WRAP-BWR-EM SYSTEM DEVELOPMENT AND APPLICATIONS

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SUMMARY

The WRAP-EM system is a complete computational system for analysis of loss-of-coolant accidents (LOCAs) in light water power reactors. The system has been developed for use by the Nuclear Regulatory Commission in evaluating and interpreting reactor vendor evaluation model methods and results. The initial WRAP system was developed at the Savannah River Laboratory (SRL) in 1977. WRAP is a modified version of the RELAP4/MOD5 code with a modular structure, JOSHUA input templated records, automatic steady-state initialization procedure, and dynamic dimensioning capabilities. During 1978, the WRAP system was extended to provide a complete evaluation model (EM) capability for boiling water reactors (BWRs). The BWR-EM system has been extensively evaluated and is currently being verified by comparison with LOCA experiments and with vendor EM calculations. A similar capability for pressurized water reactors (PWRs) has been completed recently.

Modular extensions to the WRAP system have provided the additional computational capabilities required for complete EM analysis of LOCAs in BWRs. These additional modules include GAPCON for calculation of initial fuel conditions, NORCOOL for analysis of the reflood of the core, and MOXY for the calculation of the fuel assembly hot plane temperature. In addition, a BWR steady-state initialization procedure has been developed to provide the initial operating state of the reactor system. To fully automate the LOCA analysis, data interfaces between the various modules are provided via the JOSHUA Operating System. The interfaces include the transfer of initial fuel conditions, as calculated by GAPCON, to WRAP for the thermal-hydraulic initialization of the reactor system and to MOXY for initialization of the fuel hot plane analysis. At the end of blowdown, the WRAP/NORCOOL interface performs a system renodalization and transfers the modified database from WRAP to NORCOOL. The WRAP/MOXY interface provides MOXY with time-dependent data from the blowdown, and the NORCOOL/MOXY interface provides data from the reflood calculation.

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Initial evaluation of the BWR system indicated that modifications and improvements to each computational step were necessary for an adequate EM analysis capability. These include:

- RELAP4/MOD5 updates
 1. Generalized GEXL correlation
 2. Improved slip modeling
 3. Jet pump smoothing
- MOXY - addition of the Thom nucleate boiling correlation for initialization of the hot plane analysis.
- GAPCON - addition of the Thom nucleate boiling correlation for calculation of initial fuel conditions.
- NORCOOL - improvement of the reflood initialization procedure as well as physical model improvements.

RELAP4/MOD5 and MOXY were modified by the Idaho National Engineering Laboratory (INEL) and implemented in the WRAP-EM system at SRL. GAPCON was modified by SRL. NORCOOL was modified at the Riso National Laboratory in Denmark (with assistance from SRL)

Evaluation of the consistency of the fuel models in WRAP (RELAP), MOXY, and GAPCON led to the above-mentioned modifications to GAPCON and MOXY. With these modifications, the fuel model consistency has been re-examined using Hope Creek design fuel data. The results of the studies indicate excellent agreement between GAPCON and WRAP (RELAP4): volume-average fuel temperatures calculated by the two codes differed by less than 2%. GAPCON and MOXY also agree closely when the gap heat transfer coefficient is supplied to MOXY from the GAPCON calculation. In both cases, the GAPCON results are more conservative than the WRAP and MOXY predictions. Inconsistencies that increase with fuel burnup are noted when the gap heat transfer coefficient is calculated by MOXY. The MOXY procedure for calculating gap heat transfer coefficients is currently being examined to determine the source of these inconsistencies.

The BWR steady-state procedure developed at SRL employs a time-dependent solution of the thermal-hydraulic equations under the constraints of constant core power, total water mass in the system, steam dome specific volume, and feedwater junction specific enthalpies. The uniqueness of the solution obtained by the BWR steady-state procedure has been demonstrated, and consistency with the transient solution has been established. The procedure requires approximately 15 to 20 CPU minutes on an IBM 360 Model 195 computer for a typical BWR nodalization.

