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"Nodal Sensitivity in Modeling a Large Pressurized Water
Reactor System for a Loss-of-Coolant Accident"

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Nodal Sensitivity in Modeling a Large Pressurized Water Reactor System for a Loss-of-Coolant Accident

I. INTRODUCTION

In recent years much effort by the Atomic Energy Commission, its regulatory agencies, and the nuclear power industry has been directed toward developing the capability of assessing the performance of engineered safety systems designed to mitigate the consequences of a loss-of-coolant accident in a nuclear power reactor. To help provide this capability, computer codes have been developed that attempt to predict the course of a loss-of-coolant accident and the effectiveness of the emergency safety systems. The extent of geometrical description used in digital computer codes for predicting the consequences of a loss-of-coolant accident in a large pressurized water reactor system has evolved from simple codes employing only three control volumes^(1,2) to present generation codes in which the number of control volumes or nodes is limited only by the core storage of the computer⁽³⁾. The increased nodal description has frequently resulted in an increase in computation time. In order to avoid extensive computation time for safety analysis calculations, the geometrical description is often reduced by lumping similar system components together into a common control volume. This practice is valid only if the predicted results are not significantly affected by the reduction in geometrical description. Unfortunately, many analyses have been made without investigating the degree of geometrical description necessary to insure convergence to the proper solution. The present study was undertaken to determine the effect of the number and distribution of control volumes used in describing a large pressurized water reactor system on the solution of the conservation equations of mass, momentum, and energy. A version of the RELAP3 code⁽³⁾, dimensioned for two-hundred control volumes, was utilized in this study. A large pressurized water reactor system of current design subjected to a loss-of-coolant accident (100% offset shear of inlet piping) was described with several computer models that exhibited wide variation in both the number and distribution of the control volumes. The blowdown behavior calculated by the RELAP3 code using the various models was then compared. The number of variables that could be compared were numerous and led to difficulty in establishing criteria for comparing the effects of nodal description on the solution to the governing equations. In the interest of brevity and because of their importance and sensitivity to small perturbations in pressure, core flow rates and core surface temperatures were chosen as the common variables for comparison among the various models. The study has been divided into two general areas: modeling of components in which no energy is transferred across the boundary and modeling of components in which energy is transferred across the boundary (the core and steam generators). As initially suspected and as later verified by the study, strong coupling was exhibited among the conservation equations when energy was transferred into or from the fluid.

The intent of this paper is not to provide a description and interpretation of the events associated with a loss-of-coolant accident but rather to illustrate the need of adequate nodal description in loss-of-coolant accident analyses, and to furnish some general guidelines for modeling large pressurized water reactors.

II. MODELING OF REACTOR PRIMARY SYSTEM COMPONENTS IN WHICH HEAT IS NOT TRANSFERRED TO OR FROM THE COOLANT

In the reactor primary system during normal operation the temperature of the coolant changes markedly in passing through the core and the steam generators. In the other components of the reactor primary system, the coolant temperatures approach an isothermal condition and the main changes are hydraulic in nature. To evaluate effects of increasing the extent of geometrical description for those reactor system components which are almost isothermal, a series of computer models for a large PWR system of current design were developed. Computer models with 3, 5, 15, 16, 19, and 37 control volumes were utilized for these comparisons. In the 15-, 16-, 19-, and 37-volume models the number of core volumes and steam generator volumes remained constant and only the number of volumes used to describe the passive components were changed.

Figures 1 through 6 depict the 3-, 5-, 15-, 16-, 19-, and 37-volume models, respectively. As can be readily seen in Figures 1 and 2, the 3- and 5-volume models are simple, each combining the unbroken loops into a single volume. In both models, the core and upper plenum are included in the same control volume.

Figures 7 through 11 present the core inlet flow, and average core surface temperatures and Figures 12 through 16 present the core outlet flows that were predicted using the various models. The general trends of the core inlet and outlet flows for the 3- and 5-volume models are similar and differ mainly in magnitude. Little difference is observed after 4 sec. The core flows predicted using the 15-volume model deviates considerably from those predicted by the 16-, 19-, and 37-volume models. The core flows predicted by the latter three models are not significantly different. The same similarities and differences exhibited in the core flows are reflected in the core surface temperature histories. The maximum surface temperature and time of occurrence of an average rod predicted by the various models is presented in Table I. Little difference in the surface temperatures was observed for the 16-, 19-, and 37-volume models. The differences between the results predicted by the 3-, 5-, and 15-volumes models and those predicted by the 16-, 19-, and 37-volume models can be explained through examination of the method used in creating the models. In creating the 3-, 5-, and 15-volume models, dissimilar components, for example steam generator and piping, had to be represented with a single volume. In contrast, the 16-, 19-, and 37-volume models represent increased nodal description of similar components.

