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TRANSIENT THERMAL ANALYSIS OF PWR'S BY A SINGLE PASS PROCEDURE USING A SIMPLIFIED NODAL LAYOUT

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Reports Issued Under this Contract

A. Topical Reports

P. Moreno, C. Chiu, R. Bowring, E. Khan, J. Liu, N. Todreas "Methods for Steady State Thermal/Hydraulic Analysis of PWR Cores," Report MIT-EL-76-006, Rev. 1, July 1977 (Orig. 3/77).

J. Liu, N. Todreas, "Transient Thermal Analysis of PWR's by a Single Pass Procedure Using a Simplified Nodal Layout," Report MIT-EL-77-008, Rev. 1, February 1979 (Orig. 6/77).

J. Liu and N. Todreas, "The Comparison of Available Data on PWR Assembly Thermal Behavior with Analytical Predictions," Report MIT-EL-77-009, Rev. 1, February 1979 (Orig. 6/77).

B. Papers

C. Chiu, P. Moreno, R. Bowring, N. Todreas, "Enthalpy Transfer Between PWR Fuel Assemblies in Analysis by the Lumped Subchannel Model," forthcoming in Nuclear Science & Engineering.

P. Moreno, J. Liu, E. Khan, N. Todreas, "Steady State Thermal Analysis of PWR's by a Single Pass Procedure Using a Simplified Nodal Layout," Nuclear Science & Engineering, Vol. 47, 1978, pp. 35-48.

P. Moreno, J. Liu, E. Khan and N. Todreas, "Steady-State Thermal Analysis of PWR's by a Simplified Method," American Nuclear Society Transactions, Vol. 26, 1977, p. 465.

ABSTRACT

PWR accident conditions and analysis methods have been reviewed. Limitations of the simplified method with respect to analysis of these accident conditions are drawn and two transients (loss of coolant flow, seized rotor) identified as candidates for analysis by this method. These transients have been examined in detail by this one-stage approach (the simplified method). It is concluded that the steady state one-stage simplified method can be applied to the above two transient conditions.

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Chapter 1

INTRODUCTION

The ideal way to analyze a PWR reactor core utilizing the lumped subchannel approach is by taking each radial node in the analysis at least as an actual subchannel. In this approach, the number of radial node is so large that there is no available computer which could handle this problem. Therefore, alternative approaches are needed.

Mass and momentum exchange occurs by both (a) turbulent mixing and (b) cross flow due to lateral pressure gradients. The success of any design code is dependent on how well these two phenomena are modeled. Since not much cross flow effect exists in PWR cores under steady state conditions, there is no need for complicated subchannel analysis methods. Because of this, an alternative procedure called the simplified method for steady state condition was initiated by Pablo Moreno.[1]

The simplified method is a one-stage approach which is different from other multi-stage approaches. For this method, the hot channel (most limiting location) needs to be identified before detailed analysis proceeds. For the detailed analysis, fine mesh (each radial node represents an actual subchannel) nodes are placed around the hot channel, and coarse mesh (each radial node represents a region which is formed by lumping subchannels) nodes are placed outside the fine mesh. In this way, the MDNBR of the PWR core can be evaluated. The computer code chosen was COBRA IIIC[2] in its M.I.T. version, COBRA IIIC/MIT.[3]

Preliminary investigations of the effects of both fine mesh and coarse mesh arrangements on the MDNBR were performed by Pablo Moreno. Based on his concept and later detailed analysis, the recommended procedure for the simplified method was finalized for steady state.[4]

The steady state simplified method can yield very satisfactory results. How good the simplified method will be under transient condition is the next key question which needs to be answered. Due to the large differences between different transients, the thermal analysis method for each can not be the same. Therefore, the identification of the characteristics of different types of transient is an essential part of the transient analysis work. Chapter 2 reviews many PWR accident situations and the different methods typically used by utilities in their assessment. From this review the applicability of the simplified method for each type of transient is evaluated.

Chapter 3 presents the results obtained by the simplified method under two transient conditions: Loss of Coolant

Flow and Seized Rotor. It was concluded that the former can be conservatively analyzed by the transient simplified method and the later can be analyzed by steady state simplified method assuming corewide uniform flow reduction. Assessment of the effect of realistic inlet plenum flow maldistribution requires accurate prescribed core flow inlet distribution.

The PWR data which have been used in this study are described in Appendix A.

Appendixes B is the tabulation and channel layout figures of the cases analyzed in this study.

Appendix C presents a qualitative description and proposed guidelines for the identification of the hot rod for cases with complicated radial and axial power distributions. Definitions of MDNBR and radial power peaking factor are also clearly stated.

CHAPTER 2

ANALYSIS METHODS FOR DIFFERENT TRANSIENT CONDITIONS

2.1 Introduction

The simplified method not only is based on MDNBR as the criterion but also cannot analyze either transients coupled with reactor system or transients with substantial feedback. Moreover, some transients can be analyzed simply although conservatively by a steady state approach. Therefore, only a limited number of transient conditions can or need be analyzed by the transient simplified method. This is demonstrated in Table 2.1 which indicates that the transient simplified method can be applied to only the loss of coolant flow, excess load and group CEA withdrawal transients.

The basis of the conclusions of Table 2.1 follows from the discussion of analysis methods for different transient conditions utilized by the Yankee Atomic Power Company for the Maine Yankee plant which are presented in this chapter[5].

The overall description, the limiting criteria, the analysis method and assumptions and the important results are the main items described for each type of accident. The possibility of using the simplified method for each type of transient is also discussed.

2.2 Full Length CEA Drop Incident

2.2.1 Overall Description

The result of this incident is a sudden drop in the core power level and an asymmetric power distribution. The maximum power peaking can be as high as 129.6% of the initial value. Due to the action of the negative temperature coefficient, the plant will return to the initial power level after approximately 60 seconds. During the transient period, both pressure and temperature decrease.

2.2.2 Limiting Criteria

The limiting criteria for this incident are DNBR and fuel centerline melt. The worst case full length rod drop incident, with respect to both criteria was identified as one initiated from near full power. Also, the worst case CEA drop with respect to DNB was found to be the minimum worth CEA that results in the maximum increase in peaking (0.1% $\Delta\rho$). The worst case with respect to fuel centerline melt criteria is one initiated from approximately 80-90%

power with Bank 5 inserted.

2.2.3 Analysis Method

This incident was analyzed by a code modeling the reactor system (like the GEMINI-II model). DNBR calculations were performed using COBRA IIIC steady state model.

2.2.4 Assumptions

The DNBR analysis assumes that the core power returns to the initial value following the transient. This incident analysis was based on the following assumptions:

(a) No credit was taken for the turbine runback feature,

(b) CEA drops in 0.05 seconds,

(c) Reactor is in manual control,

(d) No credit for pressurizer control (heaters or charging),

(e) Constant feedwater enthalpy.

2.2.5 Analysis results

The worst case full length CEA drop incident that could occur during Cycle 3 results in a DNBR greater than 1.57.

2.2.6 Applicability of the Simplified Method

Since the DNBR analysis was performed by a steady state model, the steady state simplified method is sufficient.

2.3 Loss of Coolant Flow Incident

2.3.1 Overall Description

The limiting loss of coolant flow incident is the simultaneous loss of all three reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature in core region. (T_{in} = constant). This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

2.3.2 Limiting Criteria

The three pump loss of coolant flow incident was analyzed using the CHIC-KIN computer code. The three pump flow coastdown used was taken from plant data. DNBR calculations were carried out using the COBRA IIIC computer code.

2.3.3 Assumptions

The moderator temperature coefficient was conservatively assumed to be zero, since the assumption of no temperature feedback effect is more conservative than use of a negative feedback effect. Reactor trip is considered in the analysis.

2.3.4 Analysis

Minimum DNBR of about 1.85 occurs about 3.3 seconds.

2.3.5 Applicability of the Simplified Method

Since DNBR is the limiting criterion, the simplified method can be applied to this transient condition.

2.4 Seized Rotor Accident

2.4.1 Overall Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced.

2.4.2 Limiting Criteria

This accident is similar to the loss of coolant flow accident. DNBR is the criterion for this accident analysis.

2.4.3 Analysis Method

DNBR calculations for this accident were performed with the COBRA IIIC computer code.

2.4.4 Assumptions

A conservative seized rotor analysis was performed assuming that the three pump flow rate instantaneously decreased to the two pump flow rate (2/3 of full flow) and that the reactor power remains at the initial full power value. Uniform inlet flow was assumed.

2.4.5 Analysis Results

The results of this analysis indicates that the minimum DNBR is 1.42, and therefore, no fuel damage is predicted.

2.4.6 Applicability of the Simplified Method

The simplified method can be used under seized rotor accident condition.

2.5 Excess Load Incident

2.5.1 Overall Description

This is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The consequences of the incident in the reactor core are a decrease in the core temperature which means reactivity insertion in the core (or increase in the power of about 8%-9%) and a small decrease in the pressure (2000 psia) of the coolant.

2.5.2 Limiting Criteria

It is a cold water transient. The hot zero power condition is the limiting case. DNBR is the criterion for the analysis.

2.5.3 Analysis Method

DNBR calculations for this incident can be performed with the COBRA IIIC computer code.

2.5.4 Assumptions

The worst excess load incident results from the full opening of the Dump and Bypass Valves at hot standby conditions. The analysis was based on full flow as the initial condition.

2.5.5 Analysis Results

DNBR's are always greater than or equal to 2.

2.5.6 Applicability of the Simplified Method

As long as the forcing functions are available, the simplified method can be used under this incident condition.

2.6 Loss of Load Incident

2.6.1 Overall Description

The major result of this incident is the temperature and pressure increase in the coolant loop. The power rate will decrease because of the negative moderator (~0. at BOC) and Doppler reactivity insertion due to the increase in temperature.

2.6.2 Limiting Criteria

Over pressure is the limiting criterion for the loss of Load incident. DNBR is also a criterion for the incident but not primarily.

2.6.3 Analysis Method

Thermal performance under this incident can be evaluated by using the COBRA IIIC computer code but not typically.

2.6.4 Assumptions

A conservative analysis can be performed by assuming a constant power rates, no events caused by turbine trip (no reactor trip, no turbine trip), only high pressure trip, and only secondary loop steam release (safety valves).

2.6.5 Analysis Results

Analysis of the loss of load incident showed the worst case to result in a high pressurizer pressure trip with a minimum DNB ratio of 1.9.

2.6.6 Applicability of the Simplified Method

Because the analysis criterion for the simplified method is DNBR, and the key concern for this incident is over pressure, it is not necessary to use the simplified method.

2.7 CEA Withdrawal Incident

2.7.1 Overall Description

This is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of a group of control element assemblies due to malfunction of the reactor control or control drive systems. The result is a rapid increase in the reactivity of the core and consequently the neutron flux and heat generation. This increase causes a reactor trip. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief of safety valve set point, there is a net increase in the reactor coolant temperature and consequently pressure.

Negative temperature coefficient tends to hold down the power.

2.7.2 Limiting Criteria

DNBR is the criterion for the analysis of the CEA withdrawal incident.

2.7.3 Analysis Method

The GEMINI-II along with the COBRA IIIC computer code can be used in the analysis.

2.7.4 Analysis Results

The CEA withdrawal incident which produced the lowest DNB ratio was the full power, end-of-life case.

2.7.5 Applicability of the Simplified Method

The simplified method can be used for this power transient type incident. Since the power transient is not a local phenomenon for the grouped CEA withdrawal incident.

2.8 Boron Dilution Incident

2.8.1 Overall Description

Boron dilution in the coolant reactor system changes the reactivity of the core due to its large neutron absorption rate. The incident consists of boric acid or demineralized water flow rates deviating from preset values as a result of system malfunction. Substantially this incident is one of uncontrolled reactivity rate and therefore the analysis is quite similar to the CEA uncontrolled withdrawal incident.

2.8.2 Limiting Criteria

This incident involves the length of time to approach criticality and the length of time necessary to lose shutdown margin. This length of time is the critical concern of this incident.

2.8.3 Analysis Method

The analysis was performed by hand calculation.

2.8.4 Assumptions

Inadvertent boron dilution is postulated to occur under a variety of plant conditions.

2.8.5 Analysis Results

The results from the hand calculation show that the length of time to approach criticality is within the design limit.

2.8.6 Applicability of the Simplified Method

Since the limiting criterion is not DNBR, the simplified method is not applicable for the analysis of this transient condition.

2.9 Steam Line Rupture Accident

2.9.1 Overall Description

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. With negative moderator and fuel reactivity coefficients, the cooldown will produce a positive reactivity addition. If the most reactive rod control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive rod control assembly to be stuck in its fully withdrawn position.

2.9.2 Limiting Criteria

The prevention of the core from returning to criticality and causing fuel damage is the prime concern in this accident.

2.9.3 Analysis Method

The system analysis computer code has to be used in the analysis.

2.9.4 Assumptions

The assumptions made in the analysis are consistent with maximizing the primary system cooldown and subsequent reactivity additions. The fastest cooldown which results in the most rapid reactivity addition, occurs when the break is at a steam generator nozzle. This break location is assumed for the cases analyzed.

2.9.5 Analysis Results

The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

2.9.6 Applicability of the Simplified Method

The prime consern is not the minimum DNBR but the possibility of a return to power. The simplified analysis method is not sufficient under the steam line rupture accident, since the system code is needed to perform the analysis.

2.10 Loss of Feedwater Flow Incident

2.10.1 Overall Description

A loss of normal feedwater results in a reduction in capability of the secondary system to remove the heat generated in the reactor core (same consequences as a loss of heat sink). The immediate result is the increase in coolant temperature and pressure.

2.10.2 Limiting Criteria

The availability of ample time for providing emergency feedwater is the key issue in the analysis of this accident.

2.10.3 Analysis Method

The analysis can be accomplished by running the system computer code with a model of the steam generator.

2.10.4 Assumptions

Feedwater flow is instantaneously reduced to zero.

2.10.5 Analysis Results

Twenty minutes of ample time for emergency feedwater is sufficient to ensure the existence of a heat sink.

2.10.6 Applicability of the Simplified Method

The simplified method is not applicable for determing the time required for providing the emergency feedwater since the system code is needed.

2.11 CEA Ejection Accident

2.11.1 Overall Description

The consequence of this accident is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

2.11.2 Limiting Criteria

Fuel center line melting temperature (incipient melting, not total melting) is the criterion for the analysis.

2.11.3 Analysis Method

Core power traces and channel enthalpy could be obtained by runing CHIC-KIN computer code.

2.11.4 Assumptions

Feedback effect is included for gross core analysis, but not included for the hot channel analysis.

2.11.5 Analysis Results

Partial power core or zero power core sometimes is even worse than full power core case.

2.11.6 Applicability of the simplified Method

Since fuel center line melting temperature is not the criterion which the simplified method is designed to calculate, the simplified method is not applicable to this method.

2.12 Other Incident Conditions.

2.12.1 Idle Loop Start Up Incident

Since idle loop start-up is not permitted in the Maine Yankee core, no analysis regarding to this incident has been performed.

2.12.2 Part Length CEA Drop Incident

Part length CEA drop incident was not analyzed because the use of part length CEA's during power operation has been prohibited by the technical specification.

2.13 Conclusion

From the descriptions in this chapter, it is clear that for some long term transients, the thermal analysis can be performed by quasi-steady state method. This means that the steady state simplified method is sufficient. Neutronic/Thermal-hydraulic coupling code and system code which is not the tool utilized by the simplified method are needed for some transient analysis. For certain transient that the analysis criterion is not MDNBR, the simplified method is not applicable. Therefore, only limited numbers of transient conditions can or need be analyzed by the transient simplified method as shown in Table 3.1.

TRANSIENT	DNBR ANALYSIS REQUIRED?	STEADY STATE ANALYSIS SUFFICIENT?	TRANSIENT SIMPLIFIED METHOD APPLICABLE?
Full Length CEA Drop	YES	YES	NO
Loss of Coolant Flow	YES	NO	YES
Seized Rotor	YES	YES	NO
Excess Load	YES	NO	YES
Loss of Load	NO	NO	NO
CEA Withdrawal (Group)	YES	NO	YES
Boron Dilution	NO	YES	NO
Steam Line Rupture	NO	NO	NO
Loss of Feedwater Flow	No	NO	NO
CEA Ejection	NO	NO	NO

Table 2.1 Applicability of the Transient Simplified Method

Chapter 3

ACCIDENT TRANSIENTS ANALYZED BY THE SIMPLIFIED METHOD

3.1 Introduction

From the descriptions in the previous chapter, it is clear that the loss of coolant flow transient and seized rotor transient not only are short term transients but also are analyzed by the criterion of MDNBR. (Since the DNBR reaches its minimum very fast during the transient period.) Additionally, the neutronic feedback effect is not significant, since the void fraction or density changes are small. Also, no data regarding to excess load and group CEA withdrawl have been obtained for this work. Therefore, the types of transient which have been analyzed by the simplified method are the loss of coolant flow and seized rotor transient accidents.

The Maine Yankee PWR core was selected as an example core and its physical characteristics are presented in Appendix A. The computer code chosen was COBRA IIIC/MIT. Steady state correlations were assumed to be valid under transient conditions. Cases analyzed are tabulated and described in Appendix B.

3.2 Loss of Coolant Flow Accident

First, it is helpful to restate the ingredients of the one-stage simplified method. These are

a) the identification from power factors of the hot channel,

b) the necessity to represent the whole core in the analysis and

c) the insensitivity of MDNBR to the number of coarse mesh nodes.

The conclusion that the steady state simplified method is valid under loss of coolant flow transient can only be drawn if the above three ingredients hold under the transient condition. Therefore, the strategy of the approach is to confirm that the approach used for each of these three ingredients in the steady state approach is valid for this transient.

Secondly, the transient conditions and assumptions related to the analysis in this section will be described and specified.

Finally, results of different cases analyzed are presented and discussed.

3.2.1 Conditions and Assumptions[5]

The accident analyzed was a complete loss of coolant flow (3 pumps out of 3 flow coastdown). It is the most severe loss of flow accident condition. The coolant flow rate decreases to 60% of the initial value after 9 seconds. The flow forcing function which was taken from plant data is shown in Figure 3.1. System pressure is nearly constant during the transient period, but a slightly decreasing pressure forcing function had been assumed to have a conservative estimation. The inlet coolant temperature is also unchanged for this accident. Power input is assumed to be constant for the case of no reactor trip. For the case which has reactor trip, the power will drop to 14% of the initial value after 3 seconds as again is shown in Figure 3 .1.

Three different axial power distributions are shown in Figure 3 .2. Based on the definition of the MDNBR and the results which were obtained from single channel thermal hydraulic analyses (Case's 35, 36, 37.), the axial power distribution with $I_p = -0.235$ was shown to be the conservative axial shape for enthalpy or temperature analysis. By the same reason, the axial power distribution with $I_p = +0.201$ can be assumed to be the conservative axial shape for MDNBR analysis. Since the criterion of the simplified method is the MDNBR, $I_p = +0.201$ has been choosen as the axial power distribution in this analysis. Fixed power shape is justified, since density changes are small. A Thorough investigation of the effect of the axial power distribution is discussed in Appendix C.

3.2.2 The Effect of the Time Step in COBRA IIIC/MIT

The computation procedure of COBRA IIIC/MIT is that the steady-state computations are performed first to obtain initial conditions for the transient. Since the finite difference equations are stable for large time steps, those same equations are used for the steady state calculation by setting "Δt" equal to some arbitrarily large value. An iteration is performed until convergence of the flow solution is obtained. Transient calculations are performed in the same way but for a selected time step Δt. Boundary conditions and other forcing functions are set to their desired values at each new time; then, the calculation sweeps through the channel for the number of iterations required to achieve convergence on the crossflow. The converged solution is used for the new initial condition and the procedure continues for all time steps.

Three different time steps have been choosen (1, 0.5, 0.25 seconds) in the time step sensitivity study (Case 38). Results of this study are shown in Figure 3.3. As can be seen that results are grossly insensitive to the Δt value before the minimum DNB ratio occurs at 3 seconds. Therefore, one second has been choosen as the time step for the analyses performed in this study.

