

**Depressurization Accident Analyses for
the Fort St. Vrain Reactor**

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MASTER

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DEPRESSURIZATION ACCIDENT ANALYSES FOR THE FORT ST. VRAIN REACTOR

D. D. Paul

ABSTRACT

Design-basis depressurization accident analyses for the Fort St. Vrain reactor were performed using the FLODIS (Ref. 4) code. The FLODIS code models the active core, side reflector, gas annulus between the core barrel and the PCRV liner, and the PCRV cooling system. Results are presented for the Pelton circulators operating at 10,550, 8800, and 7000 rpm. Maximum temperatures of selected components are plotted as a function of time during the transient. None of the components studied exceeded the temperature at which failure or damage may occur. However, there must be sufficient mixing of the outlet gas in the lower plenum to insure the integrity of the steel liners of the steam generator inlet ducts.

Keywords: conduction, convection, orifice, Pelton circulator, plenum, refueling region, reflector, transient.

INTRODUCTION

During startup testing of the Fort St. Vrain reactor, cracks were discovered in the cervic coupling and bucket areas of the Pelton circulators. The circulators were replaced. An investigation into the causes of the cracks revealed that the problem could be eliminated by operating the Pelton circulators at a reduced speed. Since these circulators are used only during auxiliary cooling situations, a review of postulated accidents that require their operation was performed. It was concluded that the most serious accident requiring use of the Pelton circulators was the design-basis depressurization accident (DBDA). The Nuclear Regulatory Commission (NRC) asked Oak Ridge National Laboratory (ORNL) to provide an independent review of this accident.

For a DBDA, the following postulated events are assumed. The reactor is operating at 105% power. Instantaneously, the reactor depressurizes to atmospheric pressure and simultaneously the reactor is shutdown. There is a five minute delay in the startup of the Pelton circulators and then cooling is resumed. The entire transient lasts for approximately 10 hours.

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Maximum fuel temperatures occur approximately 2 hours into the transient, whereas maximum coolant exit temperatures occur approximately 4 hours into the transient. From the above figures it should be obvious that the response of the entire system is rather slow compared with other reactor types undergoing postulated accidents requiring emergency core cooling. This is a major safety advantage for HTGRs.

COMPUTATIONAL MODEL OF THE FORT ST. VRAIN REACTOR

The FLODIS code was written specifically for analyzing the Fort St. Vrain reactor. The model includes the active core, side reflector, gas annulus between the core barrel and the PCRV liner, the PCRV cooling system, top plenum, bottom plenum, and orifice devices. Essentially, the model includes the upper half of the PCRV above the core support floor as shown in Fig. 1.

The FLODIS code calculates the flow distribution among the 37 refueling regions of the Fort St. Vrain reactor. Furthermore, the code calculates, based on the mesh spacing, how the flow will distribute within a refueling region. There is an important distinction to be made here. Intra-regional distribution of coolant flow is not handled by any other codes which model multiregion HTGR cores. Both natural and forced convection flows are accommodated with the FLODIS calculation.

The temperature distribution is calculated throughout the entire reactor core. Heat is allowed to flow between all adjacent grid spaces; the entire core temperature distribution is calculated as a three dimensional heat transfer problem. Heat may flow from the active core, through the side reflector blocks, may be transported to the gas flowing within the annulus between the core barrel and PCRV liner, and may be conducted to the PCRV cooling system. Overall, the problem is a very complicated heat transfer - fluid flow calculation.

A unique feature of the FLODIS code is that it models the entire system with a rectangular grid as shown in Fig. 2. A typical refueling region is represented by 4 rectangular mesh spaces; whereas a partial refueling region is represented by 3 rectangular mesh spaces. There are

