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SIMULATION OF THE CHERNOBYL ACCIDENT

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An analysis of the April 26, 1986 accident at the Chernobyl-4 nuclear power plant in the Soviet Union is presented. The peak calculated core power during the accident was 550000 MW_t. The analysis provides insights that further understanding of the plant behavior during the accident. The plant was modeled with the RELAP5/MOD2 computer code using information available in the open literature. RELAP5/MOD2 is an advanced computer code designed for best-estimate thermal-hydraulic analysis of transients in light water reactors. The Chernobyl-4 model included the reactor kinetics effects of fuel temperature, graphite temperature, core average void fraction, and automatic regulator control rod position. Preliminary calculations indicated the effects of recirculation pump coast down during performance of a test at the plant were not sufficient to initiate a reactor kinetics-driven power excursion. Another mechanism, or "trigger" is required. The accident simulation assumed the trigger was recirculation pump performance degradation caused by the onset of pump cavitation. Fuel disintegration caused by the power excursion probably led to rupture of pressure tubes. To further characterize the response of the Chernobyl-4 plant during severe accidents, simulations of an extended station blackout sequence with failure of all feedwater are also presented. For those simulations, RELAP5/MOD2 and SCDAP/MOD1 (an advanced best-estimate computer code for the prediction of reactor core behavior during a severe accident) were used. The simulations indicated that fuel rod melting was delayed significantly because the graphite acted as a heat sink.

1. Introduction

Chernobyl, Unit 4, was a 3140 MW_t (1000 MW_e), RBMK-1000 class reactor operated in the USSR prior to a severe accident on April 26, 1986. The reactor was a boiling-light-water-cooled, graphite-moderated type, and contained 1661 vertical zirconium pressure tubes within the graphite stack. Each tube contained 18 fuel rods with a 7-m heated length. The reactor coolant system consisted of an intricate arrangement of pressure tube inlet and outlet connectors, horizontal drum-type steam separators, condensate inlet and outlet headers, and recirculation pumps. The following description of the accident sequence is a condensation of information provided in the Soviet report on the accident [1].

The accident occurred during the performance of a plant test designed to investigate pumped recirculation flow response following isolation of steam from a turbogenerator. Under the procedures for the plant test, half of the recirculation pumps were powered by the affected turbogenerator. At the initiation of the test, the shutdown control valves of the affected turbine were closed, causing the turbine to coast down. As a result, electrical output of the associated generator declined. Since half of the recirculation pumps were powered by this generator, the output of the pump motors, and hence the core flow, declined accordingly.

As the core flow declined, the core average void fraction increased. An increase in core power resulted because the coolant void reactivity coefficient is positive in the RBMK-1000 reactor design. Sufficient and timely control rod negative reactivity was not available to compensate for the increase in core power because of a series of operator actions. This caused a runaway power excursion that destroyed the plant.

The combination of many factors was required to provide the conditions necessary for the power excursion. The plant test was intended to be performed at a core power of 700 to 1000 MW_t. The operators attempted to achieve a power level in this range; however, control of the reactor proved elusive and, in the interim, core power is reported to have fallen to 30 MW_t. Eventually, the operators were able to achieve a relatively stable operating power of 200 MW_t. As a result of operating at very low powers for extended periods, xenon poisoning of the reactor was significant. The operators responded by decreasing the reactivity reserve (by withdrawal of manual control rods) below its ac-

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Table 1
Chernobyl accident sequence of events

Local time April 26, 1986	Event
1:19:00	The reactor power is stabilized at 200 MW_t with 57000 m^3/hr core flow.
1:19:10	To remedy a low separator level indication, the feedwater flow is increased.
1:19:30	Feedwater flow is stabilized at about 4 times its initial rate. Manual regulator control rods are withdrawn as necessary to compensate for the resulting decrease in core average void fraction.
1:22:00	Feedwater flow is decreased. Desired separator level has been achieved.
1:22:30	Feedwater flow stabilized near its initial rate.
1:23:04	Test begins. Shutdown control valves to affected turbine generator are closed. Turbine and recirculation pump coastdowns begin.
1:23:40	The AZ-5 push button (manual reactor trip) is depressed.
1:24:00	Two explosions are reported to destroy the plant.

ceptable range. Furthermore performing the test at 200 MW_t required the operator to disengage the local automatic control (LAR) rods. Operation of all recirculation coolant pumps simultaneously resulted in a violation of regulations intended to avoid pump cavitation. The operators elected to defeat automatic safety systems affecting reactor trip and emergency coolant injection to prevent their spurious operation during the test. Thus, at the beginning of the test, the reactor was operating outside acceptable limits for reactivity reserve and pumped recirculation flow, and with its reactor protection and safety injection functions disabled.

A detailed sequence of events starting at 1:19:00 a.m. on April 26, 1986 (4 min, 4 s before the beginning of the test) is listed in table 1. At 1:23:40, the operator initiated a manual reactor trip. The reactivity response time is about 3 s. The accuracy of the reported time of the explosion in table 1 is questionable because if the reactor trip was initiated at 1:23:40, then by 1:24:00 negative reactivity insertion should have been sufficient to prevent a power excursion.

The effort reported herein consisted of assembling a RELAP5 model of the Chernobyl-4 reactor and analyzing the accident sequence to provide an understanding of plant response during the accident. In addition, to further understand the Chernobyl-4 plant response during another type of severe accident, a combined RELAP5 and SCDAP analysis was performed for a hypothetical station blackout sequence with failure of all feedwater.

Results from the accident simulation and station blackout analyses are presented in this paper. Precursors leading to conditions at the time of the accident are also discussed.

2. Model descriptions

This section describes the RELAP5/MOD2 [2] and the SCDAP/MOD1 [3] computer code models of the Chernobyl-4 plant. The models were constructed with information obtained from refs. [1], and [4] through [12]. Ref. [4] provided the most information on the design of the reactor core region. Ref. [5] provided fuel assembly details. Ref. [6] provided information on conditions expected during plant operation. Ref. [7] provided most of the information on the coolant loop design. In some instances, data provided in the references proved inconsistent. Ref. [11] provided an information standard from which the inconsistencies could be resolved. The RELAP5 model is described first, followed by the SCDAP model.

2.1. RELAP5/MOD2 model description

RELAP5/MOD2 is an advanced thermal-hydraulic computer code designed for best-estimate analysis of transients in light water reactors and related experimental systems. The code, which is based on a two-fluid hydrodynamic model, allows unequal phasic velocities and temperatures within calculational cells. RELAP5 models represent the physical processes of critical flow, vertical and horizontal stratified flow, and heat transfer from structures. The code also has models representing reactor point kinetics, control systems, and components such as valves, pumps, and accumulators.

The RELAP5 model consists of a single coolant loop representation of the 3140 MW_t light-water-cooled, graphite-moderated, Chernobyl-4 reactor. The model contains 29 hydrodynamic volumes, 29 hydrodynamic junctions, and 38 heat structures. A nodalization diagram of the model appears in fig. 1.

