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Author(s): J. R. Larson

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Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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INTERIM REPORT

NRC Research and Technical Assistance Report

422 166

June 15, 1979

Mr. R. E. Tiller, Director
Reactor Operations and Programs Division
Idaho Operations Office - DOE
Idaho Falls, ID 83401

TRANSMITTAL OF "AN ASSESSMENT CF THE CAPABILITY OF THE EXISTING HEAT TRANSFER DATA BASE AND OF DESIGNATED DATA SOURCES TO PROVIDE DATA OVER THE CALCULATED HEAT TRANSFER VARIABLE RANGES DURING A LOCA BLOWDOWN (A6047)", CAAP-TR-048 - JAD-131-79

Ref: Status Summ ry Report, WRSR, Office of Nuclear Regulatory
Research, Ap 4, 1979

Dear Mr. Tiller:

Attached is the subject Preliminary Assessment Report. This transmittal is in partial fulfillment of the Node associated with the Review of Experimental Programs of the referenced Buff Book.

Very truly yours,

Original Signed by

J. A. Dearien, Manager
Code Assessment and
Applications Program

vjd

Attachment:
As stated

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Central File

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CAAP-TR-048

FOR U.S. NUCLEAR REGULATORY COMMISSION

CODE ASSESSMENT AND APPLICATIONS PROGRAM

AN ASSESSMENT OF THE CAPABILITY
OF THE EXISTING HEAT TRANSFER DATA BASE AND
OF DESIGNATED DATA SOURCES TO PROVIDE DATA OVER THE
CALCULATED HEAT TRANSFER VARIABLE RANGES DURING A LOCA BLOWDOWN

MAY 1979

BY

J. R. LARSON

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

IDAHO OPERATIONS OFFICE UNDER CONTRACT DE-AC07-76IDO1570

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EG&G Idaho, Inc.
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Prepared for the
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Under contract No. EY-76-C-07-1570
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INTERIM REPORT

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ABSTRACT

Data for transition boiling, film boiling and critical heat flux are compared to code calculated heat transfer variable ranges to determine whether the data are sufficient to support the calculations during the blowdown phase of a postulated loss of coolant accident. The data were selected from the Heat Transfer Data Bank. RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1 calculated heat transfer variables of mass flux, pressure, quality, heat flux, and surface temperature are used in the study for the core and steam generator components of PWR and Semiscale systems. The various heat transfer regimes used by the codes for the core and steam generator are determined and a comparison of the usage is presented. Test facilities designated to provide heat transfer data during blowdown are evaluated with respect to their capability to provide data for heat transfer correlation development and testing. Conclusions and recomendations concerning the calculated heat transfer variable ranges and the data are presented.

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SUMMARY

Heat transfer correlations used to calculate the thermal response of heat transfer surfaces, particularily surfaces in a nuclear reactor core during a postulated loss-of-coolant accident, have been used outside the range of the data upon which they have been based. To evaluate the magnitude of the code calculation beyond the data base, existing code calculations made for a typical PWR, and for the Semiscale Mod-1, and Semiscale Mod-3 experimental systems were evaluated. The code cases considered were RELAP4/MOD5 calculations for the PWR; RELAP4/MOD6 calculations for the PWR, Semiscale Mod-1 Tests S-04-5 and S-06-1, and Semiscale Mod-3 Test S-07-1; and TRAC-P1 calculations for the PWR and Semiscale Mod-3 Test S-07-1. All cases were for a large double ended break.

The ranges of the calculated heat transfer variables (coolant mass flux, pressure, quality, and surface heat flux and temperature) were determined for the various heat transfer regimes (subcooled liquid convection, nucleate boiling, critical heat flux, transition boiling, film boiling, single phase vapor convection, and two phase convection). Heat transfer surfaces considered were the core hot spot, core average surface and the steam generator primary and secondary sides. The calculated ranges of the variables were compared to selected data in the Heat Transfer Data Bank (a subset of the NRC/RSR Data Bank maintained at the INEL) for the onset of the critical heat flux, transition boiling and film boiling in the core and film boiling on the primary side of the steam generator. The comparisons were made by plotting the calculated values of the variables against each other (for example heat flux versus mass flux) and overlaying the data values for the same variables. Data considered in the comparison consisted of a few available points for transition boiling obtained in vertical tubes, about 4600 points for film boiling in vertical tubes, about 100 film boiling points for vertical rod bundles and about 5200 rod bundle critical heat flux points.

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For transition boiling the comparison of calculated variable ranges with the data showed that while the available data were in proximity to the calculated ranges there was very little overlapping with the calculated ranges. For film boiling the comparison showed that most of the tube data were taken at conditions far removed from the code calculated variable ranges. Rod bundle data from a single source best approximated the calculated variable ranges. However, large areas of data void exist over the calculated ranges and some combinations of variables (heat flux versus mass flux) were completely void of data. For the critical heat flux the calculated variable values were generally lower in value than the selected data. (An exception was an early critical heat flux calculated by RELAP4/MOD5 with the B&W-2 correlation likely caused by an inverse mass flux dependence of the correlation.)

Several facilities have been designated to provide heat transfer information during the blowdown phase of a postulated loss-of-coolant accident. These facilities, Semiscale Mod-1 and Mod-3 systems, Thermal Hydraulic Test Facility, and Two Loop Test Apparatus were evaluated to determine their capability to provide data during blowdown for development and testing of heat transfer correlations. Of the facilities only Semiscale Mod-3 has a potential for providing useful data during transient testing. Development of assessment procedures to utilize additional core measurements in the Mod-3 system coupled with an uncertainty analysis is necessary before a conclusion about its suitability can be reached.

The study also compared the usage of the various heat transfer regimes by the codes for the core and steam generator. The code heat transfer logic and correlation usage was found to be inconsistent for the steam generator primary or secondary side.

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The conclusion reached was that heat transfer correlations for transition boiling and film boiling are being applied to conditions outside the data range upon which the correlations were developed and tested.

Separate effects tests are recommended to obtain data for the transition and film boiling regimes. Film boiling tests should be conducted at steady state to eliminate uncertainty due to transient operation and the need for the codes to evaluate the local coolant and surface conditions. The tests should be with rod bundles preferably with a 5x5 rod array or larger and with a length of 2 meters. Transition boiling tests could initially be conducted with a facility simpler in concept than for film boiling, perhaps consisting of only a single tube. Again steady state operation would be preferable over transient operation. An uncertainty analysis should be conducted to aid the facility designs to ensure usefulness of the data.

Perhaps useful data can be obtained by changing the mode of operation of the facilities reviewed, that is, from transient blowdown to steady state. Before this conclusion can be drawn additional review of the facilities is necessary to determine their capability to operate in the steady state mode and to determine the range of heat transfer variables that could be achieved. Also an uncertainty analysis for the steady state operating mode would be necessary. This uncertainty analysis is recommended.

The critical heat flux data in the Heat Transfer Data Bank should be updated by means of a literature search and the critical heat flux correlations tested against the data.

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TABLES

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I. INTRODUCTION

Heat transfer correlations used to calculate the thermal response of heat transfer surfaces during the blowdown phase of a postulated loss-of-coolant accident (LOCA) have been used outside the range of the data upon which they have been based. This report compares the calculated heat transfer variable ranges over which the correlations are used during a blowdown to selected data to determine whether the data is sufficient to support the calculations. The report also evaluates the ability of present facilities to provide additional data for development of new correlations or assessment of existing correlations in the variable range where little or no data exist.

Code calculations considered include RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1. The calculations were for a typical PWR and the Semiscale Mod-1 and Mod-3 systems. All calculations were for large double ended breaks. The heat transfer variables of pressure, quality, mass flux, surface heat flux and temperature were examined to determine the range over which they were being used in the various heat transfer regimes specified in the codes. Comparisons of the calculations with the data were made for the core and steam generator primary side.

The Heat Transfer Data Bank, a subset of the NRC/RSR Data Bank[1] maintained at the INEL, is currently limited to selected data describing the heat transfer regimes of transition boiling, film boiling, and critical heat flux. These regimes have the most impact on peak cladding temperature and are probably the most difficult to calculate accurately.

Section II briefly describes the three facilities that have been designated to provide basic heat transfer information, Semiscale, Thermal Hydraulic Test Facility (THTF) and Two Loop Test Apparatus (TLTA). An evaluation of their capability to provide satisfactory information during blowdown is included.

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Section III defines the code calculations considered in this study, summarizes the heat transfer regimes and correlations used in the codes, and illustrates the application of the regimes in the core and steam generator of a PWR. The code noding is tabulated in Appendix A.

Section IV compares the calculated ranges of the variables to the ranges of the data. For the regimes that were not compared to data, the calculated variable ranges are tabulated in Appendix B.

Section V contains the conclusions based on the study and recommendations for obtaining additional data to support correlation application where data presently do not exist.

A brief discussion of the methods used to obtain the heat transfer variables from the codes is presented in Appendix C.

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II. EVALUATION OF EXPERIMENTAL FACILITIES

The following paragraphs briefly describe the facilities that have provided data for a simulated reactor core during blowdown for heat transfer correlation assessment. A brief evaluation of their capability to provide satisfactory data during blowdown is also provided.

1. SEMISCALE

The Semiscale experimental program is part of the investigation of the thermal and hydraulic phenomena accompanying a hypothesized LOCA in a pressurized water cooled nuclear reactor. The two significant systems are considered herein, Mod-1 and Mod-3. One objective of the program is to obtain information necessary to evaluate analytical models used to calculate heat transfer coefficients during blowdown.

1.1 Mod-1 Facility Description

The Semiscale Mod-1 system, shown in Figure 1, is a small size simulation of a four-loop PWR. The system consists of a pressure vessel with core simulator, upper and lower plenums, and downcomer; an intact loop with steam generator, pump, and pressurizer; a broken loop with simulated steam generator and simulated pump; coolant injection accumulators; high and low pressure coolant injection [illegible]s; and a pressure suppression system with a suppression tank, header, and heated steam supply system.

The core simulator contains 40 electrically heated rods. The rod diameter is typical of PWR rods, however, the heated length is only 1.68 m. Ten power steps in each rod provide a slightly bottom skewed

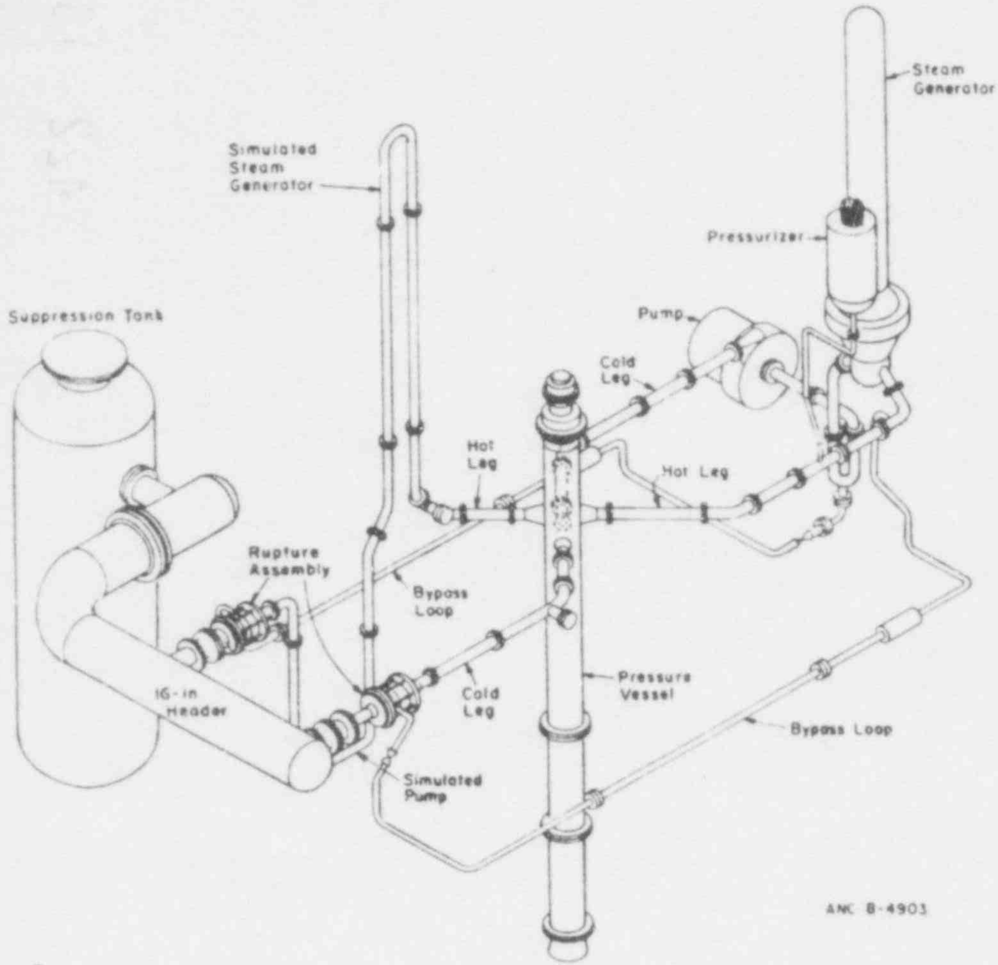

Fig. 1 Semiscale Mod-1 system.

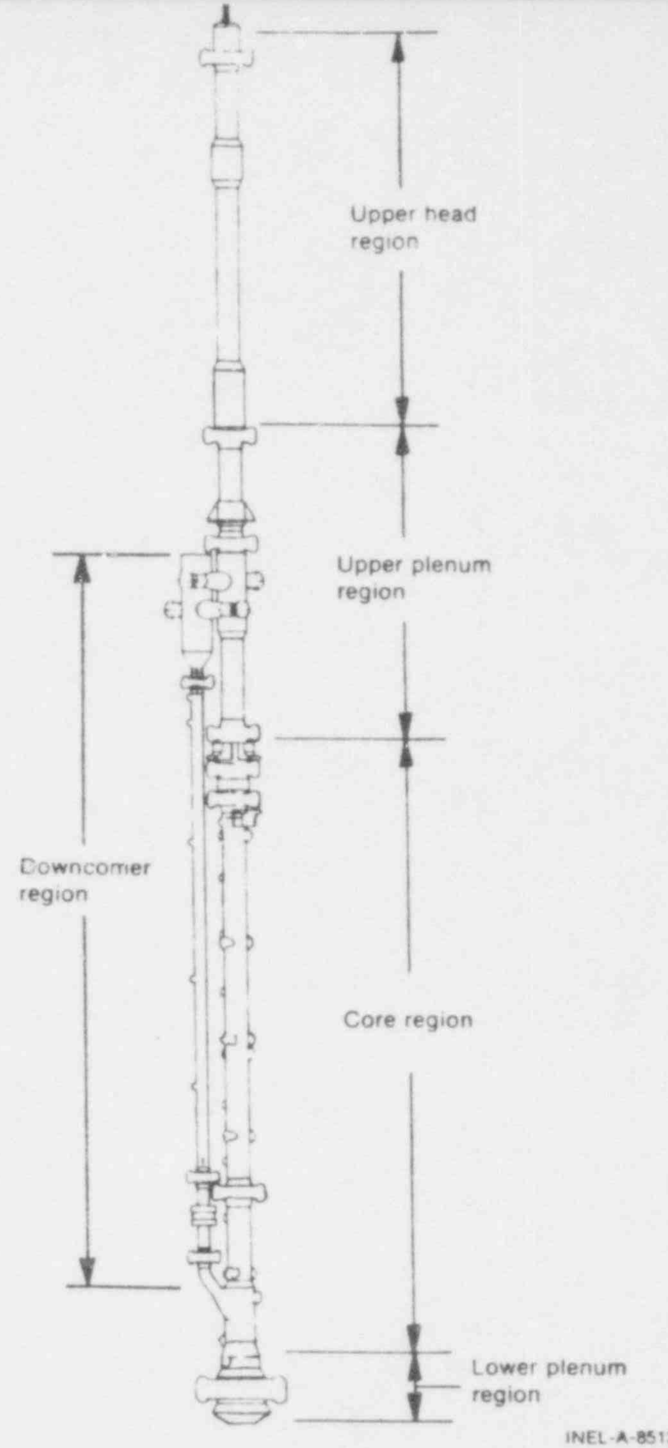

Fig. 2 Semiscale Mod-3 vessel showing major regions.

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axial power profile. A radial power peaking factor can be applied to the center four rods. A more detailed description of the Semiscale Mod-1 system may be found in Reference 2. Coolant measurements pertinent to the evaluation of core heat transfer characteristics are made in spools located in the hot leg piping and a location at the core inlet. Rod cladding temperature measurements are also made.

1.2 Mod-1 Data Evaluation

To evaluate the suitability of using Semiscale Mod-1 data for assessment of post-CHF analytical heat transfer models an extensive analysis[3] was performed with data from Test S-02-9. In summary the analysis was performed using the RELAP4 and INVERT (an inverse heat conduction code) codes along with system measurements. The uncertainty in the measurements was propagated through the codes to determine an uncertainty in the core local fluid conditions. Uncertainties in the analytical methods were not considered.

The effect of an uncertainty in local fluid conditions on calculated cladding temperature was then determined by use of several film boiling heat transfer correlations. The local condition fluid data were considered satisfactory if their uncertainty did not result in a change in calculated peak cladding temperature of 27.8 K for a postulated double ended cold leg break in a PWR. The measurements of inlet flow and the heat flux at the axial peak power location had the most significant effect on the calculated values of the local fluid conditions, particularly enthalpy. Because the nominal mass flow rate was near zero over much of the transient, small errors in absolute value resulted in large relative errors in the local coolant enthalpy.

Data for only 2 of the 15 seconds analyzed were found to be satisfactory. Satisfactory data were obtained only when the mass flow rate was high. The report[3] concluded that additional analysis of other Semiscale Mod-1 test data would not be cost effective.

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1.3 Mod-3 Facility Description

The Mod-3 system is a two loop representation of a four loop PWR. It differs from the Mod-1 system in that the simulated fuel rods are full length, the broken loop is active instead of passive and the downcomer is external to the vessel instead of an annular path inside the vessel. The number of core rods is reduced to 24.

Measurement stations are located at the core exit and in the downcomer. The stations are closer to the core than in the Mod-1 system; however simulation of the PWR requires that the lower plenum be located between the downcomer measurement and the core inlet. Additional measurements are made in the core region of the coolant temperature and density and the rod cladding temperature.

An illustration of the Mod-3 vessel is shown in Figure 2. A detailed description is given in Reference 4.

1.4 Mod-3 Data Evaluation

The instrumentation for the Mod-1 and Mod-3 systems is essentially the same. The uncertainties associated with each measurement are also approximately the same for each system[4,5]. Potential for improvement in determining core local conditions would be expected in the Mod-3 system over the Mod-1 system primarily because of the closer location of the measurement stations to the core and the additional core coolant density measurements providing check points between the measurement stations. Presently a procedure has not been developed to use intermediate check measurements to reduce the uncertainty in the local conditions.

Heat transfer coefficients derived from the Mod-3 data would still be subject to errors inherent with transient operation and code uncertainties. The uncertainty in the flow and local conditions is unknown but might not be significantly different than obtained for the Mod-1 system[3].

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2. THERMAL HYDRAULIC TEST FACILITY (THTF)

The objective of the THTF is to provide heat transfer data for a simulated PWR core during blowdown.

2.1 Facility Description

Figure 3 shows an isometric of the THTF. The system contains a vessel, pump, pressurizer, heat exchanger, pressure-suppression system, and piping. The vessel contains a core simulator, upper and lower plenums, and downcomer. The core simulator in the THTF contains 49 electrically heated rods. The diameter and heated length of the rods are 0.0107 and 3.66 m, respectively. A cosine axial power distribution is simulated by nine power steps in the heater rods. The power profile is radially uniform within the core. A more detailed description of the THTF may be found in Reference 5.

Pertinent fluid condition measurents are made during blowdown in vertical piping spools located in the inlet and outlet to the test vessel, a reentrant type with an annular downcomer. Rod temperature measurements were made at the centerline and between a double outer wall.

2.2 Data Evaluation

Measurement accuracy of the THTF has been compared to that of Semiscale Mod-1[5]. The accuracy of Semiscale measurements was found to be better than for similar measurements in THTF. Efforts to evaluate the core coolant local conditions and characterize the rod coolant heat transfer have been unsucessful[7]. Factors adversely affecting the attempt were determined to be the remote location of the measurement spools with respect to the core, inadequate measurement accuracy, and uncertainty in analytical code methods.

