Modification of RELAP5-3D in Accordance with Appendix K of 10 CFR 50.46

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Abstract

In light water reactors, particularly the pressurized water reactor (PWR), the severity of a LOCA will limit how high the reactor power can operate. Although the best-estimate LOCA licensing methodology can provide the greatest margin on the PCT evaluation during LOCA, it generally takes more resources to develop. Instead, implementation of evaluation models required by the Appendix K of 10 CFR 50 upon an advanced thermal-hydraulic platform also can gain significant margin for the PCT calculation. The compliance of the current RELAP5-3D code with Appendix K of 10 CFR50 has been evaluated, and it was found that there are ten areas where code assessment and/or further modifications were required to satisfy the requirements set forth in the Appendix K of 10 CFR 50. All of the ten areas have been further evaluated and the RELAP5-3D has been successfully modified to fulfill the associated requirements. To verify and assess the development of the Appendix K version of RELAP5-3D, nine kinds of separate-effect experiments were adopted. Through the assessments against separate-effect experiments, the success of the code modification in accordance with the Appendix K of 10 CFR 50 was demonstrated. Another six sets of integral-effect experiments will be applied in the next step to assure the integral conservatism of the Appendix K version of RELAP5-3D on LOCA licensing evaluation.

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1. Introduction

The Loss of Coolant Accident (LOCA) is one of the most important design basis accidents (DBA). In light water reactors, particularly the pressurized water reactor (PWR), the severity of a LOCA will limit how high the reactor power can operate. In the regulatory analysis (USNRC, 1987), it was estimated that if the peak cladding temperature (PCT) during a LOCA decreases by 100°F, it would be possible to raise the plant power by 10%. The revision of 10 CFR50.46 in 1988 stated that two kinds of LOCA licensing approaches can be accepted, namely the **realistic** and **Appendix K** methodologies. The realistic licensing technique describes the behavior of the reactor system during a LOCA with best estimate (BE) codes. However, the BE analysis method and inputs must be identified and assessed so that the uncertainties in the calculated results can be estimated to a high confidence level. Alternatively, an ECCS evaluation model (EM) also can be developed in conformance with the required and acceptable features of the Appendix K of 10 CFR 50. The Appendix K approach will guarantee the conservatism of the calculation results, instead of answering the analytical uncertainty. It is widely believed that the realistic approach can more precisely calculate the sequences of a LOCA accident, and therefore provides a greater margin for the PCT evaluation. However, the development of the realistic LOCA methodology is long and costly, and the safety authority is highly demanding in their approach to evaluate uncertainties. For instance, Westinghouse took about 50 man-years over 10 years to develop their best-estimate large break LOCA methodology, and it is the only company to date that has acquired the final approval from the U.S. regulatory authority in 1995 using the new realistic large

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break LOCA methodology. Regarding the Appendix K LOCA methodology, it is quite interesting that comparisons of the calculated PCT obtained using early thermal hydraulic codes versus the advanced thermal-hydraulic codes show that the advanced codes calculated a significantly lower PCT than the early ones for the same set of conditions required by Appendix K of 10 CFR 50. For instance, the PCT of Taiwan's Maanshan Nuclear Power Plant calculated by the latest Westinghouse Evaluation Model BASH (Westinghouse, 1987) is 445°F (2170°F→1725°F) lower than that of 1981's calculation (Taipower Company, 1982).

Although the realistic LOCA methodology can provide greater margin for PCT evaluation, Appendix K requirements along with an advanced thermal-hydraulic platform still can offer significant margin. Besides, the Appendix K LOCA methodology can be developed with fewer resources. Therefore, a program to modify RELAP5-3D in accordance with Appendix K of 10 CFR 50 was launched by INER (Institute of Nuclear Energy Research, Taiwan), and it consists of six sequential phases of work. It includes (1) RELAP5-3D compliance evaluation and EM models as well as assessment data collection; (2) Individual model implementation and stand-alone verification; (3) Model integration to generate the Appendix K version of RELAP5-3D (RELAP5-3D/K); (4) Integral assessment of the new developed RELAP5-3D/K; (5) LOCA licensing analysis with RELAP5-3D/K for the Taiwan's Maanshan Power Plant; and (6) Licensing submittal covering both RELAP5-3D/K development and plant specific application for approval. In this paper, the compliance of RELAP5-3D against the Appendix K of 10 CFR 50 has been evaluated, and all the required Appendix K been successfully implemented into the best estimate version of

RELAP5-3D.

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2. Compliance Evaluation of RELAP5-3D with Appendix K of 10 CFR50.46

Section I of Appendix K can be divided into four subsections: A) sources of heat during the LOCA; B) swelling and rupture of the cladding and fuel rod thermal parameters; C) blowdown phenomena; and D) post-blowdown phenomena. The RELAP5-3D code in its current state has a number of models that enable it to meet many of the Appendix K requirements with no modification. Based on the configuration of the RELAP5-3D, actions required to meet Appendix K requirements fall within three categories: Category 1- required models are missing and must be added to the RELAP5-3D; Category 2- requirements can be satisfied by preparing the correct input for the code for a required analysis; Category 3- requirements that can only be satisfied by performing a series of parametric or sensitivity calculation once all the Appendix K required models are present in the code. For category 1, there are ten areas in the RELAP5-3D need to be further assessed and/or modified to achieve conformance with Appendix K requirements. Those ten areas are:

2.1 Fission Product Decay

Appendix K requires that the heat generation from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS standard (October 1971). As stated in the RELAP5-3D documents (RELAP5-3D Code Development Team, 1998), the user can select the decay power model based on either the 1973 ANS proposed Standard or 1979 ANSI/ANS standard. Therefore, the required 1971 ANS standard would be inserted into the RELAP5-3D.

2.2 Metal –Water Reaction Rate

Appendix K requires that the rate of energy release, hydrogen generation, and

cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation, while the metal-water reaction model included in the existing RELAP5-3D is based on the Cathcart model. Consequently, the code does not meet the Appendix K requirement and the Baker-Just model would replace the built-in Cathcart model.

2.3 Discharge Model

Appendix K requires that for all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by the Moody model. The critical flow models included in RELAP5-3D do not include the Moody model. Consequently, the required Moody model would be inserted into RELAP5-3D.

2.4 ECC Bypass Model

As stated in Appendix K, the ECC water injected into the inlet lines or the reactor vessel during the bypass period shall be subtracted from the reactor vessel calculated inventory. The RELAP5-3D code does not have a "bypass model" to comply with Appendix K requirements. However, the proposed method for meeting the Appendix K bypass model requirements is to use: (a) the counter-current flow limiting (CCFL) model to prevent penetration of ECC liquid down into the reactor vessel, and (b) an on-line ECC water subtraction scheme to remove the ECC water that is injected into the primary system before the end of ECC bypass.

2.5 Critical Heat Flux during Blowdown

The RELAP5-3D code uses the 1986 AECL-UO Critical Heat Flux Lookup Table to calculate the critical heat flux point. This correlation is not among the approved correlations set forth in Appendix K. The set of Appendix K CHF correlations used in

RELAP4/MOD7 (Stephen R. Behling, 1981) would be adopted, which includes B&W-2, Barnett and modified Barnett correlations. These sets of correlations were on the approval list of Appendix K and they can cover the right range of interest.

2.6 Post-CHF Heat Transfer during Blowdown

In the present version of RELAP5-3D, transition boiling is modeled using the Chen correlation, and film boiling is modeled using the Bromley correlation for the conductive mechanism and the Sun-Gonzales-Tien correlation for the radiation mechanism. These correlations are not among the approved list of Appendix K. Therefore, RELAP5-3D has to be modified by inserting approved correlations. Referring to the Appendix K associated model structure of RELAP4/MOD7, Groeneveld 5.7 for high flow and modified Bromley for low flow would be adopted for film boiling heat transfer. As for the transition boiling, the McDonough, Milich and King correlation would be adopted.