The Chernobyl-4 reactor core contained 1661 vertically-oriented pressure tubes surrounded by graphite

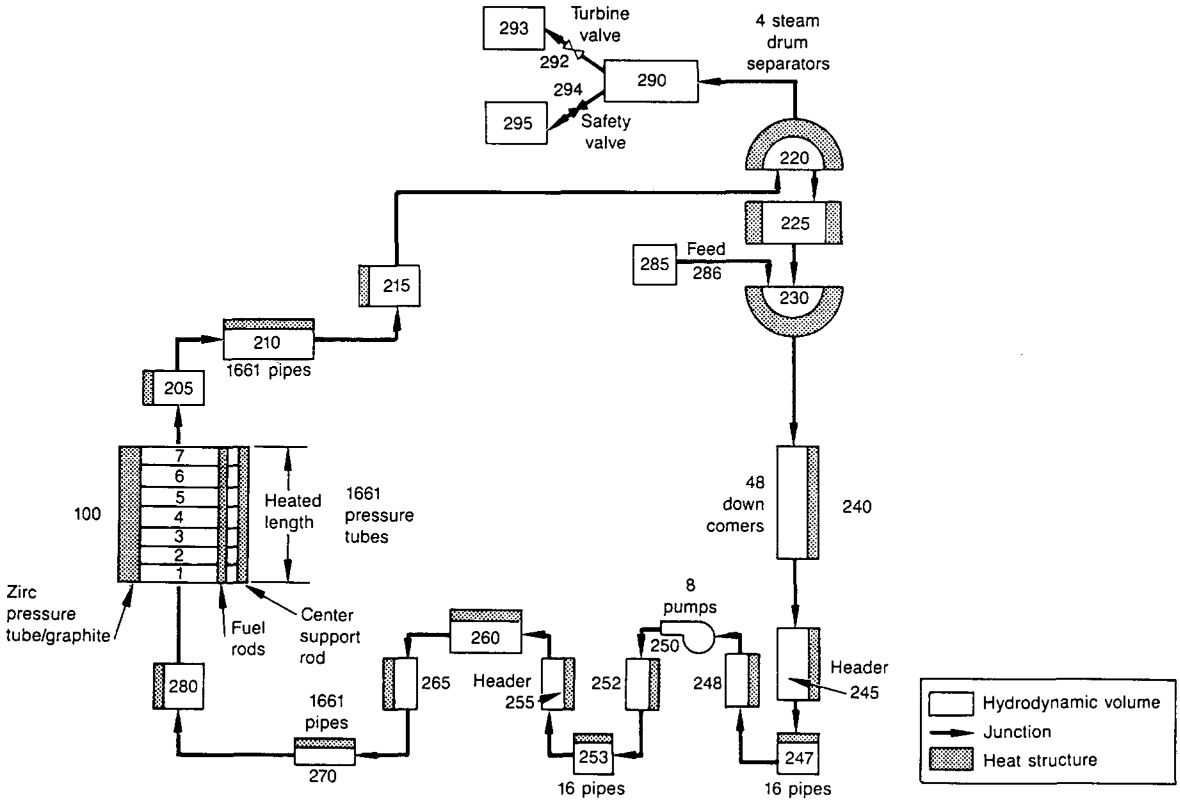

Fig. 1. Nodalization of the RELAP5 Chernobyl model.

moderator blocks. The core is represented by a RELAP5 pipe component with seven equal-height cells over the span of the fuel rod heated length. The six core internal junctions simulate the flow losses through grid spacers. Extra form losses were added at the internal core junctions to achieve the desired pressure drop across the pressure tube. Heat structures attached to each of the core hydrodynamic cells represent the fuel rods, pressure tube wall/graphite moderator, and central support rods within the pressure tubes.

The fuel rods are simulated by seven equal-height heat structures with a convective boundary condition on the outside surface. These heat structures represent the UO_2 fuel, gap, and zirconium cladding arrangement of the fuel rod. The fuel rod gap conductance was adjusted to achieve the desired average fuel temperature. The fuel rod axial power profile is a chopped cosine shape with a peak-to-average power ratio of 1.40.

The zirconium pressure tube and graphite are also simulated with seven equal-height heat structures. Cover gas and control rod cooling of the graphite are not modeled. The heat structures are cylindrical representations of the zirconium tube, the gas-filled gap, and graphite arrangement. The total graphite volume modeled is 966.8 m^3 with a total graphite mass of 1.615×10^6 kg. These heat structures have convective inside and adiabatic outside boundary conditions. Power distribution over the graphite thickness is assumed to be flat and the axial power profile within the graphite is identical to that described above for the fuel rods. The graphite gap conductance was selected to provide the desired average graphite temperature. At full power, 95% of core thermal power is produced within the fuel rods and 5% within the graphite.

Graphite thermal conductivity depends upon the physical properties of the graphite. The exact composition of the graphite in the Chernobyl-4 plant is not known. The thermal conductivity used in this model (33.81 W/m-K) represents the average of the three conductivities given in ref. [9] for (1) homogeneous, (2) extruded/parallel, and (3) extruded/perpendicular graphite irradiated to saturation at 873 K.

RELAP5 allows the computation of core power based on a point reactor kinetics formulation. Four reactivity components are considered in the Chernobyl-4 RELAP5 model: fuel temperature (Doppler), graphite temperature, core average coolant void fraction, and control rod position. A delayed neutron fraction of 0.005 and a neutron generation time of 1.785×10^{-5} s are assumed in calculating core power.

Reactivity feedback coefficients, obtained from ref. [1], are applied against core average quantities for fuel temperature, graphite temperature, and coolant void fraction at each axial level. A total negative reactivity of 0.0042 was assumed for the 12 automatic regulator control rods and is included based on information for the Leningrad-1, RBMK-1000 reactor. During the RELAP5 calculation of the Chernobyl-4 accident sequence, the automatic regulator control rods were modeled by assuming that they attempted to control the core power at 200 MW_t.

The coolant loop model consists of the physical components of the reactor from the heated core outlet through the steam drum separators and recirculation pumps, and back to the heated core. While the plant loop design is complex, the RELAP5 model is simplified by combining parallel flow paths and components (piping, steam drums, and pumps) into a single coolant loop. In creating the input, standard RELAP5 modeling techniques were used.

Three volumes and heat structures represent the 1661 pipes from the heated core section to the steam drum separators (see fig. 1). Three vertically-oriented volumes model the steam drum separators. Steam separation occurs within the top volume. The steam leaves the separator through the steam lines (component 290) and enters the turbines (Component 293) through the turbine valves (Component 292). Valve 294 simulates the steam safety valves. Feedwater enters the lower steam drum cell. Heat structures simulate the stainless steel walls of the steam drums. Forty-eight downcomer pipes connect the drums to four pump inlet headers. Sixteen pump inlet pipes connect the headers to eight recirculation pumps. Four RELAP5 components represent this configuration as shown in fig. 1. Cylindrical heat structures represent the stainless steel pipe walls in this region.

A RELAP5 pump component represents the eight recirculation pumps in the plant. Ref. [7] provides data on pump rated speed, head, and flow. Since a minimum of pump information was available, the pump model uses additional rated conditions from typical U.S. light water reactor coolant pumps and the RELAP5 built-in Westinghouse pump homologous curves.

Sixteen pipes connect the pumps to the pump outlet headers and 1661 pipes connect the headers to the heated core inlet. Seven RELAP5 components represent this portion of the plant. The inlet junction to Component 255 is a check valve to prevent backflow to the pumps.

2.2. Full-power steady state

The RELAP5 Chernobyl-4 model was driven to full power steady state conditions using steady state controllers. A constant pressure condition represented the turbine inlet, and the pump speed was adjusted to provide the desired core coolant flow. Table 2 compares RELAP5/MOD2 full power conditions with actual plant operating conditions. The comparison indicates the RELAP5 model well represents the thermal-hydraulic conditions of the Chernobyl-4 reactor during steady full-power operation.

2.3. SCDAP Chernobyl fuel channel model description

The Severe Core Damage Analysis Program (SCDAP/MOD1) is a best-estimate calculational tool for predicting light water reactor core behavior during severe accidents. SCDAP focuses on determining the physical, thermal, and chemical state of the core during such accidents. The code is capable of simulating phenomena such as core heatup (including radiation heat transfer), cladding deformation and relocation, and hy-

Table 2
RELAP5 – Calculated and desired full-power steady state conditions

Parameter	Desired	RELAP5
Core inlet temperature (K)	543.0	544.8
Core outlet quality	0.145	0.141
Core thermal power (MW)	3140.0	3140.0
Fuel rod power (MW)	2983.0	2983.0
Graphite power (MW)	157.0	157.0
Graphite average temperature (K)	873.0	872.1
Graphite peak temperature (K)	1023.0	1020.6
Fuel average temperature (K)	819.0	813.1
Core coolant flow (kg/s)	10417.0	10733.0
Pump inlet pressure (MPa)	7.186	7.208
Steam separator pressure (MPa)	7.0	7.019
Pump head (MPa)	1.50	1.59
Reg. valve differential pressure (MPa)	0.61–1.39	0.92
Pressure tube differential pressure (MPa)	0.66	0.53
Feedwater temperature (K)	443.0	443.0
Feed/steam flow rate (kg/s)	1525.0	1525.0
Separator liquid volume (m^3)	unknown	222.0 [a]

[a] The liquid volume in the model represents steam drum separators that are approximately 2/3 liquid-filled.

