UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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MEMORANDUM FOR: Harold Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER #109 WRAP-BWR-EM (WATER REACTOR ANALYSIS PACKAGE - BOILING WATER REACTOR - EVALUATION MODEL)

I. INTRODUCTION: OVERVIEW OF WRAP-BWR-EM DEVELOPMENT

The WRAP-BWR-EM computer code system (Ref. 1) is designed to provide NRC with the capability to perform audit calculations for Loss of Coolant Accidents (LOCAs) in boiling water reactors. This code system is an outgrowth of, and replacement for, the original WREM system (Ref. 2) of Evaluation-Model-Audit codes. The work was performed in response to a request (Ref. 3) from the Office of Nuclear Reactor Regulation to provide a complete Evaluation Model (EM) code package for audit capability. A companion Research Information Letter (RIL) is being issued for the WRAP-PWR-EM system for pressurized water reactors.

The main thrust of the system development was to use existing computer codes, each code calculating a specific facet or portion of a BWR LOCA, and provide automatic data transfer and interfacing between those codes. Doing this allows the calculation to proceed smoothly through a complete LOCA sequence. Considerable effort also went into making the system format user convenient (Ref. 4) and also providing an automatic initialization for the LOCA transient (Ref. 5).

It is worth mentioning that the Office of Nuclear Regulatory Research (RES) management of the WRAP program at the Savannah River Laboratory (SRL) was aided by the successful interaction between SRL and other groups of workers. First, NRR personnel aided in the WRAP program from the beginning. Although they could not respond in writing to all requests for review of models and interfaces, their contributions were helpful (Ref. 6). Second, the check-out and verification of the completed WRAP-PWR-EM system was performed in parallel efforts by both Savannah River Laboratory (SRL) and the Idaho National Engineering Laboratory (INEL) (Ref. 7). The assistance that INEL provided to SRL throughout this project was very important. Third, the RISO Institute in Denmark at which one of the codes in the package was developed, provided assistance in resolving various problems associated with the installation of NORCOOL at SRL.

II. DESCRIPTION OF CALCULATION SCHEME

A. Codes Forming The WRAP Package

The WRAP system for BWR-EM analysis comprises several computer codes which have been developed to analyze individual phases of a LOCA. These codes include GAPCON (Ref. 8) for calculation of initial fuel conditions, TWRAM (the SRL analog of RELAP4/MOD5 (Ref. 9)) for analysis of the system blowdown, NORCOOL (Ref. 10) for analysis of the reflood phase, and MOXY (Ref. 11) for the calculation of the fuel assembly hot plane temperature.

The GAPCON module calculates preaccident thermal conditions of the fuel rods. For a given fuel rod, GAPCON determines the gap conductance, temperatures, pressures, and stored energy as a function of the power history of the rod. These data are then used as initial conditions for the transient fuel models.

GAPCON calculations can be performed as part of the WRAP-EM modular path or separately from the other calculations. In either procedure, GAPCON results may be stored in a data library. The stored GAPCON data are then used as input to WRAP and MOXY calculations. The GAPCON data transferred to WRAP require only two fuel rod descriptions--an average rod for the hottest fuel bundle and an average rod for the other fuel bundles.

The BWR system blowdown is calculated by the well-known RELAP4/MOD5 code (Ref. 9). The initial WRAP system was based on RELAP4/MOD5/Version 65. The initial step in the EM development required updating WRAP to RELAP4/MOD5/Version 74 and then implementing several modifications to provide an EM treatment of the BWR blowdown calculation. The latter included:

- o Vertical slip modeling modifications (Ref. 12) necessary to properly model gravity-induced velocity differences between liquid and gas phases,
- o Jet pump modeling modifications (Ref. 12) required to eliminate the discontinuity in the jet pump momentum equations that develops as either drive or suction flow becomes zero,
- o Addition of the GEXL correlation (Ref. 13) for determination of the boiling transition location, and
- o Corrections to the fuel rod plenum temperature calculations (Ref. 12).

Several other corrections (Ref. 14) were made which included the proper calculation of potential energy contributions to the junction enthalpy when flow reverses as well as other minor coding modifications.

The MOXY module is used to perform thermal analysis of a planar section of a BWR fuel assembly during a LOCA. The module calculates the temperature distribution of each fuel rod in a BWR fuel assembly at a single axial level of the assembly, usually the level with the hottest axial temperature. The code models describe heat transfer by conduction, convection, and radiation and heat generation by fission product decay and the metal-water reaction. Fuel-rod swelling and rupture are considered along with energy transfer across the fuel-cladding gap. Heat transfer in BWR assemblies is calculated during the three stages of a LOCA; blowdown, core heatup, and emergency cooling. Only during blowdown are the time-dependent data for power, heat transfer coefficient, and coolant temperature required. For the remainder of the transient, built-in data as required by 10 CFR 50, Appendix K (Ref. 15) guidelines for EM analysis are used.

The NORCOOL code (Ref. 10) was developed for the evaluation of emergency core cooling systems in BWRs. It is applicable to the regime following the termination of reactor blowdown where there is near-pressure equilibrium between the reactor vessel and the containment. The code is used to analyze operation of the spray and reflood systems through the refill and reflood phases.