2.7 Prevention from Returning to Nucleate Boiling and Transition Boiling Heat Transfer Prior to Reflood

Appendix K requires that after CHF is first predicted at any axial fuel rod location during blowdown, the calculation shall not use nucleate boiling and transition boiling heat transfer correlations at that location subsequently during the blowdown, unless justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA. The RELAP5-3D code does not contain any logic to prevent returning to nucleate boiling or transition boiling once CHF has occurred. Consequently, logical prevention would be inserted into the current heat transfer model selection logic of RELAP5-3D.

2.8 Core Flow Distribution during Blowdown

Appendix K requires that calculations of average flow and flow in the hot region shall take into account cross flow between regions. Furthermore, the calculated flow shall be smoothed to eliminate any rapid oscillations with period less than 0.1 seconds. The cross flow between regions can be properly calculated by using the built-in cross flow junctions connecting regions laterally. However, the capability of RELAP5-3D to calculate lateral flow mixing resulted from fuel blockage would be further assessed.

2.9 Reflood Rate for PWRs

As required in Appendix K, the ratio of the total fluid flow at the core exit plane to the total fluid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data. The PSI reflood model (G. Th. Analytis, 1996) in RELAP5-3D calculates the time and rate of flooding of the core, by taking into account the thermal and hydraulic characteristics of the core and of the reactor system. However, whether the carryover fraction calculated by the model is complied with applicable experimental data needs to be further verified. To determine whether the model is in conformance with Appendix K requirements can only be determined by performing applicable code assessment studies.

2.10 Refill and Reflood Heat Transfer for PWRs

Appendix K requires that for reflood rates of 1 inch/sec or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results. Whether the PSI reflood model used in the RELAP5-3D is acceptable must be determined by comparison with FLECHT data for a range of parameters consistent with the transient of interest. However, with the

reflood rate less than 1 inch/sec, heat transfer coefficients shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur. The RELAP5-3D reflood model does not calculate heat transfer by steam cooling only when reflood rates are less than 1 inch/sec.

3. Code Modifications and Assessments to Satisfy Requirements of Appendix K of 10 CFR 50

The best-estimate version of RELAP5-3D was modified and assessed to fulfill requirements set forth in the Appendix K of 10 CFR 50. Separate-effects experiments were applied to assess specific code models and assure each modification working properly. The separate-effects assessment cases for each modification are summarized in Table 1.

3.1 Metal-Water Reaction Rate

Since melting of fuel cladding is not the applicable domain, the parabolic rate low from the Baker-Just model (Louis Baker, Jr., 1962) would be applied to calculate the fuel oxidation from zirconium-water reaction:

$$(\frac{dr}{dt}) = \frac{B}{Ro - r} Exp(-\frac{G}{T_s}) \tag{3.1}$$

The above original form of Baker-Just model was re-derived, and the final form used for coding is:

$$AP = 6.98 \times 10^{-5} \times EXP(\frac{-22898.8}{T_s}) \Delta t \tag{3.2}$$

$$DRP1 = (DRP^2 + AP)^{1/2} \tag{3.3}$$

Once the oxidation thickness has been evaluated, the associated amount of reaction heat added to the cladding and hydrogen generation also would be calculated. The Cathcart data (J.V. Cathcart, 1977) was used to assess the implementation of the Baker-Just models into RELAP5-3D. Cathcart measured the isothermal reaction rates of Zircaloy-4 tubes in steam at elevated temperatures. After the specified oxidation time, the tube was removed and the oxide thickness was measured using standard metallographic techniques. Typical assessment calculations are shown in Figure 1 and

Figure 2. It can be seen that at a higher bath temperature (1500°C), the conservatism of the Baker-Just model is very clear. However, at the bath temperature around 1000°c, the Baker-Just model matches the data very well.

3.2 Discharge Model

The Moody model for the calculation of two phase choked flow and the Henry Hauske model for the single phase liquid choked flow were added to RELAP5-3D to make a break flow evaluation model. Regarding applying the Moody model, the stagnation conditions (p_0, h_0) need to be derived from the cell center immediately upstream of the exit plane. The stagnation enthalpy can be calculated from the cell center properties as:

$$h_0 = (h_f + \frac{v_f^2}{2})(1-x) + (h_g + \frac{v_g^2}{2})x \tag{3.4}$$

where the local enthalpies, fluid velocities and flow quality are evaluated at the equilibrium condition at the cell center. By assuming an isentropic process, the stagnation pressure can then be obtained from the local entropy defined by the cell center properties and the stagnation enthalpy through steam table iteration:

$$P_o = P_o(h_o, s(h, P)) \tag{3.5}$$

Data from Marviken Test 22 (L. Erickson, 1979) was used to assess the implementation of the Moody model. Marviken Test 22 was a full-scale critical flow test. The break was connected to the bottom of a large pressure vessel. The pressure vessel, which was originally part of the Marviken Nuclear Power Station in Sweden, was 5.2 meters in diameter and 24.6 meters tall. The vessel initially contained regions

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of subcooled liquid, saturated liquid and a steam dome. The assessment calculations against measured break flow and pressure are shown in Figure 3 and Figure 4. The conservatism of the Moody model in two-phase choked flow was demonstrated.

3.3 ECC Bypass Model

During the ECC bypass period, the emergency coolant will be held in the upper downcomer region, will accumulate in the inlet lines, and will then leave RCS through the break without taking decay heat from the reactor core, until the vapor flow from the core can no longer sustain the water. The downcomer flooding model derived from the UPTF full-scale test (Siemens, 1988) was applied to determine when the ECC water could penetrate the downcomer through the RELAP5-3D regular CCFL input process. The UPTF downcomer flooding model is:

$$j_g^{*1/2} + 2.193 j_f^{*1/2} = 0.6208 \tag{3.6}$$

According to the requirement, before the end of the bypass period all the injected ECC water needs to be removed from the system. To fulfill the ECC subtraction requirement, a set of time dependent junction and volume (TMDPJUN, TMDPVOL) would be connected to the cold leg of the broken loop close to the downcomer. Equal amount of injected ECC water will be forced to be on-line removed from the reactor system by this artificial set of TMDPJUN and TMDPVOL. The boron transport calculation of RELAP5-3D can indicate when the end of ECC bypass takes place. This boron model will trace the transport of the borated ECC water. Once the borated ECC water penetrates the downcomer and reaches the lower plenum, a signal of the end of ECC bypass will be generated and the ECC subtraction scheme via the TMDPJUN and TMDPVOL will be automatically terminated. The comparison of actual injected ECC water in the LOFT L2-5 (C.B. Davis, 1998) and the one

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calculated by the Appendix K model is shown in Figure 5.

3.4 Critical Heat Flux during Blowdown

Three correlations suggested by Appendix K of 10 CFR 50, B&W-2, Barnett and Hughes(modified Barnett), were implemented into the best estimate version of RELAP5-3D to cover the pressure range of interest. For the high-pressure range (P>10.34 MPa), B&W-2 was applied; for the medium pressure range (8.96 MPa>P>6.89MPa), Barnett correlation was applied; for the low-pressure range (P < 5 MPa), the modified Branett was adopted. For pressures between ranges, interpolation by pressure was applied to calculate the correspond CHF:

$$q_{CHF} = \frac{(P_H - P)q_{CHF_L} + (P - P_L)q_{CHF_H}}{P_H - P_L} \qquad (3.7)$$

where index H and L represent the high and low ends of the interpolation range. Rod bundle heat transfer tests (G.L.Yoder, 1982) performed in the Thermal-Hydraulic Test Facility (THTF) at Oak Ridge National Laboratory (ORNL) were used to assess the CHF model and film boiling heat transfer. These tests were performed using an 8 × 8 fuel bundle. The rod geometry was representative of 17 × 17 fuel bundles, and the full-length bundle was electrically heated and had uniform axial and radial profiles. Three tests were used for assessment the CHF calculation, which include tests 3.07.9B, 3.07.9N and 3.07.9W. The range of conditions during this test was representative of those expected during a large break LOCA. A typical comparison of the location first experiencing CHF is shown in Figure 6. It can be seen that the CHF location predicted by the EM models was conservatively lower.