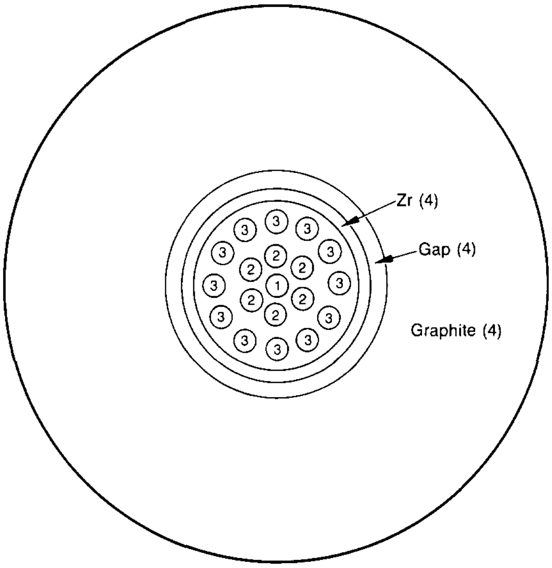

Fig. 2. Nodalization of the SCDAP Chernobyl model.

drogen generation. SCDAP also simulates cladding melting, UO_2 dissolution and relocation, and fission product release.

A single fuel channel in the Chernobyl-4 reactor is represented by the SCDAP model. The same modeling philosophy and dimensions used in the RELAP5 model are also used in the SCDAP model where possible. Fig. 2 shows a nodalization diagram of the SCDAP model. The model represents the central support rod and the eighteen fuel rods in two vertically-stacked subassemblies, the zirconium pressure tube, and the graphite block surrounding one pressure tube. These components are modeled in detail through the entire heated length of the fuel channel. Simple hydrodynamic models represent the regions above and below the heated length. Thermal-hydraulic initial and boundary conditions calculated by the RELAP5 model drive the SCDAP model.

Four groups, or components divide the fuel channel for computation purposes. These components correspond to the central support rod, the inner ring of six fuel rods, the outer ring of 12 fuel rods, and the surrounding zirconium pressure tube and graphite moderator. The components are linked together for radiation heat transfer calculations by view factors and path lengths. The support rods, fuel rods, and the inner surface of the pressure tube are all exposed to the same coolant flow conditions.

The central support rod was modeled with one component to allow representation of both the zirconium alloy tube and the stainless steel support rod inside this tube. The eighteen fuel rods in the fuel channel were modeled with two groups. One group models an inner ring of six fuel rods. The other group simulates an outer ring of twelve fuel rods. This division is made to account for possible radiation heat transfer differences between the inner and outer rods of the fuel assembly. The zirconium alloy pressure tube, gap, and graphite moderator were modeled with one SCDAP component divided into three layers. For the transient SCDAP calculation, the shroud is unpowered and the outer surface is assumed to be an adiabatic boundary.

3. Analysis of the Chernobyl accident

Section 3.1 presents results of the RELAP5 calculation of the Chernobyl accident sequence. Section 3.2 provides a comparison of code-calculated and plant data during the accident. Coolant pump cavitation effects are discussed in section 3.3 and alternate trigger mechanisms for the power excursion are discussed in section 3.4.

3.1. RELAP5 calculation of the Chernobyl accident

The RELAP5 model described in section 2 was used to perform a simulation of the Chernobyl accident sequence. First, a steady state calculation was performed to achieve conditions consistent with those described in the Soviet report [1] at time 1:19:00. These known conditions include a core power of 200 MW_t, a core flow rate of 15.74 m^3/s, a steam separator pressure of 6.849 MPa, steam and feedwater flow rates of 97.8 kg/s, and a feedwater temperature of 438 K. Other RELAP5-calculated conditions were: core average and

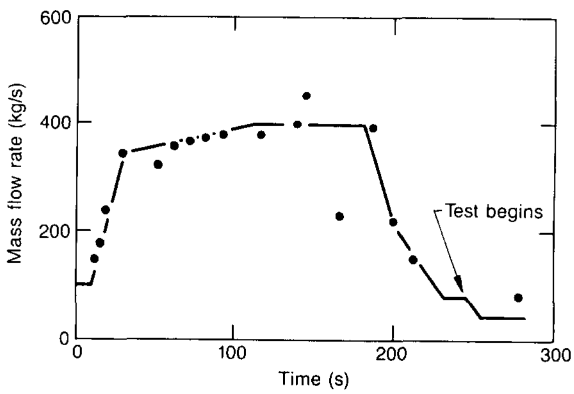

Fig. 3. Feedwater flow during the Chernobyl accident.

exit void fractions of 1.72% and 7.04% respectively, average fuel and graphite temperatures of 572.1 K and 578 K respectively, a recirculation pump head of 1.724 MPa, and subcoolings of 2.42 K at the pump inlet and 5.68 K at the core inlet. The ability of the RELAP5 model to match the broader range of known full power steady conditions (as presented in section 2) lends confidence to using the model in this low power application.

In the discussion of the accident sequence that follows, time zero is the time of the 200 MW_t steady state at 1:19:00. In the figures that follow, the points marked with a "●" represent the Chernobyl-4 plant data provided from fig. 4 of ref. [1], and continuous lines represent RELAP5-calculated data. A discussion comparing the code-calculated and plant data is given in section 3.2.

Table 3
RELAP5 – Calculated sequence of events for the Chernobyl accident simulation

Local time (HH:MM:SS)	Calculation time (s)	Event
1:19:00	0	Steady 200 MW_t plant conditions.
1:19:30	30	Feedwater flow stabilized at about 4 times the initial rate.
1:21:00	120	Reactor kinetics implemented in the model following withdrawal of manual regulator control rods.
1:22:00	180	Desired separator level attained, decrease in feedwater flow begins.
1:22:50	230	Feedwater flow stabilized at about 0.8 times initial flow rate.
1:23:04	244	Plant test begins. Recirculation pump flow coastdown initiated.
1:23:36	276	Effects of pump cavitation started.
1:23:38	278	Effects of pump cavitation fully implemented.
1:23:43.17	283.17	Calculation terminated. Melting of fuel in progress.

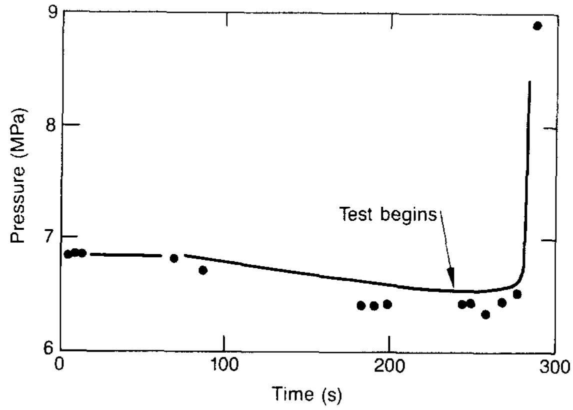

Fig. 4. Steam separator pressure during the Chernobyl accident.