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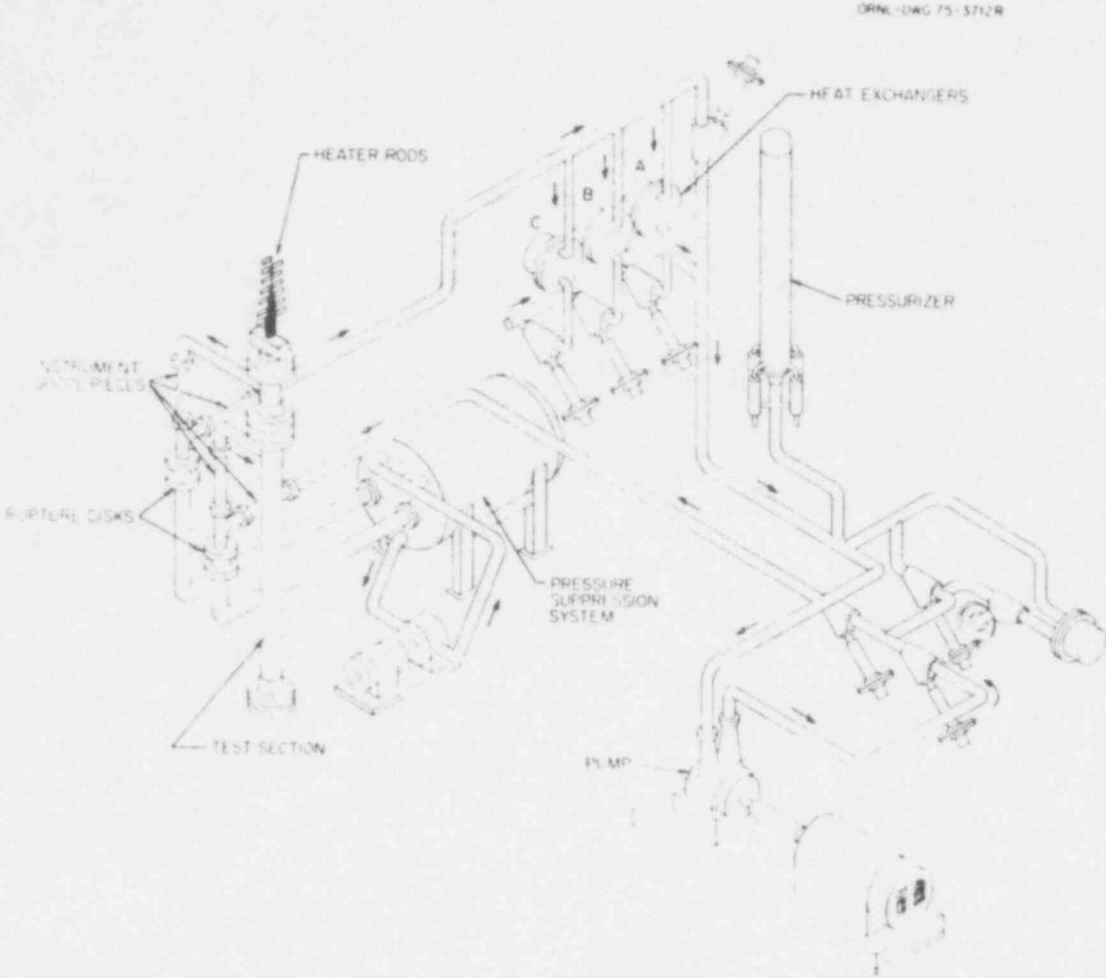

Fig. 3 Thermal-hydraulic test facility.

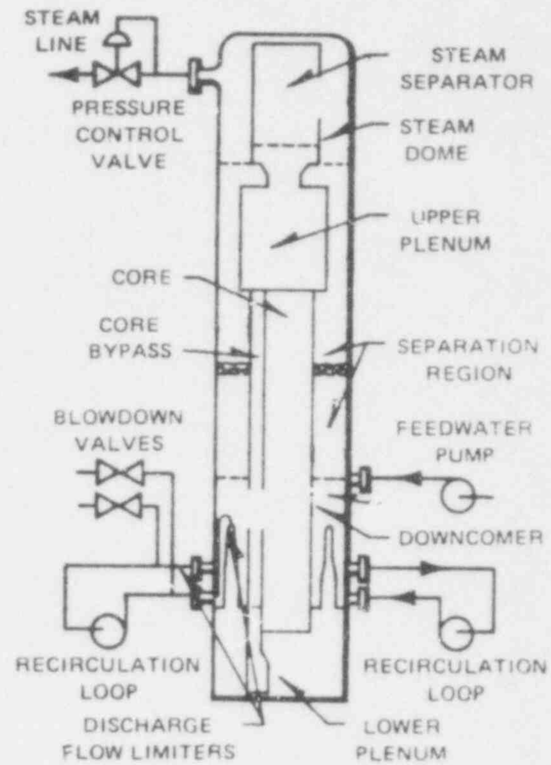

Fig. 4 Two loop test apparatus schematic.

Modifications were proposed to improve measurement accuracy and move the measurement locations closer to the core. These modifications would likely not result in better information than achievable from the Semiscale Mod-1 system.

3. TWO LOOP TEST APPARATUS (TLTA)

The TLTA experimental program is to provide basic information on the performance of a BWR system under LOCA conditions. The program consists of two phases, a completed blowdown heat transfer phase and a phase extending the LOCA transient to include emergency core cooling (BWR BD/ECC). The objective of the first phase (BWR Blowdown Heat Transfer Program) was to obtain transient core heat transfer data[8].

3.1 Facility Description

The TLTA facility is a simulation of a BWR. The system consists of a simulated vessel and internals containing a full size fuel bundle consisting of skin type electrical heaters, a core flow bypass, and external systems such as recirculation loops, jet pumps, and spray cooling systems. The system and fuel bundle have gone through several modifications to represent a BWR/4 (7x7 fuel rod bundle), a BWR/6 (8x8 fuel rod bundle), and lastly adding emergency core cooling spray systems.

The measurement system employed in the TLTA is significantly different from that used in Semiscale and the THTF. The TLTA measurement sysem at the bundle inlet is based primarily on pressure difference devices to give a volumetric average density and orifices to give flow rates. A turbine flowmeter is also stationed at the core inlet. The bundle outlet contains a turbine flowmeter for volumetric

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flow measurement. Measurements are also made of the coolant temperature and pressure and the heater rod temperature. Further description is presented in Reference 9.

3.2 Data Evalution

The measurement system has not been evaluated to the same rigor as that of Semiscale and THTF. This evaluation is thus subjective. The measurement system has the capability for measuring core inlet flow conditions when the lower plenum is subcooled, that is, up to the end of the flow coast down period. After this time the inlet density and enthalpy cannot be determined and an unmeasurable countercurrent flow occurs in the bundle.

The core bypass flow rate is not measured and a density or quality at the upper plenum exit is not determined. Thus insufficent information is available to perform a component evaluation using RELAP4 or TRAC-P1 to obtain the local core coolant conditions.

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III. CODE CALCULATIONS AND HEAT TRANSFER REGIMES

The following section describes the code calculations used in the study and the heat transfer regimes assumed by the code to calculate heat transfer from surfaces to the coolant.

1. CODE CALCULATIONS

Table I lists the particular calculations, that is, the code and system considered in this study. The codes used were RELAP4/MOD5[10],

TABLE I

CALCULATIONS CONSIDERED

System	Code		
	RELAP4/MOD5	RELAP4/MOD6	TRAC-P1
PWR	X[13]	X[14,a]	X[15,16,b]
Semiscale Mod-3 Test S-07-1		X[17]	X[c]
Semiscale Mod-1 Test S-04-5		X[14]	
Semiscale Mod-1 Test S-06-1		X[14]	

a Calculation limited to 15 seconds.

b The calculation used in this report was conducted with a revised model of the one described in Reference 14. Reference 15 presents the results of the blowdown calculation.

c Calculation limited to 18 seconds.

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RELAP4/MOD6[11], and TRAC-P1[12]. Calculations for the cases considered had been previously completed. They all represented large double ended cold leg breaks. Intermediate and small break analyses were not available for consideration. Break size does affect the rate of change in the variables; however, this effect was not considered. Break size may have a more important effect on the relative duration of the various heat transfer regimes.

Three different systems were considered, a large four loop PWR, the Semiscale Mod-1 (Tests S-04-5 and S-06-1) system and the Semiscale Mod-3 (Test S-07-1) system. The PWR geometries for RELAP4/MOD5 and RELAP4/MOD6 were identical with a broken and intact loop modeled. The PWR geometry for the TRAC-P1 run was based largely on Reference 13[15] with three intact and one broken loop modeled.

Rod power was included as a parameter. For the RELAP4 PWR calculations an average and hot channel were evaluated, and for the RELAP4/MOD6 Semiscale calculations a low power test (Test S-06-1) was evaluated. The axial peak power location was used in all cases. Further detail of the volume and heat slab noding is provided in Appendix A.

The RELAP4 calculations were on magnetic tape which did permit computerized data processing at small time intervals. The TRAC-P1 output was printed at intervals approaching one second. These relatively large intervals limited definition of the variable ranges. The variables at onset of CHF also could not be defined for the TRAC-P1 calculations because the output straddled the time of occurrence. Output for all calculations was limited to the normal output. Output of the void fraction would have been helpful in determination of the RELAP4 heat transfer regimes.

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2. HEAT TRANSFER REGIMES

The heat transfer correlations used in the codes are organized and applied in general according to the type of heat transfer mechanism or regime occurring between the surface and fluid. Also the data base is categorized by regime, therefore the comparison of the variable ranges calculated by the codes with the data is accomplished by regime. The regimes are classified in Table II along with the correlations and heat transfer mode identification used by each code by specific regime. The regimes generally apply to any heat transfer surface in the reactor except for the ones denoted for the steam generator which are applied only to the secondary side of the steam generator. Included as a regime is the critical heat flux which really is a criterion for transition between regimes.

Table II reveals that the codes do not use identical correlations for a particular regime. Also a particular correlation may be applied in several regimes. The RELAP4/MOD5 and RELAP4/MOD6 codes require selection of the correlations to be used in transition and film boiling regimes. The TRAC-P1 code permits little choice in application of a correlation for a particular regime. The criteria for changing correlations also differ between the codes. Correlation references and heat transfer logic may be found in the user manuals[10,11,12] and computer listing of the codes.

This study was not concerned with particular correlations or the logic for their selection but was concerned with the ranges of the heat transfer variables used for each regime.

3. CODE USAGE OF HEAT TRANSFER REGIMES

This section illustrates the code usage of the various regimes during blowdown for the core and steam generator.

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TABLE II

CODE BLOWDOWN HEAT TRANSFER IDENTIFICATION AND CORRELATIONS BY REGIME

Regime	RELAP4/MOD5	RELAP4/MOD6	TRAC-P1
Subcooled Liquid Convection	1 (Dittus-Boelter)[a]	1 (Dittus-Boelter)	1 (Laminar, Dittus-Boelter)
Nucleate Boiling	2 (Thom)	2 (Chen, Modified Chen)	2 (Chen)
Transition Boiling	4 (McDonough, Milich, King)	3 (Modified Tong-Young) 4 (Modified Condie-Bengston) 9 (Modified Hsu)	3 (Fit to CHF and Minimum Film Boiling Points)
Film Boiling	5 (Groeneveld) 6 (Modified Bromley) 9 (Dougall-Rohsenow)	5 (Groeneveld) 6 (Condie-Bengston) 9 (Bromley-Pomeranz)	4 (Bromley, Radiation Modified Dittus-Boelter)
Single Phase Vapor-Convection	7 (Free Convection-Radiation) 8 (Dittus Boelter)	7 (Free Convection-Radiation) 8 (Dittus-Boelter) Film Boiling Correlations[b]	6 (McAdams, Dittus-Boelter)

a Number refers to code identification of particular heat transfer mode. Correlations are described and referenced in the code manuals.

b Film boiling correlations are sometimes applied to other regimes.

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TABLE II (Cont'd)

CODE BLOWDOWN HEAT TRANSFER IDENTIFICATION AND CORRELATIONS BY REGIME

Regime	RELAP4/MOD5	RELAP4/MOD6	TRAC-P1
Two Phase Convection	7 (Free Convection-Radiation) 3 (Shrock-Grossman)	7 (Free Convection-Radiation) Film Boiling Correlations[b]	7 (Modified Laminar, Modified Dittus-Boelter)
Steam Generator Natural Convection		11 (Natural Convection)	
Steam Generator Condensation			11 (Chato) 12 (Collier)
Critical Heat Flux	B&W-2, Barnett, Modified Barnett	Tong, Hsu and Beckner, Modified Zuber	Zuber, Biasi

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3.1 Core

Figure 5 shows the heat transfer regime as a function of time after blowdown initiation as calculated by RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1 for the core hot spot in a PWR. The TRAC-P1 calculated regime during the first second is unknown because of the large time intervals. The RELAP4/MOD6 calculation was not continued after 15 seconds.

The figure does illustrate that the same regimes are used over essentially the same time periods. Differences do exist and are primarily caused by small differences in calculated coolant mass flow rates and quality as well as different logic employed to select the regime.

The ranges of the variables (mass flow rate, quality, pressure, surface heat flux and temperature) would be expected to be similar for each regime for all codes. The similarity by regime is confirmed if one examines the variable ranges by regime.

3.2 Steam Generator

The heat transfer regime calculation for the primary and secondary side of the intact loop steam generator is presented in the following sections.

3.2.1 Primary Side. Figure 6 shows the regime calculation for the primary side of the intact loop steam generator as calculated by the three codes for the PWR. As the RELAP4/MOD6 calculation extended only to 15 seconds, the expected regime has been projected on the basis of results for Semiscale Tests S-07-1 and S-04-5.

Obvious differences in the behavior of the three codes are caused in part by the heat transfer logic applied by the codes. The recommended TRAC-P1 logic for the primary side of the steam generator

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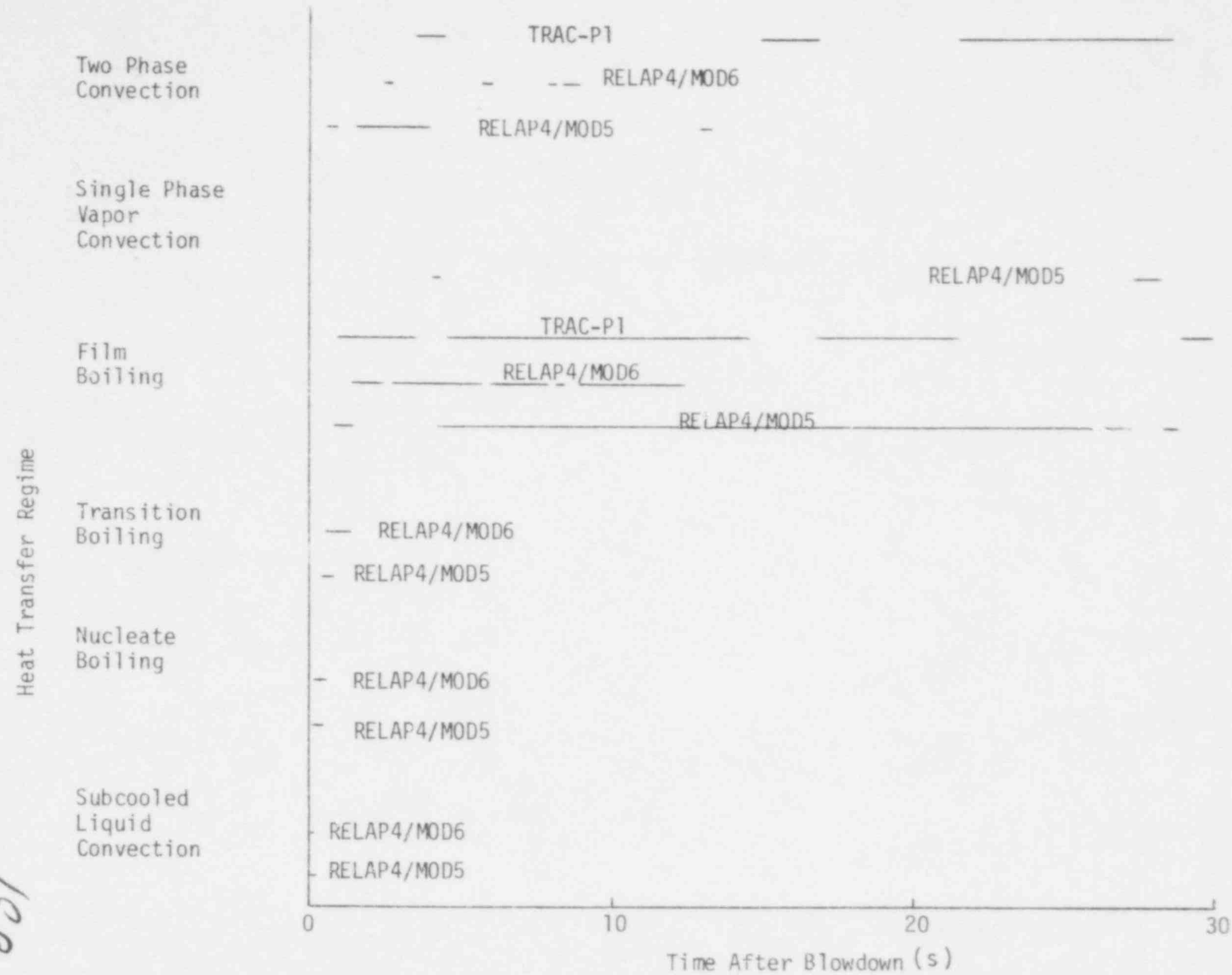

Fig. 5 Heat transfer regime usage during blowdown of PWR core calculated by RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1.

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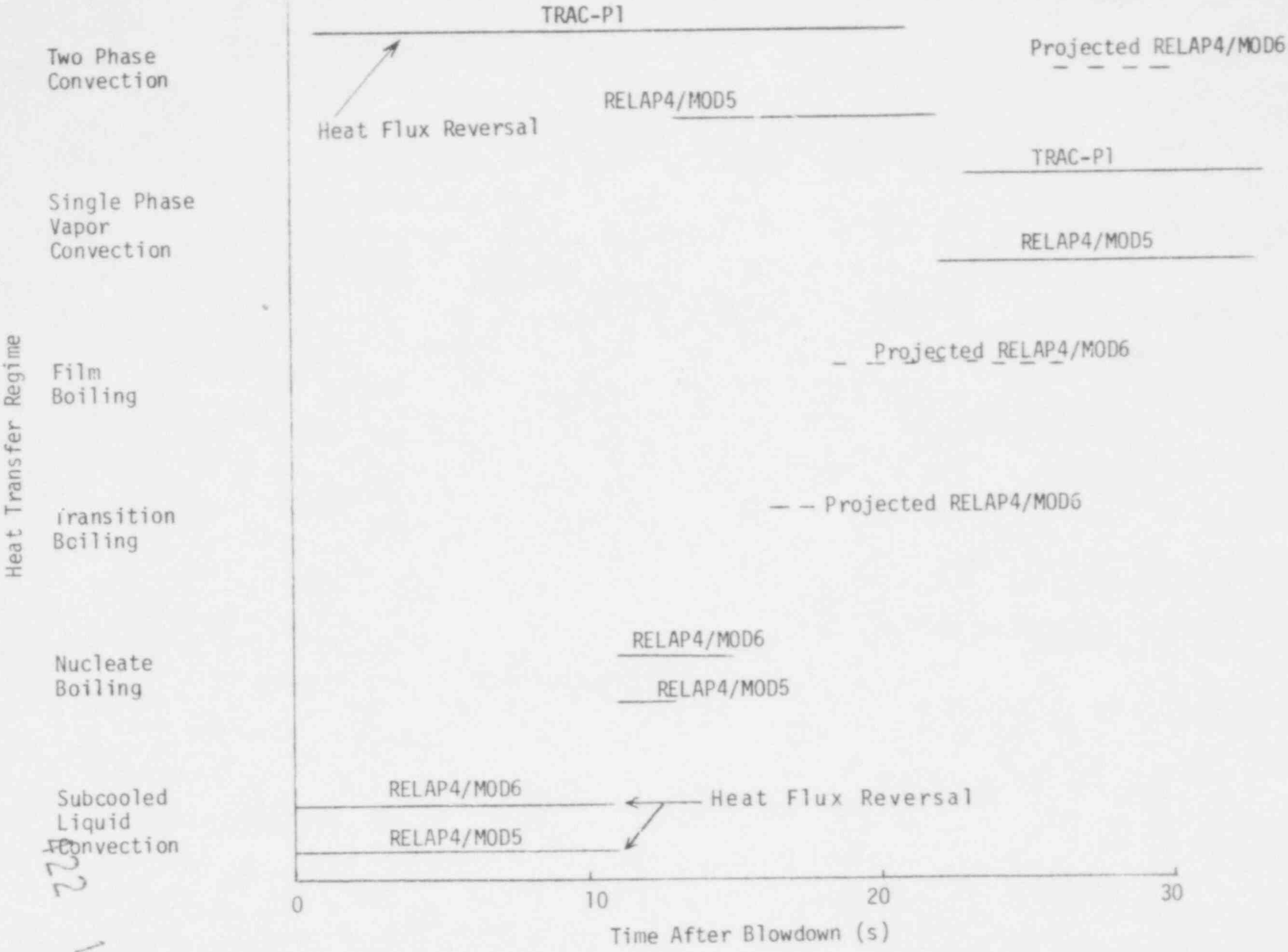

Fig. 6 Heat transfer regime usage during blowdown of PWR intact loop steam generator primary side calculated by RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1.

initially permits only two-phase convection which is calculated to last until dryout occurs resulting in single phase vapor convection. The TRAC-P1 calculated heat flux direction reverses at about 3 to 6 seconds (depending on node location) thereafter providing heat to the primary coolant. Nucleate boiling and CHF is not permitted even after the heat flux changes direction.

