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PARAMETRIC REACTIVITY TRANSIENT ANALYSES FOR THE FFTF NUCLEAR PROOF TEST REACTOR

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ABSTRACT

Fault tree techniques have been used to identify possible failure paths within the NPTR which could lead to core disassembly. The analysis of the various faults has led to formulation of design requirements, protective system requirements, and administrative restraints required to prevent accidents from these faults.

Transient analyses were performed using the heat transfer-nuclear kinetics codes, Nutiger-II, FORE-II, and MELT-II. To verify results, inter-comparison studies were made between the codes. The codes were in good general agreement. Each code was found to exhibit different advantages and disadvantage.

Inherent reactivity feedback effects were assessed in the analysis. With the assumed core parameters, there appears to be sufficient Doppler to prolong a nuclear transient to allow protective action to prevent fuel from melting. The use of average values of the feedback coefficients smeared over the entire core does not appear to be an acceptable method with spacially dependent temperatures.

In the thermal analysis, the fuel pin gap coefficient and sodium film coefficient do not appear to be highly sensitive parameters for transient analysis. Power transients resulting from reactivity insertions of from 2\$/sec to 20\$/sec have been examined in detail. Sodium will be molten

before fuel melting occurs for accidents within this range. For the smaller ramp rates (< 4$/sec), sodium may even reach vaporization temperatures before any fuel melts.

Power transients terminated by effective protective action were investigated. It is believed possible to design a scram system, with the present state of the art, to prevent sodium from melting for a reactivity ramp up to at least 6$/sec. This same system would prevent fuel melting for a reactivity ramp up to 15$/sec.

Sodium thermal expansion will play a very important role in a core disassembly. When the average sodium temperature exceeds 250 °F, physical core distortion must result to relieve expansion pressures.

Rupturing of the fuel assembly cans during a transient increases the probability of a sodium fire. Pressures and temperatures from a sodium fire could easily exceed 20 psig and 1000 °F.

The design basis accident has not been identified. However, the lower limit is a sodium fire involving hot liquid sodium and possible sodium vapor. A fuel vapor explosion would require a large initiating reactivity ramp rate (> 20$/sec) with at least 3$ total reactivity worth. No mechanism for intorduction of a reactivity insertion of this characteristic has been identified other than a dropped fuel assembly into a vacant core position. This mechanism is discounted as it is believed that sub-criticality of the reactor can be guaranteed during refueling.

It is conceivable that a minor accident could be aggravated into an explosive accident by failure of protective system and positive feedback mechanisms. The possibility of this occurring is dependent upon what effects the confined sodium has on the core. It is desirable that the sodium would take the core to a disassembly condition or termination mode. Additional analysis will be necessary before this can be guaranteed.

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1.0 INTRODUCTION

1.1 BACKGROUND

The Nuclear Proof Test Reactor (NPTR) is a proposed zero power reactor which, at its normal operating condition, will reproduce neutronic characteristics of the Fast Test Reactor (FTR). The Nuclear Proof Test Reactor (NPTR) will also be capable of brief powered runs of up to several kilowatts.

A design requirement of the NPTR is that it have the capability to accept for test, FTR fuel, poison rods, and experiments in any of its grid locations. This requirement essentially requires the NPTR to have the same macro-geometry as the FTR. Currently, the NPTR fuel pin size and fuel assembly geometry are analogous to FTR fuel. The major physical difference between the reactors is that sodium will be frozen and canned in the NPTR.

Previous analysis of NPTR safety has been based on models which were representative of a shutdown FTR. It was recognized that accident and power transient phenomena would be considerably different for the NPTR. With this viewpoint an accident analysis was initiated for the NPTR which was essentially divorced from the FTR accident analysis.

1.2 OBJECTIVES

The primary objectives of this analysis were:

- Identify potential accident initiating mechanisms.

- Determine static and dynamic reactivity effects for identified reactor abnormalities.
- Parametrically investigate power transients for a range of reactivity insertions.
- Parametrically investigate effects of variances in different core parameters.
- Determine scram system requirements.
- Identify design basis accident.

With the exception of the last item all objectives were obtained. Deferral of the NPTR project to a later date terminated the NPTR accident analysis before this phase was complete.

1.3 METHOD OF ANALYSIS

A detailed mechanistic approach was applied in the preliminary safety analysis of the NPTR. Identification of accident initiating mechanisms was through the development of fault tree logic diagrams.

After the initiating mechanisms were identified, their static and dynamic reactivity effects were quantified. A parametric analysis was then performed bracketing the identified range of reactivity insertions and insertion rates. The results were analyzed for interplay of reactivity feedbacks, core physical behavior, and possible accident aggravation effects.

The heat transfer-nuclear kinetics codes NUTIGER-II, FORE-II, and MELT-II were utilized for the reactivity transient studies and determination of scram system requirements.

2.0 SUMMARY OF RESULTS AND CONCLUSIONS

Fault tree techniques were found to be extremely useful in identifying accident initiating mechanisms. Once these faults were identified, design requirements, protective system requirements, and administrative restraints were formulated to prevent accidents from occurring.

Transient analyses were performed using the heat transfer-nuclear kinetics codes NUTIGER-II, FORE-II, and MELT-II. Base case comparisons indicate that they can essentially be used on an interchangeable basis with only minor discrepancies which are explainable. Each code was found to exhibit different advantages and disadvantages. A comprehensive analysis would require the utilization of all three codes.

Inherent reactivity feedback effects were assessed. With the assumed core parameters, there is sufficient Doppler ($T \frac{dk}{dT} = -0.004$) to turn over a nuclear transient to allow protective action to prevent fuel from melting for any credible transient.

Power transients resulting from reactivity insertions of from 2$/sec to 20$/sec were examined in detail. Sodium will be molten before fuel melting occurs for accidents within this range. For smaller ramp rates ($\sim$ 4$/sec), sodium may even reach vaporization temperatures before fuel melting begins.

Scram parameters were analyzed and it is believed possible to design a scram system, with the present state of the art, that will prevent sodium from melting for a reactivity insertion of up to 6$/sec. This same system would prevent fuel melting for reactivity ramp up to 15$/sec.

Sodium thermal expansion will play a very important role in a core disassembly. When the average sodium temperature exceeds 250 °F, physical core distortion must result to relieve expansion pressures. The rupturing of a fuel assembly can during a transient would greatly increase the probability of a sodium fire. However, it is possible that sodium expansion forces may disrupt the core to a mode which will terminate a transient.

The design basis accident has not been identified. However, the lower limit is a sodium fire involving hot liquid sodium and possible sodium vapor. A fuel vapor explosion would probably require a large initiating reactivity ramp rate (> 20$/sec) with at least 3$ reactivity worth. No mechanism for introduction of reactivity insertion of this characteristic has been identified other than a fuel assembly dropped into a vacant core position. This mechanism is discounted as it is believed that subcriticality of the reactor can be guaranteed during refueling.

It is conceivable that a minor accident could be aggravated into a fuel vapor explosion accident by the failure of protective systems and positive feedback mechanisms. The possibility of this occurring is dependent upon what effects the confined sodium has on the core. Additional analysis is recommended in this area.

3.0 ACCIDENT IDENTIFICATION AND ANALYSIS

To start the NPTR accident analysis, a fault tree was constructed for core disassembly. This fault tree was expanded downward until the basic initiating accidents or events were identified. Each individual branch of the tree was analyzed to determine what design requirements, protective system requirements and/or administrative procedures could be used to reduce the probability of the initiating event from occurring or progressing to a core disassembly.

For the purpose of discussion, the NPTR core disassembly fault tree, (Figures 1a-1d), was broken into five accident categories.

- Improper Control/Safety Rod Motion
- Improper Startup Operation
- Neutron Spectrum Shift
- Shutdown Criticality
- Core Compaction

Each category was analyzed in detail to determine the static and dynamic reactivity characteristics associated with it. These results were used in the subsequent parametric transient analysis. The result of the fault tree analysis were:

3.1 IMPROPER CONTROL/SAFETY ROD MOTION

Control/safety rod motion created by an object falling onto the core can be prevented by design requirements coupled with administrative controls and operational procedures. The individual worth of a single rod should not exceed 50% of the shutdown margin.

Expulsion of a control/safety rod by internal forces is considered incredible. Due to the low heat generation rates at operational conditions no mechanism can be identified which is capable of generating sufficient energy to expel a rod by internal forces. The inadvertent uninhibited withdrawal of a control/safety(s) at normal withdrawal speeds would give reactivity insertion rate of from 0.05$/sec to 0.60$/sec depending upon the number of rods involved. For this accident to occur and proceed to core damage would require operator error and failure of protective systems.

3.2 IMPROPER STARTUP ACCIDENT

Criticality during safety rod withdrawal would require a gross fuel loading error and failure to note improper neutron response during initial withdrawal. For the accident to proceed to core damage would require failure of protective system.

A tertiary reactivity control system has been proposed for NPTR for use during core loading. For criticality to occur during movement of the tertiary system would require a grossly overloaded core or the inadvertent removal of part or all of the other poison systems. If criticality was achieved during tertiary system movement from a grossly overloaded core there would be no protective system scram as all poison rods would be in. However, the rate of withdrawal coupled with the large Doppler safety margin would allow ample time for either automatic or manual reversal of the tertiary system. It is incredible that the series of errors needed to overload the core to this extent could occur and go unnoticed.

3.3 NEUTRON SPECTRUM SHIFT

A neutron spectrum shift could result from a moderating material being accidentally inserted into the core. This could be in the form of an experimental error or core flooding, either internal or external. The introduction of a liquid moderator (H_2O or hydrocarbon, etc.) into the core voids and the flooding of one "dry" (sodium absent) fuel assembly could give a reactivity input of up to 12$. Flooding of core voids would give only 9$.

The reactivity insertion rate would be dependent upon the method of moderator insertion. It is possible that a transient resulting from spectrum shifting would be self terminating as the liquid moderating material in the flooded void spaces would be forced out by thermal expansion effects. The possibility of a sodium fire would still exist.

The possibility of accidental insertion of a liquid moderator into the core must be limited by design. The quantity of liquid moderator allowed into the reactor cell must be limited to safe amounts. All potential moderating materials must be under direct administrative control with strict accounting procedures.

3.4 SHUTDOWN CRITICALITY

A shutdown criticality accident occurring from a dropped fuel assembly can be precluded by the guaranteed subcriticality of the reactor. A dropped fuel element during the life of the reactor is deemed to be credible; however, the resulting reactivity ramp rate can be design limited to a certain degree.

Credit may be taken from contact loading, i.e., knowledge of control rod location, visual core observation, etc; however, uncertainties are introduced by unknown worths of new fuel and test assemblies. In addition, the tertiary shutdown system provides the required assurance of guaranteed subcriticality. For the reactor to go critical during tertiary system withdrawal would require gross loading errors and failure to monitor reactor status.

3.5 CORE COMPACTION

Core compaction due to external falling objects can be made incredible by design requirements coupled with administrative controls and operational procedures.

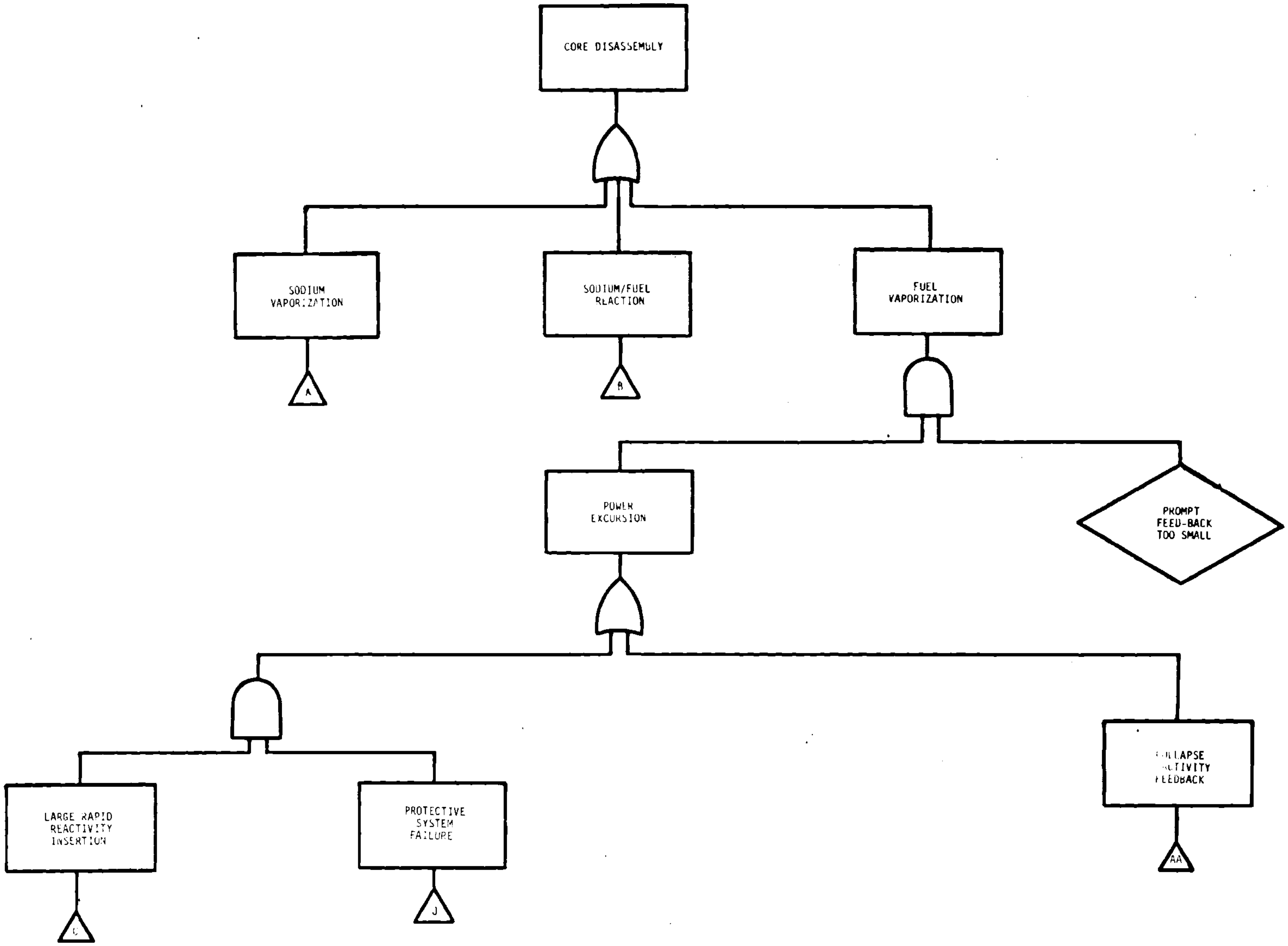
Core compaction from a loss of coolant accident is possible only with gross overpower. Fuel melting can be avoided and thus core compaction due to a power/heat imbalance at normal operational or shutdown condition is not possible. The maximum short term fission power of the NPTR is 2 kW. The inherent heat generation of the Pu fuel will be about 3 kW. This gives a maximum power of 5 kW. Therefore, the temperature will depend on thermodynamic rather than neutronics conditions.