NORCOOL consists of two basic models: a fuel-rod model and a model for two-phase flow. One-dimensional heat conduction equations are solved by the fuel rod model. The two-phase flow model is based on a solution of the multifield conservation equations for mass, momentum, and energy as well as the equation of state. The flow regimes covered are single-phase liquid, bubbly flow, inverse annular flow, film flow, and dispersed flow. Thermodynamic equilibrium is not assumed; i.e., steam is allowed to be superheated and water subcooled. The fuel-rod model and the two-phase flow models are coupled through a number of physical models and heat transfer correlations which include conduction, convection, and thermal radiation effects. Two-dimensional axial conduction is treated in the rewetting front.

B. Code Interfaces For Data Transfer

The integration of the GAPCON, RELAP, MOXY, and NORCOOL modules to form the WRAP-EM system required the defining and programming of the following interfaces:

- GAPCON/RELAP,
- GAPCON/MOXY,
- RELAP/MOXY,
- RELAP/NORCOOL, and
- NORCOOL/MOXY.

These interfaces automate the computational steps required to perform a complete LOCA analysis from break through reflood.

In general, hot assembly and average assembly fuel-pin conditions as calculated by GAPCON are transferred to WRAP via the GAPCON/RELAP interface. The fuel-pin conditions for the hot assembly are also transferred to MOXY via the GAPCON/MOXY interface. The RELAP/NORCOOL interface is a transfer of the data from RELAP at end of blowdown to NORCOOL for initialization of the reflood calculation. Transient hot assembly data during blowdown and reflood are passed to MOXY via the RELAP/MOXY and NORCOOL/MOXY interfaces, respectively. A more detailed description of the data transferred by each interface is presented in Appendix I and Reference 4.

C. BWR Steady-State Procedure

The RELAP4/MOD5 code provides no explicit procedure for initializing the transient thermal-hydraulic calculation. Instead, the user is required to generate the initial-system state by a series of hand calculations to produce estimates of the state variables and then short transient runs to evaluate the reasonableness of the estimates.

In the extension of the system for BWR-EM analysis, an automatic steady-state procedure has been developed for boiling water reactors. The BWR steady-state procedure employs a time-dependent solution of the thermal-hydraulic equations without perturbation. Instead of specifying the volume variables and junction flows required by RELAP, the WRAP user specifies the following quantities:

- Core power,
- Total water mass in the system,
- Steam dome specific volume, and
- Feedwater junction specific enthalpies.

The procedure then computes the:

- Thermodynamic state of all control volumes, and
- Flow rates for all junctions including the feedwater and steam line.

The uniqueness of the solution obtained by the BWR steady-state procedure has been demonstrated. The procedure requires approximately 15 to 20 minutes computer (cpu) time on an IBM 360 Model 195 for a typical BWR nodalization. A detailed description of the procedure is provided in Reference 5.

D. WRAP-BWR-EM Analysis Sequence

The BWR-EM analysis scheme is shown in Figure 1. The calculation proceeds along two paths, one for thermal/hydraulic analysis and the other for fuel rod response analysis. This scheme has been reviewed by NRR personnel (Ref. 16).

At selected times during the calculation, information is automatically passed from the thermal/hydraulic analysis to the fuel rod response analysis. These include fuel thermal conditions for the hot plane analysis by the MOXY module and the times at which the following events occur during blowdown:

1. Time of end of lower plenum flashing,

2. Time when core sprays are at rated flow,

3. Time for end of critical flow at the break. The above times are used in applying the 10 CFR 50, Appendix K rule (Ref. 15) for specifying convective heat transfer coefficients for BWR fuel under spray cooling.

At the end of blowdown, system renodalization is performed by the RELAP/NORCOOL interface routine and the reflood phase of the accident is calculated by the NORCOOL module. The time to hot-plane quench is passed to MOXY for use in determining the end time for the hot-plane analysis.

III. WRAP-BWR-EM VERIFICATION RESULTS

Calculations to verify the capability of the WRAP-BWR-EM system were performed and compared with test data as well as with vendor calculations. All calculations, except the special MOXY/GE comparison, were performed at SRL with automatic interfacing between the separate codes, and with the same code versions at INEL (Ref. 7) with hand interfacing between codes. The latter procedure was used to uncover and correct coding and input errors.

MOXY calculations were compared to a General Electric (GE) EM analysis for a BWR/6 LOCA (Ref. 17). Data supplied by NRC for a BWR/6-218 reference study were used for the MOXY calculations. Data used for the GE calculations were not completely specified (Ref. 17). Nevertheless, there was good general agreement between the MOXY results and the GE results. MOXY predicted essentially the same time-dependent behavior for the peak clad temperature (PCT) as did the GE methods. For the input values used in this study, all values of PCT predicted by MOXY were within 7 percent of the GE value. However, sensitivity studies performed with MOXY showed variations in PCT as large as 200°F, depending on values of input data which were estimated from plots in the GE report. There was general agreement with the GE results for the time-dependent behavior of the average fuel temperature and the fuel-pin pressure. However, MOXY predicted much higher initial average fuel temperatures and fuel-plenum gas pressures than reported in the GE work. This was probably due to differences in the initial fuel conditions such as power levels and distributions, cold gas pressure, and time in reactor. The objective of this study was not so much to exactly duplicate the GE results, but to demonstrate that MOXY could calculate the same general fuel-temperature behavior; this demonstration was achieved.