3.5 Post-CHF Heat Transfer during Blowdown

Two correlations suggested by Appendix K of 10 CFR 50 were adopted to calculate film boiling and transition boiling heat transfer. For the stable film boiling,

Groeneveld 5.7 was applied, while the McDonough-Milich-King correlation was used for transition boiling heat transfer. Once CHF has occurred, the greater heat flux would be applied which were calculated by the either the film boiling or transition boiling correlations. As stated in Appendix K, the Groeneveld correlation shall not be used in the region near its low-pressure singularity. As suggested by INEEL (Richard R. Schultz, 1999), for high flow ($j_g^{*1/2} + j_f^{*1/2} > 1.36$ for up flow, $j_g^{*1/2} + j_f^{*1/2} > 3.5$ for downflow) if pressure is less than 1.38 MPa, the modified Dittus-Boelter correlation can be used to replace the Groeneveld correlation. If the core flow is not high, the modified Bromley correlation by Hsu with convection can be used to correct the low-pressure singularity. Typical assessments against THTF tests for film boiling heat transfer of the EM model are shown in Figure 7 and Figure 8. As for the assessment of transition boiling heat transfer, THTF transition test with power ramping (THTF-303.6AR) was adopted. A typical comparison is shown in Figure 9.

3.6 Prevention from Returning to Nucleate Boiling and Transition Boiling Heat Transfer prior to Reflood

As required by Appendix K, during the blowdown phase once CHF occurs, transition boiling and nucleate boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, unless the reflood phase of the transition has been entered. Assessment of the artificial prevention algorithm is shown in 10. This figure depicts the mode change with and without the prevention algorithm. It can be seen that nucleate boiling heat transfer was successfully prevented by the algorithm, which modifies the existing heat transfer logic.

3.7 Core Flow Distribution during Blowdown

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To fulfill the requirement of taking into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture, the feature of the cross flow junction of the RELAP5-3D would be applied. In cross flow junctions, the transverse momentum convection terms are neglected. Therefore, there is no transport of x-direction momentum due to the flow in the transverse direction. To assess the calculation of core flow distribution under flow partial blockage, two EPRI flow blockage tests (A. Tapucu, 1984) were adopted in which single-phase liquid and two-phase air/water were used for a range of blockages and flow conditions. The comparisons of the calculated channel pressure distribution for both blocked and unblocked channels of the two-phase test against measurements are shown in Figures 11 and 12.

3.8 Reflood Rate for PWRs

According to Appendix K of 10 CFR 50, the calculated carryover fraction and mass in bundle needs to be verified against applicable experimental data. In the existing PSI reflood model of RELAP5-3D, the modified Bestion correlation was used for interfacial drag in vertical bubbly-slug flow at pressures below 10 bars to replace the EPRI correlation. Above 20 bars the EPRI correlation was used. Between 10 and 20 bars the interfacial drag was interpolated. To assess the performance of the PSI model in the best estimate version of the RELAP5-3D, five FLECHT-SEASET tests (M. Loftus, 1980) (31504, 31203, 31302, 31805 and 33338) were adopted. For the first four forced reflood tests, the flooding rates ranged from 0.81 inch/s to 3.01 inch/s. As for the last gravity-driven reflood test, the flooding rate was up to 11.8 inch/s during the accumulator injection period. Typical assessments were shown in Figures 13 and 14. Through the assessments against five reflood tests, it was found

that the PSI model could predict the flooding rate reasonable well but with enough conservatism.

3.9 Refill and Reflood Heat Transfer for PWRs

During reflood phase, the RELAP5-3D PSI model was adopted to fulfill the Appendix K requirement for a flooding rate greater than 1 inch/sec with necessary modifications. In the PSI model, a modified Weisman correlation calculating the heat transfer to liquid and a modified Dittus-Boelter correlation calculating the heat transfer to vapor replace the Chen transition boiling correlation. As for film boiling, heat transfer to liquid uses the maximum of a film coefficient contributed by the modified Bromley correlation, and a Forslund-Rohsenow coefficient. In addition, radiation to droplets is added to the final film-boiling coefficient to liquid. The heat transfer to vapor for film boiling is the same as the one for transition boiling, which was calculated by the modified Dittus-Boelter ($h_{Ditt}\alpha_g$). As required by the Appendix K of 10 CFR 50, when the flooding rate is less than 1 inch/s, only steam cooling in the PSI model was allowed. Assessment calculations were performed to against the five FLECHT SEASET tests discussed in section (8). To bound the peak cladding temperature (PCT) span on each measured fuel rods at the same elevation, the calculated heat transfer coefficient calculated by the original PSI model was reduced by a factor of 0.6 for the flooding rate greater than 1 inch/sec to ensure reasonable conservatism. Typical comparison of the PCTs is shown in Figures 15. While the comparison of heat transfer coefficients is shown in Figures 16.

4. Conclusions

Although the best-estimate LOCA methodology can provide the greatest margin for the PCT evaluation during a LOCA, it takes more resources to develop. Instead, implementation of evaluation models required by Appendix K of 10 CFR 50 upon an advanced thermal-hydraulic platform can also gain significant margin on the PCT calculation but with less resources. The best estimate version of RELAP5-3D has been evaluated against the requirements of the Appendix K of 10 CFR 50, and ten major areas of the current RELAP5-3D were identified to be further assessed and/or modified, which included (1) fission product decay, (2) metal-water reaction rate, (3) discharge model, (4) ECC bypass during blowdown, (5) critical heat flux, (6) prevention to return to nucleate boiling and transition during blowdown, (7) post-CHF heat transfer during blowdown, (8) core flow distribution during blowndown, (9) reflood rate calculation for PWRs, and (10)refill and reflood heat transfer for PWRs.

All the above required models have been successfully implemented into RELAP5-3D and verified against associated separate-effect tests. The final package of the modified RELAP5-3D satisfying requirements set forth in the Appendix K of 10 CFR 50 for the LOCA evaluation is summarized in Table 2. To further assess the integral performance of the Appendix K version of the RELAP5-3D, another six sets (Table 3) of integral-effect experiments will be applied in the next step. Through the next assessment program, the integral conservatism of RELAP5-3D/K for the LOCA calculation will be quantified and assured to be ready for licensing application.

Nomenclature

r, R_o = radius and original radius of unreacted metal

t = time

$$B = \frac{10^{-6} * A}{2 * \rho_m^2}$$

A = pre-exponential factor, $29.5 * 10^6 (mg/cm^2)^2/sec$

ρ_m = metal density

$$G = \frac{\Delta E}{R}$$

ΔE = activation energy, 45.5 kcal/mole

R = gas constant, 1.987 cal/(mole)(°K)

T_s = oxide surface temperature

DRP1 = the depth the reaction has penetrated the cladding at the end of a time step

DRP = the depth the reaction has penetrated the cladding at the start of a time step

h_o = stagnation enthalpy

h_f = liquid enthalpy

h_f = vapor enthalpy

v_f = liquid velocity

v_g = vapor velocity

x = flow quality

p_o = stagnation pressure

s = entropy

j_g^* = dimensionless gas superficial velocity

j_f^* = dimensionless liquid superficial velocity

q_{CHF} = critical heat flux

p = pressure

p = pressure

Reference

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Table 1 Cases for Separate-Effect Assessments

Case	Phenomenon/Model	Applicable Appendix K Section
Cathcart oxidation data	metal-water reaction	I.A.5
Marviken Test 22	critical flow	I.C.1.ab
ORNL THTF Tests 3.07.9B 3.07.9N, 3.07.9W	critical heat flux	I.C.4
ORNL THTF Tests 3.07.9B 3.07.9N, 3.07.9W	film boiling	I.C.5
ORNL THTF Test 3.03.6AR	transition boiling	I.C.5
EPRI flow blockage Run 4 and Run 8	core blockage and cross flow	I.C.7.a
FLECHT-SEASET Tests 31504, 31203, 31302, 31805 and 33338	refill and reflood rates	I.D.3
FLECHT-SEASET Tests 31504, 31203, 31302, 31805 and 33338	refilled and reflood heat transfer	I.D.5

