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THERMAL-HYDRAULIC INSTABILITIES IN PRESSURE TUBE GRAPHITE-MODERATED BOILING WATER REACTORS

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THERMAL-HYDRAULIC INSTABILITIES

IN PRESSURE TUBE GRAPHITE - MODERATED

BOILING WATER REACTORS

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ABSTRACT

Thermally induced two-phase instabilities in non-uniformly heated boiling channels in RBMK-1000 reactor have been analyzed using RELAP5/MOD3 code. The RELAP5 model of a RBMK-1000 reactor was developed to investigate low flow in a distribution group header (DGH) supplying 44 fuel pressure tubes. The model was evaluated against experimental data.

The results of the calculations indicate that the period of oscillation for the high power tube varied from 3.1s to 2.6s, over the power range of 2.0 MW to 3.0 MW, respectively. The amplitude of the flow oscillation for the high powered tube varied from +100% to -150% of the tube average flow. Reverse flow did not occur in the lower power tubes. The amplitude of oscillation in the subcooled region at the inlet to the fuel region is higher than in the saturated region at the outlet. In the upper fuel region and outlet connectors the flow oscillations are dissipated.

The threshold of flow instability for the high powered tubes of a RBMK reactor is compared to Japanese data and appears to be in good agreement. This work was sponsored by the United States Department of Energy under Contract DE-AC06-76RLO 1830.

INTRODUCTION

On March 24, 1991, the Unit 3 reactor at Leningrad Nuclear Power Plant, a 1000 MW pressure tube graphite moderated reactor, was automatically shutdown because of a pressure tube rupture in the upper part of the reactor core cavity [1,2]. The rupture occurred due to a failure of the inlet flow control valve to one of the core pressure tubes. It was estimated from a post-accident review, that this failure resulted in flow reduction of the inlet flow to less than 10% of the initial tube flow. The flow reduction initiated a fuel temperature excursion and also elevated the pressure tube wall temperature due to radiative heat transfer between the fuel rods and tube wall. Approximately 40-45 seconds after the inlet valve failure, the pressure tube ruptured. The reactor shutdown was initiated 3.7 seconds after the pressure tube rupture due to the high core cavity pressure.

Similar events are also possible for a partial break of the distribution group header, when quasi-stagnation or flow fluctuation at near zero pressure drop ΔP occurs. At this condition, the post-dryout heat transfer under low flow is not sufficient to prevent a pressure tube wall temperature excursion. The purpose of this paper is to validate RELAP5 for two-phase flow dynamic instability problems in RBMK reactors. The work includes two related accident analysis:

- Blockage of coolant at the pressure tube inlet.
- Blockage of DGH or partial break of DGH.

The general characteristics of the RBMK type reactor are as follows:

- Thermal core power 3200 MW.
- 1661 fuel tubes, 7 m active core, average linear heat flux 153 W/cm.
- Operating pressure 7 MPa.
- 37,600 tonne/hr total loop flow, an average of 6.288 kg/s per tube.
- 40 DGH with 42 pressure tubes in each.

The reactor has four steam drum separators, two hydraulic loops common at the steam header and 8 main circulation pumps (PCP), (6 operating, and 2 reserved).

RELAP5 Models for RBMK

Two RELAP5 models were developed that represent a 1/4 core and 1/2 core of an RBMK reactor. With these two models, minor modifications were made specific to the transient being simulated. For both models, the nodalization is setup to perform a detailed calculation of an affected core region (for a single or multiple tube rupture or blockage). The balance of the core is lumped into a single tube to allow the RELAP5 model to predict needed fluid conditions in the steam drum and inlet distribution headers. It was felt that a simple single tube model, with boundary conditions for these regions, would not allow sufficient degrees of freedom in the calculation to provide accurate results. The nodalization schemes for both models are shown in Figures 1 and 2. RBMK design data were provided by [1,2,3].

The 1/4 core model assumes a 1/4 core symmetry for the RBMK, contains two parallel fuel regions for the reactor core, and uses boundary conditions for the main coolant pump (Figure 1). A 1/4 core model is the minimum size needed to include a steam drum model, and is readily adaptable for assuming conservative core power distributions (i.e. assuming high/low power regions). The two fuel region model allows for one or more 'affected' tubes (fuel channels) to be modeled separate from the intact core for events such as tube rupture or blockage. The 1/2 core model contains four parallel fuel regions for the core, and a pump model to provide a complete loop simulation. The 1/2 core representation allows a more accurate calculation of the core average conditions as the RBMK core is split in-two hydraulically.

For both models, the affected tube is modeled hydraulically using 9 inlet connector volumes, 16 axial fuel volumes (14 active fuel regions), 6 upper tube volumes, and 5 outlet connector volumes. This nodalization allows for detailed pressure and temperature monitoring, and ease of defining the tube rupture location for different events without significant changes to the base model. The intact core is modeled using 5 inlet connector volumes, 7 axial fuel volumes (5 active fuel regions), 5 upper tube volumes, and 5 outlet connector volumes (these outlet connectors are set up to allow for future model expansions as needed). In the three channel model, a third channel is modeled hydraulically with the same detail as the affected core. Overall, the two channel model (1/4 core) represents 416 fuel channels, typically a single 'affected' channel and 415 lumped channels. The four channel model (1/2 core) represents 830 fuel channels, typically one or more 'affected' channels, two sets of parallel channels for the balance of the 44 tubes on one DGH, and the remaining 786 lumped channels.

