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AN EFFICIENT METHOD FOR SIMULATION OF BWR SEVERE ACCIDENT SEQUENCE EVENTS BEFORE CORE UNCOVERY*

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An Efficient Method for Simulation of BWR Severe Accident Sequence Events before Core Uncovery*

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The Oak Ridge National Laboratory (ORNL) has participated in the Severe Accident Sequence Analysis (SASA) program since it was established in 1980 by the Containment Systems Research Branch of the Nuclear Regulatory Commission. The SASA program at ORNL has researched potentially severe accidents at Boiling Water Reactors (BWRs), with the objective of establishing as realistically as possible the sequence of events and consequences of each accident. The Browns Ferry Unit 1 BWR has been utilized with the full cooperation of the Tennessee Valley Authority as the example plant for the accident studies.

In order to obtain a realistic prediction of the consequences of a severe accident, it is necessary to analyze both the initial and final phases of the accident sequence. The initial phase takes place before significant fuel damage, typically over a period of hours, and sets the circumstances and timing of the possible progression to the final, fuel damage phase of the accident. An ORNL-developed simulation of the Browns Ferry BWR is used as a scoping tool to help the investigators to gain an understanding of the initial phase of each accident, including the effect of operator actions. This simulation, designated BWR-LACP (Boiling Water Reactor - Loss of AC Power) to incorporate a description of the first accident sequence studied, has been upgraded and expanded to meet the needs of each subsequent accident sequence studied at ORNL. The purpose of this paper is to discuss the attributes of the BWR-LACP simulation and to give specific examples of the results that have been obtained.

The most basic assumption of the BWR-LACP code is that the reactor vessel, internals, and fuel are undamaged. The thermal-hydraulic conditions are calculated for both reactor vessel and primary containment. The programming provides flexibility to model the effect of operator actions. These facts, combined with economical computer running time, allow BWR-LACP to complement the calculational capability provided by an arsenal of other computer codes, such as RELAP-5 and the ORNL-modified version of MARCH, that are capable of analyzing some aspect of the initial phase of BWR severe accidents. BWR-LACP is implemented in the IBM CSMP (Continuous System Modeling Program) simulation language, which allows the solution of large scale problems involving non-linear differential equations.

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An adequate simulation code for the initial phase of BWR severe accidents requires only a relatively simple primary system model coupled to a primary containment model. The task of calculating the reactor coolant system pressure and water level is aided by the simplicity of the BWR primary coolant system design. The voiding that accompanies boiling is expected in both normal operation as well as during accident sequences. The reactor vessel internals, as illustrated by Fig. 1, are designed such that boiling enhances the natural circulation of coolant through the core.

Special features of the BWR design require special consideration in modeling. The reactor vessel water level measurement is utilized to determine the timing of automatic and manual control of reactor vessel injection. The Main Steam Isolation Valves (MSIVs) usually close early in the accident sequence and the reactor vessel thus becomes an isolated pot of boiling water steaming through the safety relief valves, which discharge to the pressure suppression pool (PSP). As illustrated by Fig. 2, the BWR Mark I containment volume is relatively small. The large mass of water in the PSP provides the sole heat sink in most severe accident sequences. The more responsive nature of the BWR containment, coupled with safety system actions that are automatically keyed to the primary containment pressure, make it essential that containment response be simulated in an accident sequence study.

The BWR-LACP code simulates all interacting plant systems that determine accident sequence development. The functioning of the injection systems, the reactor vessel SRVs, and the primary containment cooling systems are especially important to the course of an accident sequence. Several systems, for example, the main condenser heat sink or the main feedwater pumps, do not need to be simulated because the MSIVs typically go shut in a severe accident sequence and, in effect, disconnect or disable those systems. Automatic and manual control of the injection systems, SRVs, and containment cooling systems are modeled. Automatic control is simulated on a functional level. The simulation of manual control is based on a review of the plant emergency procedures, operator training manuals, interviews with operators, and upon observation of operator response during simulator working sessions. The control room indications of vessel water level influence both manual and automatic control of injection systems. Two level instruments are simulated: the Emergency Systems range and the Post-Accident Flooding range. Possible systematic errors due to changing reactor vessel pressure and drywell atmosphere temperature are approximated by the model.