Table 2 Final Package of RELAP5-3D to Satisfy the Appendix K of 10 CFR 50

Model Requirement	RELAP5-3D Subroutines	Status
Fission Heat	rrkin & rkin	Apply the existing model of the code
Decay of Actinides	rkin	Apply the existing model of the code
Fission Product Decay	rrkin & rkin	*Change to 1971 ANS Standard Model*
Metal-Water Reaction Rate	qmwr	*Change to Baker-Just correlation*
Swell & Rupture of the Cladding and Fuel Rod Thermal Parameters	madata, gapcon, cplexp, ruplas, plstrn, kloss	Apply the existing model of the code
Discharge Model	jchoke	*Change to Moody model*
End of Blowdown	none	*Apply the CCFL model suggested by UPTF test along with on-line ECC water subtraction scheme*
Frictional Pressure Drops	fwdrag	Apply the existing model of the code
Momentum Equation Requirements	vexplt (semi-implicit)	Apply the existing model of the code
Critical Heat Flux	chfcal & chftab	*Change to B&W-2, Barnett, & modified Barnett correlations*
Prevent Return to Nucleate Boiling	htrcl	*Modify the existing heat transfer selection logic*
Post-CHF Heat Transfer Correlations: Film Boiling	pstdnb & suboil	*Change to Groeneveld 5.7, modified Dittus-Boelter, & modified Bromley correlations*
Post-CHF Heat Transfer Correlations:Transition Boiling	pstdnb	*Change to McDonough, Milich, & King correlations*
Prevent Return to Transition Boiling Heat Transfer Prior to Reflood	pstdnb	*Modify the existing heat transfer selection logic*
Pump Model	pump	Apply the existing model of the code
Core Flow Distribution During Blowdown		*Apply the existing cross-flow junction model of the code by assessments and apply core flow smoothing if necessary*
Calculation of Reflood Rate for PWRs	rhtcmp, htrcl, qfmove	*Applying the existing PSI model by performing separate-effect assessments*
Steam Interactions with ECC Water	eccmxj & eccmxv	Apply the existing model of the code
Refill and Reflood Heat Transfer for PWRs	rhtcmp, htrcl, qfmove	*Modify the existing PSI model*

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Table 3 Cases for Integral-Effect Assessments

| Case	L2-3	L2-5	L3-7	S-LH-1	IIST	L8-2
Break Size	200%	200%	0.1%	5%	0	23%
Break Location	Cold leg	Cold leg	Cold leg	Cold leg	None	Cold leg
Notes	RCP Running	RCP Tripped	Without Core Heatup	With Core Heatup	Natural Circulation	Restart of RCPs

List of Figures

Figure 1. (Thomas K.S. Liang, et al.)

Figure 2. (Thomas K.S. Liang, et al.)

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Figure 3. (Thomas K.S. Liang. et al.)
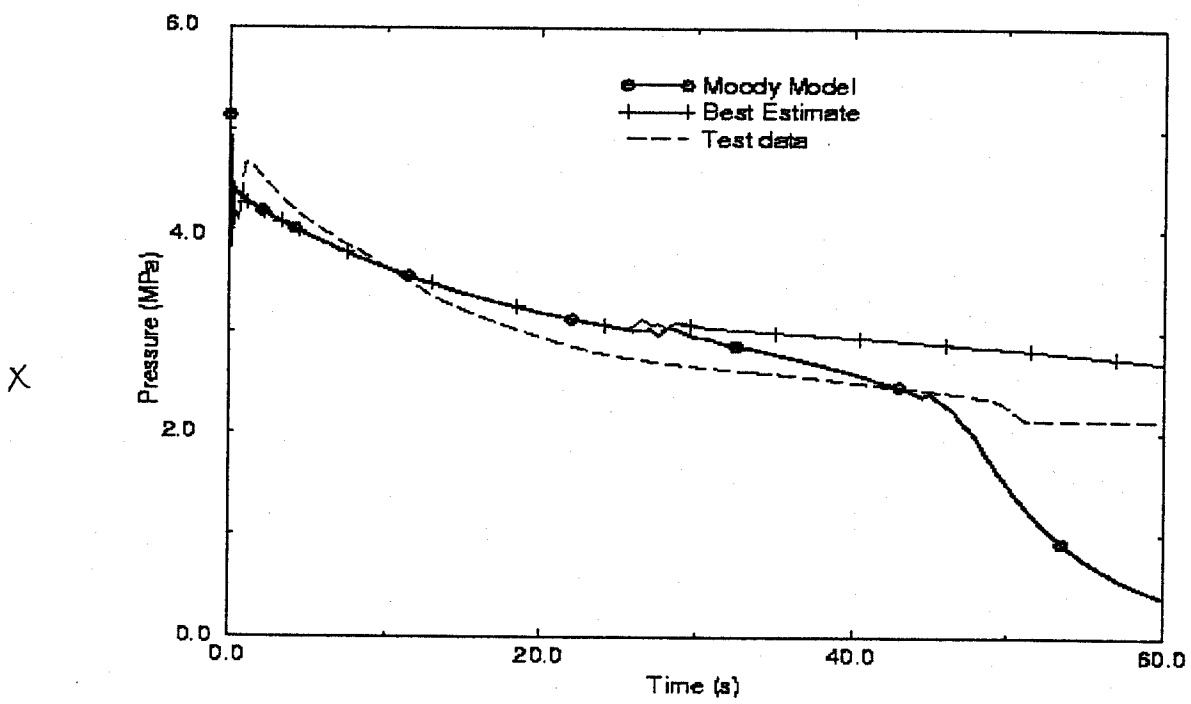
Figure 4. (Thomas K.S. Liang, et al.)