The steam drum separator is modeled using 14 volumes. This is shown in Figures 1 and 2. This modelling detail allows for a more accurate inventory calculation, and in particular, a more accurate prediction of the fluid conditions for reverse flow into the affected tube(s). There are additional volumes for the inlet sparger volume (for feedwater return), an outlet downcomer for coolant return to the main coolant pumps (MCPs), and steam piping volumes leading to the turbines. The turbines and feedwater return pumps are not modeled explicitly. They are approximated using time dependent volumes to supply the necessary boundary conditions, with the fluid conditions taken from plant operating data. The steam drum is sized to represent a single drum for the 1/4 core model, and two steam drums for the 1/2 core model.

The heat structures modeled include the fuel pins and carrier rod, pressure tube and surrounding graphite, and the inlet and outlet connector piping walls. No heat structures are modeled at this time for the steam separator. The affected tube for both models contains two fuel pin heat structures that represent an equivalent of 6 and 12 fuel pins lumped together to represent the 18 fuel pins per rod bundle. This allows radial power peaking to be modeled for the 6 inner and 12 outer fuel rings of the fuel bundle. The unaffected tubes are modeled with a single heat structure representing an equivalent of 18 fuel pins lumped together.

For the affected core, the RELAP5/MOD3 radiative heat transfer model is used. Radiative heat transfer between the inner fuel ring, outer fuel ring, carrier rod, and tube wall is modeled. Appropriate view factors were calculated for each heat structure component. Preliminary calculations for the tube blockage event were made to investigate the surface emissivity for the fuel cladding and tube wall. Emissivity values of 0.5, 0.6 and 0.7 were evaluated. This range was considered to be typical for Zr (the cladding and tube wall material). An average value of 0.6 was chosen for the calculations presented here, as the preliminary results did not show a strong dependence over this range of emissivity.

The 1/2 core model was developed to investigate low flow induced oscillation (Figure 2). The model contains 4 core regions, three within the affected DGH representing 44 tubes, and one for the balance of the core. The three affected tube regions were defined as 4 high power tubes (ranging from 2.2 MW to 3.0 MW per tube), 18 medium power tubes (set at 2.2 MW per tube), and 22 low power tubes (set at 1.6 MW per tube). This distribution was based on previous work done at PNL for post-Chernobyl neutronics analysis [4].

The low flow condition for the affected DGH was simulated by defining a time dependent junction at the inlet to the DGH to provide the desired flow conditions. Total power for the 4 tube core region was set at a predetermined power for each case analyzed (2.2 MW, 2.4 MW, 2.6 MW, and 3.0 MW). The model was run to achieve a steady state solution for full power/full flow, and then flow reduced to the affected DGH slowly until the point of flow instability was seen. The point of flow instability was defined as an oscillation amplitude of +/-30%. Flow to the DGH was then held constant at the point of instability to observe the "stabilization" of the flow instability.