The accident sequence begins with an increase in the feedwater flow rate as shown in fig. 3. The RELAP5-calculated sequence of events is shown in table 3. The initial 120 s of the transient calculation was performed without reactor kinetics feedback implemented; the core power was held constant at 200 MW_t. During this period, the operators withdrew manual regulator rods as needed to maintain core power. Following 120 s, the core power was calculated by the combined point reactor kinetics feedback effects of core average void fraction, fuel average temperature (Doppler), graphite average temperature, and automatic regulator control rod position.

The reactivity of the 12 automatic regulator rods (rod groups AC1, AC2, and AC3, consisting of 4 automatic regulator rods each) was simulated. The average position of these rods is known at 1:19:00, and their

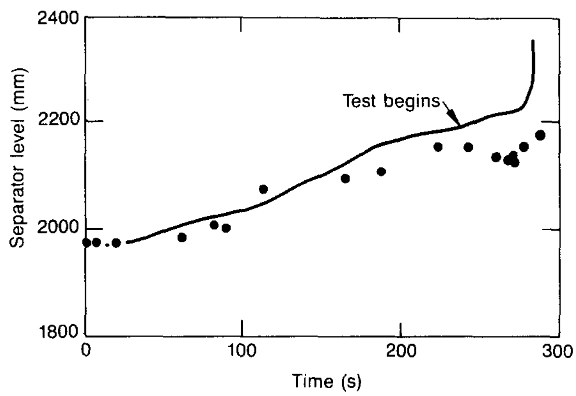

Fig. 5. Steam separator level during the Chernobyl accident.

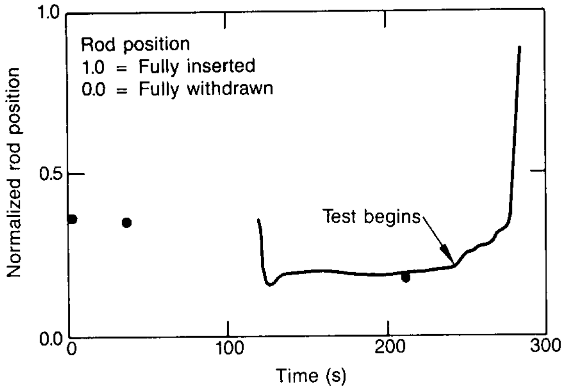

Fig. 6. Automatic regulator rod position during the Chernobyl accident.

position was assumed to not change between 1:19:00 and 1:21:00. In the model, rod position was changed, at up to the maximum rod travel rate, to attempt to control the power at 200 MW_t throughout the remainder of the transient.

At 30 s, the feedwater flow was stabilized at a flow rate approximately 4 times its initial rate. With the control rod position adjusting to maintain a constant core power, the higher feedwater flow caused the steam flow rate to fall, and this caused the coolant system pressure to decrease (fig. 4). With the feedwater flow rate exceeding the steam flow rate, the steam separator level began to increase as shown in fig. 5.

The calculated normalized automatic regulator control rod position is shown in fig. 6. Immediately following 120 s, when reactor kinetics control of the core power was implemented, the control rods automatically

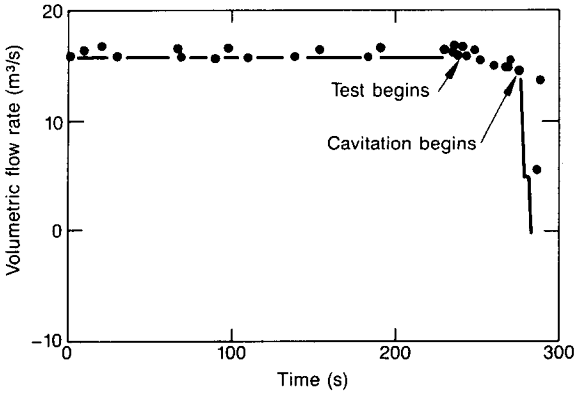

Fig. 7. Recirculation pump flow during the Chernobyl accident.

withdrew about 16%. This movement resulted from implementing reactor kinetics as the transient proceeded; rod withdrawal was required immediately to compensate for the declining core average void fraction. To maintain a constant core power, the control rods were inserted or withdrawn as needed to balance the reactivity generated by changing core void fractions (the coolant void reactivity coefficient is positive). Fig. 7 shows the recirculation pump volumetric flow rate. The recirculation pump speed was held constant, providing a constant pumped flow rate to the core up to the start of the test at 1:23:04 (244 s).

At 180 s, the desired steam separator level had been attained, and the operators throttled feedwater. This arrested the decline in the measured coolant system pressure and prevented further increase in the measured steam separator level. The decreasing feedwater flow also caused an increase in core average void fraction resulting in a positive reactivity insertion. To compensate for the increased power, insertion of the automatic regulator control rods began (fig. 6).

At 1:23:44 (244 s), the plant test began. The control rods were about 20% inserted at this time, as compared to 36% at 0 s; the difference caused by the falling core average void fraction over this period. When the test began, coastdown of half of the recirculation pumps started. Plant data in fig. 4, ref. [1], indicates the pumped flow rate declined in an approximately linear manner over the initial 40-s period of the test. The power excursion was experienced when only about 22% of the flow loss expected during the test had occurred. The coastdown of the affected recirculation pumps over the first 40 s of the test period caused a reduction in core flow and an increase in core average void fraction that could be compensated by inserting only a small fraction of the available automatic regulator control rod reactivity.

Many preliminary RELAP5 calculations were performed in an attempt to obtain a core power excursion during the first minute the test. The findings of all these calculations were similar: control rod reactivity is more than sufficient to compensate for all of the core void increase resulting from a coastdown of the core flow from 100% to 50%. Following the completion of the flow coastdown, the control rod was 90% inserted.

This finding was significant because it is inconsistent with data in ref. [1] indicating the power excursion and explosion occurred within the first minute of the test. A power excursion in the RELAP5 calculation during this time frame required a higher rate of core void formation than could be accomplished through pump coastdown alone.

The most likely manner to accelerate void formation is to accelerate the decrease in core inlet flow rate. The pump flow plant data presented in ref. [1] is credible from the start of the test up to 276 s (1:23:36). During this period, data points are frequent and show a reasonable scatter. However, after 276 s, only two pump flow data points are available. Furthermore, one of these points, at 287 s (see fig. 7), indicates a pump flow rate of only about 30% of the initial flow. This is significantly lower than expected given the trend of flow coastdown between 244 and 276 s. This data point is questionable as to whether it precedes or supersedes the power excursion.

A possible mechanism for accelerating the increase in core void fraction is recirculation pump cavitation. The RELAP5 calculation assumed the plant data point for pump flow at 287 s was an indication of pump cavitation, and the model was modified to simulate it. The modification consisted of reducing the pump speed from that needed to provide the prescribed flow coastdown to a constant speed providing 30% of the initial flow. The modification was implemented between 276 and 278 s. A further discussion of pump cavitation is provided in section 3.3; other possible trigger mechanisms for a power excursion are discussed in section 3.4.

The recirculation pumps were assumed to cavitate for two reasons. The first is that the Soviet report [1] indicated the pumps were operated in violation of plant requirements to avoid cavitation. Second, the RELAP5-calculated results indicated low subcooling at the inlet to the recirculation pumps. The minimum net positive suction head (NPSH) of the recirculation pumps is 0.6 MPa [11]. Minimum NPSH is a pump design parameter indicating the pressure above saturation required at the pump inlet to avoid cavitation. This criterion includes engineering safety factors so degradation in pump performance typically will not occur unless a significant violation of the criterion occurs. A minimum NPSH of 0.6 MPa is the equivalent of 5.6 K of subcooling at the pump inlet. During full-power operation, the minimum NPSH requirement is well exceeded; the RELAP5 full-power steady state calculation described in section 2 indicates 16.4 K subcooling at the recirculation pump inlet.