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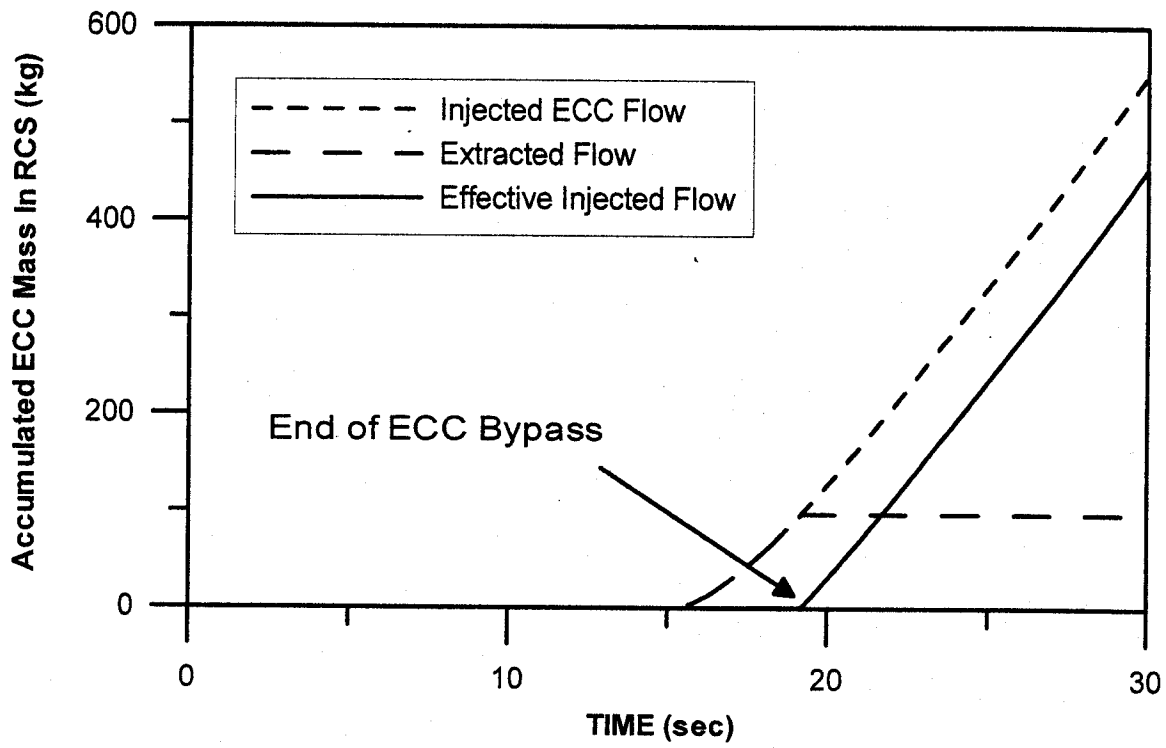
Figure 5. (Thomas K.S. Liang, et al.)
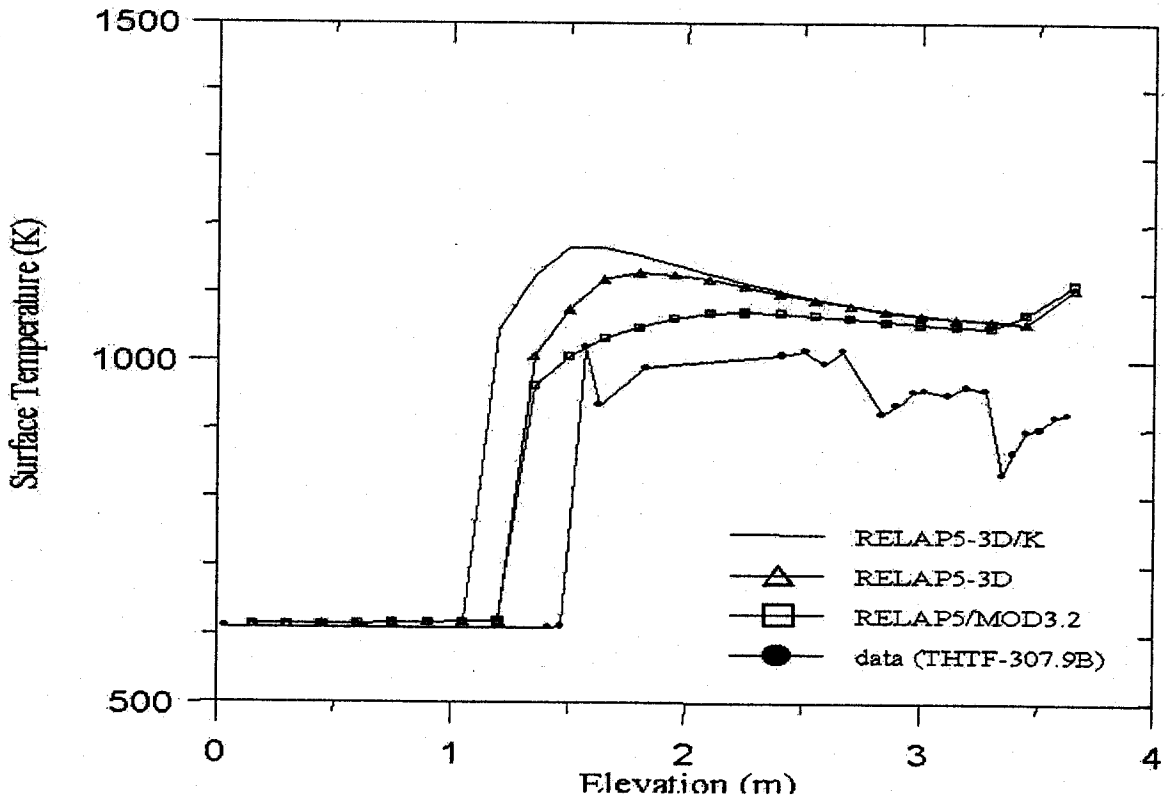
Figure 6. (Thomas K.S. Liang, et al.)
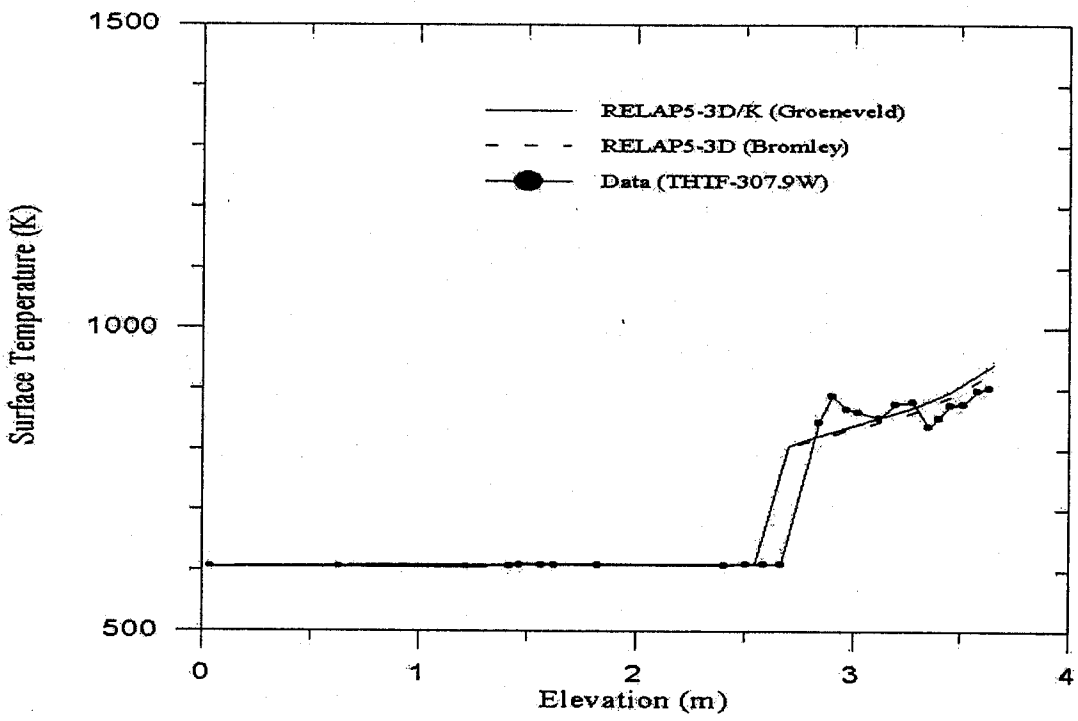
Figure 7. (Thomas K.S. Liang, et al.)
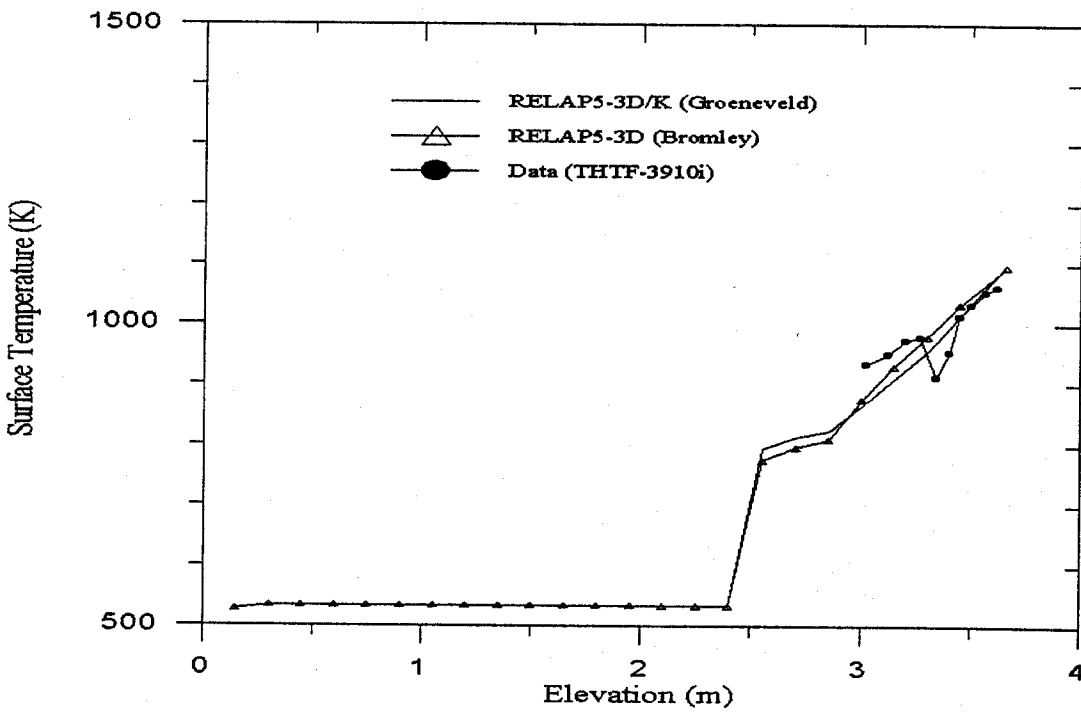
Figure 8. (Thomas K.S. Liang, et al.)

