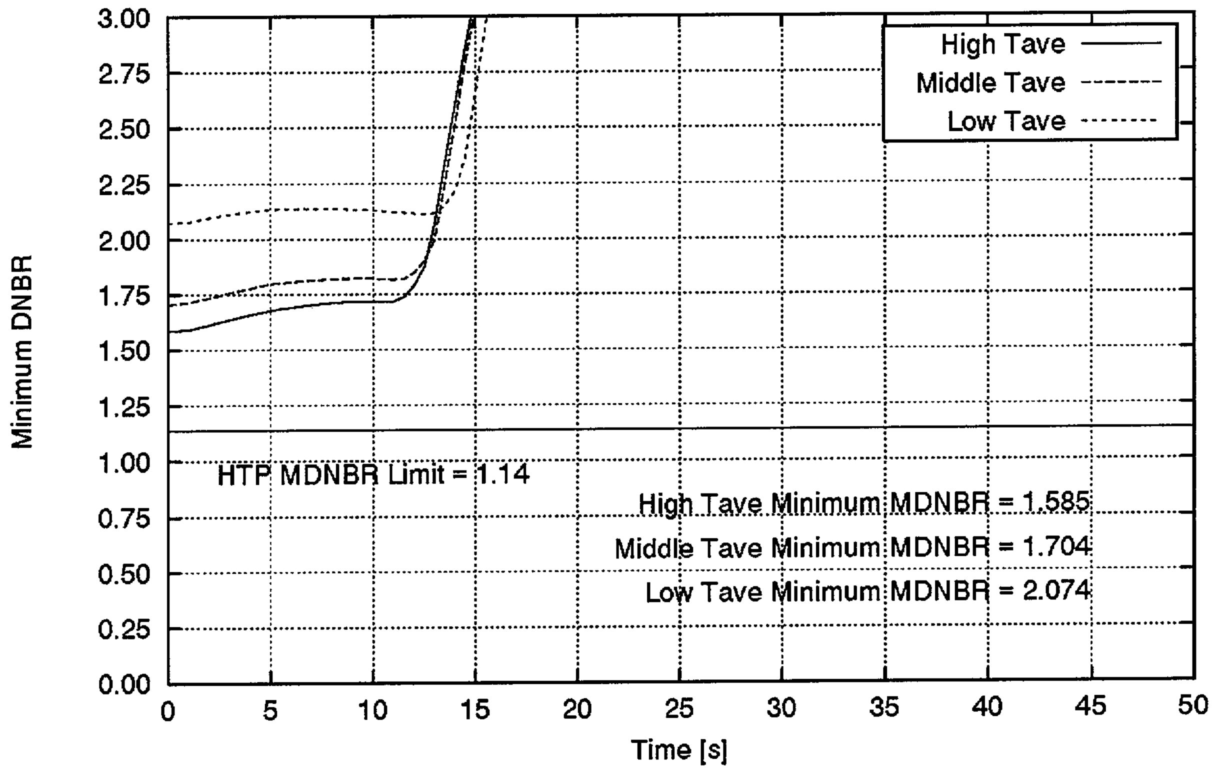


Figure 6.8.10-15
Loss of External Electrical Load - BOC Manual Control
Minimum DNBR vs. Time

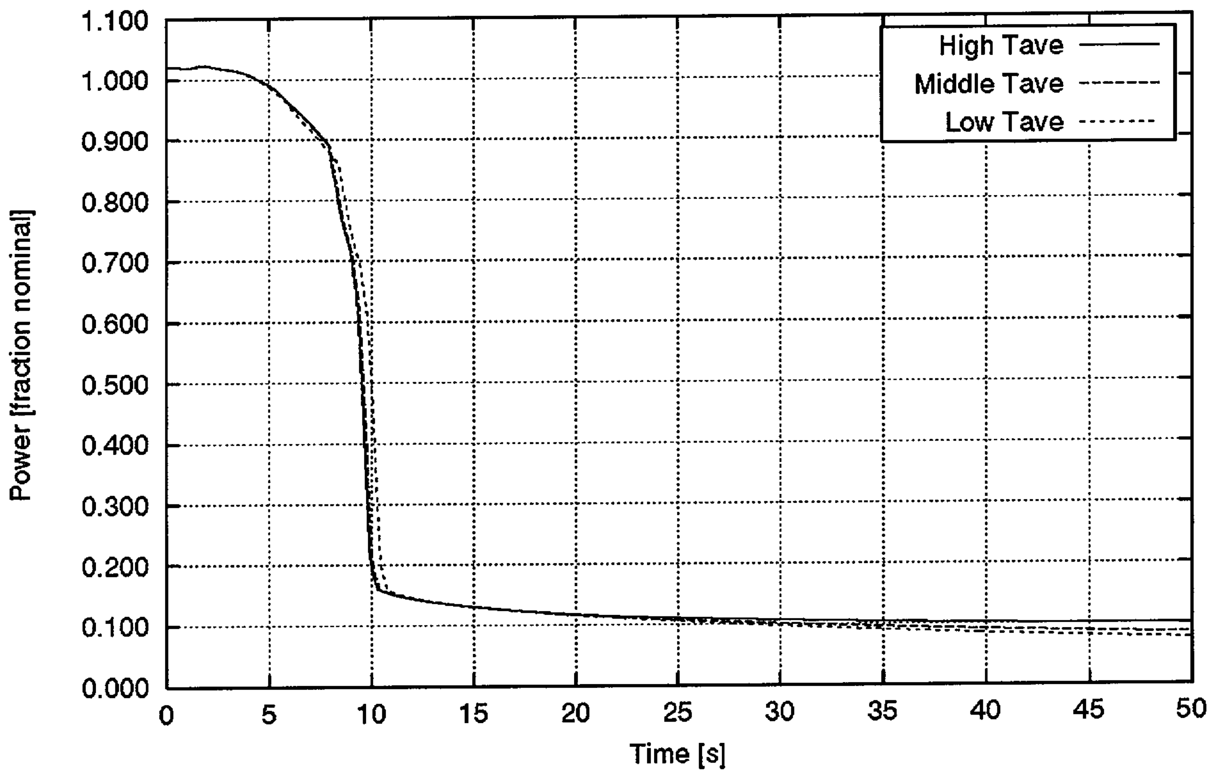


Figure 6.8.10-16
Loss of External Electrical Load - EOC Manual Control - High Pressure
Reactor Power vs. Time

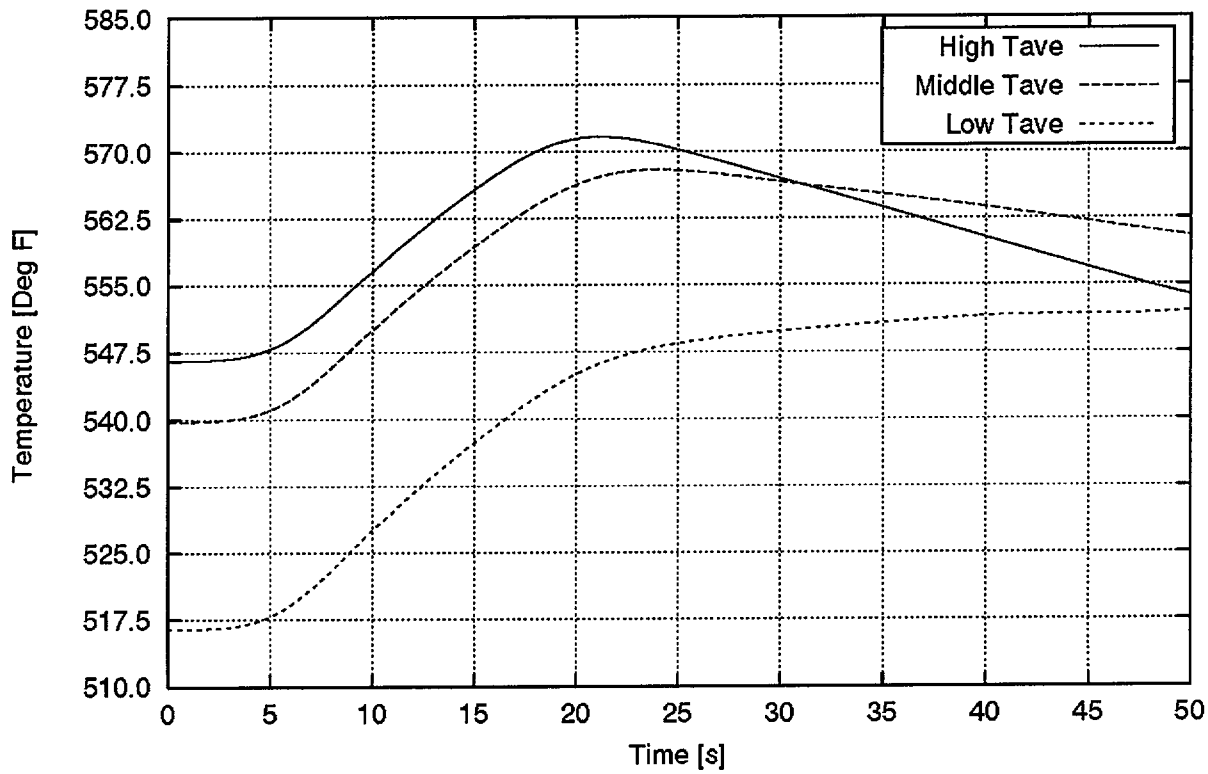


Figure 6.8.10-17
Loss of External Electrical Load - EOC Manual Control - High Pressure
T_{inlet} vs. Time

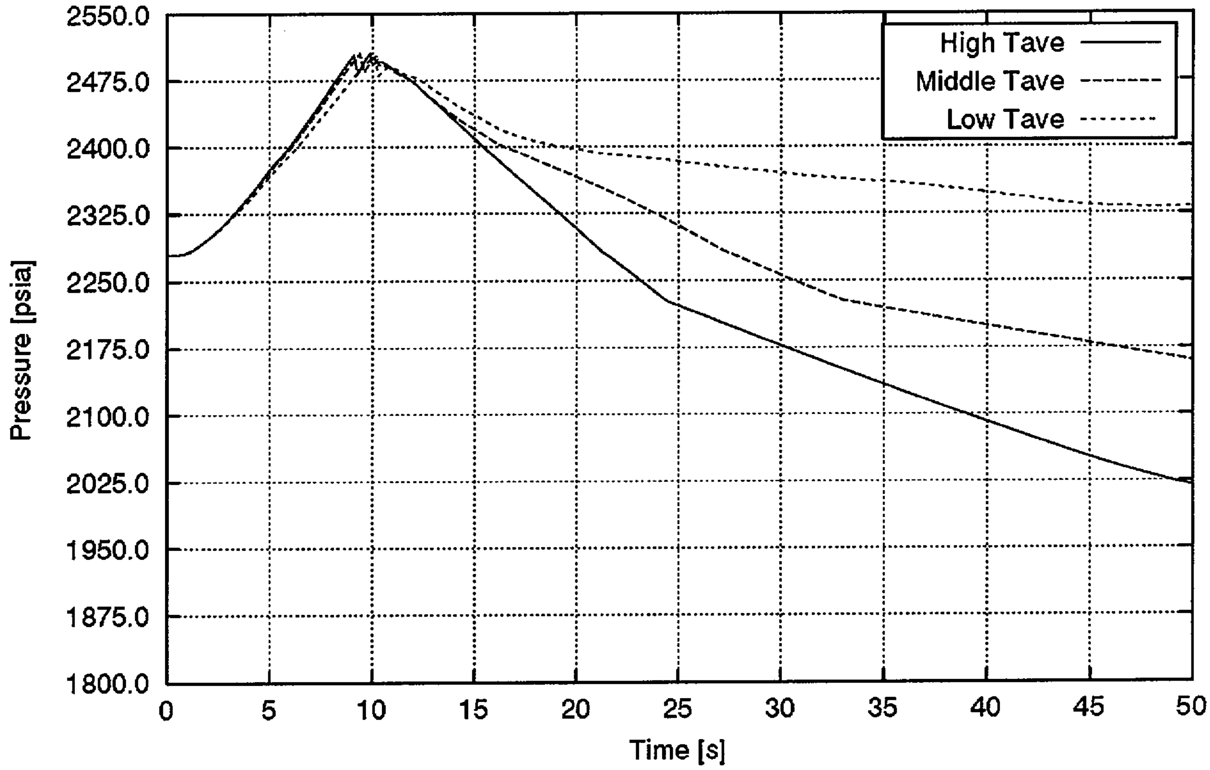


Figure 6.8.10-18
Loss of External Electrical Load - EOC Manual Control - High Pressure
Pressurizer Pressure vs. Time