The mathematical and physical bases for the BWR-LACP models are described in detail in Ref. 1. Major considerations are outlined as follows. The core heat production includes heat released by decaying fission products and actinides, as specified by the 1979 ANS Standard. An optional point kinetics model can be invoked for Anticipated Transient without Scram (ATWS) calculations. Enthalpy and mass are calculated for each of four liquid or two-phase volumes and for the steam space within the reactor vessel. The natural in-vessel circulation is calculated by balancing the driving head against the overall unrecoverable pressure drop. Solution of the differential equations for the steam space specific volume and enthalpy allows the vessel pressure to be determined by a steam table look-up routine. The flow of steam from the reactor vessel, through the SRVs to the pressure suppression pool (PSP) is modeled as a critical flow of dry steam; the reactor vessel internals separate moisture from the steam before it exits the reactor vessel.

The thermodynamic calculations for the primary containment include mass and energy balances for one liquid region to describe the PSP and for two atmosphere regions to describe the wetwell and drywell atmospheres. The nitrogen and water vapor gases of the atmosphere regions are treated as perfect gases. Although it has proven adequate for most SASA studies, the one-node PSP model can be replaced with an ORNL-developed multi-node PSP model (Ref. 2) that calculates the variation of pool temperature with depth and with angular position around the major axis of the toroidal pool.

The Severe Accident Sequence Analysis (SASA) program at ORNL has performed detailed studies of five BWR accident sequences: Station Blackout (Ref. 3), Small Break LOCA Outside Primary Containment (Ref. 4), Loss of Decay Heat Removal (Ref. 5), Loss of Injection (Ref. 6), and Anticipated Transient without Scram (ATWS) (Ref. 7). The use of BWR-LACP to investigate the initial phase of each accident has resulted in important findings relevant to safety. The balance of this paper is devoted to examples of the types of insights that have been gained. During three of the studies (Station Blackout, SBLOCA Outside containment, and ATWS) it was decided that certain of the BWR-LACP calculations were important enough that they should be validated against the results of a more sophisticated code. RELAP-5 analyses performed by SASA program personnel at Idaho National Engineering Laboratory were utilized for this purpose. In each case, the RELAP results support the conclusions reached by use of the much simpler scoping tool, BWR-LACP.

The BWR-LACP calculations for the Station Blackout study demonstrated the importance of depressurizing the reactor vessel before battery failure. The capability to calculate both reactor vessel and primary containment response over the approximately eight hour period before core uncovery allowed investigators to fully understand that depressurizing the reactor vessel limits drywell temperature, reduces the total number of required SRV actuations, and extends the time to core uncovery after battery failure.

The use of BWR-LACP on the Small Break LOCA Outside Primary Containment study uncovered potential operator difficulties in recognizing the break condition immediately after the scram and, subsequently, in preventing reactor vessel overfill after depressurization. A substantial liquid line break with an initial leak rate of 550 gpm (34.66 l/s) is within the capability of the Reactor Core Isolation Cooling (RCIC) system, the smaller of the two high pressure injection systems; without the high drywell pressure that would accompany breaks inside the primary containment, such a leak could initially go unnoticed by control room operators. Reactor vessel overfill is a possibility because there is no automatic high water level cut-off of the very high capacity, low pressure, condensate and condensate booster pumps. In the absense of loss of offsite power, these pumps continue running after reactor scram and operator action would be required to turn them off. Vessel overfill would be a safety concern if enough water were pumped into the main steam lines to cause the subsequent failure of the steam-turbine-driven high pressure injection systems.