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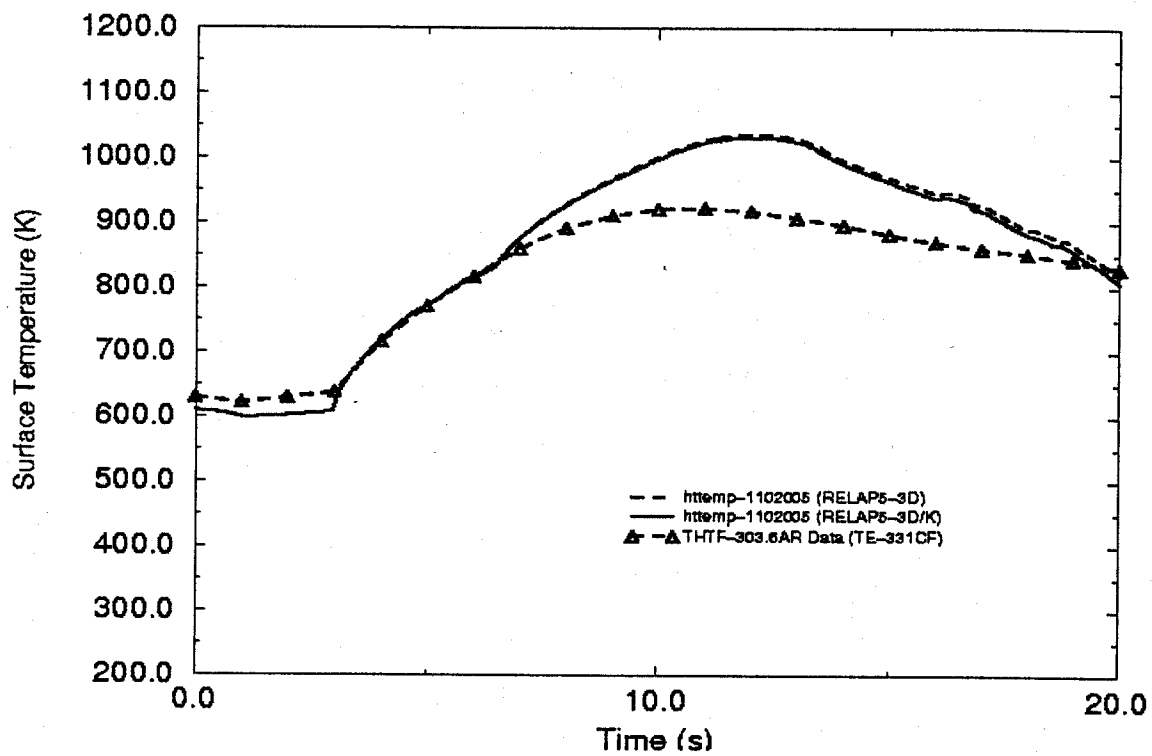
Figure 9. (Thomas K.S. Liang, et al.)
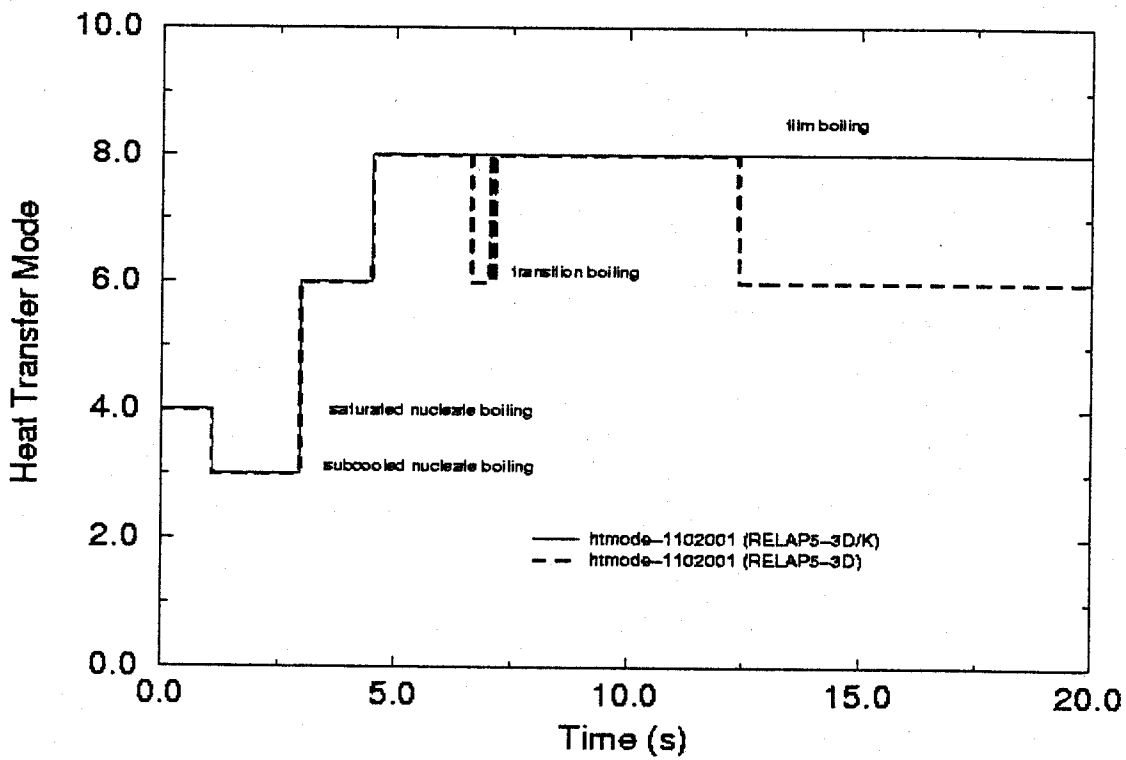
Figure 10. (Thomas K.S. Liang, et al.)

