WOLF CREEK
NUCLEAR OPERATING CORPORATION

Forrest T. Rhodes
Vice President
Engineering & Technical Services

May 14, 1991

ET 91-0074

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Revision to Technical Specification 3.1.3.4 - Control Rod Drop Time

Gentlemen:

The purpose of this letter is to transmit an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS), Unit No. 1. This proposed license amendment revises Technical Specification 3.1.3.4 to increase the maximum allowed control rod drop time from 2.2 to 2.7 seconds.

This proposed change is needed to support the planned use of Westinghouse VANTAGE 5H fuel at WCGS. The control rod guide thimble diameter incorporated in the VANTAGE 5H fuel design is slightly smaller than in the fuel design currently used at WCGS. This reduced thimble diameter is expected to result in a slight increase in control rod drop time which is accommodated by this amendment request. The Westinghouse VANTAGE 5H fuel design has previously been reviewed and approved by the NRC on a generic basis and this proposed revision is similar to amendments approved for other facilities.

Attachments I through III provide the Safety Evaluation, Significant Hazards Consideration Determination, and Environmental Impact Determination supporting the requested change. Attachment IV provides the revised technical specification page.

In accordance with 10 CFR 50.91, a copy of this application, with attachments is being provided to the designated Kansas State Official.

A001
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P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831

This proposed license amendment is needed to support loading of Westinghouse VANTAGE 5H fuel during the upcoming refueling outage. Therefore, Wolf Creek Nuclear Operating Corporation requests approval of this proposed amendment prior to the fifth WCGS refueling outage which is currently scheduled to begin in September of 1991.

If you have any questions concerning this matter, please contact me or Mr. H. K. Chernoff of my staff.

Very truly yours,

Forrest T. Rhodes
Vice President
Engineering & Technical Services

FTR/jra

Attachments: I - Safety Evaluation
II - Significant Hazards Consideration Determination
III - Environmental Impact Determination
IV - Proposed Technical Specification Change

cc: G. W. Allen (KDHE), w/a
L. L. Gundrum (NRC), w/a
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R. D. Martin (NRC), w/a
D. V. Pickett (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Forrest T. Rhodes, of lawful age, b[illegible]orn upon oath says that he is Vice President Engine[illegible]ervices of Wolf Creek Nuclear Operating Corporation; th[illegible]oregoing document and knows the content thereof; that h[illegible]me for and on behalf of said Corporation with full powe[illegible]o; and that the facts therein stated are true a[illegible] of his knowledge, information and belief.

By Forrest T. Rhodes
Forrest T. Rhodes
Vice President
Engineering & Technical Services

SUBSCRIBED and sworn to before me this 14 day of May, 1991.

Marline Heachman
Notary Public

MARLINE HEATHMAN
NOTARY PUBLIC
STATE OF KANSAS

Expiration Date August 4, 1994

ATTACHMENT I

SAFETY EVALUATION

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SAFETY EVALUATION

1.0 INTRODUCTION

Wolf Creek Nuclear Operating Corporation (WCNOC) is proposing a change from the current technical specification maximum Rod Cluster Control Assembly (RCCA) drop time to the dashpot of 2.2 seconds to a new value of 2.7 seconds. The purpose of this proposed change is to support the planned use of Westinghouse VANTAGE 5H fuel assemblies at Wolf Creek Generating Station (WCGS). Westinghouse VANTAGE 5H fuel is an evolution of previous Westinghouse fuel designs employing a low pressure drop Zircaloy grid. This fuel design has previously been submitted to the NRC[1,2].

The VANTAGE 5H fuel assembly control rod guide thimbles have an inside diameter of 0.442 inches while the current fuel design has an analogous dimension of 0.450 inches. The increased hydraulic resistance of the smaller thimble diameter is expected to result in a slight increase in the RCCA drop time. To account for this effect, the safety analyses will use an increased RCCA drop time. The increased RCCA drop time value of 2.7 seconds was chosen to reasonably bound the expected increase without requiring extensive safety analysis.

2.0 EVALUATION METHODOLOGY AND SUMMARY OF RESULTS

Loss-of-Coolant Accident (LOCA) and non-LOCA safety analyses and evaluations confirm the acceptability of a 0.5 second increase in the RCCA drop time to the dashpot to 2.7 seconds.

For non-LOCA transients, transient specific safety evaluations confirm that minimum Departure from Nucleate Boiling Ratios (DNBRs) meet the DNBR safety limit and there is no increase in fuel rod failures. Evaluations and limited sensitivity studies support the conclusion that all other applicable safety analysis acceptance criteria continue to be met.

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[1] Davidson, S. L. ed. et al., "VANTAGE 5 Fuel Assembly Reference Core Report", WCAP-10444-P-A, September 1985

[2] Davidson, S. L. ed. et al., "VANTAGE 5H Fuel Assembly", WCAP-10444-P-A, Addendum 2A, February 1989

For the WCGS licensing basis LOCA analyses, evaluations were performed considering the LOCA hydraulic forces, large break LOCA, small break LOCA, hot leg switchover, and long term cooling calculations.

All LOCA and non-LOCA safety analysis conclusions remain valid for this proposed change. In all cases, the effect did not result in any design or regulatory limit being exceeded. Therefore, based upon the evaluations described above, it is concluded that there is no significant impact on the safety analysis due to this proposed technical specification change and the conclusions presented in the WCGS Updated Safety Analysis Report (USAR) remain valid.

It should be noted that WCNOC has previously submitted a license amendment request which proposed a slight (approximately 2%) reduction in the required Reactor Coolant System (RCS) thermal design flow[3]. This amendment request is currently under review by the Nuclear Regulatory Commission staff. In conjunction with the proposed reduction in RCS thermal design flow, the safety analysis limit DNBR was increased from 1.30 to 1.32. The analyses and evaluations for the proposed increase in RCCA drop time are consistent with the reduced RCS thermal design flow and more conservative safety analysis limit DNBR included in the previous amendment request.

3.0 EVALUATION

3.1 Non-LOCA Accidents

This section summarizes the reanalysis and evaluations performed in support of the proposed technical specification change to increase the RCCA drop time to 2.7 seconds. This longer drop time affects the results of fast non-LOCA limiting transients such as partial and complete loss of flow, RCCA withdrawal from subcritical, locked reactor coolant pump rotor and RCCA ejection. These fast non-LOCA transients were reanalyzed.

Non-LOCA events not mentioned above were not reanalyzed for the increased rod insertion time for one or more of the following reasons:

1) The transient results are insensitive to the rod drop time.

2) A reactor trip was not assumed or explicitly modeled in the analysis.

3) The reactor trip has no effect on the minimum or maximum value of the critical parameter of interest.

4) The event may be impacted, but the magnitude of the impact is small when compared to the margin to the design basis limit.

[3]Letter ET 91-0042, dated March 5, 1991, from F. T. Rhodes to the U. S. Nuclear Regulatory Commission

The following pages contain a summary of the results for transients that were reanalyzed and an evaluation of the impact of the increased RCCA drop time on all other non-LOCA events. The discussion of events is arranged in the order that they appear in the WCGS USAR.

3.1.1 Feedwater System Malfunctions (USAR 15.1.1 & 15.1.2)

This event is analyzed to show that the DNB design basis is met. For the limiting case, the minimum DNBR occurs immediately after the turbine is tripped. A reactor trip occurs from the turbine trip signal, although the reactor is tripped after the minimum DNBR has occurred. Therefore, an increase in the RCCA drop time will not have any impact on the minimum DNBR.

3.1.2 Excessive Increase in Secondary Steam Flow (USAR 15.1.3)

This event is analyzed to show that the DNB design basis is met following a step load increase from full power. Cases are analyzed at Beginning-of-Life (BOL) and End-of-Life (EOL) conditions with and without rod control. In all cases analyzed, the reactor stabilizes without a reactor trip. Therefore, the increased control rod drop time will have no effect on this event.

3.1.3 Inadvertent Opening of a Steam Generator Relief or Safety Valve (USAR 15.1.4) and Steam System Piping Failure (USAR 15.1.5)

The inadvertent opening of a steam generator relief or safety valve case is analyzed to show that the DNB design basis is met. The steam system piping failure cases are analyzed to show that the core remains intact and in place and that the radiation doses do not exceed the guidelines of 10 CFR 100. This is demonstrated by showing that the DNB design basis is met, even though DNB and possible clad perforation are not necessarily unacceptable for the piping failure cases.

The limiting steamline depressurization and steamline break cases are analyzed from hot shutdown initial conditions. The transient is started assuming the reactor is tripped and the core is at the minimum design shutdown margin. Therefore the 0.5 second increase in the rod insertion time from 2.2 to 2.7 seconds will have no effect on the results of these analyses.

3.1.4 Loss of External Electrical Load/Turbine Trip (USAR 15.2.2 & 15.2.3)

This event is analyzed to show that the DNB design basis is met and that primary and secondary side system pressures do not exceed 110% of design values. Four cases are analyzed for BOL and EOL, with and without automatic pressurizer pressure control.

The increased RCCA drop time will not result in system pressures exceeding 110% of design values. Pressure transients from the current analysis of record were evaluated based on an RCCA drop time sensitivity study. In all cases, there was ample margin to account for the slight increase in peak primary and secondary side pressures due to the slower drop times.

The increased rod drop time will not result in a DNBR below the design limit. The DNBR transients for the BOL without pressure control and both EOL cases never fall below the initial value. During the limiting case for DNB, BOL with pressure control, the DNBR initially rises and then decreases slightly with the minimum DNBR occurring shortly after the reactor trips. Based on sensitivity studies of this limiting case, the decrease in minimum DNBR is very small when the RCCA drop time is increased by 0.5 seconds. The DNBR remains above the limit value.

3.1.5 Loss of Non-emergency AC Power to the Station Auxiliaries (USAR 15.2.6)

This event is analyzed to show that adequate heat removal capability exists via natural circulation flow, as aided by the Auxiliary Feedwater System, to remove core decay heat and stored energy following reactor trip. This is demonstrated by ensuring that the RCS heatup is terminated prior to the time when coolant expansion causes the pressurizer to become filled with water. The RCS volumetric expansion is not affected since the total RCS flow and vessel outlet temperature remain the same. This transient is a slow long-term heatup event and this aspect of the transient is not sensitive to the rate at which the rods are inserted during a reactor trip. With respect to DNB criterion, this event is bounded by the Complete Loss of Forced Reactor Coolant Flow analysis which was reanalyzed and shown to be acceptable.