An analysis (BNWL-766)[1] has shown that the reactor operating at a power level of 5 kW without cooling will result in no more than 6 °F temperature rise per hour. This is a fairly conservative estimate, as the calculational model assumed that there was very little radial heat flow. The balance of the heat was assumed to flow out the ends.

This low temperature rise will allow the core to be unloaded in the event that there is a complete loss of core cooling capacity. If the core remained without cooling for a prolonged period, only localized sodium melting would result. There is insufficient heat generation to do extensive melting. It has been calculated that 18 kW of heat would be required to hold the core at a temperature of 250 °F.

The scope of experimental accidents which could lead to reactivity insertions were also reviewed. It is difficult to conceive an experiment that could lead to gross core failure.

FIGURE 1a. *Fault Tree for NPTR Core Disassembly*



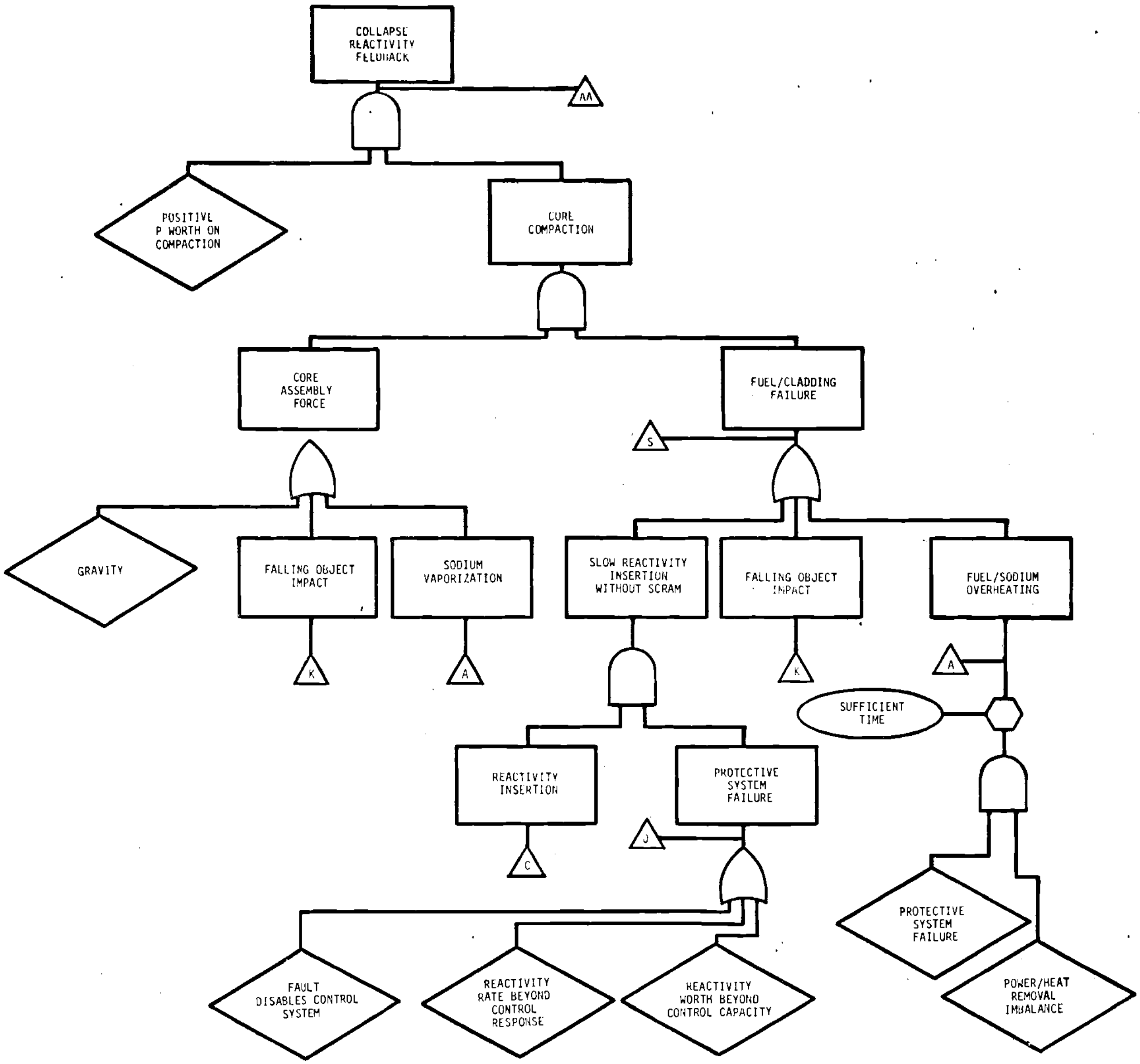

COLLAPSE REACTIVITY FEEDBACK
AA
POSITIVE P WORTH ON COMPACTION
CORE COMPACTION
CORE ASSEMBLY FORCE
FUEL/CLADDING FAILURE
S
GRAVITY
FALLING OBJECT IMPACT
SODIUM VAPORIZATION
SLOW REACTIVITY INSERTION WITHOUT SCRAM
FALLING OBJECT IMPACT
FUEL/SODIUM OVERHEATING
K
A
K
A
SUFFICIENT TIME
REACTIVITY INSERTION
PROTECTIVE SYSTEM FAILURE
J
C
PROTECTIVE SYSTEM FAILURE
POWER/HEAT REMOVAL IMBALANCE
FAULT DISABLES CONTROL SYSTEM
REACTIVITY RATE BEYOND CONTROL RESPONSE
REACTIVITY WORTH BEYOND CONTROL CAPACITY

FIGURE 1b.

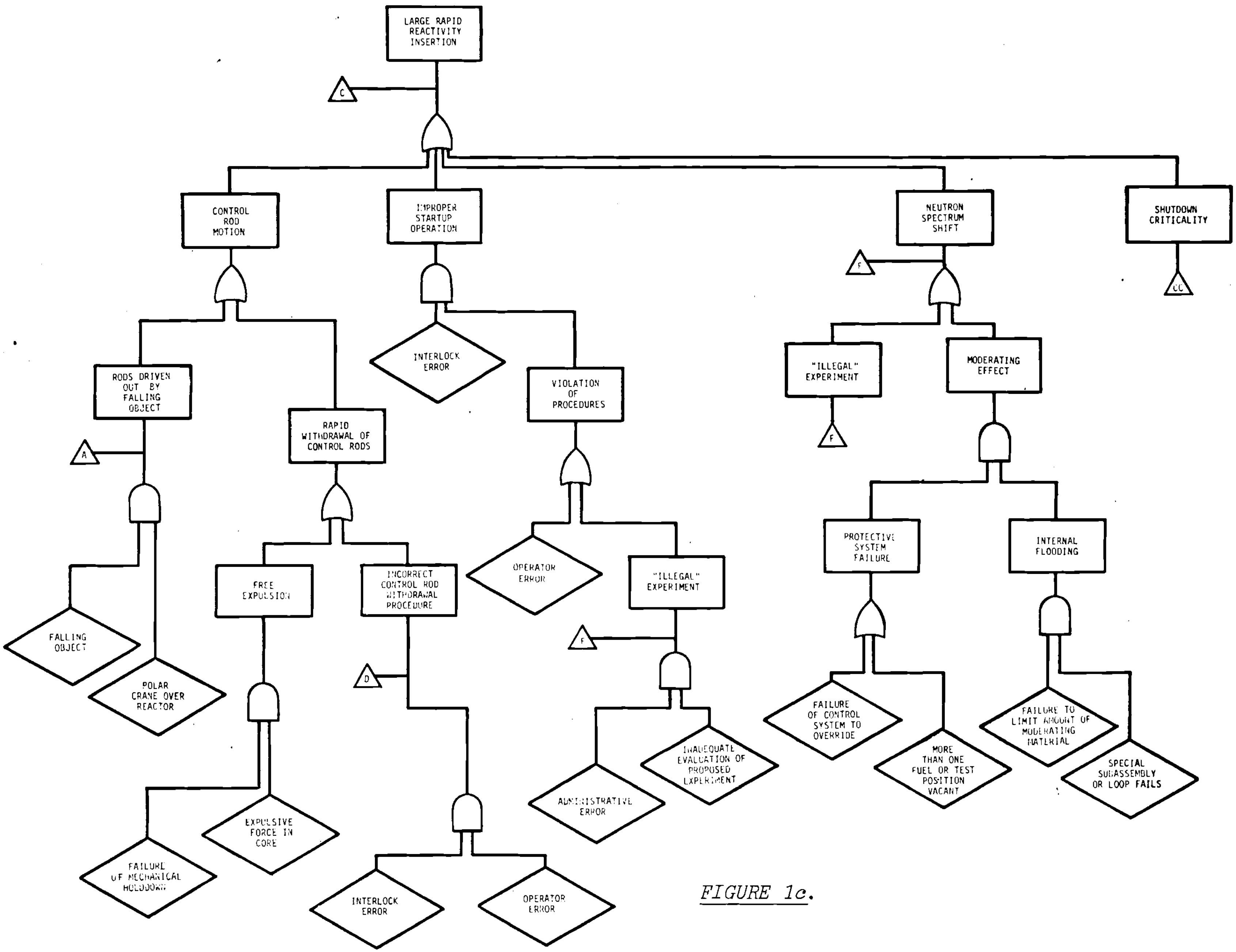

LARGE RAPID REACTIVITY INSERTION
C
CONTROL ROD MOTION
IMPROPER STARTUP OPERATION
NEUTRON SPECTRUM SHIFT
SHUTDOWN CRITICALITY
CC
RODS DRIVEN OUT BY FALLING OBJECT
A
FALLING OBJECT
POLAR CRANE OVER REACTOR
RAPID WITHDRAWAL OF CONTROL RODS
FREE EXPULSION
EXPULSIVE FORCE IN CORE
INCORRECT CONTROL ROD WITHDRAWAL PROCEDURE
D
INTERLOCK ERROR
OPERATOR ERROR
INTERLOCK ERROR
VIOLATION OF PROCEDURES
OPERATOR ERROR
"ILLEGAL" EXPERIMENT
F
ADMINISTRATIVE ERROR
INADEQUATE EVALUATION OF PROPOSED EXPERIMENT
F
"ILLEGAL" EXPERIMENT
F
MODERATING EFFECT
PROTECTIVE SYSTEM FAILURE
FAILURE OF CONTROL SYSTEM TO OVERRIDE
MORE THAN ONE FUEL OR TEST POSITION VACANT
INTERNAL FLOODING
FAILURE TO LIMIT AMOUNT OF MODERATING MATERIAL
SPECIAL SUBASSEMBLY OR LOOP FAILS

FIGURE 1c.