As part of the checkout program of the WRAP-BWR-EM system, companion calculations were conducted at INEL and SRL for a pump suction line break in a BWR/4-type plant (Ref. 18). The Hope Creek plant was used for the reactor model. RELAP4/MOD5 and MOXY input were created at INEL. These input cards were converted to JOSHUA records suitable for use by the WRAP system of codes at SRL. The results of the SRL blowdown calculations with the WRAP module TWRAM were in excellent agreement with the reported RELAP4/MOD5 blowdown calculations conducted at INEL. Good agreement also was obtained between SRL-MOXY and INEL-MOXY calculations of the fuel heatup following the beginning of lower plenum flashing.

A verification calculation was also performed to demonstrate the application of the package to the analysis of a loss-of-coolant accident for a General Electric BWR/6-218 design (Ref. 19). The problem selected by NRC was a double-ended guillotine break of the pump suction line of a recirculation loop. This same calculation is the base case large break for a BWR/6 EM reference study currently being supported at SRL by NRR. The WRAP results were compared to an analogous GE calculation and the ongoing NRR study will perform parametric calculations to investigate the differences in the WRAP and GE calculations.

The nodalization of the reactor system used in this study (Ref. 19), as well as many of the model options employed were originally developed at INEL as part of a best-estimate study. Appropriate modifications were made to create an EM problem. Representative fuel data were obtained from NRC (See Ref. 19). All computational modules in the WRAP-BWR-EM system were exercised. A steady state was not computed using the BWR steady-state module in WRAP since INEL had already established steady-state input parameters. However, a supplementary calculation with WRAP yielded a steady-state very similar to the INEL input.

The reasonable comparison between the WRAP and GE calculated pressures is shown in Figure 2. However, the clad temperature histories in the WRAP and GE calculations were quite different, as shown in Figure 3. GE calculated departure from nucleate boiling (DNB) at the high-powered plane at 10 sec; WRAP did not calculate DNB at that location until 48 sec. The cause is probably a combination of two effects: 1) the predicted core flow during the first 10 sec of the transient, and 2) the use by GE of the GEXL correlation which is not applicable to low or reverse flow situations. More detail is given in Appendix II. As mentioned above, an ongoing parametric study is investigating these discrepancies, as part of the audit function of WRAP-BWR-EM.

Finally, WRAP-BWR-EM calculations were compared with data from two TLTA tests: 6007 (no ECC) and 6406 (with ECC), as presented in Reference 20.

These two tests were chosen for simulation with WRAP because the measured pressure decay was faster without ECC (6007) than with ECC (6400), which was the opposite of what was expected. The WRAP calculations predicted a faster depressurization for Test 6007 than for Test 6406, in agreement with the GE test results (Fig. 4). WRAP satisfactorily calculated both the TLTA experiments, within the constraints of the EM specifications. The Moody critical flow model with a discharge coefficicent $C_D = 1.0$ gave higher break flows and therefore more rapid depressurization than measured. The WRAP calculated peak clad temperatures were above the measured data and showed the same general behavior toward the end of blowdown. At early times in the transient, WRAP predicted anomalously high clad temperatures because of the Barnett and modified Barnett critical heat flux (CHF) correlations specified in the INEL input. A short run with the Hench-Levy (GE) correlation gave more reasonable results and this correlation is recommended for future calculations.

IV. RECOMMENDATION FOR WRAP-BWR-EM USE BY NRR

Development and verificaton of the WRAP-BWR-EM system is now complete and it is recommended for use by NRR. In early CY 80, the WRAP system was made operational on the Harry Diamond Laboratory computer, located in a Maryland suburb of Washington, and a special remote terminal was installed in Bethesda so that NRR personnel could perform their own calculations with WRAP. An indoctrination class in the use of WRAP was also held in Bethesda by SRL personnel in early CY 80.

Currently, most of the NRR use of WRAP is being performed at SRL. These include a BWR-LOCA sensitivity study and other audit calculations, funded by NRR. RES intends to supply maintenance of the WRAP system over the next few years, to improve modeling and to solve any coding problems that are identified by NRR use.

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Enclosures:
1. Appendices I and II
2. Figures 1 through 4

