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AEC Research and
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UC-81, Reactors - Power
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thermal analysis of SM-1 core III

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APAE-105
Supplement 1

AEC Research and
Development Report
UC-81, Reactors, Power
(Special Distribution)

THERMAL ANALYSIS OF SM-1 CORE III

By:
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Approved by:
J. G. Gallagher

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ABSTRACT

This report covers the thermal analysis performed on the SM-1 Core III for both steady state and transient conditions. SM-1 Core III will be used as a test for Type 3 elements in a PM-2A Core. The steady state analysis indicated minimum DNBR's for both design and scram conditions are above the minimum criteria of 1.5. Local nucleate boiling was noted in the hot internal channels and lattice passage at scram power conditions. Loss of flow transient results indicate DNBR's above 1.5, insuring that the core is safe from burnout. Bulk boiling was noted in the hot channels and lattice passage at scram power condition.

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SUMMARY

A thermal analysis was performed under Subtask 3.6 of the FY 1962 Program for Engineering Support and Development of Army Pressurized Water Reactor Power Plants to insure the thermal safety of the SM-1 Core III. The SM-1 Core III will be used as a test for Type 3 elements in a PM-2A Core. The fundamental criterion for acceptable thermal design is the minimum departure from nucleate boiling ratio (DNBR). The minimum DNBR at design power conditions and scram power conditions for concurrent transient and steady state analyses is currently specified at 1.5.

The ART-02 code formerly used in loss of flow transient analyses was updated by utilizing the ART-04 code. Thermal conditions were investigated in the lattice passage between two stationary elements.

The steady state thermal analysis indicates that the SM-1 Core III will operate safely at design conditions of 10.77 Mw and scram power 13.45 Mw with minimum DNBR's above 1.5. The minimum DNBR for steady state conditions at 13.45 Mw is 3.53. Within the lattice passage the minimum DNBR is 3.7. For a loss of flow transient the minimum DNBR is 3.7 and 3.4 in the lattice passage. Limited amounts of local nucleate and bulk boiling were found in this core. No boiling was found in the nominal channels analyzed.

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THERMAL ANALYSIS OF SM-1 CORE III

1.0 INTRODUCTION

A thermal analysis using Type 3 elements in a 37 element PM-2A configuration was performed to determine whether any problems or hazards are present in the operation of the SM-1 Core III. The SM-1 Core III is to be a 37-element core as a test of a Type 3 Core [1] for use in the PM-2A.

The SM-1 configuration described in APAE-105 [5] was reduced to 37 elements by inserting dummy elements in the vacated element positions. The core heat flux is 14 percent greater than the full 45 element Type 3 SM-1 core due to a decrease in effective core heat transfer area resulting from eight less elements and a five rod bank position change. Each of the dummy elements was orificed to receive approximately 75 percent less flow than its respective position in the SM-1 Type 3 core. This resulted in modification to the present core flow distribution and core pressure drop.

The selection of a minimum DNBR for safe operation of the SM-1 Core III is in accordance with the published criteria. A minimum DNBR of 1.5 has been established for steady state and loss of flow transient analysis at design conditions.

The most critical element positions within the core have been analyzed as well as the most critical lattice passage between two stationary elements. Steady state analyses were performed using the WAPD Code STDY-3 [2] and transient analyses using the WAPD Codes ART-02 [3] and ART-04 [4].

2.0 METHODS OF ANALYSIS

Core flow distribution and hot channel factors were established for use in the STDY-3 code, as described in APAE-105 in the steady state analysis.[5] Radial and axial power distributions for SM-1 Core III are presented in reference (11).

A loss of flow transient analysis was performed using the ART-02 code and its modification ART-04. Both of these codes subject the reactor to a variation in reactivity due to control rod motion if requested on loss of pump. Reactor kinetic calculations were performed to determine core power as a function of time. The ART-04 code updates the ART-02 by utilizing an advanced slip flow model instead of a fog flow model. The important correlations and equations used in both of these codes are presented in APAE-105.[5]

3.0 CORE HYDRAULICS

3.1 Core Pressure Drop

A discussion of the core pressure drop and the effects of dummy elements are given in APAE-114.[1] A summary of the pressure drop analysis is repeated in Table-1 for the reader's convenience.

TABLE-1
SUMMARY OF PRESSURE DROP ANALYSIS

Δ p Core Ft-H_2O	Total Flow Thru 8 Corner Positions, GPM	Flow Thru 37 Elements, GPM	Lattice Flow, GPM	By-Pass Flow, GPM	Pump Flow, GPM	Pump Head, Ft-H_2O
5.537	0	3165	685	315	4165	28.39
5.355	80	3113	674	315	4182	28.19
5.180	160	3062	663	315	4200	28.03
5.018	240	3013	653	315	4221	27.87
4.75	363	2932	635	315	4245	27.60

3.2 Core Flow Distribution

A calculation was made to establish the increase in flow through the individual fuel elements caused by orificing the flow through each of the eight dummy elements to 10 gpm. The final SM-1 Core III flow distribution used in the thermal analysis is shown on Fig. 1.

4.0 POWER DISTRIBUTION

In order to place the analysis of the SM-1 Core III on the most conservative basis, both steady state and loss of flow transient analyses were performed at the scram power level. Scram power level is 13.45 Mw for the SM-1. Core average heat flux, peak heat flux, core heat transfer areas for both the SM-1 Type 3 core and SM-1 Core III are shown in Table 2.

