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SYSTEM TRANSIENT RESPONSE TO LOSS OF OFF-SITE POWER[1]

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SYSTEM TRANSIENT RESPONSE TO LOSS OF OFF-SITE POWER

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A simultaneous trip of the reactor, main circulation pumps, secondary coolant pumps, and pressurizer pump due to loss of off-site power at the High Flux Isotope Reactor (HFIR) located at the Oak Ridge National Laboratory (ORNL) has been analyzed to estimate available safety margin. A computer model based on the Modular Modeling System code has been used to calculate the transient response of the system. The reactor depressurizes from 482.7 psia down to about 23 psia in about 50 seconds and remains stable thereafter. Available safety margin has been estimated in terms of the incipient boiling heat flux ratio. It is a conservative estimate due to assumed less than available primary and secondary flows and higher than normal depressurization rate. The ratio indicates no incipient boiling conditions at the hot spot. No potential damage to the fuel is likely to occur during this transient.

1. INTRODUCTION

The HFIR is a research reactor designed to generate high thermal neutron fluxes (4×10^{15} neutrons cm^{-2} s^{-1}) originally at 100 MW and currently at 85 MW. The reactor is a pressurized light water moderated and cooled reactor and does not generate electricity. The reactor vessel is under water and pressurized by a pressurizer pump unlike commercial pressurized water reactors (PWRs) that have pressurizers. The reactor core is composed of 540 fuel plates distributed among two concentric annular regions. It has a height of 24 inches and a diameter of 17.1 inches. The primary purpose of HFIR is to produce isotopes for use in the heavy element research programs and also to provide experimental facilities on site.

The construction of HFIR was completed in 1965. It had been operated until it was shut down in November 1986 due to concerns mainly over increased pressure vessel embrittlement. The operational conditions were revised (85 MW and 482.7 psia), and HFIR started operating in April 1989. The HFIRSYS computer model has been developed to analyze operational and small break loss-of-coolant type system transients. The model is based on the Modular Modeling System (MMS) computer code whose development has been sponsored originally by EPRI and currently by the Babcock and Wilcox (B&W) Corporation through international and domestic MMS users. [1] The HFIRSYS computer code has been verified and validated against the available experimental data.

Loss of off-site power resulting in automatic trip of all of the pumps and reactor has been studied to estimate whether or not incipient boiling conditions are likely to occur at a hot spot located at the exit of the core. A leak with a 0.5-in. diameter represents leakage from the primary coolant system during the transient. (The system normally depressurizes much slower than the rate caused by a 0.5-

in.-diameter leak.) Conservative results are further encouraged by less than available primary and secondary flow rates. Boiling conditions, although could occur at low temperatures, is considered undesirable conservatively. The ratio of the heat flux required to cause boiling to heat flux at the hot spot is used as an indicator of estimated safety margin. The results of this study as well as a brief description of HFIR is presented.

2. HIGH FLUX ISOTOPE REACTOR

The HFIR is a pressurized light water moderated and cooled research reactor. A schematic of the primary coolant system is provided in Fig 1. The primary coolant system has the reactor vessel, four heat exchanger loops with heat exchangers (coolers), and main circulation pumps. The letdown lines and pressurizer pump discharge line connect the primary coolant system to the low pressure system which contains the deaerator, pumps, prefilters, demineralizers, after filters, primary coolant head tank, and interconnecting piping. Water from the heat exchangers travels in a 20-in. cold leg and passes through a strainer before it enters the top of the reactor vessel through two diametrically opposed 16-in. lines. The hot leg is a single 18-in. line. The primary heat exchangers are parallel-counter-flow (shell and U-tube) type and are mounted vertically. Only three heat exchangers are required for full-power operation. The four main circulation pumps are vertical shaft centrifugal pumps; each pump takes its suction from an individual heat exchanger. The discharge pipe is a 10-in. line and connected to 20-in. cold leg. Each pump delivers about 5500 gpm during normal operation. They are driven by 600-hp ac motors and 3-hp dc pony motors under normal operational conditions. During shutdown cooling only the dc pony motors are needed. If the reactor vessel pressure is below 249.7 psia, the main circulation motors are tripped; however, the dc pony motors continue to operate. The pump speed is about 1800 rpm during normal operation.