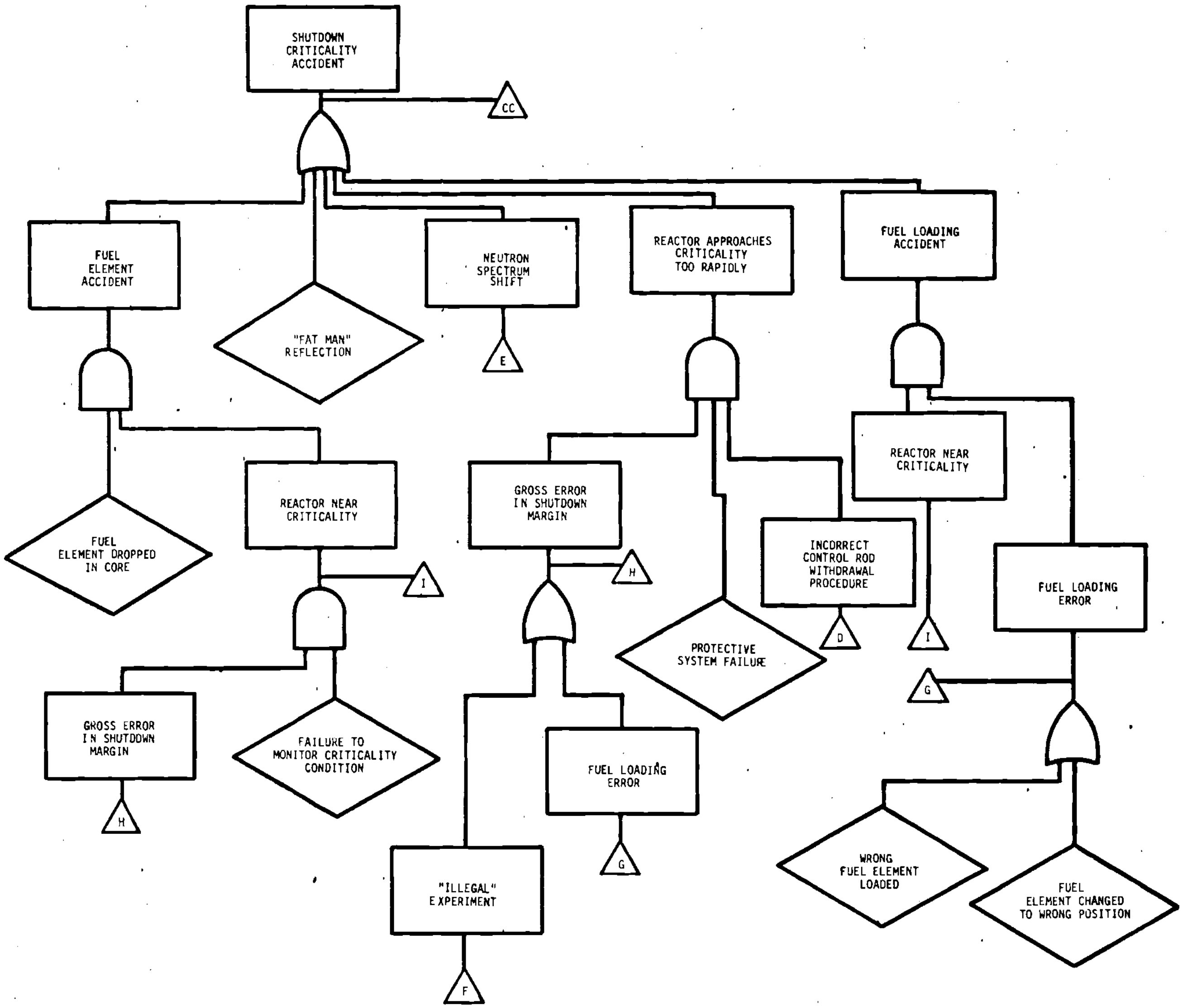
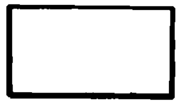

SHUTDOWN CRITICALITY ACCIDENT
CC
FUEL ELEMENT ACCIDENT
"FAT MAN" REFLECTION
NEUTRON SPECTRUM SHIFT
E
REACTOR APPROACHES CRITICALITY TOO RAPIDLY
FUEL LOADING ACCIDENT
FUEL ELEMENT DROPPED IN CORE
REACTOR NEAR CRITICALITY
I
GROSS ERROR IN SHUTDOWN MARGIN
H
INCORRECT CONTROL ROD WITHDRAWAL PROCEDURE
D
PROTECTIVE SYSTEM FAILURE
REACTOR NEAR CRITICALITY
I
FUEL LOADING ERROR
G
GROSS ERROR IN SHUTDOWN MARGIN
H
FAILURE TO MONITOR CRITICALITY CONDITION
FUEL LOADING ERROR
G
"ILLEGAL" EXPERIMENT
F
WRONG FUEL ELEMENT LOADED
FUEL ELEMENT CHANGED TO WRONG POSITION

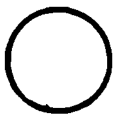
FIGURE 1d.

TABLE 1. Fault Tree Symbols

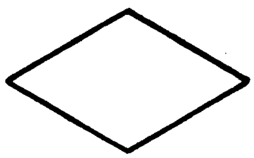
Events:



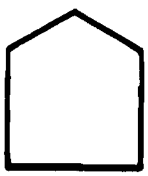
An event (usually a fault or malcondition) expressed in functional terms.



An event of which fault sequence is terminated for lack of information or consequences.



An event described by a basic component or part failure (these are the "independent" events).

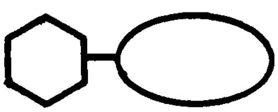


An event that is normally expected to occur.

Gates:

AND gate - Requires coexistence of all gate inputs for output.

OR gate - Requires anyone gate input for output.



Inhibit Gate - If input event occurs and the condition is satisfied an output event will be generated.

Other Symbols:

Transfer symbol used to transfer an entire sequence of events to another part of tree.

4.0 FUEL PIN GAP COEFFICIENT AND SODIUM FILM COEFFICIENT

4.1 SODIUM FILM COEFFICIENT

Several transients were investigated to assess the importance of the sodium film coefficient. A pseudo-coefficient was used as the sodium is stagnant in the NPTR. The value used corresponds to the value estimated for a conduction model.

NUTIGER-II was used to compare the pseudo-film coefficient with the conduction model. No significant difference resulted as temperature differences were in terms of 1 or 2 °F in the sodium.

A constant film coefficient of 30,000 BTU/hr ft^2 °F was used in NUTIGER-II and MELT-II. A variable film coefficient was used in FORE-II. Its magnitude ranged from 50,000 at 160 °F to 25,000 at Na temperature just below 1650 °F. The choice of film coefficient in the NPTR is relatively unimportant, as the dominating link in the chain is the gap coefficient.

4.2 FUEL PIN GAP COEFFICIENT

The NPTR will have essentially green fuel. (No burnup and no sintering). Its gap coefficient could be as low as 300 BTU/hr ft^2 °F at 170 °F. For the normal zero power operation of the NPTR, this will not create any fuel temperature problems.

During a power transient the gap coefficient will increase due to fuel swelling. Figures 2, 3, and 4 show the correlation between the hot spot fuel temperature and the gap coefficient associated with the hot spot segment. This calculation was made with FORE-II. Results are compatible with hand calculations. A gap coefficient of 1000 BTU/hr ft^2 °F is calculated for the hot spot node just before melting begins. As melting progresses, the gap coefficient increases quite rapidly as the fuel expands out to the clad. The coefficient may increase to 5000 BTU/hr ft^2 °F.

The use of a constant gap coefficient of 1000 does not give appreciable difference in the fuel temperature or power trace when compared to the variable coefficient.

However, sodium temperatures are initially overestimated before fuel melting and then underestimated after fuel melting occurs.

If the sodium feedback coefficient gives a large negative feedback, the difference in sodium temperatures could create significant error. However, calculations have indicated that the sodium feedback coefficient does not appear to be an important feedback contributor. Also, the time to significant core distortion due to sodium expansion effects may be prematurely estimated when a large gap coefficient is used.