The RELAP4 logic results in subcooled convection until about 11 seconds. During this interval the quality of the primary coolant approaches 15% and a very high void fraction (not accounted for in the correlation form applied). At about 11 seconds the RELAP4 calculated heat flux reverses direction. For RELAP4/MOD6 the subcooled convection regime switches to nucleate boiling. A critical heat flux is calculated after about 13 seconds with RELAP4/MOD6 resulting in post-CHF heat transfer regimes. With RELAP4/MOD5 the subcooled convection switches to a two-phase convection regime followed by dryout and eventually single phase vapor convection.

3.2.2 Secondary Side. Figure 7 shows the regime calculation for the secondary side of the intact loop steam generator as calculated by the codes for the PWR. The RELAP4 heat transfer logic calls for nucleate boiling to occur until the heat flux reverses direction. With the heat flux reversal the RELAP4/MOD5 logic results in subcooled liquid convection. The RELAP4/MOD6 logic has an additional option permitting natural convection after the heat flux reversal. The RELAP4/MOD6 code calculation for the PWR did not employ the logic option for natural convection on the secondary side. Consequently the behavior of both RELAP4 calculations for the PWR is the same. The option was used in other RELAP4/MOD6 calculations resulting in the projected curve shown by the dashed line.

The TRAC-P1 regime calculation differs also as from an initial nucleate boiling period, CHF occurs with resulting post-CHF heat transfer until the heat flux reverses direction. Then condensation is assumed to occur.

422 280

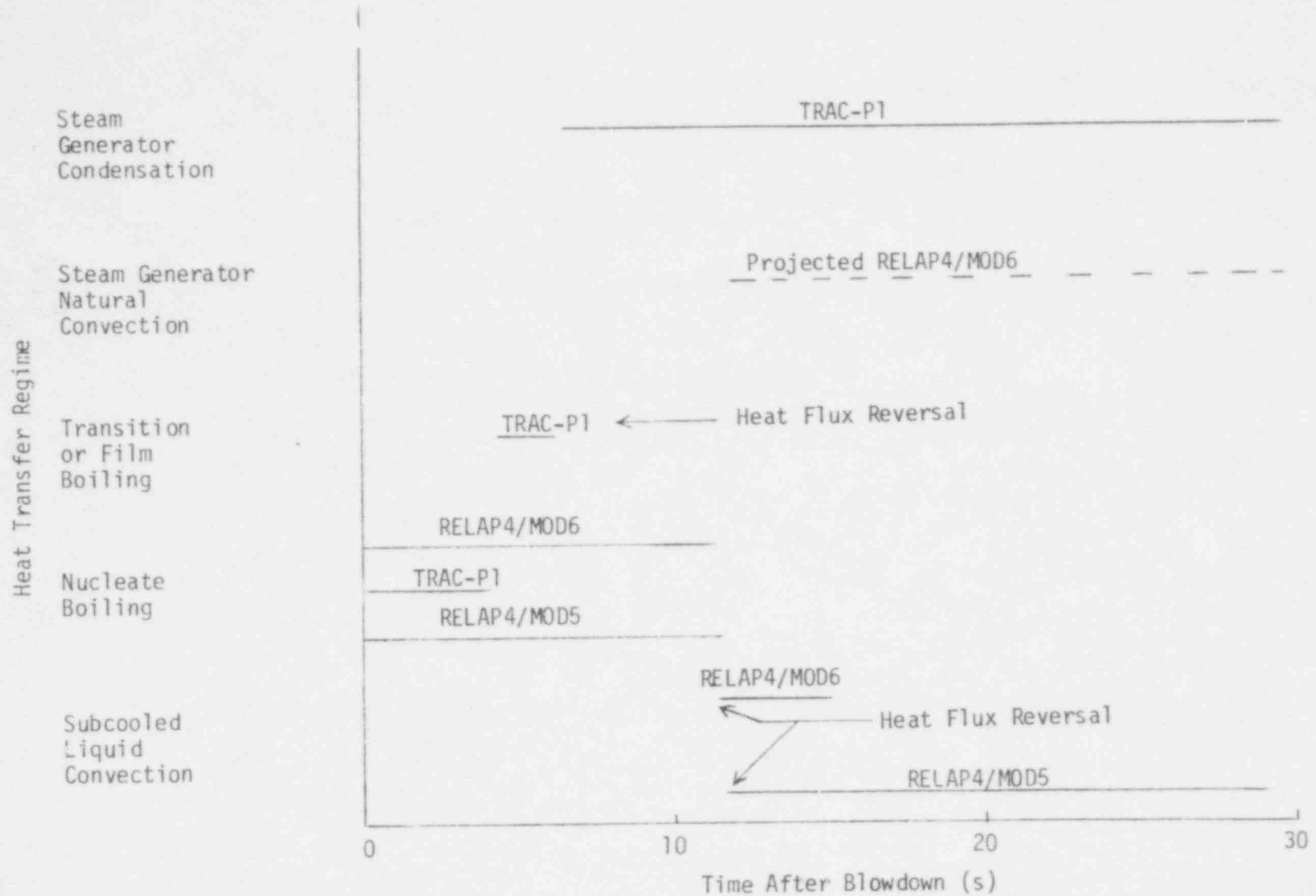

Fig. 7. Heat transfer regime usage during blowdown of PWR intact loop steam generator secondary side calculated by RELAP4/MOD5, RELAP4/MOD6 and TRAC-P1.

IV. COMPARISON OF MEASURED AND CALCULATED PARAMETER RANGES

This section provides a comparison of the ranges of the calculated variables (coolant pressure, mass flux and quality and surface temperature and heat flux) with the measured data. Comparisons are made for the core transition and film boiling regimes, the steam generator primary side film boiling regime, and the core CHF. The comparisons are made in two steps; first the calculated ranges of the variables are plotted one against another and shown with a box encompassing the vast majority of the calculated points; second, the data are shown with the box containing the calculated points superimposed. The plotting technique visually shows the conditions used in the codes and for which the data were obtained. Areas void of data within the calculated ranges are readily apparent.

For the regimes where selected data are not available for comparison, the calculated parameter ranges are tabulated in Appendix A for the core and steam generator primary and secondary sides.

To qualitatively assess the importance of using the correlation outside the data range, the fraction of time the code is outside the data range has been tabulated for each comparison where appropriate.

The data used for comparison with the code calculations are from vertical rod bundles and tube experiments from various investigators. The tube data[18] includes the transition and film boiling regimes although the vast majority of points are for film boiling. The rod bundle data[19,20,21,22] covers only the film boiling regime. The data were complied and screended by the Heat Transfer Data Bank Manager. The original tube data sources are in Reference 18. The CHF data encompassed 151 separate sources[23] with additional selected data for a LOFT type 25 rod assembly tested at Columbia University.

422 208

1. CORE TRANSITION BOILING COMPARISON

The calculated ranges of the variables are shown beginning with Figure 8 for heat flux versus wall temperature. The relative density of the points indicate the variable values that were most often used during the calculation. Calculations for a particular RELAP4 case were taken at 40 msec intervals and generally trace a path on the plots as the variables change. Case labels are placed on the traces to aid in identification where possible. The time intervals for which data were available for TRAC-P1 calculation were too large and straddled the variable values except for one point for the Semiscale Test S-07-1 case.

The usage of the transition boiling regime was primarily confined to the cases calculated by RELAP4/MOD6. The calculated duration of the regime was also quite short, that is, 0.2 sec for the PWR RELAP4/MOD5 calculation. The low power case (Semiscale Test S-06-1) conducted for the Semiscale Mod-1 system remained in the transition boiling regime the longest, about 6 seconds.

A box is drawn to encompass the calculated variable ranges in Figure 8. The box enclosing the calculated points is redrawn on Figure 9 below which compares the calculated variable ranges with the transition boiling data.

As can be seen, the data lie along the lower temperature range of the code calculations. The data are very limited and are from an experiment conducted inside a single vertical tube[24]. All data simultaneously fit within the variable ranges specified for the coordinate boundaries of the plots, i.e., heat flux from

422 203

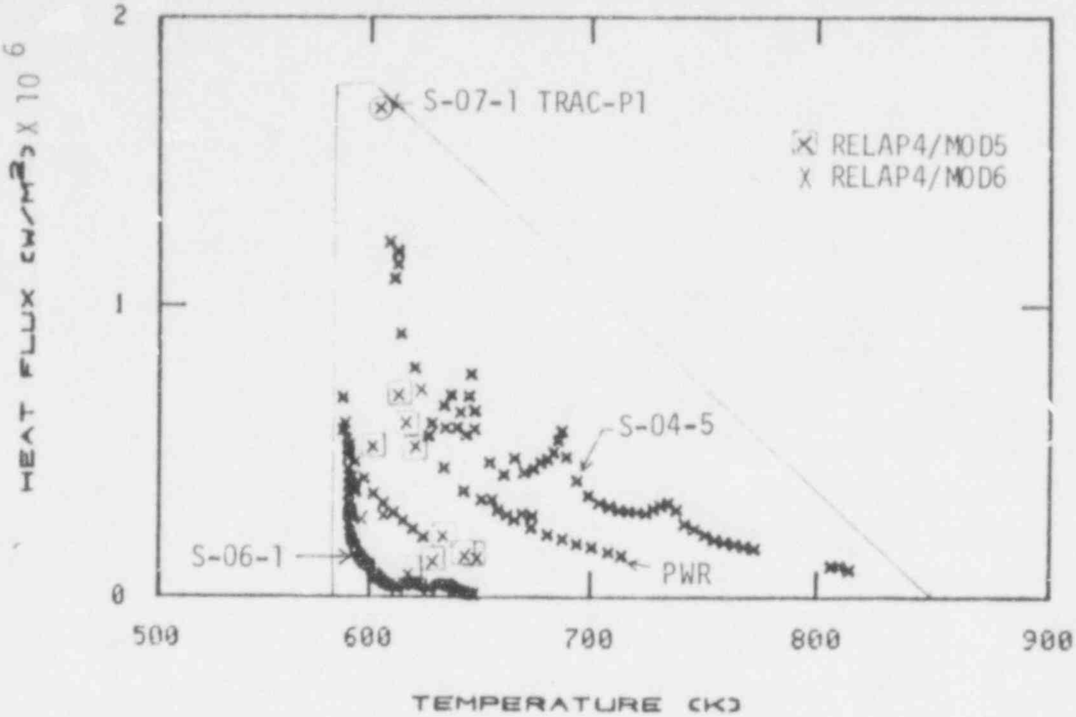

Fig. 8 Heat flux vs. surface temperature as calculated by the codes for the core for the transition boiling regime.

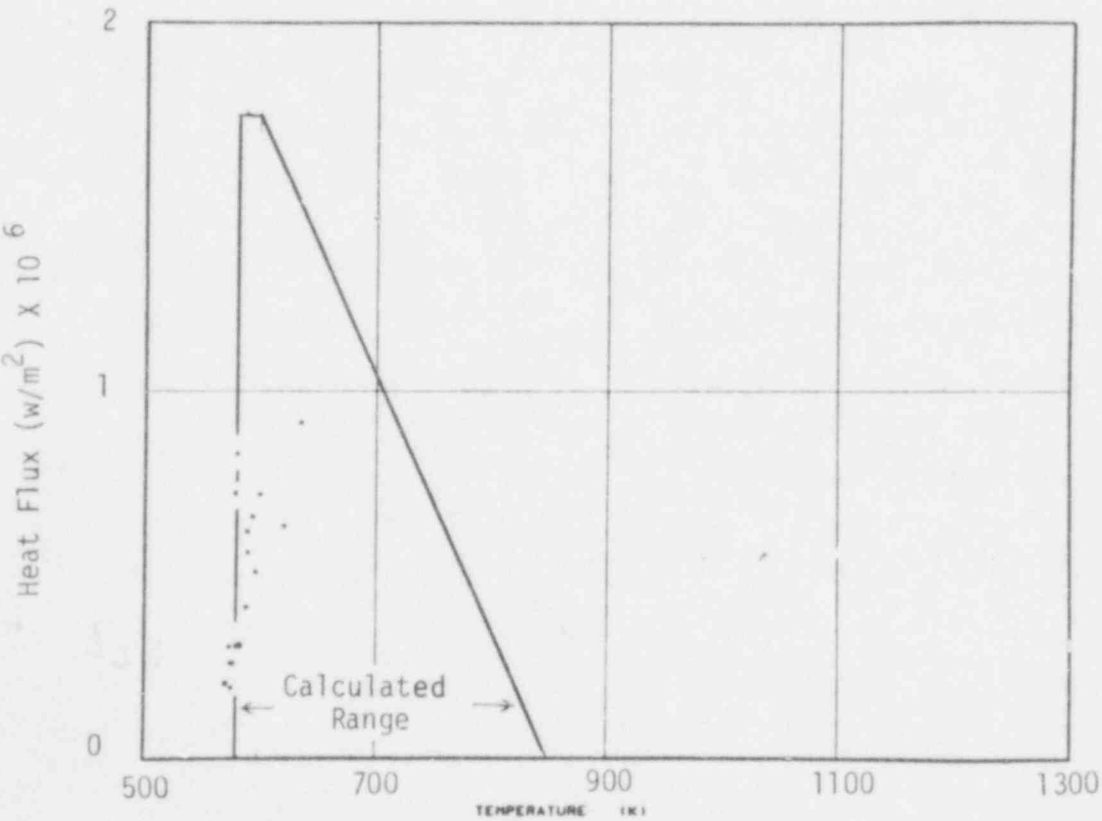

Fig. 9 Comparison of calculated variable ranges of heat flux and pressure with selected tube data for transition boiling.

422 204

0 to 2×10^6 W/m^3 and surface temperature from 500 to 1300K, pressure from 0 to 15 MPa, quality from 0 to 1.2, and mass flux from 0 to 2500 $kg/s\text{-}m^2$.

The variable magnitudes chosen for the data are abritrary but in all cases exceed the calculated variable ranges. The transition boiling data have been questioned as to whether they actually represent the transition boiling regime[25]. If the data are later found not to be for transition boiling the results and conclusions based on the comparison presented would not be changed.

Figure 10 shows the calculated variable ranges for the plot of heat flux versus pressure. The trace formed by the calculation is obvious for some cases and is identified. The box drawn in the figure encloses all the calculations. The box is redrawn in Figure 11 where again the calculated ranges are compared with the data. None of the data fall within the calculated ranges.

The process is repeated again in Figures 12 and 13 for the quality versus temperature parameters. The traces formed by the calculations are readily distinguishable in Figure 12. The comparison of the calculated ranges with the data in Figure 13 again indicates little overlapping of the data and calculated ranges.

Figures 14 and 15 show the comparison of the calculated ranges of heat flux and mass flux with the data. The traces formed by the calculations are not as distinct as for the previous parameter combinations. No overlapping of the data with the calculated parameter ranges is shown in Figure 15.

422 205

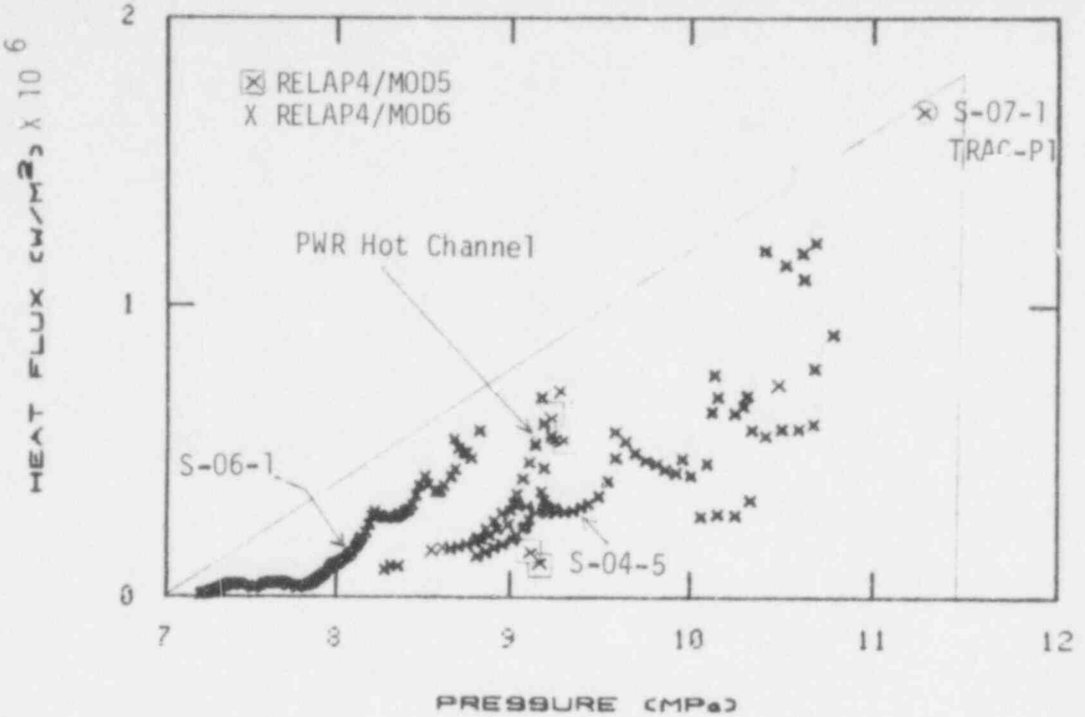

Fig. 10 Heat flux vs. pressure as calculated by the codes for the core for the transition boiling regime.

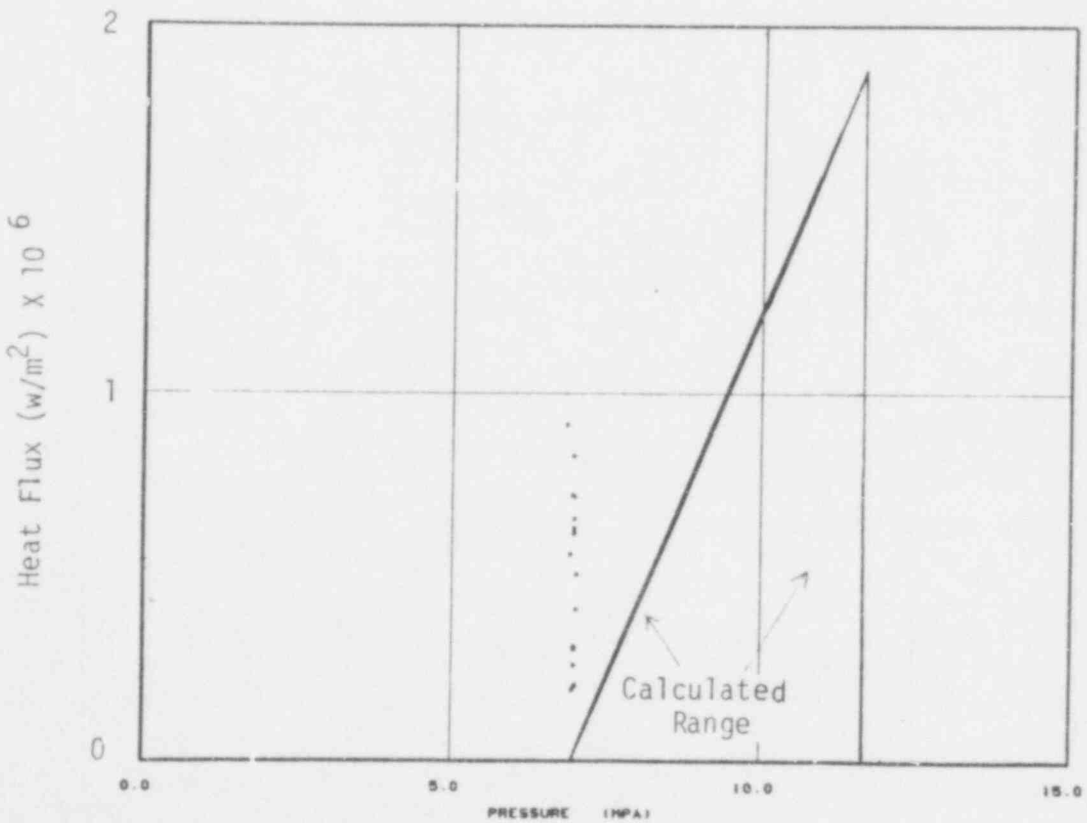

Fig. 11 Comparison of calculated variable ranges of heat flux and pressure with selected tube data for transition boiling.