At the beginning of the test, the RELAP5-calculated pump inlet subcooling was only 2.4 K because of the high recirculation flow rate and low feedwater flow rate. At 276 s, the subcooling was also 2.4 K and falling. The assumptions that pump cavitation began at this time and resulted in a degradation in pump flow to 30% are arbitrary. The assumptions, however are not contradicted by the plant data and, based on NPSH considerations, may be appropriate. As will be shown next, these pump cavitation assumptions result in a RELAP5-calculated power excursion at approximately the reported time.

With the degradation in pumped core flow at 276 s, the rate of increase in the core average void fraction accelerated. Reactivity insertion from the increasing core void was only partially offset by control rod movement during the period from 276 to 280 s. Fig. 8 shows the components of RELAP5-calculated reactivity. As a result of the net positive reactivity insertion, the core power increased from 200 MW_t at 276 s to 576 MW_t at 280 s. The increase in core power caused a runaway power excursion. Increasing power resulted in an increasing core void fraction which, with a positive void reactivity coefficient, resulted in still higher power.

The core power increased to extremely high levels, peaking at 5.5×10^{11} W (550 000 MW) at 282.94 s as shown in fig. 9. The increased power caused a pressurization of the reactor coolant system as shown in fig. 4. As the pressure rose, the rate of increase in the core average void fraction was slowed, allowing the insertion of negative reactivity from increasing fuel temperatures to momentarily exceed the positive reactivity insertion due to increasing core average void fraction. This caused the power to decrease after 282.94 s. The core power then declined to 1.48×10^{11} W (148 000 MW) at 283.02 s. By this time, the steam safety relief valves had opened. This slowed the pressurization rate, accelerated formation of core void, and caused the core power to increase again. A second power peak of 4.35×10^{11} W (435 000 MW), was calculated at 283.16 s. The power declined thereafter because the fuel temperature negative reactivity insertion again exceeded the core void positive reactivity insertion.

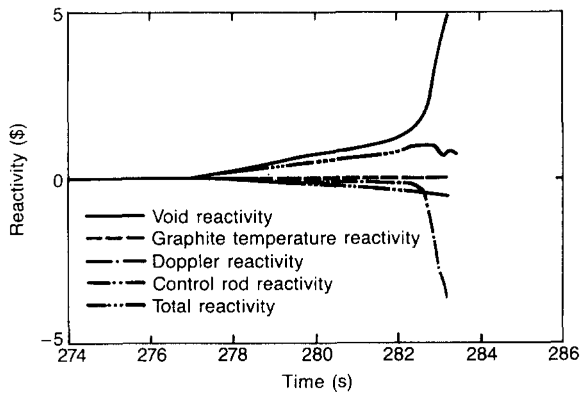

Fig. 8. Reactivity components during the power excursion.

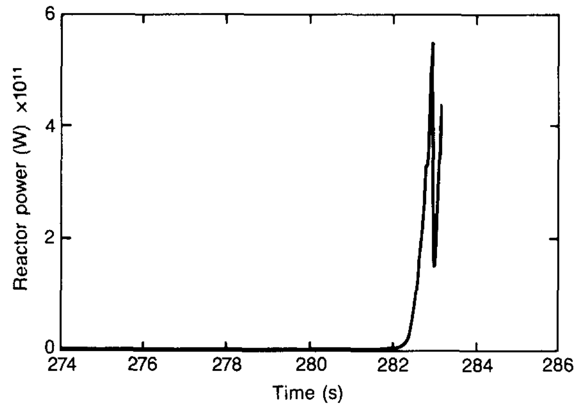

Fig. 9. Reactor power during the power excursion.

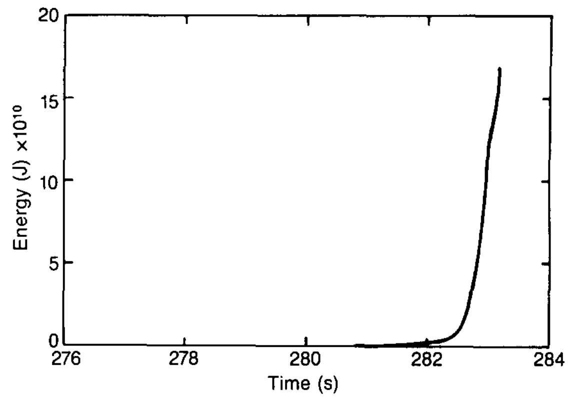

Fig. 11. Energy release during the power excursion.

The RELAP5 calculation was terminated at 283.17 s when one of the fuel temperatures reached 3116 K. This temperature is above the melting point of UO_2. At the end of the calculation, the core power was 3.91×10^{11} W (391 000 MW), the steam separator pressure was 8.42 MPa, the core average void fraction was 83.8%, and the core average fuel temperature, shown in fig. 10, was 2077 K. The average graphite temperature at this time was 582.6 K, and the automatic regulator rods were 88.2% inserted. During the final seconds of the calculation, the control rods were calculated to insert at their maximum rate of travel. Pumped core flow, shown in fig. 7, fell significantly below 30% of initial flow when the increasing core power produced a localized pressurization in the core.

Ref. [13] provides results from tests performed in the Power Burst Facility at the Idaho National Engineering

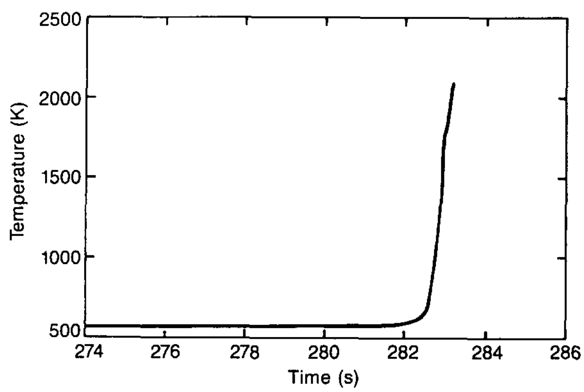

Fig. 10. Core average fuel temperature during the power excursion.

Laboratory simulating power excursions expected during reactivity initiated accidents. The report indicates that fuel rod failure is expected following an increase in the fuel rod enthalpy of about 240 cal/g. For Chernobyl-4, this corresponds to an increase in fuel energy of 1.90×10^{11} J. Fig. 11 shows the RELAP5-calculated net fuel energy deposition after 276 s. At the end of the RELAP5 calculation, the fuel rod energy input, net of heat loss to the coolant, was 1.69×10^{11} J and rising rapidly. The RELAP5 calculation indicates the reported explosion in the Chernobyl plant was likely initiated by a reactivity-induced core power excursion of sufficient magnitude to cause rapid disintegration of the fuel rods. Continuing the RELAP5 calculation further would be inappropriate because the fuel rod geometry would be expected to change and interactions between coolant, cladding, and fuel would be significant. Modeling of these severe accident phenomena is beyond the capabilities of the RELAP5 computer code. Because of the rapid fuel rod disintegration process, the simulation was not continued also with the SCDAP computer code.

The RELAP5-calculated core average void fraction response during the last few seconds of the calculation is shown in fig. 12. At the end of the calculation, total core dryout is imminent, and the core inlet flow, characterized by the recirculation pump flow in fig. 7, was stagnant. Under these conditions, a steam explosion resulting from interaction of molten fuel and core coolant would be unlikely because of the limited volume of liquid remaining in the core. When the core becomes completely dry, however, steam production could cease, allowing reintroduction of liquid into the core by the recirculation pumps. Another explanation for reintroduction could be cessation of pump cavitation caused by the general repressurization of the primary coolant

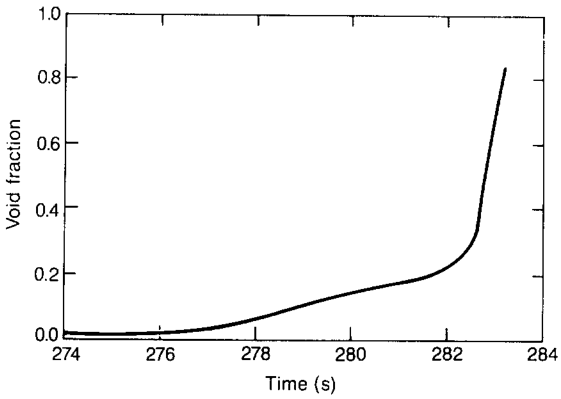

Fig. 12. Core average void fraction during the power excursion.

system at the time of the power excursion. Reintroduction of large quantities of liquid into the core at this time, when the pressure tubes contain molten fuel, could produce a steam explosion that could rupture the pressure tubes.