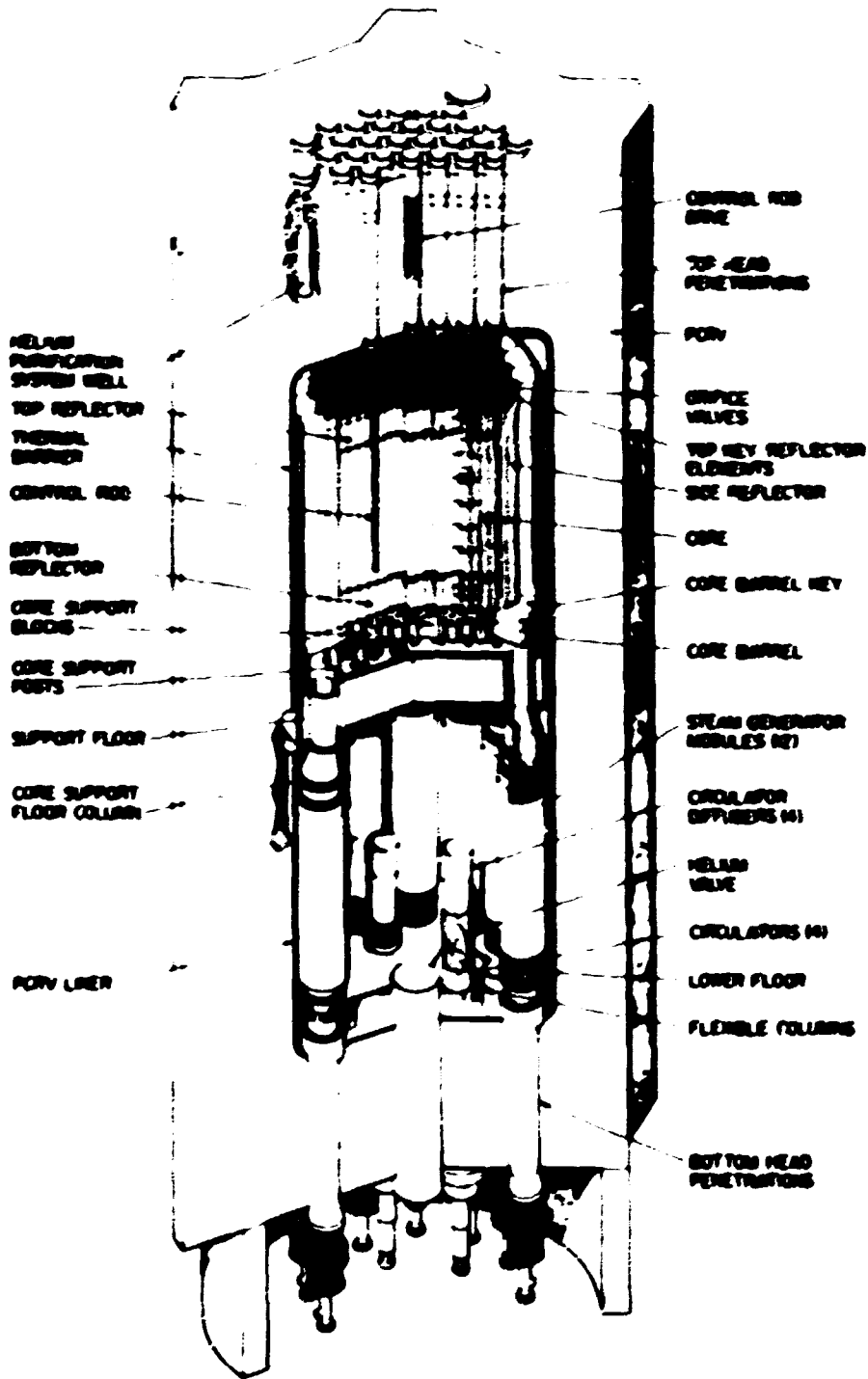


Fig. 1. Reactor arrangement.