3.2.3 Cases Analyzed

The purpose of the cases analyzed in this section is to check the results of minimum DNBR with the simplified method channel layout.

In order to achieve this goal, two major questions were investigated:

1) Is the layout of nodes specified for the steady state simplified method sufficient for transient analysis?

To assess this, two series of cases were investigated. The first series (Series one) was a scoping study to confirm that differences in DNBR magnitude between different nodal arrangements did not change with time during the transient. This conclusion then permits confident extrapolation of the results of the steady state nodalization studies to loss of flow transients. The second series of cases (Series two) utilized the test core to verify this conclusion.

2) Does the location of MDNBR move during the loss of flow transient?

A series of cases (Series three) utilizing the test core was run to investigate this question.

3.2.3.1 Effect of the Transient Condition on MDNBR (Series One)

The accident without reactor trip was selected to examine the most severe thermal behavior together with the loss of flow transient condition. As mentioned in

Reference 4, the results of the minimum DNBR obtained from COBRA IIIC/MIT are different due to different nodal layouts. As noted under 1) above the question now to be considered is the changing magnitude of the differences as a function of time under the loss of coolant flow transient condition.

Two major input parameters will vary due to different channel models – the channel area and the channel power. To explain this, three different cases of channel layout along with the numerical values of the channel radial peaking factors are illustrated as an example in Figure 3.4. It is supposed that Case A is the actual subchannel layout, Case B and Case C are two different channel layouts to model the real subchannel case (i.e. Case A). The channel area and the channel power are different between Case's B and C. So the direct effect of arbitrary different channel layouts can be examined by inputting different channel areas and powers to the code. A series of cases (Series one) were performed by varying the channel areas and channel powers to examine the effect of different channel layouts under the transient condition. Results of this series of cases were plotted on Figure's 3.5, 3.6 and 3.7.

Figure's 3.5 and 3.6 are results of MDNBR from a two-channel analysis (Case 39). Figure 3.5 is the result of a case which has a constant ratio of the two channel areas and a constant power input to the hot channel but with a varying value of the power input to the cold channel. Under the steady state condition (t = 0), the MDNBR decreases as the cold channel power increases. Results of MDNBR at each second during the whole transient period were plotted on the same figure. The MDNBR decreases as the transient time increases.

Figure 3.6 is the result of a case which has a constant power ratio of the two channels but with a varying magnitude of the cold channel area. Similarly, under the steady state condition (t = 0.), the MDNBR increases as the flow area of the cold channel increases. These increases are much smaller than the decreases as shown in Figure 3.5, since the effect of power input is more significant than the channel area which is represented by the nodal point. Results of MDNBR at every second of the transient period (9 seconds) were also plotted on the same figure. The MDNBR decreases as the time of transient increases. The important message from these two figures is that the trend of MDNBR under steady state condition due to different channel area or power is preserved under the loss of coolant flow transient condition.

This transient behavior also can be confirmed by another three case study as shown in Figure 3.7. Case 40, Case 41 and Case 42 have totally different channel layout. The trend of the differences of the MDNBR between these three different cases under steady state was indicated in the figure by the curves labeled t=0. This is similar to the trend of the differences of the MDNBR between these three different cases under transient condition which was indicated in the figure as at the end of the transient period (9 seconds). Therefore, the trend of MDNBR is preserved and the loss of coolant flow transient condition does not have any major effect on the result obtained from different channel layout modeling methods. This conclusion is important, because it can explain the validity of the simplified method under this transient condition.

3.2.3.2 Nodal Layout Required for Adequate Transient Analysis (Series Two)

The nodal layout of the simplified method was recommended in Reference 4 for steady state analysis. A series of cases (Series two) has been analyzed by the same recommended procedure in this section to check the validity of the method under this transient condition.

Case 43 is the recommended channel layout case. For this case, layout of fine mesh and coarse mesh are shown in Figure's 3.8 and 3.9 respectively. Case 44 is the more simplified case in which the corner subchannels of the fine mesh layer which surrounds the hot channel has been eliminated. The channel layout of fine mesh and coarse mesh of this case are shown in Figure's 3.10 and 3.9 respectively. Case 45 is the base case which has more coarse mesh nodes than other cases as shown in Figure's 3.11 and 3.12. Case 46 has the same fine mesh layout as Case 45 but has a smaller number of coarse mesh nodes as shown in Figure's 3.13 and 3.14.

Figure 3.15 illustrates the differences of MDNBR between these cases under the loss of coolant flow with power trip transient condition. Due to the reactor trip, the MDNBR will change from decreasing to increasing at 3 second after the initiation of the transient. From this figure, it clearly demonstrates that the small differences of MDNBR between different cases at the initial condition are preserved during the transient period. Additionally, the simplified method always gives conservative results for the whole period of transient.

Also this accident without reactor trip transient condition has been investigated to check a more severe situation. Under this condition, results of MDNBR as a function of time obtained from the four different channel layout cases mentioned above are shown in Figure 3.16 (Case's 47, 48, 49 and 50). It is illustrated clearly that the differences between cases with different channel layout do not change considerably even under this severe transient condition.

3.2.3.3 Location of MDNBR under Transient Conditions (Series Three)

The location of MDNBR is defined as the place which the minimum DNB ratio occurs. The rod with the minimum DNBR is called the hot rod. Since MDNBR is most sensitive to linear power generation rate, the hot rod can be identified by examining the rod radial power peaking factors. The above argument was investigated and confirmed under steady state condition.[4] The effect of the transient on the ability to identify the hot rod is investigated in this section.

Three cases (Series three) have been performed to check the location of MDNBR. (Case's 51, 52, 53)

Case 51 is a two channel thermal analysis case which covers the two most limiting subchannels in the Maine Yankee core. The numbering scheme of rods and channels for the code input along with a table of the radial power peaking factors of rods and channels are shown in figure 3.17. As indicated in the figure, two half rods across channel 1 and 2 were divided in order to check the location of MDNBR. Results of MDNBR from the analysis are shown in Table 3.1. It shows that the rod numbered 2 (Also the rod which has the highest peaking factor of 1.362 with cold wall effect.) is the hot rod from the begining of the transient till the end. This means that there is only one hot rod during the whole period of transient.

Case's 52 and 53 are whole core analysis for checking the location of MDNBR. As mentioned before, because of a symmetrical core, a one eighth core segment was selected Figure's 3.18, 3.19 and 3.14 illustrate the channel layout and numbering scheme for the two cases. The rod radial power peaking factors for the two cases are listed in Table's 3.2 and 3.3. As indicated in Figure 3.14, the layout of the coarse mesh is the same for the two cases. Since the purpose of the cases analyzed here is to investigate the possibility of the migration of the hot rod, the only difference between Case 52 and Case 53 is the layout of fine mesh nodes. The layout of fine mesh in Case 52 is such that the rod with the highest radial power peaking factor which is located in a channel which has cold wall effect can be evaluated by the simplified method.

The layout of the fine mesh in Case 53 is such that the highest power rod which is located in a channel which does not have cold wall can be evaluated. It had been confirmed by prior two channel analysis that the hot rod is the rod which possesses the highest radial power peaking factor and is located in the channel which has cold wall.

The results obtained from Case 52 are listed in Table 3.4. Rod numbered 9 is the rod with the highest peaking factor located in the channel which has the presence of a cold wall. Rod numbered 10 is the highest power rod located in a channel without the presence of a cold wall. From the table it is clear that rod numbered 9 is the hot rod for the whole period of transient.

The results obtained from Case 53 are listed in Table 3.5. The hot rod is the rod numbered 8 which is the highest power rod with cold wall effect. Even though the layout of fine mesh nodes has been changed from Case 52 to Case 53, the hot rod is still the same rod.

From the results, it can be concluded that the location of hot rod will not migrate under the loss of coolant flow transient. The method of identifying the hot rod or hot channel developed under steady state condition is still valid.

3.2.4 Analysis of the Results

In order to have a more detailed explanation of the results mentioned in section 3.2.3, individual diversion cross flows for different cases have been examined. A typical case as shown in Figure 3.20 is the diversion cross flow as a function of time across from the hot channel to one of the four adjacent channels at the elevation of MDNBR. The difference between Case's 46 and 45 and the difference between Case's 43 and 44 both are small. There is a comparatively large difference between Case 46 and Case 43. The above results can be explained from the channel layout point of view. The reason is because Case 46 and Case 45 both have the same layout of fine mesh nodes, and Case 43 and Case 44 both have the same layout of coarse mesh nodes.

Figure 3.21 indicates that the differences of total cross flow from hot channel at the location of MDNBR (Summation of the individual cross flow at certain elevation) for the four cases are not as far away as the individual cross flow for the four cases as already shown in Figure 3.20. The differences are still noticeable. Since the total cross flow mentioned above was calculated at only one specific location, the differences of the total cross flow at a fixed location between different cases do not play an important role in determining the hot channel axial mass flow rate. Figure 3.22 shows that the differences of the hot channel axial mass flow rate at the location of MDNBR between cases with different model of channel layout are not noticeable. The hot channel axial mass flow rate is an important factor to determine the specific enthalpy which again is very important to predict the critical heat flux. The very small differences of the hot channel axial mass flow rate between cases with different channel layout is the reason why there are only small differences of MDNBR between cases with different channel layout. Figure 3.22 shows clearly that the small differences are preserved throughout the whole period of transient. Therefore, the differences of MDNBR between cases are preserved under this transient condition. The effect of this transient on the development of the simplified method is insignificant.

The changes of the individual cross flow pattern with time along the path of the hot channel were also examined. Figure's 3.23 and 3.24 show that the cross flow patterns are different due to different channel layout and different transient time. These two figures also show that the effect due to the transient on the thermal analysis results is much less than the effect due to the different channel layout methods. Effect of the different channel layout methods on the thermal analysis results have been thoroughly investigated.(4) The conclusion and the recommended method which were developed in Reference 4 are indeed valid under the loss of coolant flow transient condition.

3.3 Seized Rotor Accident

This accident again can be analyzed by performing the thermal hydraulic computer code COBRA IIIC/MIT transient runs. The transient conditions and assumptions have been described in Chapter 2. Three cases (54, 55 and 56) with different channel layout which are similar to the channel layouts of Case's 43, 44 and 45 as mentioned in section 3.2.3 have been performed again in this section. In order to have a more severe condition, no reactor trip has been considered in the analysis. Since the nature of the transient condition and the thermal analysis is similar to the loss of coolant flow cases, only results of MDNBR are presented in this section as shown in Figure 3.25.

Because flow goes instantaneously to 2/3 of the initial value and the power remains constant, the MDNBR drops immediately to about 2.5. Subsequent changes of MDNBR with time are very small because of a constant power and flow. This leads to the conclusion that the seized rotor accident without reactor trip transient condition can be analyzed by a steady state or a quasi-steady state method. The differences of MDNBR due to different channel layout are also small and preserved during the whole period of transient. Therefore, this type of transient can be analyzed by the simplified method as recommended in Reference 4.

The above analysis was based on an uniform inlet flow distribution. Actually this is not the real condition for a seized rotor accident. Nonuniform inlet flow distribution should be utilized to confirm the ability of the simplified method under this transient condition. This requires a calculation using the base case nodal layout but for the whole core crossection since now a sector of the core inlet associated with the seized rotor loop has lower flow than the core average. This calculation is considerably more expensive than the 1/8 core calculations made in this chapter but can be done with COBRA IIIC and is necessary to confirm the common assumption of uniform inlet flow.

3.4 Transport Coefficent N_H under Transient Conditions

An attempt was made to correct the results obtained in the previous sections with the consideration of transport coefficient as defined in Reference 4. Case's 57 and 58 with transport coefficient $N_H = N$ (see Reference 4) under conditions of loss of coolant flow and seized rotor respectively were performed. As shown in Table's 3.6 and 3.7, results from these two cases were compared to those with $N_H = 1$ under the same condition and channel layout. The transport coefficient N_H is not important under these two transient conditions, since values of MDNBR do not change much for cases with N_H.

TIME sec.	MDNBR	ROD
0	3.163	2
1	3.020	2
2	2.863	2
3	2.726	2
4	2.591	2
5	2.485	2
6	2.356	2
7	2.266	2
8	2.161	2
9	2.096	2

Table 3.1 Identification of Hot Rod (Case 51)

Rod Number	1	2	3	4	5	6	7	8	9	10	11
Peaking Factor	.8547	1.	1.117	.9793	.979	.681	.679	.949	1.362	1.362	1.084

Rod Number	12	13	14	15	16	17	18	19	20	21
Peaking Factor	1.183	1.198	1.2	1.004	0.	1.273	1.281	1.353	1.281	1.274

Table 3.2 Power Inputs for Case 52

Rod Number	1	2	3	4	5	6	7	8	9	10	11
Peaking Factor	.8547	1.	1.015	1.	.681	.679	.672	1.362	1.362	.969	1.101

Rod Number	12	13	14	15	16	17	18	19	20	21
Peaking Factor	1.198	1.2	1.186	1.004	0.	1.273	1.281	1.353	1.281	1.274

Table 3.3 Power Inputs for Case 53

TIME sec.	MDNBR	
	ROD 9	ROD 10
0	3.223	3.717
1	3.021	3.431
2	2.859	3.209
3	2.718	3.016
4	2.582	2.830
5	2.460	2.663
6	2.321	2.466
7	2.204	2.300
8	2.091	2.120
9	1.991	1.994

Table 3.4 Comparison of MDNBR between Rod 9 and 10 (Case 52)

TIME sec.	MDNBR	
	ROD 8	ROD 9
0	3.229	3.739
1	3.027	3.452
2	2.864	3.229
3	2.724	3.036
4	2,588	2.850
5	2.466	2.633
6	2.326	2.484
7	2.209	2.318
8	2.096	2.142
9	1.916	2.011

Table 3 .5 Comparison of MDNBR between Rod 8 and 9
(Case 53)

TIME sec.	$N_H = 1$	$N_H = N$
0	3.215	3.219
1	3.016	3.018
2	2.858	2.855
3	2.717	2.715
4	2.583	2.579
5	2.462	2.456
6	2.322	2.317
7	2.206	2.201
8	2.092	2.084
9	1.999	1.993

Table 3.6 Comparison of MDNBR under loss of Coolant Flow Transient Condition (Case 47 versus Case 57)

TIME sec.	N_H = 1.	N_H = N
0	2.322	2.315
1	2.290	2.284
2	2.283	2.276
3	2.276	2.269
4	2.268	2.262
5	2.261	2.255
6	2.226	2.220
7	2.194	2.188
8	2.164	2.158
9	2.127	2.122

Table 3.7 Comparison of MDNBR under Seized Rotor Transient Condition. (Case 54 versus Case 58)

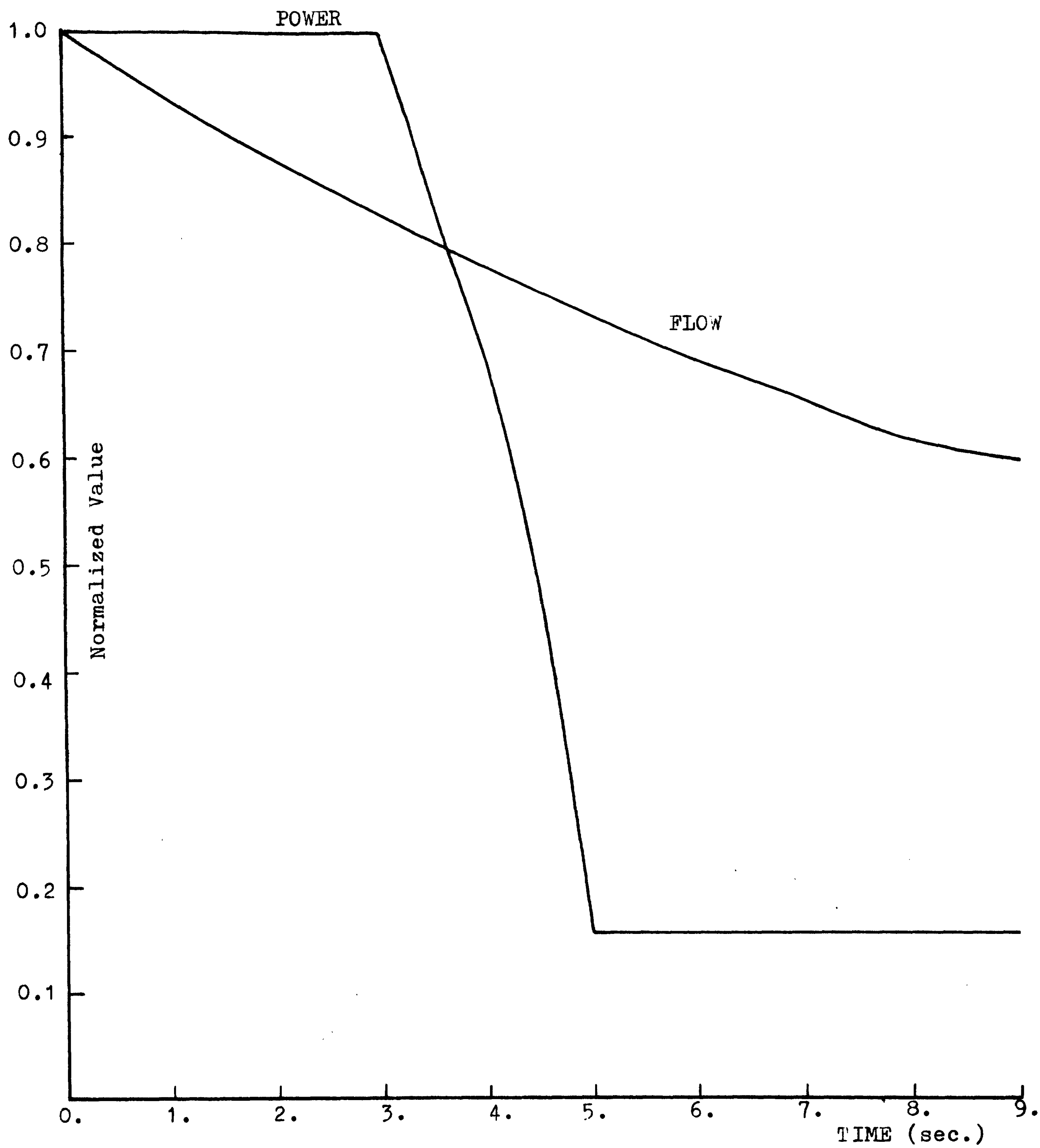


Figure 3.1 Power and Flow Transient Forcing Functions

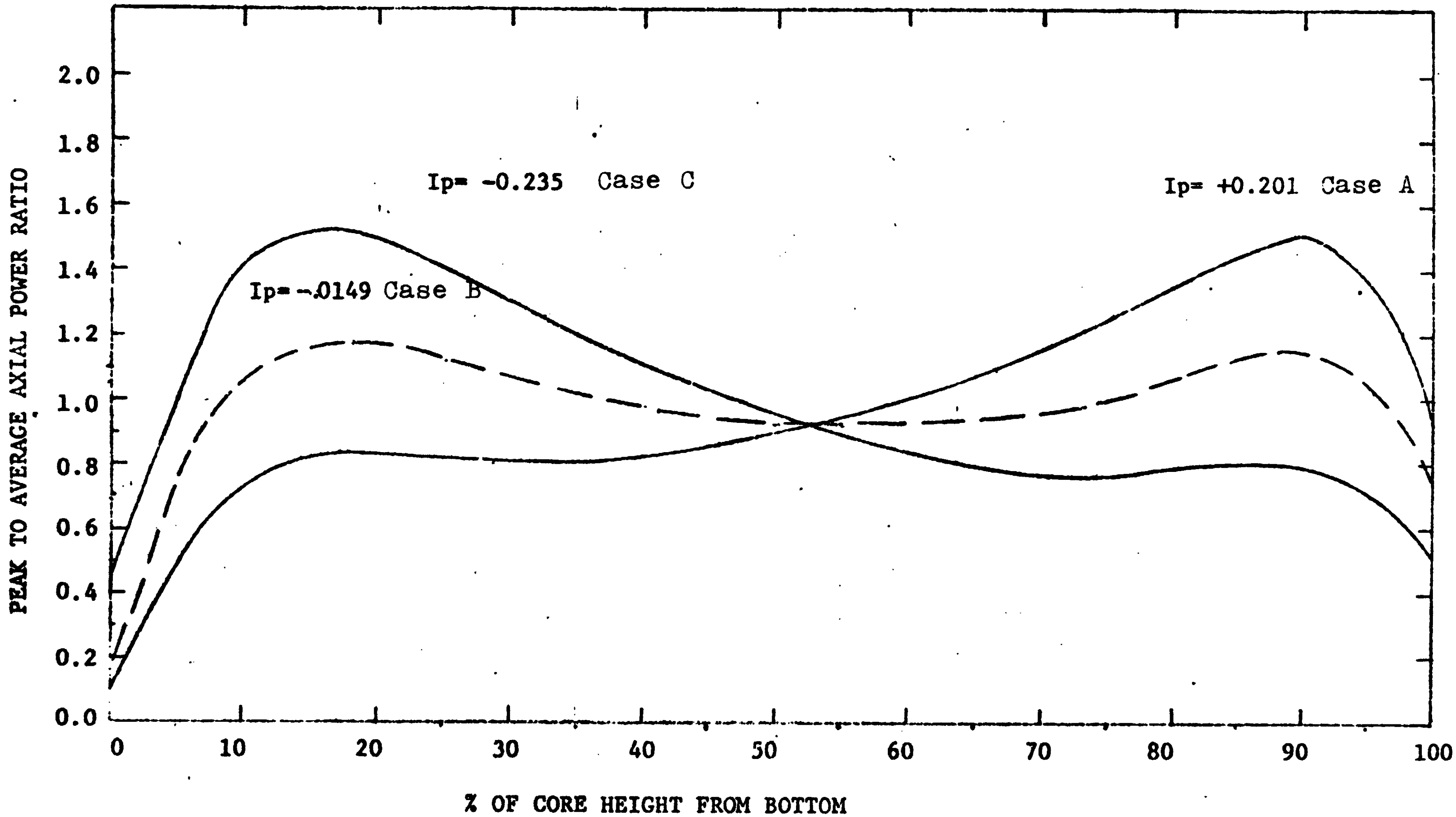


Figure 3.2 Core 2 Axial Power Distributions Used in Safety Analysis (Maine Yankee)