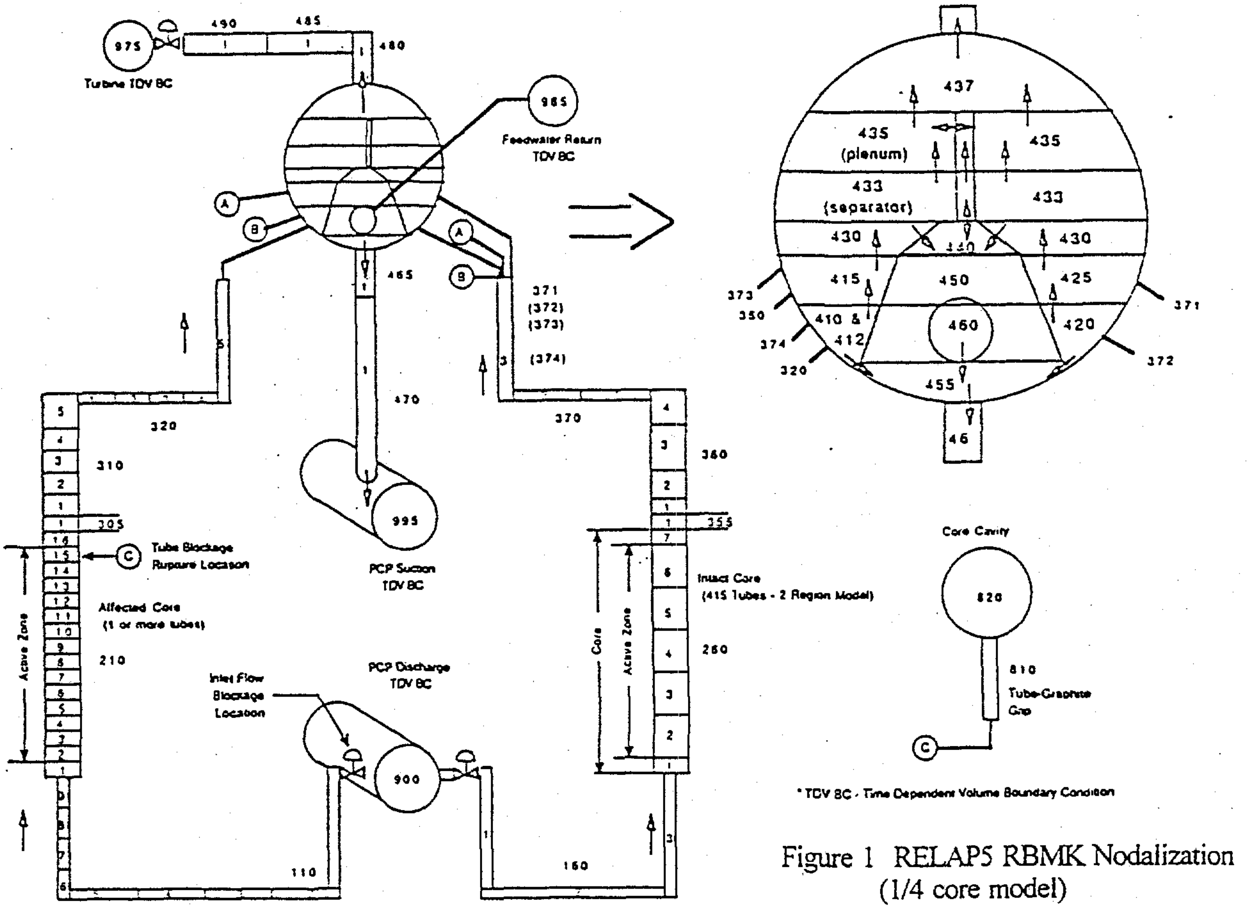

Figure 1 RELAP5 RBMK Nodalization
(1/4 core model)

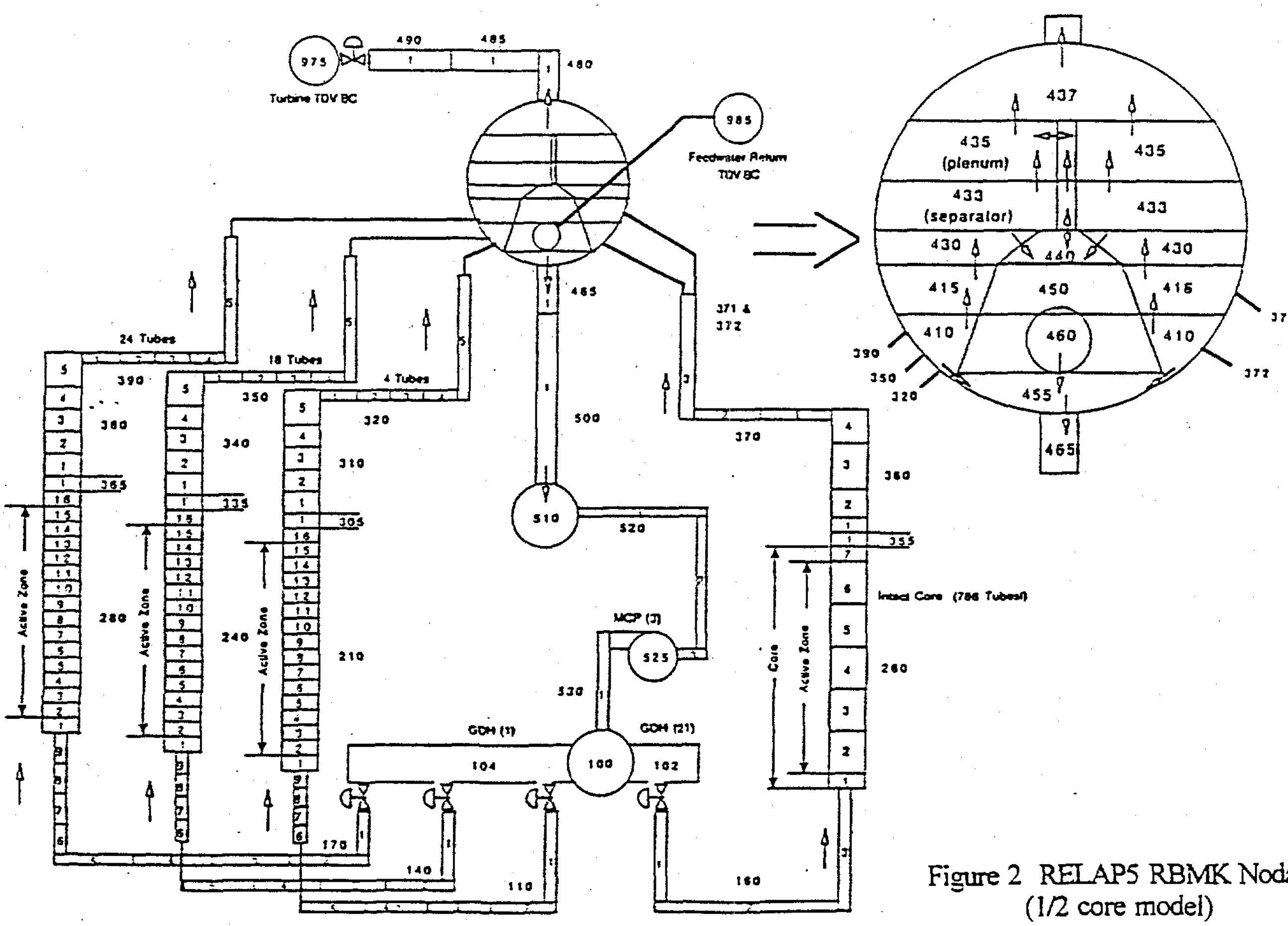

Figure 2 RELAP5 RBMK Nodalization
(1/2 core model)

CODE VERIFICATION

The first stage of verification includes calculation for steady-state parameters in the RBMK and some transient calculation against known experimental data. Limited results for code verification were presented at [5]. Results from the investigation of a tube blockage are presented.