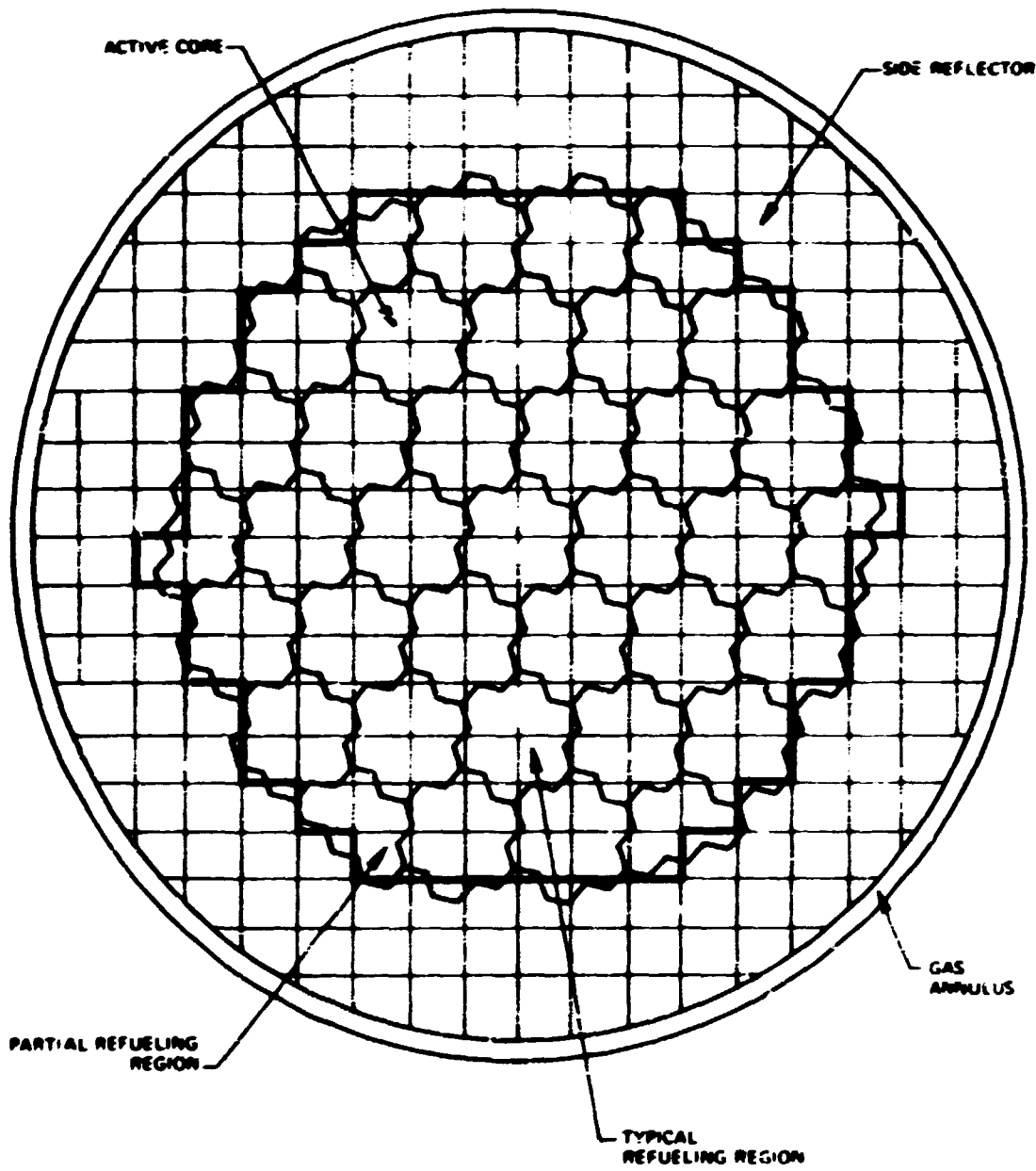


Fig. 2. Mesh spacing in FLODIS code.

1-2 mesh spaces in the active core, 146 mesh spaces in the side reflector, and 52 mesh spaces in the gas annulus between the core barrel and PCRV liner. Axially there are 20 mesh spaces. Counting both core and coolant nodes, the entire system is represented by 12,560 nodes.

This model differs significantly from the RECA code, which represents a refueling region by a single hexagonal mesh space. RECA uses 8 axial nodes. Again counting both core and coolant nodes, the entire system, as represented by RECA, uses approximately 600 nodes. The total number of nodes used to represent the system gives an indication of the complexity of the analysis, since both the computer memory and time requirements increase drastically with the number of nodes. This comparison illustrates the relative size of the two codes.

A flow diagram of a typical refueling region is shown in Fig. 3. For each of the 37 refueling regions, the FLODIS code calculates the flow, \dot{W}_i , entering the orifice. However, the code goes one step further in that it now distributes the flow within a refueling region such that

$$\dot{W}_i = \dot{W}_1 + \dot{W}_2 + \dot{W}_3 + \dot{W}_4.$$

Depending on the temperature gradient across a refueling region, the sub-region flows may vary considerably. For the DBDA analyses in this report, as much as a 15% difference in flow between subregions was observed. This result is important because the subregion with the highest temperature will obtain the least flow relative to the other subregions.

The orifice loss coefficients are set based on given steady-state requirements. Experimental data obtained from General Atomic (Ref. 2) were used to determine minimum and maximum orifice loss coefficients together with orifice areas at steady-state conditions. Following depressurization and a reduction in coolant flow, the orifice loss coefficients are assumed to remain constant. Some experimental evidence exists for other types of orifices which indicate that this assumption might not be valid.² This is definitely one area where further experimental work should be done.

Two limitations exist on the use of the FLODIS code. These are computer memory and computer time restrictions. The FLODIS code needs 540 K of computer memory. For a DBDA analysis, the execution time is