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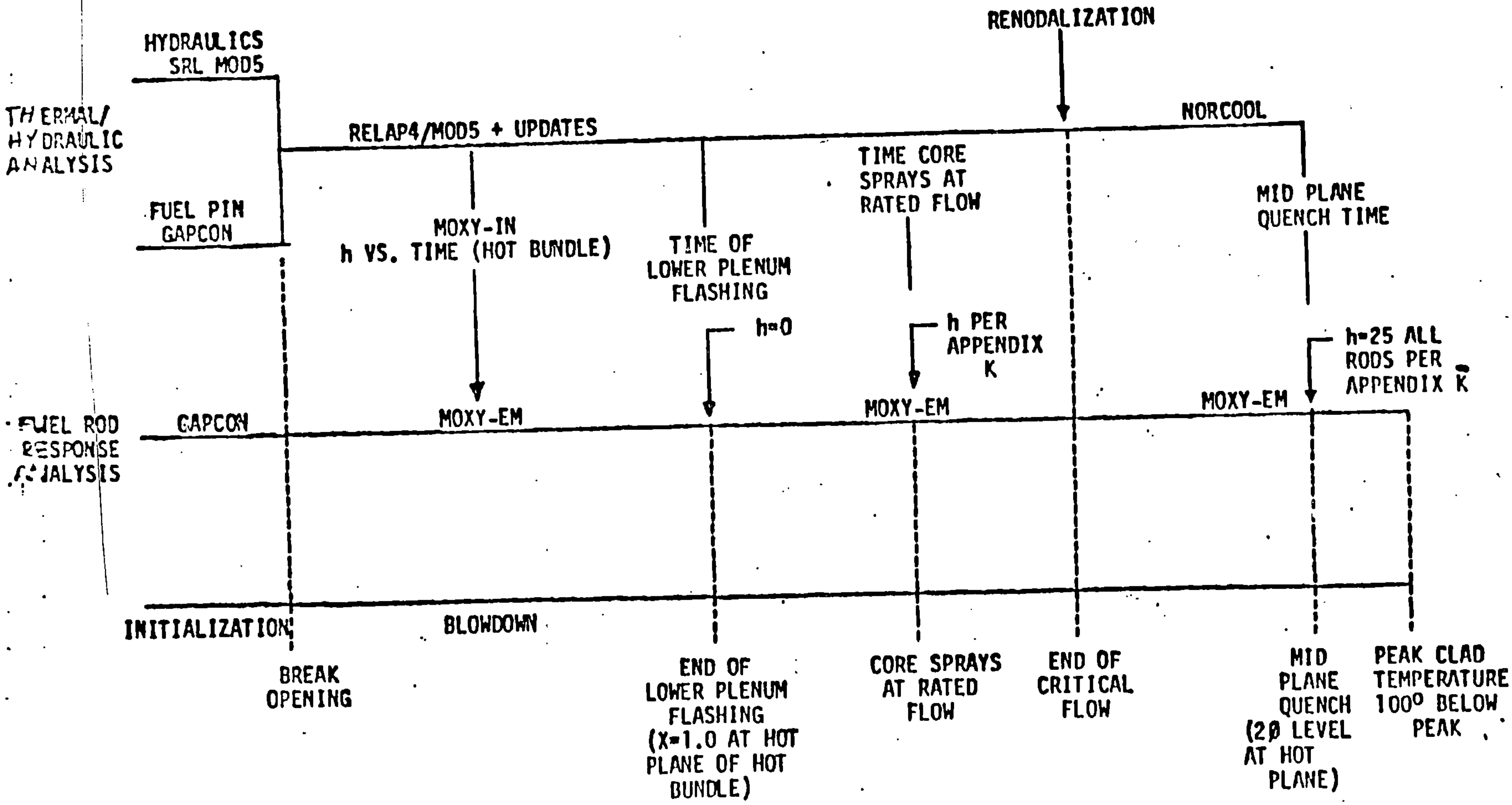
FIGURE 1