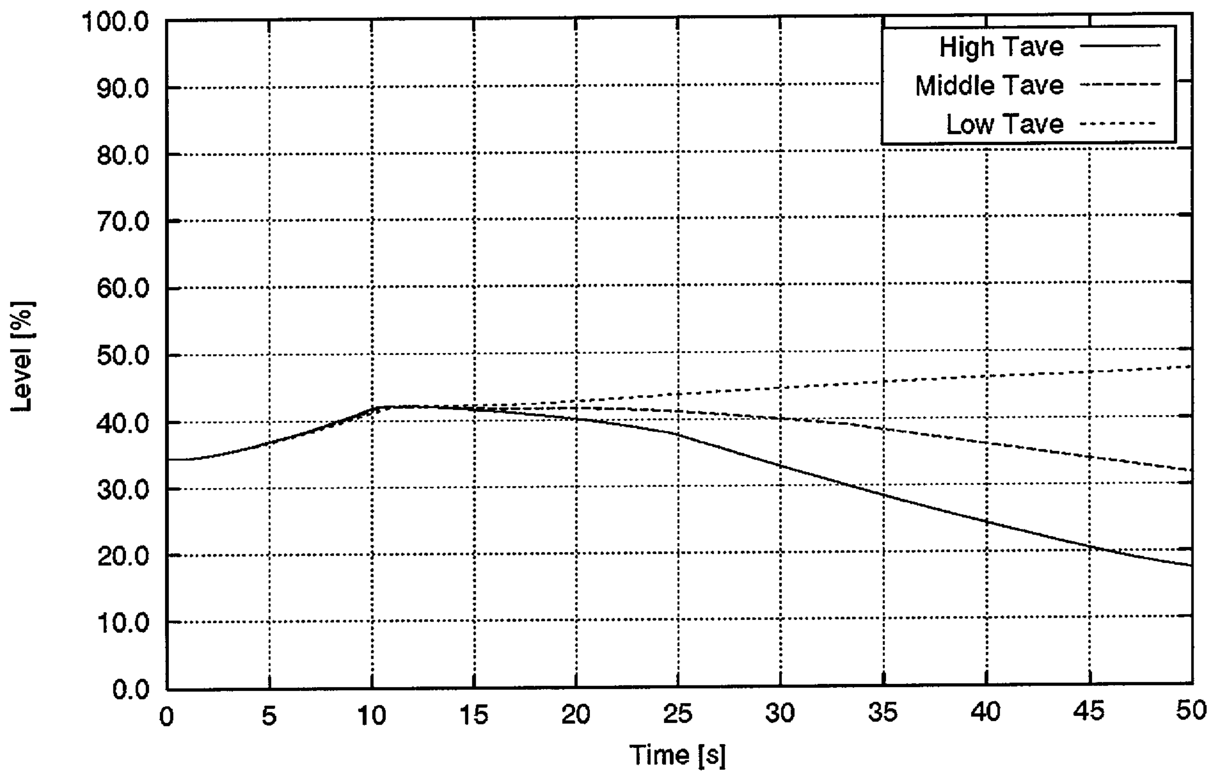


Figure 6.8.10-19
Loss of External Electrical Load - EOC Manual Control - High Pressure
Pressurizer Water Level vs. Time

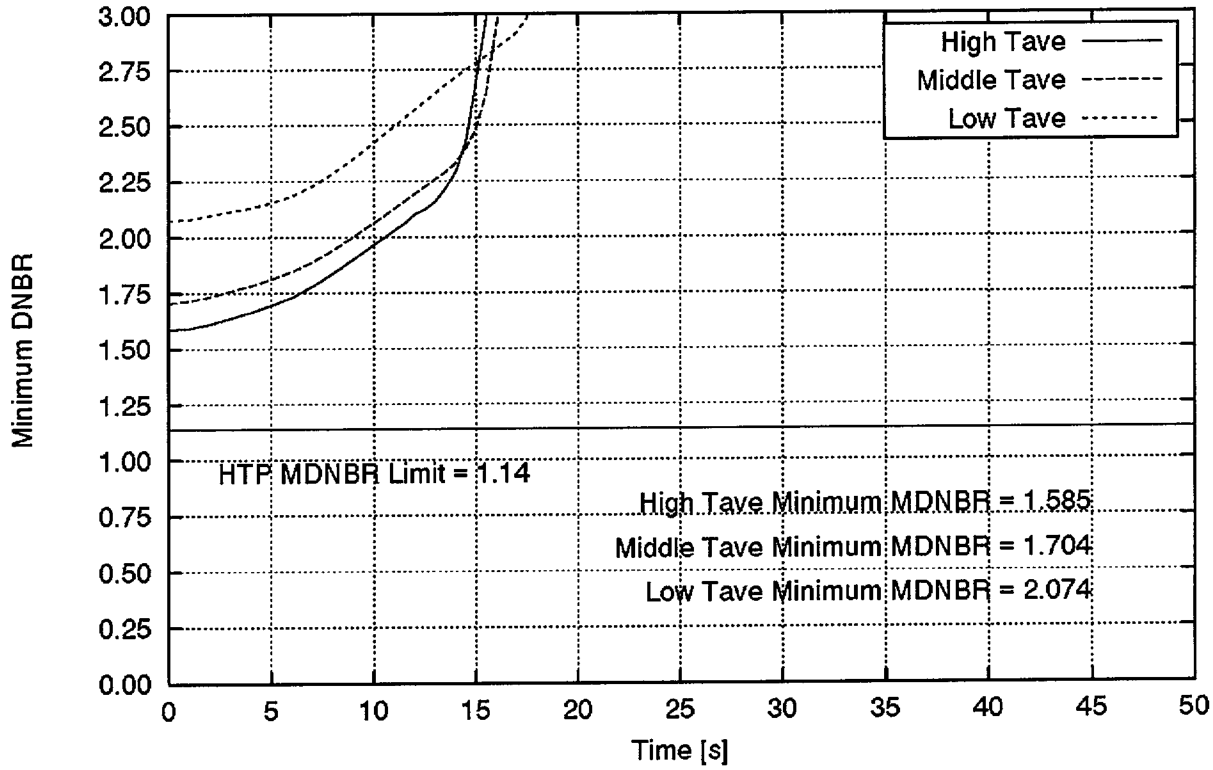


Figure 6.8.10-20
Loss of External Electrical Load - EOC Manual Control
Minimum DNBR vs. Time

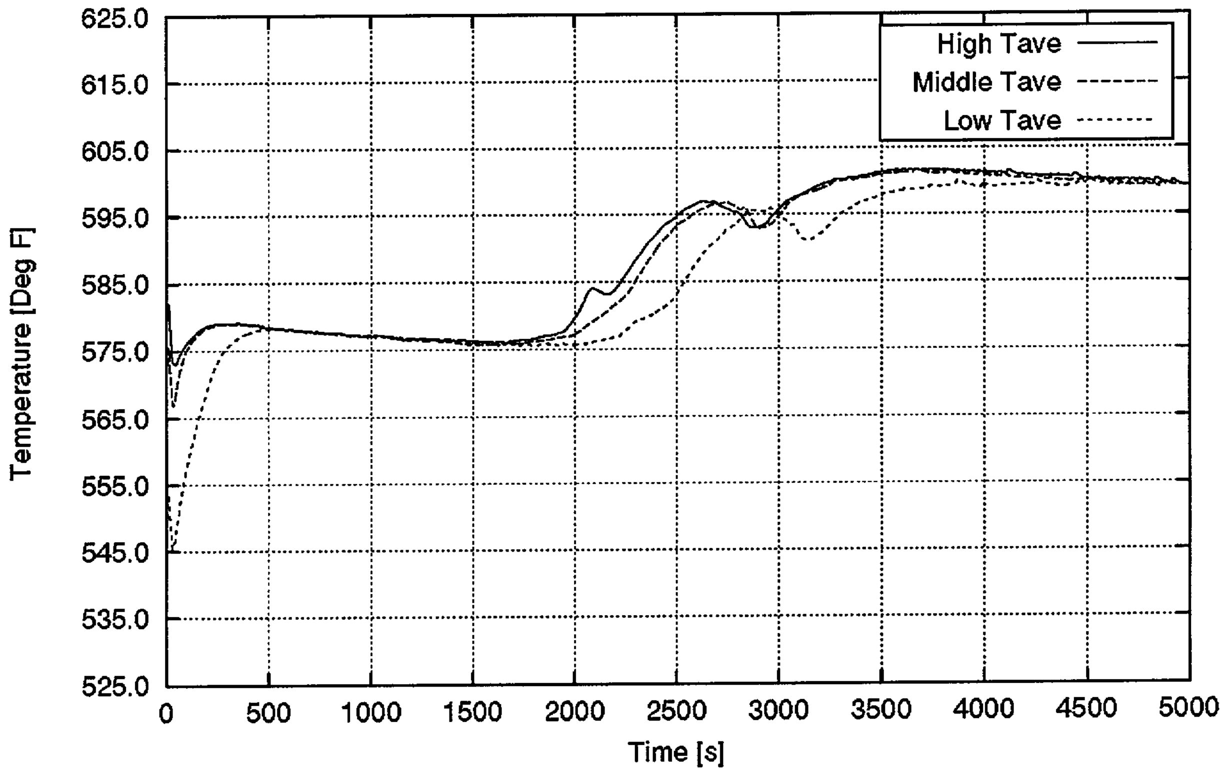


Figure 6.8.11-1
Loss of Normal Feedwater - High Pressure
T_{avg} vs. Time

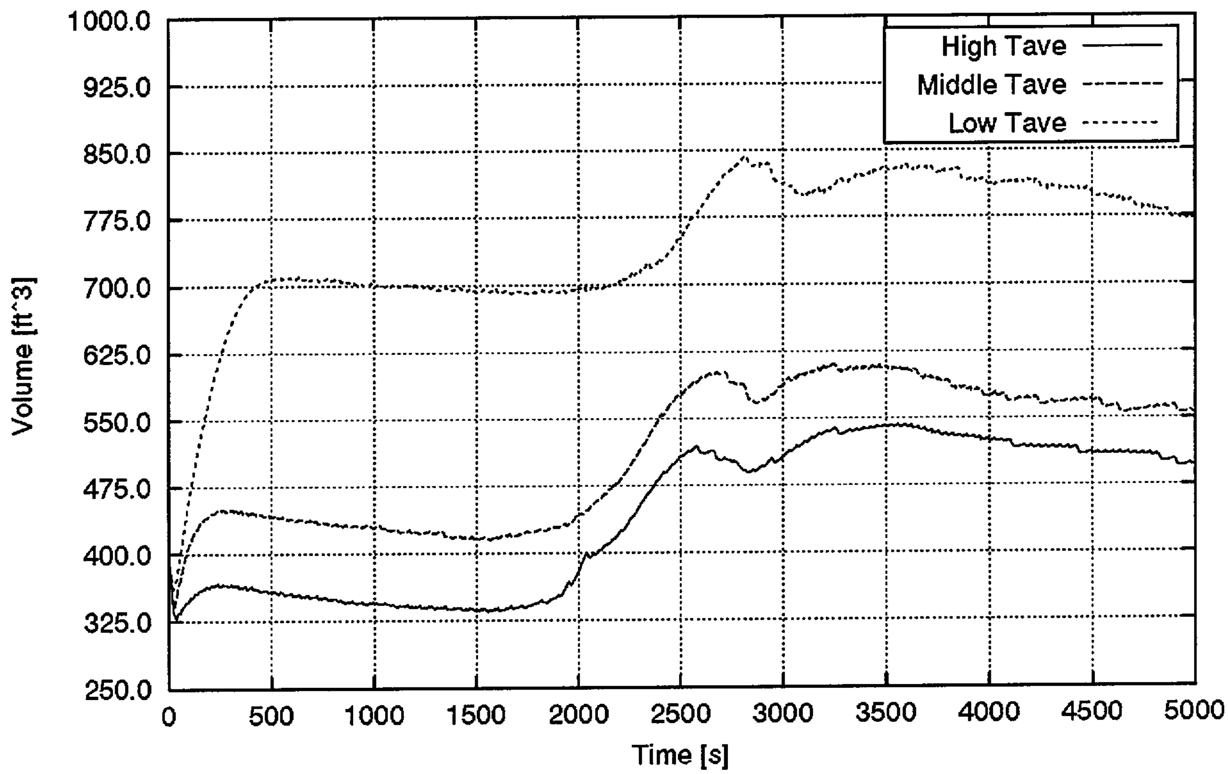


Figure 6.8.11-2
Loss of Normal Feedwater - High Pressure
Pressurizer Liquid Volume vs. Time