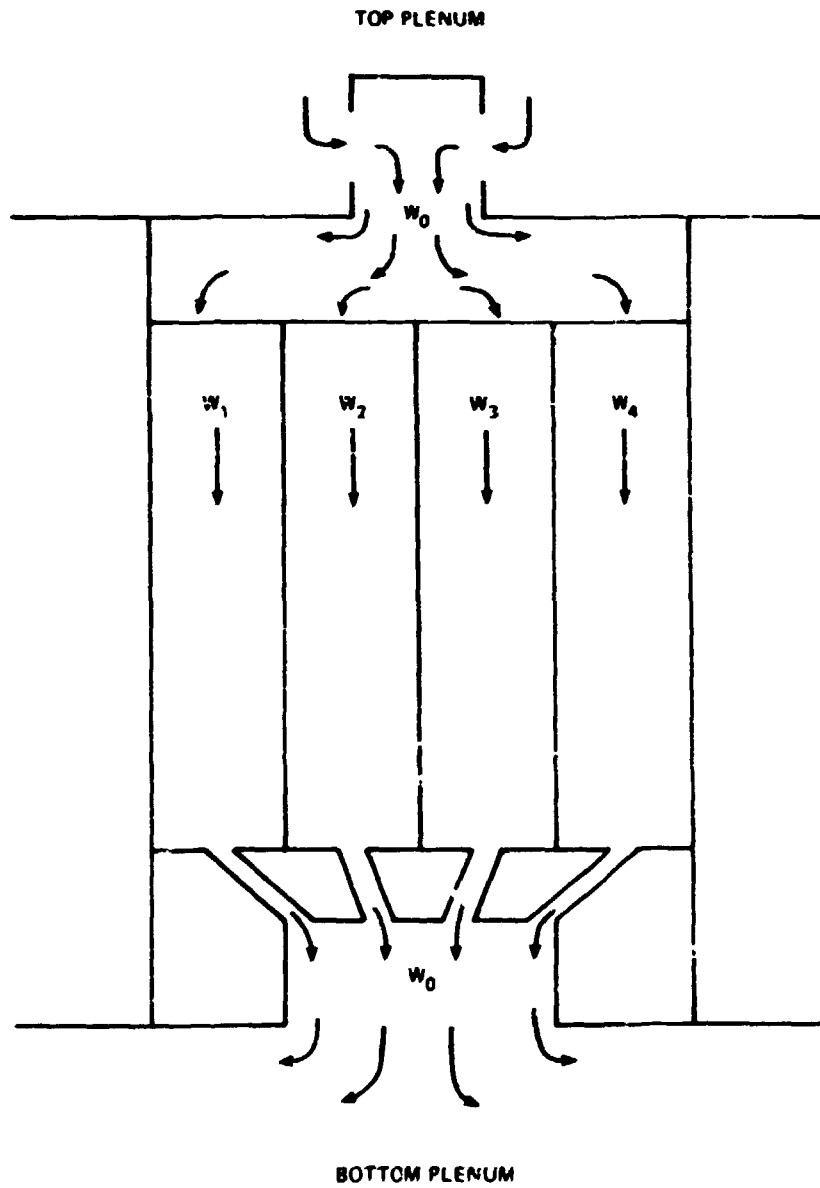


Fig. 3. Refueling region flow diagram.

40 minutes on the IBM 360/91 computer at ORNL. Total cost for the job was \$110. However, the setup of a test case is not difficult because the code was written specifically for analyzing the Fort St. Vrain reactor. All of the input required to define the geometry of the reactor has been incorporated internally in the code. This has allowed for detailed modeling of the entire system, since general purpose routines need not be devised.

For details concerning the differential equations, method of solution, and calculation of conductances as solved by FLODIS, documentation of the code has been made in a previous report (Ref. 4).

DEPRESSURIZATION ACCIDENT ANALYSES

Three separate cases of depressurization accidents were analyzed. The only difference between the cases was the choice of the circulation speed. General Atomic originally specified the Pelton circulator speed to be 10,440 rpm. During inspection prior to startup of the reactor, cracks were discovered in the curvic coupling and bucket areas of the Pelton circulators. The circulators were replaced and the high speed trip setting was reduced to 8800 rpm. To be conservative General Atomic analyzed the DBDA with a circulator speed of 7000 rpm. Analyses of the design-basis depressurization accident at these three circulator speeds were performed. Values for the circulator total flow and gas temperature were supplied by General Atomic.

Figure 4 shows maximum fuel temperatures during the transient for the three different circulator speeds. Also shown on the graph is the critical safety limit for fuel particles. This is the temperature above which there is a rapid deterioration of the fission product barrier. The above analysis indicates that the temperatures are low enough so as to insure that the design criteria will not be exceeded.