BWR ANALYSIS SCHEME



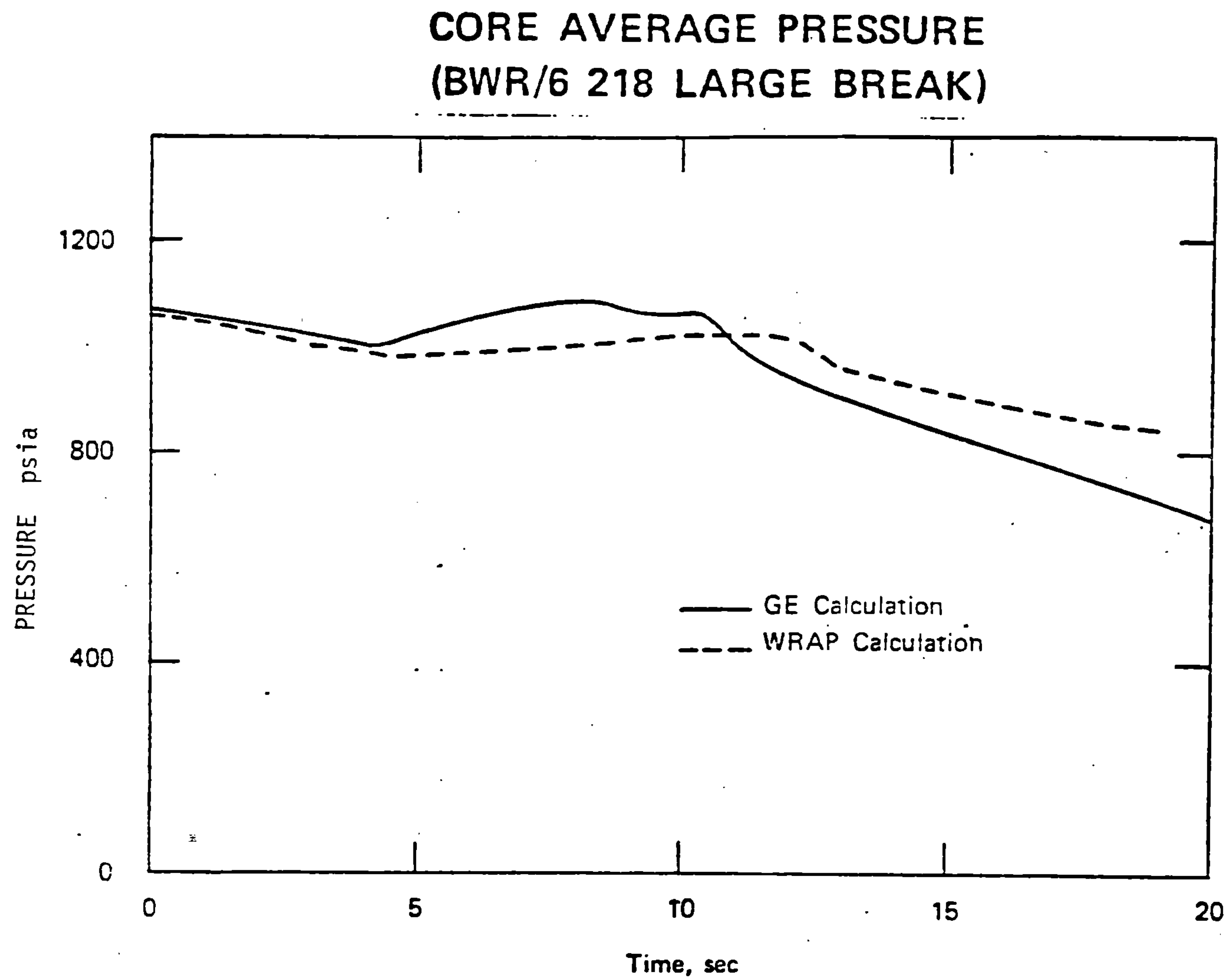

CORE AVERAGE PRESSURE
(BWR/6 218 LARGE BREAK)
1200
800
400
0
PRESSURE psia
0
5
10
15
20
Time, sec
GE Calculation
WRAP Calculation