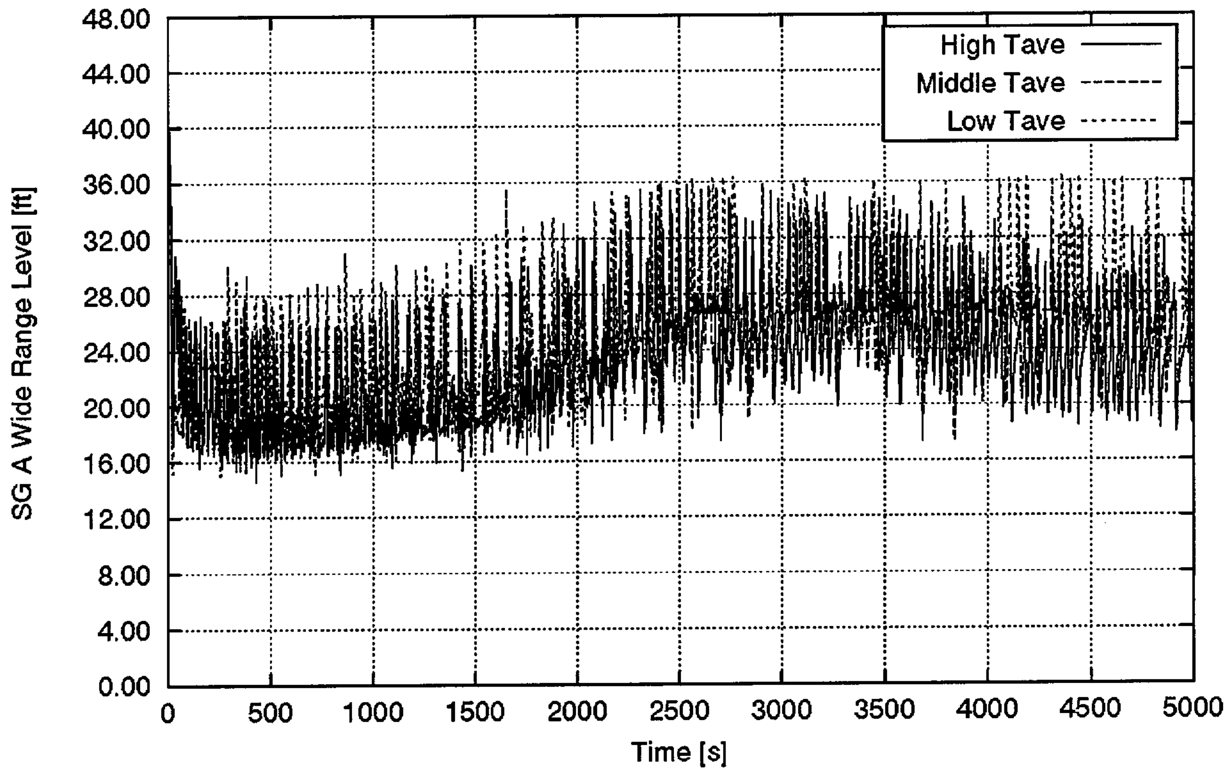


Figure 6.8.11-3
Loss of Normal Feedwater - High Pressure
SG A Wide-Range Level vs. Time

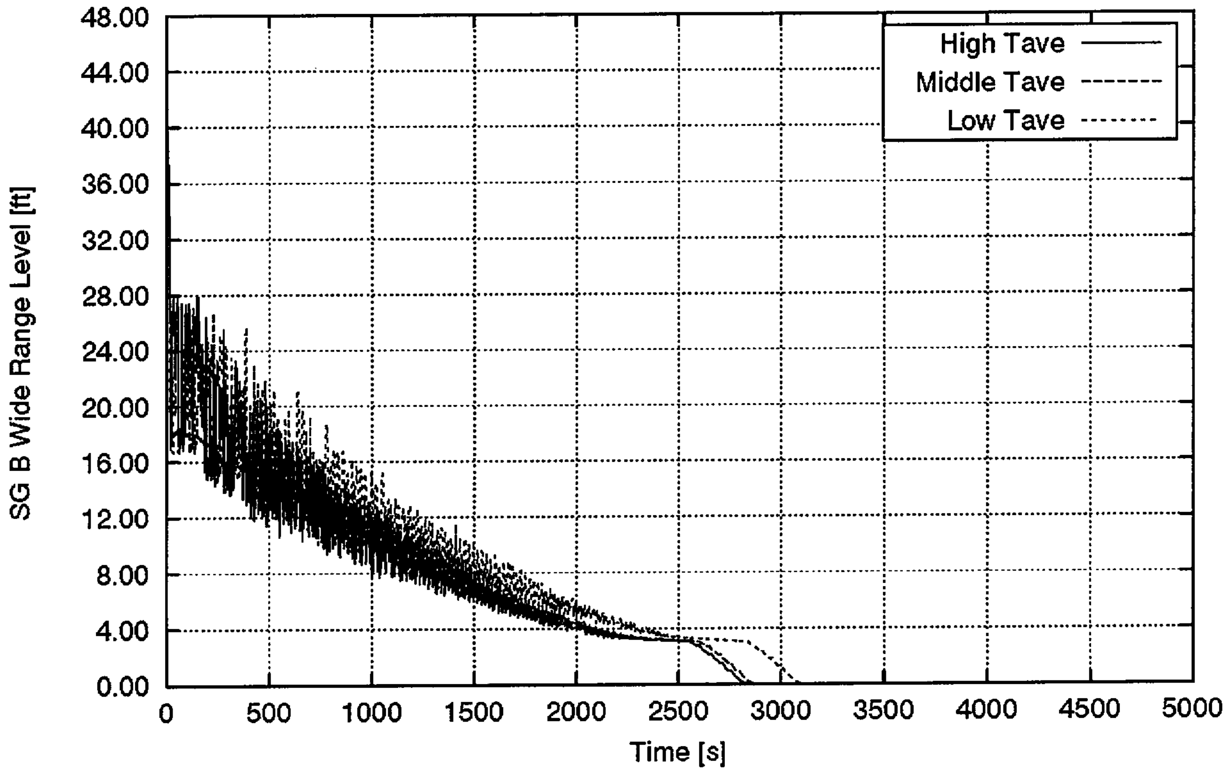


Figure 6.8.11-4
Loss of Normal Feedwater - High Pressure
SG B Wide-Range Level vs. Time

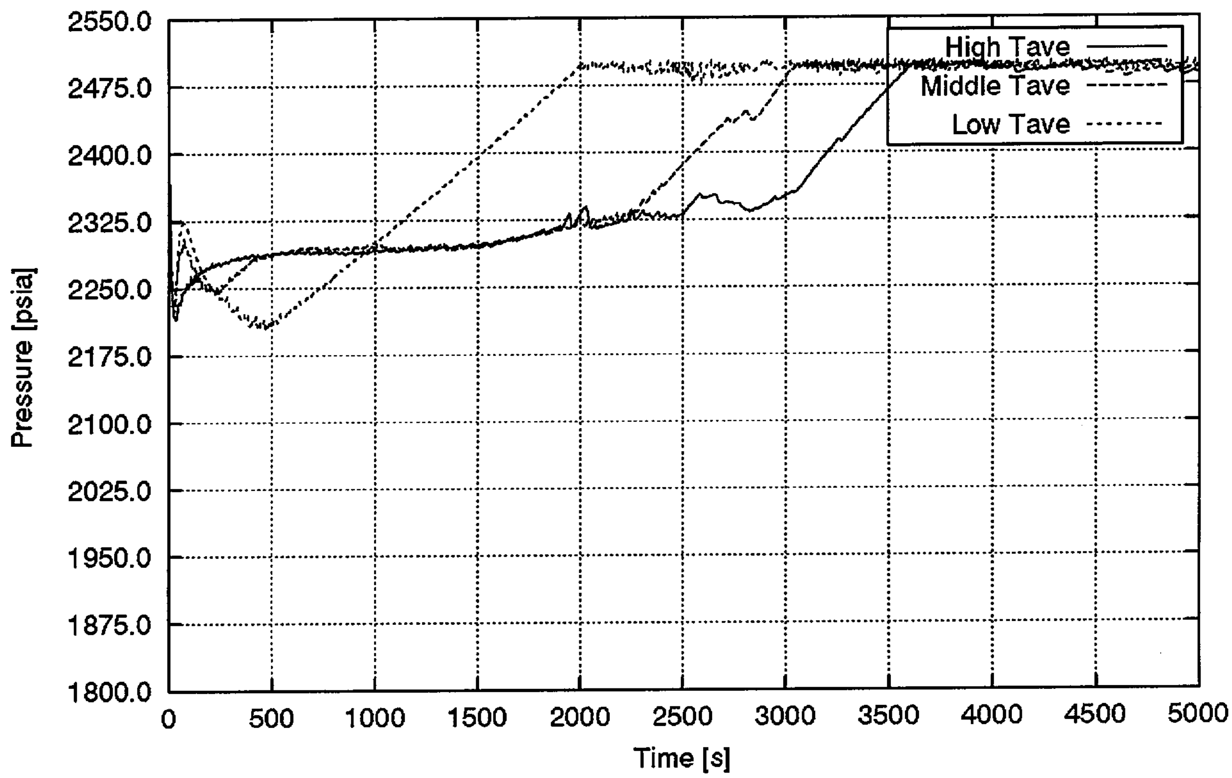


Figure 6.8.11-5
Loss of Normal Feedwater - High Pressure
Pressurizer Pressure vs. Time