Figure 2 provides a detailed view of the vertical cross sections of the reactor vessel and core. Only one of the two vessel inlets are shown. There is only one vessel outlet. The flow paths are not shown in the figure. The coolant enters the core from above and flows downward into the vessel outlet. The flow is split into mainly target, control plate, beryllium reflector, and core regions. The normal operational vessel inlet flow is about 16700 gpm. At 85.3 MW the vessel inlet and outlet temperatures are 120°F and 156°F. The reactor vessel is submerged in a pool. It has an internal diameter of 94 inches and a height of 19 feet including the narrow outlet portion of the vessel. There is a two-in.-check valve between the vessel and the pool, and the valve is connected to the vessel through a two-in. line. The check valve opens allowing water from the pool into the inlet plenum if the reactor pressure is below about 24 psia. There is a total of 540 fuel plates distributed between inner and outer annuli. The active fuel length is 20 inches. The channel width between two fuel plates and the fuel plate thickness is 0.050-in. each.

Heat is removed by secondary water and dissipated to the atmosphere by an induced-draft cooling tower. The secondary flow is controlled to keep the reactor inlet temperature at 120°F. There are three main secondary coolant pumps. Two of these operate during the normal operation and have a combined capacity of about 26000 gpm. An auxiliary pump has a capacity of 6000 gpm at its high speed setting.

The control valves in the letdown lines control letdown flow to keep reactor inlet plenum pressure at 482.7 psia. The letdown flow through the letdown valves goes into the letdown header also and through the cleanup system and returns to the primary head tank. The pressurizer pump forces the coolant from the primary head tank back into the primary coolant system at about 120 gpm. If the reactor pressure falls below 407.7 psia, the block valves are automatically closed preventing any further flow through the letdown lines.

3. HFIRSYS COMPUTER MODEL

The HFIRSYS computer code was developed based on the Modular Modeling System (MMS) code, whose development was originally sponsored by the Electric Power Research Organization (EPRI) and currently by the Babcock & Wilcox Corporation (B&W) through domestic and international user organizations of MMS. The MMS code provides plant component modules and tools for analysis of fossil and nuclear power plant dynamics. [1] It supplements the existing large thermal hydraulic codes like RETRAN, TRAC, and RELAP by providing capabilities for scoping safety analyses and operational transients. Additional models such as the reactor vessel and core have been developed and modifications to the MMS component modules have been made to represent HFIR. The MMS code is based on the Advanced Continuous Simulation Language (ACSL). The macro capability of ACSL provides modular capability. Each plant component (pumps, valves, pipes, etc.) is represented by a module (macro) and can be used as many times as necessary. Modules interact through input and output variables; this allows replacing modules by more or less complicated ones.

The MMS code utilizes the lumped parameter approach. Solid structures and liquid volumes are represented by nodes. The number of nodes in a component module depend on specific component models; however, nodes in various components are large in general. The independent variables for fluid nodes are selected as pressure and enthalpy. The other properties are described as their functions. A significant assumption made in application of pressure and enthalpy derivative equations to nodes is that the rate of change of the property leaving the node is approximately equal to the rate of change of the average property. The derivative equations are modified to account for the structural elasticity of the HFIR primary coolant system in HFIRSYS. A detailed description of the code and models used to represent components are provided in a report. [2]

The HFIRSYS transient model does not include all of the details of the primary coolant and low pressure systems. There are four heat exchanger (cooler) loops, three of which are necessary for normal operation. In HFIRSYS, two of the three heat exchanger loops are combined into a single loop, and the other loop includes one cooler and a main circulation pump. This allows studying transients including trip of either one, two, or three main circulation pumps. A schematic drawing of the HFIRSYS model with component modules is shown in Fig 3.