Initial evaluation of the BWR-EM system for a large break in the recirculation line (pump suction side) of a BWR-4 and a BWR-6 plant design indicated an early departure from nucleate boiling (DNB) for the hot fuel bundle that is not characteristic of BWR behavior. This discrepancy was traced to an inaccurate critical heat flux correlation. With the generalized GEXL correlation, the behavior is predicted better. The BWR system has also been evaluated to assess the calculational capabilities for small break analysis. Initial indications are that small breaks can be analyzed, but further study is planned to assess the adequacy of the physical models for describing the two-phase phenomena that develop in extremely slow blowdown transients.

Completion of the evaluation of the WRAP-BWR-EM system is planned for December 1979. This will include detailed comparisons with TLTA experiments (6007 and 6406) as well as comparisons with vendor-computed LOCA results for the Peach Bottom reload and for a generic BWR-6 plant. As mentioned above, extensive evaluation of the calculational capabilities for small break analysis is also planned. After completion of the BWR verification studies, reference and sensitivity studies for generic BWR-6 and BWR-4 plants will be performed. Also, the BWR-EM system will be used to evaluate licensing concerns as requested by NRC.

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SLIDE 1

WRAP-BWR-EM System Development
and Applications

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SLIDE 2

WRAP Project Description

WRAP (Water Reactor Analysis Package)

A modular computational system for
complete analysis of loss-of-coolant
accidents in light water power reactors.

SLIDE 3

WRAP-EM System Development

- (1976-77) WRAP-User-Oriented Blowdown Analysis Code
for Routine Production Use
- (1977-78) Automatic Evaluation Model (EM) Capability
for Complete BWR LOCA Analysis
- (1978-79) Automatic EM Capability for Complete PWR
LOCA Analysis

WRAP-BWR-EM SYSTEM

- RELAP Updates
 1. Generalized GEXL Correlation
 2. Improved Slip Modeling
 3. Jet Pump Smoothing
- MOXY — Thom Nucleate Boiling Correlation
- GAPCON — Thom Nucleate Boiling Correlation
- NORCOOL
 1. Improved Initialization
 2. Improved Physical Models

SLIDE 6

Fuel Model Consistency Study Results

- GAPCON — WRAP (RELAP)
 - <2% - Maximum Fuel Temperature
 - <2% - Average Fuel Temperature
- GAPCON — MOXY
 - <1% - Maximum Fuel Temperature
 - <2% - Average Fuel Temperature
- MOXY Results are Inconsistent when Gap Heat Transfer Coefficient is Calculated Internally

SLIDE 7

BWR Steady-State Initialization Procedure

User Specifies:

- Total Water Mass in the System
- Steam Dome Specific Volume
- Feedwater Junction Specific Enthalpies

Procedure Computes:

- Thermodynamic State of all Control Volumes
- Flow Rates for all Junctions Including the Feedwater and Steam Line

SLIDE 8

BWR-4 System Model

SLIDE 9

BWR Steady-State Consistency Test:
Steam Dome Pressure vs. Time

SLIDE 10

BWR Steady-State Consistency Test:
Hot Channel Flow Rate vs. Time

SLIDE 11

Steam Dome Pressure — Recirculation Line Suction
Break in a BWR-4 Plant, Break Size = 3.67 ft^2

SLIDE 12

Water Level in Lower Plenum, Core and Upper Plenum —
Recirculation Line Suction Break in a BWR-4 Plant,
Break Size = 3.67 ft²

SLIDE 13

Fuel Rod Surface Temperature — Hot Channel —
Hot Pin — Hot Level — Recirculation Line Suction
Break in a BWR-4 Plant, Break Size = 3.67 ft²

SLIDE 14

BWR-6 System Nodalization

SLIDE 15

Recirculation Line Suction Break in a
Generic BWR-6 Plant, Break Area = 0.5 ft²

SLIDE 16

Recirculation Line Suction Break in a
Generic BWR-6 Plant, Break Area = 0.5 ft²

SLIDE 17

Future Work

- BWR Verification Studies
 1. TLTA 6007 and 6406
 2. Peach Bottom Reload
 3. BWR/6 - 218 Generic Plant
- Small Break Model Evaluation
- Technical Support to NRC
 1. Reference and Sensitivity Studies
 2. Licensing Concerns