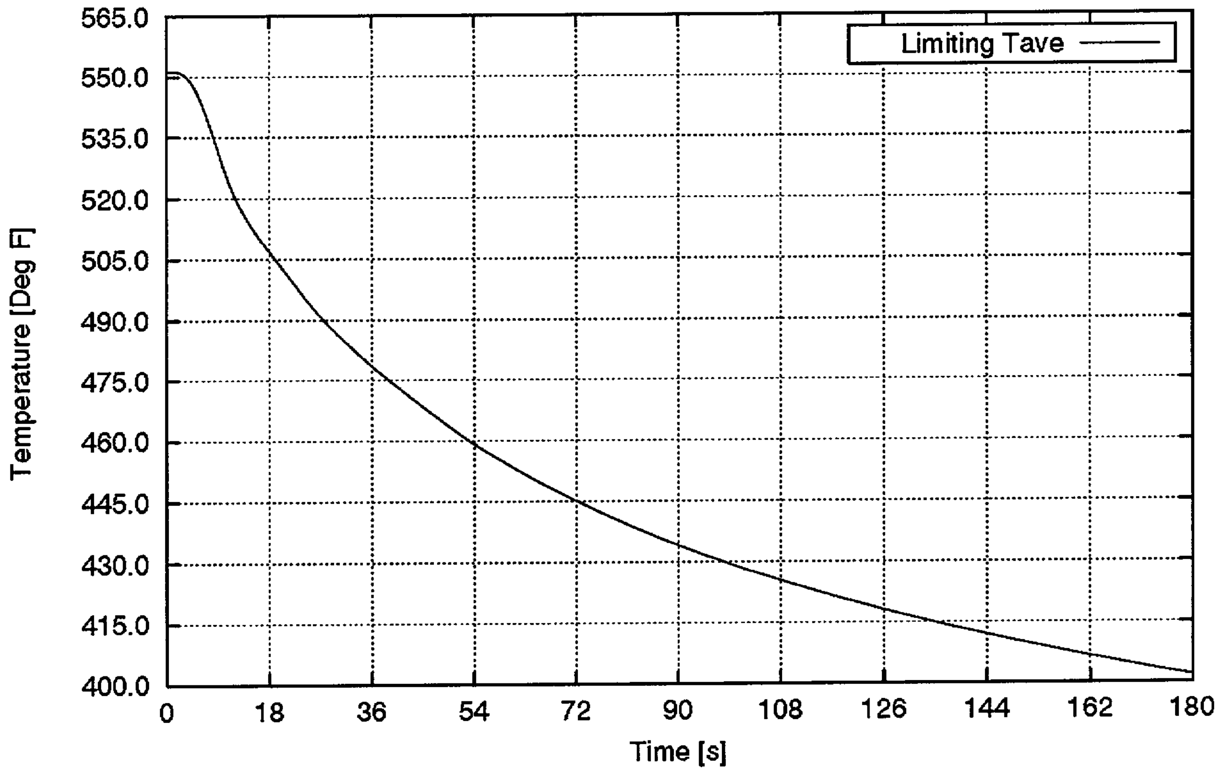


Figure 6.8.14-1
Main Steam Line Break – Limiting Core Case 14NYY0
T_{avg} vs. Time

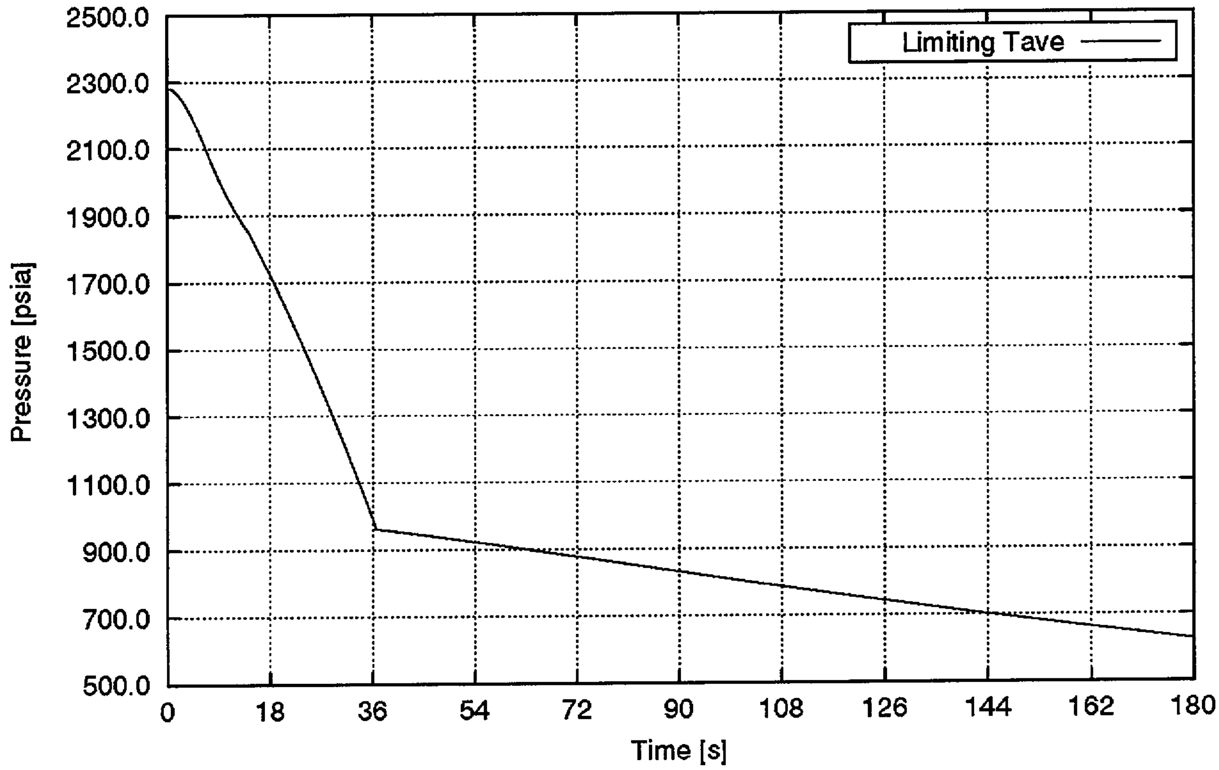


Figure 6.8.14-2
Main Steam Line Break – Limiting Core Case 14NYY0
Pressurizer Pressure vs. Time

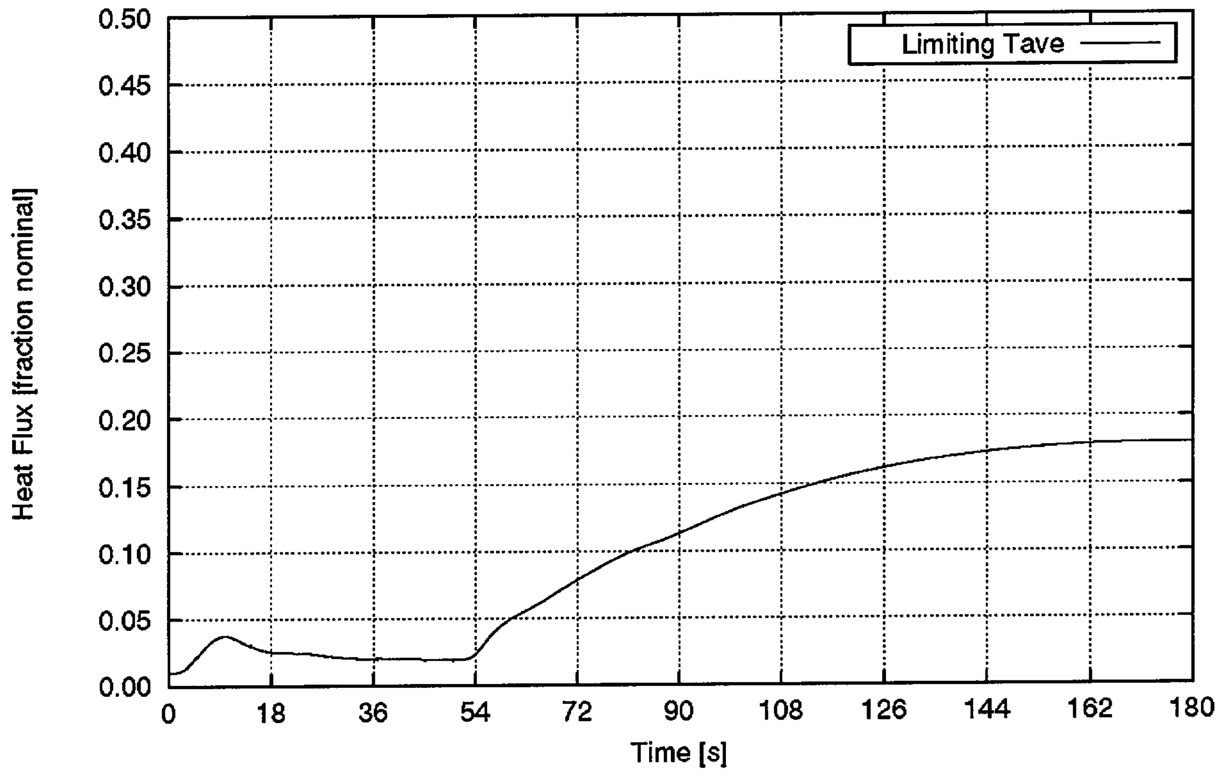


Figure 6.8.14-3
Main Steam Line Break – Limiting Core Case 14NYY0
Heat Flux vs. Time

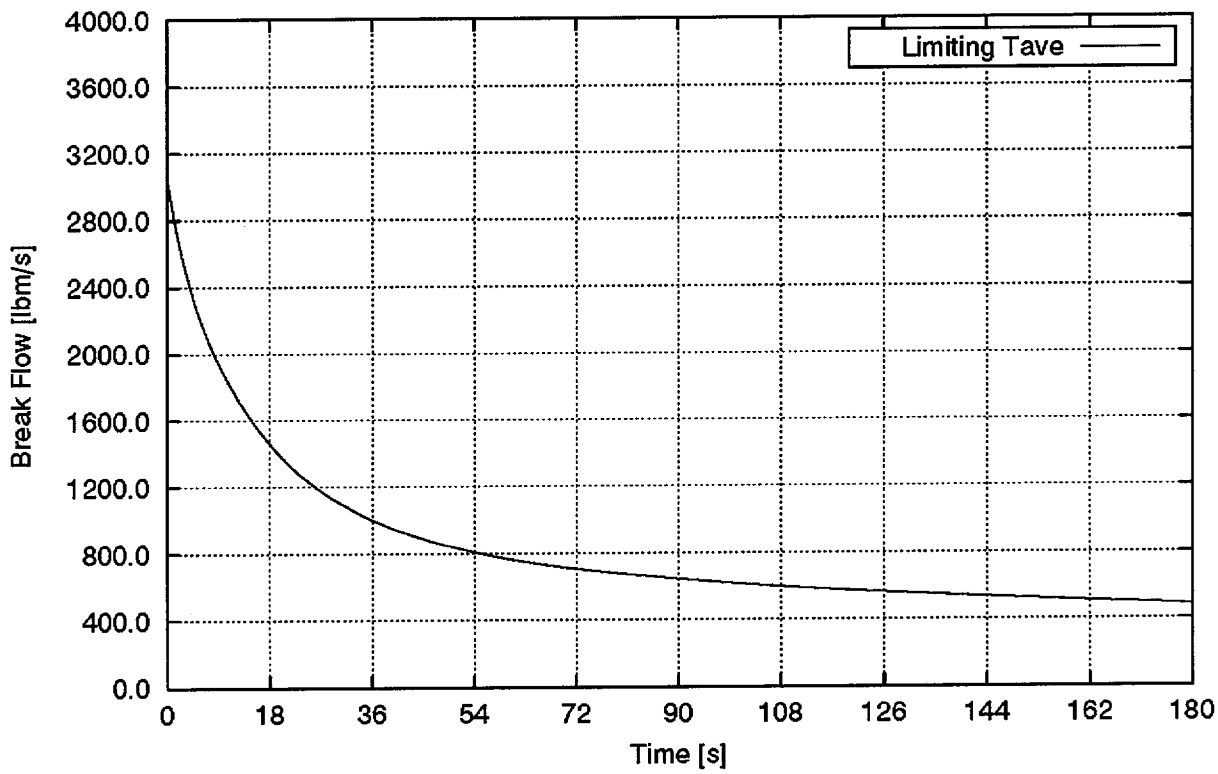


Figure 6.8.14-4
Main Steam Line Break – Limiting Core Case 14NYYo
SG B Break Flow vs. Time