	10.0	13 59.7	14 63.3	15 61.2	10.0	
10.0	22 71.3	23 95.2	100.2	25 93.7	26 70.9	10.0
31 58.7	32 93.1	94.7	34 107.1	35 106.8	36 92.7	37 60.4
41 61.2	100.6	43 113.7	100.2	45 111.6	97.8	47 66.8
51 62.2	52 91.7	53 103.4	54 112.7	94.5	56 93.9	57 70.0
10.0	62 72.1	63 98.5	100.5	65 92.3	66 72.2	10.0
	10.0	73 62.2	74 62.6	75 53.3	10.0	

Note: Values shown in element positions are in GPM
3113.9 - GPM total internal fuel element flow plus
674 - GPM lattice flow
80 - GPM total dummy element flow

Figure 1. SM-1 Core III Flow Distribution

TABLE 2
CORE POWER FOR SM-1 CORES

	Power, Mw	Core Heat Transfer Area, ft^2	Core Average Heat Flux, Btu/hr-ft^2	Core Peak Flux, Btu/hr-ft^2
SM-1 Type 3 Core	10.77	589.32	64,557	257,800
SM-1 Core III	10.77	491.7	77,369	299,743

The average and local peaking factors were used to determine the hot channel factors for both the nominal and hot channel. A description of how these peaking factors are used in the thermal analysis is contained in the literature (5).

The average and local radial peaking factors were also used to select the most critical element position within the SM-1 Core III with high burnup elements SM-2A and SM-2B in positions 37, and 51, respectively. A DNBR index was established using these factors and the individual element flow rates to determine the relative values of DNBR between elements. The most critical location was found to be at element position 37.

A study was performed to determine the effect that variations in lattice channel spacing between two stationary elements have on power peaking in the SM-1 Core III. A two-dimensional diffusion theory code was utilized in the analysis for channel spacing of 0.123 in. (nominal channel), 0.148 in. (average maximum channel) and 0.150 in. (absolute maximum channel). The hottest spot occurred in the corner of each element. The results of this analysis are listed in Table 3.

TABLE 3
LATTICE PASSAGE POWER PEAKS

Channel Spacing	Relative Power	Increase in Rel. Power Over That Of Nominal Channel	% Increase in Rel. Power Over That of Nominal Channel
0.123 in. (nom. channel)	1.317	0	0
0.148 in. (Ave. max.)	1.357	.04	3.04
0.150 in. (abs. max.)	1.369	.052	3.95

The percent increase in relative power was applied to the radial power peaks in the nominal channels to obtain the peaking in the lattice passage. These power peaks are multiples of the effects on DNBR presented in Section 6.

5.0 HOT CHANNEL FACTORS

The mechanical hot channel factors used in the steady state and transient analysis include average and local deviations for meat length, uranium content and clad thickness. The preceding factors were calculated by methods described in the literature [5].

Plate spacing appears in the hot and nominal channel description as an average and local dimension, not as an individual factor. The average and local dimension for the nominal internal channel and lattice passage is 0.123 in. A ripple growth factor of 1.3 was applied to the stationary element internal channel local dimension to obtain a maximum local dimension of 0.136 in.

A determination of the lattice passage spacing involves the separation between two elements. Worst channel spacing is then a function of core support structure, and box machining and centering tolerances as well as outer fuel plate rippling tolerances. Because of the greater number of dimensional tolerances involved, the lattice passage is, therefore, a more critical area in thermal and hydraulic considerations than the internal channels. After ripple growth, the geometry gives a maximum local lattice passage spacing of 0.149 in. and a minimum local average dimension of 0.102 in.[6]

Channel-to-channel maldistribution factors were obtained from single element flow testing of Type 3 elements.[7]

To allow for additional conservatism in the analysis, a nuclear uncertainty, a core power generation factor (to account for the fraction of heat liberated in the fuel plates) and instrument tolerances were included. All of these factors are described in reference (5).

6.0 PERFORMANCE OF SM-1 CORE III

6.1 Steady State Analysis

A steady state analysis was performed on the 10 most critical elements within the SM-1 Core III.[1] In addition, a steady state analysis was performed on a lattice passage separating two stationary elements.

At design conditions, reactor power was 10.77 Mw with an average core heat flux of 77,369 Btu/hr-ft^2 and inlet temperature of 431.7°F. At scram power conditions, reactor power was 13.45 Mw with an average core heat flux of 93,388 Btu/hr-ft^2 and inlet temperature of 429°F.