3.1.6 Loss of Normal Feedwater Flow (USAR 15.2.7)

This event is analyzed to show that adequate heat removal capability exists via the Auxiliary Feedwater System to remove core decay heat, stored energy and RCS pump heat following reactor trip. This is demonstrated by ensuring that the RCS heatup is terminated prior to the time when coolant expansion causes the pressurizer to become filled with water. The loss of feedwater transient is a slow long-term heatup event and is not sensitive to the rate at which the control rods are inserted following a reactor trip.

3.1.7 Major Rupture of a Main Feedwater Pipe (USAR 15.2.8)

This event is analyzed to show that adequate heat removal capability exists using the auxiliary feedwater system to remove core decay heat, stored energy and RCS pump heat following reactor trip. This is demonstrated by ensuring that the RCS heatup is terminated prior to the time at which the hot legs become saturated and the peak primary and secondary pressures exceed 110% of design values.

The feedline break is a long-term heatup event and is not sensitive to the rate at which the RCCAs are inserted following a reactor trip. The heatup transient continues for many minutes following the reactor trip. The 0.5 second increase in RCCA insertion time will result in an insignificant increase in the integrated heat produced by the core during the transient. No significant increase in hot leg temperature or system pressures would occur due to the increase in RCCA insertion time.

3.1.8 Partial and Complete Loss of Forced Reactor Coolant Flow (USAR 15.3.1 and 15.3.2)

The loss of flow transients are characterized by a rapid decrease in core flow. If the reactor is not tripped promptly, DNB may occur with subsequent fuel damage. Thus these transients can be sensitive to the RCCA drop time. The partial (2/4) and complete (4/4) loss of flow transients were reanalyzed with a RCCA drop time of 2.7 seconds. Figures 3.1-1 through 3.1-3 show the results of the partial loss of flow. The results of the complete loss of flow transient are shown in Figures 3.1-4 through 3.1-6. In both cases, the minimum DNBR remains above the limit.

3.1.9 Reactor Coolant Pump Shaft Seizure/Break (USAR 15.3.3 & 15.3.4)

The reactor coolant pump shaft seizure accident is postulated as an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected coolant pump is rapidly reduced leading to an initiation of a reactor trip on a low flow signal. If the reactor is not tripped promptly, the clad temperature may exceed the limit value of 2700°F and RCS pressure may increase above that which would cause stresses to exceed the faulted condition stress limits. This transient can be very sensitive to the RCCA drop time.

The locked rotor transient was reanalyzed with a RCCA drop time of 2.7 seconds. The results of the analysis are shown in Figures 3.1-7 through 3.1-10 and Table 3.1-1. Since the peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered. Since the peak clad surface temperature calculated for the hot spot remains less than 2700°F and the amount of zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability. The number of rods in DNB was also confirmed to remain below the 5% which has been used in the offsite dose calculation.

As discussed in the USAR, the results of a reactor coolant pump shaft break would be no worse than those calculated for the locked rotor event.

3.1.10 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (USAR 15.4.1)

The RCCA withdrawal from subcritical is characterized by a rapid power increase. The power excursion is retarded by Doppler feedback and the transient is terminated by a reactor trip. Due to the high rate at which the power increases, this transient can be sensitive to the RCCA drop time.

The RCCA withdrawal from subcritical event was reanalyzed with a RCCA drop time of 2.7 seconds. Transient results are shown in Figures 3.1-11 through 3.1-14. Figures 3.1-11 shows the neutron flux transient. The neutron flux overshoots the full power nominal value, but this occurs for only a very short time period. The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 3.1-12. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling in the core at all times. Figures 3.1-13 and 3.1-14 show the response of the average fuel and cladding temperature. The minimum DNBR remains above the limits at all times.

The core and the RCS are not adversely affected, since the combination of thermal power and the coolant temperature results in a minimum DNBR above the limiting value. Thus no fuel or clad damage will occur.

3.1.11 Uncontrolled RCCA Bank Withdrawal at Power (USAR 15.4.2)

This event is analyzed to show that the DNB design basis is met. The Overtemperature Delta-T (OTΔT) setpoints are not changed, so the time of reactor trip remains the same. The transient minimum DNBR occurs immediately following the reactor trip. A sensitivity study on the effect of the increased RCCA drop time was performed for the limiting DNBR case at 100% power with minimum reactivity feedback. There is a very small decrease in the minimum DNBR when the RCCA drop time is increased to 2.7 seconds. Statepoints for this case were evaluated and it was determined that the minimum DNBR remains above the safety analysis limit. This confirms that sufficient margin exists in the current licensing basis to accommodate the 2.7 second RCCA drop time.

3.1.12 Rod Cluster Control Assembly Misoperation (USAR 15.4.3)

These accidents include the following events:

1. One or more dropped RCCAs within the same group
2. A dropped RCCA bank
3. Statically misaligned RCCA
4. Withdrawal of a single RCCA

The first three of these events are analyzed to show that the DNB design basis is met. The dropped rod analysis was updated for VANTAGE 5H fuel with the increased RCCA drop time. For a 2.7 second drop time, the maximum rod worth which is considered in the analysis is 500 pcm. For rod worths above this limit, a reactor trip is assumed to occur and no further analysis is needed. The negative flux rate trip protection system might not, however, detect rod worths less than 500 pcm. Therefore analyses are performed each cycle to ensure that the DNBR design limit is met for dropped rod worths less than 500 pcm. The statically misaligned RCCA event is not impacted by the increase in the rod drop time.

For withdrawal of a single RCCA event, the number of fuel rods experiencing DNB must remain below the safety analysis limit of 5% of the total fuel rods in the core. Because the limiting time in the transient is prior to or at the time of initial insertion of the RCCAs into the core from the reactor trip, a 0.5 second increase in the RCCA drop time would have no effect on the results.

3.1.13 Startup of an Inactive Loop at an Incorrect Temperature (USAR 15.4.4)

This event is analyzed to show that the DNB design basis is met. Minimum DNBR occurs immediately following a reactor trip on low coolant flow when the power range neutron flux exceeds the P-8 setpoint. A sensitivity study was performed to address the changes in the system transient response when the RCCA drop time is increased. Relative to the licensing basis analysis, the 2.7 second RCCA drop time slightly delays the turnaround of the nuclear power transient, producing a slightly higher peak heat flux. This would then cause a very small reduction in the predicted DNBR. However, sufficient margin exists in the licensing basis analysis to ensure that the minimum DNBR limit is not violated.

3.1.14 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (USAR 15.4.6)

This accident is analyzed to determine the time available for operator action to terminate the dilution before shutdown margin is lost. No reactor trips are assumed in the analysis for shutdown and hot standby modes, and therefore an increase in the RCCA drop time does not impact the results. For startup and power operation modes, there is ample time available for the operators to terminate the dilution and the small increase in RCCA drop time causes an insignificant effect.

3.1.15 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (USAR 15.4.7)

Incorrectly loaded fuel would result in power distortion which would raise peaking factors. The USAR concludes that any significant perturbation from the intended core inventory would be detectable due to the resulting effect on power distribution. An increase in RCCA drop time would have no effect on the ability of the core instrumentation to detect such unexpected power shapes.

3.1.16 Spectrum of Rod Cluster Control Assembly Ejection Accidents (USAR 15.4.8)

The RCCA ejection transients are characterized by a rapid power burst. Due to the speed at which the power increases, this transient can be sensitive to the RCCA drop time. The transients were reanalyzed with a RCCA drop time of 2.7 seconds. The limiting criteria for this event are:

1) Average fuel pellet enthalpy at the hot spot limited to below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

2) Fuel melting limited to less than the innermost 10% of the pellet at the hot spot. (Melting is assumed to occur at 4900°F for BOL conditions and 4800°F for EOL conditions.)

The results of the analysis and a summary of parameters used in the analysis are shown in Table 3.1-2. All cases analyzed meet the acceptance criteria.

3.1.17 Inadvertent Operation of the ECCS During Power Operation (USAR 15.5.1)

The spurious operation of the safety injection system produces a negative reactivity transient causing a reduction in core power. The power reduction causes a decrease in reactor coolant average temperature and consequent coolant shrinkage. Pressurizer pressure and level decrease until the reactor is tripped. During the transient, the DNBR never decreases below the initial value. Therefore the 0.5 second increase in RCCA drop time will have no effect on the minimum DNBR.

3.1.18 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (USAR 15.5.2)

This event is analyzed to determine how long it would take the pressurizer to fill if the chemical and volume control system malfunctions. No reactor trips are assumed in the analysis, and therefore an increase in the RCCA drop time does not impact the results.

3.1.19 Inadvertent Opening of a Pressurizer Safety or Relief Valve (USAR 15.6.1)

The limiting criterion for this event is the DNB design basis. This transient is terminated by a reactor trip on OTΔT. Minimum DNBR occurs immediately following the reactor trip. A conservative evaluation of the effect of the 0.5 second increase in RCCA drop time was done based on an RCCA drop time sensitivity. The subsequent decrease in minimum DNBR is very small, and margin to the DNBR limit still exists.

3.1.20 Steam Generator Tube Failure (USAR 15.6.3)

This event is analyzed to evaluate the most severe release of secondary activity, as well as reactor activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment depends upon the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the mass of steam discharged to the environment.

These parameters are insensitive to the rate at which the RCCAs are inserted. The increase in the RCCA drop time would increase the integrated energy produced by the core by an insignificant amount. No significant change to the leakage flow or the overall system response will occur.

Additional steam generator tube rupture analyses were performed in response to License Condition 2.C(11) and will eventually replace the current USAR analyses[4]. These analyses, which are currently pending NRC review, include steam generator overfill scenarios in addition to the worst case dose scenarios. As discussed above, the increased RCCA drop time would have no significant effect on leakage flow or overall system response and therefore, steam generator overfill scenarios would also be unaffected.

3.1.21 Mass and Energy Release for Postulated Secondary Pipe Ruptures (USAR 6.2)

The mass and energy releases inside containment following a steamline break are used in the containment integrity analysis and are insensitive to the rate at which the RCCAs are inserted. The 0.5 second increase in RCCA drop time would increase the integrated energy produced by the core by an insignificant amount. The total RCS flow rate will be the same and no significant change in the overall system response will occur.