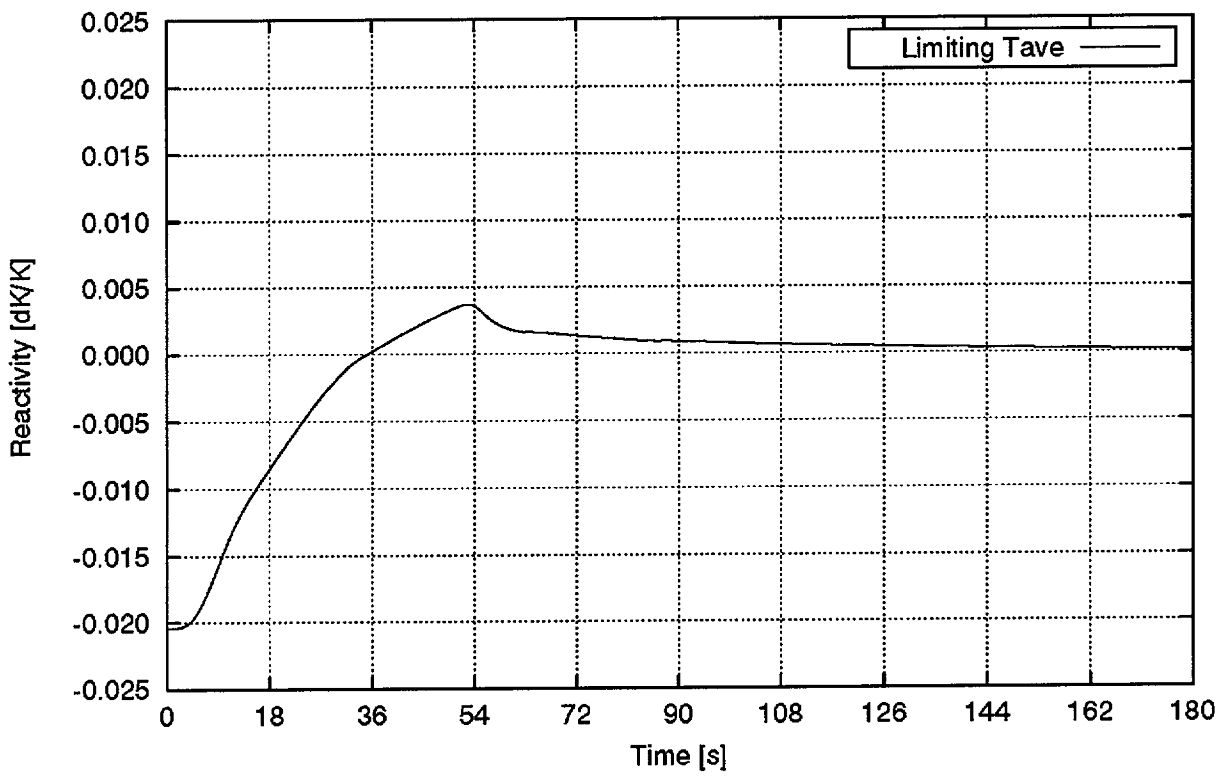


Figure 6.8.14-5
Main Steam Line Break – Limiting Core Case 14NYY0
Reactivity vs. Time

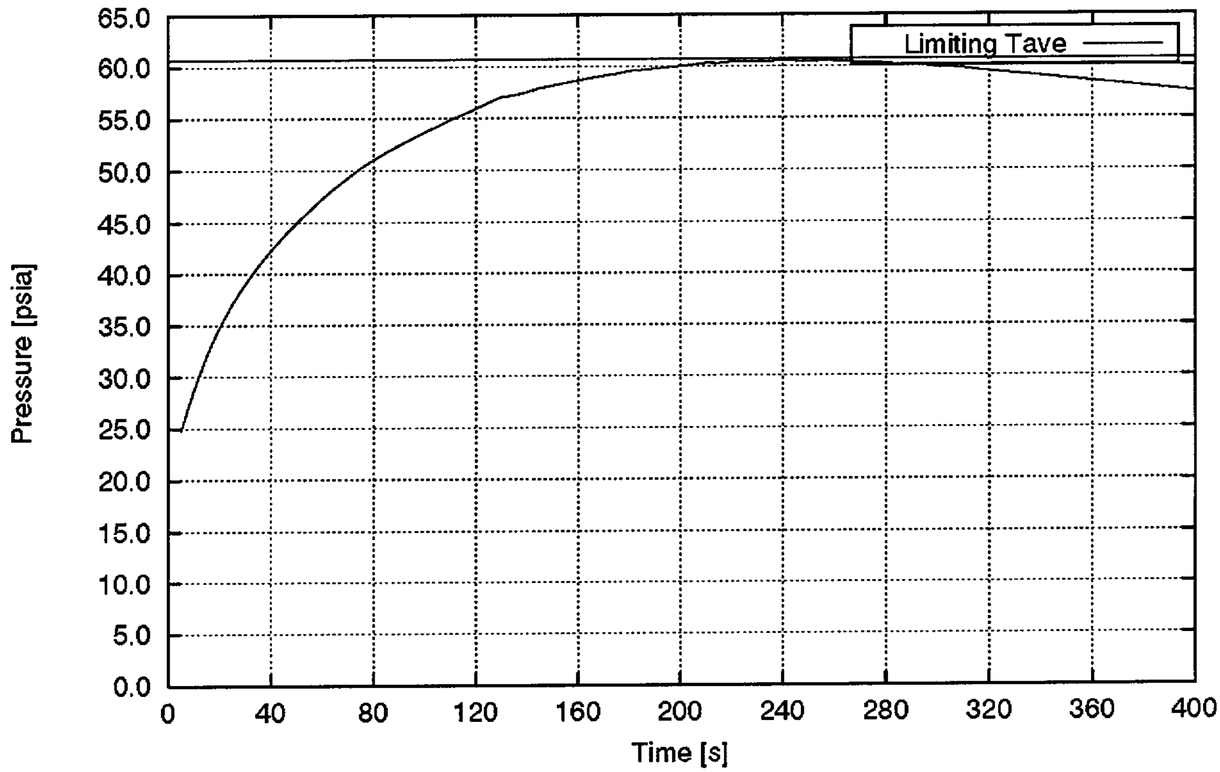


Figure 6.8.14-6
Main Steam Line Break – Limiting Containment Pressure Case 14RYY2
Containment Pressure vs. Time

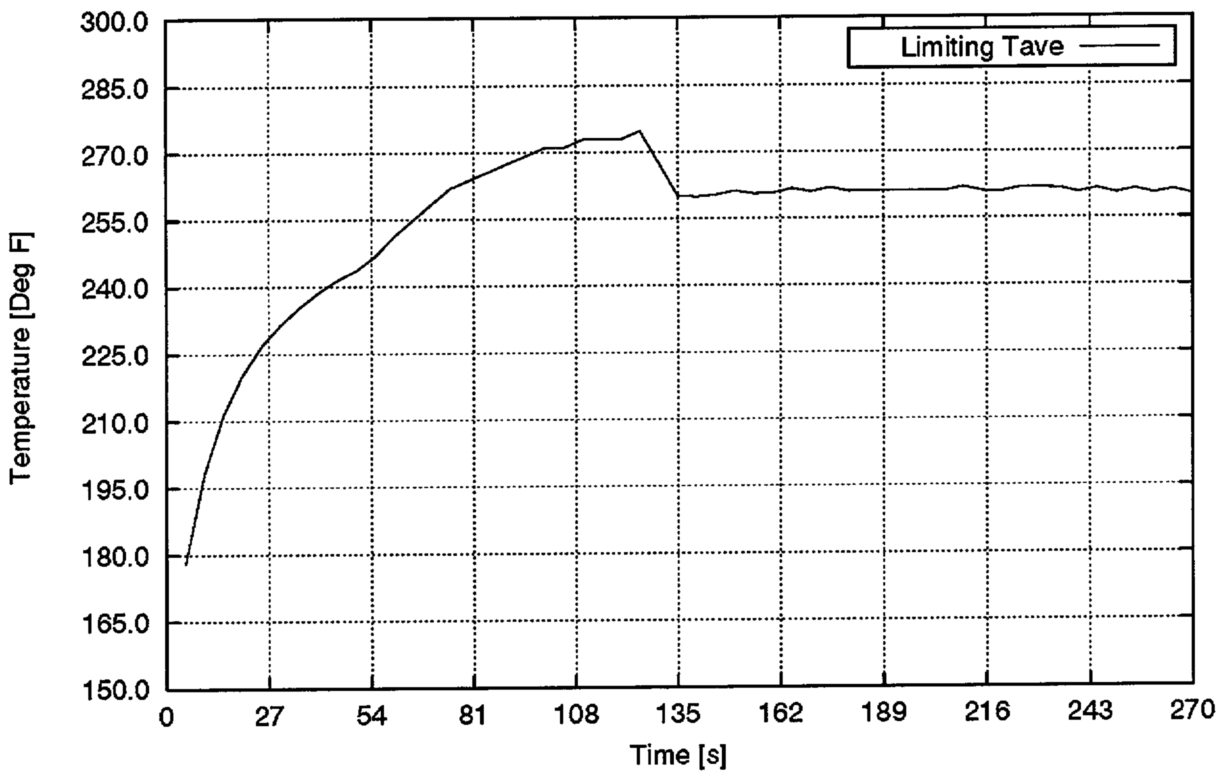


Figure 6.8.14-7
Main Steam Line Break – Limiting Containment Temperature Case 11NYY0
Containment Temperature vs. Time