Figure 2

PEAK CLAD TEMPERATURE: GE vs. WRAP

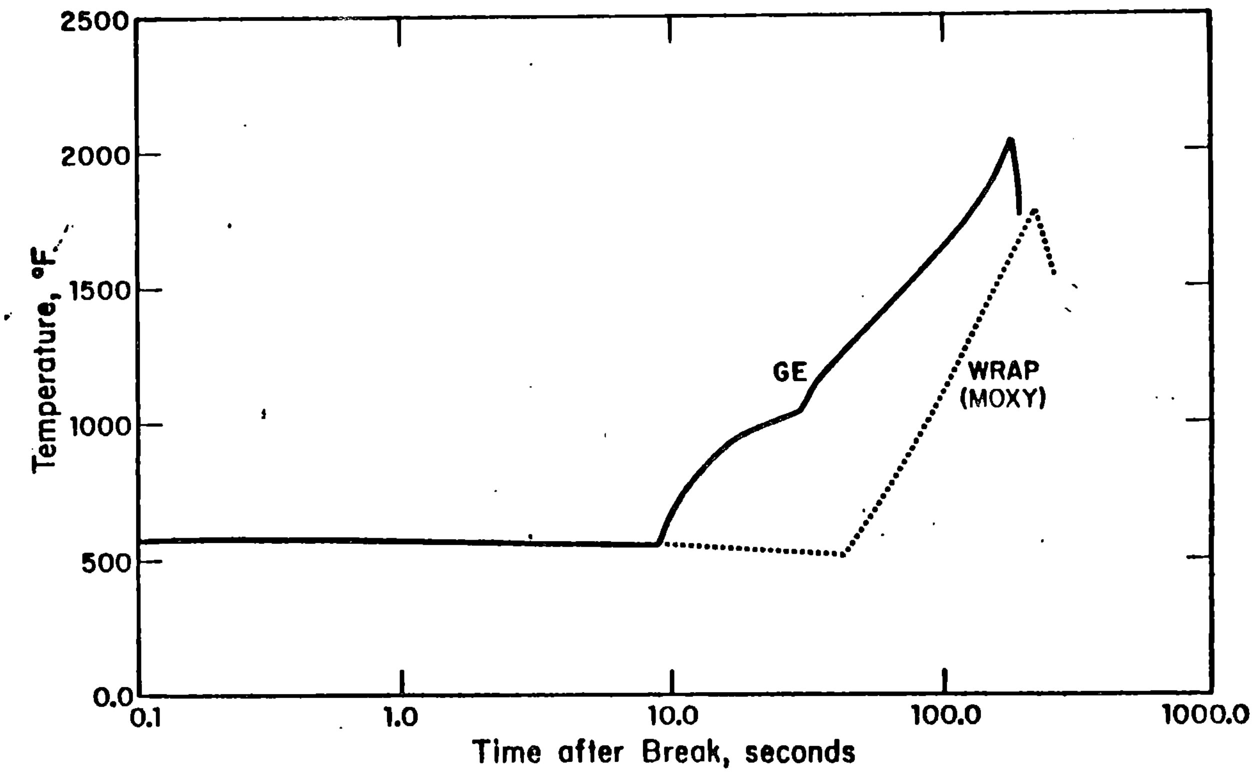
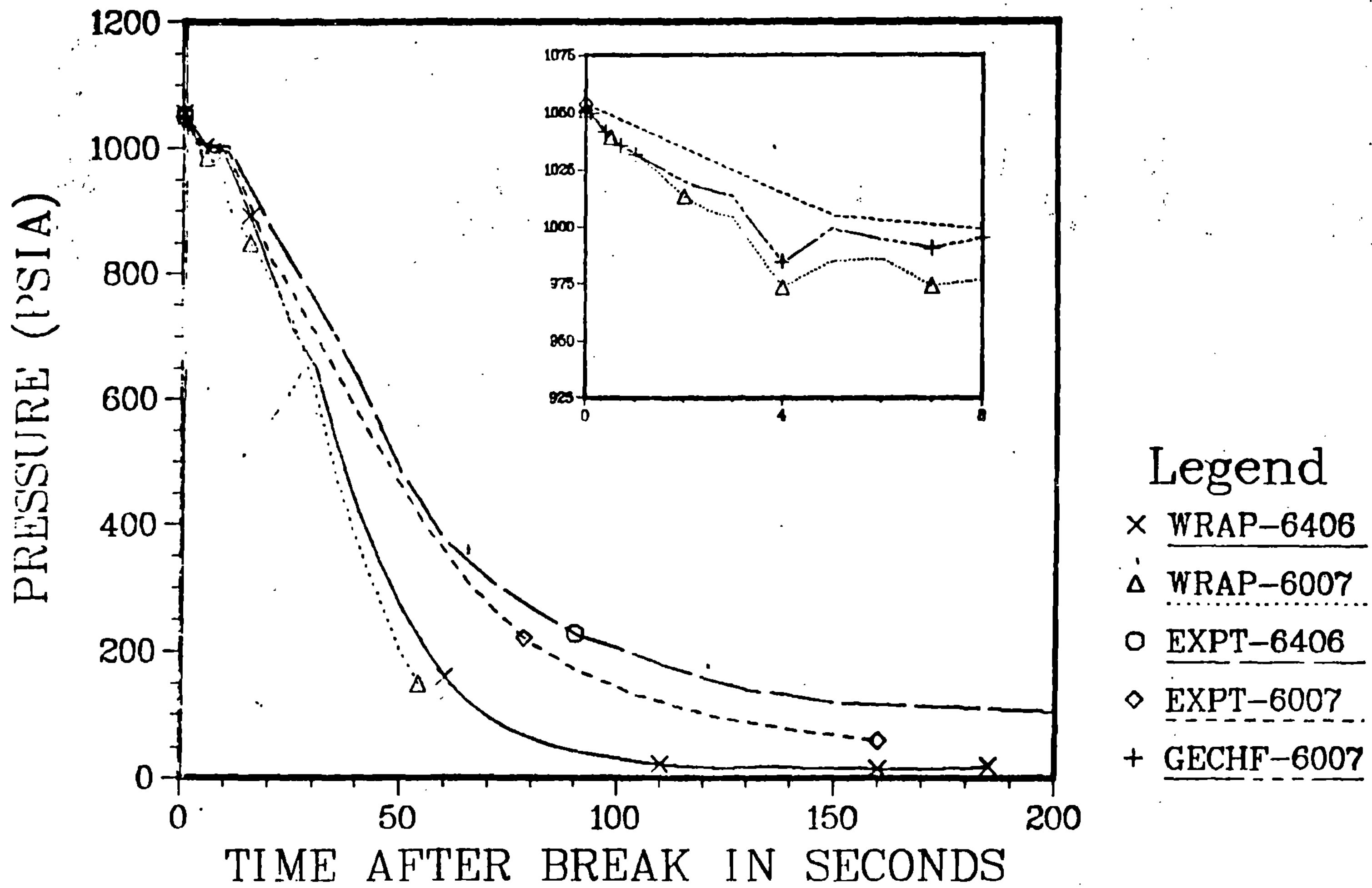


Figure 3

FIGURE 4
STEAM DOME PRESSURE



APPENDIX I: DESCRIPTION OF CODE INTERFACES

A. GAPCON/WRAP

In the WRAP-EM computational system, GAPCON is used to determine fuel rod conditions at the beginning of the LOCA analysis. These initial conditions are functions of power level, burnup, fill gas pressure, etc. The data from the GAPCON calculation are transferred via the GAPCON/RELAP interface to RELAP which calculates the blowdown phase of the LOCA.

The data transfer between GAPCON and RELAP is not straightforward because the fuel models in the two codes differ. For example, GAPCON models a single fuel pin allowing up to 20 axial nodes for detail. RELAP, on the other hand, models the complete core as one or two (or possibly three) stacks of heat slabs. One stack may represent the hot bundle; while, the other models the remainder of the core. Thus, the data transferred from GAPCON to RELAP must be collected, sorted, and interpolated by the interface before being used in RELAP.