The application of BWR-LACP to the very slowly developing Loss of Decay Heat Removal (LDHR) accident enabled investigators to confirm (or rediscover) the safety significance of the Control Rod Drive Hydraulic system (CRDHS) injection potential. Without any operator intervention, the CRDHS pumps are able to supply all needed vessel injection. The LDHR calculation extended to 40 h, without depletion of the initial condensate storage tank volume. The effect of containment back pressure is included in the BWR-LACP calculation of Residual Heat Removal (RHR) pump net positive suction head (NPSH); the calculated NPSH for the LDHR accident was sufficient for RHR pump operation. Additional BWR-LACP runs made with partial RHR system operability show that any one of the four RHR system pool coolers would be sufficient to prevent excessive PSP temperature.

The safety significance of the CRDHS injection was explored more fully in the Loss of Injection study. The main BWR-LACP contribution to this study (which consisted mostly of an extensive study of the effects of partial core uncovery in causing core damage using a specially modified version of the MARCH computer code) was a series of calculations showing how operator actions could enhance the CRDHS injection flow, and prevent or minimize the period of core uncovery. The modeling of the CRDHS pumping system was made sufficiently detailed to allow the calculation of the flow enhancement that would result from operator actions such as starting the spare pump or opening the pump discharge throttling valve.

The ORNL ATWS accident sequence analysis is the most recently completed application of the BWR-LACP code. Figures 3 through 9 are included as an example of the types of output available. Even though the MSIV-closure ATWS is a very complex event, the use of BWR-LACP as a scoping tool has provided valuable insights into the effects of operator actions on sequence development. For example, without operator action this sequence can rapidly lead to power excursions, possibly severe enough to cause fuel damage, and to the overpressure failure of the primary containment. Figure 3 illustrates the power spikes that would, if not prevented by the operators, occur after automatic depressurization of the reactor vessel and automatic actuation of large-capacity injection by the low pressure emergency core cooling and the condensate booster pumps. The BWR-LACP calculations show that timely operator action can prevent these severe consequences. The most effective operator action, and a truly essential operator action for MSIV-closure ATWS, is the initiation of the injection of sodium pentaborate solution. Additional operator actions, such as those prescribed by the BWR Owners Group Emergency Procedure Guidelines could further limit the pressure suppression pool temperature buildup, but the BWR-LACP scoping runs indicate that these actions would be hard to implement during an accident.

Figures 4 through 9 show the BWR-LACP results for an MSIV-closure-initiated ATWS in which operator action is restricted to the single act of initiating the injection of sodium pentaborate solution at 5 min. The initial effect on reactor power (Fig. 4) is slight, but when the HPCI fails after about 16 min, there is additional in-core voiding that shuts down the reactor to essentially a decay heat level, and there are no severe power excursions. As shown on Fig. 5, two indicated vessel water levels are calculated in addition to the actual downcomer annulus water level. After 50 min, the vessel water level has recovered to above the normal vessel water level, initiating an automatic trip of the RCIC system. Core inlet enthalpy is plotted along with the injection flow (Fig. 6.). Figure 7 indicates the status of the safety relief valves and demonstrates their immediate and effective control, under automatic actuation, of vessel pressure. The repeated SRV actuations cause the suppression pool water level and temperature (Fig. 8) to increase rapidly at first and then less rapidly after the HPCI system injection stops at 16 min. Drywell pressure (Fig. 9) is included although it increased only slightly in this example, due to the increased vapor pressure of the overheated suppression pool.

In summary, BWR-LACP has been a versatile tool for the ORNL SASA program. The development effort was minimal, and the code is fast running and economical. Operator actions are easily simulated and the complete scope of both reactor vessel and primary containment are modeled. Valuable insights, as discussed in this paper and in References 3 through 7, have been gained into accident sequences. A Fortran version is under development and it will be modified for application to Mark II plants.