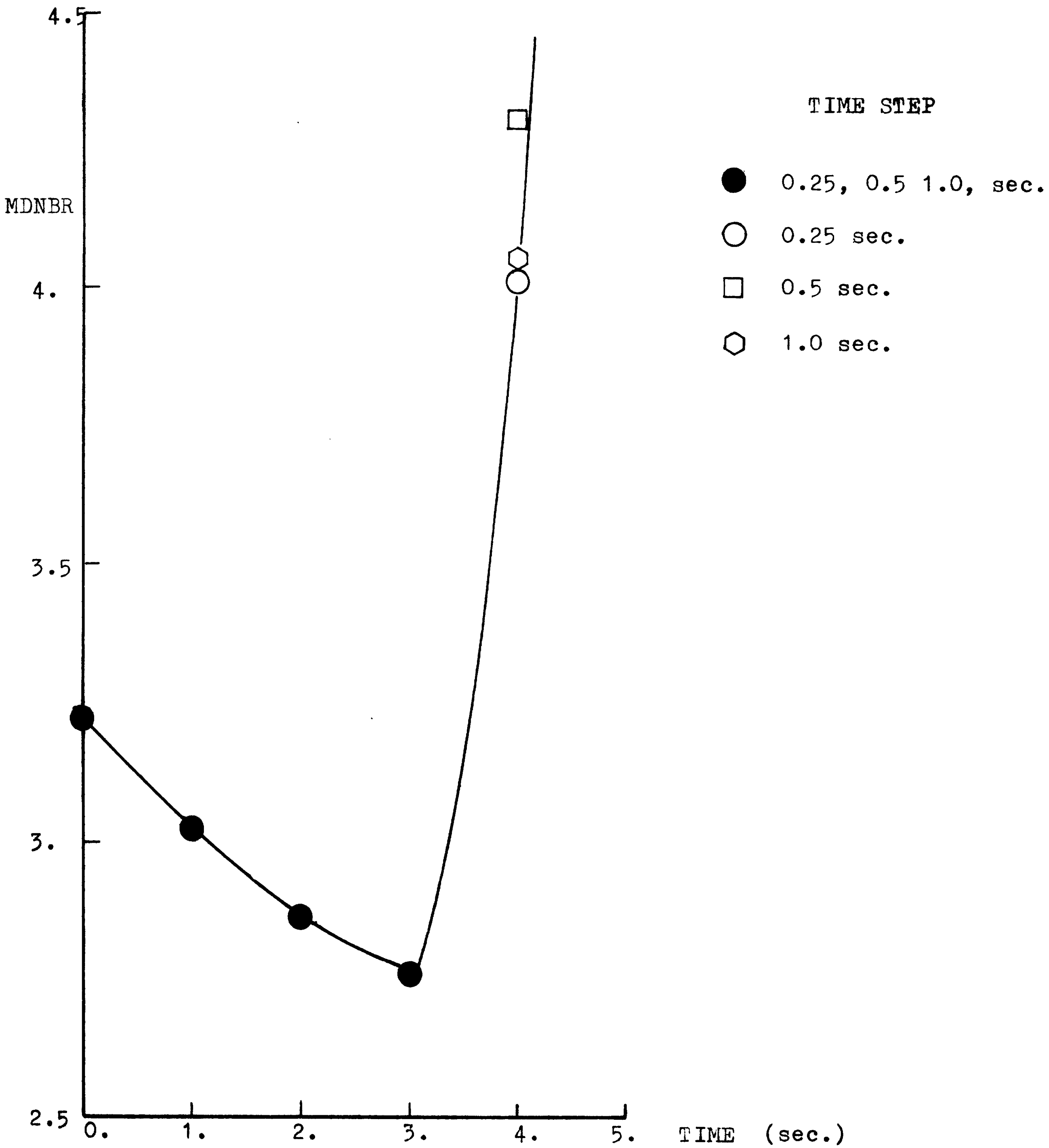
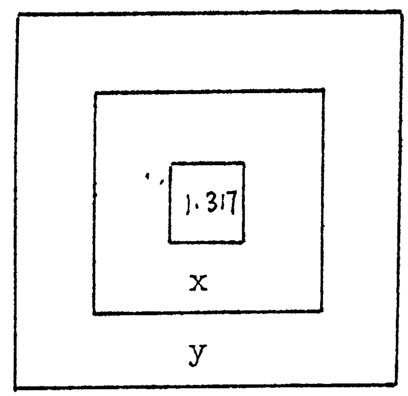


Figure 3 .3 Sensitivity of MDNBR to Time Steps (Case 43)

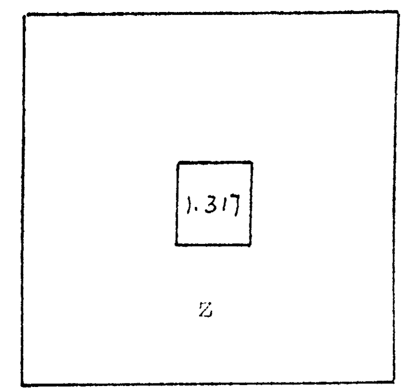
CASE A

1.307	.678	0.	.666	1.280
.979	.681	.679	.672	.946
.949	.979	1.317	.969	.921
1.183	1.198	1.20	1.186	1.158
.920	.945	1.279	.949	.933

CASE B



CASE C



Where: $x = (.681 + .679 + .672 + .979 + .969 + 1.198 + 1.20 + 1.186)/8 = .946$

$$y = \frac{(1.307 + .678 + 0. + .666 + 1.28 + .979 + .946 + .949 + .921 + 1.183 + 1.158 + .920 + .945 + 1.279 + .94}{16}$$

$$= .943$$

$$z = \frac{8x + 16y}{8 + 16} = .944$$

Figure **3.4** Example Cases

(Numbers Indicated are The Channel Radial Power Peaking Factors)

(x, y, z are the Average of Peaking Factors)