The mass and energy releases outside containment following a steamline break are used to ensure that environmental conditions used for instrument qualifications are maintained. As with the mass and energy released inside containment, the 0.5 second increase in RCCA drop time would increase the energy produced by the core by an insignificant amount. No significant change in the overall system response will occur. Therefore, data presented in WCAP-10961-P[5] will remain valid with the increased RCCA drop time of 2.7 seconds.

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[4]SLNRC 86-01, "Steam Generator Single-Tube Rupture Analysis for SNUPPS Plants Callaway and Wolf Creek", dated January 8, 1986

[5]WCAP-10961-P, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment", October 1985

Table 3.1-1

Locked Rotor Transient Results Summary

Maximum Reactor Coolant System Pressure (psia)	2617
Maximum Clad Temperature at the Core Hot Spot (^{o}F)	1754
Amount of Zr-H_2O at the Core Hot Spot (% by Weight)	0.3

Table 3.1-2

Rod Cluster Control Assembly Ejection Accident Results

Time in Life	BOL	BOL	EOL	EOL
Power Level (%)	0	102	0	102
Ejected Rod Worth (% ΔK)	0.78	0.23	0.86	0.25
Delayed Neutron Fraction	0.55	0.55	0.44	0.44
Feedback Reactivity Weighting	2.071	1.3	3.55	1.30
Trip Reactivity (% ΔK)	2.0	4.0	2.0	4.0
F_q before Rod Ejection	--	2.50	--	2.50
F_q after Rod Ejection	13.0	6.6	21.0	7.1
Number of Operational Pumps	2	4	2	4
Maximum Fuel Average Temperature (^{o}F)	3502	4080	3220	3920
Maximum Fuel Center Temperature (^{o}F)	4049	4969	3648	4867
Maximum Fuel Stored Energy (cal/gm)	149	179	135	170