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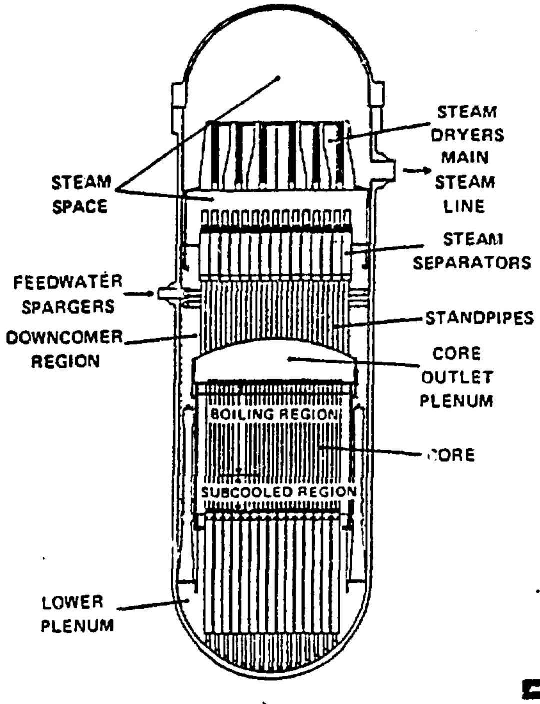

Fig. 1. Voiding is expected both in normal operation and during accident sequences.

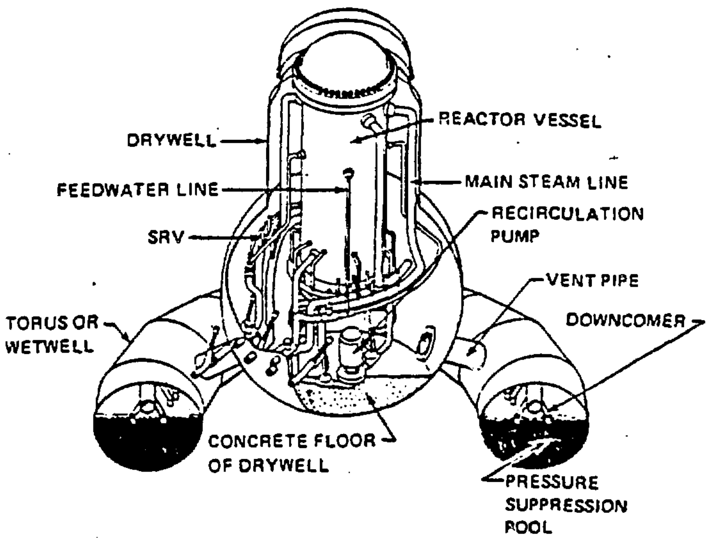

Fig. 2. The relatively small BWR primary containment plays an important part in accident sequences.

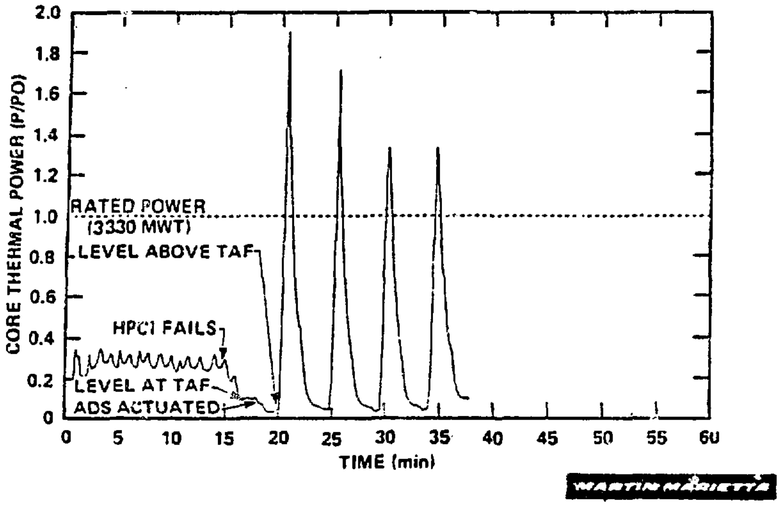

Fig. 3. Power excursions occur after HPCI system failure and ADS actuation for the case without operator action.