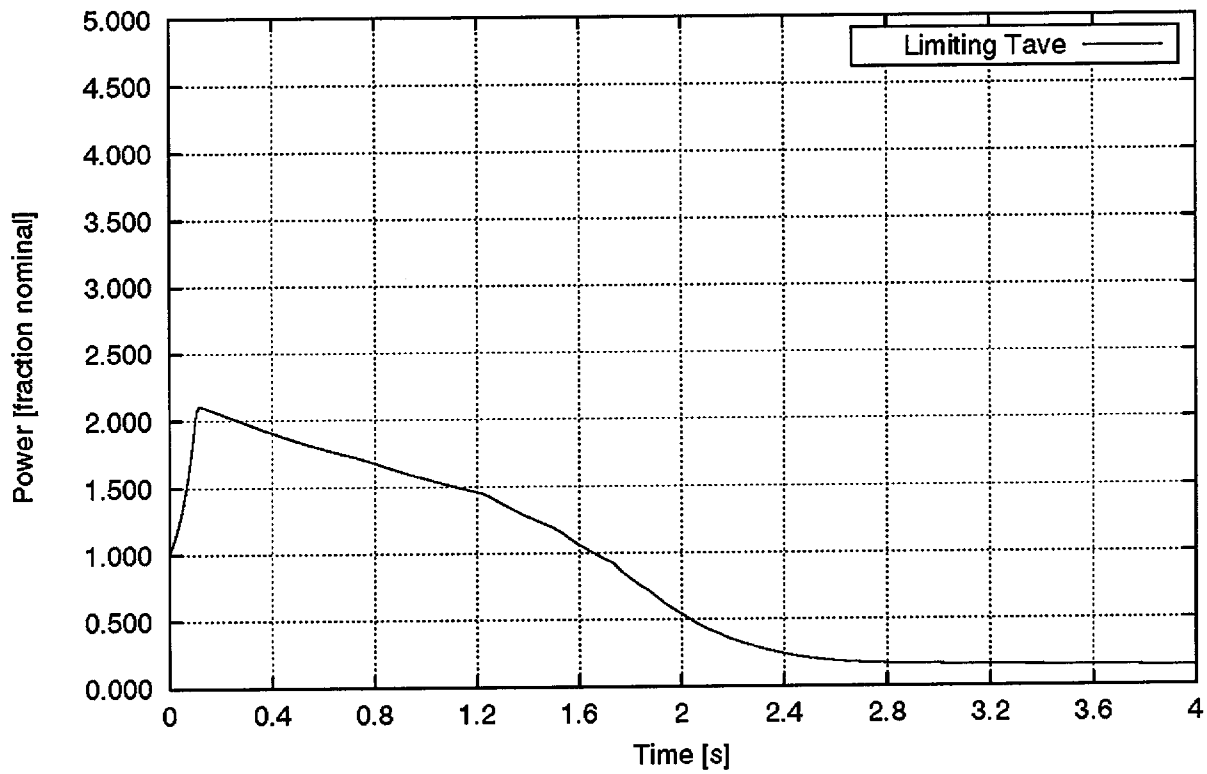


Figure 6.8.15-1
RCCA Ejection - BOC Full Power
Reactor Power vs. Time

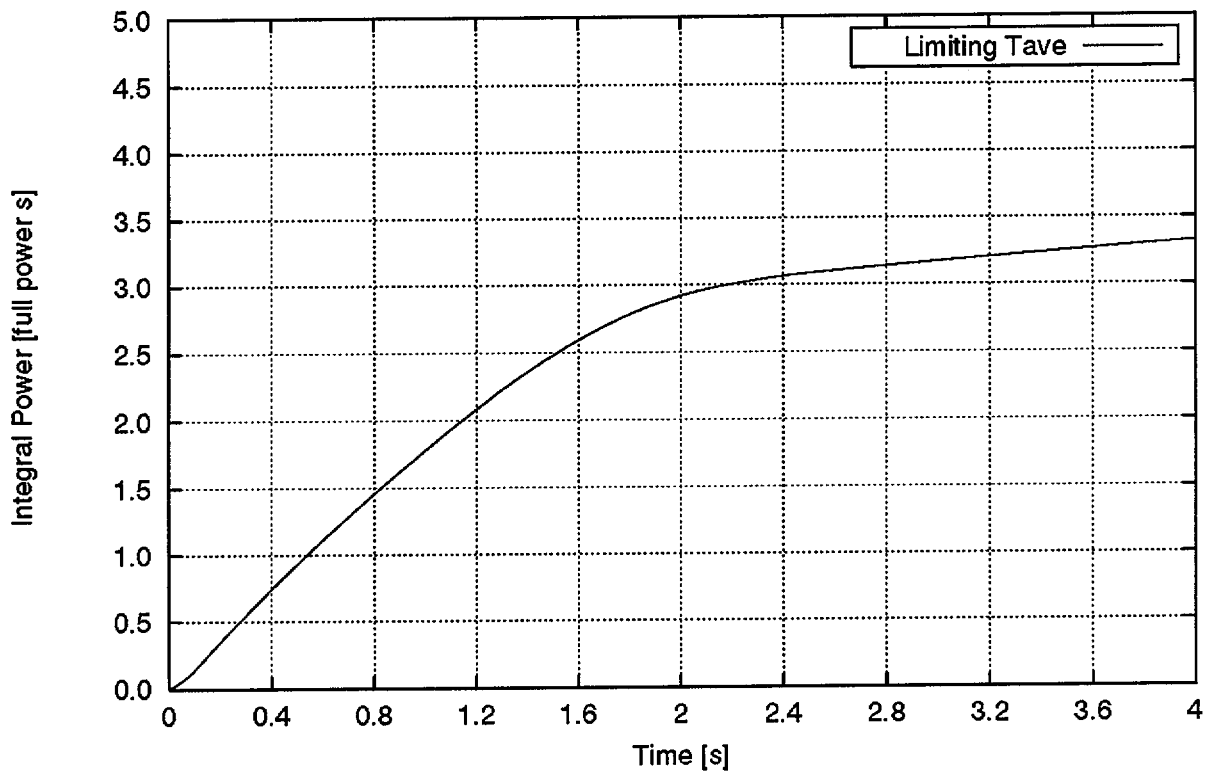


Figure 6.8.15-2
RCCA Ejection - BOC Full Power
Integral Reactor Power vs. Time

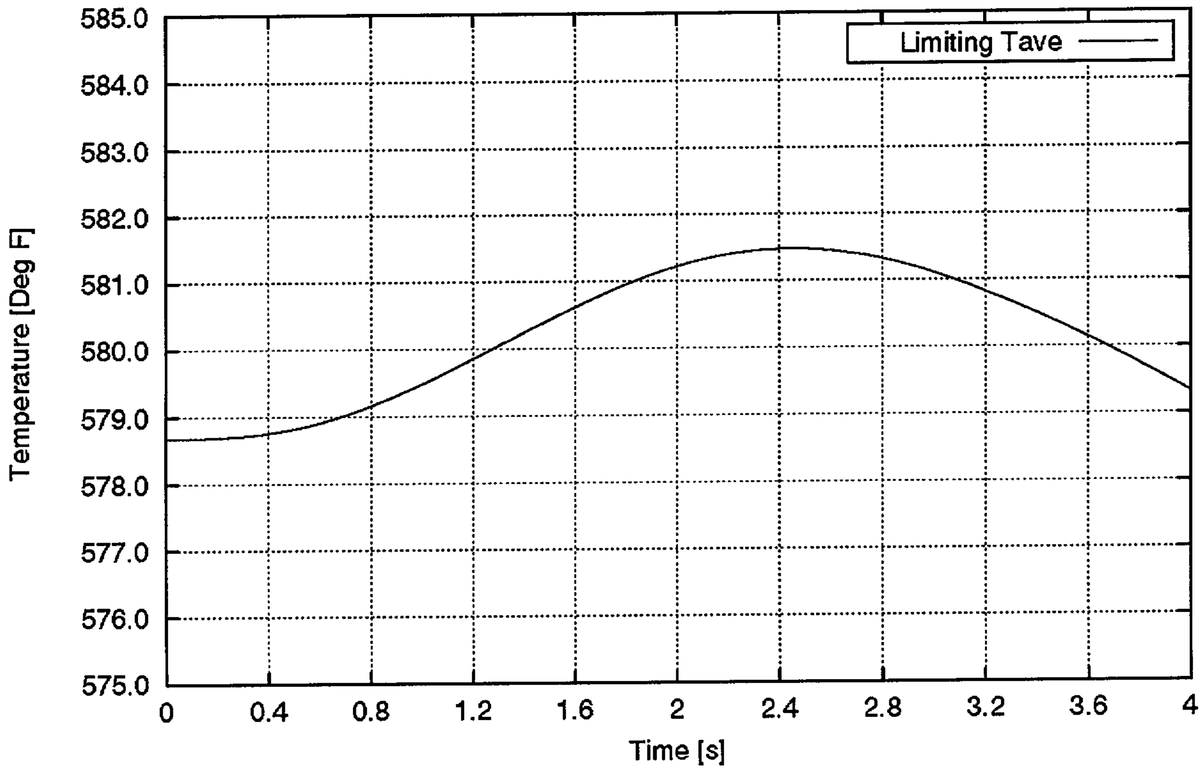


Figure 6.8.15-3
RCCA Ejection - BOC Full Power
T_{avg} vs. Time

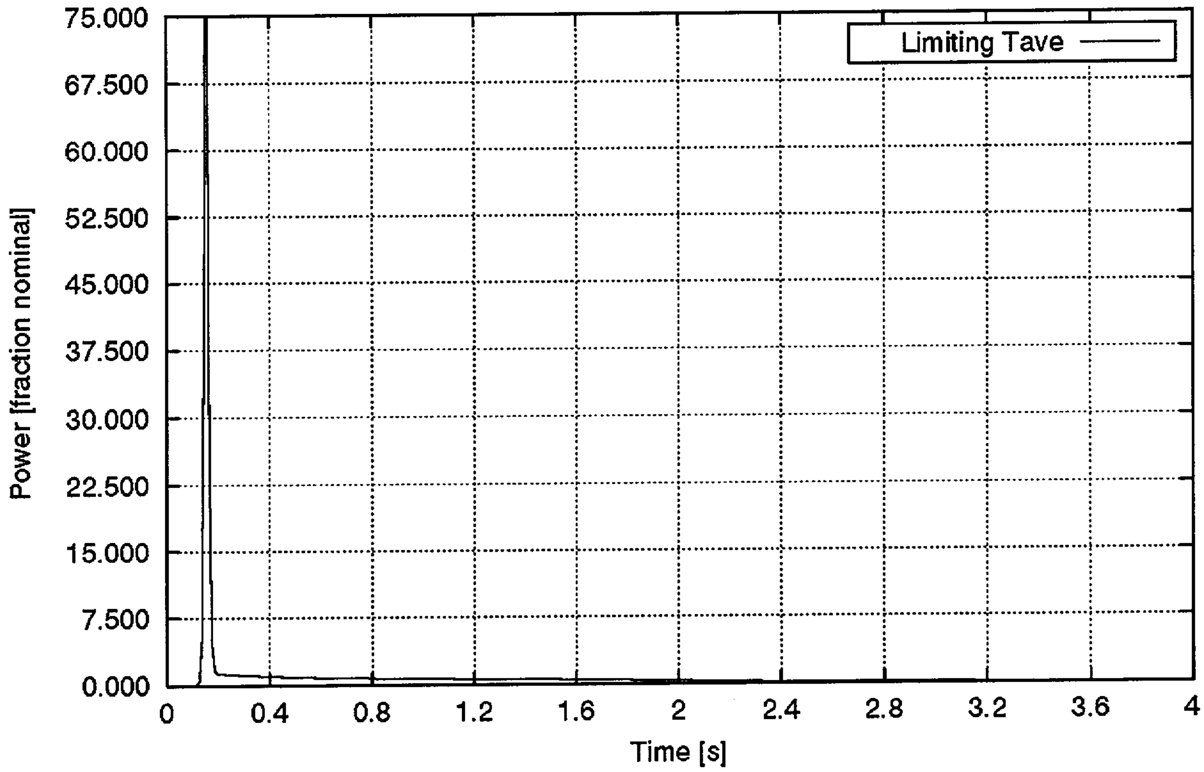


Figure 6.8.15-4
RCCA Ejection - BOC Zero Power
Reactor Power vs. Time