422 206

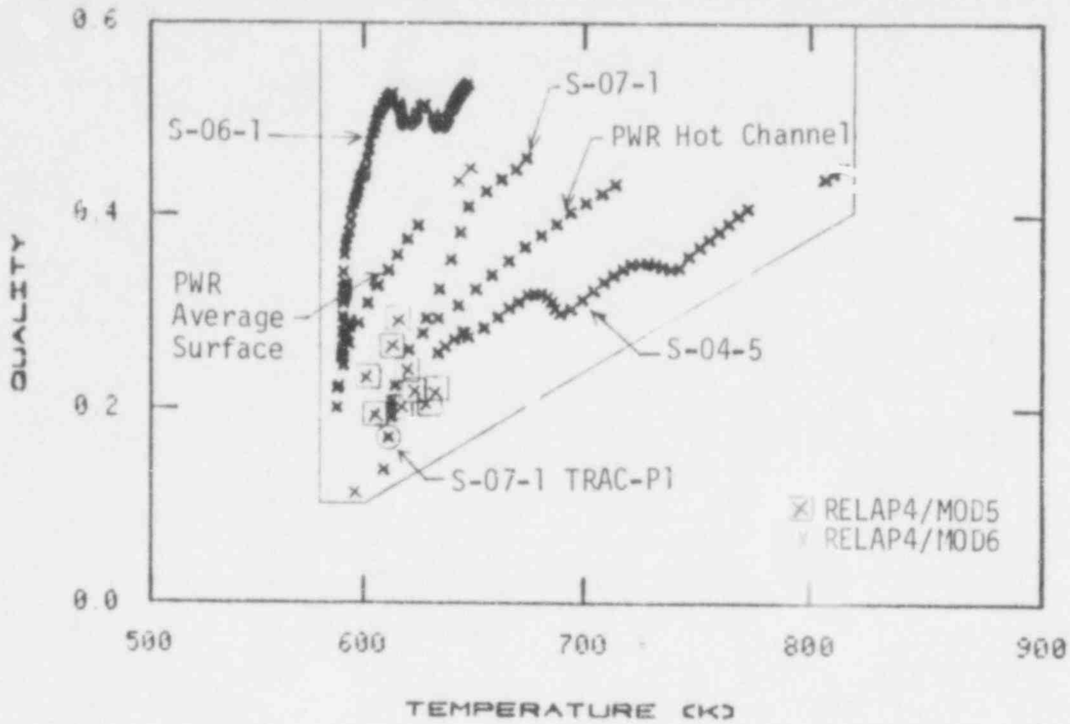

Fig. 12 Quality vs. temperature as calculated by the codes for the core for the transition boiling regime.

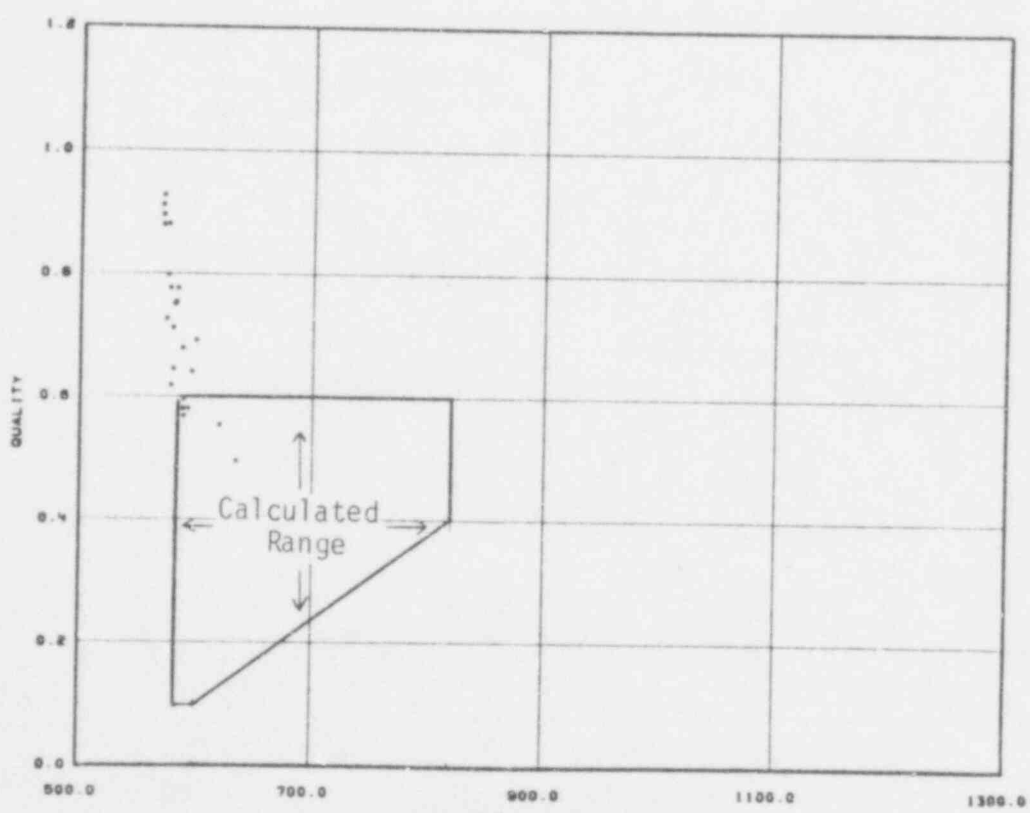

Fig. 13 Comparison of calculated variable ranges of quality and pressure with selected tube data for transition boiling regime.

422 207

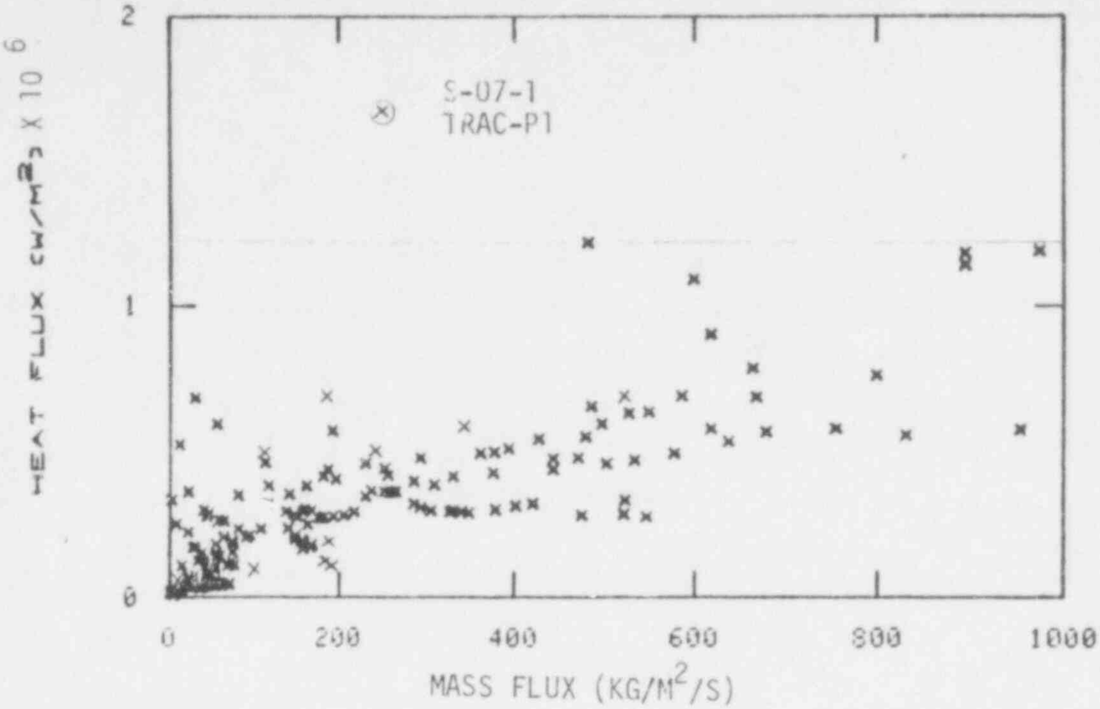

Fig. 14 Heat flux vs. mass flux as calculated by the codes for the transition boiling regime.

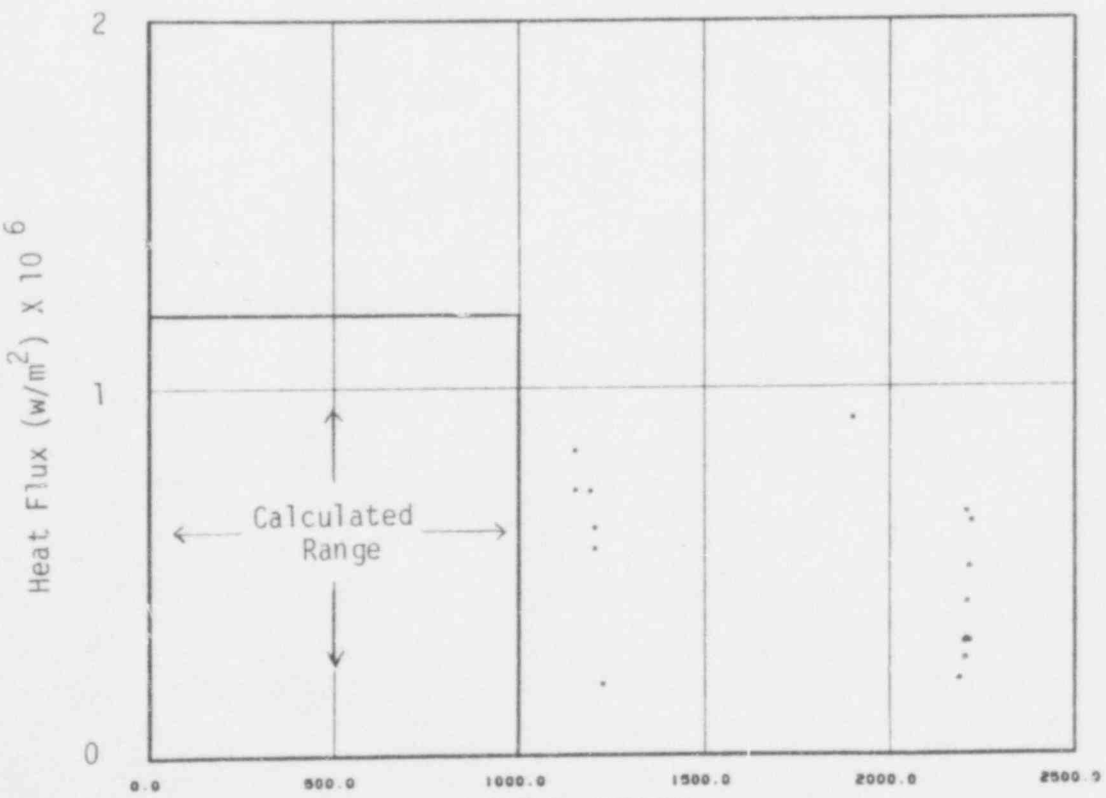

Fig. 15 Comparison of calculated variables ranges of heat flux and mass flux with selected tube data for transition boiling.

422 208

7. CORE FILM BOILING COMPARISON

The calculated ranges of the variables are shown and then compared to the data in Figures 16 through 27 by pairs for heat flux versus surface temperature, heat flux versus pressure, quality versus temperature, heat flux versus mass flux, pressure versus mass flux, and quality versus pressure, respectively. Boxes are drawn to enclose the calculated points. Again the relative density of the points indicates the frequency of the variable values over which the codes perform their calculation. The boxes enclosing the calculated ranges are redrawn on accompanying figures where the data are also plotted.

Figure 16 shows the calculated parameter ranges for the surface heat flux and surface temperature. The RELAP4/MOD6 calculation for Semiscale Test S-07-1 results in a trace apart from the bulk of the calculations and is identified; this calculation also has the highest values of heat flux. The remainder of the calculations are grouped together and no attempt was made to specifically identify them. The TRAC-P1 calculations covered the temperature range with heat fluxes less than 2.0×10^5 W/m^2.

The data from inside vertical tubes[18] and rod bundle data of Combustion Engineering[19,20] and McPherson[21,22] are plotted on Figure 17 with the box enclosing the calculated variable ranges. Different symbols are used for the tube and rod bundle data. About 1070 tube data points of a total of 4600 points and 80 rod bundle data points of a total of 100 in the data bank are plotted. The rod bundle points are nearly all from Combustion Engineering[19,20,a].

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a The basic geometry of the bundle was either 25 rods or 21 rods with a guide tube. The length was 2.1 m.

422 209

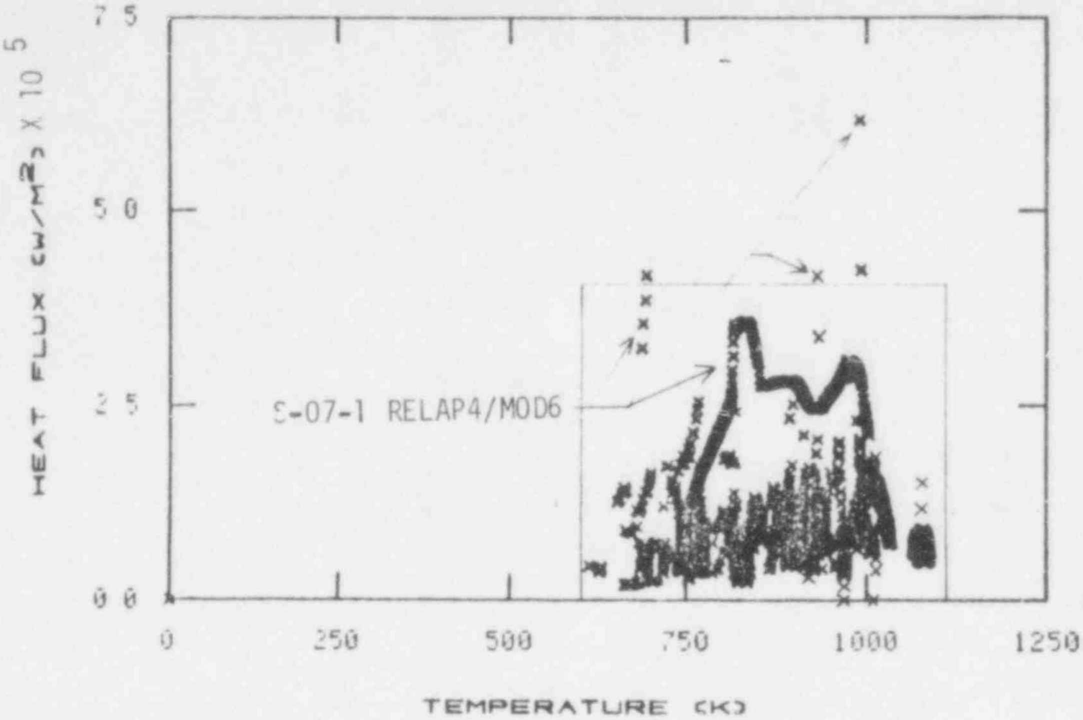

Fig. 16 Heat flux vs. temperature as calculated by the codes for the core for the film boiling regime.

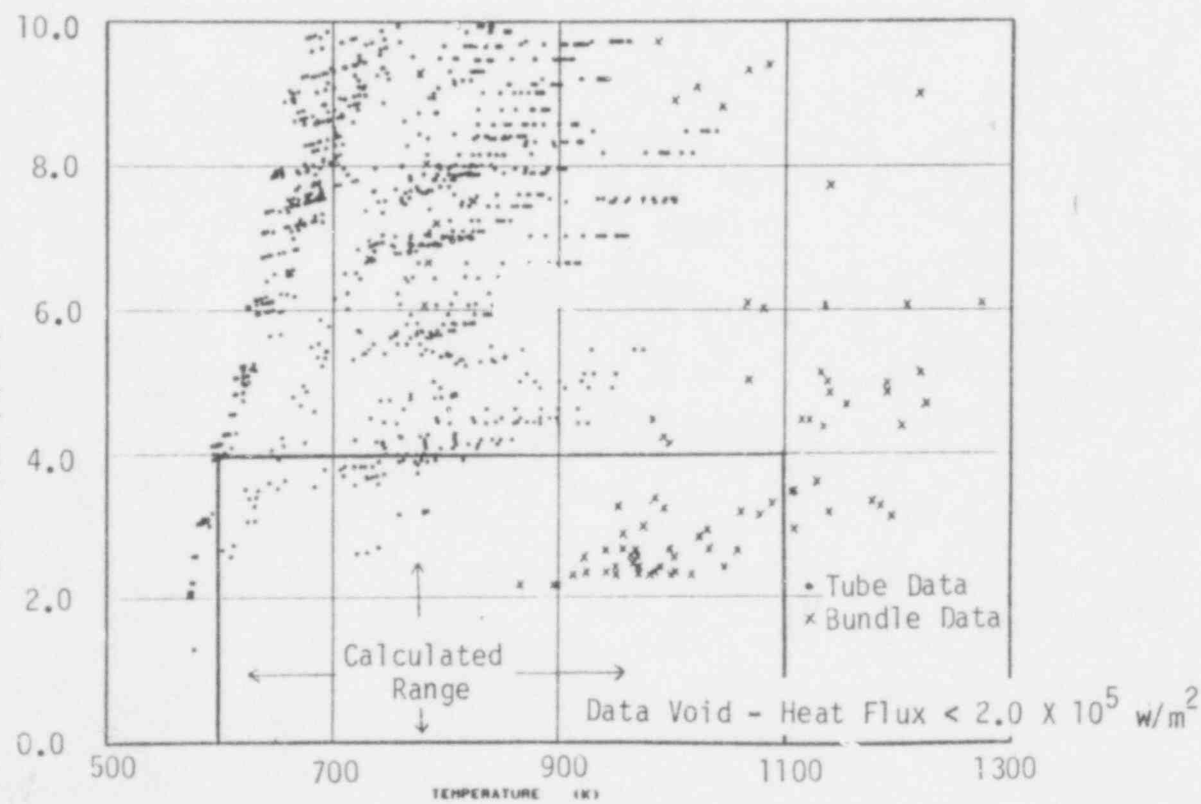

Fig. 17 Comparison of the calculated variable ranges of heat flux and temperature with selected data for film boiling.

422 200

The sorting criteria for the data were as follows: heat flux from 0 to 10^6 W/m^2, temperature from 500 to 1300 K, pressure from 0 to 15 MPa, quality from 0 to 1.2 and mass flux 0 to 1000 Kg/s-m^2. The reduction in the number of tube data points particularly indicates that much of the tube data were obtained at conditious that did not represent conditions calculated for the core.

Rod bundle and tube data have both been used in the development and assessment of correlations for application to nuclear cores during film boiling heat transfer. The assumption has been made that the physical processes were similar for both geometries and thus the tube data would apply to the rod bundle geometry. Additional rod bundle data and further analysis are needed to evaluate this assumption.

As can be seen on Figure 17 the tube and rod bundle data cover the calculated temperature range but leave a void in the heat flux parameter range below a value of 2.0 x 10^5 W/m^2. The importance of the data void region is quantified by the information presented in Table III. The table indicates the total time a particular code case was in the film boiling regime and also the elapsed time the code case was in an area void of data. The timing for the RELAP4 cases is quite accurate as calculations were available at 40 msec intervals. For TRAC-P1 the timing is approximate.

Figure 18 shows the calculated variable ranges for heat flux and pressure. All of the cases tend to cover the entire pressure range. All cases except the RELAP4/MOD6 case for Semiscale Test S-07-1 are limited to heat fluxes less than 2.0 x 10^5 W/m^2.

The calculated variable ranges are compared to the data in Figure 19. The rod bundle data cover the pressure range but not the heat flux range.