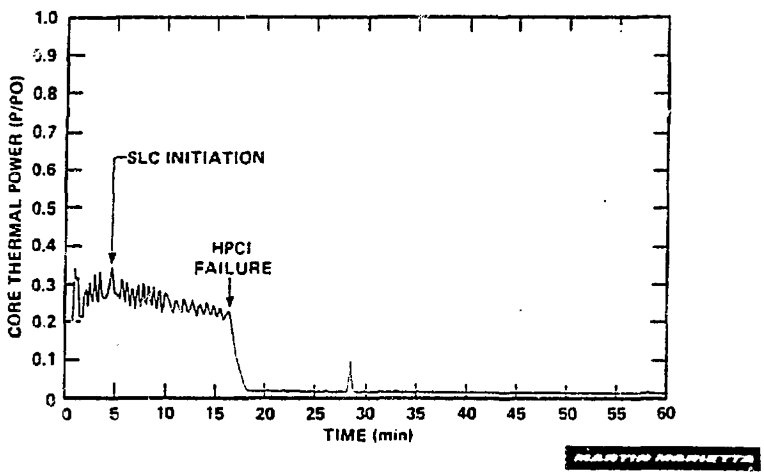

Fig. 4. With operator action to initiate boration at 5 min, the initial effect is slight but the power excursions after HPCI failure are eliminated.

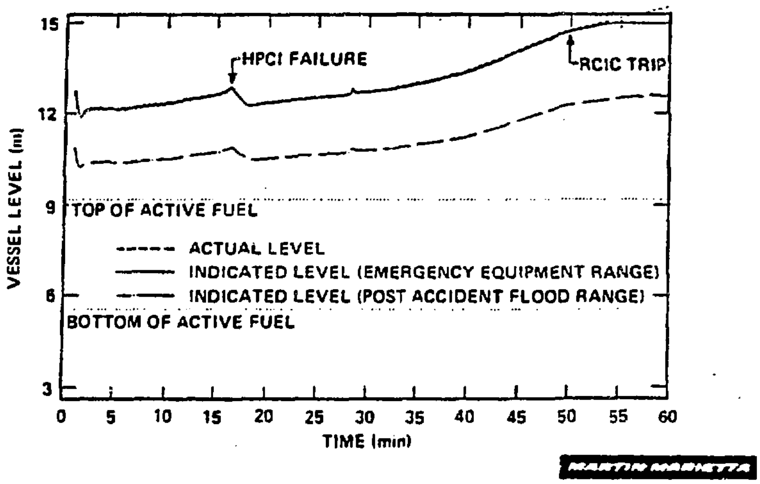

Fig. 5. The BWR-LACP results include actual reactor vessel downcomer water level and indicated water level on two control room indication systems.

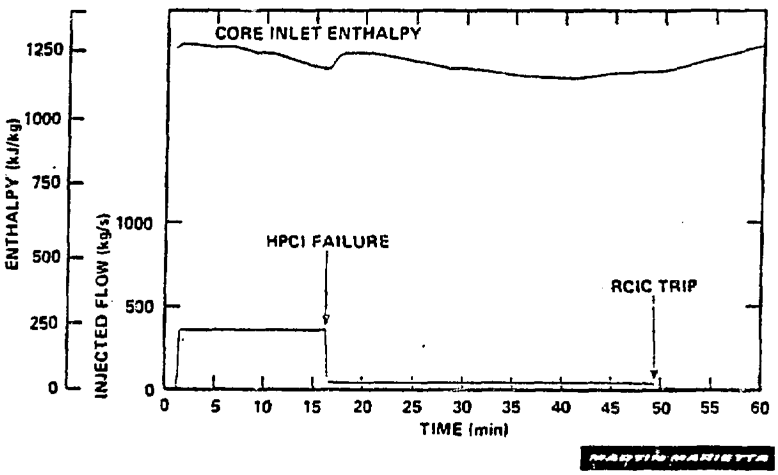

Fig. 6. The HPCI high-pressure injection system fails due to high lube oil temperature but RCIC runs until automatically tripped on high water level.