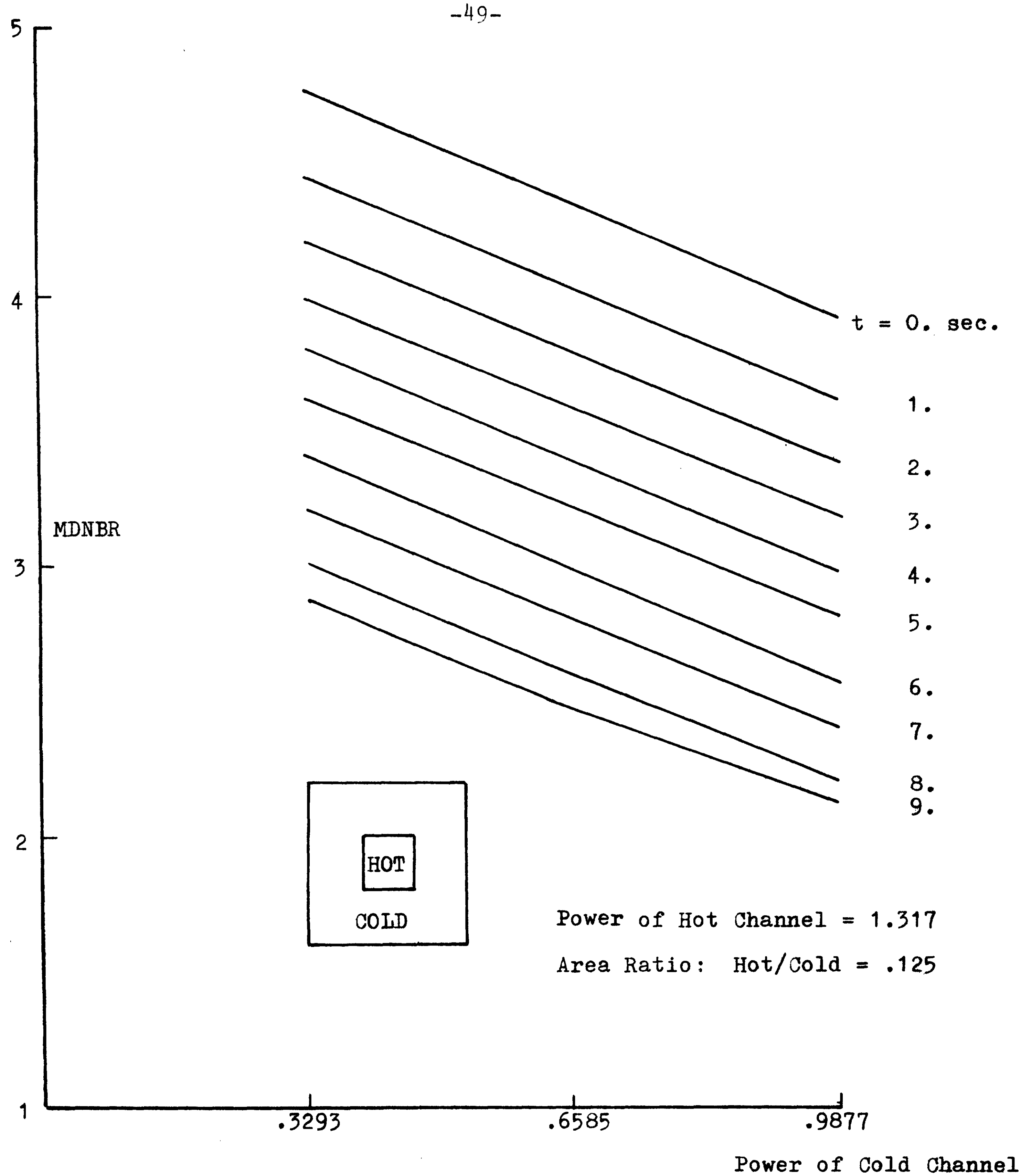


Figure 3.5 Effect of Cold Channel Power On MDNBR under Transient Condition

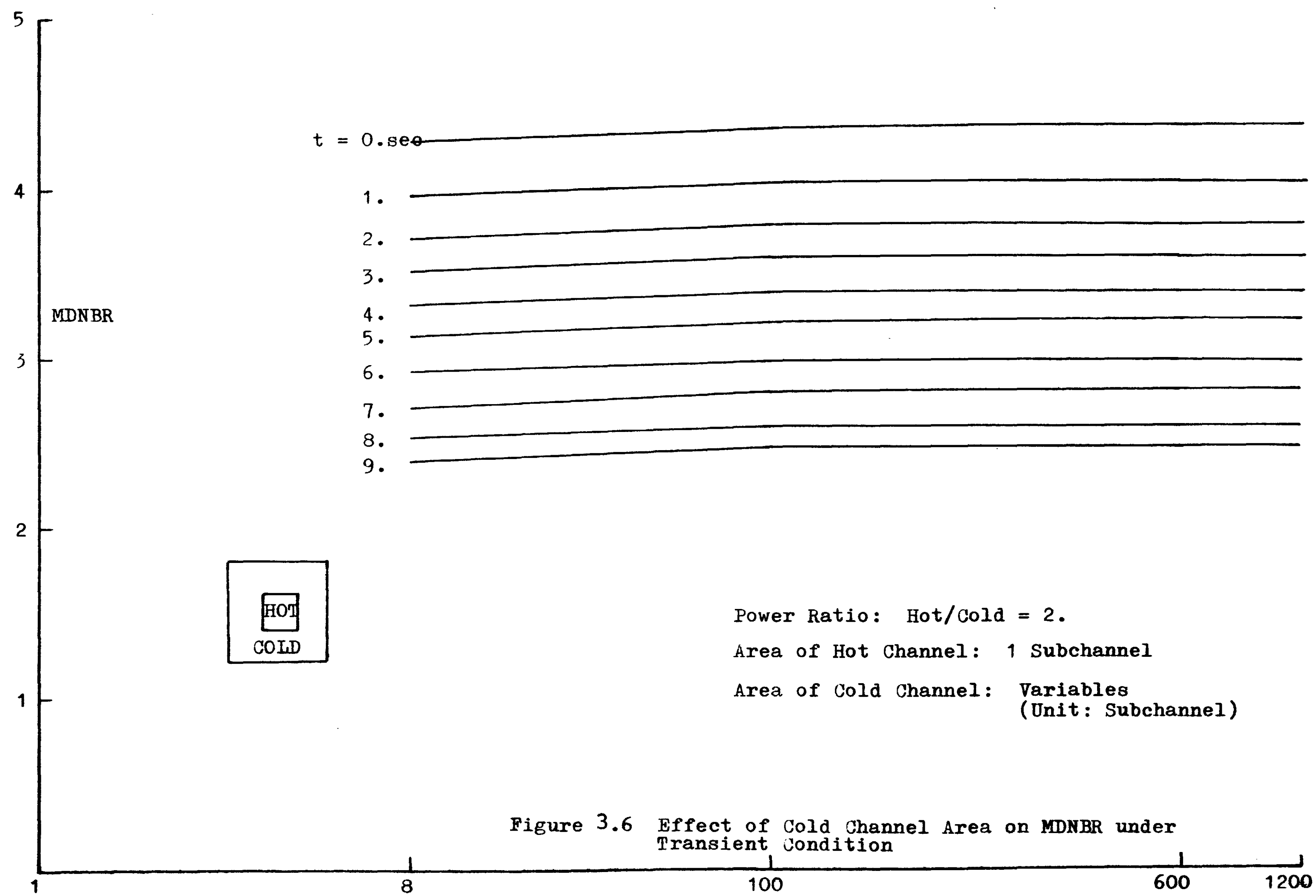


Figure 3.6 Effect of Cold Channel Area on MDNBR under Transient Condition

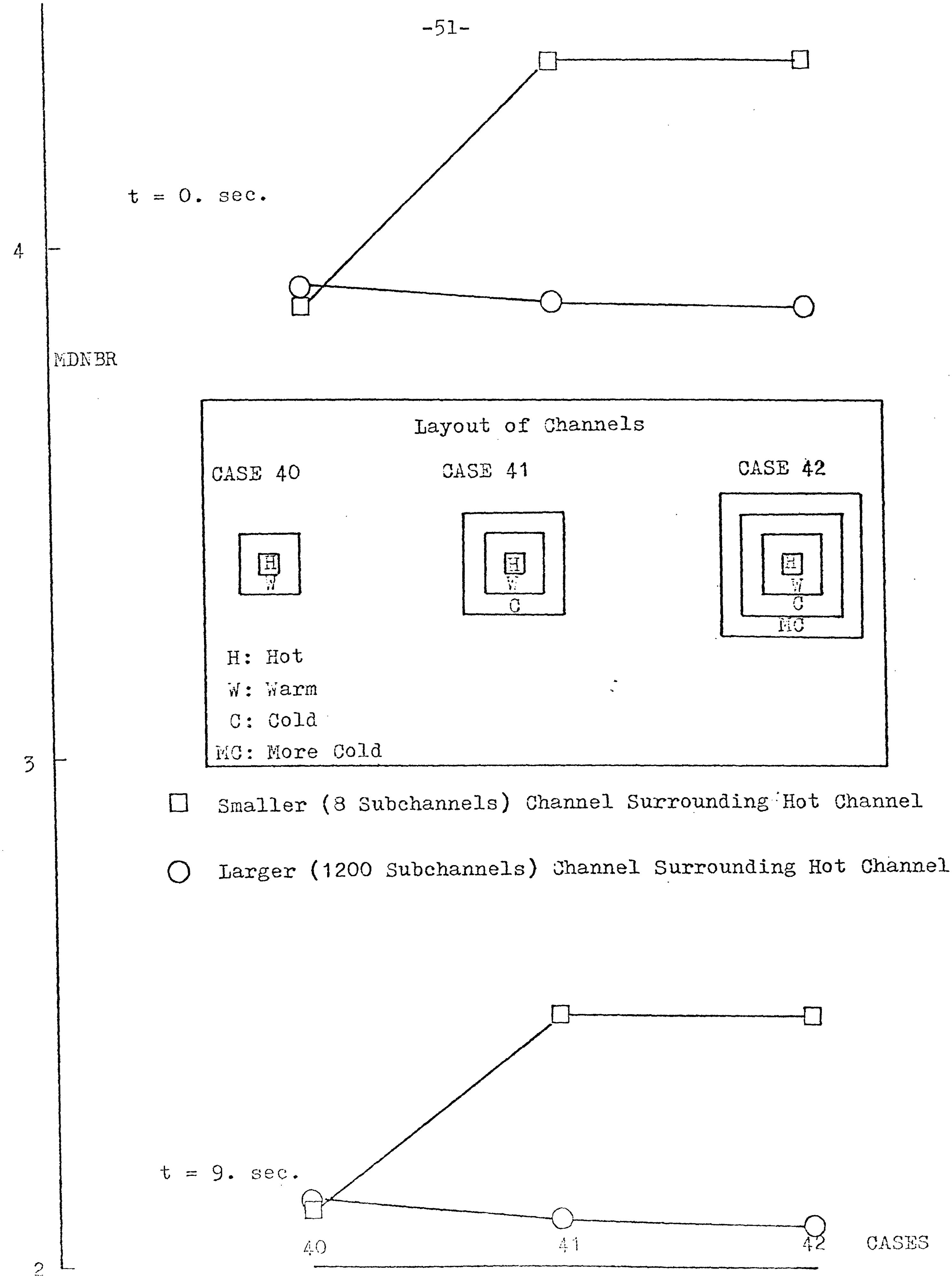


Figure 3.7 Effect of Channel Layout on MDNBR

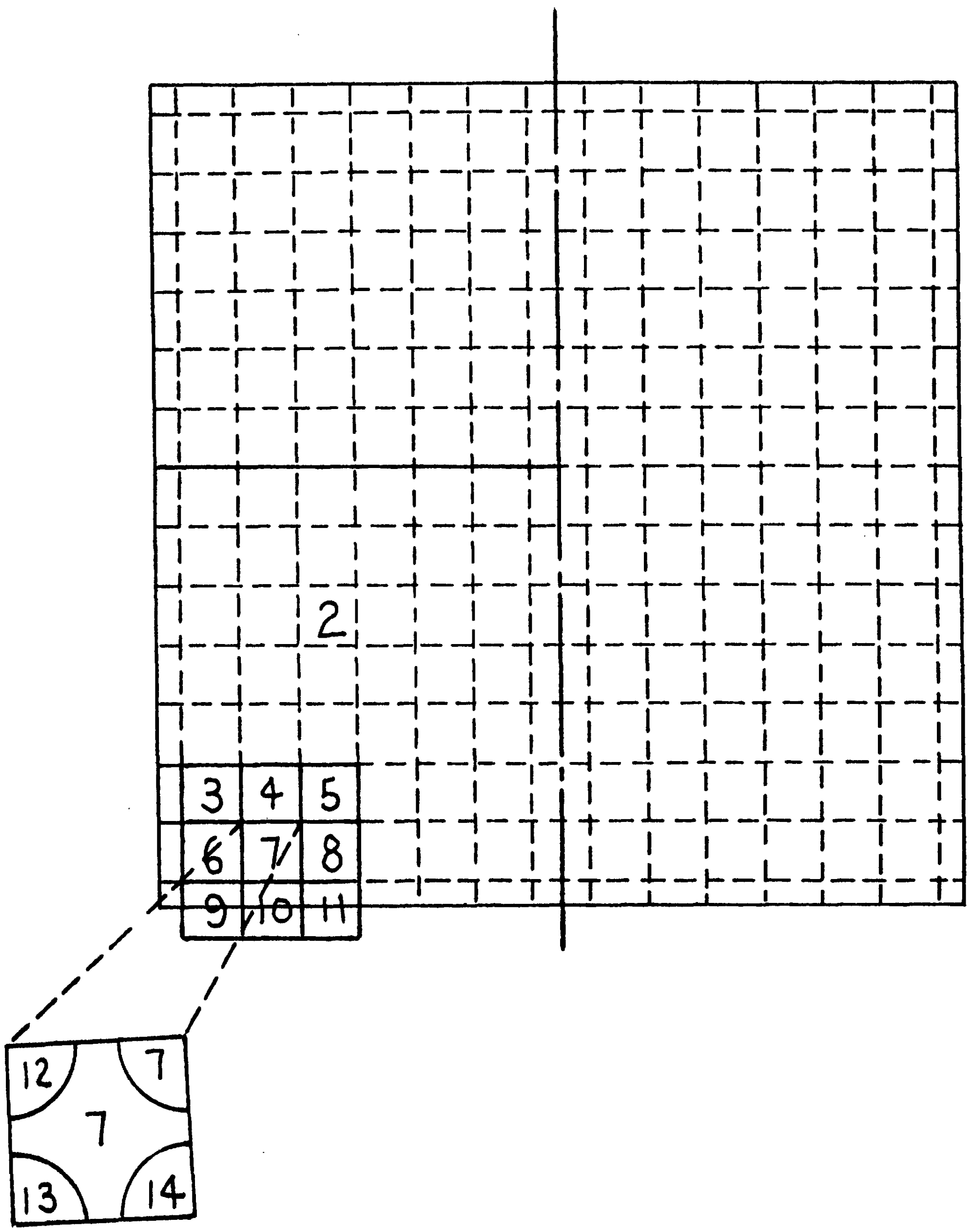


Figure 3.8 Channel Layout in Hot Assembly for Case 43

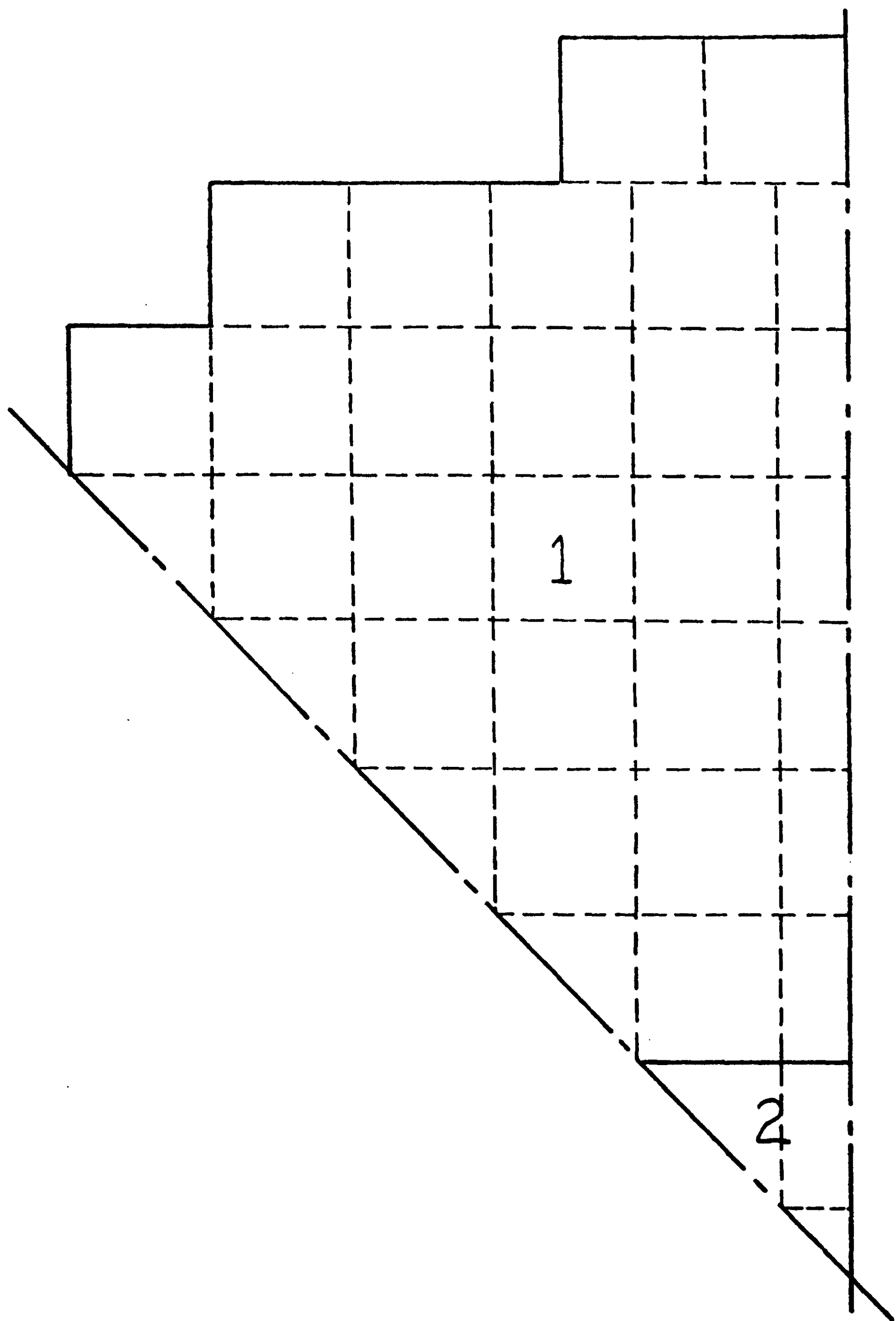


Figure 3 .9 Layout of Coarse Mesh for Case 43, Case 44

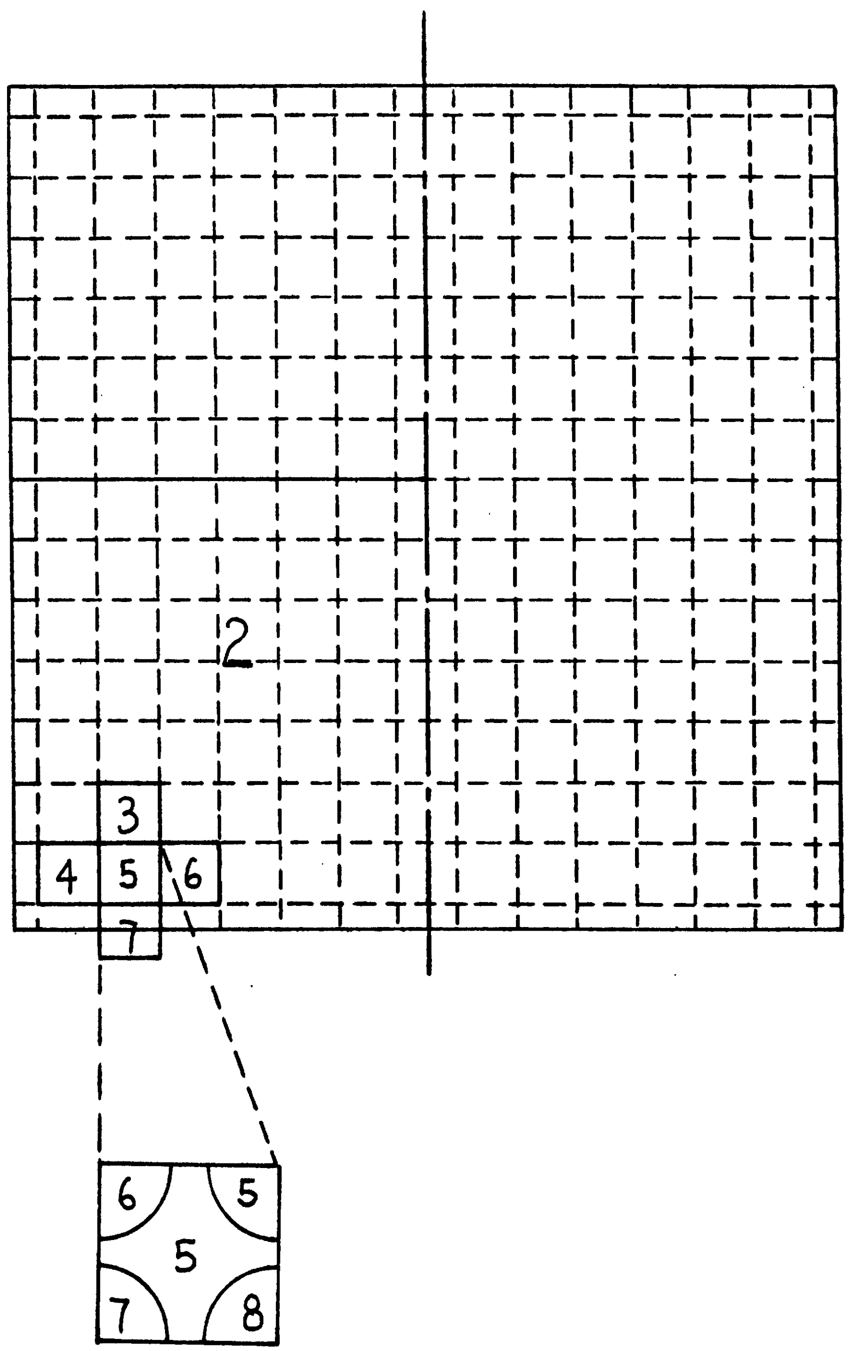


Figure 3.10 Channel Layout in Hot Assembly for Case 44

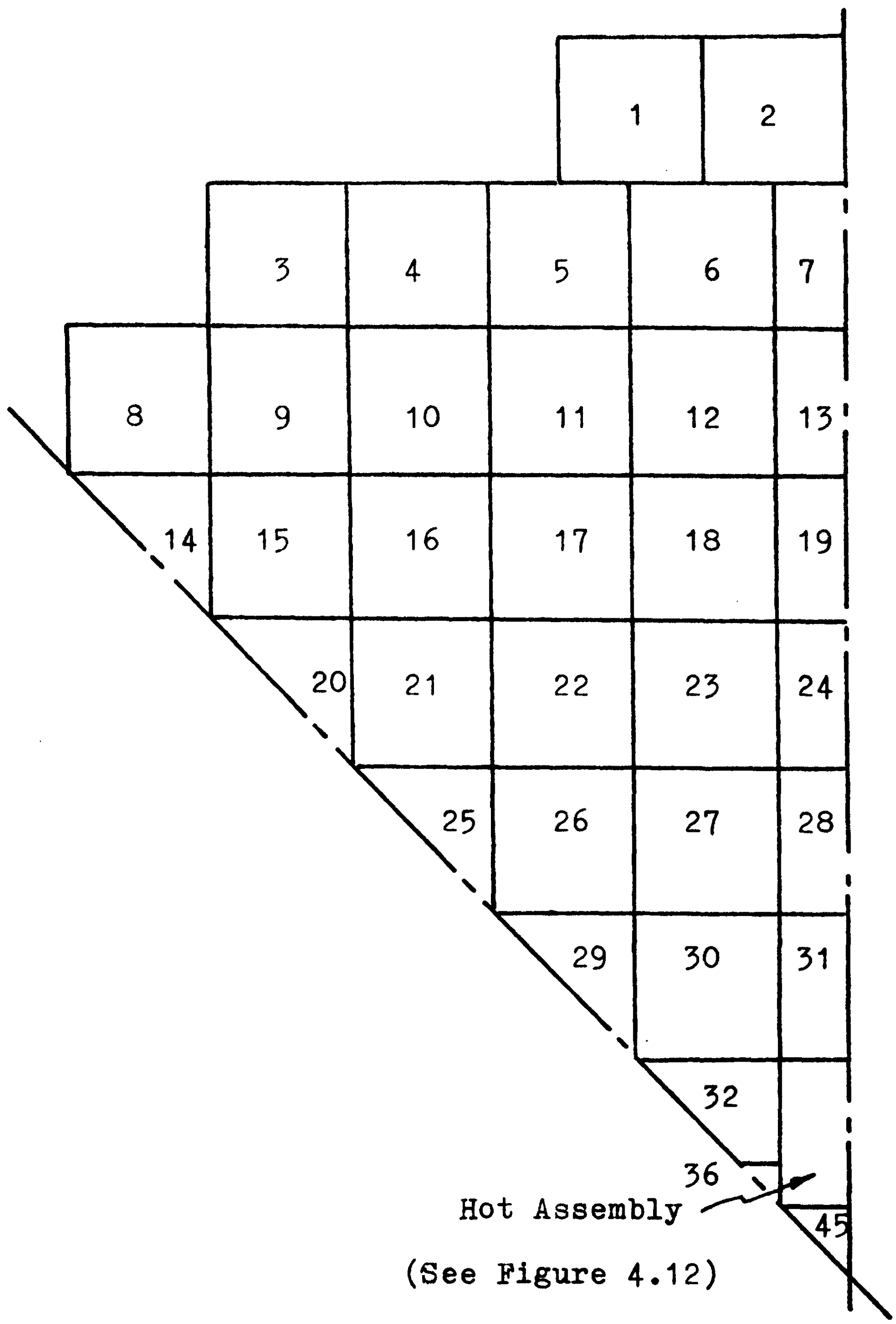


Figure 3.11 Layout of Coarse Mesh for Case 45