Pressure tube failure has not been determined to have been caused by a steam explosion. Other failure possibilities include mechanical failure as a result of fuel rod disintegration and high temperature failure as a result of contact with molten fuel.

3.2. Comparison of calculated and Chernobyl-4 plant data during the accident

Limited plant data are available from ref. [1] for the period from 1:19:00 to the time of the explosion. These data are shown in figs. 3 through 7, compared with the RELAP5 calculation of the accident sequence.

The plant data for feedwater flow, shown in fig. 3, were used to prescribe the feedwater flow boundary condition in the calculation. The RELAP5-calculated steam separator pressure is compared with plant data in fig. 4. The pressure decline in the plant data, caused by the increasing feedwater flow rate, was not duplicated entirely in the calculation. The pressure decline is a function of the steam flow rate and flow losses between the separator and constant pressure boundary condition representing the turbine. The plant turbine inlet pressure is suspected to be not well represented by the constant pressure condition assumed in the calculation.

Plant data for steam flow are not available; however, the text of ref. [1] indicates that before the beginning of the test Turbogenerator 7 was isolated. The test began by isolating Turbogenerator 8. Therefore, as the test proceeded, no outlet for steam was available and an immediate increase in coolant system pressure would have resulted. The plant pressure data, however, do not show this increase; an anomaly that has not been resolved. The comparison in fig. 4 of plant and code-calculated pressure responses during the test is marginal; improvement would require new insights into the steam conditions during the plant test.

A comparison of RELAP5-calculated separator level with plant data is shown in fig. 5. The reference point for the plant data level indication is not known; the calculated separator level is set to agree with the plant data at 0 s. The comparison is good during the period from 0 to 220 s. During the test, however, plant data indicate the level fell while the calculated level continued increasing. No explanation for this difference has been found.

Fig. 6 compares the RELAP5-calculated normalized control rod position with plant data. The plant data points shown are the average positions of automatic regulator control rod groups AC-1, AC-2, and AC-3. While the frequency of plant data points is limited, the comparison with RELAP5-calculated data is excellent. Specifically, the comparison at 211 s indicated the RELAP5 control rod modeling is adequate. By this time the manual regulator rods have been withdrawn and the automatic regulator rods have compensated for remaining effects of the feedwater flow transient. The comparison indicates this compensation was modeled correctly.

Fig. 7 compares the RELAP5-calculated recirculation pump volumetric flow rate with plant data. Note the next to last plant data point (287 s) shows the flow was significantly degraded below that expected for the lower pump speeds. As was discussed previously, this data point was used as a partial justification for assuming a significant flow reduction had been caused by pump cavitation.

In summary, comparisons of plant- and code-calculated data for steam separator level, coolant system pressure, and automatic regulator control rod position are generally in good agreement, indicating the RELAP5 model adequately represents the plant. The plant data for feedwater and recirculation pump flow rate were used as boundary conditions for the RELAP5 simulation of the accident.

3.3. Discussion of pump cavitation effects

Pump cavitation refers to a condition where voids are formed within a pumped fluid. In a centrifugal pump, such as is used in the RBMK-1000 reactor and most other nuclear plant designs, flow enters the pump axially, is accelerated radially through the action of an impeller, and exits the pump circumferentially. Cavitation occurs if the local pressure within the pump falls to the saturation pressure of the fluid being pumped. Cavitation is a condition to be avoided for two reasons. First, a degradation in the pump performance is experienced. Second, cavitation has the potential to destroy a pump. As a fluid containing voids leaves the impeller and is decelerated the voids collapse. A rapid local pressurization occurs when the fluid changes from a compressible to incompressible form, and this has been shown to cause severe pitting damage to the impeller. This local pressurization may also explain the pump performance degradation, since it disrupts the normal pressure profile through the pump.

Ref. [14], a text on cavitation, indicates a dramatic fall off in pump head and efficiency occurs when a pump cavitates. Contrasting with this indication, however, are the findings of studies involving Loss of Fluid Test (LOFT) L3-6 [15] and the accident at the Three Mile Island (TMI), Unit 2, pressurized water reactor [16]. Evidence from these studies indicates coolant pump performance degrades uniformly as coolant void increases, not dramatically as described for the onset of cavitation. A confusing issue is whether pump cavitation results in a sudden dramatic decrease in pumped flow, or if the decrease could be expected to be gradual. In the RELAP5 calculation of the accident sequence, the decrease was assumed to be sudden. The discrepancy in reported findings may be caused by a difference in pump inlet conditions. If pump inlet subcooling is decreased until cavitation occurs, the onset of pump performance degradation may be sudden. If, however, void is introduced at the pump inlet (such as from an upstream source), then pump performance degradation may be gradual. The difference between these two cases is that in the first case flow is incompressible except at the site of cavitation, in the second case flow is compressible throughout the pump.

3.4. Other possible triggers for the power excursion

As was discussed in section 3.2, analysis has shown sufficient automatic regulator control rod insertion capability is available to prevent a power excursion when the recirculation pumps coasted down during the test. Some other factor or "trigger" was needed to have caused the additional increase in core average void fraction required to initiate a runaway power excursion. In the RELAP5 calculation of the Chernobyl accident sequence the trigger was assumed to be a partial loss of recirculation pump flow due to effects of pump cavitation. However, while pump cavitation was suspected to have been the trigger, it was not shown conclusively to be the case.

Another possible, but unlikely, trigger is partial or complete loss of core inlet flow such as could be caused by inadvertent closure of valves between the recirculation pumps and the core.

The text of the Soviet report [1] indicates that during the test the reactor power slowly drifted upward. Drift may be an indication of a spurious positive reactivity insertion, possibly from xenon burnoff. As the power increases the xenon content of the core could decrease. Possibly the negative reactivity so removed could be sufficient to cause the power excursion.

Another possible trigger for the power excursion is the rapid opening of a valve in the steam line. As was discussed in section 3.2, an anomaly exists concerning steam flow during the test. Plant data indicate the coolant system pressure did not increase when Turbogenerator 8 was isolated. This implies an alternate steam sink was available, such as a bypass to condenser. If, as the test began, the operators attempted to open a turbine bypass valve to compensate for isolating Turbogenerator 8, the potential exists for a momentary loss of pressure control. The dip in pressure could result in the rapid formation of core void needed to produce a power excursion.

Next, the possibility exists for significant multidimensional effects beyond the calculational capabilities of a one-dimensional thermal-hydraulic computer code. Multidimensional effects that might be important include effects caused by the radial power shape, and time dependent axial power shapes. The RELAP5 model simulates the phenomena in an average core channel and is not capable of calculating the effects of a variable axial power shape.

Finally, the geometry of the control rods may have provided a trigger for the power excursion at the time the AZ-5 scram button was depressed. Graphite followers are located on the control rods so when a rod is withdrawn the space filled with boron carbide rod is replaced with the graphite rod follower. When the rod is fully withdrawn, only the middle 5 m of the control rod tube contains graphite; the upper and lower 1 m regions contain cold cooling water. When a scram occurs, insertion of all control rods begins, causing the lower 1 m of core to experience a positive reactivity insertion from the difference in moderation between water and graphite. Note the uppermost 1 m of core would simultaneously experience a negative reactivity insertion as cold water in the control rod tubes is replaced with boron carbide in that region. On a core-wide basis the net reactivity insertion would likely be negative. However, a potential exists for a power excursion beginning in the lower 1 m of the core. Furthermore, a power excursion producing void in the bottom of the core would likely propagate up the core because voids produced in the lower core would be swept upwards in the pressure tube coolant flow.