TABLE 4

SUMMARY OF THE RESULTS OF THE STEADY STATE THERMAL ANALYSIS OF SM-1 CORE III

Core Position	Core Power, Mw	Nominal Channel-G lb/hr-ft^2	Hot Channel Bulk Outlet Temp, $^{\circ}$F	Hot Channel Max Plate Surface, $^{\circ}$F	Minimum DNBR	Hot Channel Local Boiling	Hot Channel Bulk Boiling	% Quality
37	10.77	6.246×10^5	544	576.4	4.365	J=L	0	0
	13.45		565	577.0	3.530	J=3	J=19	2.1
51	10.77	6.461×10^5	529	576.3	4.493	J=5	J=0	0
	13.45		555	577.0	3.652	J=3	J=22	.4
62	10.77	7.460×10^5	515	576.53	4.406	J=6	J=0	0
	13.45		565	576.93	3.530	J=3	J=19	2.1
66	10.77	7.495×10^5	504	576.34	4.626	J=0	J=0	0
	13.45		522	576.88	3.788	J=5	J=0	0
31	10.77	6.097×10^5	530	576.22	4.935	J=5	J=0	0
	13.45		557	576.75	4.010	J=3	J=0	0
75	10.77	5.540×10^5	542	576.30	5.153	J=22	J=0	0
	13.45		565	576.83	4.174	J=3	J=20	1.4
15	10.77	6.354×10^5	541	576.30	5.068	J=22	J=0	0
	13.45		565	576.90	4.103	J=3	J=20	1.1
33	10.77	10.80×10^5	465.2	576.1	5.001	J=0	J=0	0
	13.45		468	576.6	4.076	J=0	J=0	0
44	10.77	11.42×10^5	471	576.7	5.090	J=0	J=0	0
	13.45		475	577.2	4.197	J=19	J=0	0
34	10.77	11.15×10^6	474	564.7	5.619	J=0	J=0	0
	13.45		479	576.6	4.705	J=0	J=0	0

* Axial increment measured from the bottom of the core at which local boiling commences

The results of the steady state analysis for the internal channels are shown on Table 4. These results indicate minimum DNBR's at design conditions are above 1.5, the currently specified minimum criteria for steady state and loss of flow transient analysis.(5)

The high burnup elements, located in positions 37 and 51, indicate both local nucleate and bulk boiling at scram power in their hot channels. Peripheral elements 75, 15 and 62 also indicate both local nucleate boiling and bulk boiling at scram power. The average channels of these elements, which were analyzed at design and scram power, did not show bulk boiling. At design power, 10.77 Mw, local nucleate boiling occurred in all elements analyzed except the control rod positions 33 and 44 and stationary element positions 34. Element positions 33, 44 and 34 did not show either bulk or local nucleate boiling due to higher element flow rates than those of the peripheral elements and better internal channel-to-channel flow distribution.

Figures 2 and 3 show for element position 37 at design conditions and scram power, the variation of meat centerline temperature, plate surface temperature and bulk water temperature with position for the hot internal channel. The inception of local nucleate boiling and bulk boiling is noted for each case.

The results of the steady state analysis for a lattice passage at scram power 13.45 Mw are listed on Table 5. The most critical lattice passage was assumed to be located between element position 37 and its adjacent stationary element. No interconnecting flow of the lattice passage analyzed with the rest of the lattice was assumed. Figure 4 shows the thermal conditions within the lattice passage.

TABLE 5

STEADY STATE RESULTS FOR SM-1 CORE III LATTICE PASSAGE

Core Power, Mw	Lattice Passage Flow, lb/hr-ft^2	Outlet Temp., ^oF	Plate Temp. Adjacent to Passage, ^oF	Meat Temp. Adjacent to Passage, ^oF
13.45	3.9×10^5	530.1	575.7	615.7

Minimum DNBR	Local Boiling	Bulk Boiling	Percent Quality
3.7	J = 6	0	0

The results in Tables 4 and 5 indicate that the SM-1 Core III will operate safely during design conditions and scram power conditions. This is based on DNBR's being above the minimum criteria of 1.5.