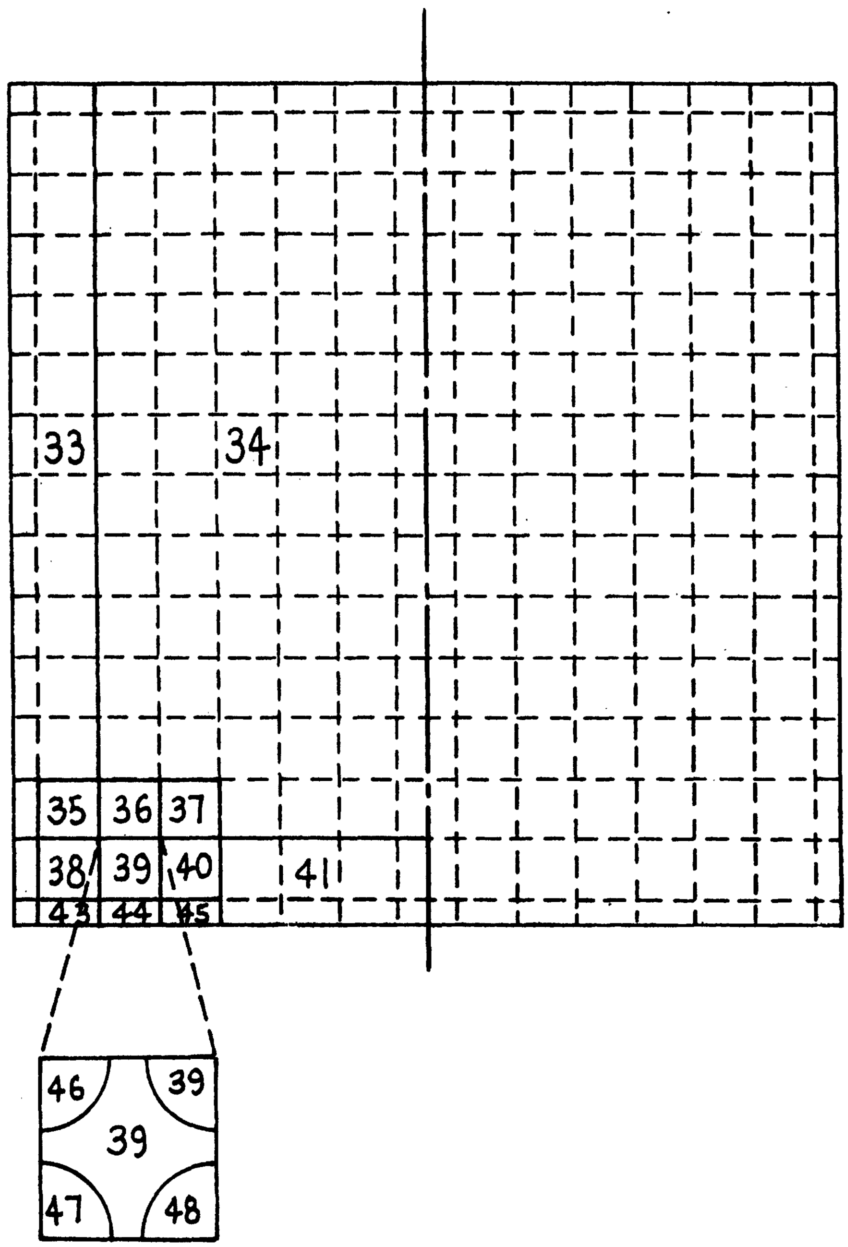


Figure 3.12 Channel Layout in Hot Assembly for Case 45

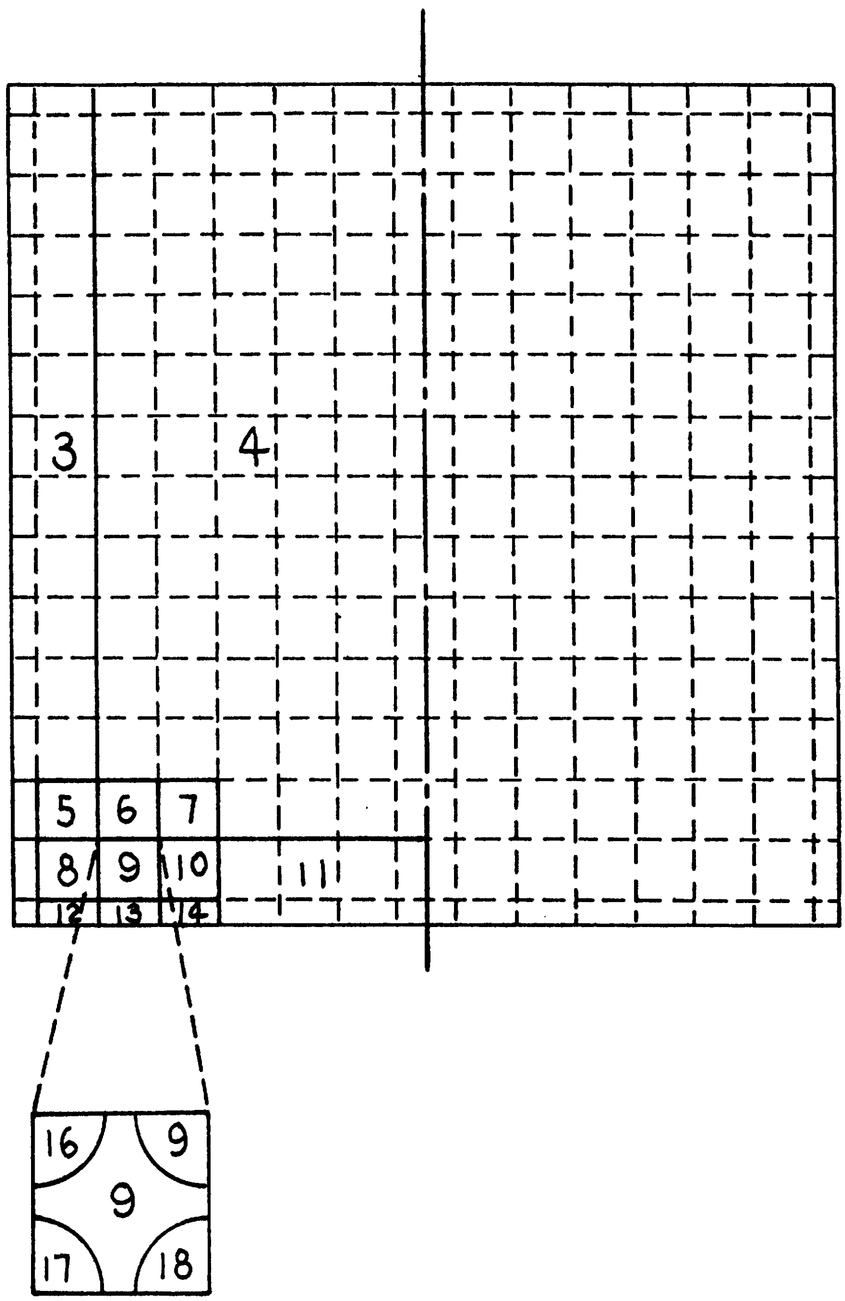


Figure 3.13 Channel Layout in Hot Assembly for Case 46

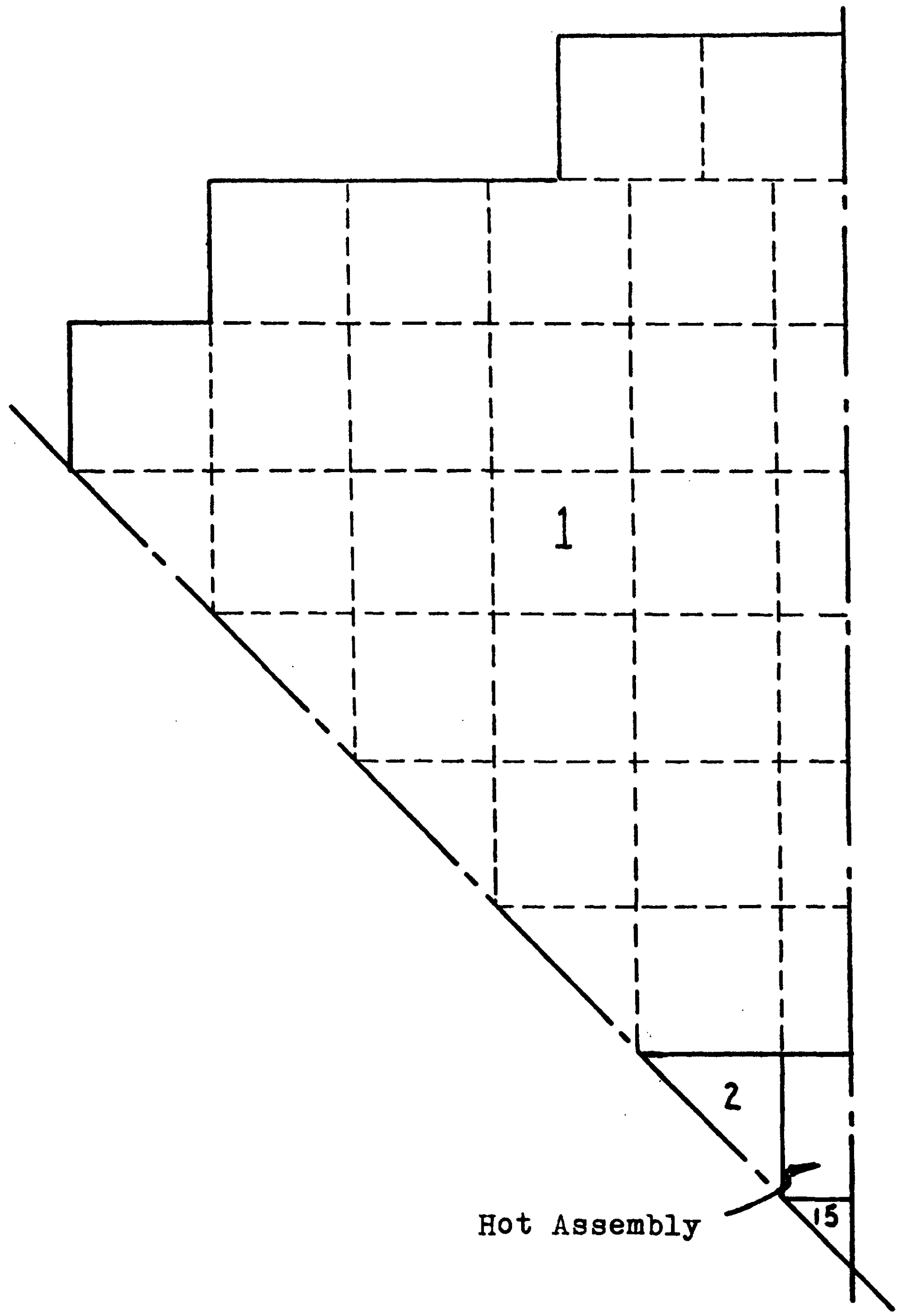


Figure 3.14 Layout of Coarse Mesh for Case's 46, 52, 53

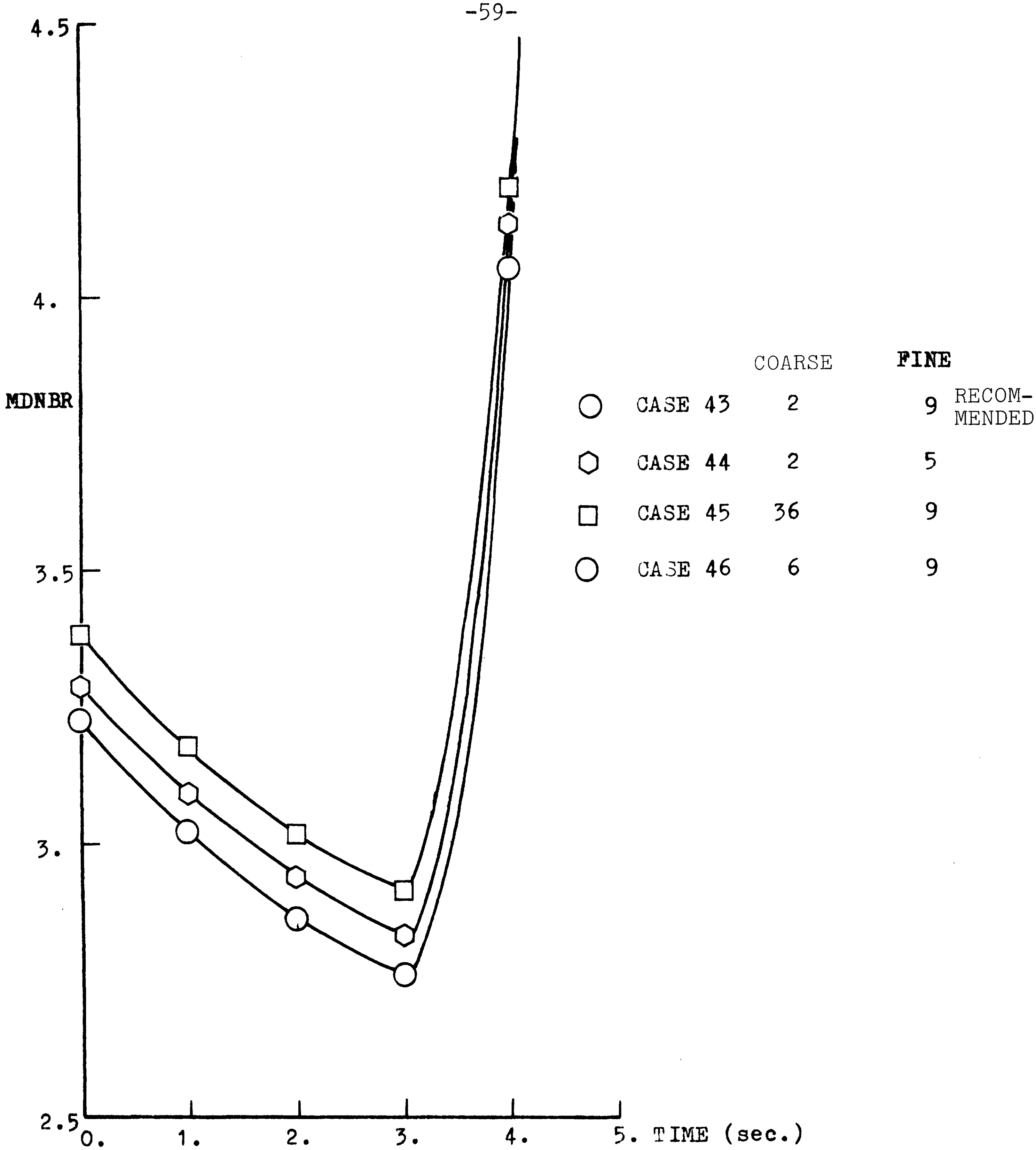


Figure 3.15 Sensitivity of MDNBR to Channel Layouts (Loss of Flow Transient with Power Trip)

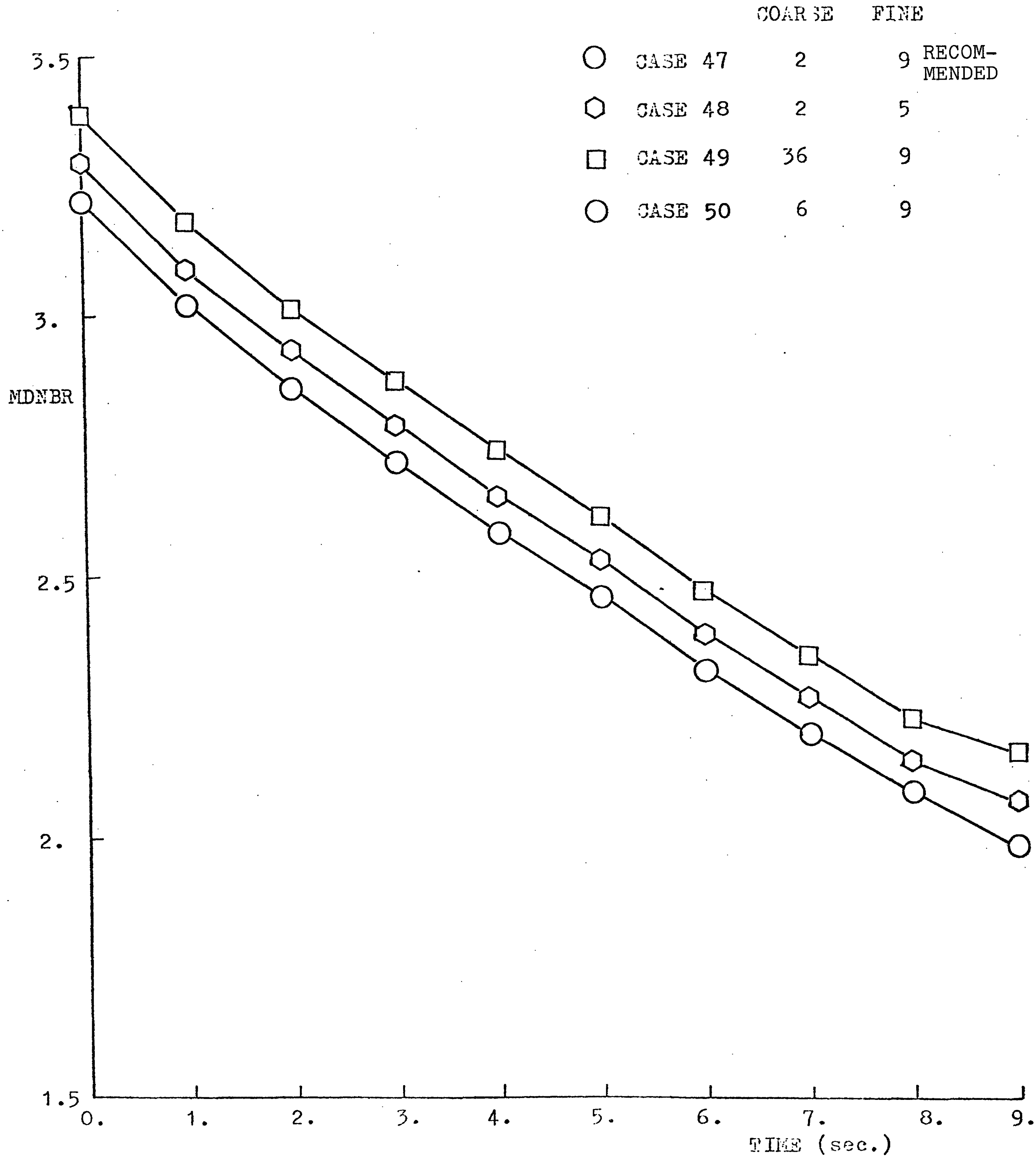
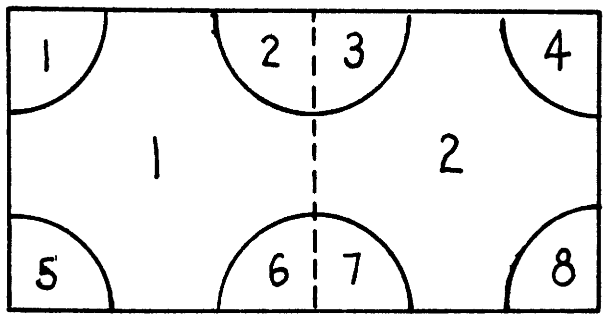


Figure 3 .16 Sensitivity of MDNBR to Channel Layouts
(Loss of Flow Transient without Power Trip)



Rod Number	1	2	3	4	5	6	7	8
Peaking Factors	0.	1.362	1.362	1.353	1.273	1.281	1.281	1.274