TABLE I

COMPARISON OF MAXIMUM SURFACE TEMPERATURE OF AVERAGE ROD
AS A FUNCTION OF THE EXTENT OF GEOMETRICAL DESCRIPTION

Number of Volumes	Maximum Surface Temperature of Average Rod (°F)	Time of Occurrence of Maximum Temperature (sec after rupture)
3	1146	2.87
5	1180	3.66
15	988	6
16	1090	6.75
19	1101	6.5
37	1115	6.25

Fig. 1. Three-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 2. Five-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

Fig. 3. Fifteen-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 4. Sixteen-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 5. Nineteen-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

Fig. 6. Thirty-Seven-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 6

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Fig. 7. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Three- and Five-Volume Models

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Fig. 8. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Five- and Fifteen-Volume Models

Fig. 9. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Fifteen- and Sixteen-Volume Models

Fig. 10. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Sixteen- and Nineteen-Volume Models

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Fig. 11. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Nineteen- and Thirty-Seven-Volume Models

Fig. 12. Core Outlet Flow Rate Calculated Using Three- and Five-Volume Models

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Fig. 13. Core Outlet Flow Rate Calculated Using Five- and Fifteen-Volume Models

Fig. 14. Core Outlet Flow Rate Calculated Using Fifteen- and Sixteen-Volume Models

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Fig. 15. Core Outlet Flow Rate Calculated Using Sixteen- and Nineteen-Volume Models

Fig. 16. Core Outlet Flow Rate Calculated Using Nineteen- and Thirty-Seven-Volume Models

From the preceding results two conclusions can be reached. First, dissimilar volumes should not be combined and secondly, increased nodal description of passive elements does not significantly affect predicted core flows and attendant surface temperatures.

III. MODELING OF COMPONENTS IN WHICH ENERGY IS TRANSFERRED ACROSS THE BOUNDARY

The components in which energy was transferred across the system boundaries were thought to be especially sensitive to nodal description due to the strong coupling between the fluid dynamics and the heat transfer. The coolant undergoes a significant temperature change in these components and in order to properly describe the energy distribution and driving forces for energy transfer rather detailed geometrical description seemed warranted because a single temperature and pressure is associated with each control volume.

Four models which differed only in the number of volumes used to describe the steam generator were used to study the effects of nodal description in the steam generator. The base case was the 16-volume model (Figure 4) used previously and which described the steam generator as a single volume. Models were created that used 22, 28, and 36 volumes in which 4, 7, and 11 volumes were used to describe the steam generator. These models are presented in Figures 17, 18 and 19.

The core inlet flows and average core surface temperatures are presented in Figures 20 through 22 and the core outlet flows are presented in Figures 23 through 25. As evidenced by these figures, the core flows and the attendant surface temperature histories are significantly different for the 16-, 22-, and 28-volume models, whereas they are quite similar for the 28- and 36-volume models. Emphasis is placed on the fact that the modeling of the steam generator has significant effect on the core flow and core thermal response. The maximum surface temperatures of an average rod for the four different models are presented in Table II. The agreement of the 28- and 36-volume models indicates convergence to the proper answer. The conclusion can be reached that 1- and 4-volume representations of the steam generators are inadequate and more detailed description is warranted. The 7-volume model of the steam generator appears to be adequate.

TABLE II

COMPARISON OF MAXIMUM SURFACE TEMPERATURE OF AVERAGE ROD
AS A FUNCTION OF THE EXTENT OF GEOMETRICAL
DETAIL USED IN MODELING THE STEAM GENERATORS

Number of Volumes	Maximum Surface Temperature of Average Rod (°F)	Time of Occurrence of Maximum Temperature (sec after rupture)
1	1090	6.5
4	1010	1.625
7	885	1.9
11	887	1.9

Of paramount interest in loss-of-coolant analyses is the behavior of the core because it in large part determines the severity of the accident. To investigate the effects of nodal description in the core, models were created using 1-, 5-,