Figure 5 shows the maximum coolant outlet temperature for the different circulator speeds. Conservatively, this temperature may be taken as the local temperature of the cast silica blocks lining the floor of the lower plenum. Again there is an adequate safety margin between the curves and the critical safety limit for cast silica blocks. The critical safety

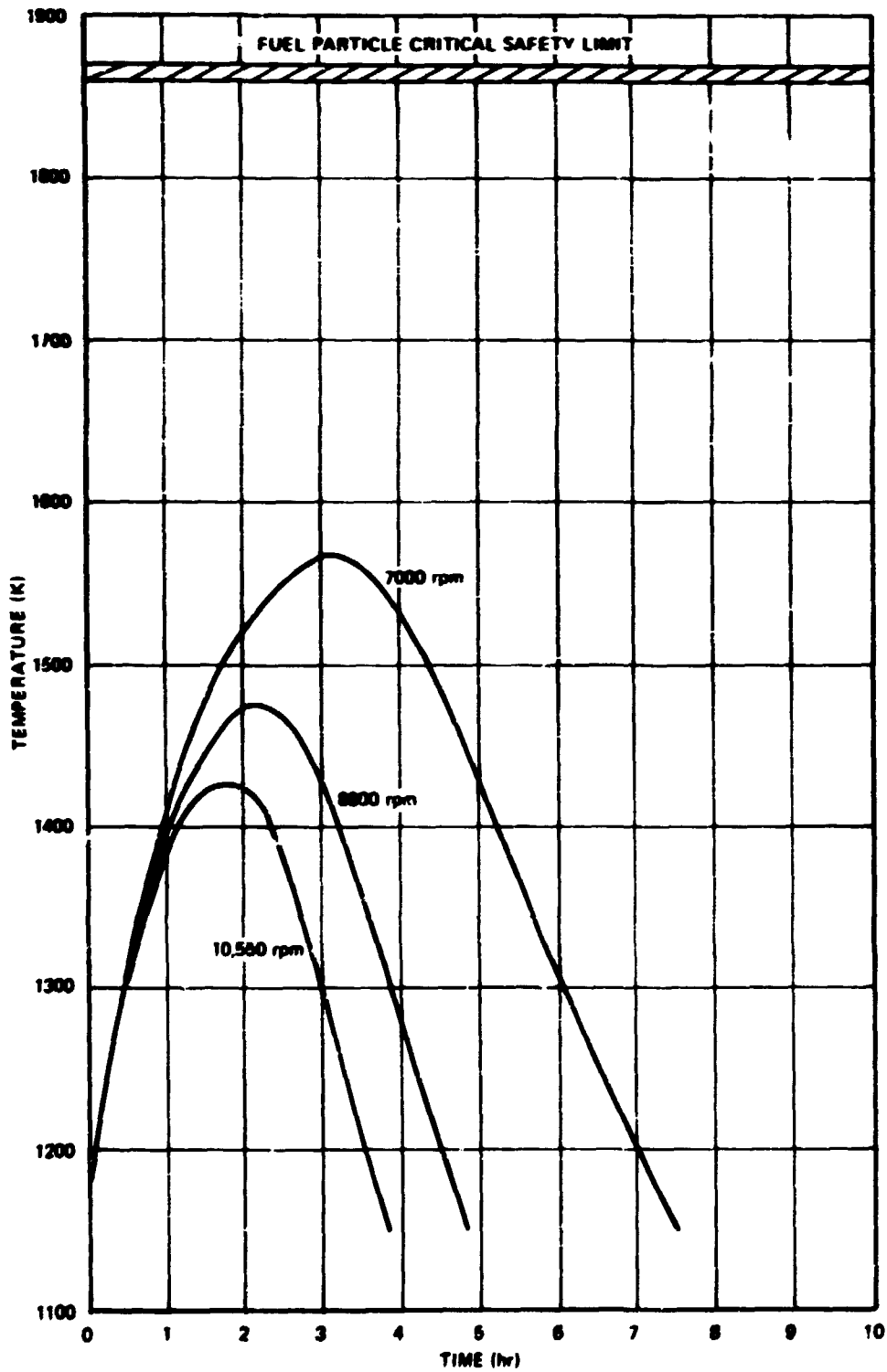


Fig. 4. Maximum core temperature during DBDA for selected circulation speeds.