Channel Number	1	2
Peaking Factors	.979	1.317

Figure 3.17 Case 51 Channel Layout and Power Inputs

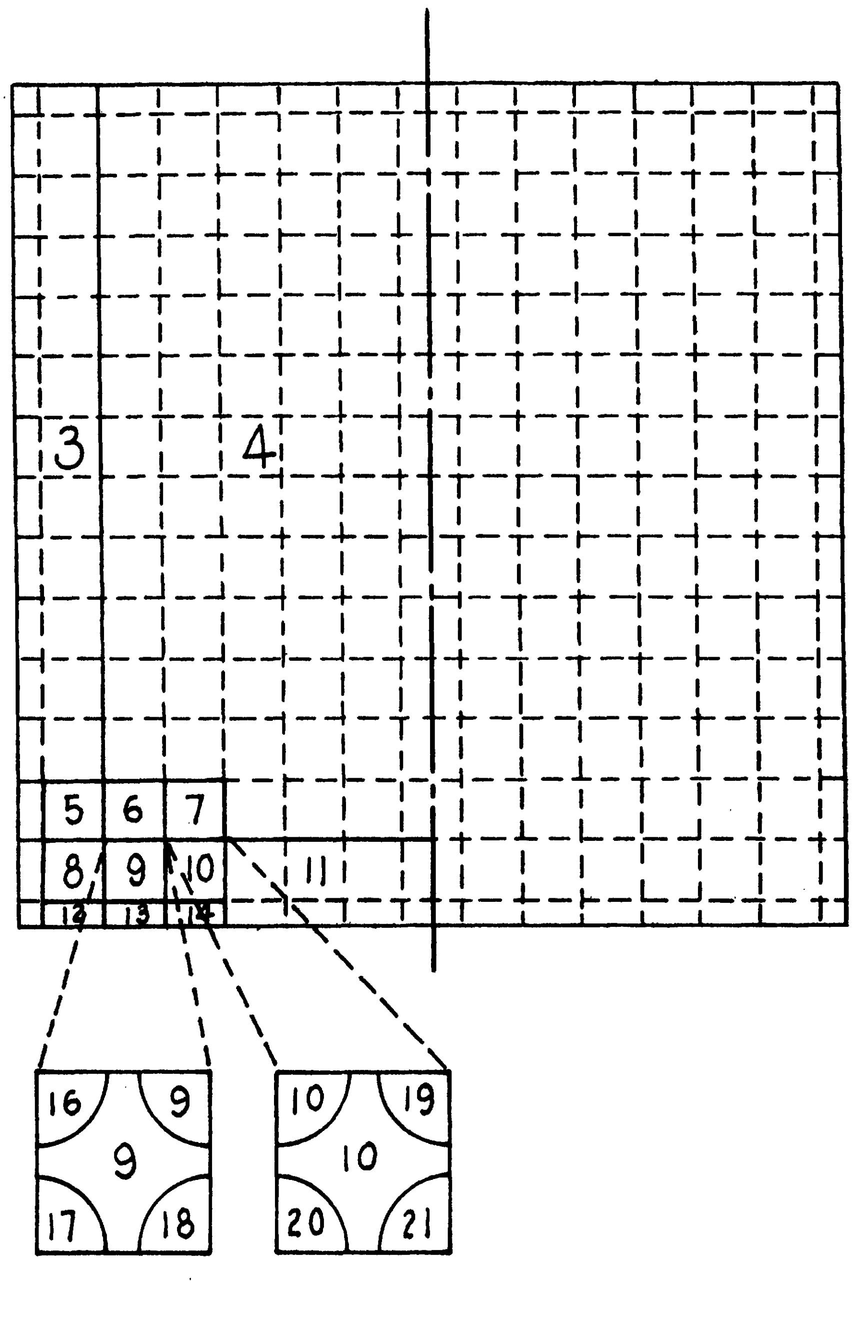


Figure 3.18 Channel Layout in Hot Assembly for Case 52

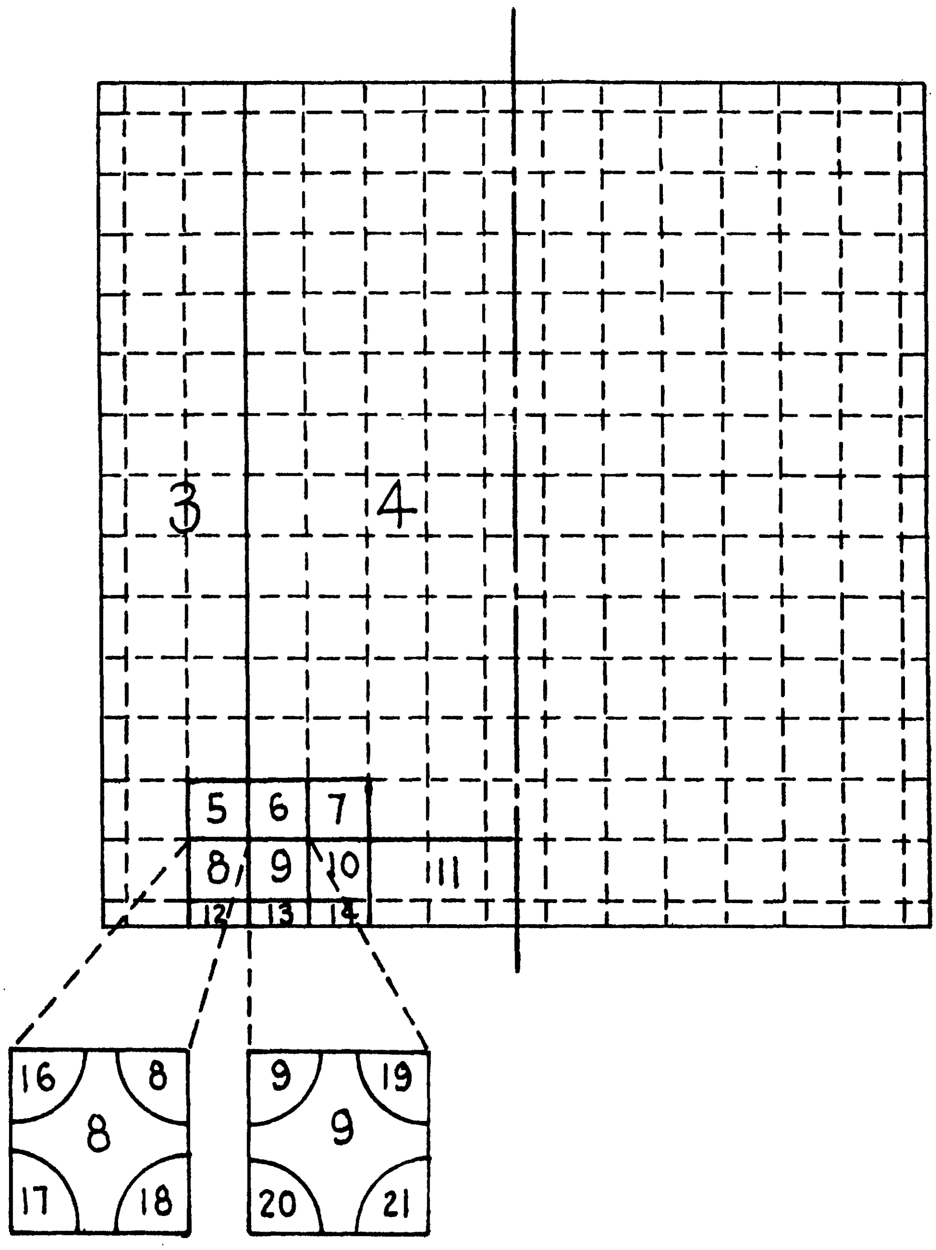


Figure 3.19 Channel Layout in Hot Assembly
for Case 53

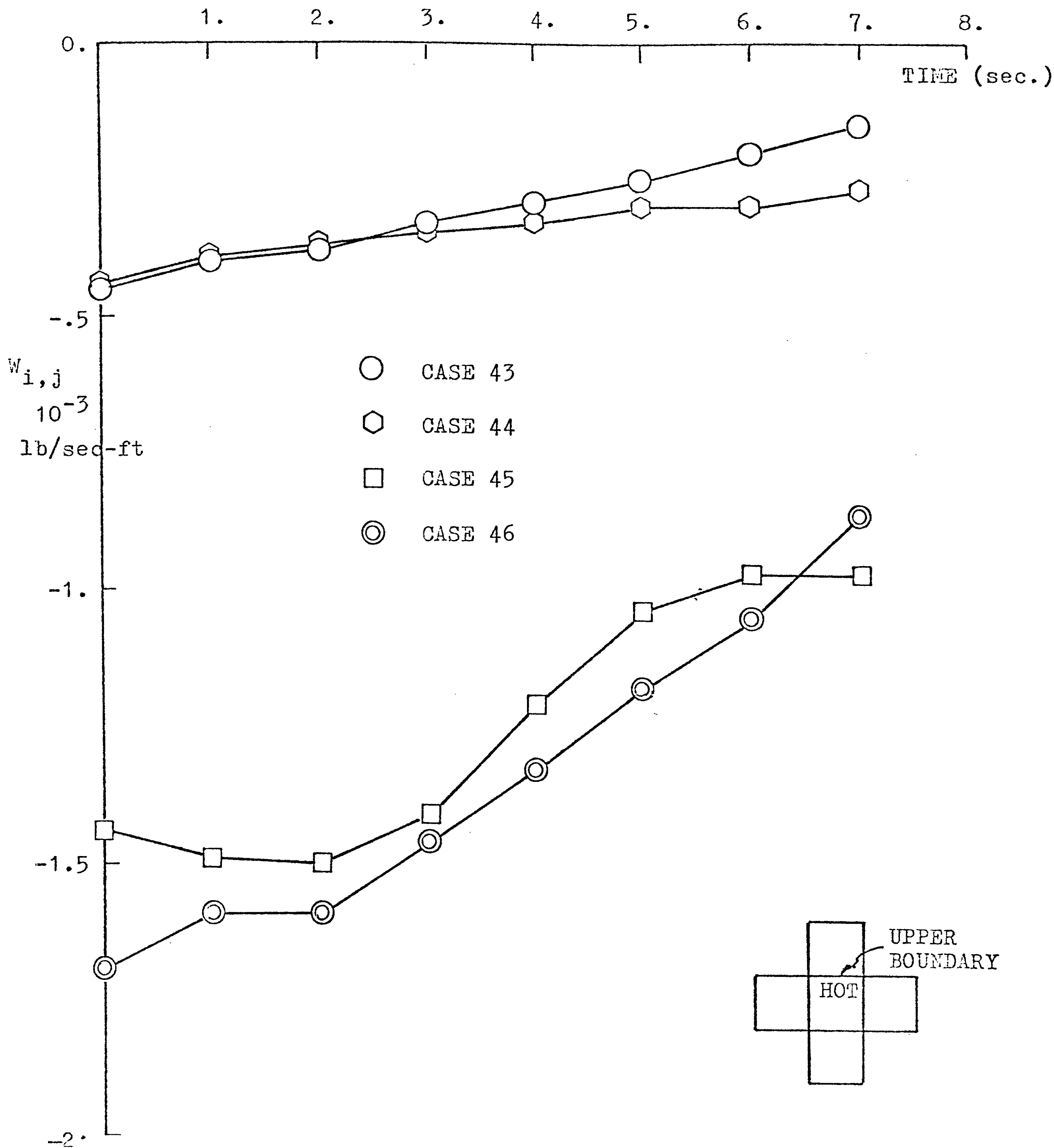


Figure 3.20 Flow Across Hot Channel Upper Boundary at Location of MDNBR

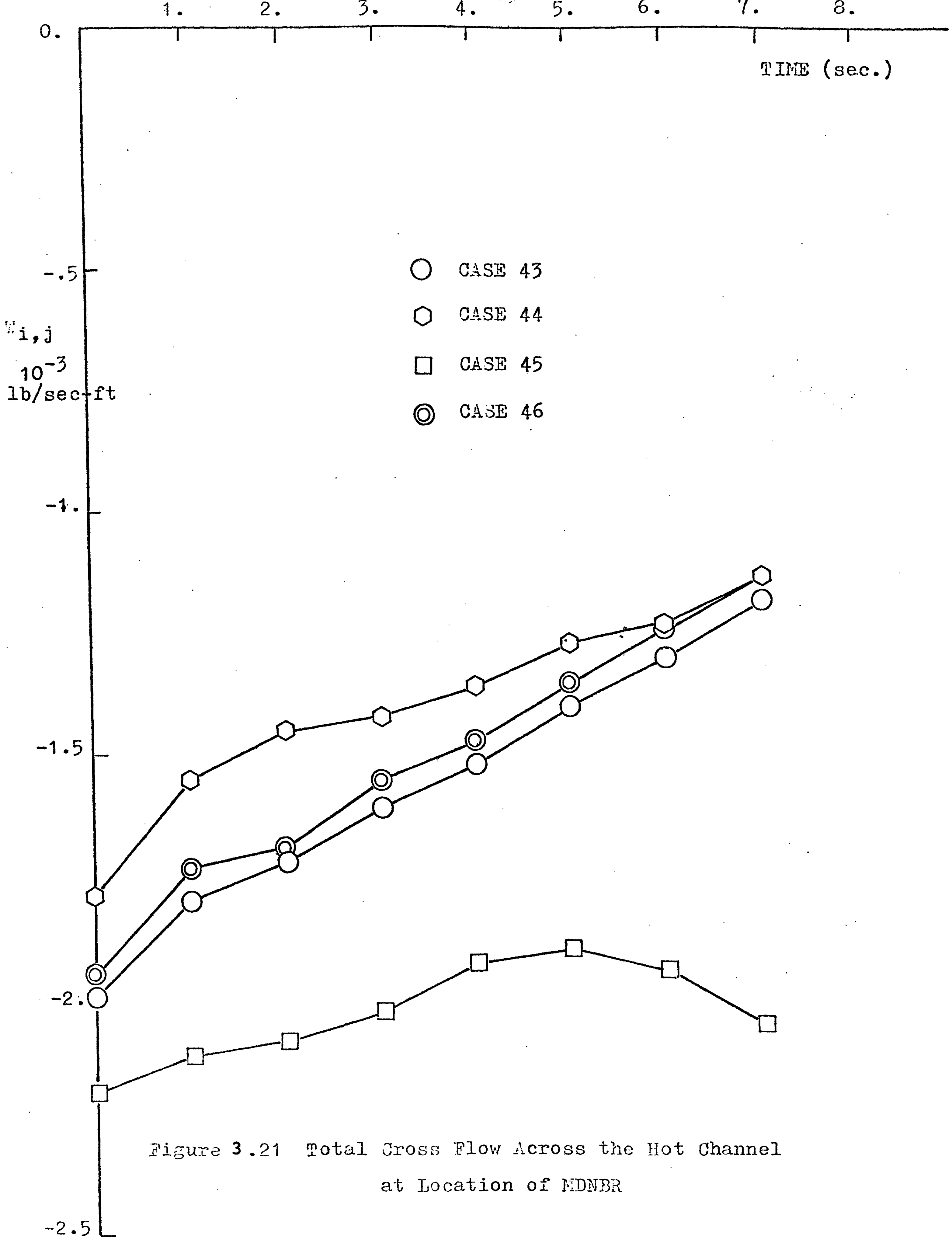


Figure 3.21 Total Cross Flow Across the Hot Channel at Location of MDNBR

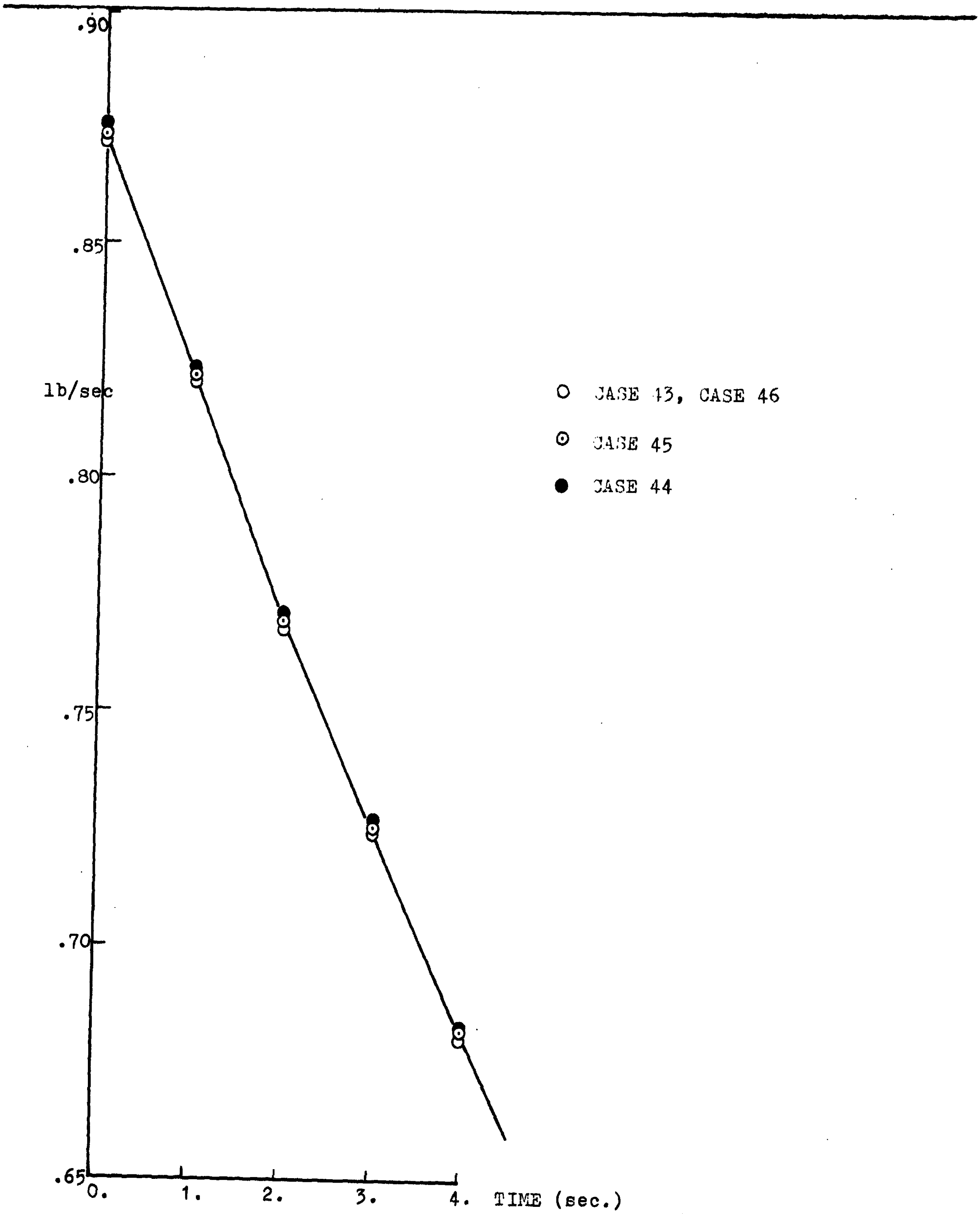


Figure 3.22 Sensitivity of Hot Channel Flow Rate at Location of MDNBR to Channel Layout (with or without Power Trip)

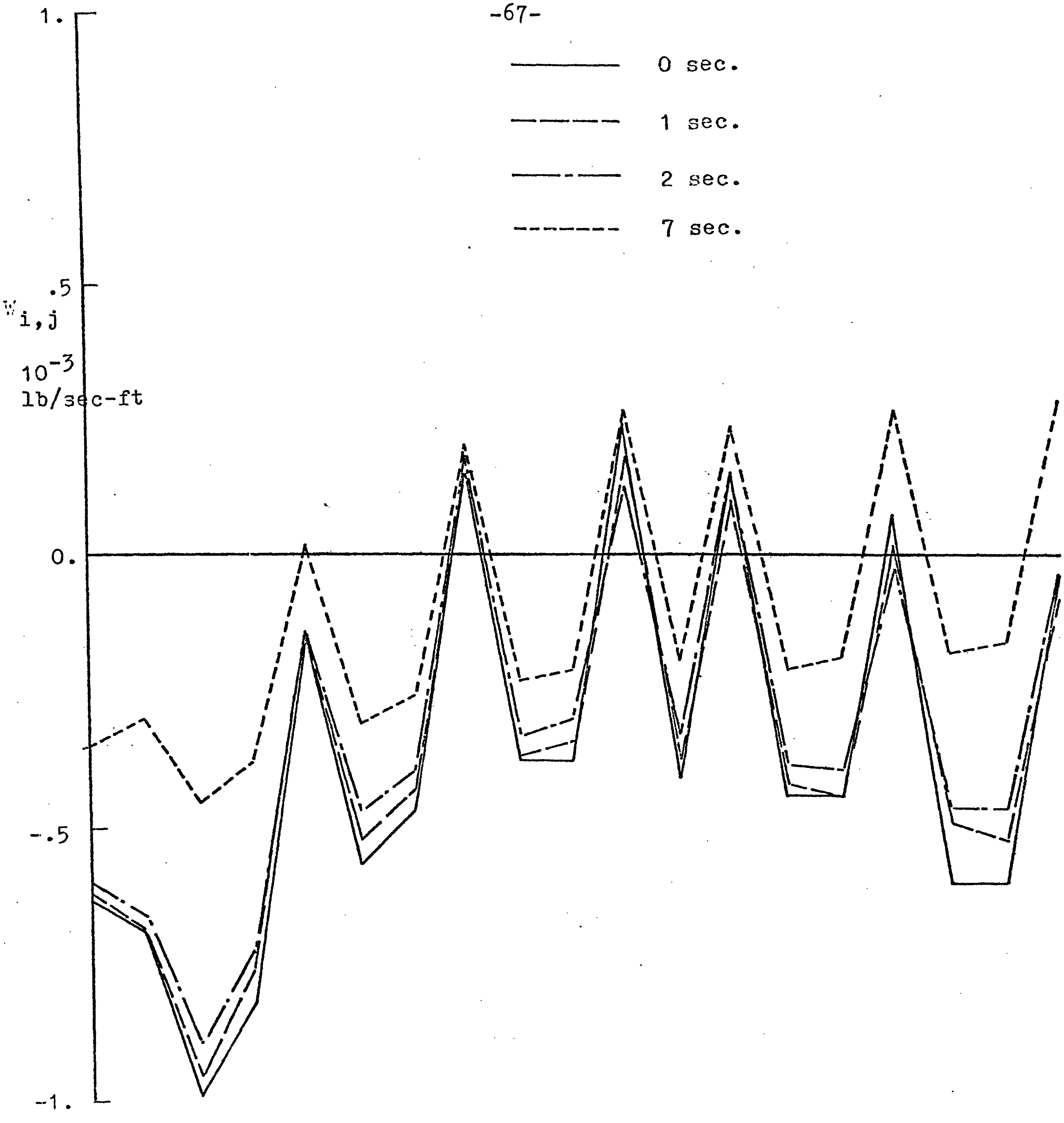


Figure 3.23 Cross Flow Pattern Along The Channel

(Case 43)

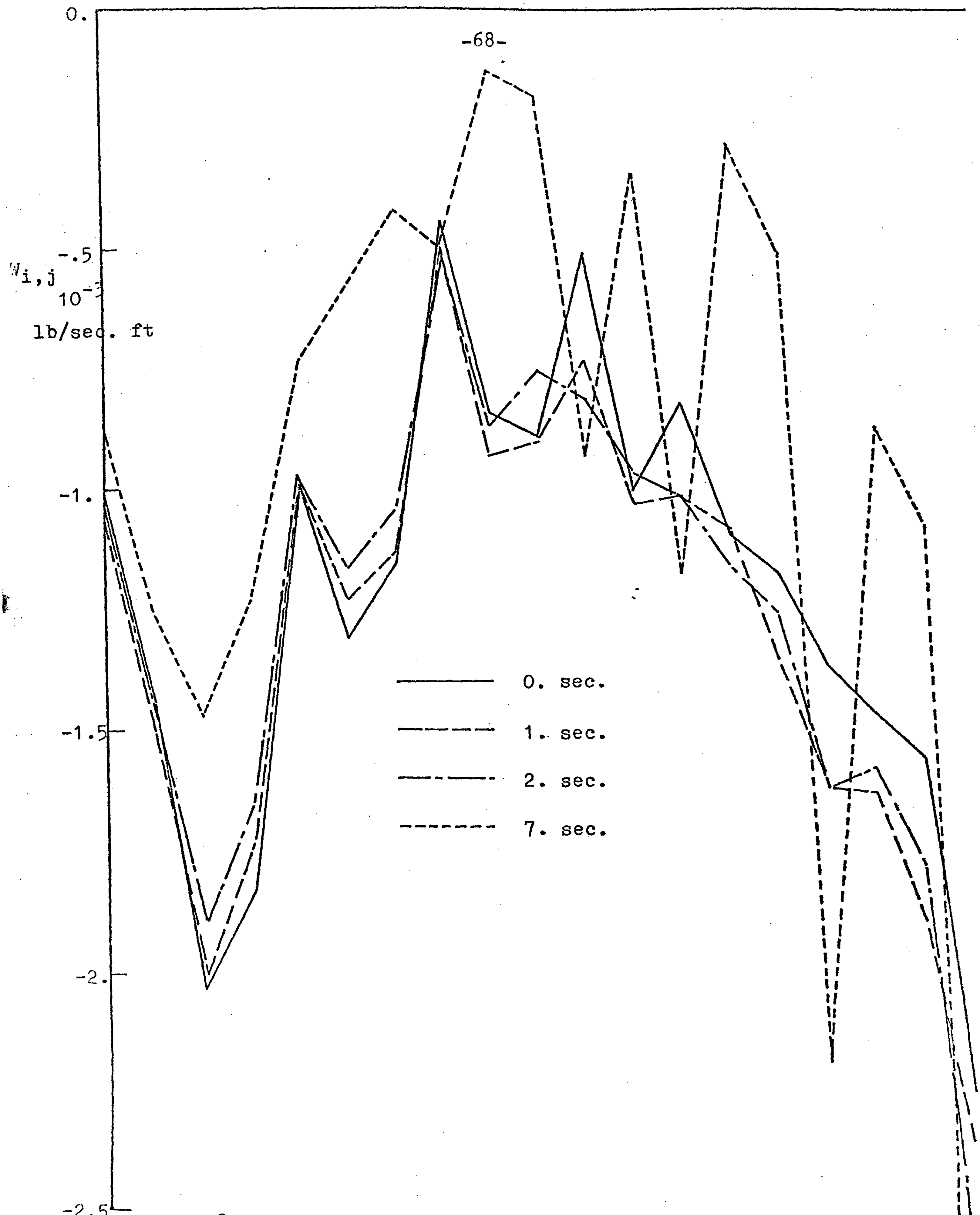


Figure 3.24 Cross Flow Pattern Along the Hot Channel (Case 46)