Fig. 17. Twenty-Two-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 18. Twenty-Eight-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 20. Core Inlet Flow Rate and Average Core Surface Temperatures Calculated Using Sixteen- and Twenty-Two-Volume Models

Fig. 21. Core Inlet Flow Rate and Average Core Surface Temperatures Calculated Using Twenty-Two- and Twenty-Eight Volume Models

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Fig. 22. Core Inlet Flow Rate and Average Core Surface Temperatures Calculated Using Twenty-Eight- and Thirty-Six-Volume Models

Fig. 23. Core Outlet Flow Rate Calculated Using Sixteen- and Twenty-Two-Volume Models

Fig. 24. Core Outlet Flow Rate Calculated Using Twenty-Two- and Twenty-Eight-Volume Models

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Fig. 25. Core Outlet Flow Rate Calculated Using Twenty-Eight- and Thirty-Six-Volume Models

and 11-volumes in the core. The 19-volume model with one core volume (Figure 5) was used as the base case and only the core volumes were changed for this study. The RELAP3 models for the 5-volume core and 11-volume core cases are presented in Figures 26 and 27. Figures 28 and 29 present the core inlet flow and the surface temperature of an average pin at the vertical midpoint of the core and Figures 30 and 31 present the core outlet flow.

The core inlet flows for the three cases are similar whereas the outlet flow for the 1-volume core model is considerably different from those of the 5- and 11-volume models. The core flows for the 5- and 11-volume core models are nearly identical. In the case of the 1-volume model a stagnation point occurred in the core between 0.25 and 1.0 second and flow was out of the core at both the top and bottom. At the corresponding time for the 5- and 11-volume models the stagnation point was in the upper plenum and flow was reversed at both the top and bottom of the core. The temperatures at the midpoint are much higher for the 5- and 11-volume core models due to both the difference in flow rate and the better description of the axial flux profile in the core. The latter appears to be the more significant because the increased flow would tend to lower the temperatures. The difference in flow rate among the 1-volume core model and those with 5- and 11-volumes is attributed to better representation of the axial pressure and energy distribution in the core and the attendant driving forces for energy transfer. The maximum temperatures for the three models are presented in Table III. Increasing the number of volumes in the core from five to eleven had little effect on the maximum temperature. These results evidence the importance of describing adequately the axial flux profile, pressure distribution, and energy distribution of the core. The 5-volume representation appears to be adequate.

TABLE III

COMPARISON OF MAXIMUM SURFACE TEMPERATURE OF AVERAGE
ROD AS A FUNCTION OF THE EXTENT OF GEOMETRICAL DESCRIPTION
USING IN MODELING THE CORE

Number of Volumes	Maximum Surface Temperature of Average Rod (°F)	Time of Occurrence of Maximum Temperature (sec after rupture)
1	1101	6.5
5	1569	8.29
11	1620	7.89

IV. CONCLUSIONS

The results of this study indicate the following criteria should be used in modeling large pressurized water reactors for loss-of-coolant analyses:

- (1) Components with dissimilar thermal characteristics should not be lumped together into a common control volume.
- (2) Detailed modeling of passive components does not appear necessary for the saturated portion of blowdown.
- (3) Components in which energy is transferred across the system boundary require detailed nodal description. For the problem investigated in this study seven volumes in the steam generator and five volumes in the core were adequate. For other problems different geometrical description may be necessary and must be determined for the particular problem of interest.

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Fig. 26. Twenty-Three-Volume RELAP3 Model of Four-Loop Large Pressurized Water Reactor

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Fig. 27. Twenty-Nine-Volume RELAP3 Model of Four-Loop Large Pressurized-Water Reactor

Fig. 28. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Nineteen- and Twenty-Three-Volume Models

Fig. 29. Core Inlet Flow Rate and Average Core Surface Temperature Calculated Using Twenty-Three- and Twenty-Nine-Volume Models

Fig. 30. Core Outlet Flow Rate Calculated Using Nineteen- and Twenty-Three-Volume Models

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Fig. 31. Core Outlet Flow Rate Calculated Using Twenty-Three- and Twenty-Nine-Volume Models

- (4) Nodal sensitivity studies should be conducted in all loss-of-coolant analyses to determine the extent of nodalization necessary and thus to insure convergence to the correct solution. Failure to do so can lead to incorrect and nonconservative predictions.

V. REFERENCES

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