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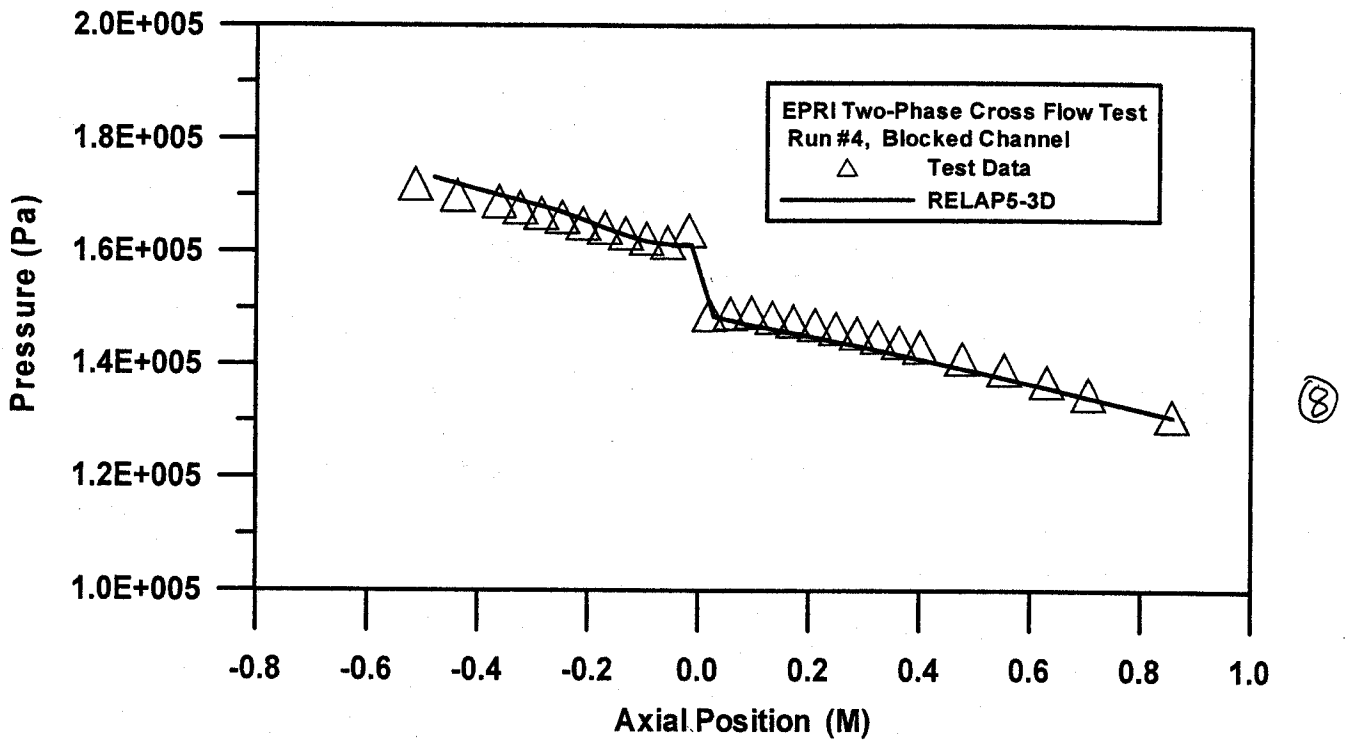
Figure 11. (Thomas K.S. Liang, et al.)
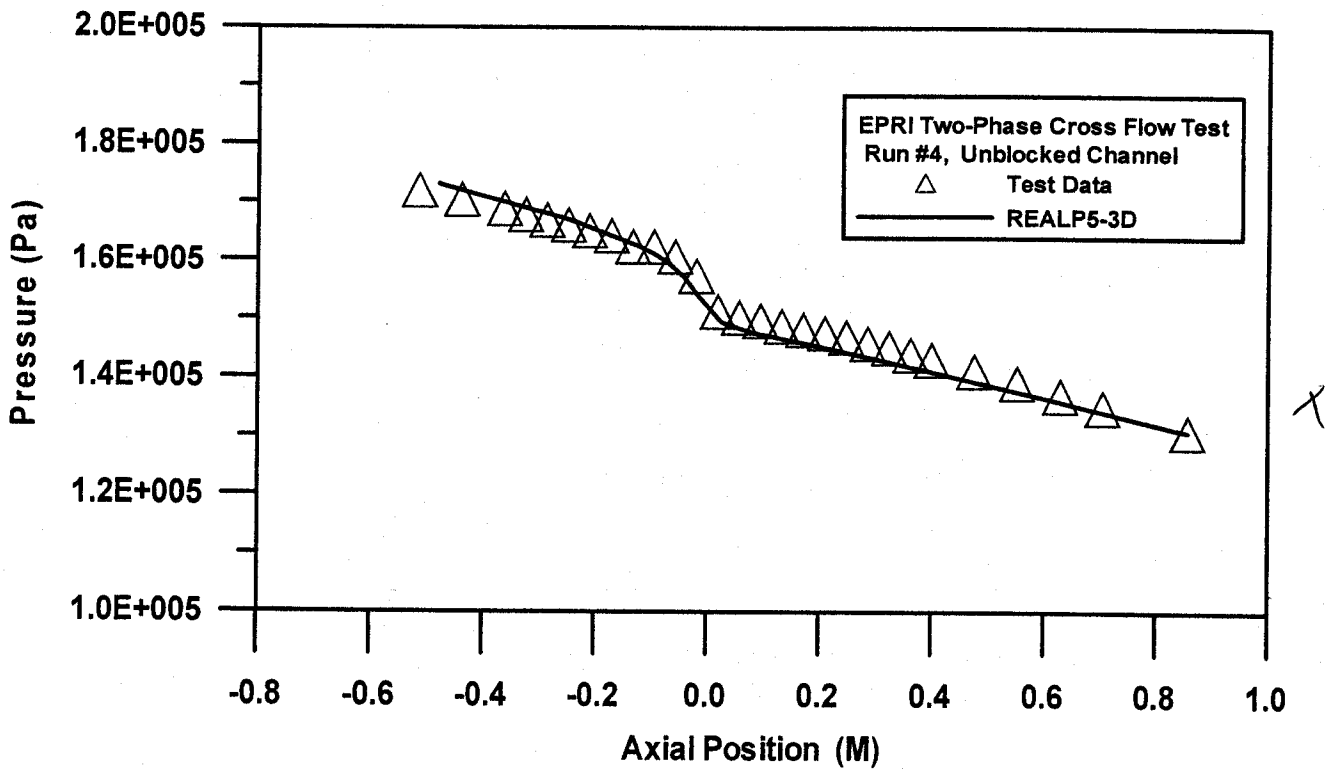
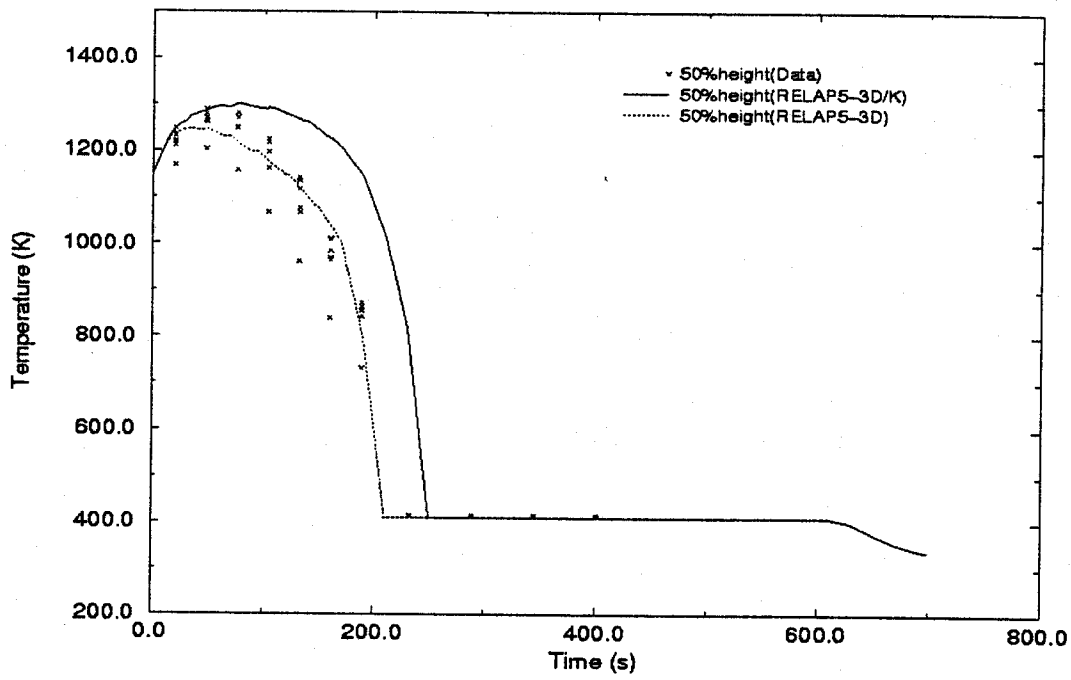Figure 12. (Thomas K.S. Liang, et al.)
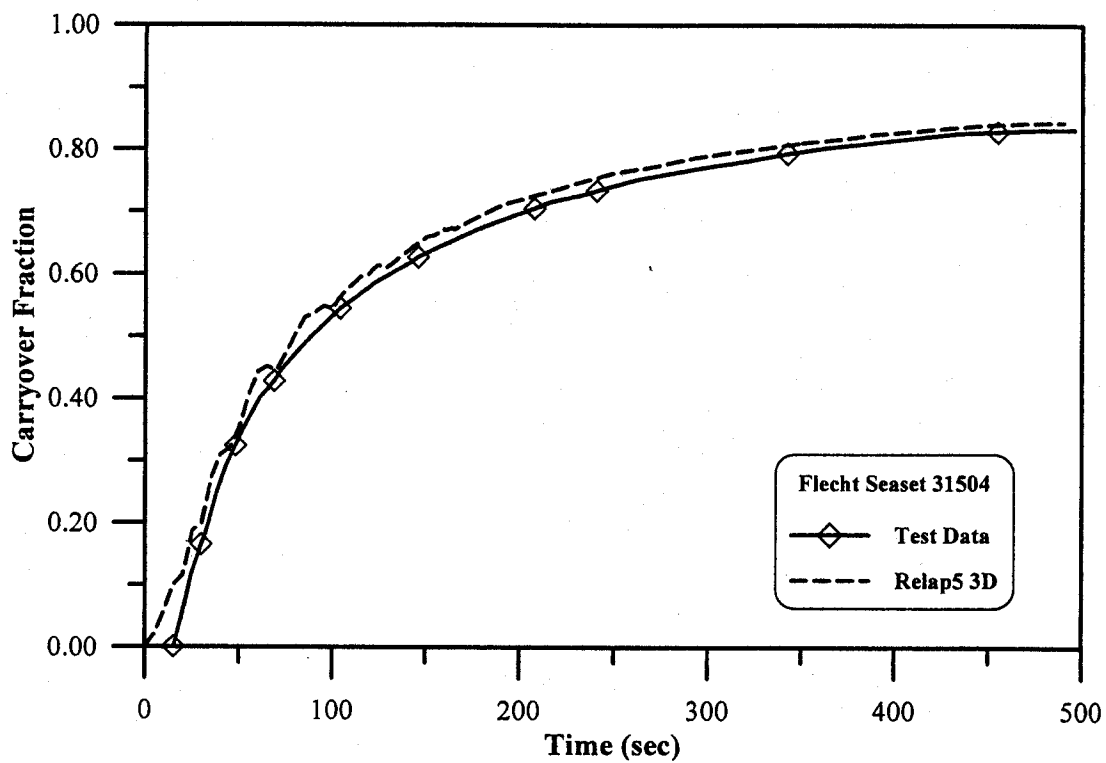
Figure 13. (Thomas K.S. Liang, et al.)