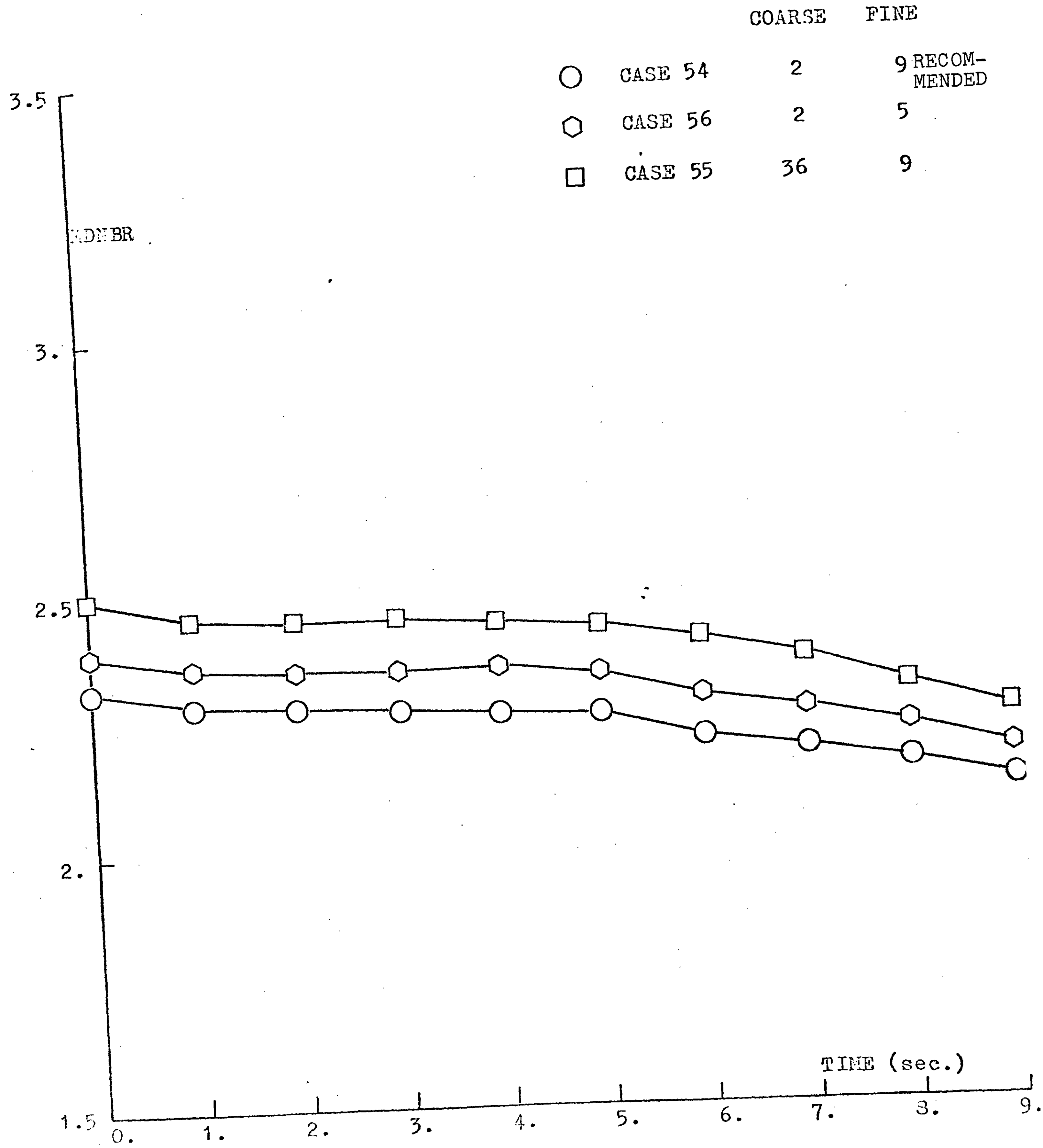


Figure 3.25 Sensitivity of MDNBR to Channel Layouts

Seized Rotor Transient Without Power Trip

Chapter 4

CONCLUSIONS

4.1 Analysis Criteria under Transient Conditions

Since neutronic feedback calculations under transient conditions have not been developed, the minimum DNB ratio was still used as the criteria for transient thermal analysis. Feedback effect should be considered. Especially for certain power transient, the effect is significant. This feedback effect can only be accomplished by combining the neutronic code and the thermal hydraulic code. Optimumal nodal pattern for both computer codes is the most important task to accomplish to apply the one stage simplified method analysis to cases with the consideration of the feed back effect.

4.2 Simplified Method Under Loss of Coolant Flow Transient Conditions

Under loss of coolant flow accident transient condition, the degree of the validity of the simplified method is just as good as the method under steady state condition. Since there is no change of the radial power distribution, and MDNBR is most sensitive to the radial power, there is no shift of the hot rod under this transient condition even with a decreasing of flow rate. Therefore, the method to identifying the hot rod under steady state condition is still valid under this transient condition. From the results of the analysis, it is clear that the differences of MDNBR under steady state condition is preserved under this transient condition.

4.3 Simplified Method Under Seized Rotor Transient Condition

Only the seized rotor accident without any consideration of the inlet flow maldistribution has been analyzed in this study. Due to the nearly constant behavior of this seized rotor with assumed uniform inlet flow transient, the analysis for this accident can be accomplished by the steady state analysis method. The effect of the inlet flow distribution is more important than other parameters(4) (e.g. crossflow resistance). Therefore, accurate inlet flow distribution is necessary to analyze a realistic seized rotor transient.

4.4 The Limitations of the Simplified Method under Transient Conditions

The one stage simplified method was developed by using the thermal hydraulic computer code COBRA IIIC/MIT. Most of the limitation of the simplified method under transient conditions are the limitations of the thermal hydraulic computer code. Transient forcing function of power, coolant flow rate, pressure, and inlet enthalpy have to be obtained from other sources of data. These data can be obtained either by running a system computer code, neutronic computer code or by taking actual plant data. For cases with significant effect of neutronic feedback, the input transient forcing functions should be modified. Most of the transient conditions mentioned in this study are involved with either the system transient behavior of the whole reactor or transient phenominum of the neutronic feedback. Most of the cases are involved with both. Limitations of the simplified method indeed exist for cases lacking information on the input forcing functions and the degree of the effect of the neutronic feedback.

Another limitation which is not related to the thermal hydraulic code is the ability to identify the hot rod under certain transient conditions which will cause the radial power distribution and axial power distribution to change. Since the simplified method is based on the accurate identification of the hot rod, if the hot rod migrates because of the change of the radial power distribution, the simplified method, in this case, is no longer valid. This transient situation was qualitatively investigated as presented in Appendix C.

4.5 Results of MDNBR

Results of MDNBR presented in this chapter were obtained by analyzing the Maine Yankee core with the data of cycle II. The reason for the higher MDNBR's than expected are merely because of the physical characteristics of the Maine Yankee core cycle II data. The control rod is next to the highest hot rod. These higher than normal results of MDNBR are also confirmed by some other investigators. For example, Emami [6] did a case with 1.5 Mlb/hr ft^2 coolant flow rate, the MDNBR is 2.31 for assembly with power peaking factor of 1.25. Therefore, the results of MDNBR presented in this chapter are reasonable even they are higher than expected. Also, the point which was emphasized in this chapter is the differences of MDNBR between different cases rather than the absolute value of MDNBR.

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(3) Robert Bowring and Pablo Moreno, COBRA IIIC/MIT Computer Code Manual to be issued as an EPRI Report.

(4) P. Moreno, C. Chiu, R. Bowring, E. Kahn, J. Liu, N. Todreas "Methods for Steady State Thermal/Hydraulic Analysis of Pressurized Water Reactor Cores" MIT-EL 76-006 March 1977.

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(6) Farideh Emani "Thermal Hydraulic Sensitivity Study" M.S. Thesis. M.I.T. 1977.

APPENDIX A

DESCRIPTION OF THE DATA USED IN THE ANALYSIS

In this appendix the data used in the present study will be given. The reactor analyzed is Main Yankee, PWR Core.

A.1 Main Yankee PWR Reactor

A.1.1 Operating Conditions

System Pressure 2100 psia

Uniform Inlet Temperature 541°F

Uniform Inlet Mass Velocity 2.48 x 10^6 lb/hr ft^2

Average Heat Flux 0.1695 x 10^6 BTU/hr ft^2

A.1.2 Dimensions of the Assemblies

7.98 inches suare 14 x 14 array

Channel Length 136.7 in (active length)

A.1.3 Dimensions of the Subchannels

Rod Diameter: 0.440 in

Rod Pitch: 0.580 in

Diameter of CEA Guide Tubes: 1.115 in

A.1.4 Axial Power Distributions

There are three different shapes of axial power distribution which are shown in Chapter 3 Figure 3 .2.

A.1.5 Radial Power Peaking Factors:

Figure's A. 1 and A. 2 show the assembly radial power peaking factor with water and without water respectively. Figure's A. 3 and A. 4 illustrate the radial power peaking of rod and subchannel respectively.

A.1.6 Spacer Data

Number of Types: 2

Single Phase Grid Coefficient

for each Grid Type

$K = 0.4605$

$K_{inlet} = 0.6448$

$K_{outlet} = 0.5544$

Spacer Grid locations are shown in Figure A.5

A.2 Thermal-Hydraulic Model

Similar thermal-hydraulic model for PWR reactors have been used in the study.

A.2.1 Mixing

The mixing coefficient β of 0.02 have been used in the analysis

The two-phase mixing coefficient is taken as equal to that of single phase.

Thermal Conduction is neglected.

A.2.2 Single-Phase Friction

It is calculated by:

$$F = \frac{0.184}{R_e^{0.2}}$$

where R_e = n Reynolds Number

A.2.3 Two-Phase Friction

The homogeneous model friction multiplies was selected to describe the two-phase pressure drop due to friction.

A.2.4 Void Fraction

It was calculated using the Levy model and a slip ratio equal to one.

A.2.5 Flow Division at Inlet

The inlet mass velocity was taken as uniform, for all channels.

A.2.6 Constants

The constants used are:

Cross-flow resistance (KIJ) = 0.5

Momentum Turbulent Factor (FTM) = 0.0

Transverse Momentum Factor (S/L) = 0.5

The CHF correlation used in all the calsulations was W-3.

A.2.7 Iteration

The flow convergence factor used was 0.01.

The number of axial steps in which the core was divided was 21.

A.2.8 Coupling Parameter

The following coupling parameter was used:

$$N_H = N$$

where

N = number of rods between the center lines of the channels making up the boundary conditions.

If no coupling parameter was used,

$$N_H = 1.$$

A.3 Forcing Functions Used in the Transient Analysis

A.3.1 Power Forcing Functions

A.3.1.1 Case A — Without Reactor Trip

This forcing function is shown in Figure A.6

A.3.1.2 Case B — With Reactor Trip

It is shown in Chapter 3 Figure 3.1

A.3.2 Inlet Flow Forcing Functions

A.3.2.1 Case A — Three Pump Coastdown Accident

It is shown in Chapter 3 Figure 3.1

A.3.2.2 Case B — Seized Rotor Accident

It is shown in Figure A.6

						1 .519	2 .653	
				3 .486	4 .660	5 .789	6 .932	7 .848
			8 .567	9 .821	10 .810	11 .872	12 .926	13 .924
		14 .567	15 .857	16 .859	17 .891	18 .931	19 .942	20 .964
	21 .486	22 .821	23 .859	24 .904	25 .943	26 .951	27 .981	28 .970
	29 .660	30 .810	31 .891	32 .943	33 .956	34 .989	35 .983	36 1.003
45 .519	37 .789	38 .872	39 .931	40 .951	41 .989	42 .986	43 1.012	44 .997
54 .653	46 .932	47 .926	48 .942	49 .981	50 .983	51 1.012	52 1.000	53 1.021
	55 .848	56 .924	57 .964	58 .970	59 1.003	60 .997	61 1.021	62 1.004

Figure A.1 Maine Yankee Average Power in Assembly with Water (Average Over Rods)

CHANNEL NUMBER

AVERAGE POWER

NO. OF RODS (FUEL)

						1 .578 176	2 .727 176	
				3 .541 176	4 .735 176	5 .943 164	6 1.113 164	7 1.039 160
			8 .631 176	9 .958 168	10 .992 160	11 .971 176	12 1.134 160	13 1.029 176
		14 .631 176	15 1.000 168	16 1.052 160	17 .992 176	18 1.141 160	19 1.049 176	20 1.181 160
	21 .541 176	22 .958 168	23 1.052 160	24 1.007 176	25 1.156 160	26 1.059 176	27 1.202 160	28 1.081 176
	29 .735 176	30 .992 160	31 .992 176	32 1.156 160	33 1.065 176	34 1.211 160	35 1.094 176	36 1.229 160
45 .578 176	37 .943 164	38 .971 176	39 1.141 160	40 1.059 176	41 1.211 160	42 1.098 176	43 1.240 160	44 1.110 176
54 .727 176	46 1.113 164	47 1.134 160	48 1.049 176	49 1.202 160	50 1.094 176	51 1.240 160	52 1.114 176	53 1.251 160
	55 1.039 160	56 1.029 176	57 1.181 160	58 1.081 176	59 1.229 160	60 1.110 176	61 1.251 160	62 1.119 176

Figure A.2 Maine Yankee Average Power in Assembly Fuel Rod Only (Average Over Fuel Rods)

1.244	1.269	1.276	1.269	1.247	1.205	1.176	1.176	1.205	1.247	1.269	1.276	1.269	1.244
1.269	0.	1.358	1.348	0.	1.220	1.172	1.172	1.220	0.	1.348	1.358	0.	1.269
1.277	1.358	0.	0.	1.330	1.224	1.172	1.172	1.224	1.330	0.	0.	1.358	1.277
1.270	1.349	0.	0.	1.327	1.227	1.182	1.182	1.227	1.327	0.	0.	1.349	1.270
1.248	0.	1.331	1.327	0.	1.236	1.212	1.212	1.236	0.	1.327	1.331	0.	1.248
1.207	1.221	1.224	1.228	1.236	1.256	1.292	1.292	1.256	1.236	1.228	1.224	1.221	1.207
1.178	1.174	1.173	1.183	1.213	1.292	0.	0.	1.292	1.213	1.183	1.173	1.174	1.178
1.178	1.174	1.173	1.183	1.213	1.292	0.	0.	1.292	1.213	1.183	1.173	1.174	1.178
1.208	1.223	1.226	1.229	1.237	1.257	1.293	1.293	1.257	1.237	1.229	1.226	1.223	1.208
1.250	0.	1.333	1.329	0.	1.237	1.214	1.214	1.237	0.	1.329	1.333	0.	1.250
1.273	1.352	0.	0.	1.330	1.230	1.184	1.184	1.230	1.330	0.	0.	1.352	1.273
1.280	1.362	0.	0.	1.334	1.227	1.175	1.175	1.227	1.334	0.	0.	1.362	1.280
1.273	0.	1.362	1.353	0.	1.224	1.176	1.176	1.224	0.	1.353	1.362	0.	1.273
1.249	1.273	1.281	1.274	1.251	1.210	1.180	1.180	1.210	1.251	1.274	1.281	1.273	1.249

Figure A. 3 Maine Yankee Hot Assembly Rod Power Peaking Factors

1.169														1.169
	1.178	1.193	1.196	1.181	1.153	1.125	1.114	1.125	1.153	1.181	1.218	1.193	1.178	
1.179	.945	.976	1.313	.966	.918	1.193	1.174	1.193	.918	.966	1.313	.976	.945	1.179
1.194	.976	.679	.677	.670	.943	1.197	1.172	1.197	.943	.670	.677	.679	.976	1.194
1.197	1.313	.677	0.	.664	1.277	1.201	1.177	1.201	1.277	.664	0.	.677	1.313	1.197
1.183	.967	.670	.665	.664	.947	1.214	1.197	1.214	.947	.664	.665	.670	.967	1.183
1.155	.919	.944	1.278	.948	.932	1.249	1.252	1.249	.932	.948	1.278	.944	.919	1.155
1.127	1.195	1.198	1.202	1.215	1.249	.960	.646	.960	1.249	1.215	1.202	1.198	1.195	1.127
1.116	1.176	1.174	1.178	1.198	1.253	.646	0.	.646	1.253	1.198	1.178	1.174	1.176	1.116
1.127	1.196	1.199	1.203	1.216	1.250	.961	.647	.961	1.250	1.216	1.203	1.199	1.196	1.127
1.156	.920	.945	1.279	.949	.933	1.250	1.253	1.250	.933	.949	1.279	.945	.920	1.156
1.185	.969	.671	.666	.665	.949	1.216	1.199	1.216	.949	.665	.666	.671	.969	1.185
1.199	1.317	.678	0.	.666	1.280	1.204	1.180	1.204	1.280	.666	0.	.678	1.317	1.199
1.197	.979	.681	.679	.672	.946	1.200	1.175	1.200	.946	.672	.679	.681	.979	1.197
1.183	.949	.979	1.317	.969	.921	1.197	1.178	1.197	.921	.969	1.317	.979	.949	1.183
	1.183	1.198	1.200	1.186	1.158	1.129	1.118	1.129	1.158	1.186	1.200	1.198	1.183	
1.174														1.174