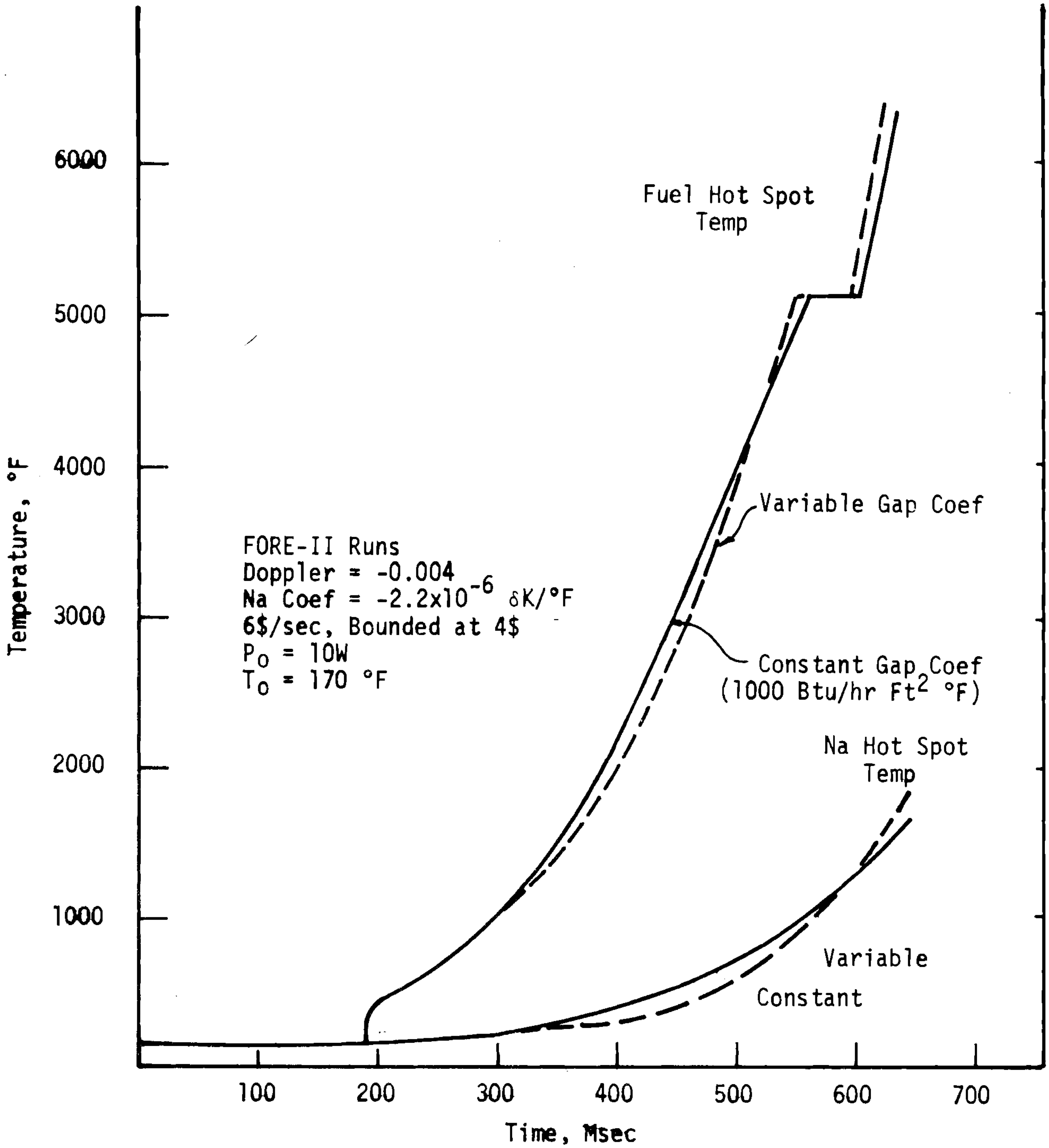


FIGURE 2. Transient Temperatures as Function of Fuel Pin Gap Coefficient

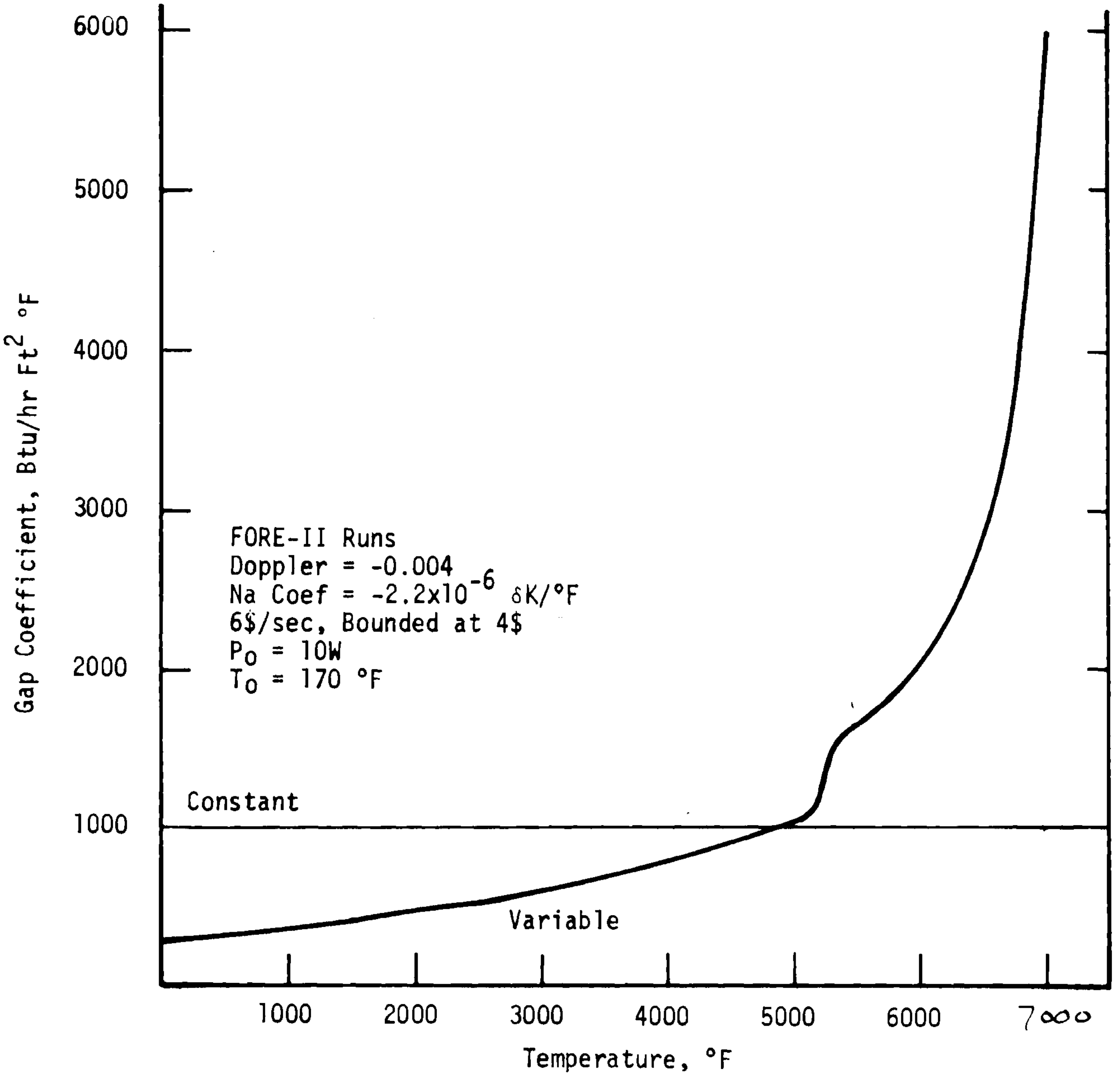


FIGURE 3. Hot Spot Gap Coefficient as Function of Hot Spot Temperature

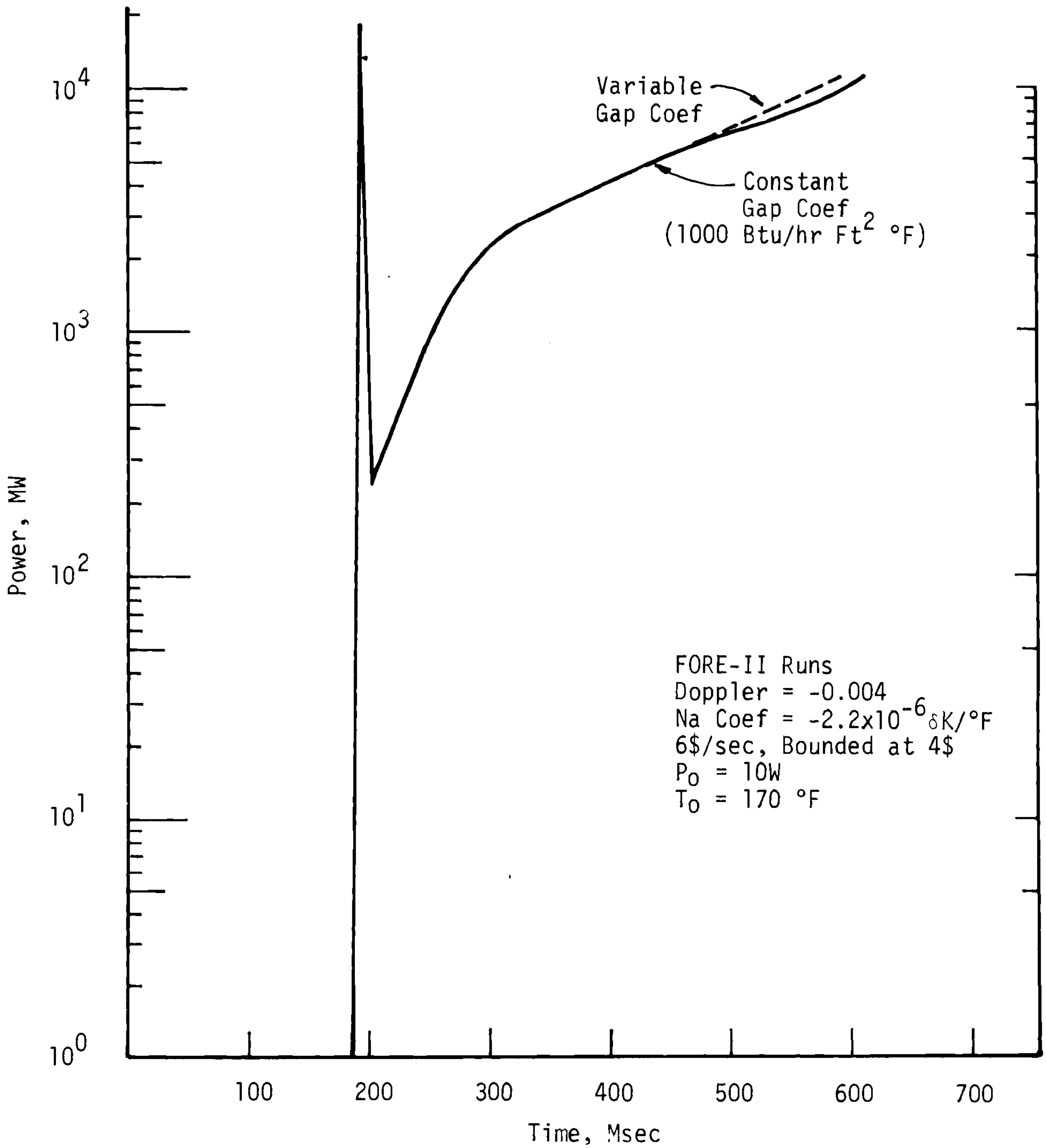


FIGURE 4. Transient Power as Function of Gap Coefficient

5.0 FEEDBACK EFFECTS

5.1 DOPPLER FEEDBACK

To assess the importance of large negative Doppler coefficient runs were made where TdK/dt was varied from -0.0025 to -0.004. (Figures 5 and 6).

FORE-II was used for this analysis as it allows the Doppler coefficient to be radially weighted. MELT-II has provisions for a complete spacial weighting of the Doppler effect, but at the present, additional information is necessary before the Doppler effect worth can be adequately described as to its spacial dependency.

A Doppler coefficient of -0.004 (compared to -0.0025) increases the time to hot spot fuel melting by 150 msec (25%) and decreases power at disassembly by a factor of 2. Sodium temperatures at disassembly time are slightly higher for the -0.004 Doppler. The additional time to disassembly allows more heat to be transferred to the sodium.

At the time of melting of hot spot fuel, the total energy input to the core from the transient and the average core temperature were constant and independent of the Doppler coefficient.

The Doppler will have little effect upon scram parameters if the scram system is required to prevent sodium from melting. A large Doppler will delay higher sodium temperatures, and for a bounded ramp, hold the fuel and sodium to substantially lower temperatures.

A large Doppler coefficient increases the time available for thermal expansion effects to provide other sources of negative (or positive) feedbacks. Up to 1.75$ Doppler feedback is available for a transient occurring at 170 °F to the hot spot fuel melting.

5.2 SODIUM FEEDBACK

The effects of sodium density change (sodium temperature coefficient) have been investigated. (Figures 7 and 8). Three cases were run with MELT-II where the sodium coefficient was varied from 0.0 to -2.2×10^{-6} $\Delta k/°F$. In these runs, the sodium coefficient was assumed to be a core average value with no spacial dependency. The Doppler was set at -0.004 to allow the maximum amount of heat to transfer to the sodium. From the results of these runs it appears that a large negative sodium coefficient could significantly reduce the power at the disassembly. Time to hot spot fuel melting could be increased 10%. However, sodium feedback would have little effect on scram parameters if sodium melting is to be prevented.

Sodium feedback at the time of hot spot fuel melting for the -2.2×10^{-6} $\Delta k/°F$ was over -0.75$.

Further investigations were then carried out to ascertain spacial effects. FORE-II allows the sodium feedback to be weighted axially. Using the sodium void worth curves from BNWL-760[2], axial weighting factors were obtained. For the equivalent base case; significantly less sodium feedback was obtained from FORE-II. Less than -0.10$ of sodium feedback was available.

MELT-II was modified to calculate sodium feedback by use of a completely spacially dependent weighting coefficient. Results from MELT-II indicated that sodium feedback could be positive, as it gave a feedback of +.05\$.

The use of an average sodium feedback coefficient appears to be a poor approximation. The portion of the core having a positive sodium feedback is the region of the core that experiences the larger temperature increase. This larger temperature increase enhances the positive portion's importance in relationship to the negative worth portion. Further studies are recommended to determine sodium density change reactivity effects. Until better information is available no attempt to take credit for the sodium density feedback coefficient should be made.

An area in which little information is available, but potentially may be very important, is feedback due to core axial and radial expansion. Calculations have indicated that a feedback of -0.19\$ will be available for a 0.1% expansion in cross-sectional area and 1.23\$ for 0.84% expansion. A 100 °F temperature increase in the sodium gives a sodium volume expansion of $\sim$ 1.5%. Assuming the axial and radial expansion due to sodium expansion is the same ($\Delta h = \Delta r$), this corresponds to 0.6% increase in cross-sectional area (Δr = 0.080 in.) or $\sim$ -1\$ feedback.

Clad axial expansion due to its thermal expansion coefficient is 0.040 in. per 100 °F temperature rise. Sodium axial expansion could be up to a factor of five larger depending upon the amount of radial expansion. Provisions could be incorporated into the fuel subassembly design which would require fuel movement out of the core upon axial sodium movement.

It is possible that up to 3-4$ of negative feedback will be available during a transient. This will greatly negate the effects of any conceivable accident.