FIGURE 3.1-1

Partial Loss of Forced Reactor Coolant Flow

Core and Loop Flow versus Time

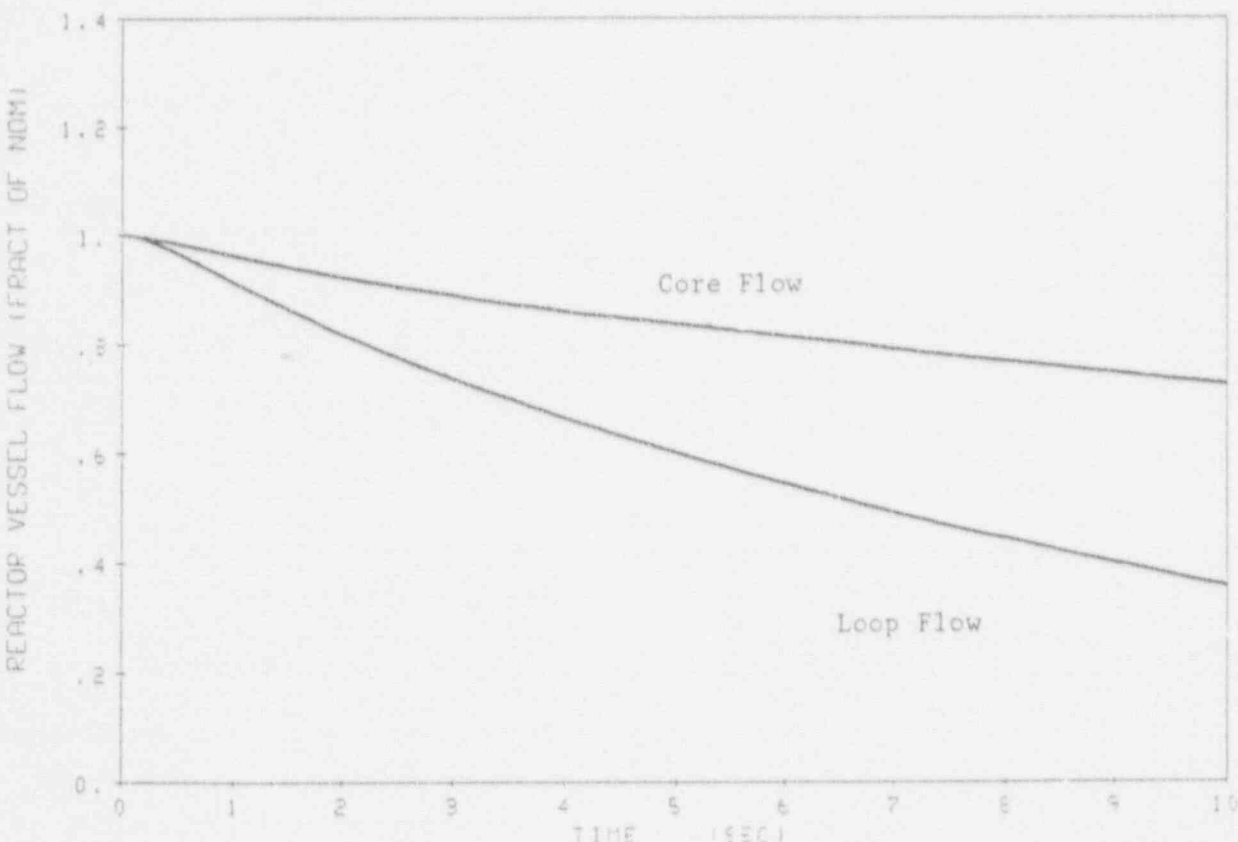


FIGURE 3.1-2

Partial Loss of Forced Reactor Coolant Flow

Heat Flux versus Time

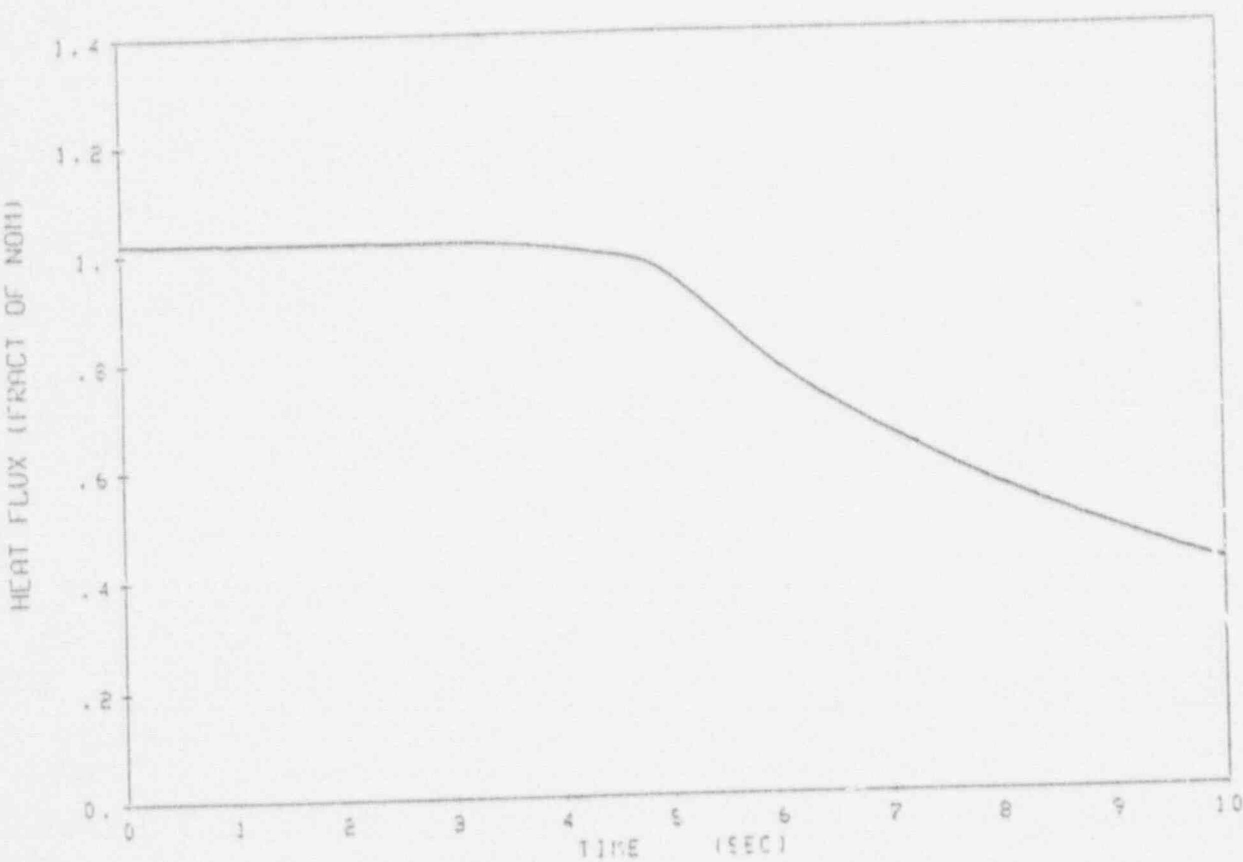


FIGURE 3.1-3

Partial Loss of Forced Reactor Coolant Flow

DNBR versus Time

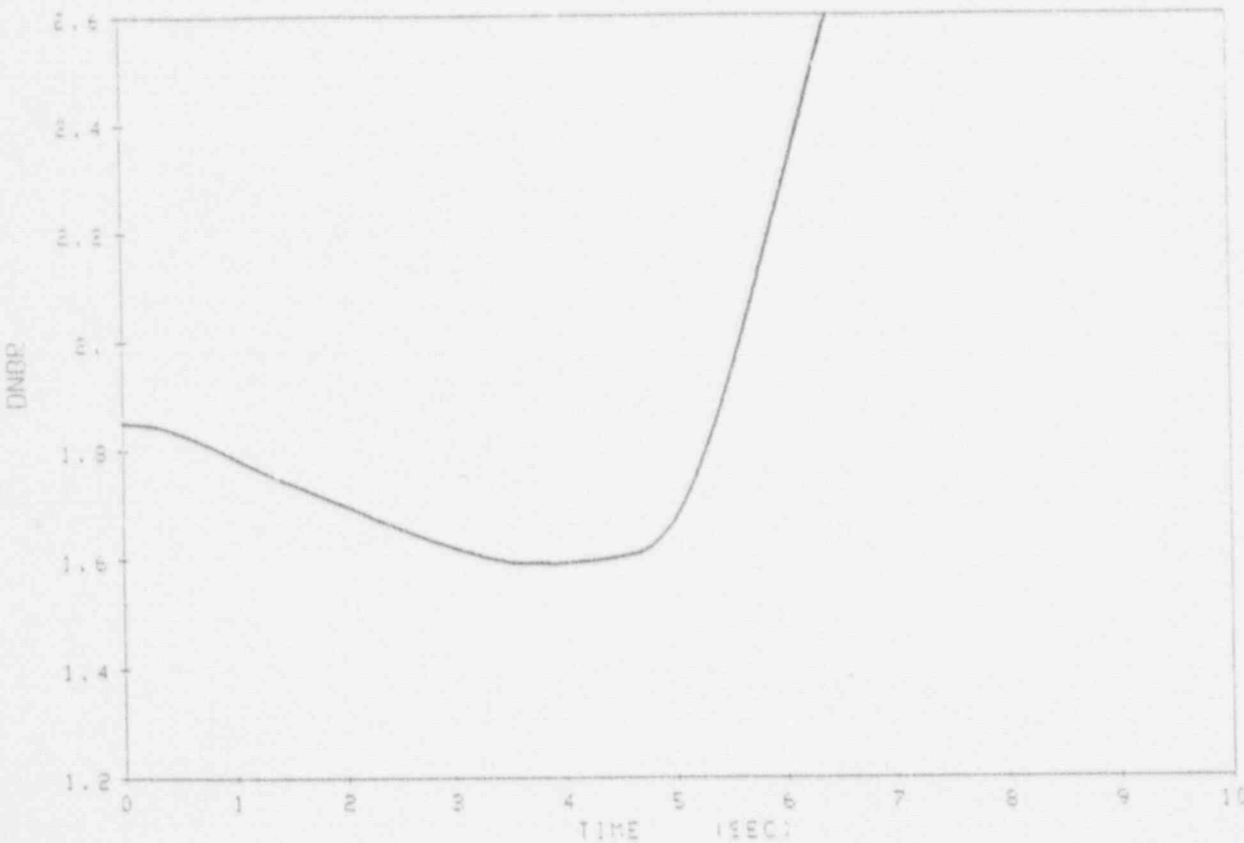


FIGURE 3.1-4

Complete Loss of Forced Reactor Coolant Flow

Core and Loop Flow versus Time

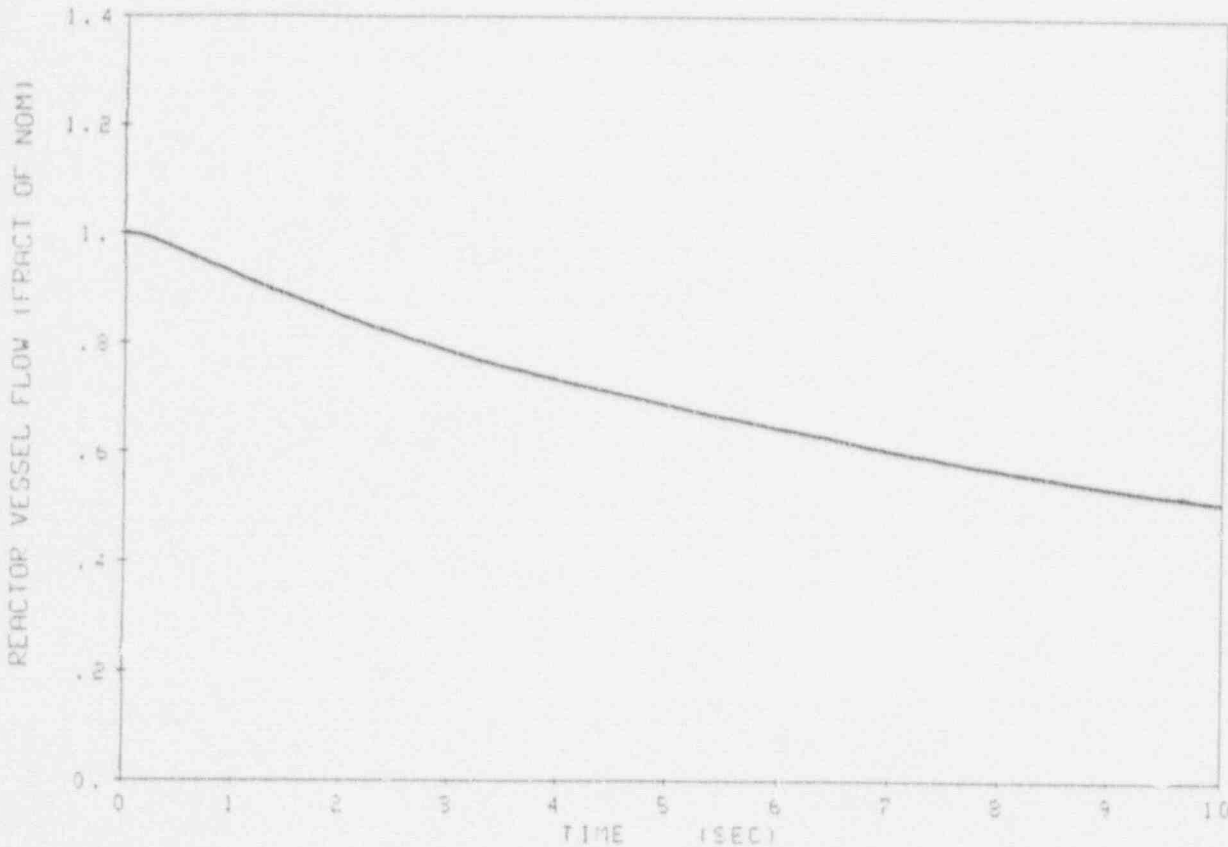


FIGURE 3.1-5

Complete Loss of Forced Reactor Coolant Flow

Heat Flux versus Time

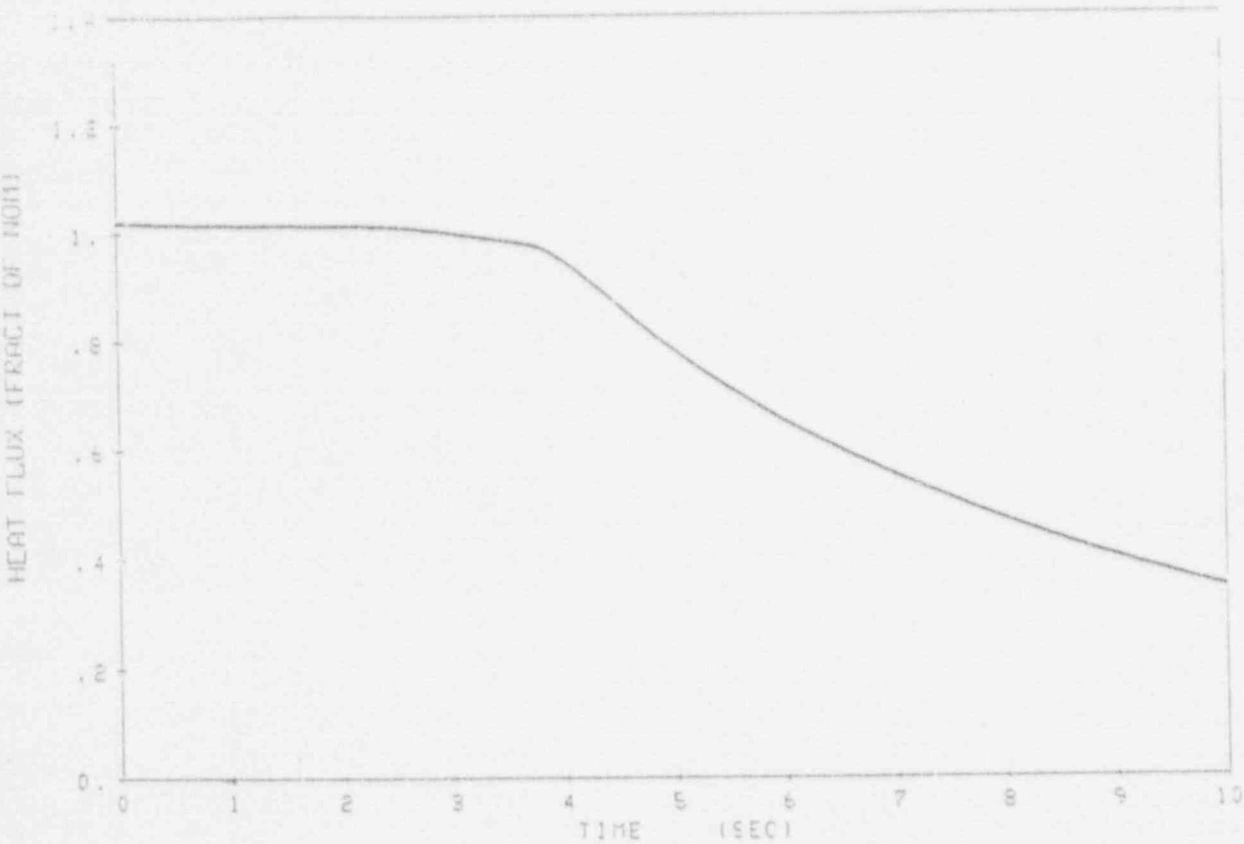