The seconds stage of verification is for low flow instability. The RBMK calculations are compared against Japanese experimental data [6] for Type II threshold of flow instability. A sensitivity study of the RELAP5 model is included with the comparison.

RESULTS AND ANALYSIS

Tube Blockage

A series of tube blockage cases were evaluated with the 1/4 core model. Briefly, the Leningrad tube rupture was initiated by a failure of the inlet flow control valve to one of the core pressure tubes. It was estimated from a post-accident review, that this failure resulted in a flow reduction of the inlet flow to less than 10% of the initial tube flow. The flow reduction initiated a fuel temperature excursion and also elevated the pressure tube wall temperature due to radiative and convective heat transfer between the fuel and the pressure tube wall. Approximately 40-45 seconds after the inlet valve failure, the pressure tube ruptured in the upper core. A reactor shutdown was initiated 3.7 seconds after the pressure tube rupture due to the high core cavity pressure.

To evaluate this event, a parametric study was performed over the potential range of inlet flow blockage. Each calculation assumed an instantaneous reduction in the inlet valve flow area to simulate the valve failure of the Leningrad event. A total of five calculations were made, varying the inlet flow blockage to obtain a range of flow reduction between 2%-10% of the initial tube flow. Initial tube flow was 6.3 kg/s. Fuel cladding and pressure tube wall temperatures were evaluated every 0.5m with the 1/4 core model. It was assumed that pressure tube failure (rupture) would occur at an average tube wall temperature of 923K (650°C). This is the temperature at which tube softening is estimated to occur that then results in tube rupture. With 0.5m volume nodalizations for the 7m active fuel region, the tube failure location was calculated to be either at the 6.25m or 6.75m core elevation, depending upon the individual case. A plot of the time to pressure tube failure was made for the five calculations, and is shown in Figure 3a. A minimum time to tube rupture of approximately 42 seconds was calculated (compared to the estimated time of 40-45 seconds).

An evaluation was also made of the general transient response, with reactor shutdown, for one of the calculations. Figures 3b through 3d show the results from an approximate 6% flow blockage calculation. Initial tube flow for this calculation was reduced from 6.3 kg/s to 0.38 kg/s. In each of these figures, a null transient is run to ensure steady state conditions have been reached prior to initiation of the blockage (e.g., the blockage occurs at 30 seconds). Figure 3b is the pressure tube inlet flow response. Figure 3c is the fuel cladding temperature response, and Figure 3d the pressure tube inner wall temperature response, for selected nodes. Refer to Figure 1 for the nodalization numbers. For this calculation, the tube was estimated to rupture 42 seconds after initiation of the blockage. A reactor shutdown was initiated 3.7 seconds after the rupture, resulting in the eventual quenching of the fuel cladding and pressure tube wall. Peak cladding temperature was calculated to be 1490K (1217°C) for this case. The general responses for this transient appear to be physically consistent.

Blockage of DGH

Low flow, high power instabilities were investigated using the 1/2 core model shown in Figure 2. The instability is initiated by reducing flow to the affected DGH, using a time dependent junction (simulating a partial blockage), while maintaining constant power. A nodalization and time step size sensitivity study was also performed. The results of the sensitivity study are presented first.

Sensitivity Study

Three areas of modeling sensitivity were investigated. These were core nodalization, outlet (steam) pipe nodalization, and time step size. The core nodalization study investigated three fuel region nodalizations; 7, 14 and 28 axial fuel nodes. The steam pipe nodalization study investigated three steam outlet pipe nodalizations; 2, 5 and 10 steam pipe nodes. The time step study was performed for three different time steps sizes; 2ms, 10ms and 12.5ms, for two different core nodalizations, 7 and 14 fuel region nodes. For the two nodalization studies, the time step size used was 12.5ms. This time step size is the inherent RELAP5 Courant limit for the model.

The nodalization study was performed by initializing the model with a 60 second null (steady state) transient, then reducing flow to the affected DGH from 276.5kg/s to 50kg/sec between 60 and 560 seconds. The 50kg/s flow is then maintained constant from 560 to 660 seconds to observe the flow instability. The time step study was performed by initializing the model with a 60 second null transient, then reducing flow to the affected DGH from