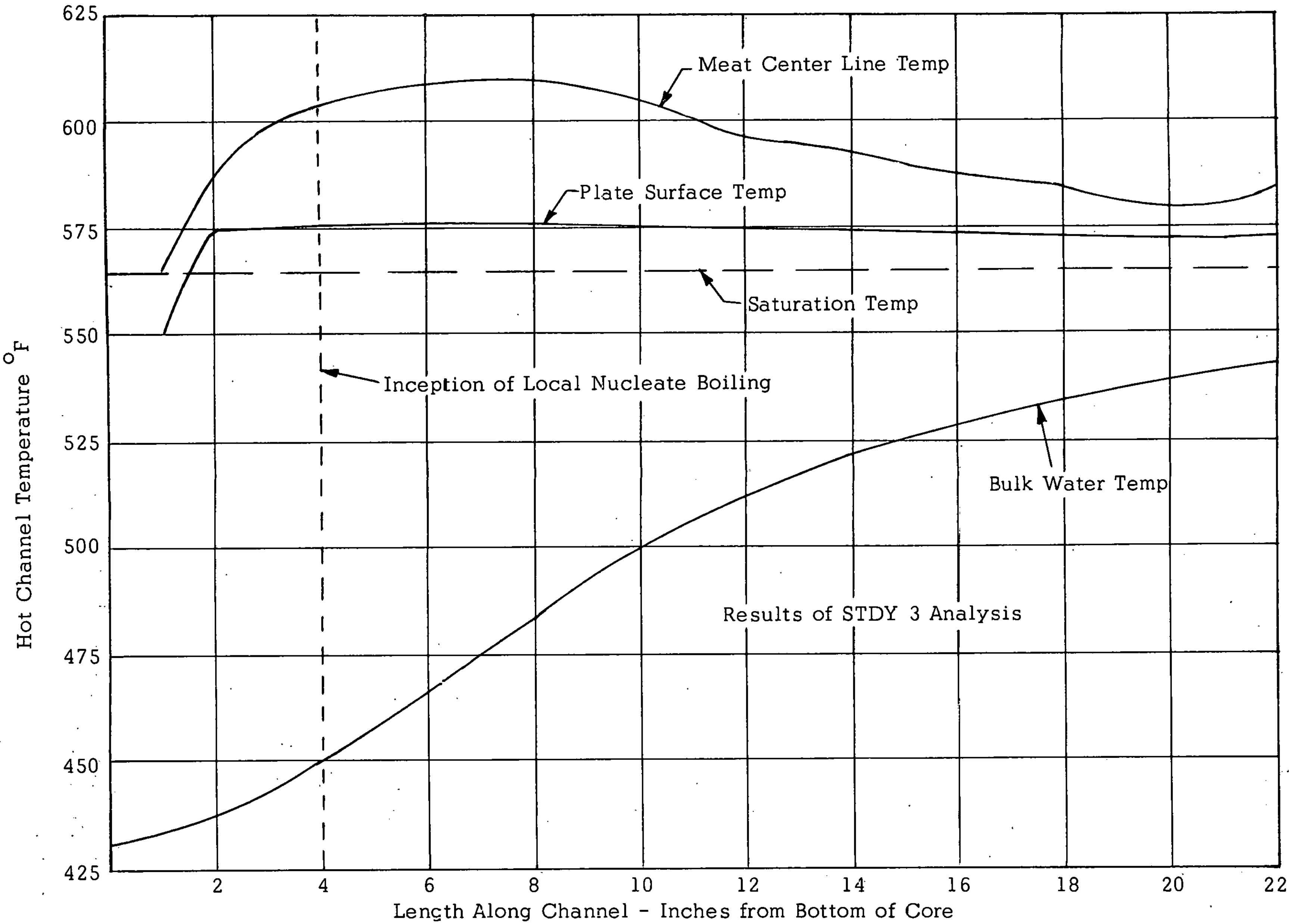

Figure 2. Hot Channel Temperature Vs Length Along Internal Channel for Element Position 37 at Reactor Power 10.77 MW

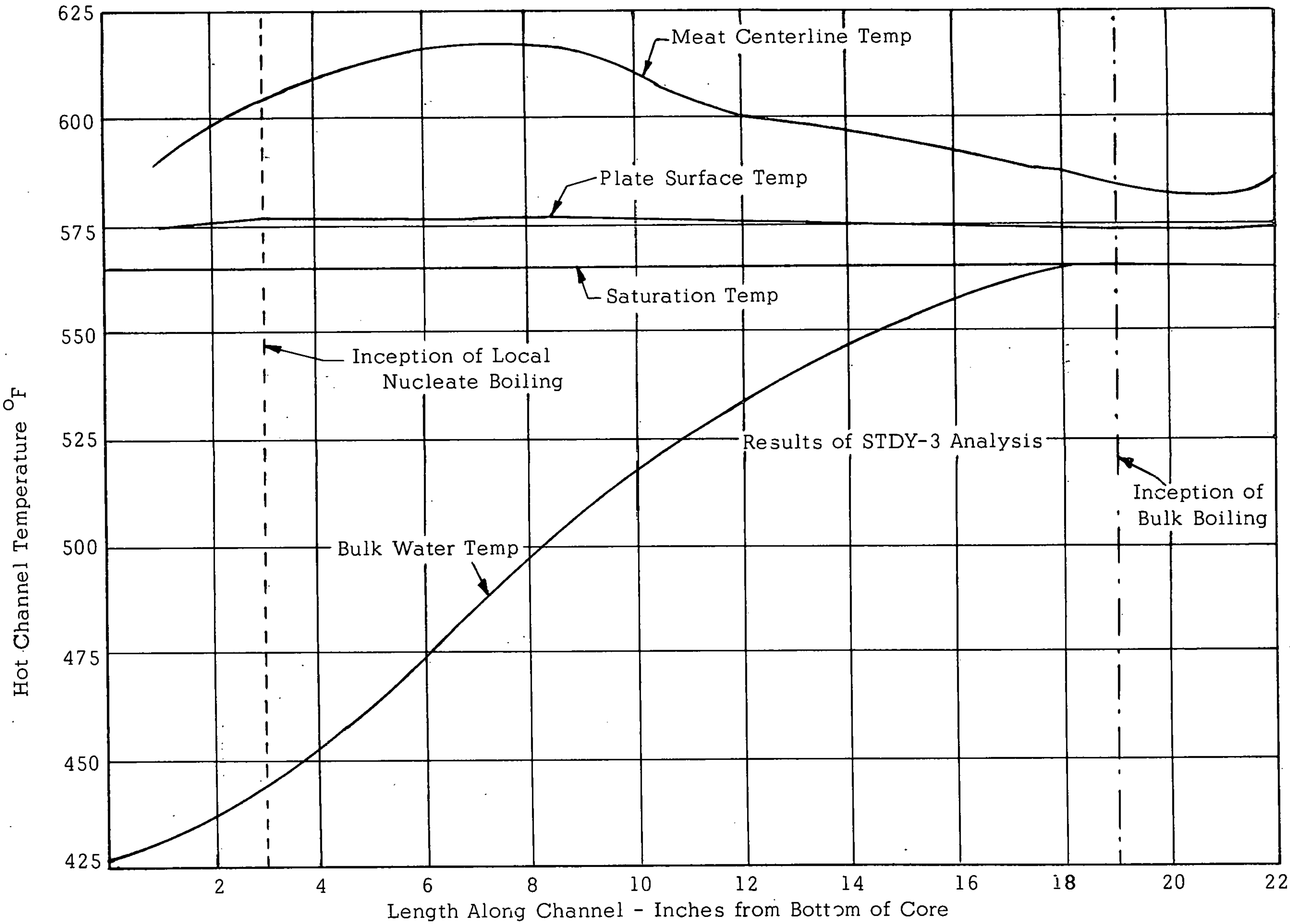

Figure 3. Hot Channel Temperature Vs Length Along Internal Channel for Element Position 37 at Reactor Power 13.45 MW