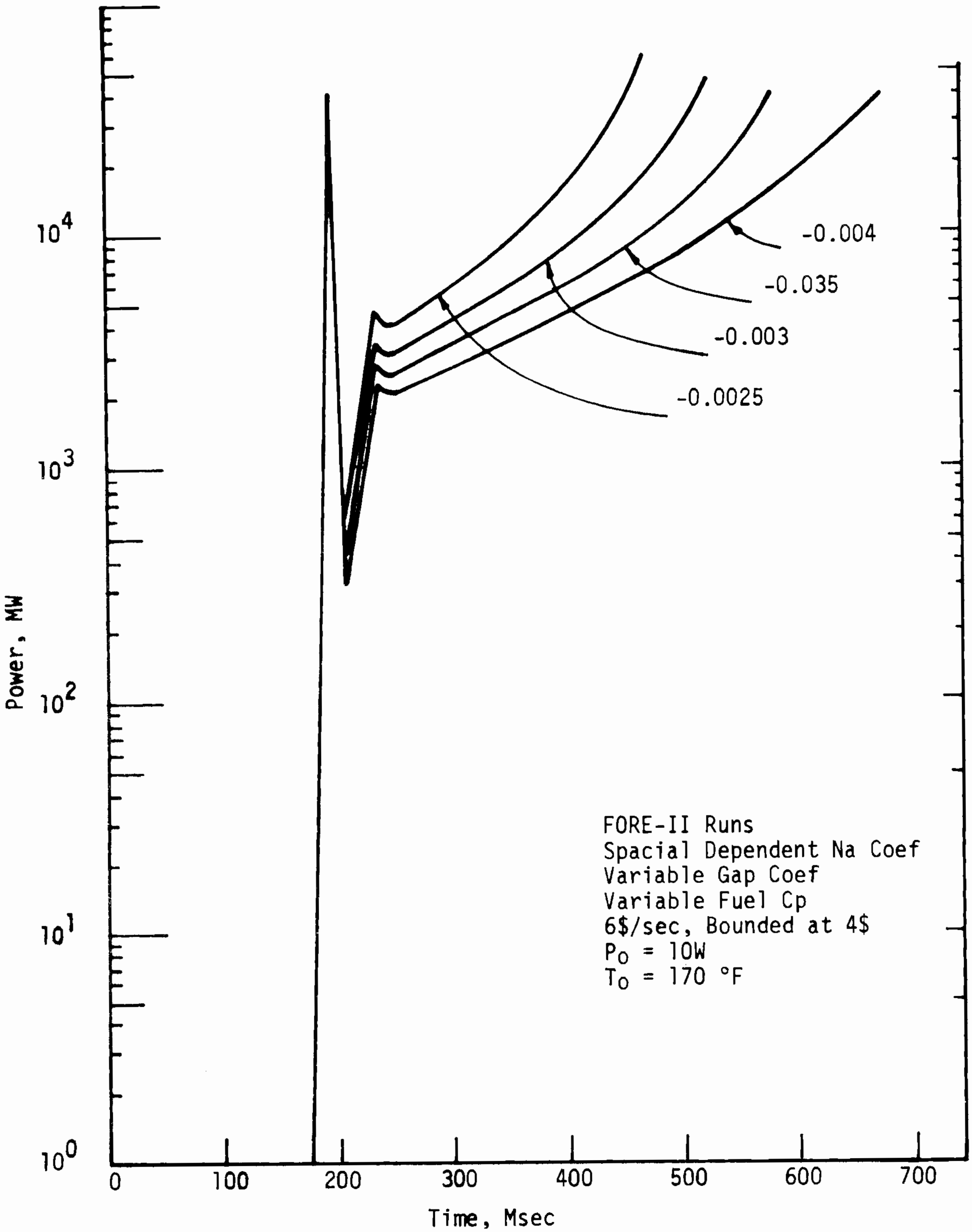


FIGURE 5. Transient Power as Function of Doppler Feedback

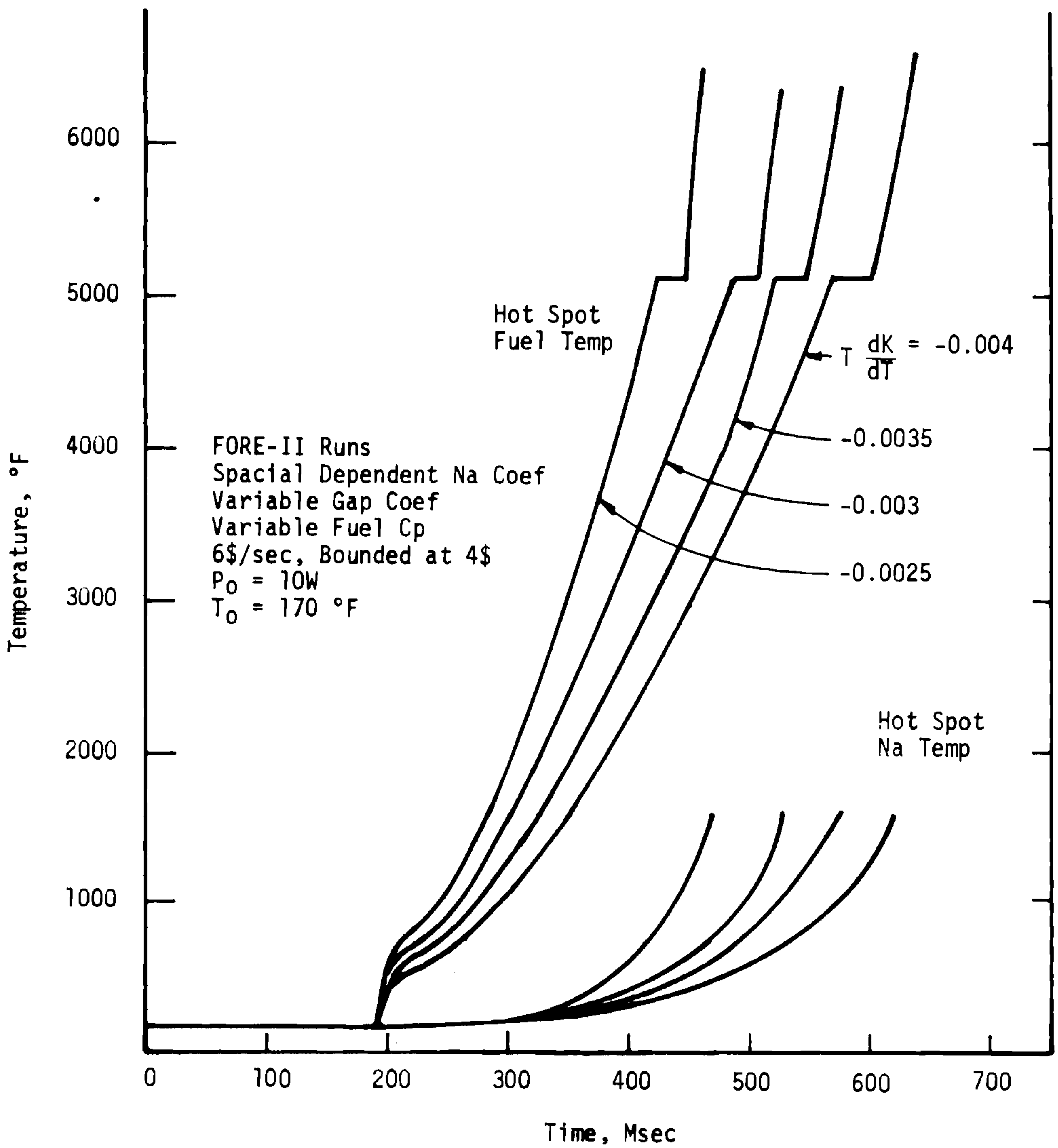


FIGURE 6. Transient Temperatures as Function of Doppler Feedback

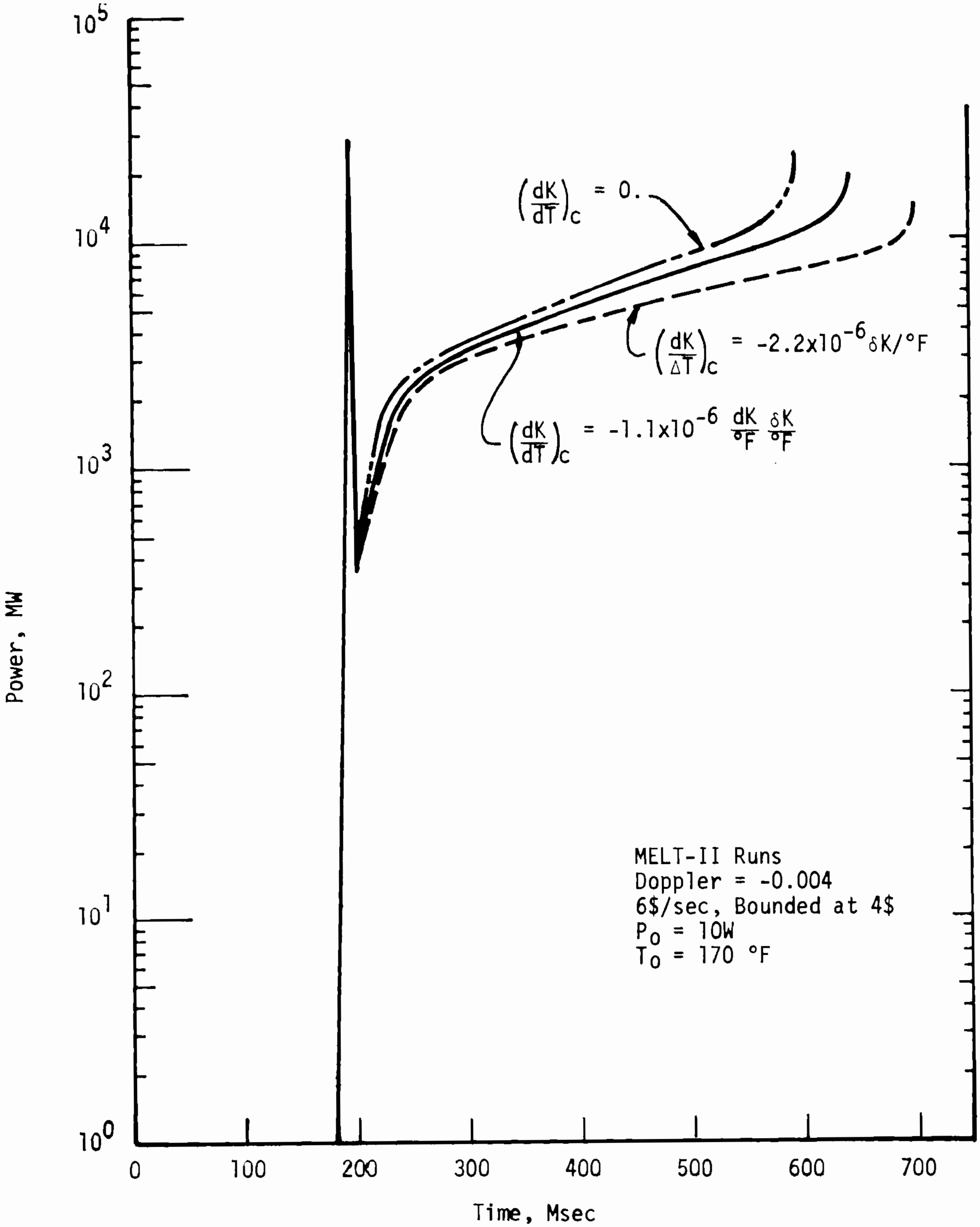


FIGURE 7. Transient Power as Function of Sodium Density Change Feedback

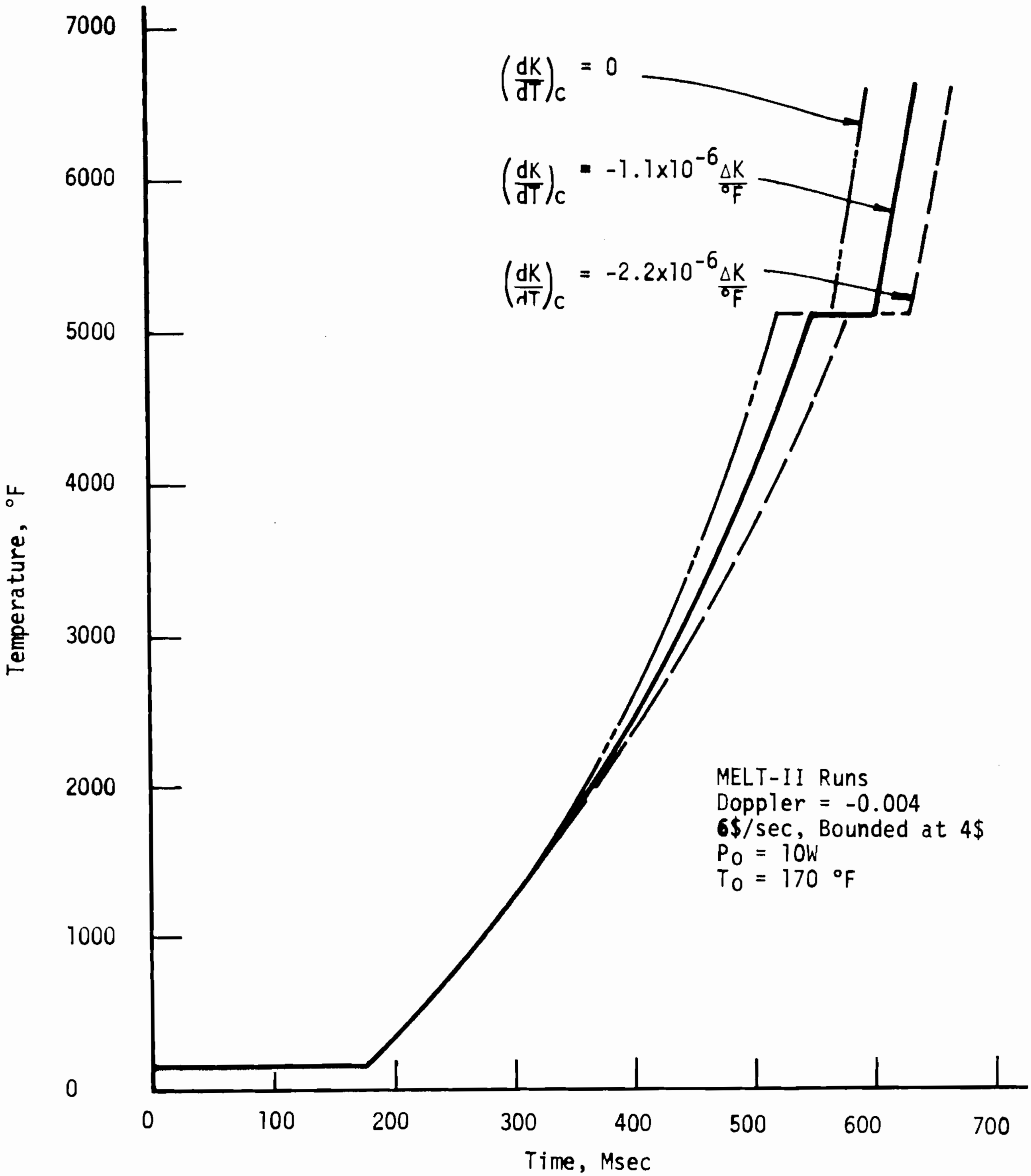


FIGURE 8. Hot Spot Transient Temperature as Function of Sodium Density Change Feedback

6.0 REACTIVITY TRANSIENTS WITHOUT SCRAM

A series of power transients were investigated where the reactivity insertion rate was varied from 2$/sec to 20$/sec. (Figures 9 and 10). This analysis was done with MELT-II.

Operating Power	10 W
Operating Temperature	170 °F
Doppler Coefficient (TdK/dT)	-0.004
Na Temp. Coefficient (Average)	-2.2×10^{-6} Δk/°F
Constant gap Coefficient	1000 BTU/hr-ft^2-°F

Several observations and conclusions were obtained from this analysis.