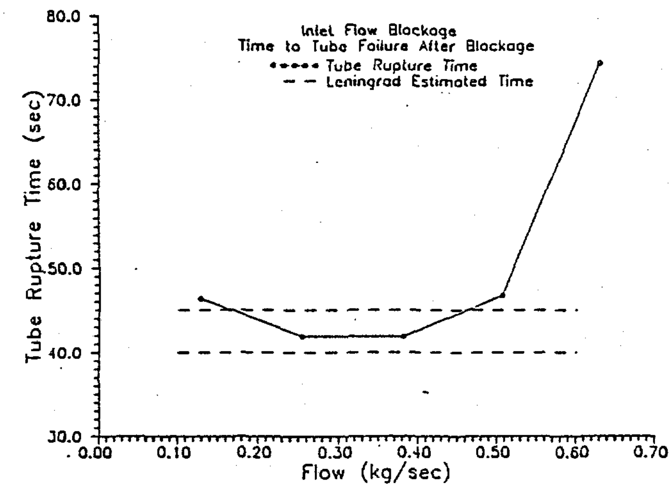

Figure 3a Time to Failure

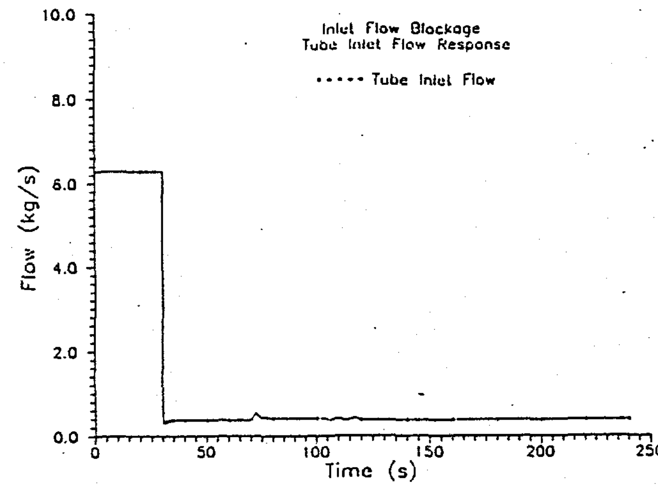

Figure 3b Inlet Flow

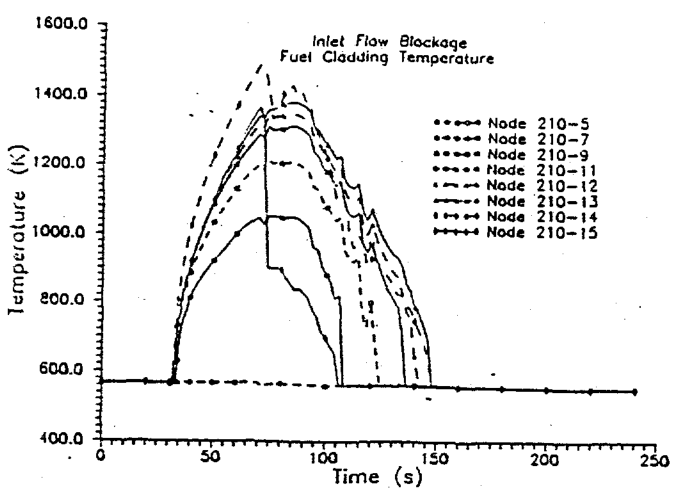

Figure 3c Cladding Temperature

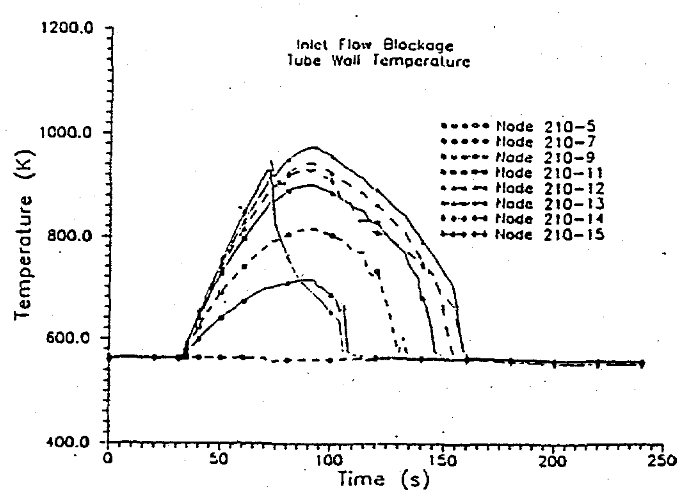

Figure 3d Tube Wall Temperature

276.5kg/s to 60kg/sec between 60 and 120 seconds. The 60kg/s flow is then maintained constant from 120 to 180 seconds to observe the flow instability (a flow of 60kg/s was chosen as this was closer to the point of instability for the DGH than 50kg/s).

The core region nodalization study was performed for three noding schemes; 7, 14 and 28 fuel region nodes. The results are shown in Figures 4a and 4b. The 14 and 28 fuel node results behave very similarly. They exhibit initiation of flow instability at very nearly the same flow, and although they differ slightly during the first 10 seconds of instability (Figure 4a, from 450 to 460 seconds), once the instability has reached a stable period they maintain similar frequencies of oscillation (although off-set slightly). The 7 node results, however, show a significantly lower point of instability initiation (Figure 4a) and frequency of flow oscillation (Figure 4b). The 7 and 14 node results do show similar amplitudes of oscillation, with the 28 node results showing a larger amplitude.

The steam pipe nodalization study was performed for three noding schemes; 2, 5 and 10 steam pipe nodes. The results are shown in Figure 5b. The results for the 5 and 10 steam pipe nodes behave very similarly. They exhibit similar initiation of flow instability and period of oscillation, differing in amplitude of oscillation only slightly during the first 10 seconds of instability (Figure 5, from 450 to 460 seconds). Once the instability has reached a stable period, they maintain similar flow oscillation amplitude and frequency. The 7 node results, however, show a significantly lower point of instability initiation, and frequency of flow oscillation. All three cases show similar amplitudes of oscillation.