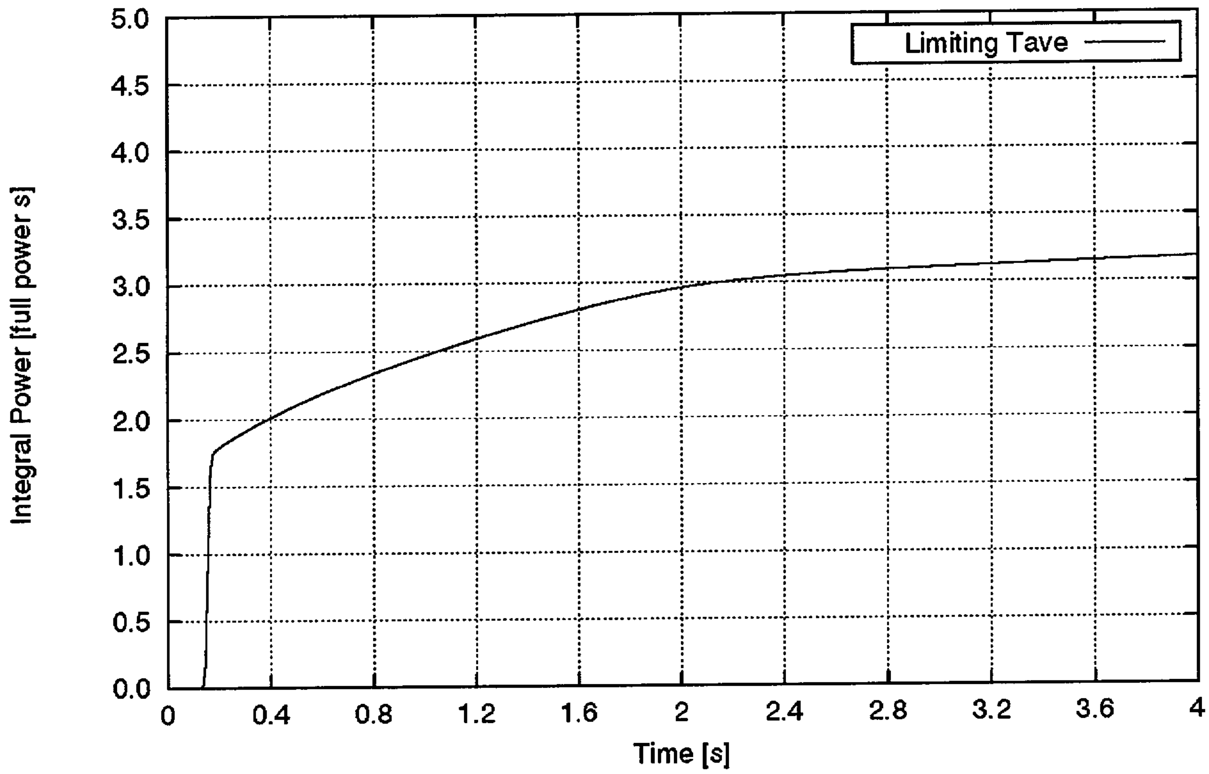


Figure 6.8.15-5
RCCA Ejection - BOC Zero Power
Integral Reactor Power vs. Time

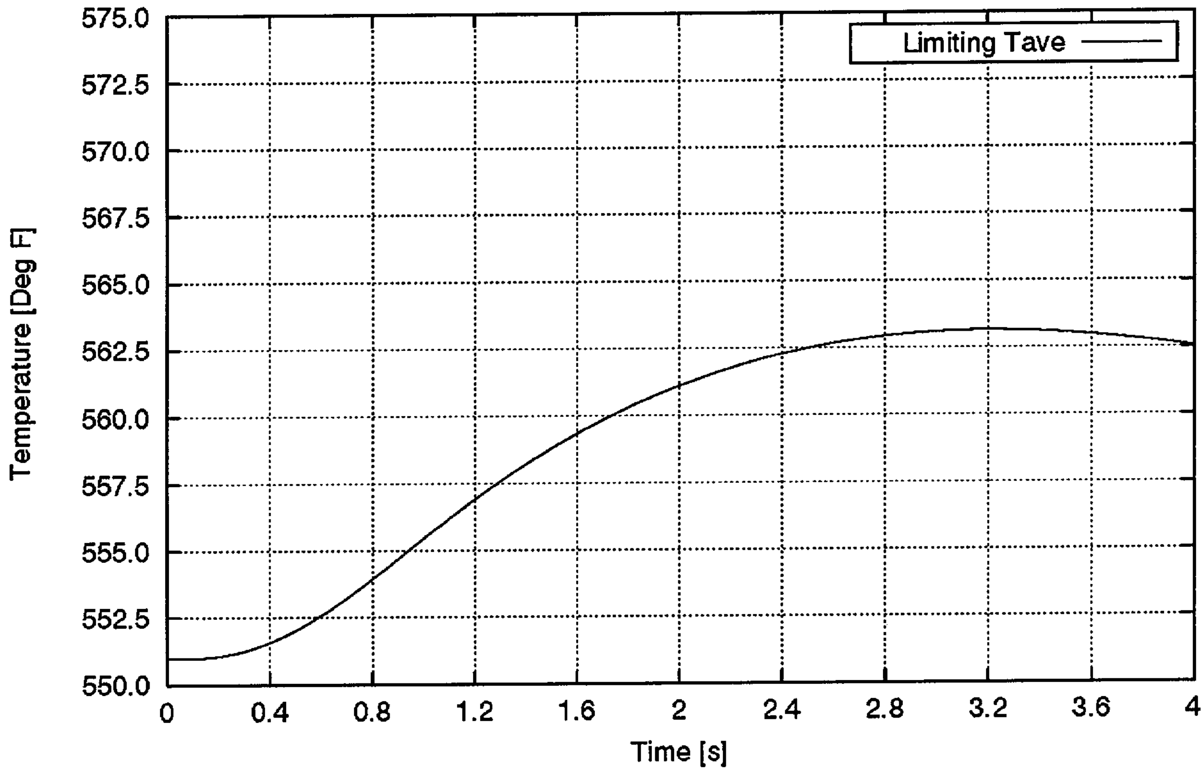


Figure 6.8.15-6
RCCA Ejection - BOC Zero Power
T_{avg} vs. Time

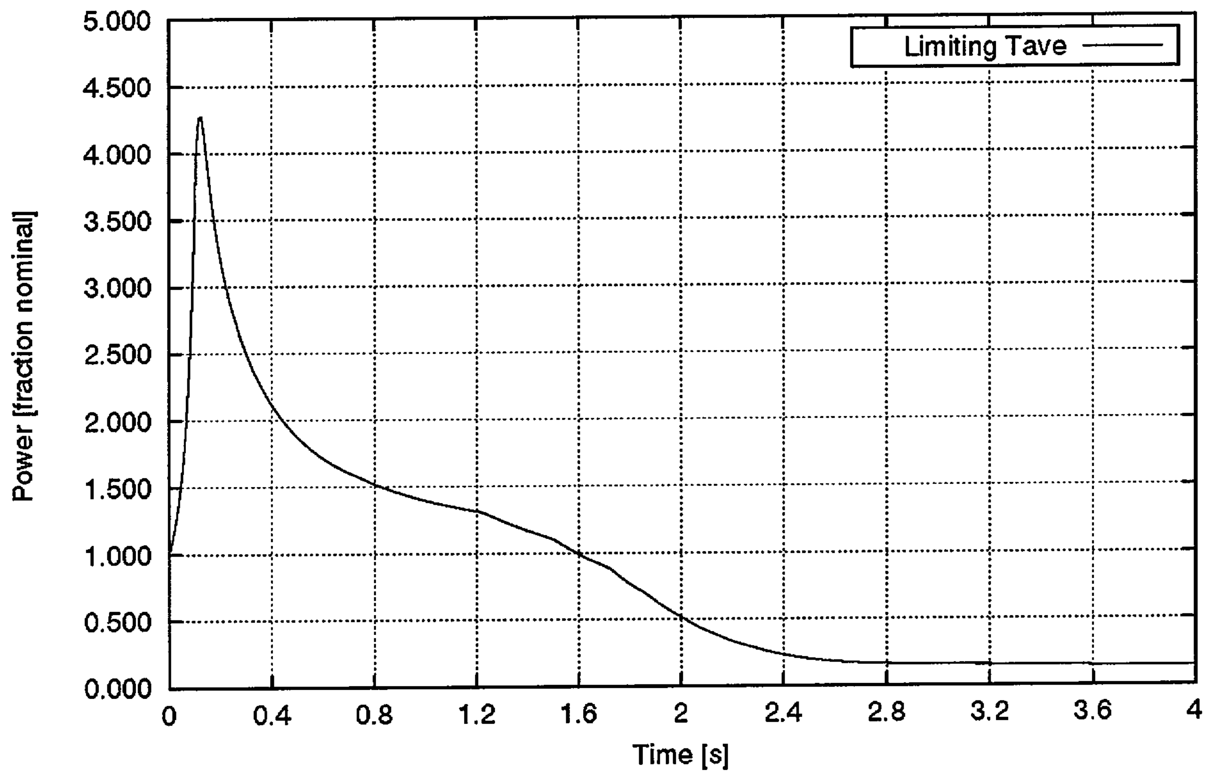


Figure 6.8.15-7
RCCA Ejection - EOC Full Power
Reactor Power vs. Time

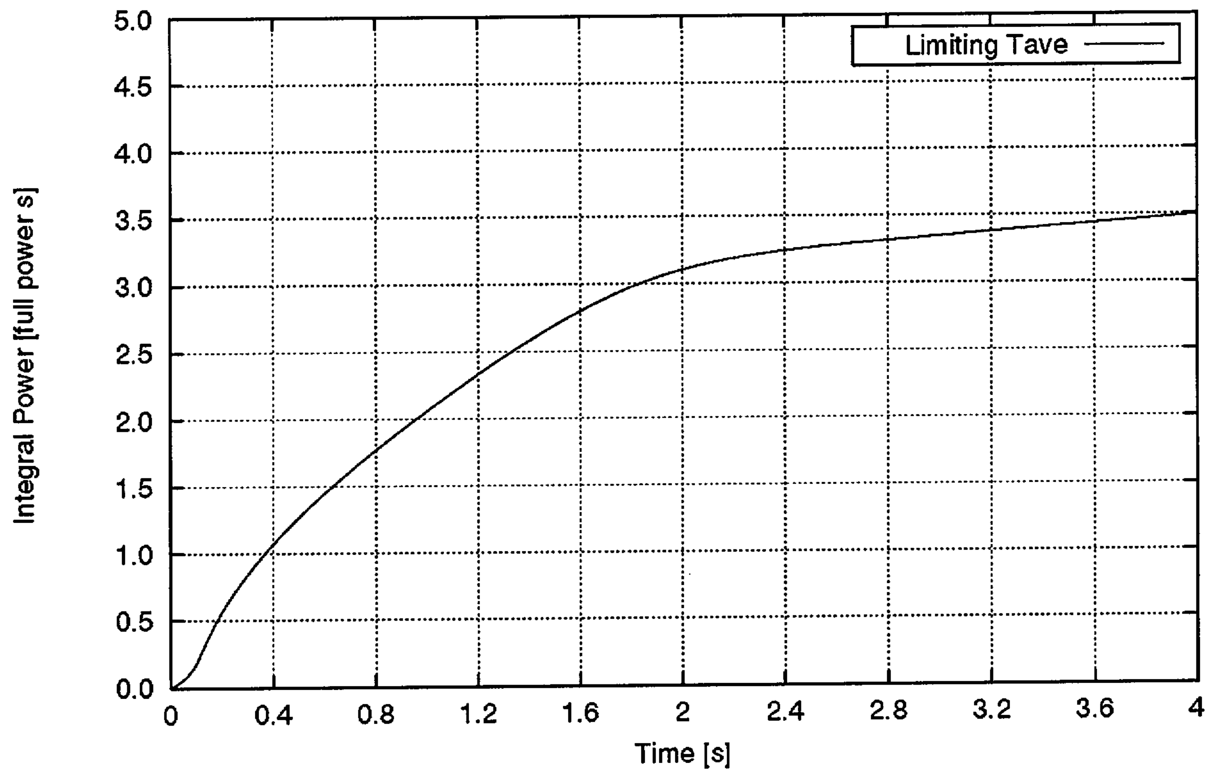


Figure 6.8.15-8
RCCA Ejection - EOC Full Power
Integral Reactor Power vs. Time