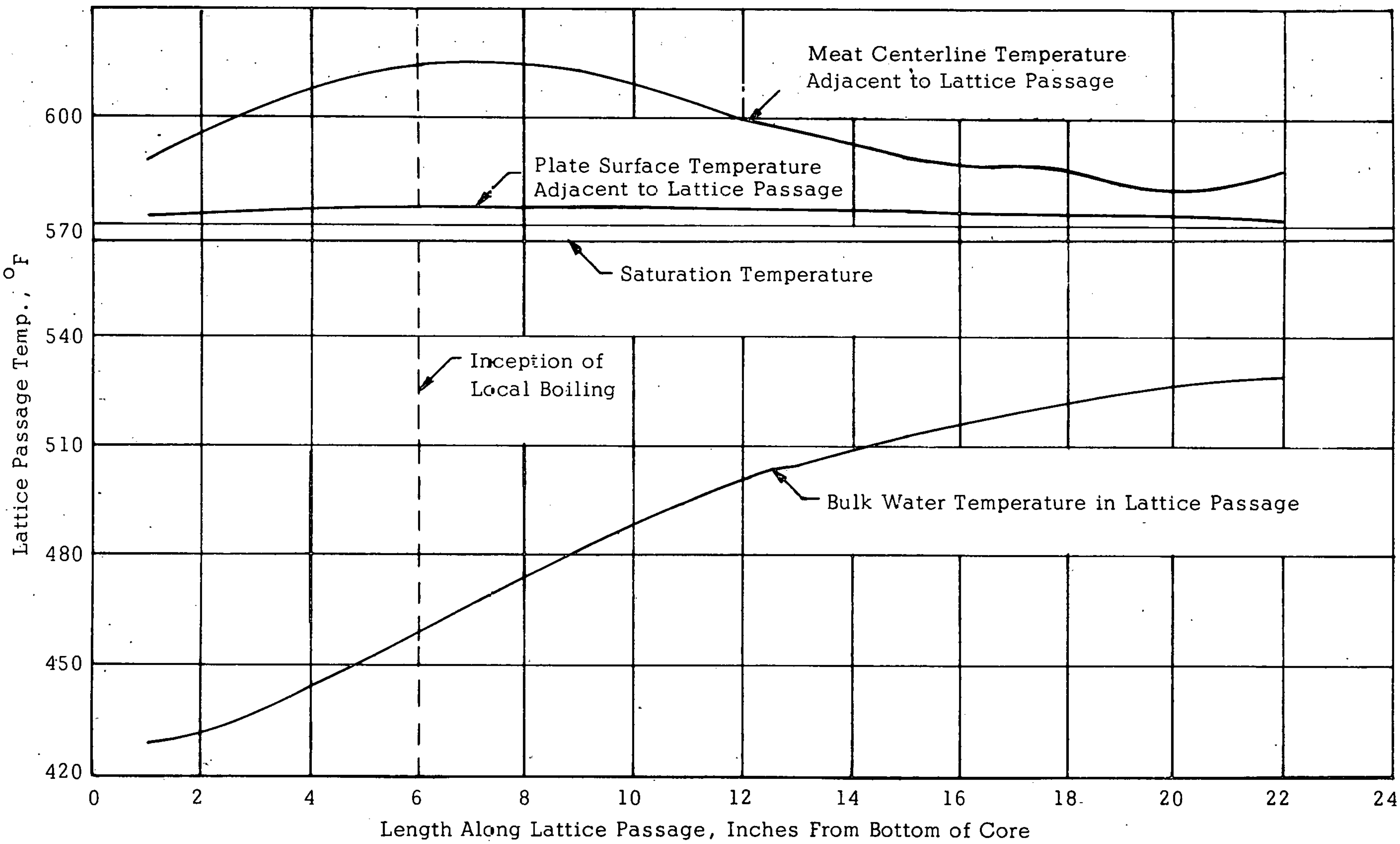

Figure 4. Lattice Passage Temperatures Vs Length Along Lattice Passage in Inches from Bottom of the Core

6.2 Transient Analysis

The minimum allowable DNBR in a reactor loss of flow transient has been set at 1.5. The loss of flow transient analysis was performed on the most critical element and the lattice passage adjacent to this element in the SM-1 Core III. From the steady state results the most critical element is located at core position 37. To insure the conservatism of the transient analysis the power level used was 13.45 Mw. If this element and its adjacent lattice passage prove safe to operate at this level it can be assumed that the entire SM-1 Core III will be safe thermally during a loss of flow transient.

The analysis assumed a "no scram" condition on low flow with the most adverse pump coastdown. A plot of flow coastdown as a function of time is given on Fig. 5. This pump coastdown is considerably more severe than that measured at SM-1 during TP-600.[8]

Previous analyses [1] utilized the ART-02 Code for the loss of flow transient. The present analysis has been updated with the use of the ART-04 Code. This code utilizes an advanced slip flow model instead of a fog flow model. With the presence of steam quality the slip flow model gives more conservative results than the fog flow model. The minimum DNBR results utilizing the ART-02 and ART-04 Codes for a coastdown of 5 sec are shown on Fig. 6. The effect of variation in flow coastdown on thermal conditions is shown in Table 6.