- Up to 2.75$ of Doppler turn around is available.
- For the range of ramps investigated sodium will melt.
- For ramp rates < 4$/sec, sodium may boil before fuel melts.
- Substantial time is available for a scram system to prevent fuel melting.
- A fast scram may prevent sodium from melting for ramps ≤ 6$/sec.
- Independent of ramp rate when 3.7$ of reactivity has been inserted into the core, hot spot fuel will begin to melt.
- Time to hot fuel melting is related to reactivity ramp rate by a hyperbolic relationship.
- Sodium feedback could be significant for slower ramp rates.

Except for power excursions from very large ramp rates, (> 50$/sec) sodium will be molten when the disassembly stage is reached. For a gap coefficient less than 1000 BTU/hr-ft^2-°F it is possible that the dividing range could be slightly lower, possible > 40\$/sec. For all rates of reactivity insertions for which mechanism have been identified in the NPTR, sodium would be molten.

For small ramps < 4\$/sec, sodium may reach vaporization temperatures before significant fuel melting has occurred. For ramps expressed in terms of ¢/sec, sodium will boil before fuel melting temperatures are reached. This leads to the possibility of Na becoming the working mechanism in the NPTR and potential exists for a sodium fire for limited accidents where the excursion is turned around before fuel melting.

It appears that sufficient time is available before fuel melting conditions are reached for a scram system to terminate the excursion for quite large ramps. To preclude core damage, a scram system that is fast enough to prevent sodium melting is desirable. Such a system appears to be feasible.

If the sodium melts it may not be possible to guarantee the integrity of the cans. Spilled sodium greatly increases the sodium fire probability.

Independent of the ramp rate, when 3.7\$ of reactivity have been inserted into the core, the hot spot fuel will have reached melting temperature. There is a hyperbolic relationship between hot spot fuel melting time and ramp rate.

The time from transient initiation to beginning of hot spot fuel melting may be expressed empirically as

$$t = \frac{\alpha}{\gamma}$$

where:

t = time, sec.

γ = reactivity ramp rate \$/sec.

α = constant which is a function of system physics, geometry, thermodynamic properties and feedback coefficients.

α = 3.7\$ for the test cases.

For a given reactivity ramp rate, the total feedback as a function of time is independent of the feedback coefficients. That is, for smaller feedback coefficients, the power increases more yielding higher temperatures, which gives an equivalent feedback. Therefore, it appears that there is a function:

$$t = \frac{\alpha ' f}{\gamma} \text{ (feedback coefficients)}$$

where:

α' = a constant independent of feedback effect.

f = a feedback term.

Analysis of runs where the Doppler coefficient were varied indicate that

$$f = (0.52 - 120.(-TdK/dT))$$

When the spacial dependence of the sodium temperature coefficient is taken into account, γ decreases to 3.31\$.

Therefore, the estimated time to hot spot fuel melting for a given ramp rate and Doppler coefficient is

$$t = \frac{3.31}{\gamma} \left(0.52 - 120. \left(\frac{TdK}{dT}\right)\right)$$

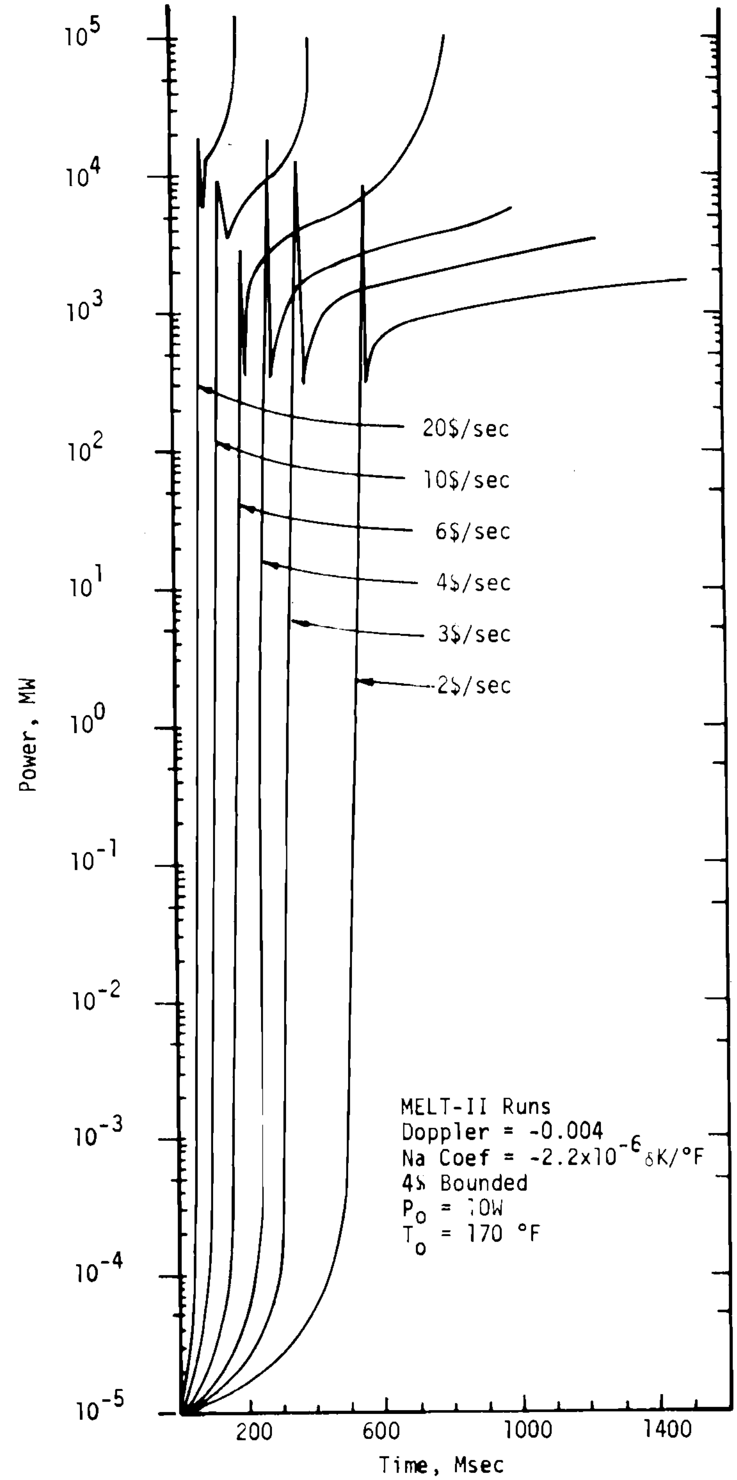


FIGURE 9. Transient Power as Function of Reactivity Insertion Rate

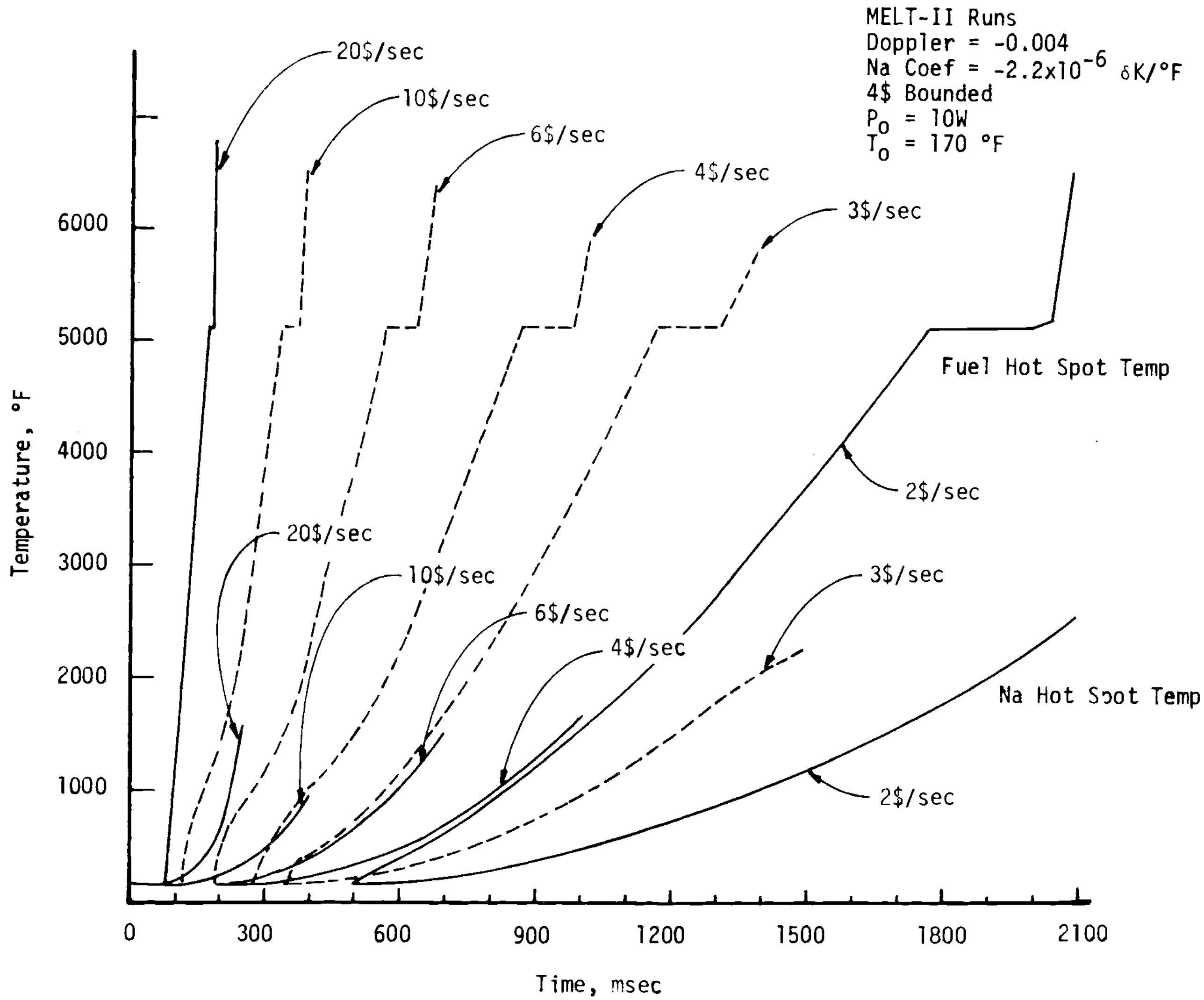


FIGURE 10. Transient Temperatures as Function of Reactivity Insertion Rate

7.0 POWER TRANSIENTS WITH SCRAM

To determine what capabilities a scram system would require to prevent core damage, a series of transients with scrams were run with NUTIGER-II. (Figures 11 and 12).

Response times and rod acceleration were varied.

Test Case Parameters

1. Nominal power = 10 W
2. Operating power = 170 °F
3. Doppler Coefficient = $-0.004\ T\ \frac{dk}{dT}$
4. 6$/sec, 4$ bounded ramp
5. 6.67$ scram worth
6. Na feedback = $2.2 \times 10^{-6}\ \Delta k/°F$
7. Response time = 50 to 200 mil sec
8. Rod accelerations = 1 and 2 G

Response Time	Rod Acceleration, G	Hot Spot Fuel Temp.,°G	HS Na. Temp.,°F
50	1	625	375
100	1	1000	∿600
200	1	2000	∿1200
50	2	170	170
100	2	580	350

A second analysis was performed using FORE-II and MELT-II. The requirement was placed on the scram system that the sodium was to be kept from melting for an instrument scram at power.

Core Model

1. Operating power = 10 W
2. Operating temperature = 170 °F
3. k_{eff} = 1.00 @ time zero

The transient was a 6$/sec ramp bounded at 4$ total insertion. This transient is more severe than any that has been identified. The following scram parameters adequately terminated the transient without sodium melting.

Scram Parameters

1. Rods 10 cm above core
2. 125% overpower factor
3. 100 msec scram system response time
4. Scrammable worth 10$
5. Rod acceleration, 3G constant for 3/4 stroke

If the rods were even with the top of the core instead of 10 cm above the core, the following could be gained.

1. 150% overpower factor, or
2. 130 msec response time, or
3. Scram worth of only 6.67$, or
4. Rod acceleration of 2G

With this scram system, fuel melting could be prevented for power excursions initiated by ramp rates in excess of 6$/sec. The practical upper bound appears to be 15$/sec.

A power excursion could occur during a power change. If the scram comes from a linear power instrument and it is assumed that the power must change by a factor of four before a scram trip occurs, the delay would be approximately 120 msec for a 6$/sec ramp. In addition, there are period scram backups which will also scram the reactor, but time delays may be in excess of 120 msec. If the reactor was scrammed under these conditions hot spot fuel could reach temperatures in excess of 1650 °F. However, the bulk temperature of the core would not exceed the sodium melting temperature. A sodium fire would be a distinct possibility under these conditions.