In summary, some trigger, beyond the coastdown of the recirculation pumps, was necessary to obtain a power excursion during the simulation of the plant test. The trigger assumed in the RELAP5 calculation of the accident was the onset of recirculation pump cavitation. This assumption is not, however, conclusively justified, and alternate trigger mechanisms have been suggested. Further details of this work appear in ref. [17].

4. Station blackout simulation

Immediately following the accident at Chernobyl-4, little was known about the sequence of events. Initial speculation on the transient concentrated on the station blackout event with loss of all feedwater. The station blackout transient, which begins with a loss of offsite power, is a risk dominant sequence leading to a core melt in light water reactors. Following loss of offsite power, the diesel generators and emergency feedwater system are assumed not to start. This transient sequence results in a boiloff of coolant inventory followed by core heatup and damage. While more recent information has indicated a station blackout was not involved in the accident (substantiated by these calculations), the plant response to this event is useful in understanding further the response of this plant design during severe accidents.

A RELAPS calculation was performed over the portion of the transient leading up to cladding–coolant reaction (and beyond to provide boundary conditions for a SCDAP calculation). A SCDAP calculation of the reactor vessel behavior beginning 3000 s into the transient was performed using coolant conditions calculated by RELAP5. SCDAP was chosen to provide information on fuel cladding oxidation, hydrogen generation, and the effects of radiation heat transfer between the fuel rods and pressure tubes. The calculated sequence of events for this transient is shown in table 4 and results of the calculations are shown in figs. 13 through 18.

Table 4
Calculated sequence of events for Chernobyl-4 station blackout with failure of emergency feedwater

Events	Time (s)
Loss of offsite power, diesel generators and emergency feedwater fail to start, reactor trip, turbine trip, recirculation pump trip	0
Turbine valve close	2
Steam relief valves open	2.5
Main feedwater flow stopped	10
Forced loop circulation ends, natural loop circulation begins, recirculation pump rotor pinwheels	195
Recirculation pump rotor stopped	389
Steam drum separators empty	2171
Loop natural circulation lost	2286
Fuel rod heatup begins at top of core	2520
Fuel rod temperature reaches 1000 K in next-to-top core cell	3000
SCDAP calculation initiated	3000
Fuel rod heatup begins at bottom of core	3256
Downcomers empty	3289
Core completely voided	3376
Peak fuel rod cladding temperature reached (1564 K) in middle of core - calculated by RELAP5	5004
All metal temperatures above saturation	5005
Calculations terminated	12000

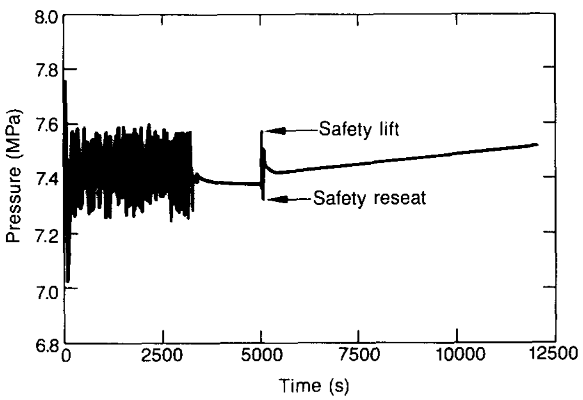

Fig. 13. RELAP5 station blackout steam separator pressure.

The steam drum separator pressure response calculated by RELAP5 is shown in fig. 13. As a result of the loss of offsite power, reactor and turbine trips occurred at time zero. The steam separator pressure increased to

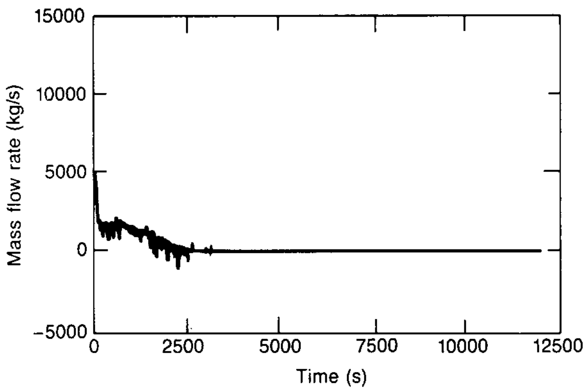

Fig. 14. RELAP5 station blackout core inlet flow.

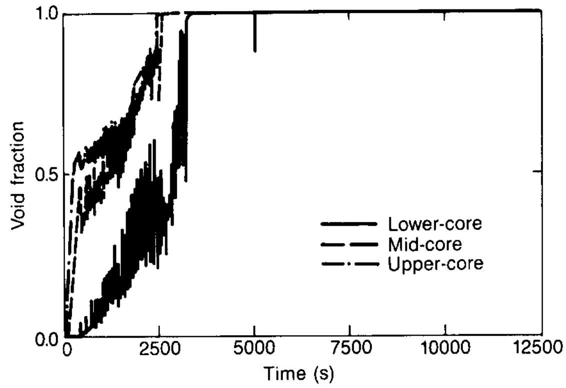

Fig. 16. RELAP5 station blackout core void fractions.

a peak of 7.81 MPa, resulting in lifting five of the six safety relief valve banks. The rising pressure caused the safety valves to lift and this caused a decrease in pressure that caused the valves to reseat. This valve cycling phenomenon explains the oscillatory pressure behavior up to about 3000 s.

The core inlet flow rate response is shown in fig. 14. The pumps were tripped at zero time as a result of the loss of offsite power. By 195 s, the pump coastdown had proceeded to the point where loop natural circulation flow was established.

The void fractions in loop components are shown in fig. 15. The steam drum separators were empty at 2171 s. As shown in fig. 14, loop natural circulation decayed gradually and was lost shortly after the steam drums emptied. Following the loss of natural circulation, the downcomers between the separators and the pump inlet headers drained; however, the core inlet sections of the coolant loop remained liquid-filled. Core boiloff progressed in a top-down manner as shown in fig. 16. At 3000 s, the next-to-stop fuel rod heat structure reached a cladding temperature of 1000 K, the onset temperature for metal-water reaction as calculated by the SCDAP code. The SCDAP calculation was started at that time.

Mid-core fuel rod cladding surface and outer graphite surface temperature responses are shown in figs. 17 and 18, respectively, for both the SCDAP and RELAP5 calculations. RELAP5-calculated graphite temperatures initially decreased because the graphite power was lost following the reactor trip. During steady state operation, the graphite is a heat source to the core coolant, and it continued to transfer heat to the fluid within the pressure tubes following reactor trip. After 3000 s, however, the fuel rods had heated up and were hotter than the graphite in both calculations. The graphite then became a heat sink and adsorbed heat from the pressure tubes. The lower cladding temperatures and higher graphite temperatures calculated by SCDAP

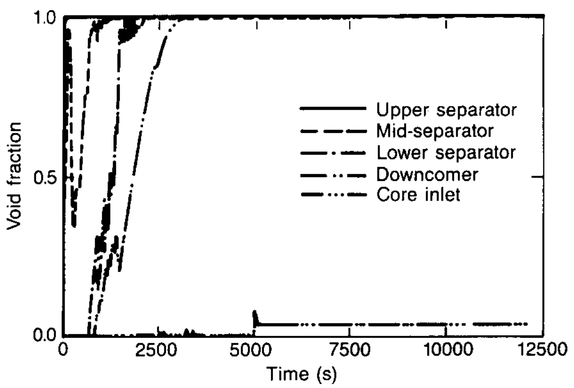

Fig. 15. RELAP5 station blackout coolant loop void fractions.