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TABLE III

ELAPSED TIME THE CODE CALCULATIONS ARE IN DATA VOID DURING FILM BOILING

Case	Time In Film Boiling (s)	Time Heat Flux < 2.0×10^5 W/m^2	Time Quality <0.4 (s)	Time In Mass Flux, Heat Flux Data Void (s)
PWR RELAP4/MOD5	22.6	22.3	8.6	22.6
PWR TRAC-P1	20.0	20.0	6.0	16.0
S-07-1 RELAP4/MOD6[a]	27.3	15.4	0	27.1
S-07-1 TRAC-P1[a,b]	10.0	10.0	8.0	10.0

a The number prior to the code name refers to the particular Semiscale test.

b Only 18 seconds available.

422 212

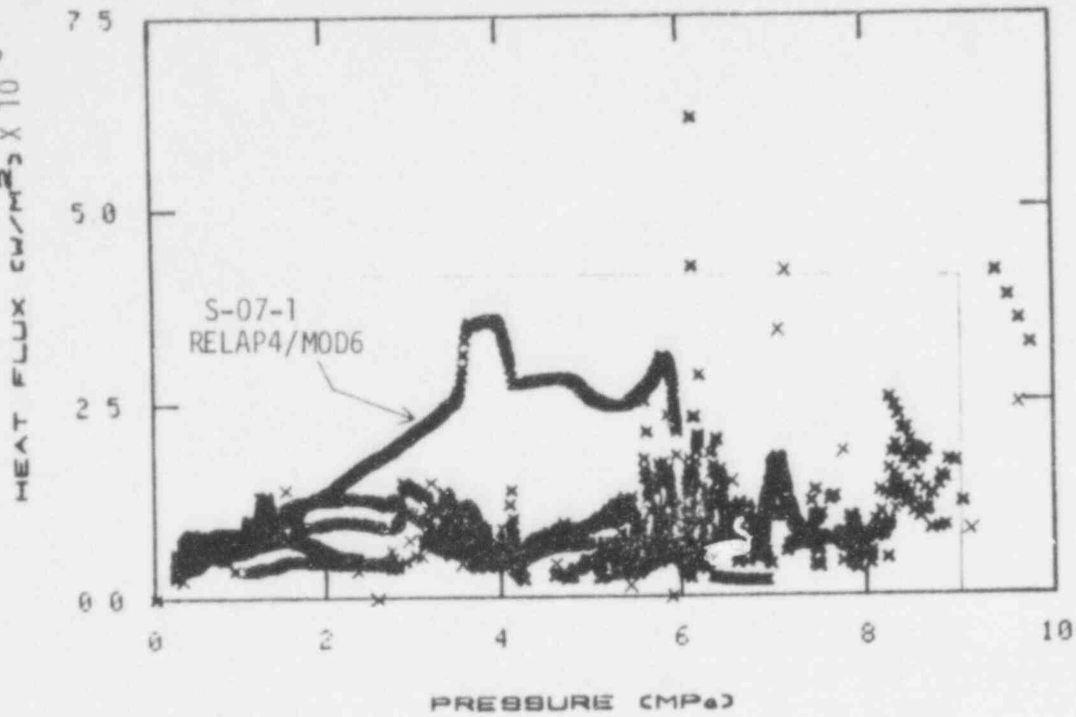

Fig. 18 Heat flux vs. pressure as calculated by the codes for the core for the film boiling regime.

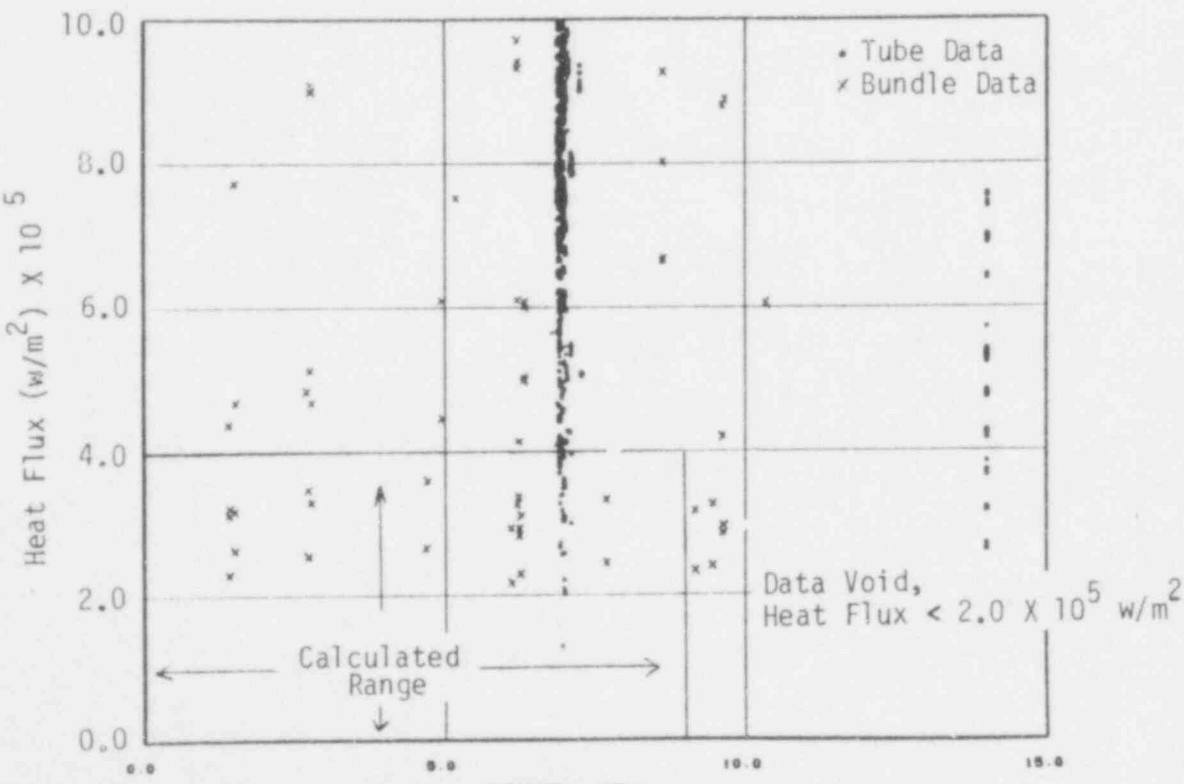

Fig. 19 Comparison of the calculated variable ranges of heat flux and pressure with selected data.

422 213

Figure 20 shows the calculated variable ranges for quality and temperature. The RELAP4/MOD6 film boiling calculations occur only with a quality value above 0.4. The RELAP4/MOD5 and TRAC-P1 calculations cover the entire quality range. Some but not all of the TRAC-P1 points are specifically identified on the figure.

The calculated quality is a flow quality (ratio of the mass flow rate of vapor to the total mass flow rate) and is limited to the range between 0 to 1. The experimental quality is a thermodynamic quality and can range from negative values to values exceeding unity. If the phases are in thermal equilibrium, the flow and thermodynamic qualities will be equal over the range 0 to 1 for cocurrent flow. The deviations from phase equilibrium are sufficiently small that the calculated and measured qualities are approximately equal and may be compared for the purpose of this task. Further discussion of quality can be found in Appendix C.

The calculated variable range for quality and temperature are shown with the data in Figure 21. The tube data are seen to cover the calculated quality temperature ranges better than for the calculated ranges of heat flux and pressure. The tube and rod bundle data cover the region with a quality higher than 0.4. Below this quality a data void exists. Table III indicates that the calculations of the Semiscale Test S-07-1 RELAP4/MOD6 case (and other RELAP4/MOD6 cases) do not fall in the data void. However, for the RELAP4/MOD5 and TRAC-P1 cases the calculations fall in the data void region for at least one third of the total time in the film boiling regime.

Figure 22 shows the calculated variable ranges for heat flux and mass flux. Except for the Semiscale Test S-07-1 RELAP4/MOD6 case the calculations are largely grouped within a mass flux of about 200 kg/s-m^2 and a heat flux of 2.0 x 10^5 W/m^2. The calculated ranges are compared to the data in Figure 23. The tube data lie completely

422 219

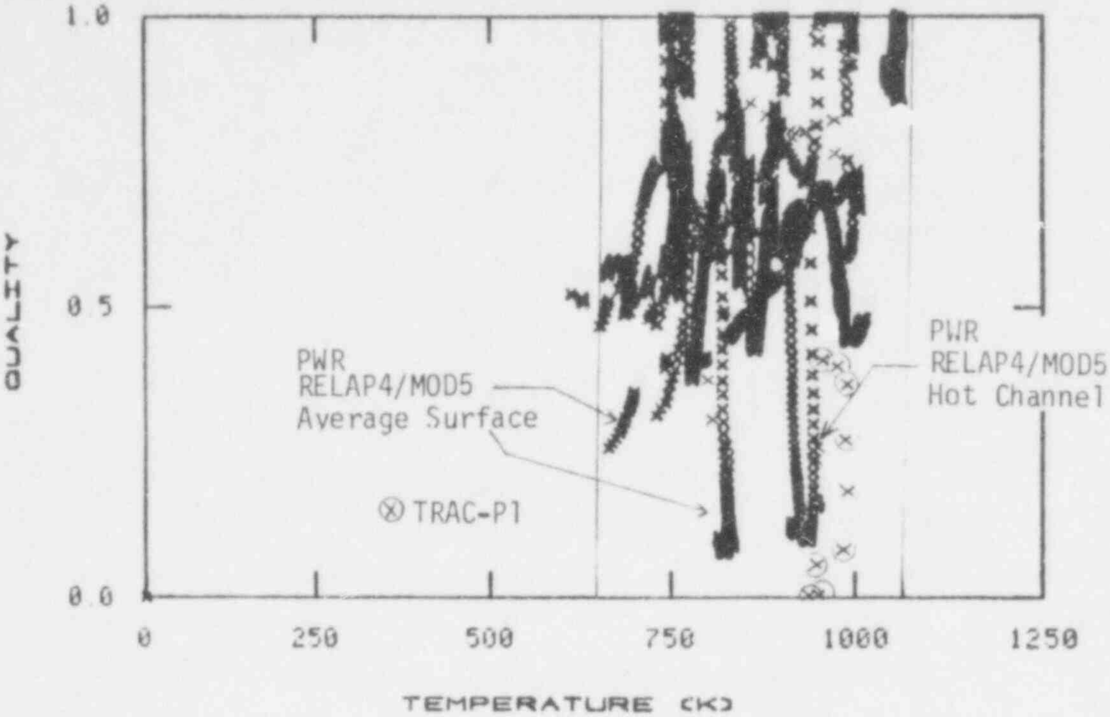

Fig. 20 Quality vs. temperature as calculated by the codes for the core for the film boiling regime.

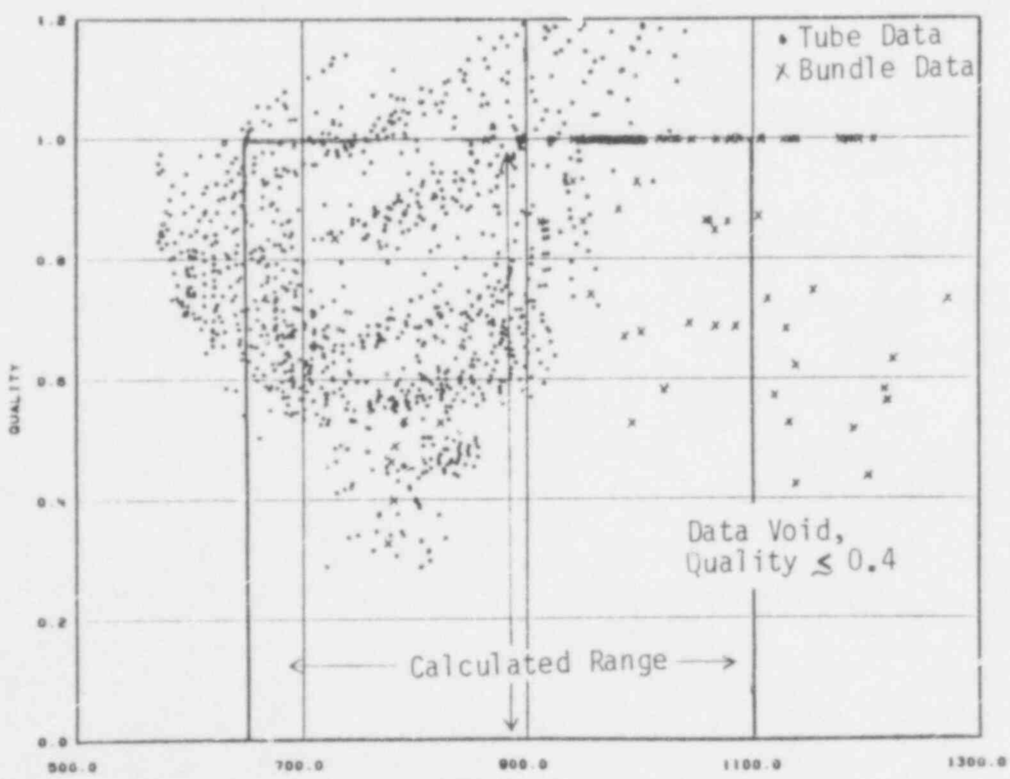

Fig. 21 Comparison of the calculated variable ranges of quality and temperature with selected data.

422 215

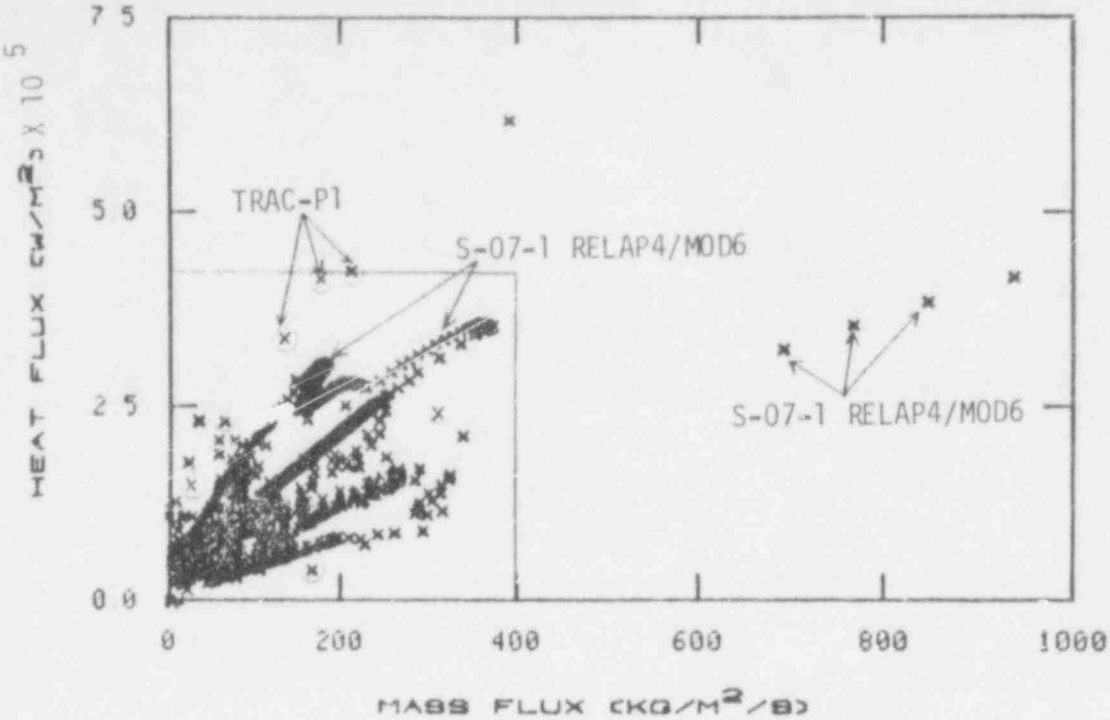

Fig. 22 Heat flux vs. mass flux as calculated by the codes for the core for the film boiling regime.

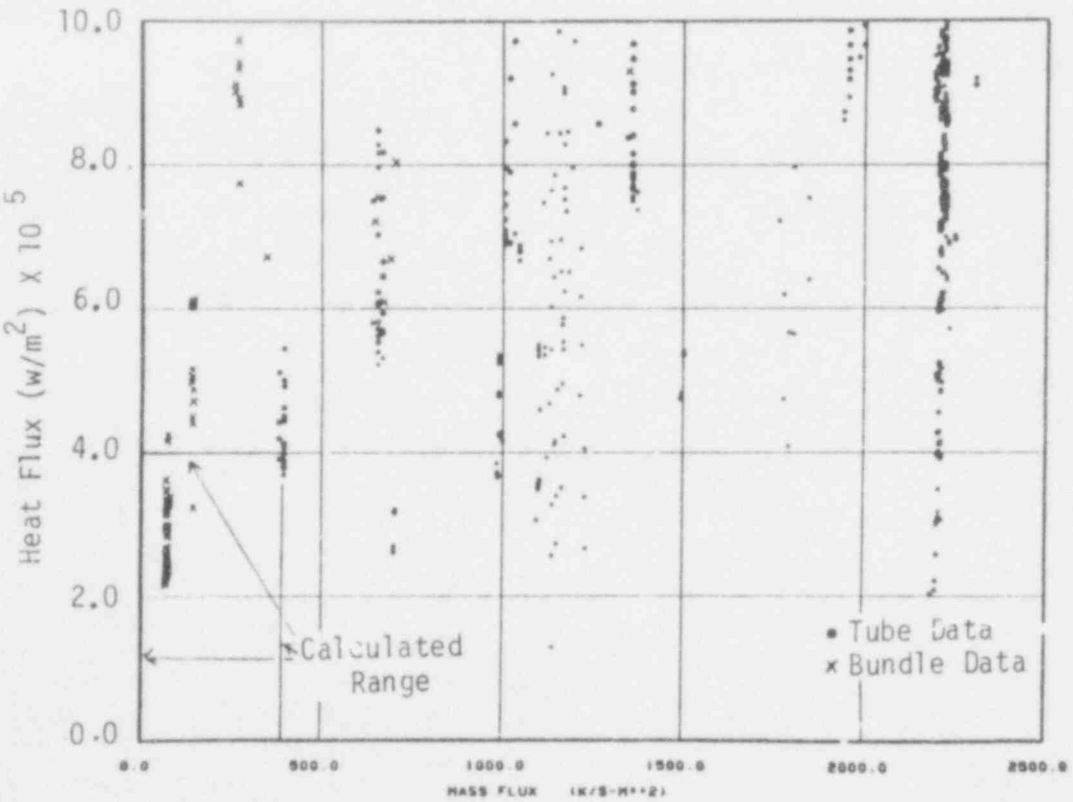

Fig. 23 Comparison of the calculated variable ranges of heat flux and mass flux with selected data for film boiling.

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without the box enclosing the calculations. The rod bundle data within the box cover a corner of the box that is void of the calculations. Thus, essentially no data overlap the calculations.

Figure 24 shows the calculated variable ranges of pressure and mass flux. A few TRAC-P1 points are indicated; however, the bulk of the TRAC-P1 film boiling points are at mass fluxes less than 100 $kg/s\text{-}m^2$ and can not be distinguished. The comparison with the data shown in Figure 25 indicates the rod bundle data covers the calculated ranges although with only about 20 points.

Figure 26 shows the calculated variable ranges of quality and pressure. The RELAP4/MOD5 and TRAC-P1 calculation ranges cover the area while the RELAP4/MOD6 calculations are at qualities above 0.4. The comparison with data is shown in Figure 27. Again the rod bundle data cover the area except for qualities less than 0.4. The tube data falling within the calculational limits were taken at essentially one pressure value.

In summary the code calculations are not identical in the variable values input to the heat transfer correlations but the values are within a finite range. The film boiling tube data do not begin to cover the variable ranges used by the codes and correlations based on the tube data are being extrapolated in their usage. The calculated variable ranges are best but far from completely covered by a single source of rod bundle data consisting of a handful of data points. Essentially no data exist at a surface heat flux less than 2.0×10^5 W/m^2, a coolant quality less than 0.4, or that simultaneously fall within the calculated mass flux and heat flux range of variable values.