FIGURE 3.1-6

Complete Loss of Forced Reactor Coolant Flow

DNBR versus Time

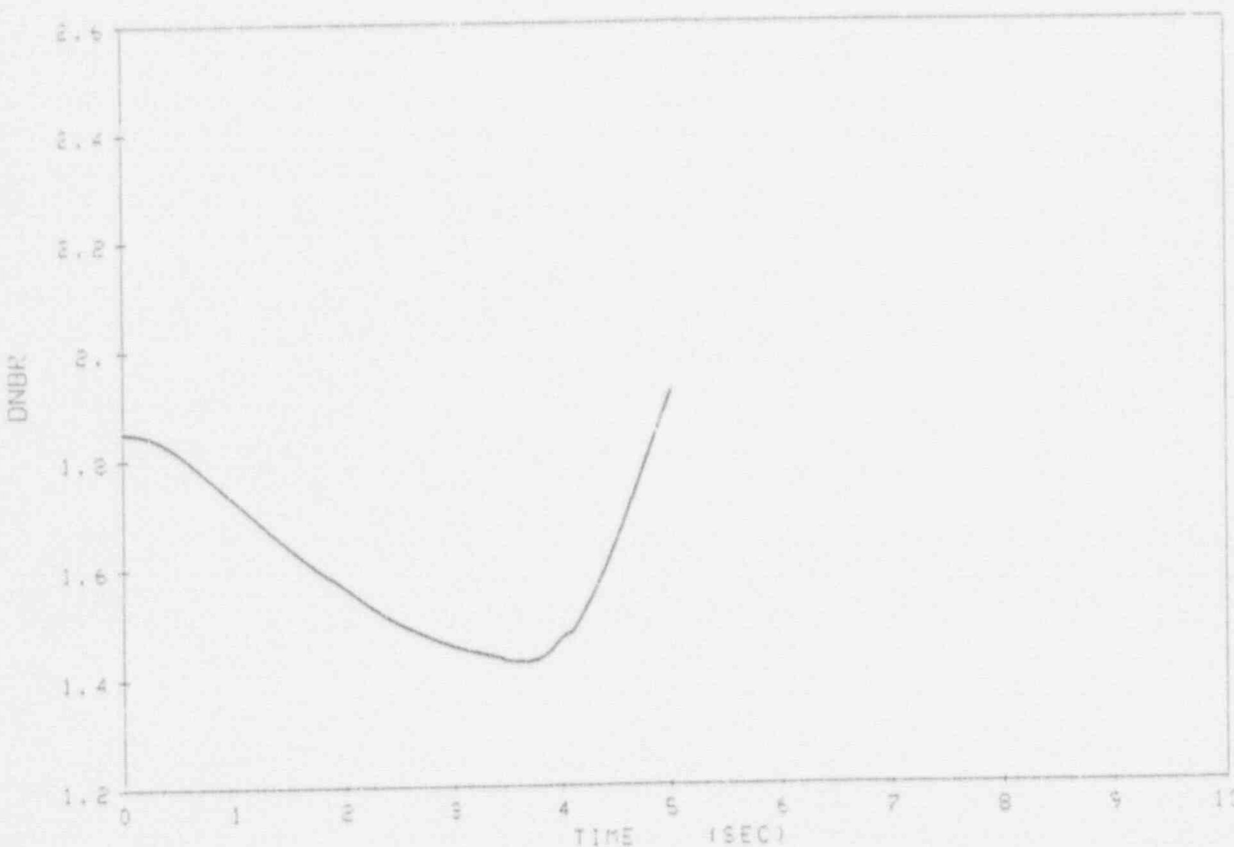


FIGURE 3.1-7

Single Reactor Coolant Pump Locked Rotor

Core Flow versus Time

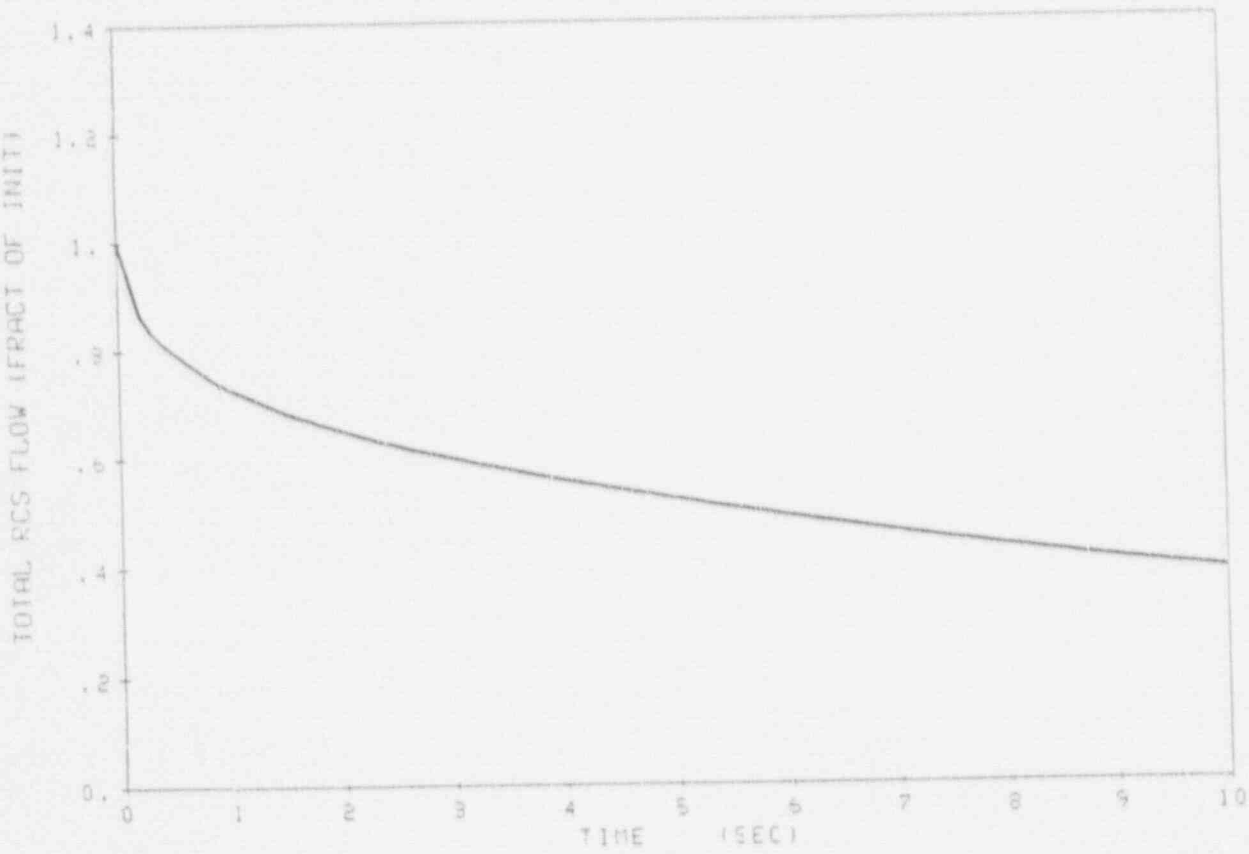


FIGURE 3.1-8

Single Reactor Coolant Pump Locked Rotor

Nuclear Power and Heat Flux versus Time

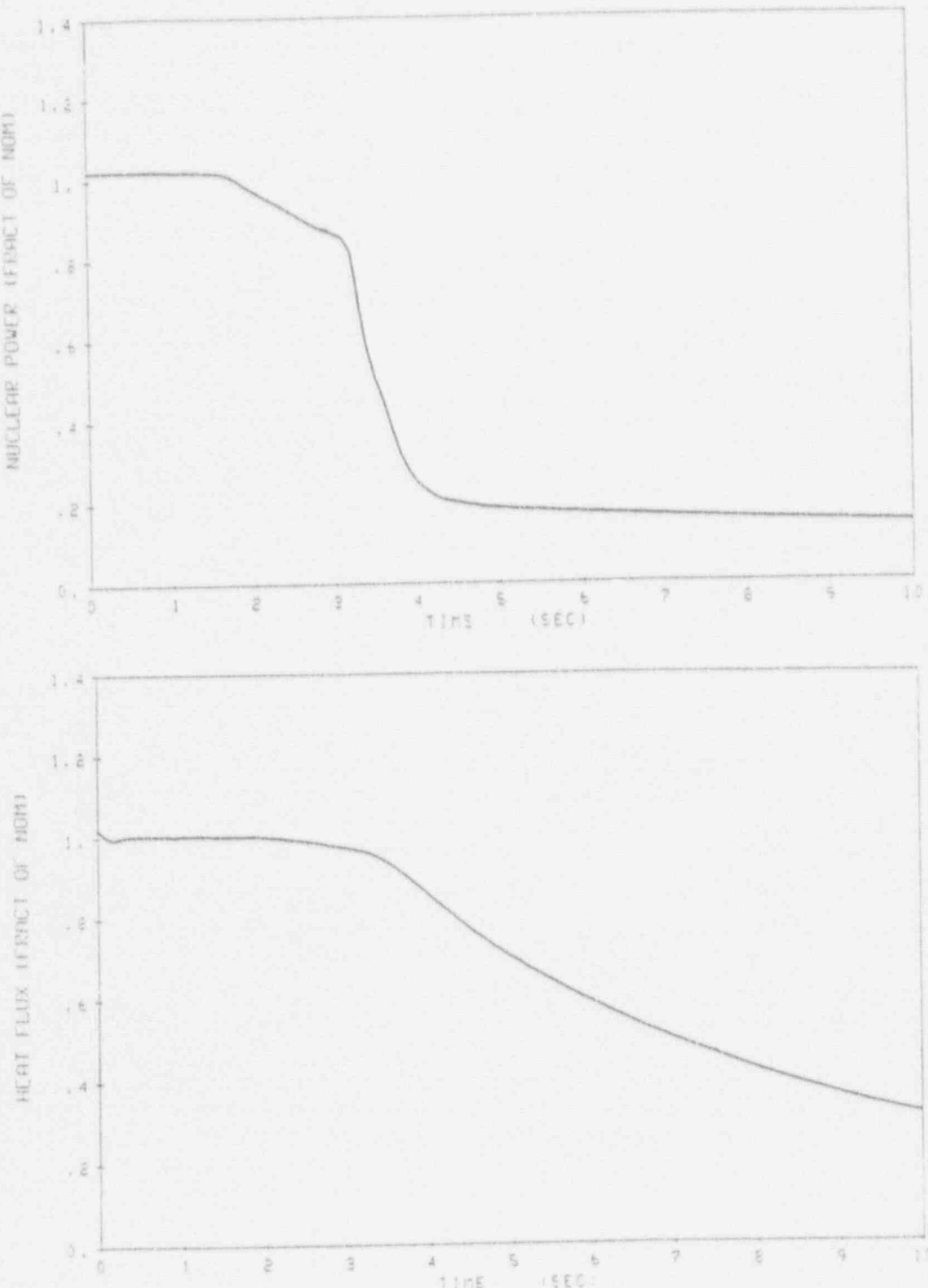


FIGURE 3.1-9

Single Reactor Coolant Pump Locked Rotor

Clad Temperature versus Time

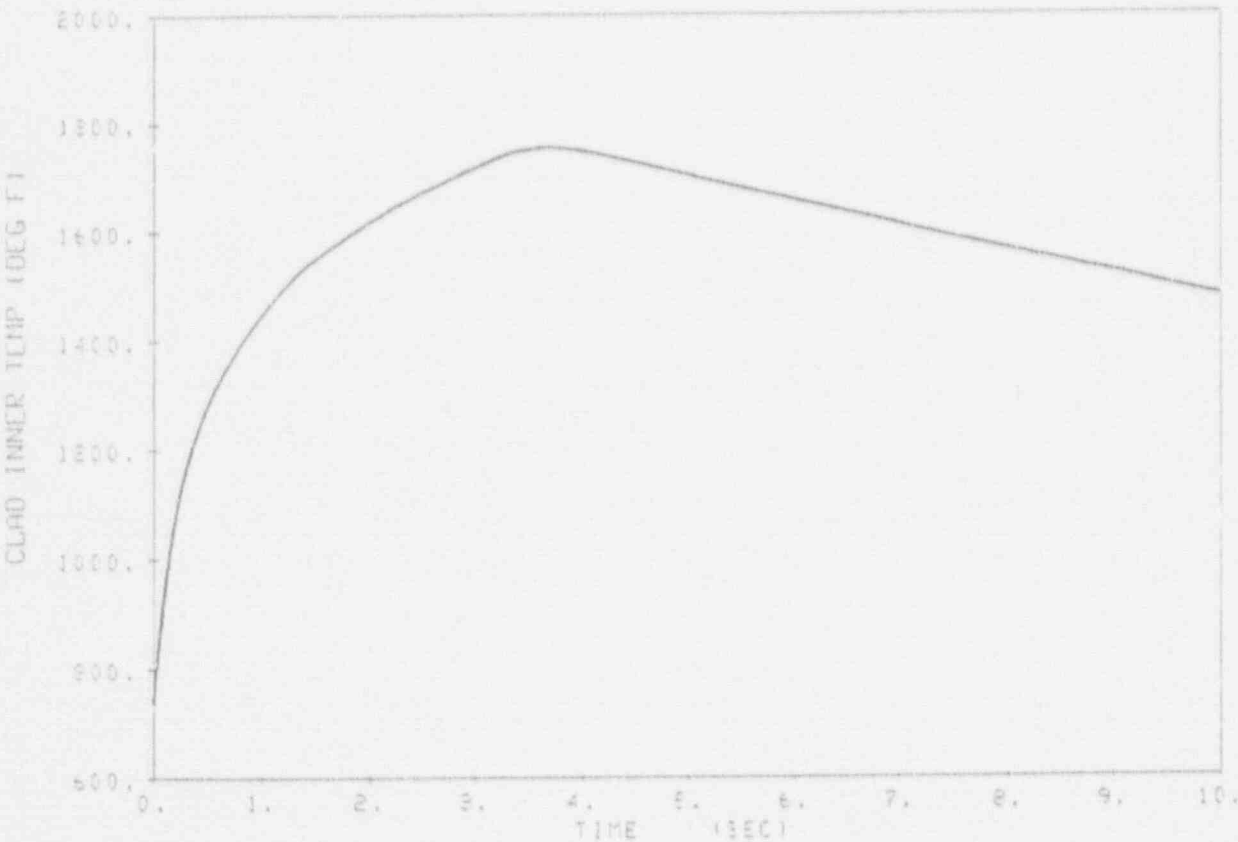


FIGURE 3.1-10

Single Reactor Coolant Pump Locked Rotor

RCS Pressure versus Time

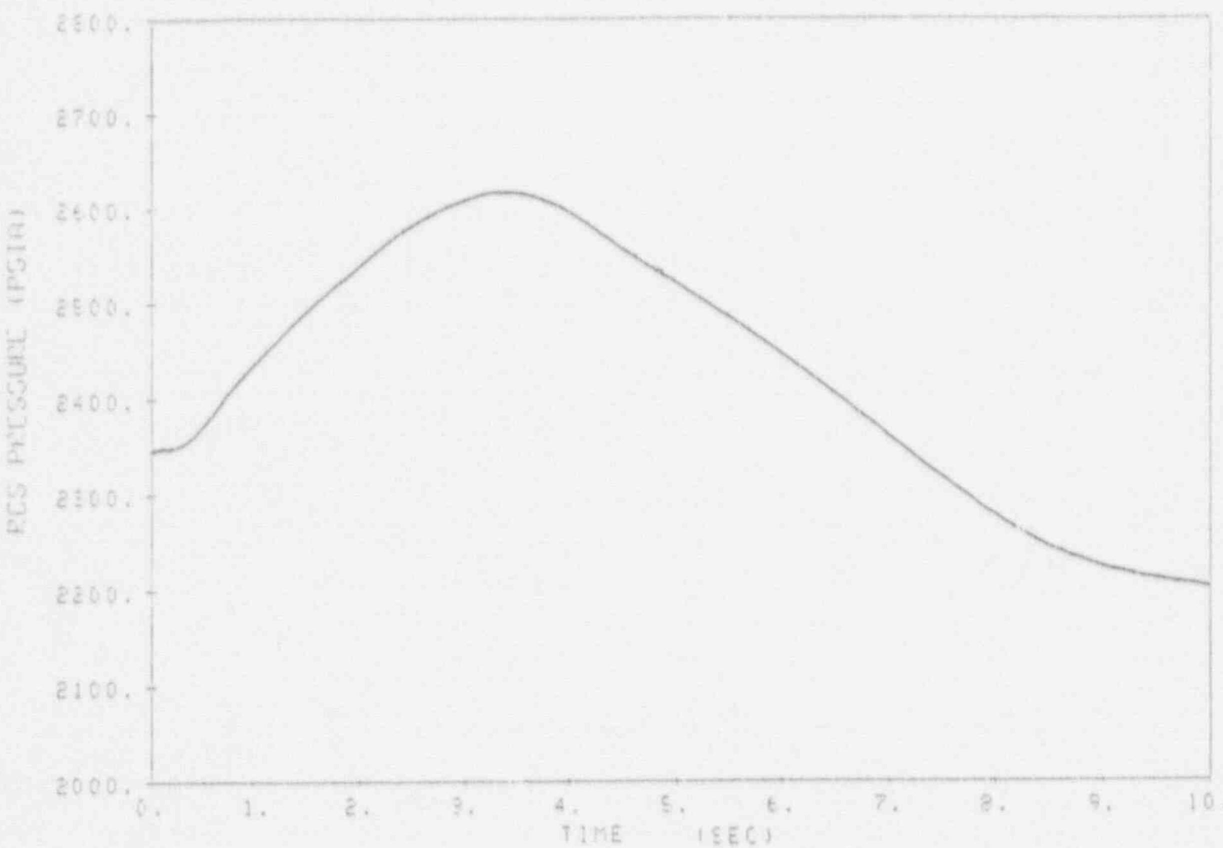