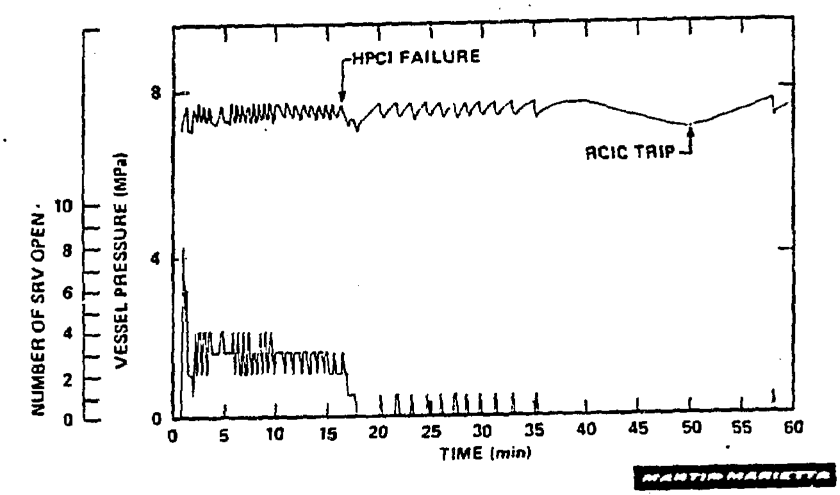

Fig. 7. The BWR-LACP results also show how automatic SRV actuation controls vessel pressure.

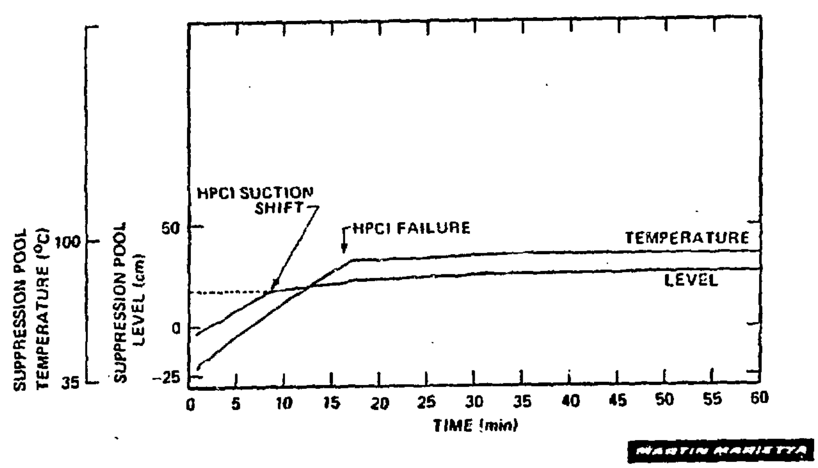

Fig. 8. The pressure suppression pool is heated rapidly at first. BWR-LACP results include both the pool level and bulk pool temperature.

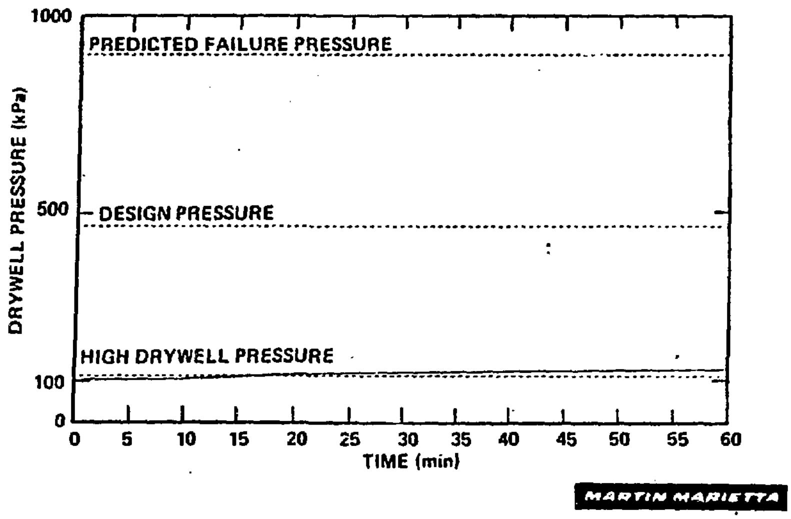

Fig. 9. Drywell pressure increases due to heatup of the pressure suppression pool.