217

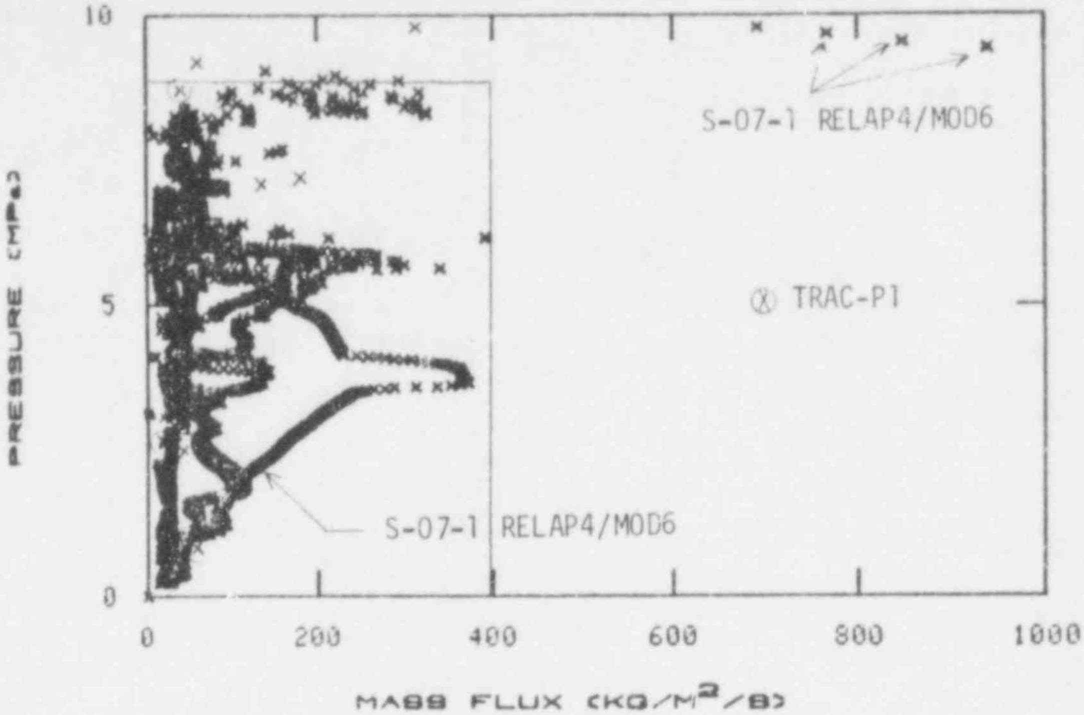

Fig. 24 Pressure vs. mass flux as calculated by the codes for the core for the film boiling regime.

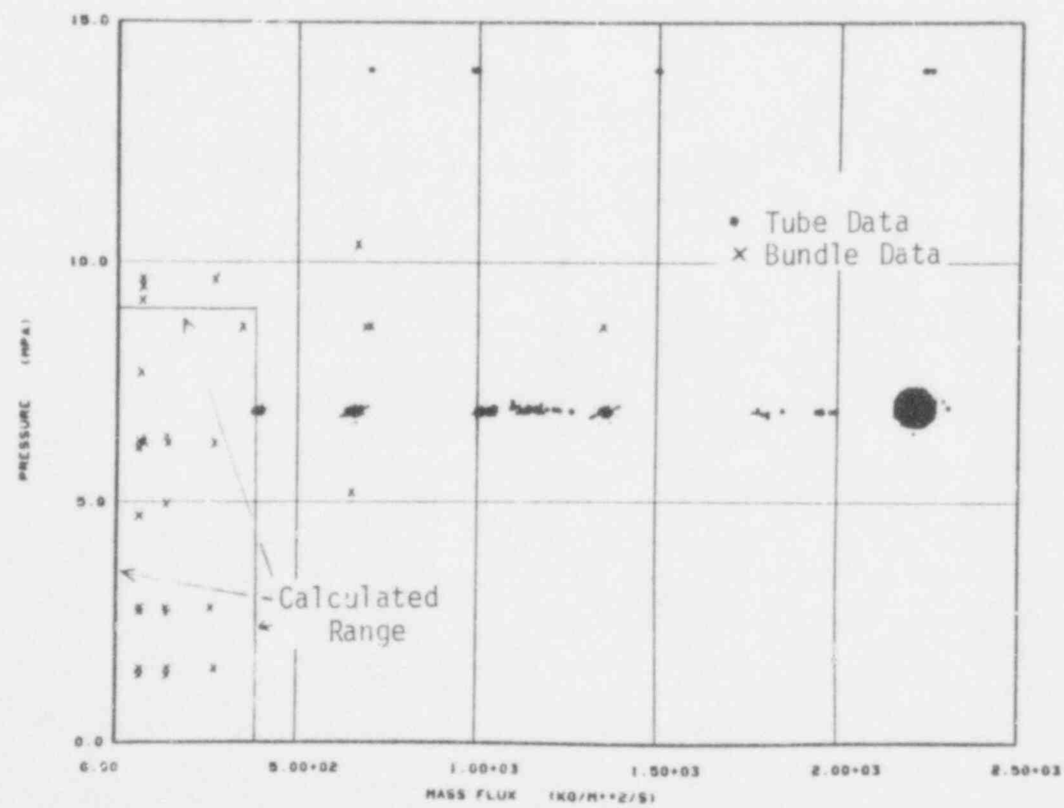

Fig. 25 Comparison of calculated variable ranges of pressure and mass flux for the core with selected data.

218

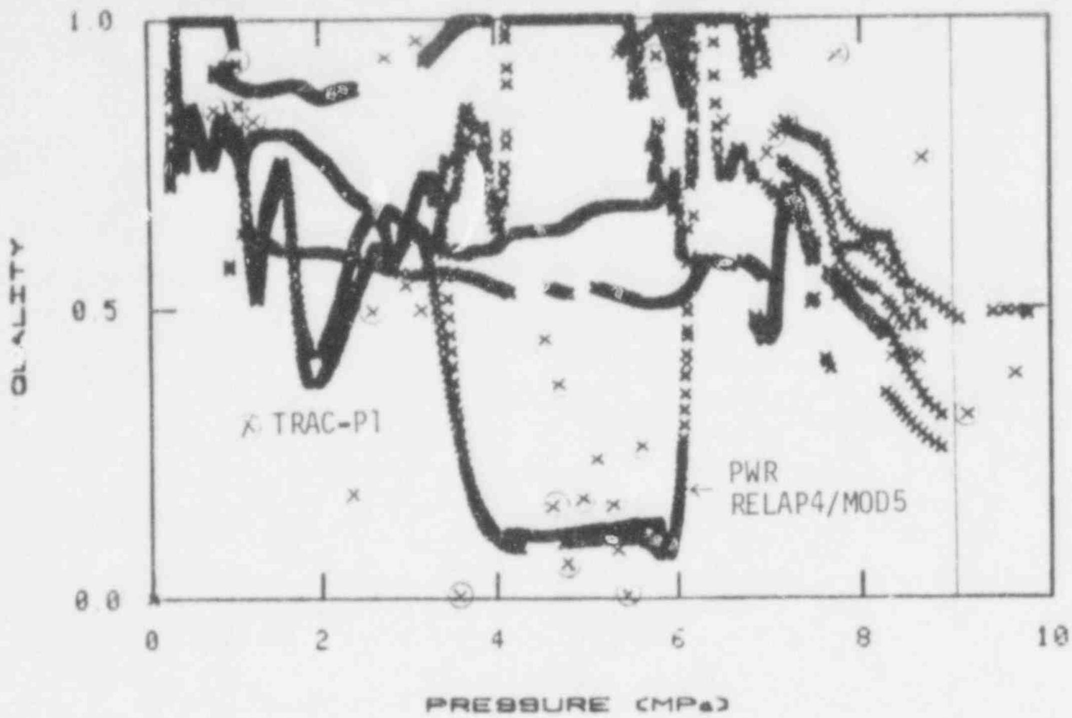

Fig. 26 Quality vs. pressure as calculated by the codes for the core for the film boiling regime.

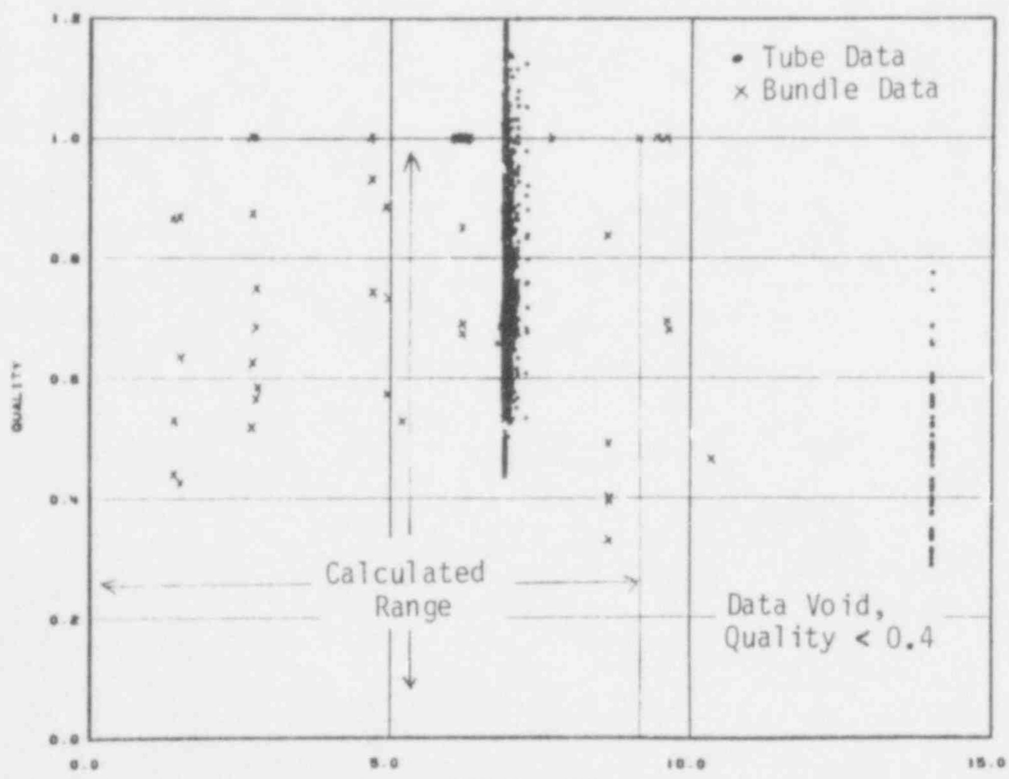

Fig. 27 Comparison of the calculated variable ranges of quality and pressure with selected data for film boiling.

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3. STEAM GENERATOR PRIMARY SIDE FILM BOILING COMPARISON

Film boiling is calculated to occur on the primary side of the steam generator by RELAP4/MOD6. As a result of the heat transfer logic RELAP4/MOD5 and TRAC-P1 do not calculate film boiling as shown in Figure 6. The ranges of the calculated variables are shown and then compared to the data in Figures 28 through 33 by pairs for the heat flux versus pressure, quality versus temperature and heat flux versus mass flux, respectively.

Comparison with similar variable plots for the core, that is, Figures 18, 20 and 22 indicate the calculated value of the variables are generally in the small end of the variable range, that is, low pressure, mass flux, heat flux and temperature while covering the same quality range. The calculated ranges are compared with the tube data values in Figures 29, 31, and 33. As can be seen, none of the data fall within the calculated ranges and the tube data is further away from the calculated variable ranges than the rod bundle data. If additional data were obtained to cover the calculated variable ranges, tube data would be desirable as the flow on the primary side of the steam generator is inside the tubes.

4. CRITICAL HEAT FLUX COMPARISON

The time for the calculated surface heat flux to equal the calculated CHF and the corresponding pertinent variable values were obtained for the RELAP4 cases for the core and the primary side of the steam generator where CHF was calculated to occur. The core is discussed in the following paragraphs. The conditions at CHF for the steam generator primary side are tabulated in Appendix B.

422 220

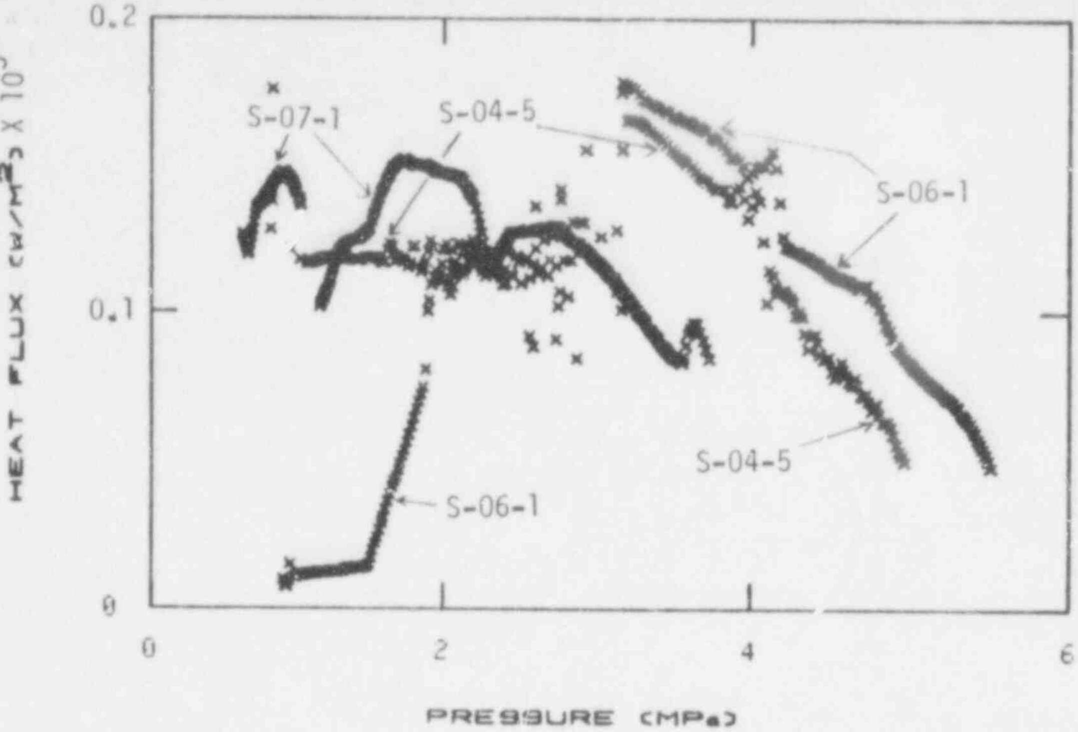

Fig. 28 Heat flux vs. pressure as calculated by RELAP4/MOD6 for the intact loop steam generator primary side for the film boiling regime.

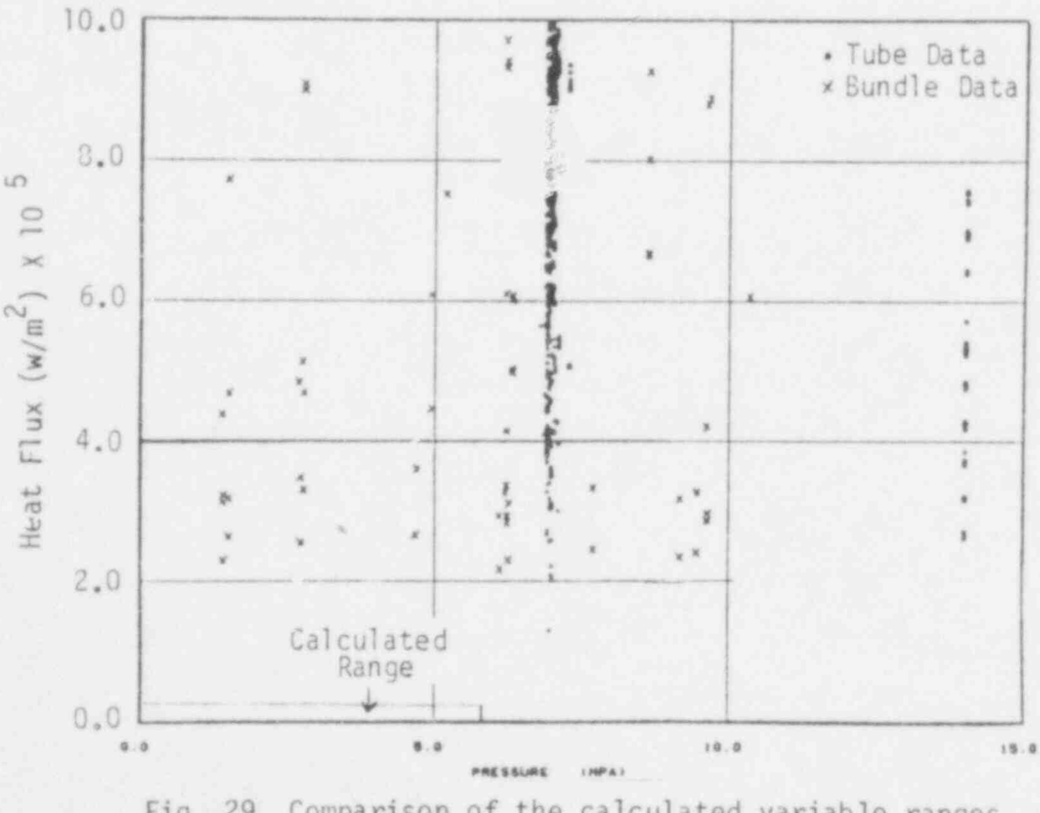

Fig. 29 Comparison of the calculated variable ranges of heat flux and pressure for the intact loop steam generator primary side with selected data for film boiling.

422 221

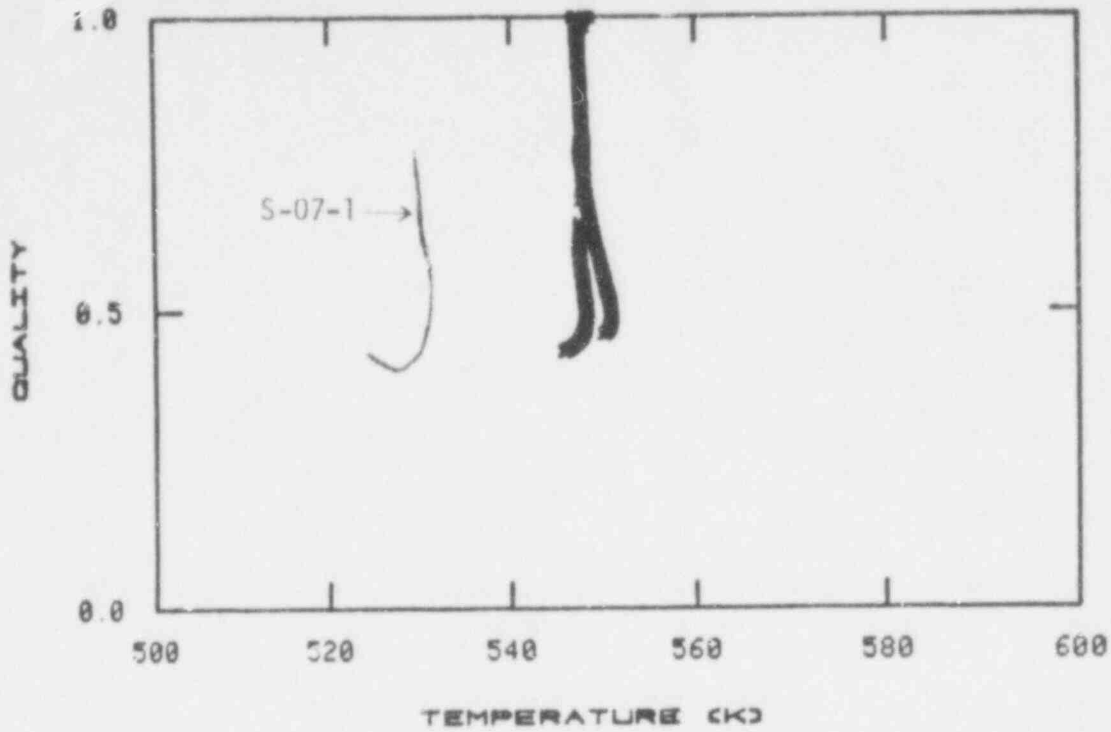

Fig. 30 Quality vs. temperature as calculated for RELAP4/MOD6 for the intact loop steam generator primary side for the film boiling regime.