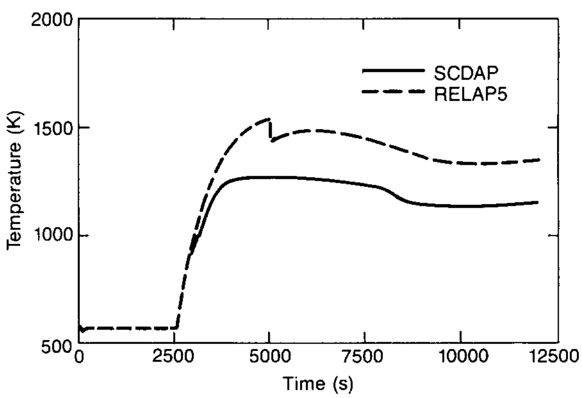

Fig. 17. Comparison of RELAP5 and SCDAP fuel rod cladding temperatures.

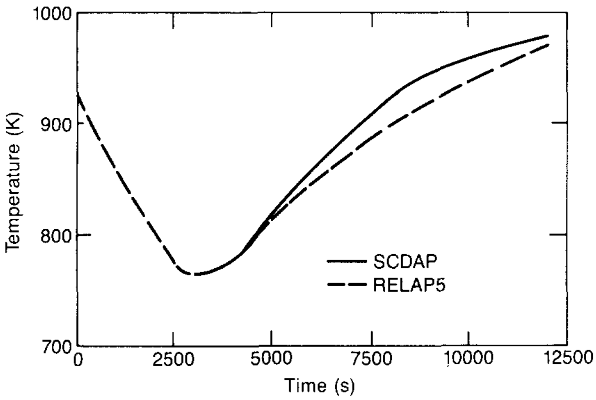

Fig. 18. Comparison of RELAP5 and SCDAP graphite temperatures.

indicate the importance of radiation heat transfer, which is not calculated by RELAP5.

The declining fuel rod temperatures from about 6000 to 9000 s indicate the core decay heat was decreasing slowly with time. At about 8000 s, the SCDAP-calculated cladding surface temperature began to decrease more rapidly. This occurred when the oxidation process slowed because an oxide layer had been built up on the outside of the fuel rod which restricted diffusion of oxygen to the cladding. A similar change in slope was not calculated with RELAP5 because the code does not calculate fuel rod cladding oxidation.

At about 9000 s, a condition was reached where fuel rod heatup resumed in order for the decay heat to pass from the fuel rods to the pressure tubes. After 12000 s (the end of the calculations), the slow heatup of fuel rod and graphite is expected to continue. A peak fuel rod cladding temperature of 1564 K was calculated by RELAP5 in the middle of the core at 5004 s.

Oxidation and the accompanying hydrogen generation were calculated by the SCDAP code beginning at 3000 s when zirconium temperatures reached 1000 K near the top of the core. About 0.07 kg of hydrogen was generated in a single fuel assembly (116 kg total for the plant) during the transient. This corresponds to oxidation of only about 5% of the fuel rod cladding and 0.3% of the zirconium pressure tube. The oxidation process did not use all of the available steam. The fuel and support rods actually became cooler after 5000 s because much of the oxidation and decay heat was being transferred to the graphite moderator.

The heatup rate during the last 3000 s of the transient was about 0.02 K/s. At that rate, the lowest temperature at which accelerated oxidation occurs (1853 K) would be reached 12 h into the accident. At this heatup rate, the oxide shell would probably be thick by that time. Accelerated oxidation would, therefore, have little effect because oxidation is inversely proportional to the amount of oxide already present. Also, little water would probably remain to provide steam to the reaction. Assuming the slow heatup rate and the small effect of accelerated oxidation, fuel melting would be expected to occur about 24 h into the accident.

In summary, the Chernobyl-4 station blackout with failure of all feedwater has been shown to lead to a fuel rod heatup beginning at about 42 minutes. The sequence of coolant boiloff and inventory depletion is comparable to that expected in a U.S. boiling water reactor. Metal–water reaction is calculated to begin at about 50 minutes, and by about 56 minutes a core-wide heatup is underway.

The graphite moderator acted as a heat source to the pressure tubes prior to core heatup. However, following the start of core heatup, the graphite acted as a heat sink that significantly limited the fuel rod temperature excursion. Continued decay heat generation is predicted to result in melting of fuel at about one day into the sequence. An important finding of this analysis is that fuel melting in the RBMK-1000 reactor design would occur much later than for other reactor designs for which fuel melting is expected in less than 4 h.

Because of a high probability of an operator restoring normal or emergency power sources over the period of a day, a station blackout with failure of all feedwater was concluded to likely not be the cause of the Chernobyl-4 accident. In light of more recent information concerning the accident, this conclusion appears obvious. However, this conclusion was meaningful at the time the analysis was performed in June 1986. The station blackout analysis is documented here because it provides a perspective of Chernobyl-4 severe accident response beyond that encountered in the accident. Further details of this work appear in ref. [17].

5. Summary and conclusions

The RELAP5 computer code simulated the April 26, 1986 accident at the Chernobyl, Unit 4 reactor. Code-calculated and plant data are in good agreement from 1:19:00 to 1:23:04 (the time the plant test began) indicating satisfactory modelling. The effects of the recirculation pump coastdown during the test are shown to be insufficient to cause the power excursion. Automatic control rod capability was more than adequate to compensate for the increase in core average void fraction as the core flow decreased. A mechanism or "trigger" was required to cause a reactivity-driven power excursion capable of destroying the plant.

Possible triggers include: (1) disruption of core flow caused by inlet valve closures, (2) a positive reactivity insertion caused by an accelerating burn off of xenon, (3) a rapid opening of a steam relief path, such as turbine bypass, (4) multidimensional radial, or time variant axial power effects, (5) a local power excursion in the lower core region, caused by a control rod moderation-induced reactivity insertion at the time of reactor trip, and (6) cavitation of the recirculation pumps. The RELAP5 simulation of the accident sequence selected recirculation pump performance degradation caused by the onset of cavitation within the pumps as the trigger. This assumption was based on the finding that the net positive suction head requirement for the pumps was violated throughout the plant test period.

With the pump cavitation trigger, a rapid power excursion was calculated. Peak RELAP5-calculated core power during the accident was 550 000 MW_t. The calculation was terminated because melting of fuel indicated a loss of fixed fuel rod geometry had occurred. At this time, core energy had increased by 1.69×10^{11} J and the core power remained elevated at 391 000 MW_t. Fuel disintegration, estimated to occur when the fuel energy increase reaches 1.9×10^{11} J (corresponding to 240 cal/g), is imminent at the end of the calculation.

From the conditions present at the end of the RELAP5 accident simulation, the Chernobyl power excursion likely led to failure of the pressure tubes. At the end of the calculation, a complete dryout of the core was approaching and the resulting decline in steam production would allow the reintroduction of a large quantity of water into the core from the action of the recirculation pumps. This reintroduction, occurring at a time when the pressure tubes contained molten fuel, would provide the potential for pressure tube failure caused by overpressurization.

A preliminary analysis of another type of severe accident sequence in the Chernobyl-4 plant (station blackout event with failure of all feedwater) indicated that sequence timing was long. Fuel melting would not occur until about one day into the accident. This analysis, performed shortly after the accident, indicated that a station blackout event was an unlikely cause of the accident. A significant finding from this analysis was fuel rod melting was delayed from less than four hours to about one day into the accident because the graphite stack acted as a heat sink for the core heat. This indicates that during this sequence more time is available for operator corrective action in plants of RBMK-1000 design than in plants of other designs.

In summary, two simulations of severe accident conditions in the Chernobyl-4 plant were performed. In a postulated station blackout sequence with loss of feedwater, core melt was found to be significantly delayed because of heat transfer to the graphite stack. Analysis of the April 26, 1986 accident simulation indicated that a mechanism, or trigger, beyond pump coastdown was required to cause a rapid power excursion. After a trigger was assumed, a power excursion with imminent fuel failure was calculated.

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