The time step sensitivity study was performed using the 7 and 14 node fuel region models (with 5 steam pipe nodes), and three different time step sizes for each; 12.5ms, 10ms, and 2ms. Figures 6a and 6b compare the time step study for the 7 and 14 node fuel regions, respectively. The 7 node model shows minor deviations between the 12.5ms and 10ms, with more significant deviations in the period of oscillation for 2ms. The 14 node model shows excellent similarity for all three time step sizes.

Blockage Calculation

The base nodalization for the DGH partial blockage was for a 14 node fuel region, a 5 node steam pipe region, and a 12.5ms time step size. The instability study was performed by initializing the model with a 60 second null transient, then reducing flow to the affected DGH from 276.5kg/s to just above the point of flow instability in 20 seconds. This point was determined with preliminary calculations for each case evaluated. The DGH flow was then slowly reduced over 40 seconds to point of instability, and then maintained constant. Four different high tube powers were evaluated; 2.2MW, 2.4MW, 2.6MW, and 3.0MW.

The results of the calculations indicate that the period of oscillation for the high power tube varied from 3.1s to 2.6s, over the power range of 2.2MW to 3.0MW. This is shown in Figure 7a. The amplitude of the flow oscillation for the high powered tube varied from +100% to -150% of the tube average flow (based on the "steady state" flow just prior to initiation of the flow instability). Figures 7b, 7c and 7d present the results for one of the cases evaluated, a tube power of 2.4MW. The lower power core regions of the affected DGH experienced the same period of oscillation, but with lower amplitude. They also did not experience reverse flow. In addition, the lowest powered core region experienced flow oscillations of smaller amplitude than the medium powered core region.

The amplitude of oscillation was referenced to the inlet flow of the fuel region, Figure 7b. In the upper fuel regions and outlet connector, the amplitude of the flow oscillation was dissipated in the upper regions of the core.

The fuel cladding and tube wall temperatures were monitored for three core elevations; the lower core, mid-core, and upper core (Figures 7c and 7d). The magnitude amplitude of the cladding temperature oscillation varied from +/-40 to +/-70K over the range of tube powers from 2.2 MW to 3.0MW, respectively. In the lower core region (node 3), temperature oscillations show alternation of wet and post-dry-out zones. In the upper core regions, where post-dryout has already occurred, temperature oscillations are due to flow and heat transfer coefficient changes. The amplitude of the tube wall temperature oscillation varied from +/-10 to +/-20K over the same range of power. Although the calculations were run long enough to produce a "stable" flow oscillation, the cladding and tube wall temperatures oscillations had not yet reached an "equilibrium" condition. For the highest power analyzed, 3.0MW, the tube wall temperature had nearly reached an "equilibrium," averaging approximately 805K, with an oscillation amplitude of +/-20K. The critical temperature for the RBMK pressure tube for tube rupture has been determined to be approximately 923K (650^{0}C). Additional calculations are needed to evaluate the potential for tube rupture. Cladding temperature is far below the critical temperature for oxidation (1473K).

The results of the calculation clearly indicate that dryout in the upper regions of the core will occur prior to oscillation of the cladding temperature. Cladding temperature rises slowly in the upper core after initiation of the flow instability, then temperature rises sharply at the dryout point (Figure 7c) and reaches a new "equilibrium" temperature (the critical heat flux of the second mode) that continues to slowly rise. The cladding temperature