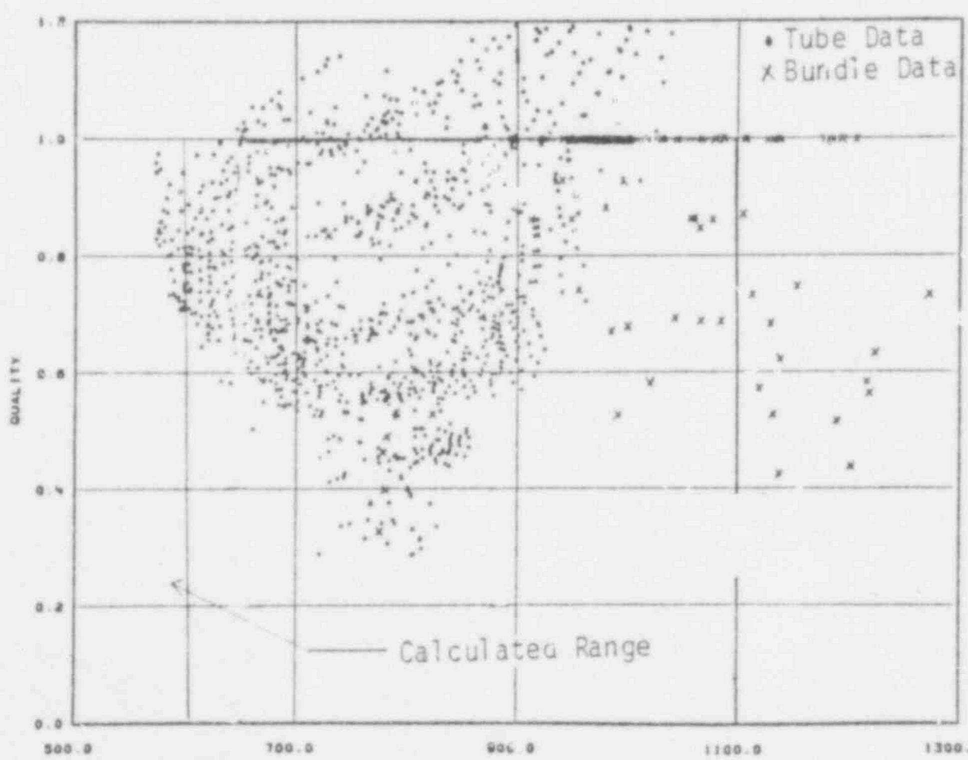

Fig. 31 Comparison of the calculated variable ranges of quality and surface temperature for the intact loop steam generator primary side with selected data for film boiling.

422 222

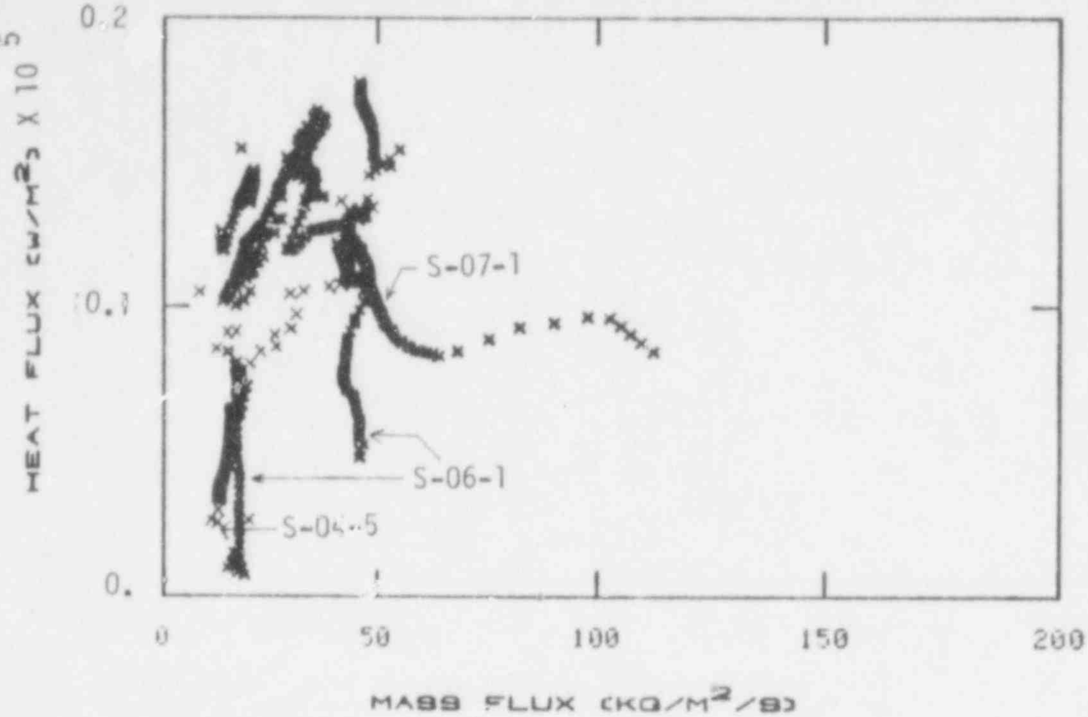

Fig. 32 Heat flux vs. mass flux as calculated by RELAP4/MOD6 for the intact loop steam generator primary side for the film boiling regime.

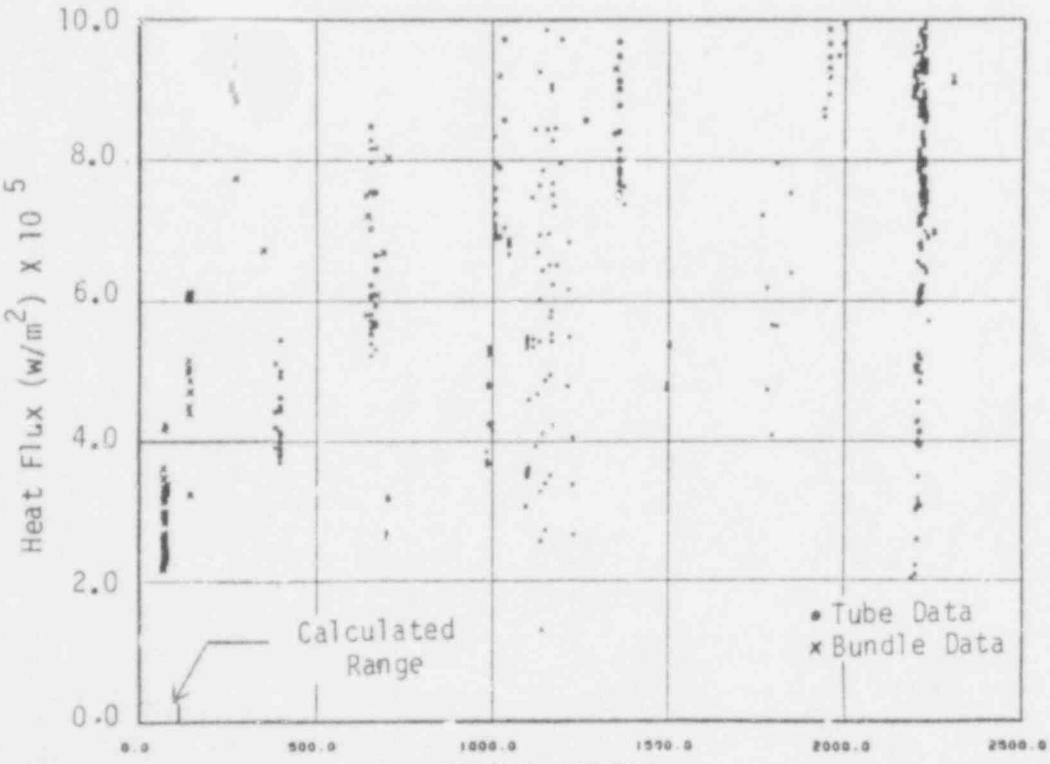

Fig. 33 Comparison of the calculated variable ranges of heat flux and mass flux for the intact loop steam generator primary side with selected data for film boiling.

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Table IV presents the calculated time of CHF along with the corresponding coolant quality, mass flux, pressure and surface heat flux. No results were available for the TRAC-P1 cases. CHF was calculated to occur twice for the hot and average channel for the PWR RELAP4/MOD5 case. A short time interval of nucleate boiling separated the calculated CHF onset. The calculation of multiple occurrences of CHF has been noted with the use of the RELAP4/MOD5 code and has been attributed to the B&W-2 correlation and its inverse dependence on mass flux[26].

CHF is calculated to occur in the RELAP4/MOD6 cases by the Modified Zuber or interpolation between the Hsu and Beckner and Modified Zuber correlations depending on the mass flux magnitude. No comparisons of the Hsu and Beckner, Zuber or Modified Zuber correlations with rod bundle data are known to the author.

The calculated values for the variables of heat flux and mass flux are compared with 102 points of CHF data[21] from the Heat Transfer Data Bank for rod bundles in Figure 34. Three of the nine calculated CHF points fall within the region where data is plotted, the initial calculation for the PWR RELAP4/MOD5 hot channel, and the calculations for Semiscale Test S-07-[illegible] and S-04-5 with RELAP4/MOD6. The mass flux for the remaining calculated points was much smaller than the mass flux for the data.

Figure 35 shows the calculated values for the variables of quality and pressure compared with the data. None of the calculated points fall within the range of the data.

The criteria for selection of the data are listed in Table V. Relaxation of the criterion on rod diameter size to include BWR rod diameters and larger would have added about 200 more points but would not have extended the data range to encompass the calculated points.

The CHF data in the Heat Transfer Data Bank is not up to date.

422 229

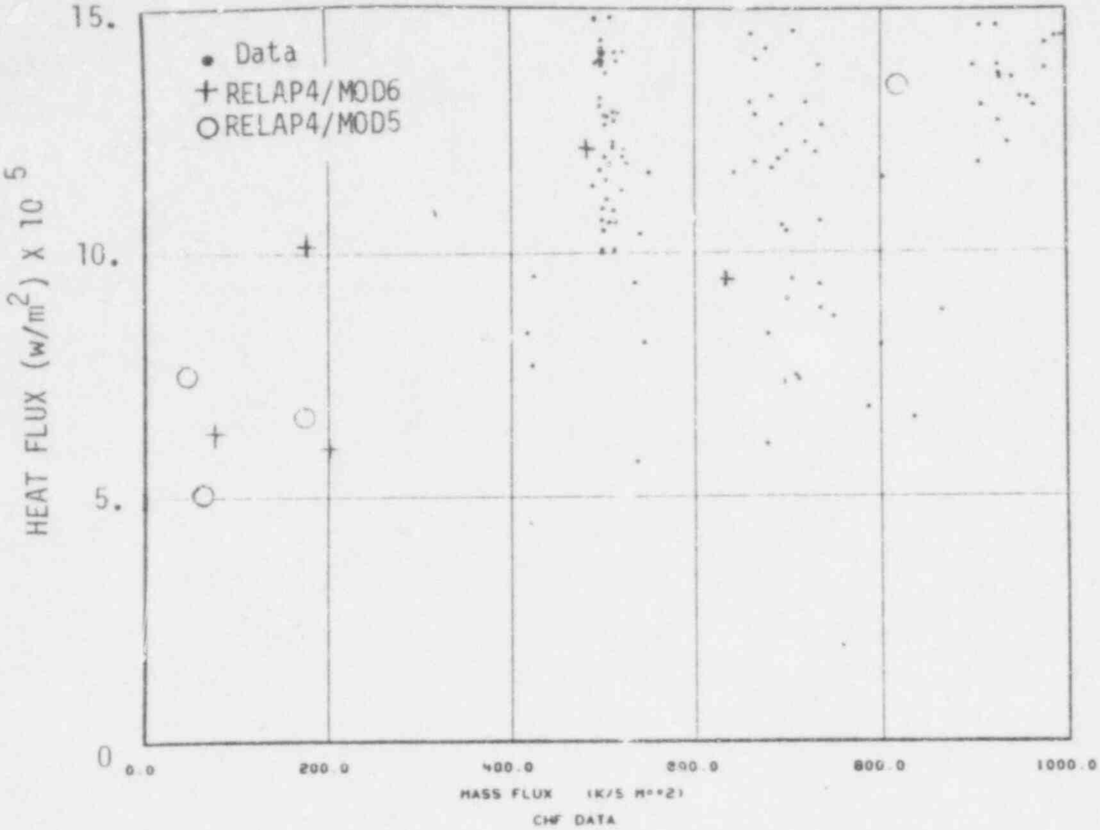

Fig. 34 Comparison of the calculated values of heat flux and mass flux at CHF with rod bundle data.

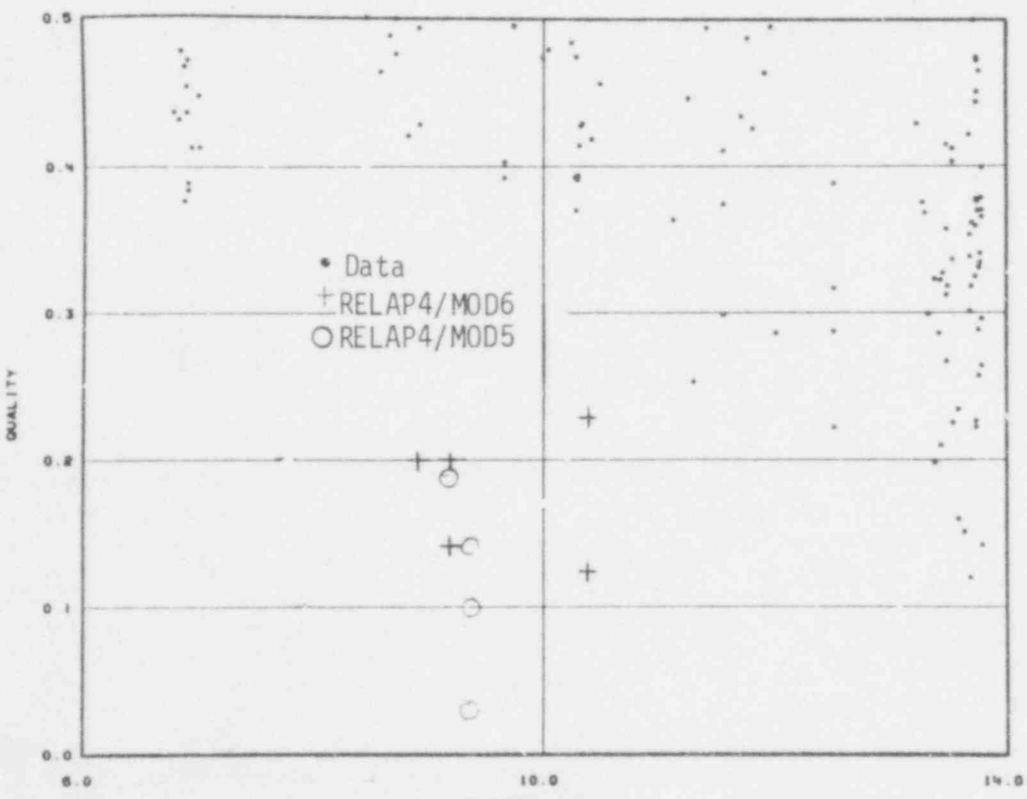

Fig. 35 Comparison of the calculated values of quality and pressure at CHF with rod bundle data.

422 225

TABLE IV

CALCULATED VARIABLE VALUES AT CHF FOR THE CORE

Case	Time After Blowdown[a] (s)	Quality	Mass Flux $\left(\frac{kg}{m^2\text{-}s}\right)$	Heat Flux $\left(10^5\ \frac{W}{m^2}\right)$	Pressure (MPa)
PWR RELAP4/MOD5					
Hot Channel	0.12	0.03	815	13.6	9.4
Hot Channel	0.4	0.14	175	7.2	9.4
Average Channel	0.36	0.10	50	8.3	9.4
Average Channel	0.72	0.18	70	5.1	9.2
PWR RELAP4/MOD6					
Hot Channel	0.41	0.14	180	10.2	9.2
Average Channel	0.75	0.20	80	6.8	9.2
S-07-1 RELAP4/MOD6[b]	0.35	0.23	635	9.3	10.7
S-04-5 RELAP4/MOD6[b]	0.31	0.13	485	12.1	10.7
S-06-1 RELAP4/MOD6[b]	0.47	0.20	200	6.4	8.9
PWR TRAC-P1[c]	<1.0				
S-07-1 TRAC-P1[b,c]	<0.4				

a The time for RELAP4 calculations was determined within 40 msec. The variable values correspond to the nominal time.

b The number prior to the code name refers to the particular Semiscale test.

c Code results not available.

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TABLE V

ROD BUNDLE CHF DATA SORTING CRITERIA

Parameter	Criteria
Rod Bundle Size	9 rods or more
Heated length	0.914 m or longer
Rod Diameter Range	9.906 - 12.45 mm
Pressure Range	6.2 - 13.8 MPa
Mass Flux Range	0 - 1000 $\frac{kg}{m^2\text{-}S}$
Quality Range	0 - 0.5
Heat Flux Range	0 - 1.48×10^6 $\frac{W}{m^2}$
Tube Surface	No wire wraps

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V. CONCLUSIONS AND RECOMENDATIONS

Analytical correlations used in the codes for calculation of the reactor core heat transfer regimes of transition boiling and film boiling during a postulated LOCA in a PWR caused by a large piping break are being applied to conditions outside the range for which the correlations have been developed and tested.

For transition boiling very limited data of questionable validity exist and essentially none fall within the calculated ranges of pressure, mass flux, quality, heat flux and surface temperature. Transition boiling is calculated to occur for less than one second during a 30 second blowdown for a large break from nominal core power. Thus, an error in calculated heat transfer magnitude is probably not critical to peak cladding temperature.

For film boiling a large amount of tube data exists but only a small fraction falls within the calculated variable ranges. Limited rod bundle data from a single source come closer to covering the calculated variable ranges. However. large data voids exist particularily at low values of heat flux, mass flux and quality. Correlations are being used with at least three of the five parameters considered outside the existing data base for calculation of film boiling heat transfer the majority of the total time, that is 20 to 27 s, the codes are calculating film boiling during a 30 second blowdown.

To quantify the importance of using the correlations over the ranges of variables where no data exist would require sensitivity analysis beyond the scope of this task.

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Limited CHF data exist within the Heat Transfer Data Bank over the variable ranges where the CHF correlations are being used in the core. However, it is not apparent that the correlations actually computing CHF in RELAP4/MOD6 and TRAC-P1 (Modfified Zuber, Zuber, Hsu and Beckner) have been testec against the available data.

Facilities (Semiscale Mod-1, THTF, TLTA) that have been developed and operated to obtain heat transfer information during the blowdown phase of a posulated LOCA have not provided data of sufficient accuracy to develop or test heat transfer correlations for transition or film boiling and the critical heat flux. The Semiscale Mod-3 facility has potential for providing information with improved accuracy over that available from the other facilities considered. Additional assessment procedures and an uncertainty analysis is needed for the facility. Better data might be obtained from the facilities by changing the transient blowdown mode of operation to a steady state mode.

Heat transfer logic and correlation usage for the primary and secondary side of the steam generator is not consistent between the codes. The film boiling regime is used in RELAP4/MOD6 for the primary side but the calculated variable ranges are not covered by film boiling tube data.

Additional separate effects data should be obtained to develop and test correlations for the transition and film boiling regimes. The Heat Transfer Data Bank should be updated to include recent CHF studies and the CHF correlations should be tested against the updated data. An analysis similar to the one reported should be conducted for the postulated small break. This analysis would disclose whether data for additional combinations of the heat transfer variables is needed to support application of the heat transfer correlations for small break studies.

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Separate effects tests for film boiling should be conducted in a steady state mode to eliminate the interpretation problems and measurement difficulties encountered with transient operation of an integral type blowdown facility. Transient integral tests need complex codes, that is, RELAP4/MOD6 or TRAC-P1 to calculate the local conditions needed. The codes themselves have uncertainties that are unknown or not quantified. Transient integral system tests also do not provide the versatility to systematically vary the heat transfer variables over the ranges of concern. Rod bundles with representative geometries are needed, that is, a minimum array size of 25 rods and two meter length. The facilities previously discussed should be re-evaluated to determine their capability to operate in a steady state mode and to determine what ranges of the heat transfer variables could be achieved. Any facility selected should be subjected to a rigorous uncertainty analysis to ensure usefulness of the data.

Because of the difficulty in obtaining transition boiling data, further evaluation is necessary before specifying hardware requirements.

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APPENDIX A

NODE IDENTIFICATION FOR CODE CASES

Table Al identifies the nodes (heat slabs, volumes, levels and cells) used for the analysis presented.

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TABLE A1

NODE IDENTIFICATION FOR CODE CASES

RELAP4 Case	Core: Heat Slab	Core: Volume	Steam Generator Primary: Heat Slab	Steam Generator Primary: Volume	Steam Generator Secondary: Heat Slab	Steam Generator Secondary: Volume	Tape Configuration Control	Reference
PWR/MOD5 and							A41172	A1
PWR/MOD6							H00060IB	
Hot Channel	16R[a]	43	25L	6	25R	27	A47703	
Average Channel	4R	37						
S-07-1	6R	39	14L	13	14R	15	T9K122	A2
S-04-5 and S-06-1	12R	39	22L	23	22R	33	T9M664 T9W540	A2

TRAC-P1 Case	Core: Core Level	Core: Vessel Level	Core: Rod	Core: Component	Steam Generator: Primary Cell	Steam Generator: Secondary Cell	Code Version: Configuration Control
PWR	3	6	10	32	5	4	20.1 Ref. A3
S-07-1	6	9	5	2	2,20	3,10	19.0 H00373IB

a The R and L designate the right and left surface of the heat slab in accordance with the RELAP4 code.