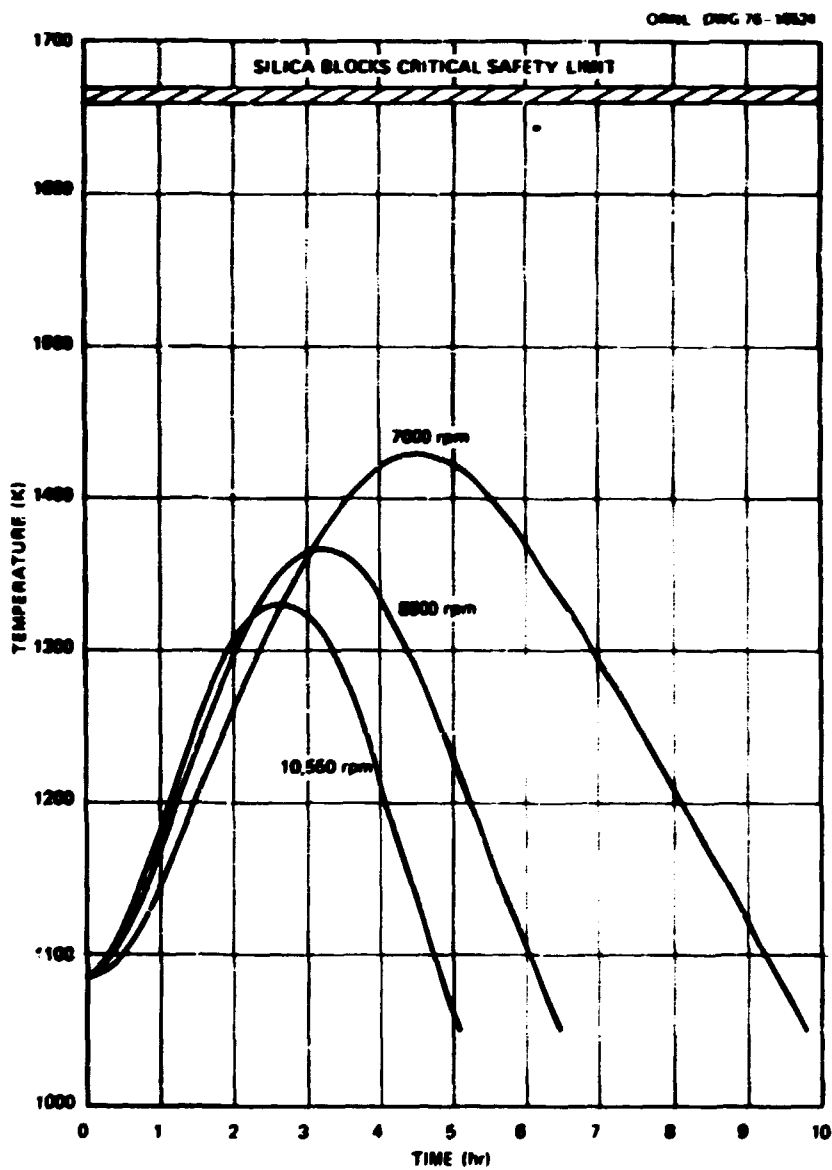


Fig. 5. Maximum coolant outlet temperature during a DBDA for selected circulator speeds.

limit for this component is the temperature which causes creep deformation of 2% at the expected operating stress level in one hour.

Figure 6 shows the mixed mean temperature of the coolant. The critical safety limit shown on this graph is for the steel liners on the steam generator inlet ducts. Again, this is the temperature above which creep deformation of 2% occurs within the structural elements in one hour. This plot shows that if the refueling region flows mix thoroughly before entering the ducts, there is an adequate safety margin for this component. However, it is interesting to consider the case of imperfect mixing. Although all experimental data made by General Atomic on mixing in the lower plenum was obtained at full flow simulations, it still provides insight into the physical processes. Data by W. E. Walker (Ref. 5) shows a mixing factor of 0.3 can be used for the Fort St. Vrain reactor under full flow conditions. The mixing factor, θ , may be defined as,

$$\theta = \frac{T_D - T_M}{T_R - T_M}$$

where

θ = mixing factor,

T_D = maximum temperature in the duct,

T_R = maximum temperature leaving refueling region,

T_M = mixed mean temperature.

Figure 7 shows the maximum temperatures in the steam generator inlet ducts for a range of mixing factors between 0.0 and 0.5, and for the Pelton circulators operating at 7000 rpm. Note that even with a mixing factor of 0.5, the critical safety limit is still not exceeded.

As an interesting sidelight, results from the DBDA at 7000 rpm are compared with those obtained from General Atomic's RECA¹ code. Figure 8 shows the comparisons for maximum fuel, maximum coolant outlet temperatures, and mixed-mean coolant outlet temperatures. Some of the differences in the two models can be inferred from this graph. Note that the maximum temperature curves are identical to about one hour in the transient. After this time there are sufficient gradients between refueling regions to reflect differences in inter-regional conduction modeling. The reasons the FLODIS temperatures peak higher than RECA temperatures are