FIGURE 3.1-11

Startup from Subcritical Condition

Nuclear Power versus Time

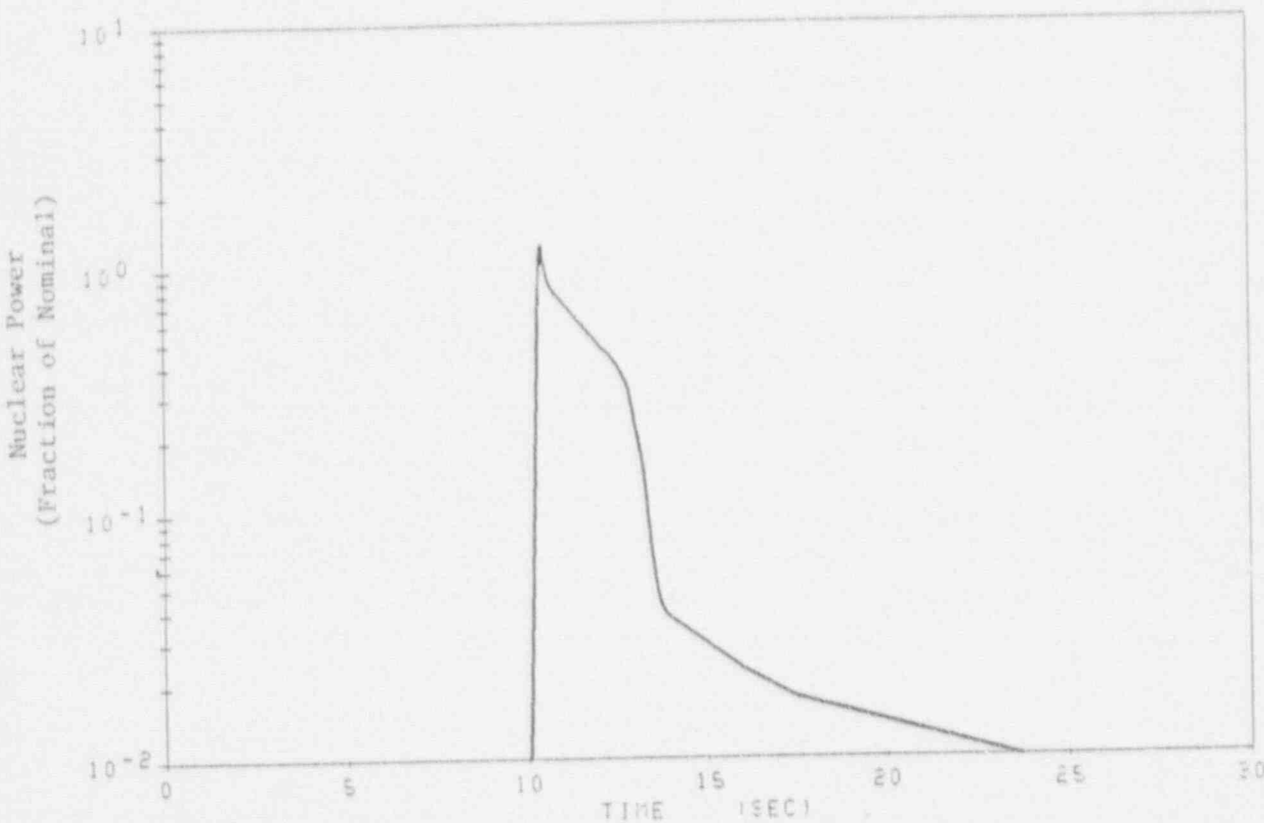


FIGURE 3.1-12

Startup from Subcritical Condition

Heat Flux versus Time

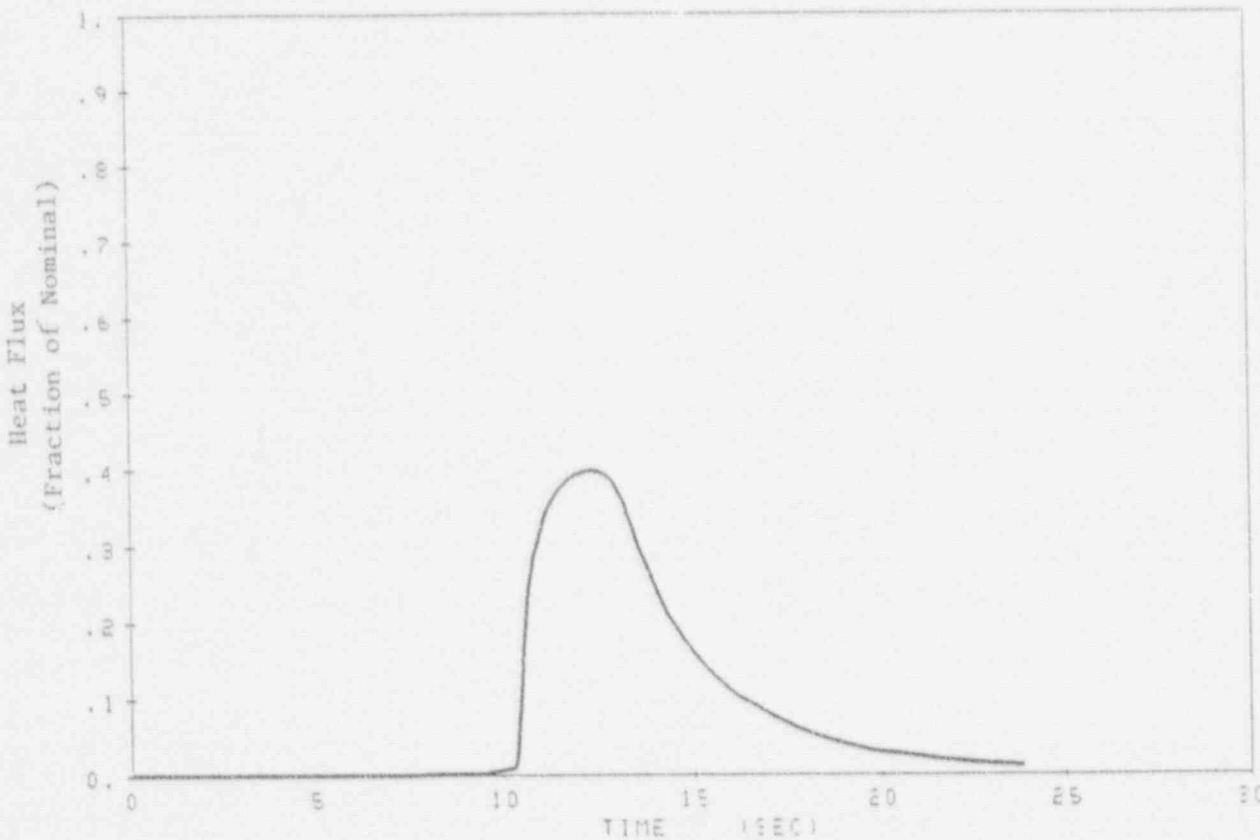


FIGURE 3.1-13

Uncontrolled Rod Withdrawal from a Subcritical Condition

Fuel Temperature versus Time

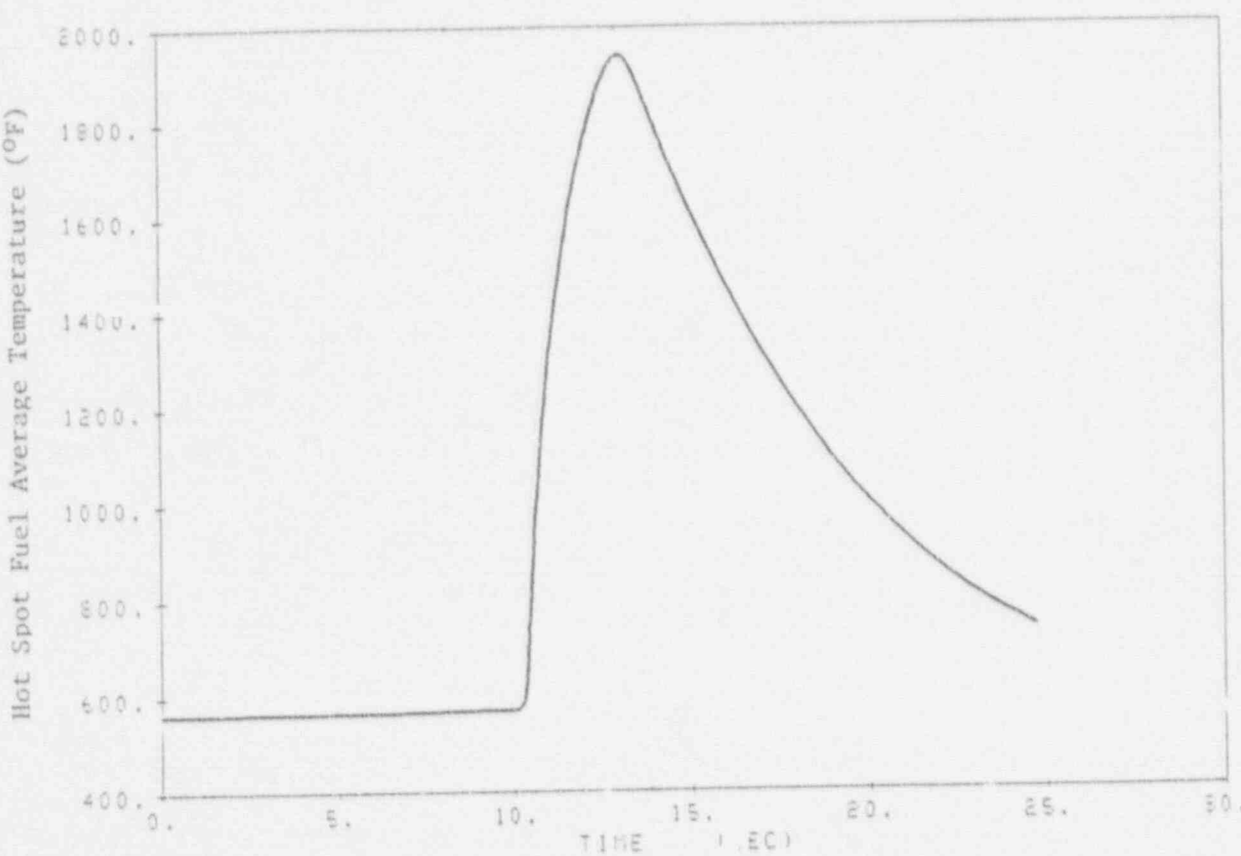
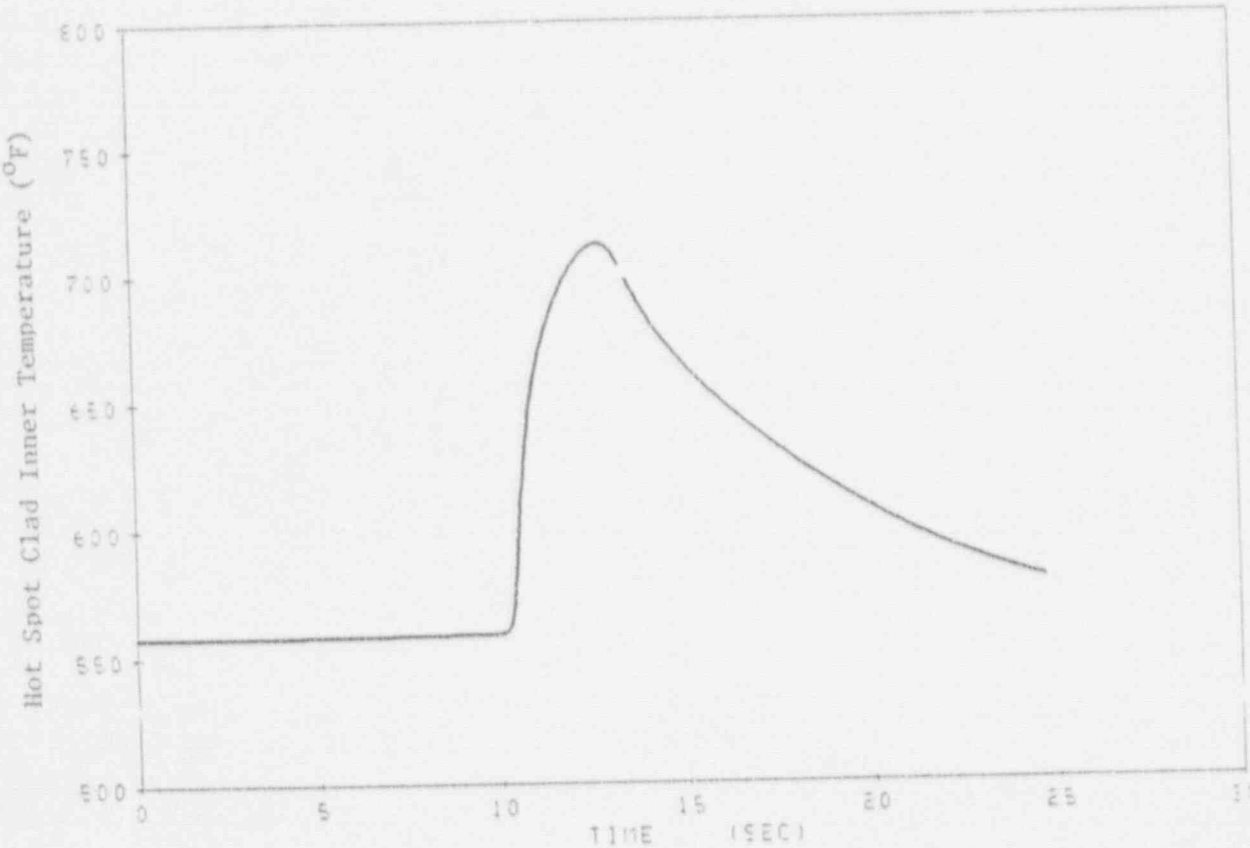


FIGURE 3.1-14

Uncontrolled Rod Withdrawal from a Subcritical Condition

Clad Temperature versus Time



3.2 LOCA and LOCA Related Analyses

This section summarizes the evaluations performed to assess the effect of VANTAGE 5H fuel with increased RCCA drop time on the WCGS licensing basis LOCA analyses. As noted in Addendum 2A to WCAP-10444-P-A, the majority of the VANTAGE 5H fuel features have no adverse effect on the licensing basis LOCA analyses due to the mechanical and hydraulic similarity to 17x17 Standard (STD) fuel. The WCAP Addendum documents that transitioning from 17x17 STD fuel to 17x17 VANTAGE 5H fuel without Intermediate Flow Mixers (IFMs) results in no transition core Peak Cladding Temperature (PCT) penalty. The only item which can potentially affect the LOCA analyses is the increase in RCCA rod drop time.