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APPENDIX A. REFERENCES

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APPENDIX B

ADDITIONAL CALCULATED HEAT TRANSFER REGIME USAGE AND VARIABLE RANGES

This appendix contains additional illustrations of the heat transfer regime usage in the core during blowdown. Figure B1 compares the RELAP4/MOD6 calculated regime usage for the different systems, a PWR, Semiscale Mod-1 and Semiscale Mod-3. Figure B2 compares the regime usage as calculated by RELAP4/MOD6 and TRAC-P1 for the Semiscale Mod-3 system.

Also included are Tables B1 through B12 showing the heat transfer variable ranges calculated for the core, and steam generator where comparisons with data were not made.

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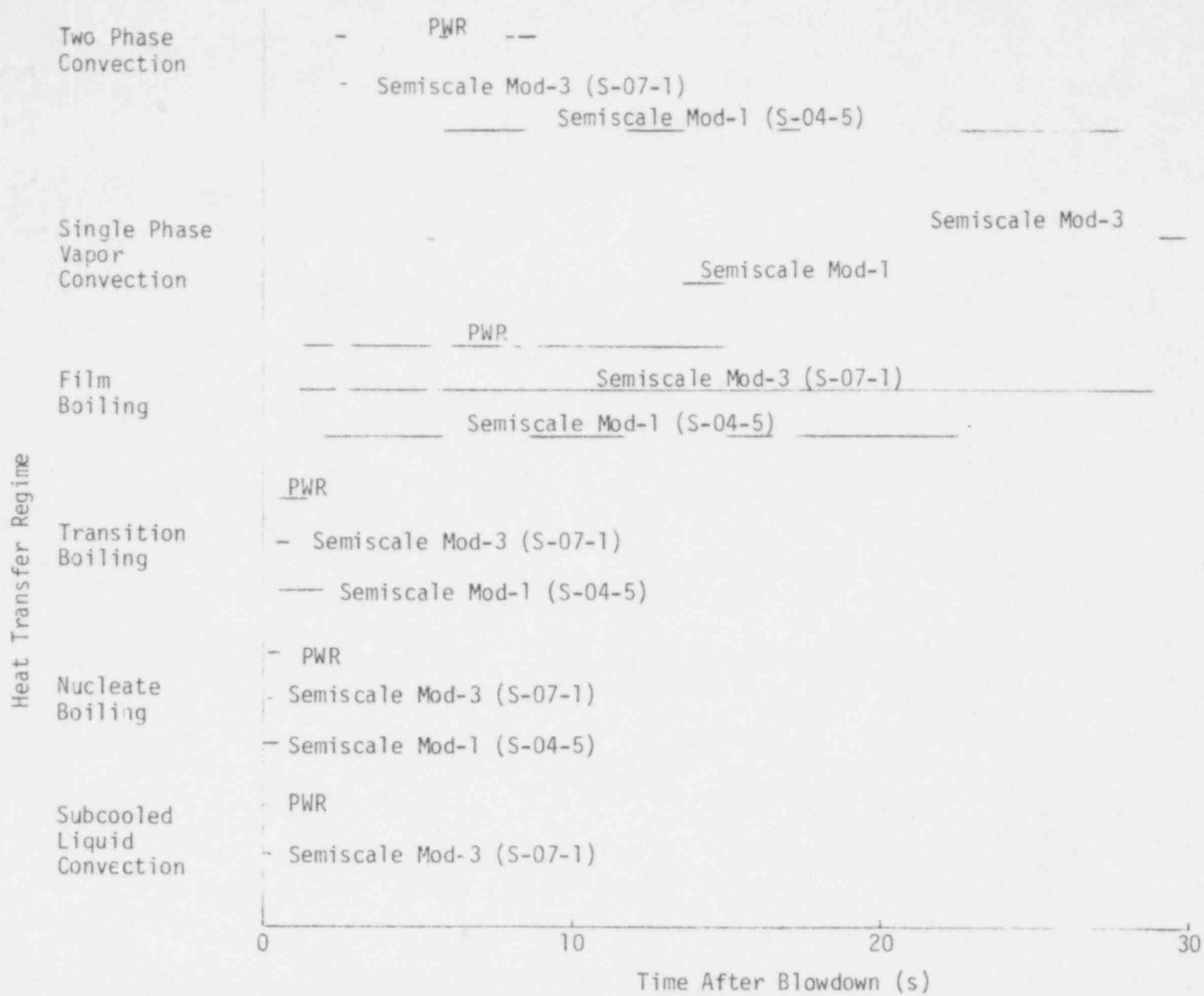

Fig. B-1 Heat transfer regime usage during blowdown for a core hot channel as calculated by RELAP4/MOD6 for a PWR, Semiscale Mod-1 and Semiscale Mod-3.

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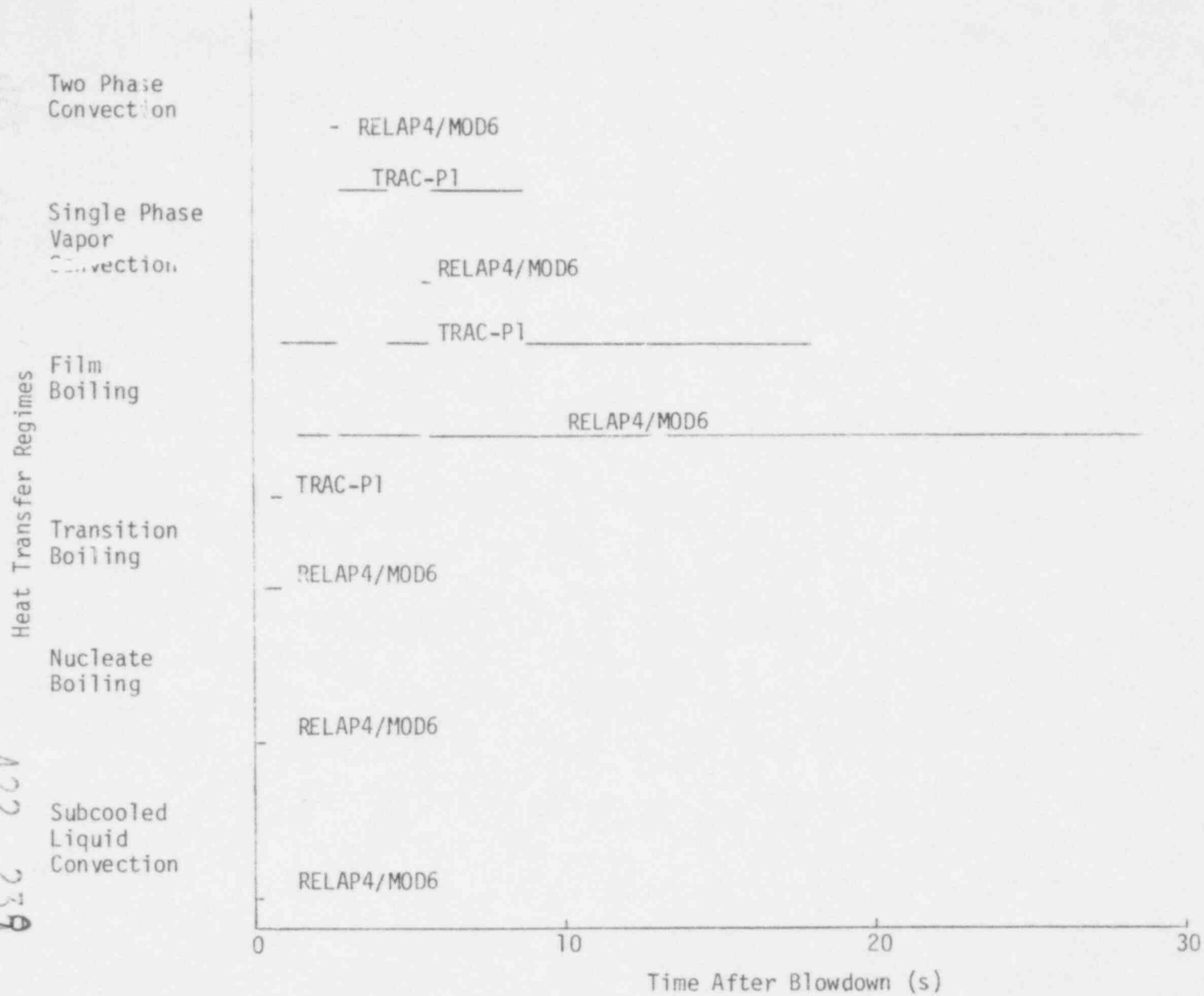

Fig. B-2 Heat transfer regime usage during blowdown for a core as calculated by RELAP4/MOD6 and TRAC-P1 for Semiscale Test S-07-1.

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TABLE B1

CORE SUBCOOLED LIQUID CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5					
Hot Channel	15.6	3680	0	610	10.6
Average Channel	15.6	3870	0	595	7.8
PWR RELAP4/MOD6					
Hot Channel	15.6	3680	0	610	10.6
Average Channel	15.6	3870	0	595	7.8
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6	15.6	3430	0	615	10.9
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	15.6	1030	0	610	4.6
S-04-5 RELAP4/MOD6[b]					

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B2

CORE NUCLEATE BOILING VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5					
Hot Channel	10.2-9.4	1000-300	0-0.2	610	14.5-10
Average Channel	10.4-9.2	1000-150	0-0.2	590	9-6.5
PWR RELAP4/MOD6					
Hot Channel	10.3-9.4	1250-250	0-0.1	600	14-9.5
Average Channel	10.3-9.2	1250- 50	0-0.17	590	8.5-5.5
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6	11.2-10.8	2600-900	0-0.15	610	12-11.4
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	11.1-8.8	4500-100 [1000-100][c]	0-0.2	590	13-6.3
S-04-5 RELAP4/MOD6	15.7-11	1900-750	0-.08	610	18-11

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

c Brackets indicate range for bulk of calculation.

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TABLE B3

CORE TWO PHASE CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5					
Hot Channel	10-4	100-2	0.1-1.0	950-650	1.5-0.05 [0.4-0.5][c]
Average Channel	8-4	150- 0	0.1-0.8	840-700	0.22-0.1
PWR RELAP4/MOD6					
Hot Channel	7-5.6	15-0	0.83-0.91	910-860	0.2-0.17
Average Channel	7.1-5.6	15-0	0.7-0.9	775-740	0.12-0.10
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6	7	9-5	0.7	870	0.175
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	7.3-0.2	14-0	6.53-0.73	800-650	0.11-0.04
S-04-5 RELAP4/MOD6	7-0.2	14-0	0.5-1	1050-1010	0.32-0.22

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

c Brackets indicate range for bulk of calculation.

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TABLE B4

CORE SINGLE PHASE VAPOR CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5					
Hot Channel	6-0.2	70-15	$\geq$ 1.0	950-830	2.4-0.4
Average Channel	0.3-0.25	27-15	$\geq$ 1.0	950-830	0.55-0.3
PWR RELAP4/MOD6[b]					
Hot Channel					
Average Channel					
PWR TRAC-P1	2.4-0.2	150-5		970-850	2 8-0.1
S-07-1 RELAP4/MOD6	0.2	9-0		750	0.05-0.01
S-07-1 TRAC-P1	8.4-6.2	120-10		940-860	3.5-0.5
S-06-1 RELAP4/MOD6[b]					
S-04-5 RELAP4/MOD6	5-3.9	10-0		1050	0.27-0.17

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

TABLE B5

STEAM GENERATOR PRIMARY SIDE TWO PHASE CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	4-1.3	125-0	0.2-0.95	530-505	1.7-0
PWR RELAP4/MOD6[b]					
PWR TRAC-P1	9.5-0.3	5200-0	0-0.95	570-490	3.6-0
S-07-1 RELAP4/MOD6	0.5-0.3	12.5-0	0.55-0.7	530	0.02
S-07-1 TRAC-P1	11.5-3	1700-0	0-1.0	580-530	2.8-0
S-06-1 RELAP4/MOD6[b]					
S J4-5 RELAP4/MOD6	0.8	13	0.98	550	0.02

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B6

STEAM GENERATOR PRIMARY SIDE SUBCOOLED LIQUID CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	15.6-5	5800-0	0-0.13	570-535	3.3-0
PWR RELAP4/MOD6	15.2-5	5800-100	0-0.13	570-540	3.3-0
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6	15-5	1800-0	0-0.19	570-535	3.3-0
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	15.6-7	1600-0	0-0.08	565	3.7-0
S-04-5 RELAP4/MOD6	15-6	2400-100	0-0.8	560-550	1.0-0

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B7

STEAM GENERATOR PRIMARY SIDE NUCLEATE BOILING VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	4.8-4.2	20-05	0.19-0.13	530	0.18-0
PWR RELAP4/MOD6	5.3-3.2	100-0	0.22-0.12	540-515	0.47-0
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6	5.1-4	180-110	0.32-0.18	530	0.4-0
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	7.1-5.7	220-50	0.4-0.08	550	0.4-0
S-04-5 RELAP4/MOD6	6.3-5.2	220-50	0.38-0.08	550	0.5-0

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B8

STEAM GENERATOR PRIMARY SIDE TRANSITION
BOILING VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5[b]					
PWR RELAP4/MOD6[b]					
PWR TRAC-P1[b]					
S-07-1 RELAP4/MOD6[b]	4.0-3.8	[illegible]05-270	0.33	525	0.35-0.15
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	5.7-5.5	50-45	0.44	550	0.35-0.05
S-04-5 RELAP4/MOD6	5.25-5	40-33	0.4		0.45-0.05

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B9

STEAM GENERATOR PRIMARY SIDE
SINGLE PHASE VAPOR CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	1.25-0.25	25-2	510-530	0.6-0.1
PWR RELAP4/MOD6[b]				
PWR TRAC-P1[b]				
S-07-1 RELAP4/MOD6[b]				
S-07-1 TRAC-P1[b]				
S-06-1 RELAP4/MOD6	1.0-0.2	15-0	550	0.008-0
S-04-5 RELAP4/MOD6	0.8-0 2	12-6	550	0.02-0.005

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B10

STEAM GENERATOR SECONDARY SIDE CONVECTION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Tempera-ture (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	4.7-4.4	b	0.07	535-510	0.6-0
PWR RELAP4/MOD6 (Same logic as MOD5)	5.5-5.1	2000-0	0.07	540	3.0
PWR TRAC-P1	7.2-5.8	b	b	560	3-0
S-07-1 RELAP4/MOD6	5 8-4.2	15-0	0.045	555-535	3-0
S-07-1 TRAC-P1	6.8-6.	b	0.9-0.1	555	3-0
S-06-1 RELAP4/MOD6	7.2-6.9	b	0.015	550	0.09-0
S-04-5 RELAP4/MOD6	6.3-5.8	25-0	0.01	550	0.25-0

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

TABLE B11

STEAM GENERATOR SECONDARY SIDE CONDENSATION VARIABLE RANGES[a]

Case	Pressure (MPa)	Mass Flux (kg/m^2-s)	Quality	Wall Temperature (K)	Heat Flux (10^5 W/m^2)
PWR RELAP4/MOD5	5.4-4.7	85-5	0.07	545-535	3-0
PWR RELAP4/MOD6	5.	80-20	0.07	540	0.02-0
PWR TRAC-P1	7.1-4.2	b	b	560-520	5-0
S-07-1 RELAP4/MOD6	5.-4.75	0	0.05	535	1.8-0
S-07-1 TRAC-P1[b]					
S-06-1 RELAP4/MOD6	6.9-5.9	0.6-0	0.015	545-525	0.07-0
S-04-5 RELAP4/MOD6	6.2	0	0.01	550	0.3-0

a Single number listed where significant range doesn't exist.

b Data not available or regime not used.

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TABLE B-12

CALCULATED VARIABLE VALUES AT CHF FOR THE PRIMARY SIDE OF THE INTACT LOOP STEAM GENERATOR

Case	TIME AFTER BLOWDOWN (s)	QUALITY	MASS FLUX $\left(\frac{kg}{m^2\text{-}s}\right)$	HEAT FLUX $(2x10^5\ W/m^2)$	PRESSURE (MPa)
S-07-1 RELAP4/MOD6	15.9	0.59	112	0.43	3.9
S-04-5 RELAP4/MOD6	12.2	0.38	42	0.5	5.2
S-06-1 RELAP4/MOD6	12.8	0.42	50	0.4	3.7

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APPENDIX C

FLOW PARAMETERS

Code calculations previously completed were used for this study. Thus, evaluation of the flow parameters was limited to the data available. The codes RELAP4 and TRAC-P1 have some fundamental differences in the assumptions built into their hydraulic models which necessitated some differences in parameter evaluation.

RELAP4 models the fluid in the core and the steam generator with phase slip at the junctions but requires a homogeneous flow in the central volume. TRAC-P1 is based on a model permitting separated flow with the phases not in equilibrium.

The quality calculated by RELAP4[C-] is a static value[C2], X_s, based on the mass of the phases present in the volume at an instant in time:

$$X_s = \frac{W_g}{W_f + W_g} .$$

The symbol W refers to weight or mass and the subscript g refers to the vapor and f refers to the liquid. This value could not be converted to other more appropriate forms as the phase velocitics were not available.

A flow quality, X_f, was computed from the TRAC-P1 output for flow in the core using the phase mass fluxes, G, as follows:

$$X_f = \frac{G_g}{G_f + G_g} .$$

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The flow quality is limited to values between zero and one for cocurrent flow. In the core the void fraction, phase velocities and phase densities were used to calculate the mass flux.

For the primary side of the steam generator the flow quality was computed from TRAC-P1 output using the phase slip ratio, S, phase densities and void fraction as follows:

$$\frac{1 - X_f}{X_f} = \frac{1}{S} \frac{\rho f}{\rho_g} \frac{1 - \alpha}{\alpha} \quad .$$

For countercurrent flow the absolute value of the slip ratio was used to obtain a flow quality equal in value to the cocurrent flow situation.

Separate effect heat transfer data has been correlated in terms of a thermodynamic quality, X_e, as this value can evaluated from an energy ballance over the apparatus. The quality, X_e, is defined as:

$$X_e = \frac{h - h_f}{h_{fg}} \quad .$$

Where the phases are separate and not in equilibrium the mixture enthalpy, h, can be determined from the phase mass fluxes and enthalpies by:

$$h = \frac{G_f h_f + G_g h_g}{G_f + G_g} \quad .$$

The enthalpy terms are evaluated at the same pressure. The definition works well fcr cocurrent flow. Negative values indicate subcooling and values greater than one indicate superheating. For countercurrent flow the absolute magnitude of the mass flux can be used to obtain an X_e equivalent to the cocurrent situation.

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Separate effects heat transfer tests generally have cocurrent flow while code calculations may be either cocurrent or countercurrent flow.

The mass flux was obtained from the RELAP4 output by dividing the mass flow rate by the flow area. The mass flowrate is the average of the net rates at the adjacent junctions. For TRAC-P1 the mass flux in the core was combining the local axial phase velocities with the void fraction and phase densities as follows:

$$G = \rho_f V_{ef} (1-\alpha) + \rho_g V_{eg} \alpha .$$

For the primary side of the steam generator the available average mixture velocity V_{em} was used with a mixture density as follows:

$$G = \rho_m V_{em}.$$

The mixture density when multiplied by the mixture velocity yields the mass flux(C3).

The heat fluxes were obtained directly except for the TRAC-P1 output for the core where the heat transfer coefficients were given. The heat flux was then calculated by combining the phase heat transfer coefficient with the appropriate temperature difference as follows:

$$q'' = h_{c_f} (t_{wall}-t_f) + h_{cg} (t_{wall}-t_g).$$

The heat transfer mode was available with the RELAP4 output and TRAC-P1 output for the core. For the TRAC-P1 steam generator, the mode was determined by evaluating the code logic.

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APPENDIX C. REFERENCES

C1. RELAP4/MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, Users Manual, Volume 1 RELAP4/MOD5 Description, ANCR-NUREG-1335 (September 1976).

C2. R. T. Lahey, Jr., Two-Phase Flow in Boiling Water Nuclear Reactors, NEDO-13388 (July 1974).

C3. TRAC-P1: An Advanced Best Estimate Computer Program for PWR LOCA Analysis, 1. Methods, Models, User Information, and Programming Details, NUREG/CR-0063, LA-7279-M5, Vol. 1, (June 1978).

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