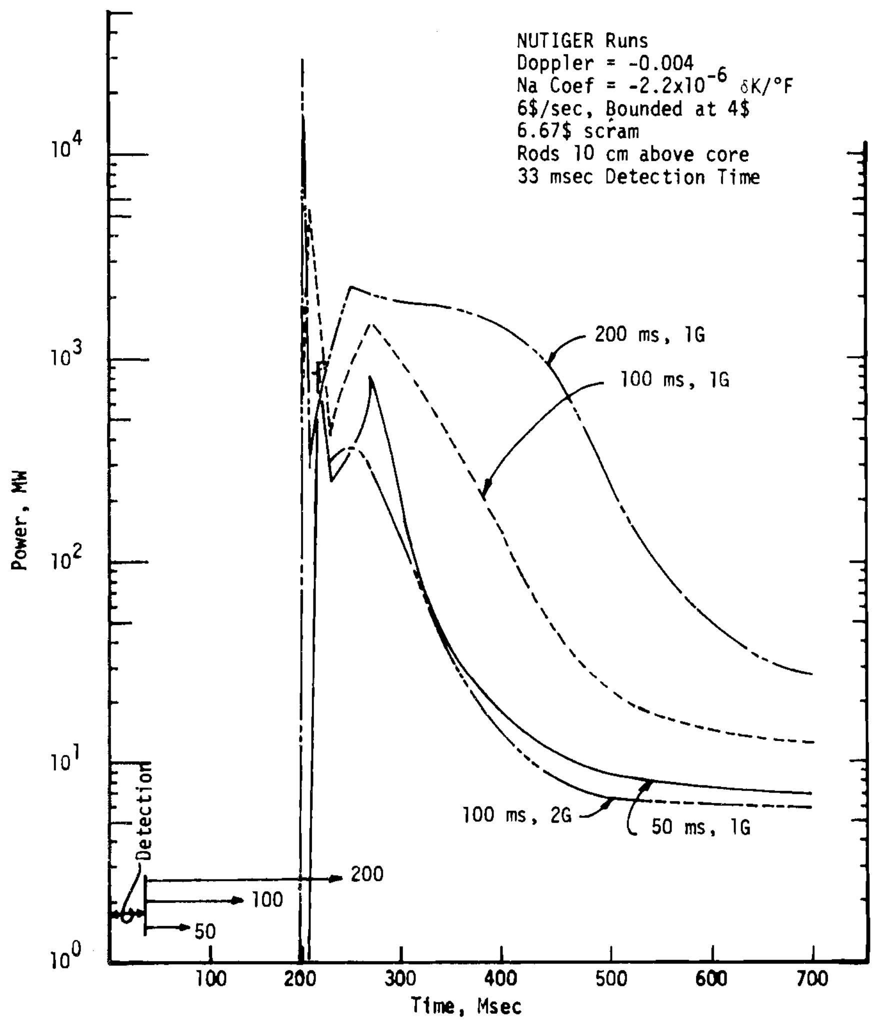


FIGURE 11. Transient Power as Function of Scram System Parameters

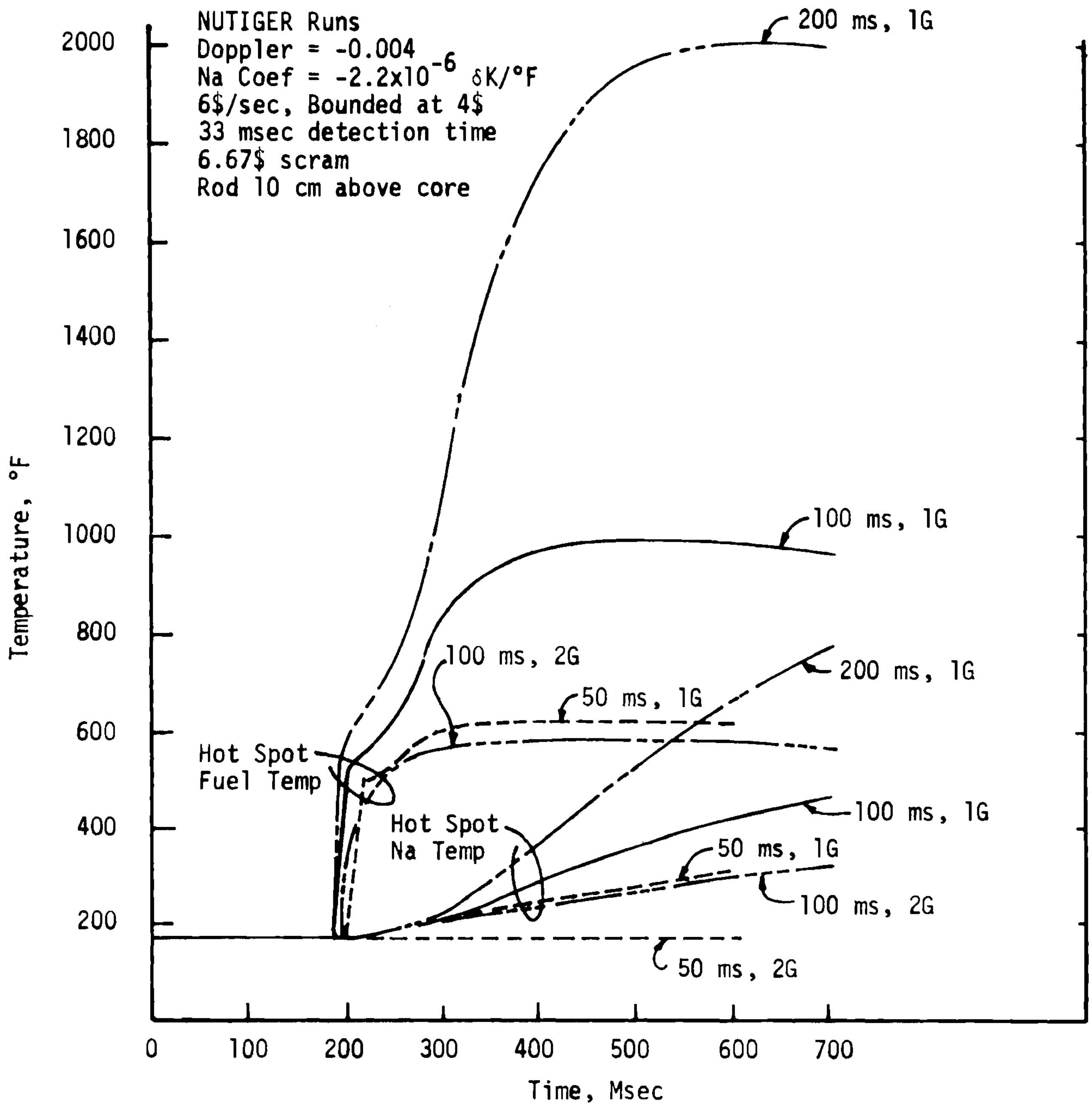


FIGURE 12. Transient Temperatures as Function of Scram System Parameters

8.0 SODIUM THERMAL EXPANSION

Sodium will play a very important part in any disassembly model for the NPTR. Except for very fast primary ramps, (in all probability at least greater than 20$/sec) there is sufficient time for heat to transfer to the sodium.

A. Padilla[3] has performed an analysis of sodium thermal expansion effect. His results may be summarized as follows:

- At an average sodium temperature of $\sim$ 250 °F all available in-core void space will be filled from expansion effects.
- Pressures in the restrained sodium will rise rapidly. At 400 °F restrained sodium would exert a pressure of 18,000 psi.
- Core distortion must result to relieve these pressures.

It is possible that the core could be designed to utilize sodium thermal expansion to force the physical removal of fuel from the core. Concepts are available that may successfully utilize sodium expansion, but none have been evaluated as yet. The assembly could also be designed to allow more expansion volume.

9.0 DESIGN BASIS ACCIDENT

The analysis of the preliminary fault trees and the parametric analysis led to formulation of a sequence of events that could occur during an accident and to the determination of areas needing further study. A second series of fault trees were generated as a result of this study.

Failure paths do exist which show the potential for small ramp rates to lead to violent core disassembly accidents. These paths can be broken in several places depending upon what happens to the sodium. It appears that a hypothetical fuel vapor explosion could be ruled physically incredible if sodium forces are sufficient to disassemble the core to a shutdown mode. However, pressure in the reactor cell that could result from a sodium fire may be substantial and may become the basis for containment requirements.

A family of fault trees shown in Figure 13 has been prepared to investigate the sequence of events which could lead to core disassembly. The potential threats to the integrity of the NPTR core range in order of increasing severity from (a) sudden expulsion of liquid sodium from driver subassemblies by thermal expansion of melted sodium; (b) sudden expulsion release of sodium vapor if the subassemblies are not ruptured by (a); (c) core disassembly molten fuel mixing with sodium if disassembly doesn't occur in (b); and (d) violent core disassembly fuel vaporization in the conventional Bethe-Tait accident.

The basic events necessary to get to the top event in fault trees (a) and (b) include a slow reactivity ramp with failure of protective action (reactor scram), or a transient ramp greater than 6$/sec with scram. The latter ramp when terminated by scram would deposit enough energy in the fuel to melt sodium but not fuel itself. Failure to scram with a reactivity insertion limited to about 2$ would lead to sodium melting. No fuel melting would occur unless significant positive feedback occurs as in fault tree (c) and then the total insertion would effectively exceed 2$. A bound is considered since 2.75$ is available as negative feedback from the Doppler effect before fuel melting. Although loss of coolant is shown in fault tree (a), the threat is not significant since ample time is available to unload the core prior to sodium melting.

If sodium vaporization does not effectively terminate the excursions, an unbounded reactivity insertion of < 2$/sec, and a fast insertion bounded at 2$, both without scram, could lead to the top event in (c) or (d) depending upon the magnitude of ramp reinforcement by the positive sodium voiding effect and/or fuel compaction feedback. The uncertainties in this effect must be resolved to determine if the excursion could initiate fuel vaporization. The top event in (c) could also be reached by a fast ramp of 2-20$/sec, with no scram and a total insertion in excess of 3$. The top event in fault tree (d) could also be reached by a very fast ramp > 20$/sec with 3$ total insertion. This is the fuel drop accident which could presumably be avoided by design.

It should be noted that no clear mechanism has been identified for the ramps considered here. The most realistic is the slow reactivity insertion without scram which would involve gross control rod withdrawal errors and protection systems failures.

A sequence of events has been outlined indicating possible modes or conditions that may exist during a DBA accident. The initiating mechanisms are not necessarily considered to be credible.

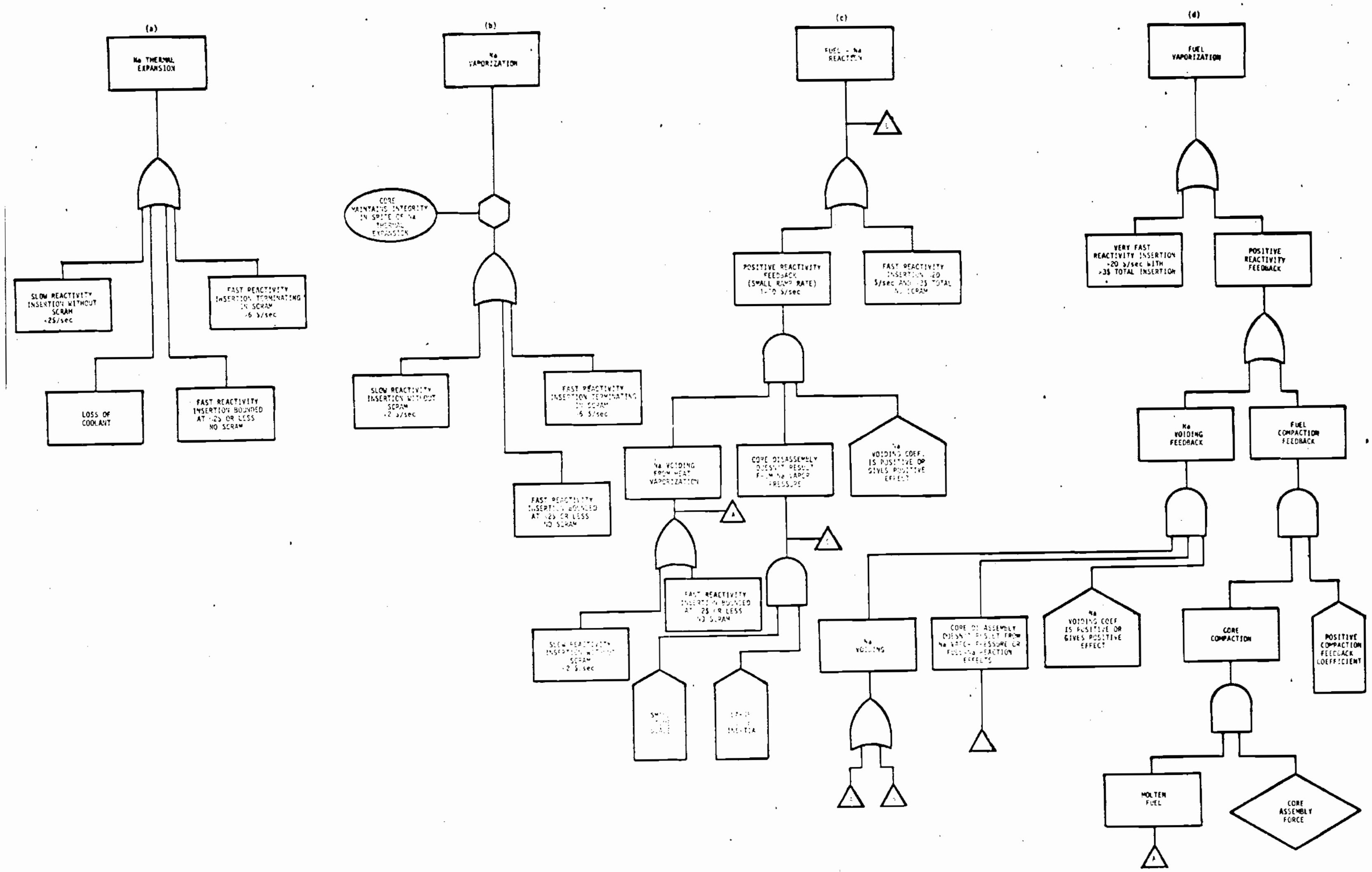


FIGURE 13. Core Disassembly Family of Fault Trees

APPENDIX A

CODE CHOICE

With the ultimate goal of determination of minimum scram system requirements and determination of the NPTR DBA, a parametric analysis was initiated for NPTR variables.