The minimum DNBR results for a coastdown of 5 sec within the lattice passage are shown on Fig. 7. The thermal conditions within the lattice passage are given in Table 6.

The results of the analysis indicate that during the critical period immediately following a loss of flow accident, the minimum DNBR within the most critical element is 3.7 and within the lattice passage 3.4. The reactor will scram because of low flow in approximately 0.050 sec.

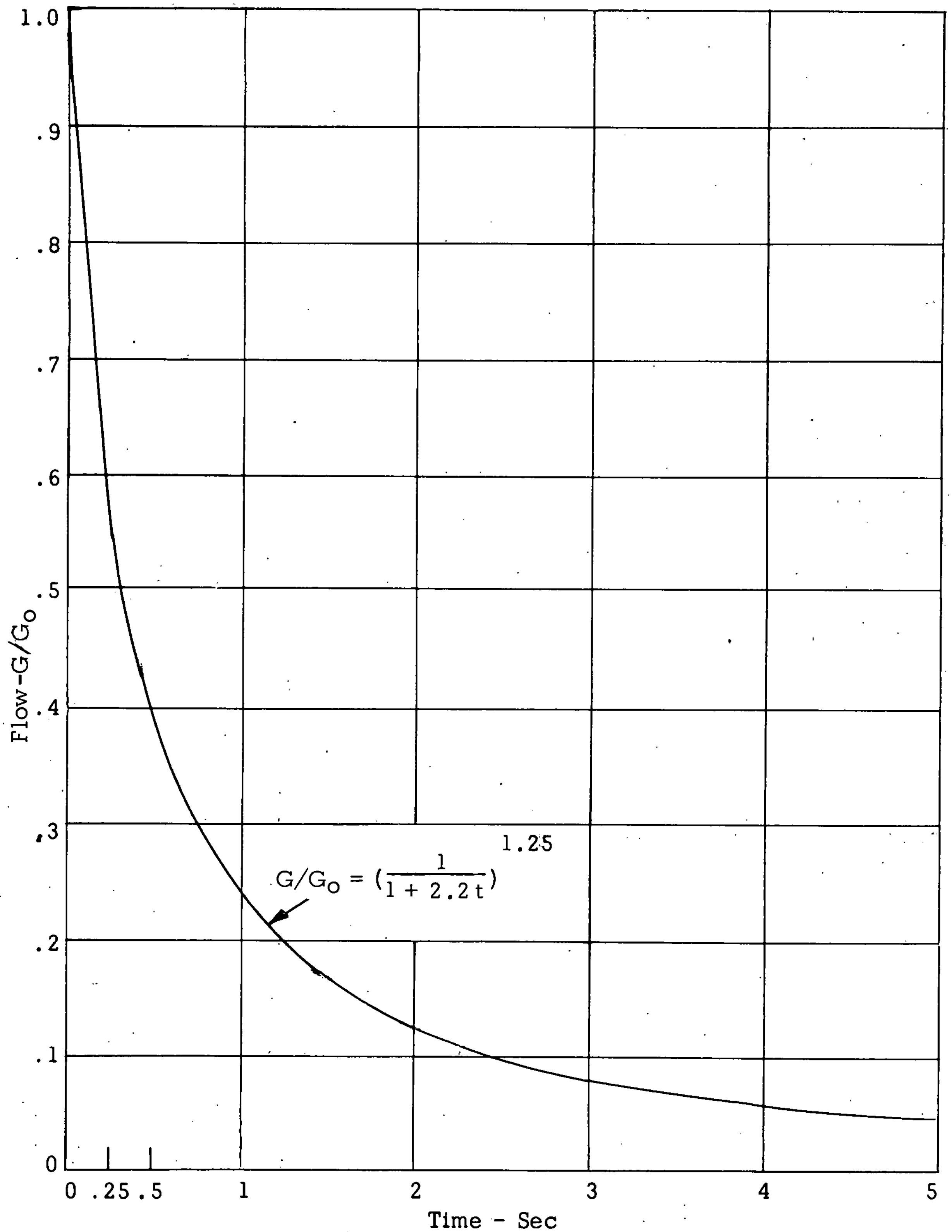

Figure 5. Flow Coastdown Investigation in Loss of Flow Analysis

TABLE 6

RESULTS OF LOSS OF FLOW ANALYSIS

Time Sec.,	Exit Enthalpy-Btu	Hot Channel Max. Surface Temp. °F	Min DNBR	Bulk Boiling J=Axial Increment	Hot Spot % Quality
ART-02					
0.0	574.27	577.5	3.74	J = 21	1.0
0.025	574.32	577.5	3.74	J = 21	1.0
0.5	585.11	577.0	4.20	J = 16	2.8
1.0	593.08	576.1	5.90	J - 17	4.1
2.0	560.63	575.0	9.02	J = 0	0.0
3.0	564.50	574.3	9.28	J = 0	0.0
4.0	577.11	573.8	11.80	J - 18	1.5
5.0	579.49	573.5	13.10	J = 19	1.8
ART-04					
0.0	575.00	577.5	3.66	J = 20	1.12
0.025	575.07	577.5	3.66	J = 20	1.13
0.5	586.05	577.0	4.19	J = 16	2.91
1.0	593.34	576.1	5.78	J = 17	4.09
2.0	561.27	575.0	8.84	J = 0	0.0
3.0	564.48	574.3	9.20	J = 0	0.0
4.0	577.10	573.8	10.93	J = 18	1.46
5.0	579.47	573.5	12.14	J = 19	1.84
ART-04 Lattice Pass.					
0.0	609.68	577.6	3.39	J = 14	6.74
0.025	609.81	577.6	3.39	J = 14	6.76
0.5	624.90	577.1	3.92	J = 13	9.21
1.0	611.34	576.2	5.39	J = 15	7.01
2.0	576.95	575.1	8.28	J = 20	1.43
3.0	579.14	574.4	8.81	J = 18	1.79
4.0	579.34	573.9	10.58	J = 20	1.82
5.0	580.20	573.6	11.96	J = 19	1.96