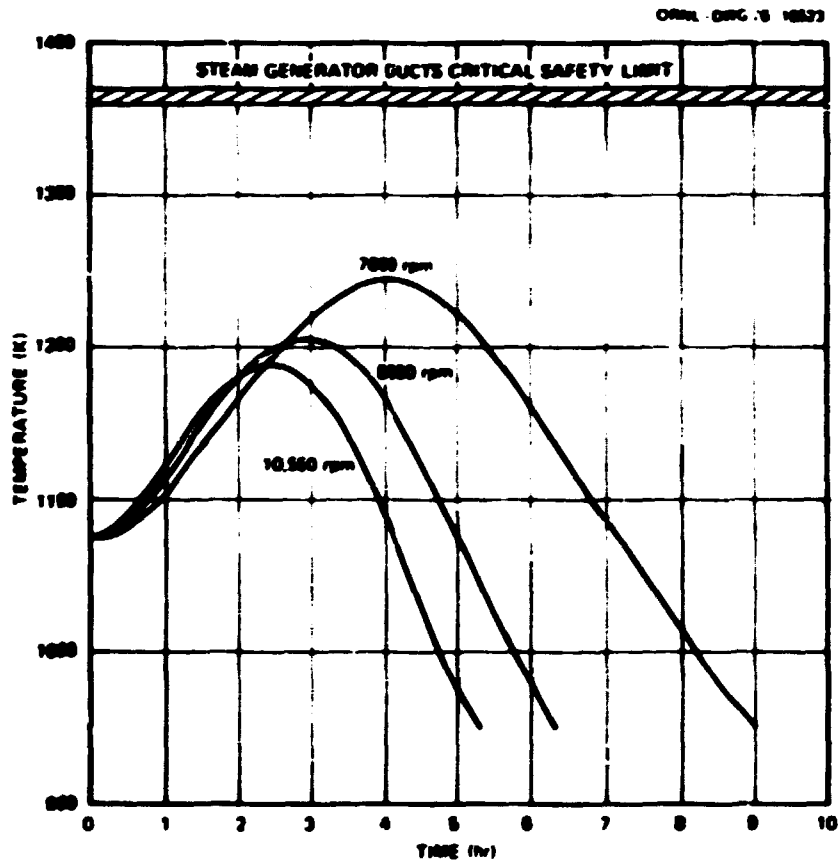


Fig. 6. Mixed mean coolant temperature during a DBDA for selected circulator speeds.

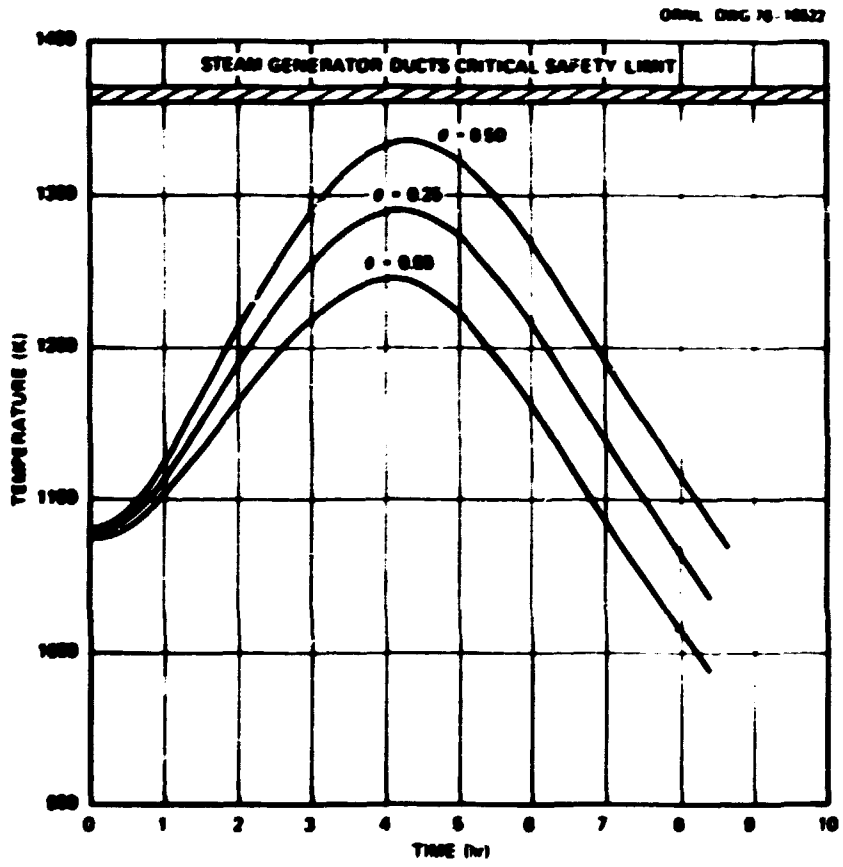


Fig. 7. Steam generator inlet duct maximum temperature for various mixing factors.