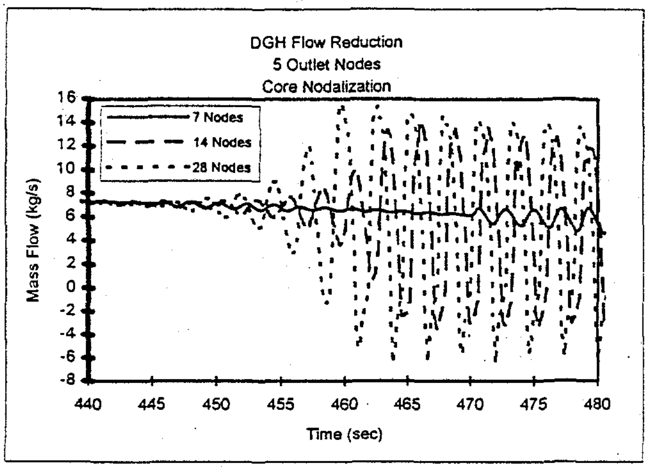

Figure 4a Core Nodalization

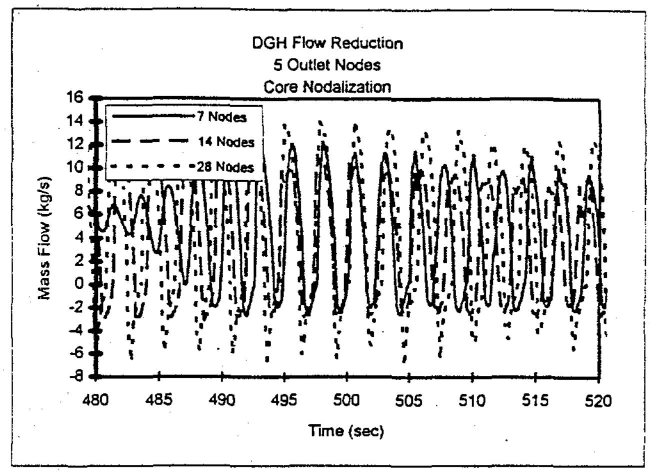

Figure 4b Core Nodalization

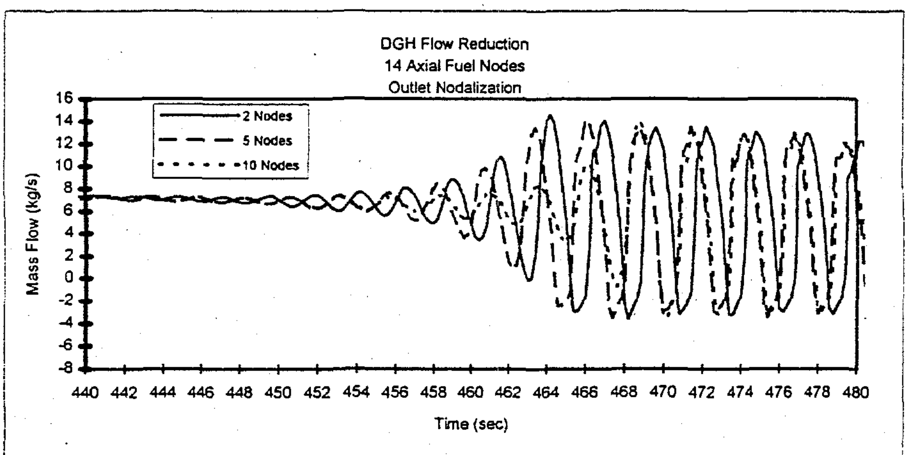

Figure 5 Outlet Nodalization

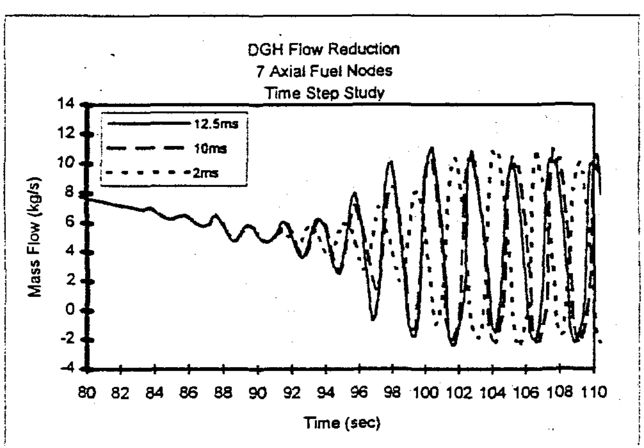

Figure 6a Time Step - 7 Fuel Nodes

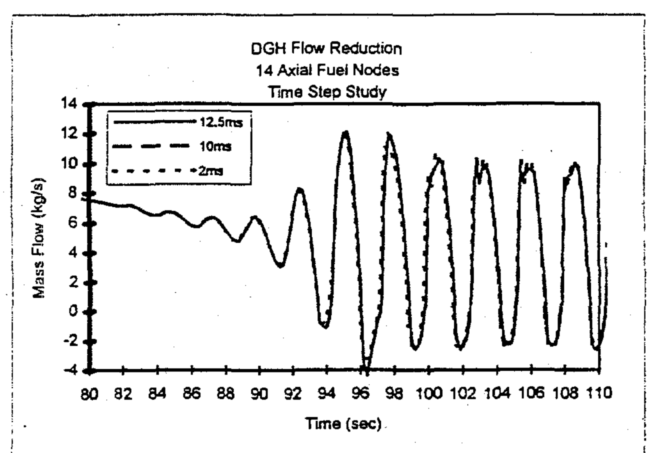

Figure 6b Time Step - 14 Fuel Nodes

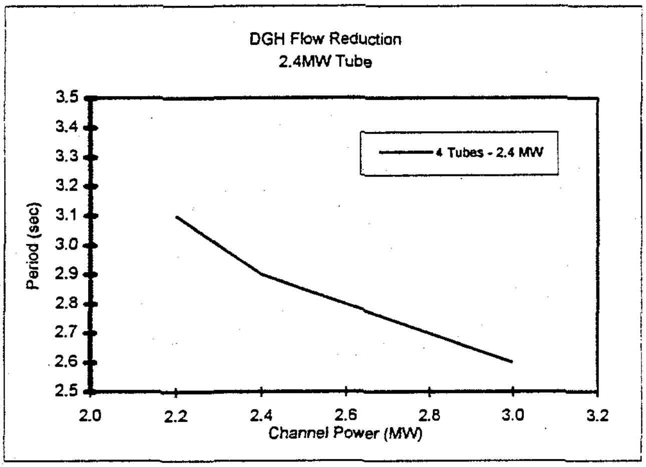

Figure 7a Period of Oscillation

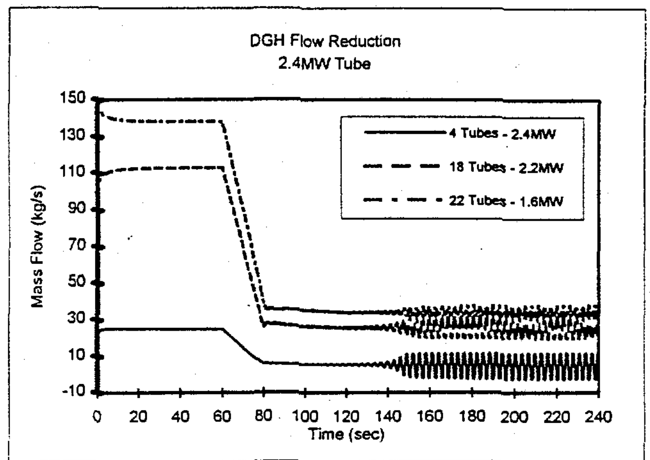

Figure 7b Tube Inlet Flow

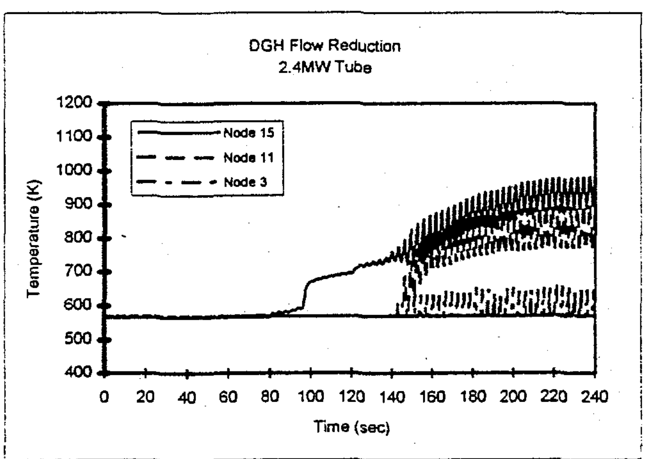

Figure 7c Cladding Temperature

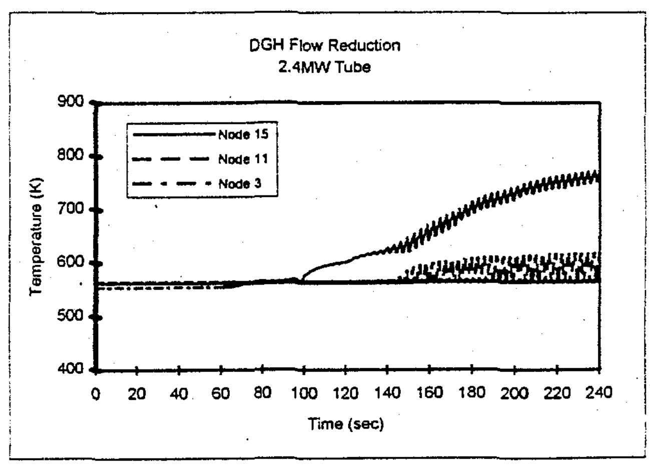

Figure 7d Tube Wall Temperature

oscillation is induced by the continued flow oscillation and moving of the boundary between the dry region and liquid.

The threshold of flow instability was calculated for each of the different powers for the high powered tube. These were compared to the data presented in Figure 8, Mochizuki [6]. The calculated threshold for the RBMK-1000 model appears to be in good agreement with this data. The data presented in Figure 8 suggest that for a DGH with high powered tubes, Type II instability is reached if flow is reduced below 1 to 2 kg/s, over the power range