Three nuclear kinetics-heat transfer codes were available for this analysis: NUTIGER,[4] FORE-II,[5] and MELT-II[6]. Each code has slightly different capabilities, advantages and disadvantages.

NUTIGER is a heat transfer code (TIGER V)[7] to which a nuclear kinetics code has been added. Its biggest advantage is that is has been in production status at BNW for over two years and results are considered reliable. One of NUTIGER's initial drawbacks was that it was limited to problems of 400 nodal capacity or less.

FORE-II was recently obtained from GEAPO. Several cases had been run prior to the start of this analysis and the results appeared to be acceptable. However, no comprehensive study of the code had been made. FORE-II was written solely for the function of nuclear kinetics-heat transfer analysis and contains many desirable options. Its largest disadvantage appeared to be its exact production status.

MELT-II is a BNW code capable of carrying a transient study into the fuel slumping stage. At the beginning of the studies reported here it was still in the debugging stage.

It was decided to set up a base case and program it for each of the three codes. The results would be compared and the analysis continued with the code most applicable to the problem.

In the light of this approach, work was begun on the three codes. NUTIGER's nodal capacity was doubled and the capability to do multiple pin runs simultaneously was added. Provisions such as allowing more than one material to melt and increase in time libraries were added. A method allowing step changes in the time step was added.

Considerable difficulty was encountered in using FORE-II. Extensive debugging and modification was necessary. A series of revisions were obtained from GE and incorporated into the code. The code is now operational; however, some of its options are not functional on the CSC UNIVAC 1108 system or have not been checked out.

During the analysis MELT-II reached a stage where it could be incorporated into the study. To increase its usefulness, it was modified to include temperature variable material properties. Problems are still appearing as various capabilities of the code are checked out, but the basic logic of the code is functioning correctly.

Advantages, disadvantages, limitations are listed in Table A-1 for the three codes.

Due to the cumbersome nature of NUTIGER-II, it is probably best used as a heat transfer program and as a baseline comparison for the other two codes. FORE-II and MELT-II, due to their varied capabilities, are both necessary to do a complete study.

TABLE A-1. Code Properties

	NUTIGER	FORE-II	MELT-II
Problem Setup	Extensive Geometry Cards	Fixed Location	Name List
Problem Flexibility	Flexible Geometry Variable Voids	Fixed Geometry	Fixed Geometry
Problem Size	800 Node Limitation	7 Axial Nodes 10 Radial Nodes 3 Channels	20 Axial Nodes 10 Radial Nodes 10 Channels
Material Capabilities	10 Materials + 2 Coolant	Total of Five	Fuel, Coolant Clad, Bond
Output Formats	Poor	Good	Excellent
Time Step (Transient)	Constant	Power Variable	Feedback Variable
Run Time (Relative)	5	4	3
Scram Input	Poor	Fair	Excellent
Time Step (After Scram)	Constant	Good	Good
Hot Spot Pin Treatment	Feedback is Independent	Feedback is Independent	Feedback is Dependent
Doppler Feedback	Spacially Constant	Radially Variable	Spacially Variable
Na Temperature Feedback	Spacially Constant	Axially Variable	Spacially Variable
Na Voiding Feedback	None	Limited	Spacially Variable

APPENDIX B

BASE CASE COMPARISON

To assess the relative capabilities of each code and to compare results, a base case was formulated. The base case was for the conical core concept. Recent comparison studies have indicated that no significant differences in transient studies exist when compared to a vertical core geometry.

Results from the comparison runs are shown in Figures B-1 and B-2. Even though minor differences are evident, the results are quite similar. Powers at time of fuel melting vary over a range of 30% and hot spot fuel melting times vary by less than 7%. Considering that the power has changed 9 orders of magnitude and temperature increased 5000 °F, agreement within these bounds is quite satisfactory.

One noticeable result is that FORE-II predicts lower power and temperatures. Minor differences are explainable from several reasons.

- Different extrapolations and approximation techniques are used.
- It was not possible to mockup the base case exactly the same for each code.
- Each code uses a different method to predict or choose its time step.

NUTIGER uses a constant time step (or one which is capable of step changes). FORE-II sets its maximum time step by a power change criteria. MELT-II uses a complicated feedback method. Runs with NUTIGER where different time steps were used, indicated that as the time step was increased, power and temperatures were overestimated by significant amounts. At the time of these runs no sensitivity study had been made on the effect of time step on stability.

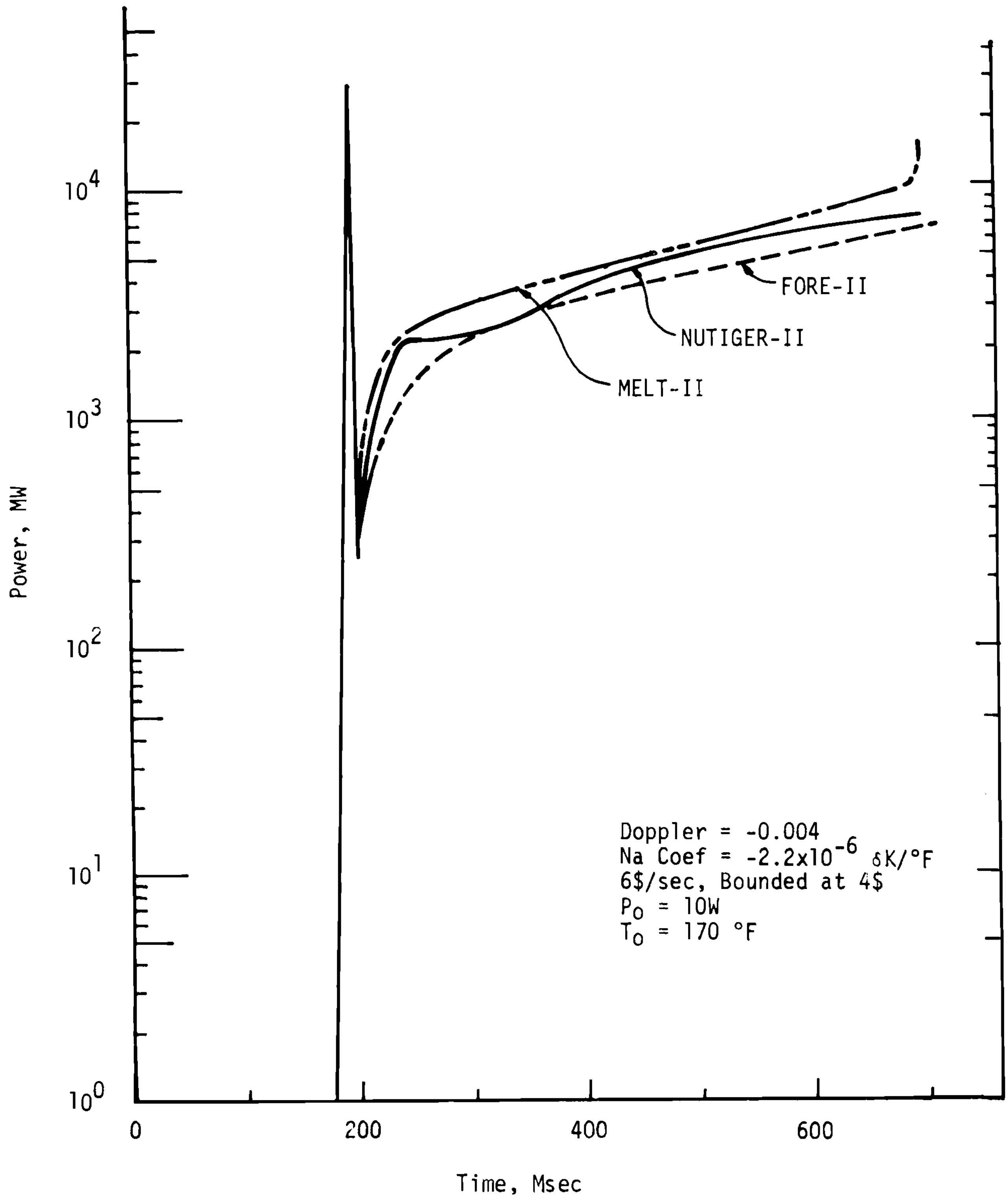


FIGURE B-1. Comparison of Transient Power From Three Computer Codes

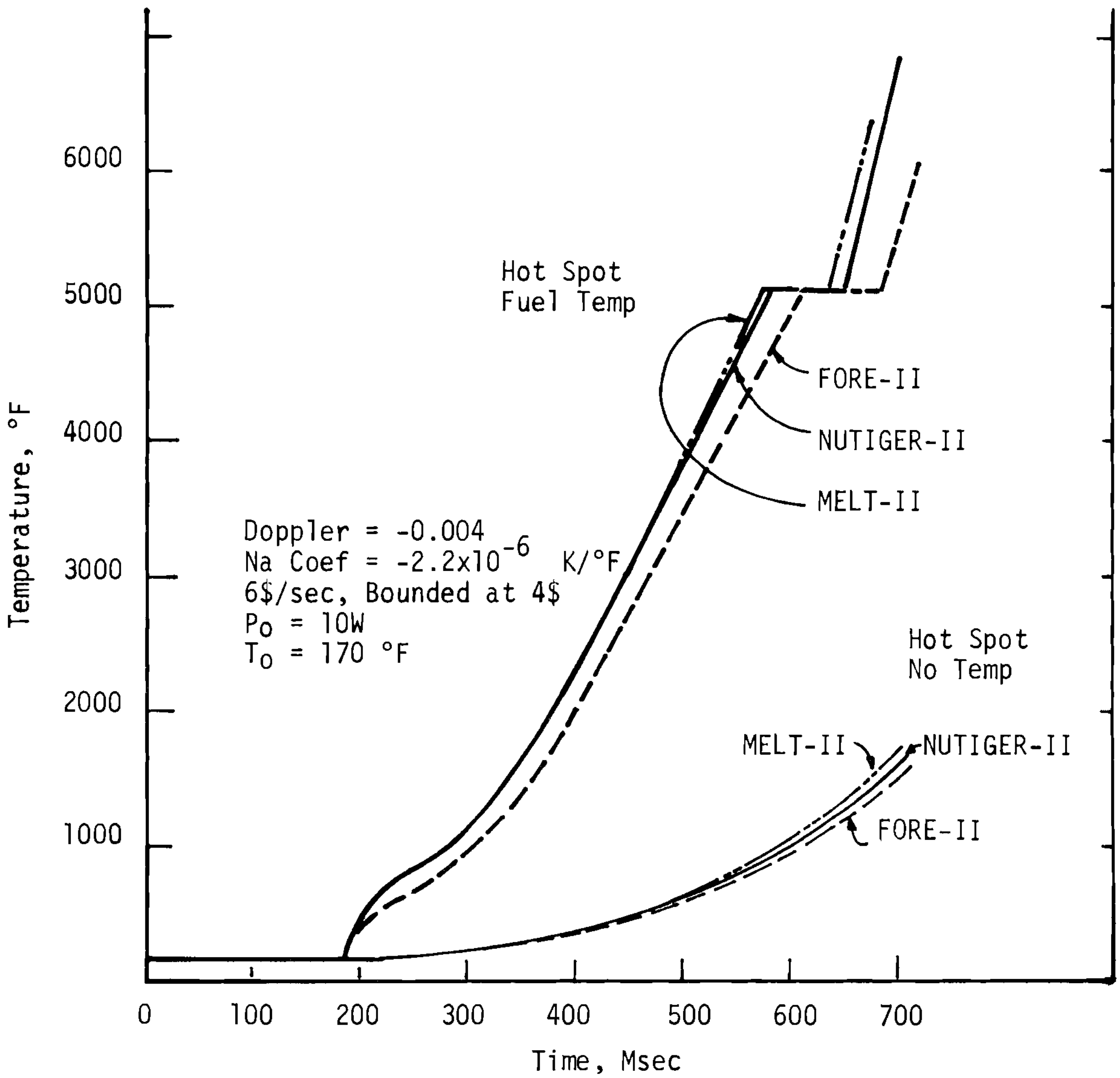


FIGURE B-2. Comparison of Transient Hot Spot Temperatures From Three Computer Codes

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