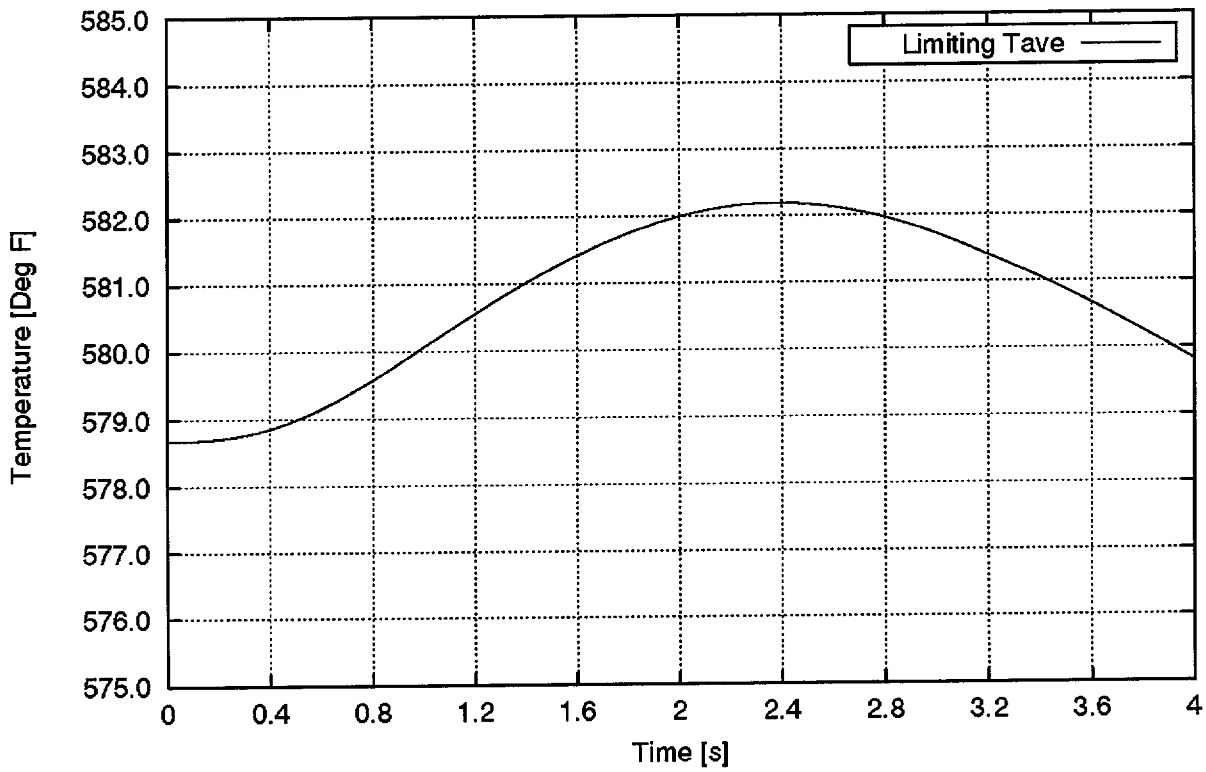


Figure 6.8.15-9
RCCA Ejection - EOC Full Power
T_{avg} vs. Time

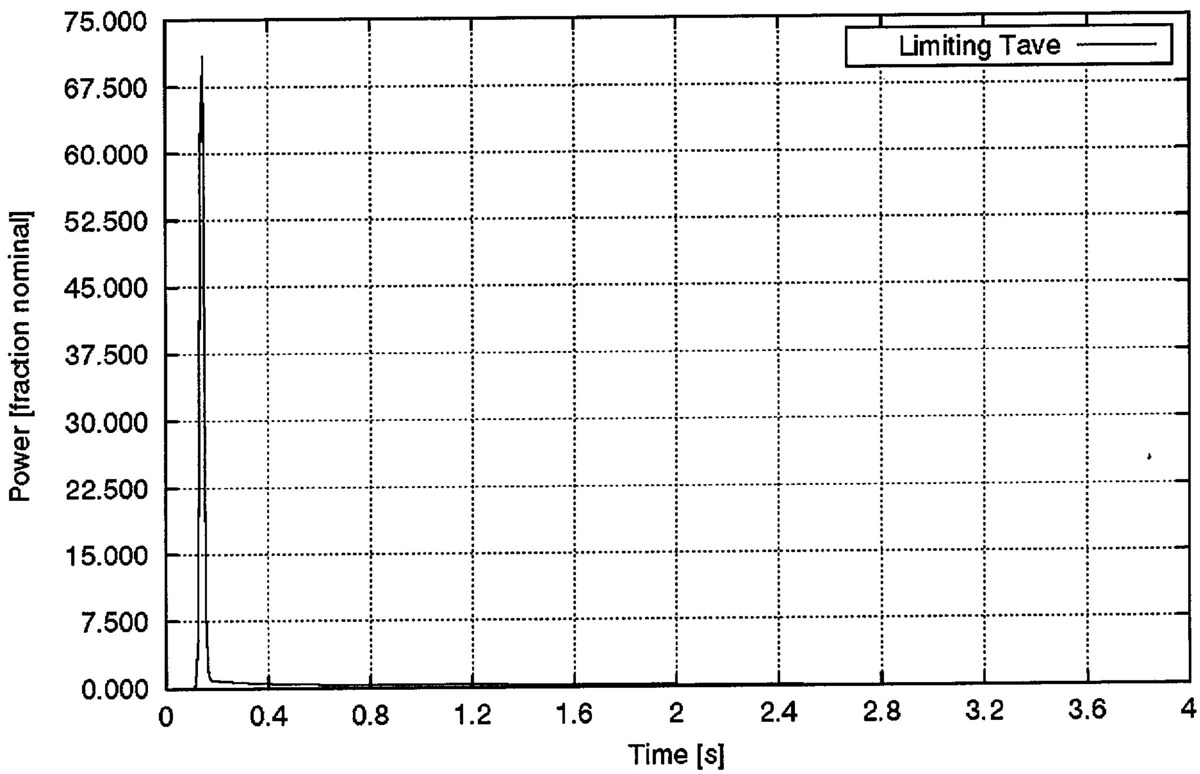


Figure 6.8.15-10
RCCA Ejection - EOC Zero Power
Reactor Power vs. Time

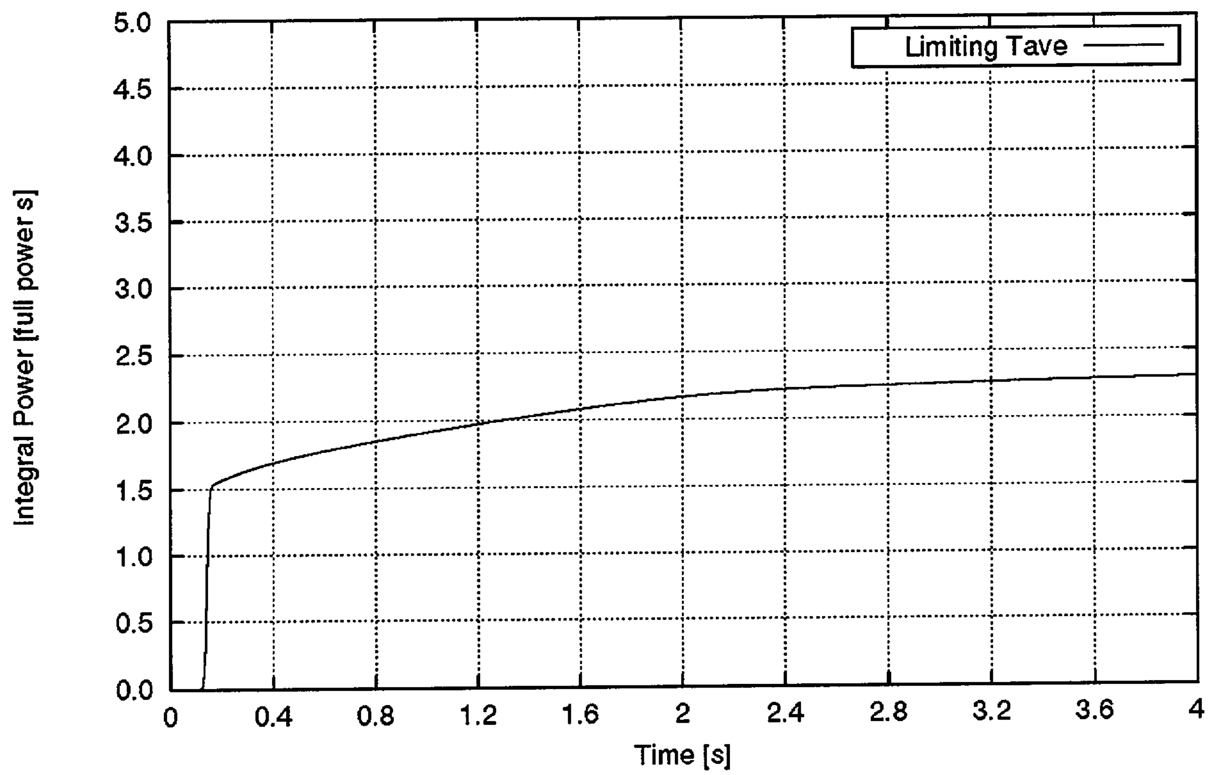


Figure 6.8.15-11
RCCA Ejection - EOC Zero Power
Integral Reactor Power vs. Time

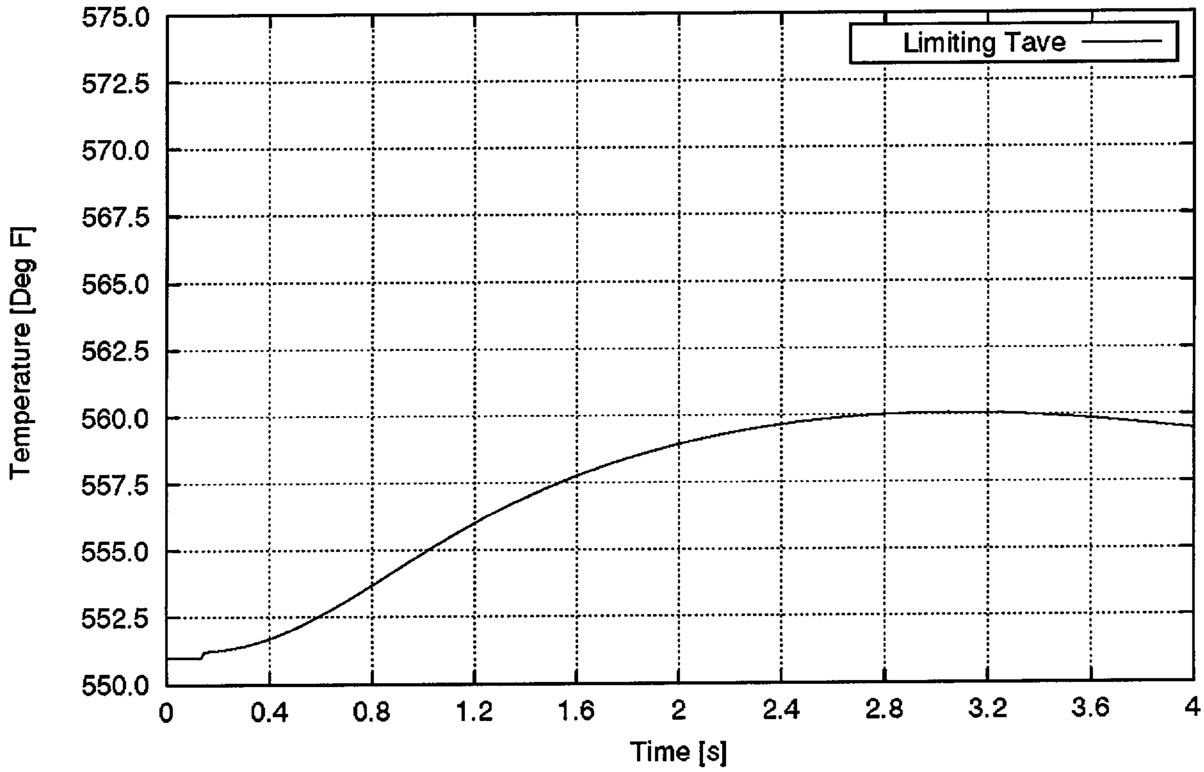


Figure 6.8.15-12
RCCA Ejection - EOC Zero Power
T_{avg} vs. Time