of 2.2 to 3.0 MW, respectively. These calculations were made for a limited power-flow range, and it is necessary to continue the analysis for flows less than 1 kg/s and powers less than 2.2 MW.

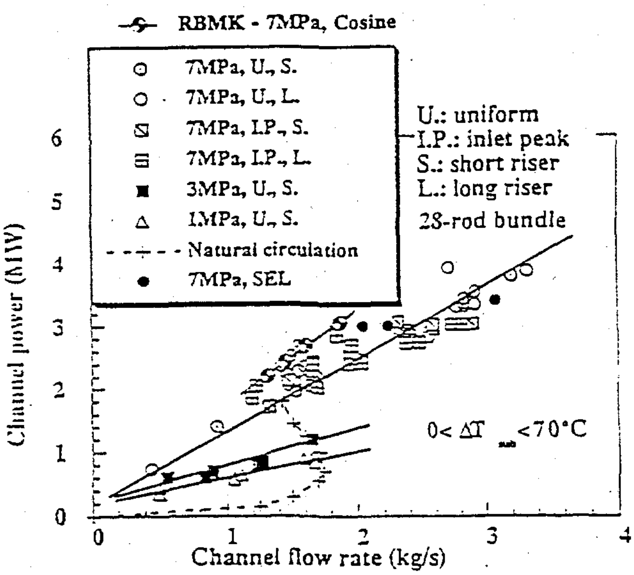

Figure 8 Type II Instability Threshold

CONCLUSIONS

Results of a single tube blockage show good agreement with the available data for the Leningrad tube rupture event. The model was able to reasonably predict the time of tube wall failure for the expected flow blockage. Comparison of the threshold of Type II flow instability shows reasonable agreement over the range of RBMK tube power investigated, and can potentially be used for safety analyses of the DGH blockage events. Modeling sensitivity studies indicate the instability analysis results were not sensitive to the nodalization scheme and time step sizes used. This was for a 14 node fuel region, 5 node outlet (steam) pipe region, and a time step size of 12.5ms. For this nodalization, there was little sensitivity to time step between 2ms and 12.5ms. The results do indicate that fewer than these number of nodes in these two regions can significantly effect the results.

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