3.2.1 Large Break Loss-of-Coolant Accidents (USAR 15.6.5)

The large break LOCA analysis for the WCGS has been performed using the Westinghouse 1981 Evaluation Model with BART[6] for 17x17 standard fuel. It resulted in a calculated PCT less than 2200°F for the limiting Cd=0.4 double-ended cold leg guillotine break.

An evaluation has been performed to consider the effects on the analysis of the VANTAGE 5H fuel with increased RCCA drop time. The large break LOCA evaluation model does not take credit for the negative reactivity introduced by the RCCAs. Instead, the reactor is brought to a subcritical condition by the presence of voids in the core caused by the rapid depressurization of the RCS. Since credit is not taken for the negative reactivity introduced by the RCCAs, the increase in rod drop time will have no effect on the current USAR large break analysis. Furthermore, sensitivity studies have demonstrated that VANTAGE 5H fuel is less limiting than 17x17 STD fuel.

Based on the above discussion, the increased RCCA drop time will not result in an increase in the PCT for the WCGS. Therefore, the change is acceptable and the resulting PCT remains within the regulatory limits.

3.2.2 Small Break Loss-of-Coolant Accidents (USAR 15.6.5)

The current small break LOCA licensing basis analysis for Wolf Creek predicted a peak clad temperature of 1790°F using the October 1975 WFLASH[7] Westinghouse Small Break Evaluation Model[8]. Since the time of that analysis, a number of safety evaluations have been performed on the small break LOCA analysis resulting in a net PCT of 1897°F.

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[6] WCAP-9561-P-A, "BART-1A: A Computer Code for the Best Estimate Analyzed Reflood Transients", 1984

[7] WCAP-8200, Revision 2 (Proprietary) and WCAP-8261, Revision 1 (Non-Proprietary), "WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR", July 1974

[8] WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), "Westinghouse Emergency Core Cooling System Small Break October 1975 Model", April 1977

The only VANTAGE 5H feature which affects the small break LOCA analysis is the increase in rod drop time. The Westinghouse small break model assumes the reactor core is brought to a subcritical condition by the negative reactivity of the RCCAs. The increase in the rod drop time to a maximum value of 2.7 seconds resultes in a penalty of 2^{o}F. The revised licensing basis for WCGS would, therefore, result in a net PCT of 1899^{o}F, well within the regulatory limits.

In addition to the PCT evaluation, an evaluation was performed to demonstrate compliance with the 10 CFR 50.46 limits associated with zirconium-water reaction:

1) Maximum Cladding Oxidation: the calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation.

2) Maximum Hydrogen Generation: the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the amount that would be generated; if all of the metal in the cladding cylinders surrounding the fuel were to react.

The evaluation showed that the 17% limit is met and the core average zirconium-water reaction is well below the 10 CFR 50.46 limit of 1%. Therefore, compliance with the 10 CFR 50.46 limits for the WCGS analysis has been demonstrated and the use of Zircaloy grids is acceptable.

3.2.3 Blowdown Reactor Vessel and Loop Forces (USAR 3.6.2 & 3.9.2)

The major factors in determining the resulting hydraulic forces from a postulated LOCA on the vessel and the internals are the reactor coolant system primary fluid temperature and pressure. Since the increased RCCA drop time does not change the primary side design temperatures and pressures which are modeled in the forces analysis, there will be no effect on the LOCA hydraulic forces.

3.2.4 Post LOCA Long-Term Core Cooling, Boron Evaluation (USAR 15.6.5)

The methods used to satisfy the requirements of 10 CFR 50.46(b)(5), "Long-Term Cooling," are defined in WCAP-8339[9]. The reactor must remain shutdown by borated water residing in the RCS and sump after a LOCA. Since credit for the RCCAs is not taken for a large break LOCA, the borated water provided by the accumulators and the Refueling Water Storage Tank (RWST) must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all RCCAs out. This is demonstrated for each reload design on a cycle specific basis.

[9] Bordelon, F. H., et al., "Westinghouse Emergency Core Cooling Evaluation Model Summary", WCAP-8339, 1974

Since the increased RCCA drop time will not affect the sources of borated and non-borated water assumed in the long term cooling calculation, it is concluded that there would be no change to the long term cooling capability of the ECCS system. As noted above, this is confirmed on a cycle by cycle basis ensuring compliance with this requirement independent of this safety evaluation.

3.2.5 Hot Leg Switchover to Prevent Potential Boron Precipitation (USAR 15.6.5)

Post-LOCA hot leg recirculation time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This recirculation time is dependent on power level, and the RCS, RWST, and accumulator water volumes and boron concentrations. The increased RCCA drop time will have no effect on the assumptions for the RCS, RWST, and the accumulators in the hot leg switchover calculation. Thus, there is no effect on the post-LOCA hot leg switchover time.

4.0 CONCLUSIONS

The results of this safety evaluation confirm the acceptability of the increase of the RCCA drop time to the dashpot from 2.2 to 2.7 seconds. The justification is based upon an evaluation of the licensing bases of the non-LOCA and LOCA analyses. It is confirmed that the transient specific minimum DNBRs meet the DNBR safety analysis limit. The evaluations also confirmed that there is no increase in transient specific calculations of fuel rod failure. Evaluations and limited sensitivity studies support the conclusion that all other applicable safety analysis acceptance criteria continue to be met and the conclusions presented in the USAR remain valid.

The increase in the RCCA drop time does not affect any of the mechanisms postulated in the USAR to cause LOCA or non-LOCA design basis events. Sensitivity studies, evaluations and minimum DNBR recalculations confirm that the transient behaviors described in the USAR do not change, and the USAR conclusions remain valid for the proposed changes. On these bases, it is concluded that the probability of occurrence or consequences of accidents previously evaluated in the USAR are not increased.

The accidents assumed to occur at the current RCCA drop time are the same as those for the proposed RCCA drop time. This proposed change does not change the plant configuration in a way that introduces a new potential hazard to the plant. For this reason, the possibility of a new accident which is different than any already evaluated in the USAR is not created.

The analyses and evaluations discussed in this safety evaluation demonstrate that all applicable safety analysis acceptance criteria continue to be met for the increased RCCA drop time. Therefore, it is concluded that there is no reduction in the margin of safety as described in the bases of any technical specification.

ATTACHMENT II

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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This proposed change has been reviewed per the standards provided in 10 CFR 50.92. Each standard is discussed separately below.

Standard I - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The increase in Rod Cluster Control Assembly (RCCA) drop time has no affect on mechanisms postulated in the Updated Safety Analysis Report (USAR) to cause design basis events. Sensitivity studies, evaluations and minimum Departure from Nucleate Boiling Ratio (DNBR) recalculations have confirmed that the transient behaviors described in the USAR are not significantly changed and that the USAR conclusions remain valid for the proposed change. On this basis it is concluded that there will be no significant increase in the probability or consequences of previously evaluated accidents.

Standard II - Create the Possibility of a New or Different Kind of Accident From any Previously Evaluated.

This proposed change involves only an increase in the maximum RCCA drop time. There are no physical modifications to the facility or changes in methods of operation. The increase in the maximum RCCA drop time will allow the use of an upgraded fuel design (Westinghouse VANTAGE 5H). However, applicable nuclear, mechanical and thermal-hydraulic fuel design criteria are unchanged. Therefore, this proposed technical specification revision does not create the possibility of a new or different kind of accident from any previously evaluated.

Standard III - Involve a Significant Reduction in the Margin of Safety.

Evaluations performed to support this proposed change demonstrate that all applicable safety analysis acceptance criteria will continue to be met and, therefore, there will be no decrease in any margin of safety defined in the technical specifications.

Based on the above, the requested technical specification change does not involve a significant increase in the probability or consequences of a previously evaluated accident, create the possibility of a new or different kind of accident, or involve a significant reduction in the margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration in accordance with 10 CFR 50.92.

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

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This amendment request meets the criteria specified in 10 CFR 51.22(c)(9). Specific criteria contained in this section are discussed below.

(i) the amendment involves no significant hazards consideration,

As demonstrated in Attachment II, this proposed amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

This proposed change involves only the maximum control rod drop time specified by the technical specifications and has no direct effect on plant effluents. This increase in rod drop time will allow the use of an upgraded fuel design (Westinghouse VANTAGE 5H). Performance of the upgraded fuel is expected to be equivalent to or better than that currently in use and therefore will have no adverse affect on the type or amount of plant effluents. The change involves no other physical modifications to the facility or change in the methods of operation. Therefore, this change will have no effect on normal plant effluents and there will be no change in the types or amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Implementation of this change will involve only administrative measures associated with control rod drop time testing. Plant operation and refueling activities will not be affected. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure associated with this proposed change.

Based on the above, there will be no significant impact on the environment resulting from this change and the change meets the criteria specified in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements of 10 CFR 51.21 relative to a specific environmental assessment by the Commission.