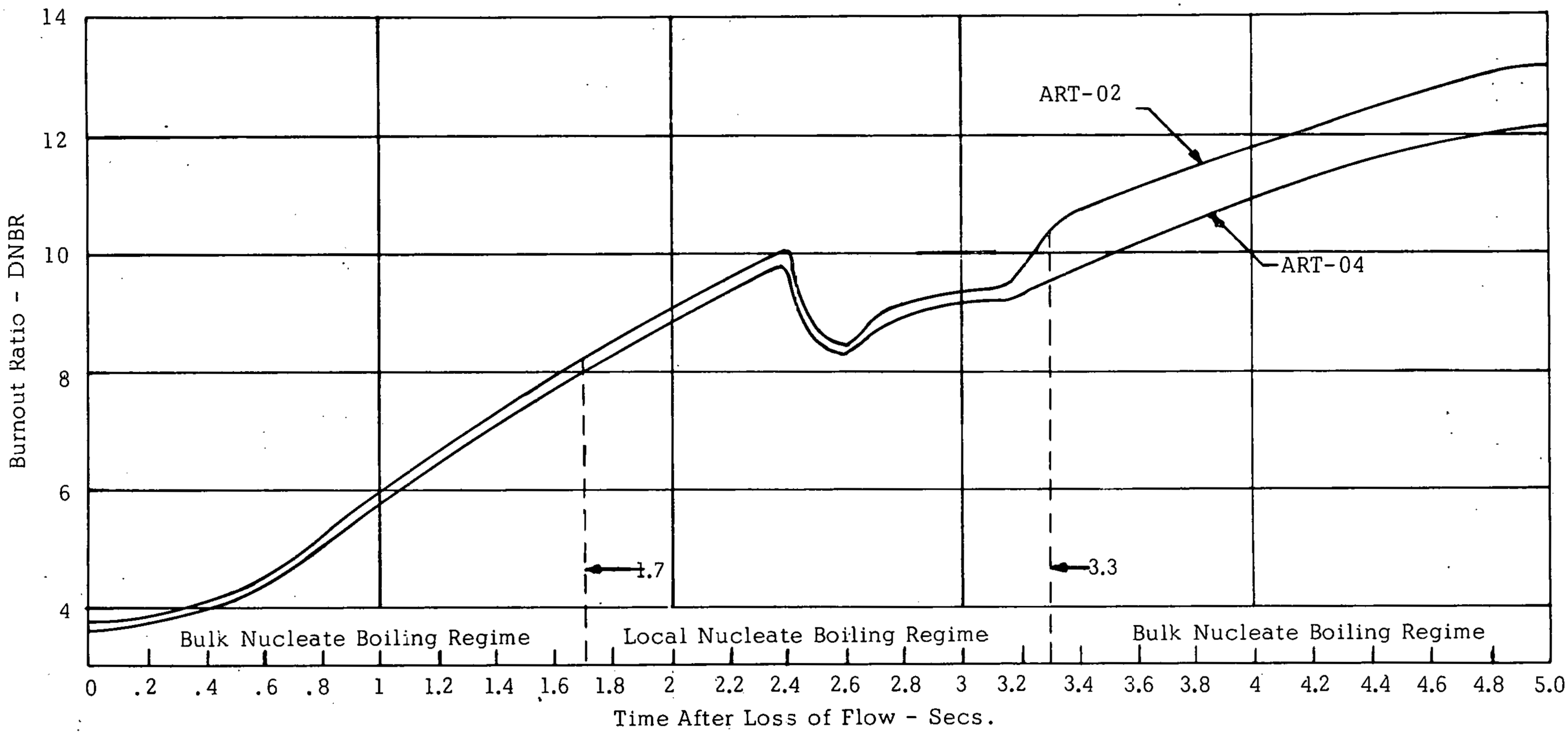

Figure 6. Minimum Burnout Ratio (DNBR) of Hot Channel Vs Time After Loss of Flow, Element Position 37. Comparison of ART-02 and ART-04 Results