Data transferred between GAPCON and RELAP is specified for a given pin(s). For example, hot channel and average channel heat slab conditions may be determined by "typical" hot rod and "typical" average fuel rod calculations done with GAPCON. In the interface, the GAPCON "typical" pin data may be scaled by the number of fuel rods per stack to obtain RELAP heat slab data. The data transferred by the interface includes the fuel rod geometry (before and after burnup), power, fuel density, and gap heat transfer factors including fission gas composition and gm-moles of fission gas. Where GAPCON data differs as a function of axial location, the RELAP heat slab data are linearly interpolated from the GAPCON data.

B. GAPCON/MOXY

Initial fuel conditions as calculated by GAPCON are passed automatically to MOXY by the GAPCON/MOXY interface. As with the GAPCON/RELAP interface, the data transfer is not straightforward because of the model differences between GAPCON and MOXY.

GAPCON models in detail a single fuel pin as a function of power level, burnup, fill gas pressure, etc. GAPCON allows up to 20 axial nodes. On the other hand, a MOXY calculation is based on a horizontal plane in a fuel assembly where the assembly contains a 7x7 or 8x8 square fuel pin array. Diagonal symmetry is assumed in the fuel array, but each pin in the remaining half assembly may differ in power and thermal characteristics. A GAPCON calculation for each pin of the MOXY model is required to properly initialize the MOXY pin array. In the case of a 7x7 fuel assembly, this requires 28 GAPCON calculations, and for the 8x8 assembly, 36 calculations for a given operating condition.

Since MOXY models a planar section of the assembly, the data transferred from GAPCON to MOXY are data for the hottest axial level. The data transfer includes a check to ensure that the hottest axial node for each GAPCON calculation is at the same elevation. The data transferred via the GAPCON/MOXY interface includes the fuel rod geometry (before and after burnup), power, fuel density, and gap heat transfer factors including fission gas composition, gm-moles of fission gas, and temperature jump distance.

C. RELAP/MOXY

Transient hot assembly data during blowndown are transferred from RELAP to MOXY via the RELAP/MOXY interface. RELAP describes the transient response of the reactor through blowdown during a postulated LOCA. RELAP typically models the complete core as one or two stacks of heat slabs. One stack may represent the hot assembly; the other the remainder of the core. MOXY, on the other hand, is used to analyze a planar section of a fuel assembly during a LOCA. Normally, MOXY, is used to analyze the hot plane in the hot assembly. Thus, data for the hottest core heat slab are passed to MOXY.

The RELAP/MOXY interface provides MOXY with input of time-dependent data for the normalized power, for the cladding-to-coolant heat transfer coefficient, and for the fluid temperature during the blowdown portion of the transient. Other quantities passed to MOXY include the times at which the following events occur in the blowdown calculation:

- o End of lower plenum flashing which is defined as the time at which dryout of the hot bundle occurs,
- o Core sprays at rated flow, and
- o End of critical flow at the break.

The above times are used in applying the 10 CFR 50, Appendix K (Ref. 15) rule for specifying convective heat transfer coefficients for BWR fuel under spray cooling.

D. RELAP/NORCOOL

Most of the input data required for NORCOOL is available directly (or may be calculated) from the data base created by execution of RELAP. The correspondence between RELAP and NORCOOL variables is provided in Table III of Reference 1.

The interface module INTNOR carries out the transfer of data from the RELAP data base into templated input records for NORCOOL. Various algorithms are built into module INTNOR. A renodalization algorithm allows the user to subdivide a RELAP volume into several identical NORCOOL nodes merely by identifying the RELAP volume name for a series of NORCOOL nodes. Each of the nodes is assigned the same state variables characteristic of the original volume. Since the nodalization of the diffuser region in NORCOOL may not be dimensionally consistent with the RELAP nodalization, several correction operations are built into the code: an equal-mesh transition zone may be created between the upper downcomer and the top of the diffuser; the top of the steam separators is equalized with the bottom of the steam dome. Nodes in the core and bypass regions are paralleled, as are those in the lower plenum and the region below the diffuser.

In a typical LOCA calculation, NORCOOL is executed when end of critical flow at the break occurs in the blowdown. At that time, the void fractions are essentially unity everywhere in the system except in the nodes close to the spray injection point and in the lower plenum. To model this situation, the interface defines a pseudo-water level at the bottom of the lower plenum, by volume averaging and collapsing void fractions required for NORCOOL to function properly. The void fraction is defined as zero below the pseudo level and unity above it. In all other nodes, the true void fraction is extracted from the data base.

E. NORCOOL/MOXY

The data interface between NORCOOL and MOXY consists of the following quantities: the hottest plane number, its elevation from the core bottom, and the time at which the quench front in the core reached the hottest plane. The latter time is used in MOXY to complete the specification of convective heat transfer coefficients according to Appendix K rules.