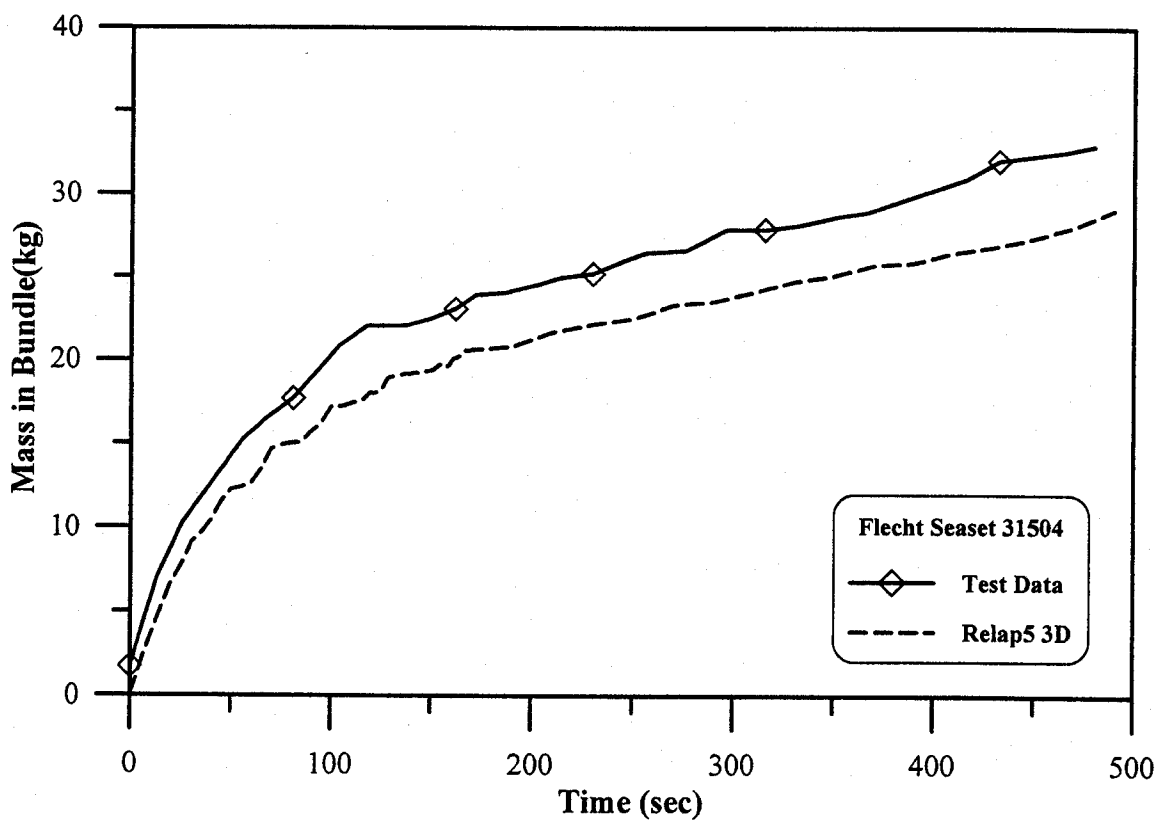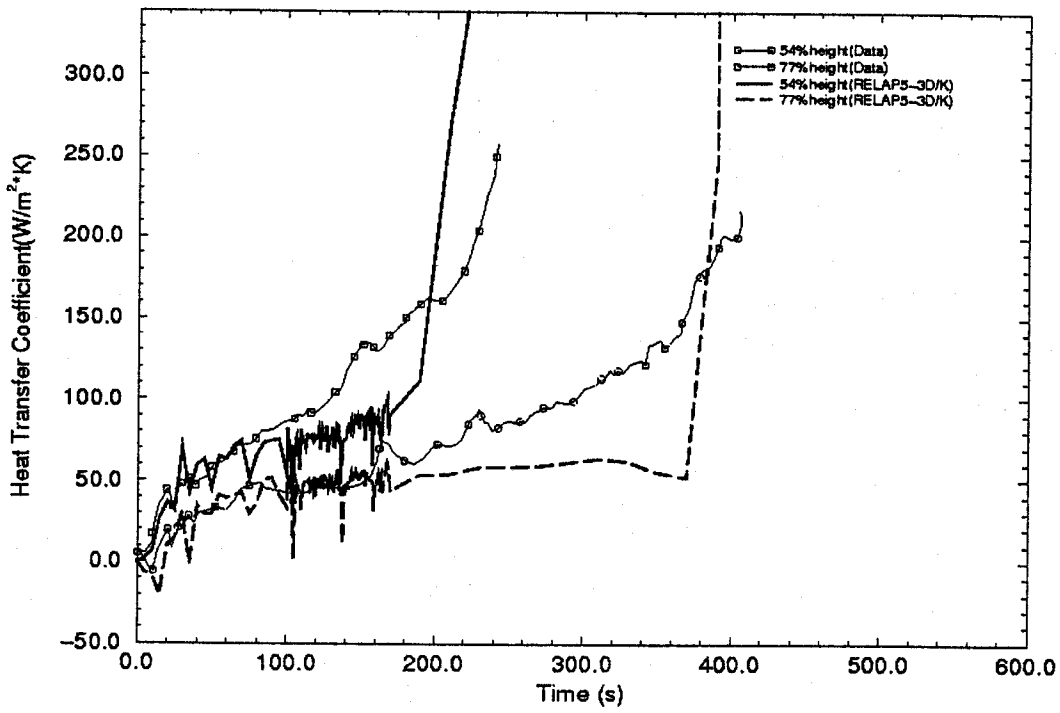

Figure 14. (Thomas K.S. Liang, et al.)

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Figure 15. (Thomas K.S. Liang, et al.)

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Figure 16. (Thomas K.S. Liang, et al.)