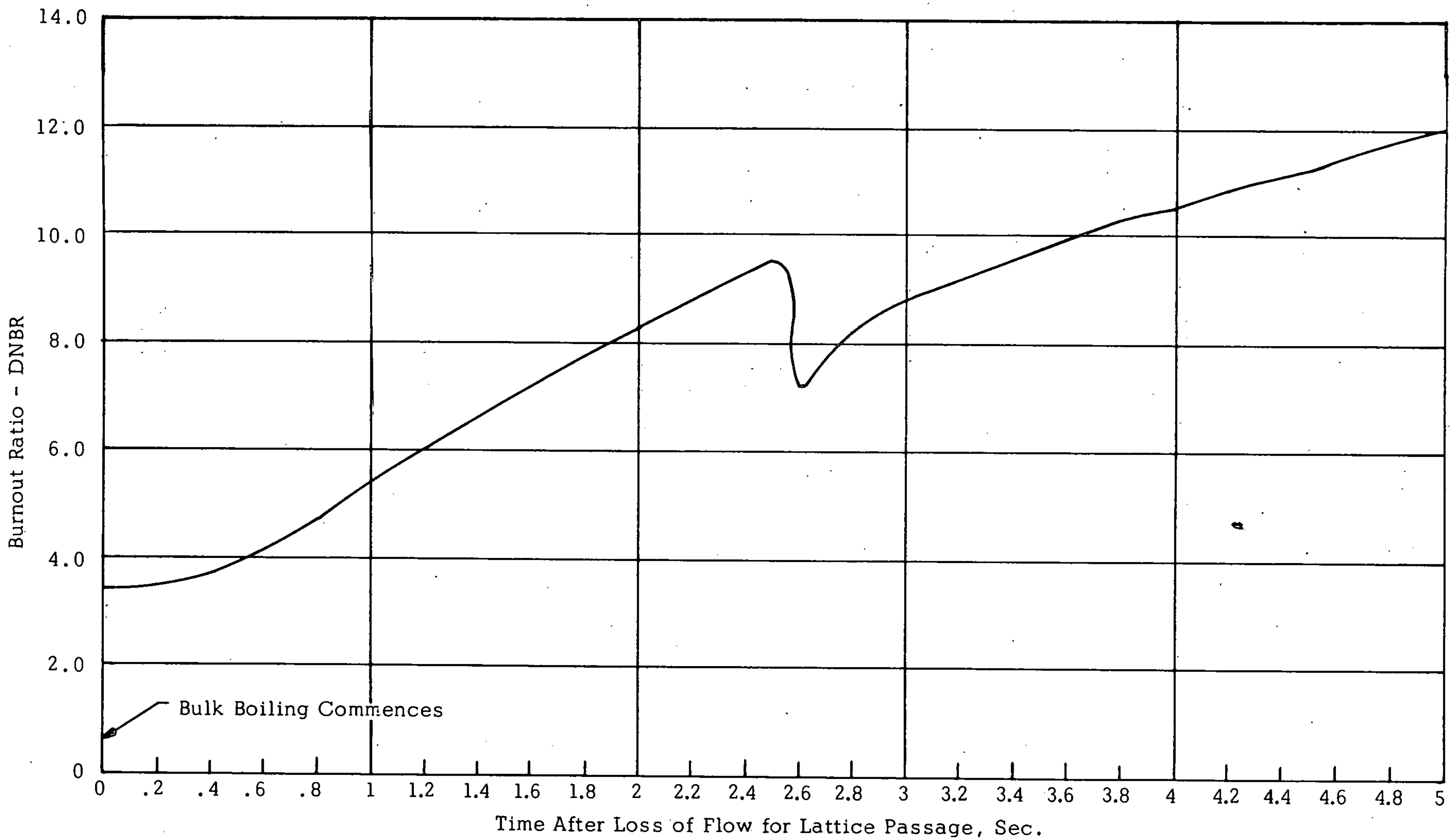

Figure 7. Minimum Burnout Ratio Vs Time After Loss of Flow for Lattice Passage

7.0 CONCLUSIONS AND RECOMMEDATIONS

1. The thermal analyis indicates the SM-1 Core III will operate safely for steady state and loss of flow transient conditions at design power 10.77 Mw and scram power 13.45 Mw.

2. The minimum DNBR for steady state conditions at 13.45 Mw is 3.53. Within the lattice passage the minimum DNBR is 3.7. The minimum DNBR for loss of flow transient condition is 3.7 internal to element position 37 and 3.4 in the lattice passage. Both of these conditions indicate DNBR's well above the minimum requirement of 1.5 for both the steady state and transient analysis.

3. Steady state analysis indicated local nucleate boiling in the hot internal channels of element positions 37, 51, 62, 75 and 15 at design power 10.77 Mw. Elements 37, 51, 31, 75 and 15 are located on the periphery of the core. A limited amount of bulk boiling was found in the exit end of the hot channels within elements 37, 51, 62, 75 and 15 at scram conditions 13.45 Mw. Bulk boiling was not found in any of the nominal channels analyzed at steady state.

4. The steady state analysis indicated at scram power, 13.45 Mw, local nucleate boiling occurred within the lattice passage formed by element position 37, and its adjacent stationary element. No bulk boiling was noted.

5. During the loss of flow transient analysis bulk boiling was evident at scram power, 13.45 Mw, in element position 37 during 0.25 to 1.7 sec after the start of pump coastdown. Bulk boiling did not appear in the hot channel until 1.6 sec later and continued for the remainder of the 5 sec transient. Minimum DNBR occurred at the start of the flow coastdown.

6. During the loss of flow transient within the lattice passage, bulk boiling was evident at the start of the flow coastdown and continued throughout the 5 sec interval for scram power conditions, 13.45 Mw. Minimum DNBR occurred at the start of the flow coastdown.

The following recommendations are made to supplement the analysis:

1. The improved two-dimensional transient codes such as XITE[9] or TITE[10] should be utilized. These codes account for two-dimensional transverse flow effects and will update the ART-04 transient results.

2. Metallurgical and radiochemical analyses should be made to determine effects of nucleate boiling in cladding material.

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