Figure A.4 Maine Yankee Hot Assembly Channel Power Peaking Factors

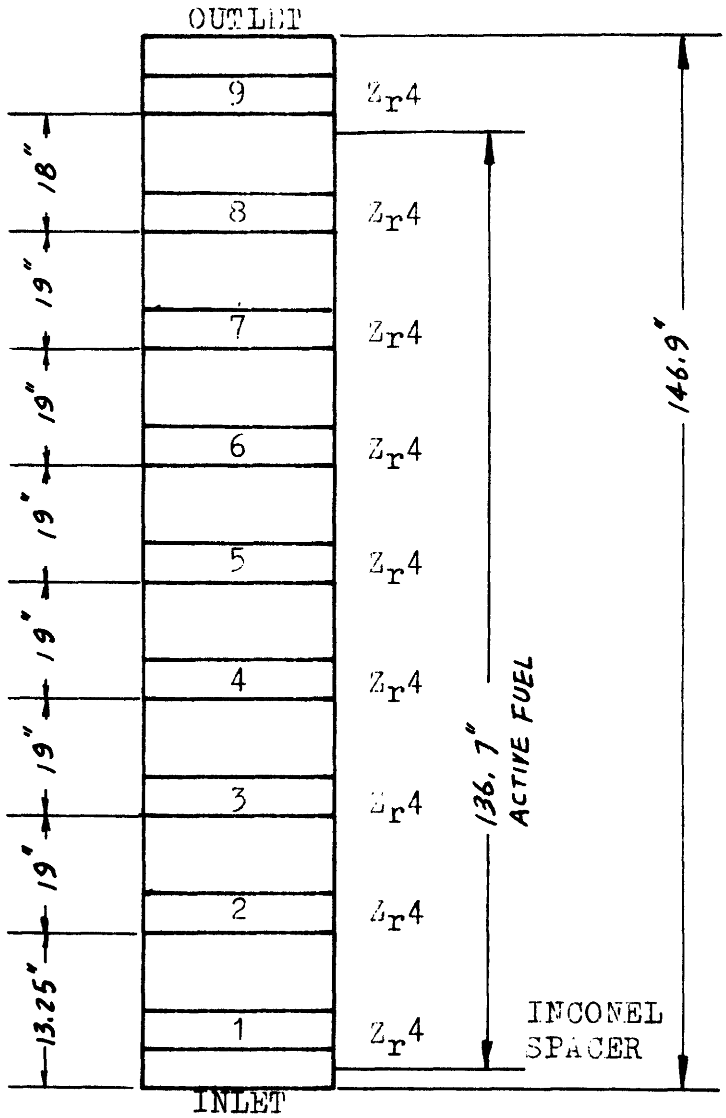

Figure A.5 Spacer Grid Locations

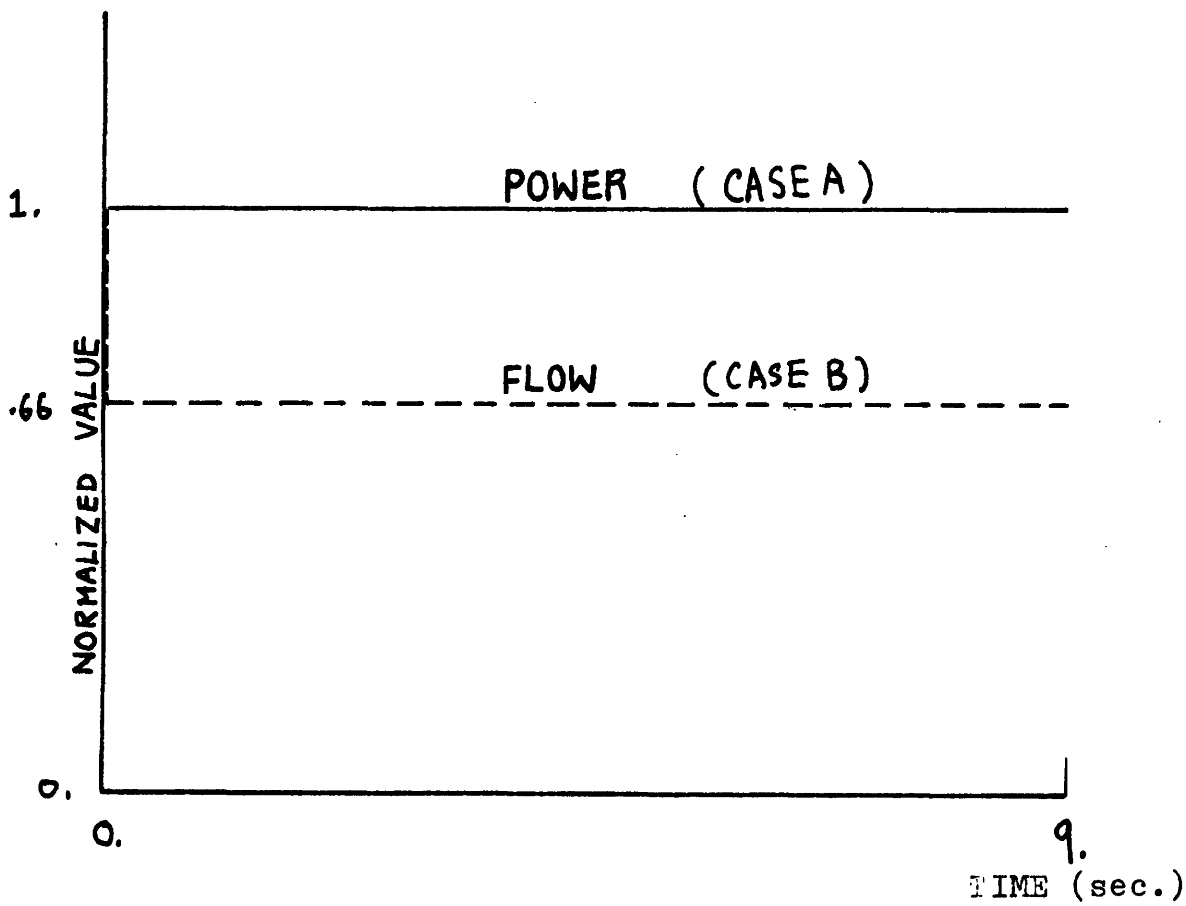

Figure A.6 Power and Flow Forcing Functions

Appendix B

CASES ANALYZED UNDER TRANSIENT CONDITIONS

All of the cases analyzed under transient conditions were based on the Maine Yankee PWR core. All the data used were presented in Appendix A. In this appendix a summary of cases analyzed under transient conditions is presented in tabulated form. The pattern of thermal model used for each case is shown in the figures. Assigned case numbers are the same as in the text. Six groups of cases have been analyzed in the transient study, these are:

(1) Cases analyzed in order to investigate the axial power distribution: Case's 35, 36, 37,

(2) Cases analyzed in order to check the sensitivity of time step: Case 38,

(3) Cases analyzed in order to check the loss of coolant flow transient effect on MDNBR: Case's 39, 40, 41, 42,

(4) Cases analyzed in order to check the location of MDNBR: Case's 51, 52, 53,

(5) Cases analyzed in order to assess the validity of the simplified method for loss of coolant flow and seized rotor transient conditions: Case's 43 through 50,

(6) Cases analyzed in order to check the importance of transport coefficient (N_H) under loss of coolant flow and seized rotor transient conditions: Case's 57, 58.

Case No.	Channel Model		Axial Power Distribution	Transport Coef.	Flow Transient Forcing Function	Power Transient Forcing Function
	Total Nodes	Figure				
35	2	B.1	C	1.	A	A
36	2	B.1	B	1.	A	A
37	2	B.1	A	1.	A	A
38	2	B.1	A	1.	A	B
39	2	B.2	A	1.	A	A
40	2	B.2	A	1.	A	A
41	3	B.3	A	1.	A	A
42	4	B.4	A	1.	A	A
43	11	3.9, 3.8	A	1.	A	B
44	7	3.9, 3.10	A	1.	A	B
45	45	3.11, 3.12	A	1.	A	B
46	15	3.13, 3.14	A	1.	A	B
47	11	3.8, 3.9	A	1.	A	A
48	7	3.9, 3.10	A	1.	A	A
49	45	3.11, 3.12	A	1.	A	A
50	15	3.13, 3.14	A	1.	A	A
51	2	3.17	A	1.	A	A
52	15	3.14, 3.18	A	1.	A	A
53	15	3.14, 3.19	A	1.	A	A
54	11	3.8, 3.9	A	1.	B	A
55	45	3.11, 3.12	A	1.	B	A
56	7	3.9, 3.10	A	1.	B	A
57	11	3.8, 3.9	A	N	A	A
58	11	3.8, 3.9	A	N	B	A

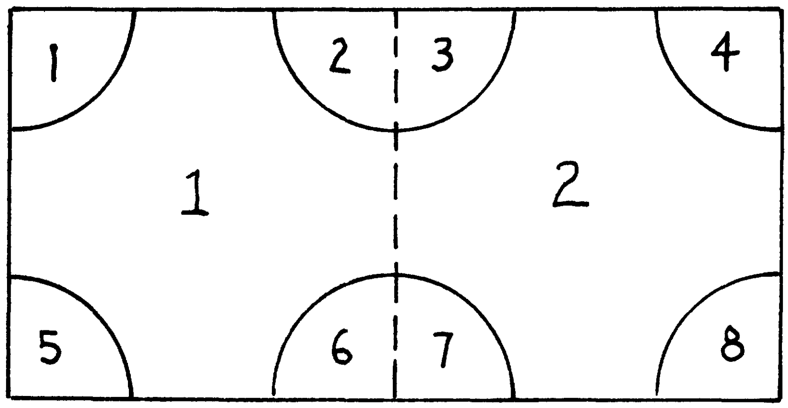

Figure B.1

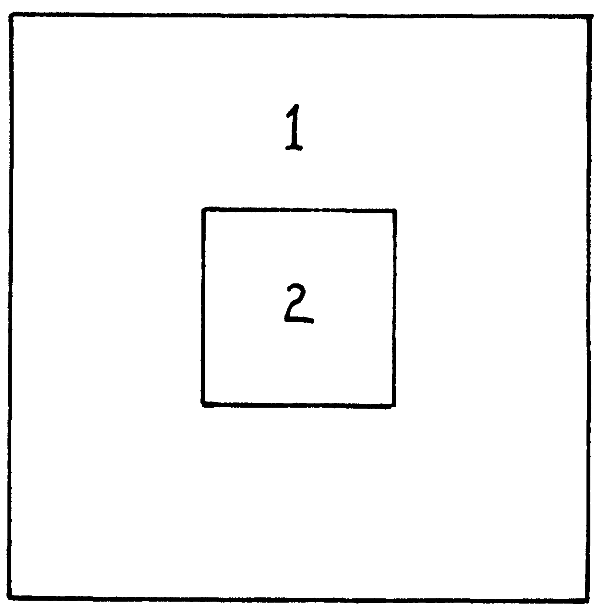

Figure B.2

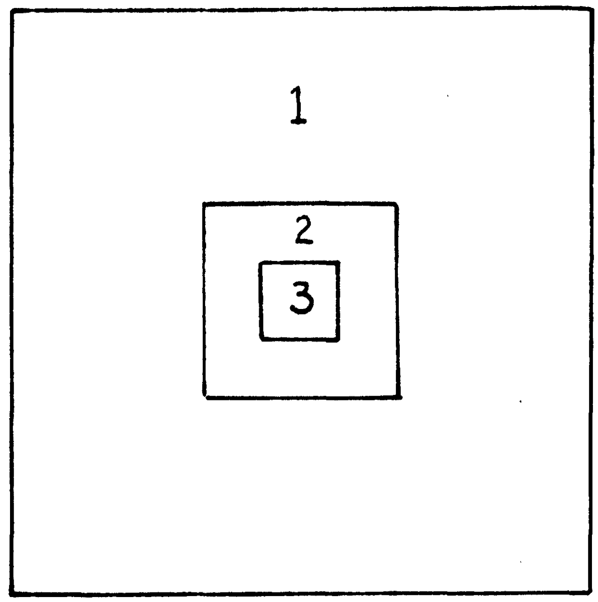

Figure B.3

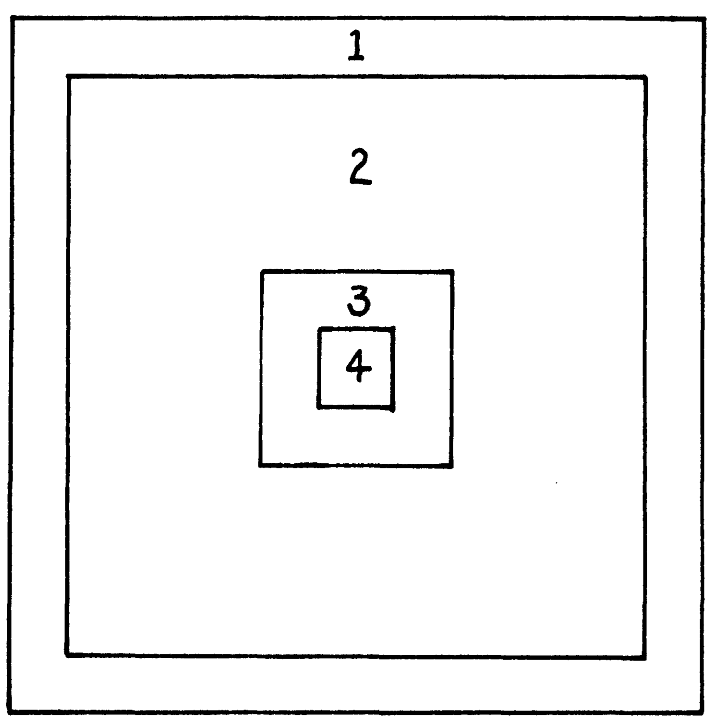

Figure B.4

Appendix C

IDENTIFICATION OF THE HOT ROD FOR CASES WITH COMPLICATED POWER DISTRIBUTION

C.1 Introduction

In this appendix, definition of the minimum departure from nucleate boiling ratio (MDNBR) is given first. From this definition, the influence of power distribution on MDNBR is discussed. A simple example case was set up in order to have a qualitative description of a proposed guide line to identify the hot rod under condition of a complicated power distribution due to cases like control rod insertion, fuel depletion,... etc.

C.2 MDNBR

The mechanism of the boiling crisis differs in the various flow patterns. In subcooled boiling, a flow pattern indicative of a PWR, the bubble boundary layer flows parallel to the wall with a liquid core flowing at the center of the tube. The local void fraction shows a peak near the surface. When the bubble layer separates from the wall, a stagnant fluid forms under the layer. Due to the high heat flux at the surface, this stagnant fuid evaporates resulting in a vapor blanket on the heated wall. Thus the local boiling heat-transfer rate is suddenly reduced and the flow boiling crisis converts the nucleate boiling into the film boiling. Hence, this type of boiling crisis is also called Departure from Nucleate Boiling (DNB).

The consequences of DNB, if the reactor is not immediately shutdown, is potential overheating of the clad and fuel pellet. Therefore the thermal design of a reactor core is limited by the DNB heat flux. The DNB ratio (DNBR) is a measure of the thermal design margin. The DNB ratio is:

$$DNBR = \frac{\text{Predicted DNB Heat Flux}}{\text{Actual Local Heat Flux}} = \frac{q_1''}{q_2''} \qquad (C-1)$$

The minimum DNB ratio (MDNBR) is the minimum of the ratio of q''_1 to q''_2. DNBR must be evaluated in a number of different channels in the core since it is a function of axial and radial power shapes which may be different in different channels. The core average DNBR is not a safety related item as it is not directly related to the MDNBR in the core, which occurs at some elevation in the limiting flow channel. The MDNBR in the limiting flow channel will be downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

C.3 Influence of Power Distribution

The core power distribution which is largely established at beginning of life by fuel enrichment, loading pattern, and core power level is also a function of variables such as control rod worth and position, and fuel depletion throughout lifetime.

Given the local power density q' (KW/ft) at a point x, y, z in a core with N fuel rods and height H, the radial power distribution can be characterized by the radial power peaking factor F:

$$F = \frac{\text{Individual Rod Power}}{\text{Average Rod Power}} = \frac{\int_0^H q'(x_0, y_0, z_0)\, dz}{\frac{1}{N} \sum_{i=1}^{N} \int_0^H q'(x, y, z)\, dz} \quad (C-2)$$

The way in which F is used in the DNB calculation is important. It is obvious that the location of MDNBR will depend on the enthalpy rise to that point and the local power at that point. Basically, the maximum values of the rod integral (integration of power axially for each rod) is used to identify the most likely rod for MDNBR.

An axial power profile is obtained which when normalized to the design value of the maximum radial power peaking factor, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical of distributions found in hot assemblies. In this manner worst case axial profiles can be combined with worst case radial distributions for reference DNB calculations. Actual axial power shape can only be obtained by the excore nuclear detectors.

C.4 Guideline to Identify Hot Rod

From the previous discussions, it is seen that the influence of the power distribution on MDNBR is not only a complicated problem but also is an important one. In order to have an accurate identification of the hot rod, detailed power shapes (3D) are necessary. In this section, a single channel comparison case along with a detailed power distribution were assumed. As shown in Figure C.1, two isolated geometrically identical rods A and B with axial power shapes of $f_A(z)$ and $f_B(z)$ respectively both have the same channel length H, and same channel flow area. The following relations are also assumed to be true:

$$\int_0^H f_A(z)\,dz > \int_0^H f_B(z)\,dz \qquad \text{(C-3)}$$

$$\left|f_A(z)\right|_{H_A} < \left|f_B(z)\right|_{H_B} \qquad \text{(C-4)}$$

Where: $\left|f_A(z)\right|_{H_A}$ and $\left|f_B(z)\right|_{H_B}$ are the local values of the heat flux at the location of MDNBR Channel's A and B respectively.

If only relation (C-3) is available, according to the hot rod identification method recommended in the simplified method (Reference 4, Appendix G), rod A is the hot rod. (This means that the MDNBR of rod A is less than the MDNBR of rod B.) But if in addition to relation (C-3), relation (C-4) is also available, the identification of the hot rod is not easy. This is because that the denominator of the term defined in (C-1) of rod A is less than that of rod B, and the accumulated power input of rod A is larger than that of rod B. Under this complicated situation, the hot rod can only be identified by a quantitative analysis.

Owing to the complicated situation above, accurate identification of the hot rod is hard to achieve. Even so, a qualitative guideline to identify the hot rod under different power distributions was proposed as the followings:

(a) Identify the maximum integral individual rod power shape $\int_0^H f_A(z)\,dz$, also check the axial shape of this power distribution $f_A(z)$.

(b) In case some lower values exist in the shape of $f_A(z)$ at the upper position of the channel, identify and check the second maximum integral power

$\int_0^H f_B(z)\,dz$.

(c) If the rod with integral power $\int_0^H f_B(z)\,dz$ is in the hot assembly, and $|f_B(z)| > |f_A(z)|$ at certain position in the upper portion of the rod, a comparison analysis should be made.

(d) Otherwise, rod with the power shape of $f_A(z)$ is the hot rod.

C.5 Conclusions

If 3D rod power distribution is not available, combination of worst axial and radial power profiles can be assumed for reference DNB calculations. Otherwise, accurated MDNBR can be obtained by inputing available detailed 3D power distribution to the code. For the simplified method, the identification of the hot rod can be achieved according to the guideline described in this Appendix.

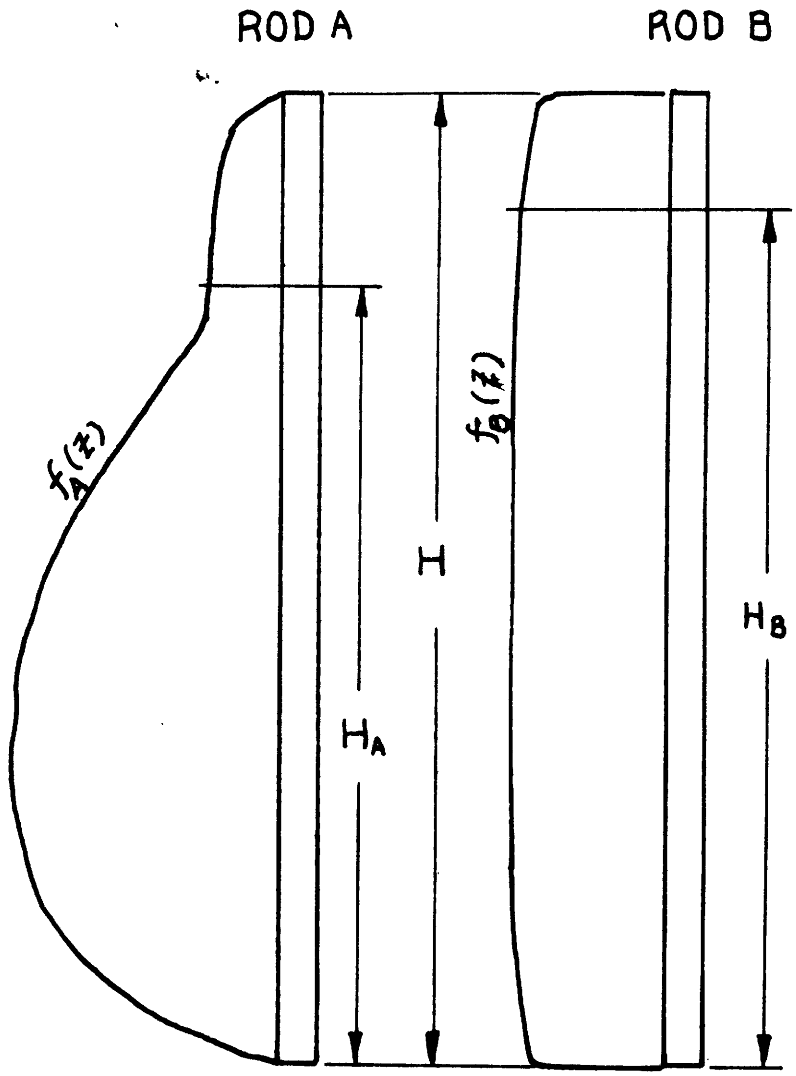

Figure C.1 Axial Power Distribution of Two Example Rods