APPENDIX II: COMPARISON OF GE AND WRAP CLAD TEMPERATURE FOR BWR/6 LOCA

The difference in peak-clad temperature between WRAP and GE is shown in Figure 3. The obvious deviation at 10 seconds indicates that GE calculates departure from nucleate boiling (DNB) at the highest powered plane earlier than WRAP. In addition, lower plenum flashing does not seem to rewet the plane, indicating the presence of a heat transfer lockout in their fuel analysis program. The WRAP (MOXY) analysis shows that the highest powered plane remains in nucleate boiling until well after lower plenum flashing. DNB is not reached until 48 seconds at which time the hot plane uncovers and remains uncovered until quenching at 216 seconds predicted by NORCOOL. The difference in temperatures at early times (∿10 seconds) is thought to be due to core flow discrepancies. WRAP computes a core flow spike at about 9 seconds while the GE analysis shows flow stagnation and reversal. This flow difference will cause a considerable variation in the heat transfer and it occurs just before the GE analysis indicates DNB. The reverse flow seen by GE is particularly important since it is not clear how it affects the critical heat flux correlation they use. GE employs the General Electrical Critical Quality (X_c) - Boiling Length Correlation (GEXL) which is apparently invalid at low mass flow rates ($<1 \times 10^5$ lbm/hr-ft^2) and reverse flow. Thus, their calculated CHF during these flow regimes may be very low.

The WRAP calculation used the Hench-Levy CHF correlation since the input data required for the GEXL correlation were not available at that time. A subsequent WRAP calculation employing the GEXL correlation (using 3.0-inch heat slabs as recommended by NRC) indicated that the heat flux at the highest powered slab remained well under the CHF through the 10 reactor seconds that the problem was executed.

These two tests were chosen for simulation with WRAP because the measured pressure decay was faster without ECC (6007) than with ECC (6400), which was the opposite of what was expected. The WRAP calculations predicted a faster depressurization for Test 6007 than for Test 6406, in agreement with the GE test results (Fig. 4). WRAP satisfactorily calculated both the TLTA experiments, within the constraints of the EM specifications. The Moody critical flow model with a discharge coefficicent $C_D = 1.0$ gave higher break flows and therefore more rapid depressurization than measured. The WRAP calculated peak clad temperatures were above the measured data and showed the same general behavior toward the end of blowdown. At early times in the transient, WRAP predicted anomalously high clad temperatures because of the Barnett and modified Barnett critical heat flux (CHF) correlations specified in the INEL input. A short run with the Hench-Levy (GE) correlation gave more reasonable results and this correlation is recommended for future calculations.

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Enclosures:
1. Appendices I and II
2. Figures 1 through 4

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CRESS:SS	RSR:ADB	RSR:ADB	RSR	RES	RES	RES
1SHOTKIN:Job	LShotkin:mw	SFabic	LSTong	TEMurley	JLarkins	RBMinogue
11/21/80	11/ /80	11/ /80	11/ /80	11/ /80	11/ /80 1/5/81	1/6/80

See concurrence, next page

These two tests were chosen for simulation with WRAP because the measured pressure decay was faster without ECC (6007) than with ECC (6400), which was the opposite of what was expected. The WRAP calculations predicted a faster depressurization for Test 6007 than for Test 6406, in agreement with the ~~General Electric (~~GE~~)~~ test results (Fig. 4). WRAP satisfactorily calculated both the TLTA experiments, within the constraints of the EM specifications. The Moody critical flow model with a discharge coefficicent $C_D = 1.0$ gave higher break flows and, therefore, more rapid depressurization than measured. The WRAP calculated peak clad temperatures were above the measured data and showed the same general behavior toward the end of blowdown. At early times in the transient, WRAP predicted anomalously high clad temperatures because of the Barnett and modified Barnett (CHF) correlations specified in the INEL input. A short run with the Hench-Levy (GE) correlation gave more reasonable results and this correlation is recommended for future calculations.

IV. RECOMMENDATION FOR WRAP-BWR-EM USE BY NRR

Development and verificaton of the WRAP-BWR-EM system is now complete and it is recommended for use by NRR. In early CY 80, the WRAP system was made operational on the Harry Diamond Lab computer, located in a Maryland ~~suburb of~~ Washington, and a special remote terminal was installed in Bethesda so that NRR personnel could perform their own calculations with WRAP. An indoctrination class in the use of WRAP was also held in Bethesda by SRL personnel in early CY 80.

Currently, most of the NRR use of WRAP is being performed at SRL. These include a BWR-LOCA sensitivity study and other audit calculations, funded by NRR. RES intends to supply maintenance of the WRAP system over the next few years, to improve modeling and to solve any coding problems that are identified by NRR use.

Thomas Murley, Acting Director
Office of Nuclear Regulatory Research

Enclosures:
1. Appendices I and II
2. Figures 1 through 4

cc: D. F. Ross, NRR
P. Check, NRR
T. Speis, NRR
G. W. Knighton, NRR
R. Audette, NRR

CRESS:SS	RSR:ADB LS	RSR:ADB SF	RSR CEJ LST	RES
1SHTKN:Job C	LShotkin:per	SFabic	LSTong	TEMurley
9/22/80	9/29/80	9/29/80	9/7/80	9/ /80

R. Audette of NRR is now using the completed WRAP-BWR-EM package. He was informed on 10/1/80 that the RIL is completed and agreed that the research has reached a stage for transmittal of a RIL.