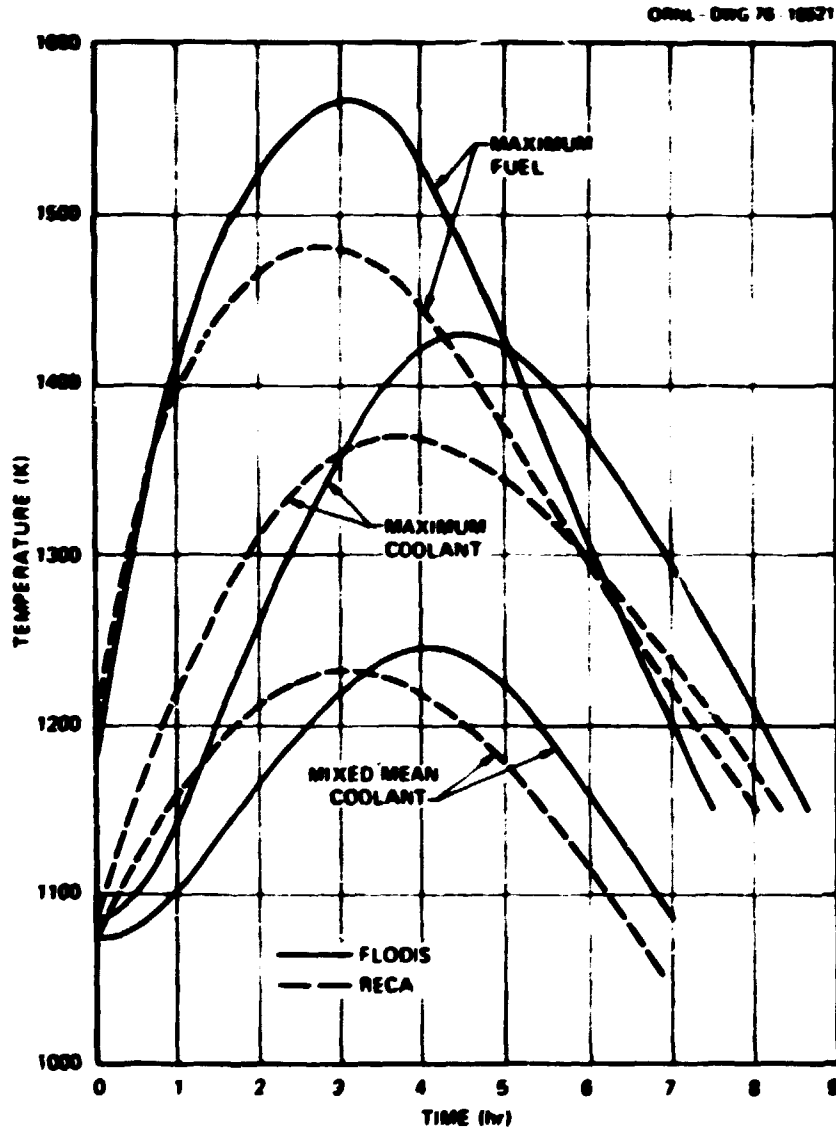


Fig. 8. Comparison of FLODIS and RECA temperature during DBDA with 7000 rpm circulator speed.

due to: (1) the fine mesh spacing allows gradients within a refueling region to be determined, and (2) the distribution of coolant within a refueling region according to the finer mesh spacing is more realistic of the actual physical process.

Note also that the coolant temperature curves vary significantly in shape early in the transient. Two reasons exist to explain this phenomena. First, the two codes use entirely different convective heat transport models. FLODIS uses the exponential approach method (Ref. 4) whereas RECA uses the endpoint weighting method (Ref. 2). The exponential approach method is valid from a physical standpoint and is numerically stable under all flow conditions. The endpoint weighting method is numerically stable but does not have a physical basis. Secondly, the modeling of the core support block is probably the cause of the differences in shape of the curve at the outset of the transient. FLODIS has 2 nodes each for the bottom reflector, upper level of the core support block, and lower level of the core support block for a total of 6 nodes. RECA lumps the same region into one node and therefore the coolant fails to see the axial temperature gradient which develops in this region after accident initiation.

Finally, the curves of the mixed-mean outlet coolant temperatures give an indication of the heat removal process during the transient. The areas under these curves are directly proportional to the heat removed from the entire reactor. RECA predicts considerably more heat removal earlier in the transient. This causes RECA fuel and coolant temperatures to peak at an earlier time than the corresponding FLODIS curves.

CONCLUSIONS

Results predicted by the FLODIS code for a DBDA show that the response of the system is such that the reactor and components maintain their integrity throughout the transient. In modeling the system two areas were noted where additional work could be done. At low flows the orifice loss coefficients have not been verified, and the mixing of outlet gas streams in the lower plenum has not been studied. Both of these

uncertainties are important for predicting the overall response of the Fort St. Vrain reactor core.

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