Verification and validation of HFIRSYS have been performed. The in-house developed coding has been reviewed to verify that the coding portrays the intended mathematical equations (model). Validation efforts, limited to areas where experimental data are available, have indicated satisfactory performance of HFIRSYS.

4. LOSS OF OFF-SITE POWER

Loss of off-site power resulting in automatic trip of all the pumps and reactor has been studied using HFIRSYS. The calculation was carried out for only two minutes. At the end of this time period the secondary flow, primary flow, and pressure was stable. Meanwhile the reactor decay heat continually decreased. The reactor power and pressure are 85.3 MW and 482.7 psia in the model. The reactor vessel inlet and outlet temperatures are about 120°F and 156°F and the primary coolant flow rate is about 16300 gpm in the model. The pressurizer pump and letdown flows are about 120 gpm.

The loss of off-site power transient was started by tripping the reactor, main circulation pumps, secondary coolant pumps, and pressurizer pump. A 0.5-in.-diameter leak is simultaneously formed at the discharge of the main circulation pumps to represent leakage from the primary coolant system. The depressurization rate caused by this leakage is much higher than normal depressurization rate. The system depressurizes in response to decreasing pressurizer pump flow and leakage and stabilizes in about 50 seconds. The block valves are automatically closed at about 407.7 psia preventing any further flow through letdown lines. Once the reactor pressure falls below about 24 psia, the check valve between the reactor pool and the vessel opens allowing water from the pool into the reactor. The check valve flow offsets leakage from the system and the pressure remains stable. The main circulation pumps coast down and the primary coolant flow decreases down to the flow generated by one main circulation pump driven by the small dc pony motor. Only one of the three dc pony motors operating at a low speed and running on batteries is assumed to be operational. This results in a primary coolant flow rate less than the normal shutdown flow rate. The auxiliary secondary coolant pump running on diesel generators provide shutdown flow. The secondary flow is assumed to decay to less than one third of the available flow generated by the auxiliary pump at its high speed setting.

As the vessel inlet plenum pressure stabilizes about 23 psia, due to low flow and static head, the pressure is higher in the vessel outlet plenum. The inlet and outlet pressures of the vessel are shown in Fig. 4. The pressure drop through the core is initially large. If the system depressurizes very quickly, boiling could occur at the exit of the core although the core exit temperature may be low. Incipient boiling at a hot spot located at the exit of the core is considered undesirable conservatively. The ratio of the heat flux required to cause incipient boiling to heat flux at the hot spot continually calculated and results are shown in Fig. 5. The incipient boiling heat flux ratio initially increases to about 16 because the heat generation in the core is reduced after the reactor trip. The peak value gradually decreases as the primary coolant flow decreases to a low and stable value. The minimum value of the incipient boiling heat flux ratio does not fall below its normal operational value during the transient. A straight line (ratio=1) indicating incipient boiling is included in the figure as a reference. Incipient boiling at the hot spot is estimated not to occur. After about 30 seconds, it continually rises because the flows are stable and reactor decay heat gradually decreases. The ratio of the critical heat flux to heat flux at the hot spot (departure from nucleate boiling) is also shown in the figure. The critical heat flux is calculated from an in-house developed correlation for HFIR. The reactor vessel inlet and outlet temperatures and saturation temperature calculated at the core exit pressure are shown in Fig. 6. The reactor vessel exit temperature decreases from its steady state value of about 156°F while the inlet temperature stays almost unchanged. The saturation temperature calculated at the core exit pressure and vessel exit temperature never approach each other. No potential damage to fuel is likely to occur during this transient.

5. SUMMARY

Loss of off-site power at HFIR located at ORNL has been analyzed to estimate safety margin using HFIRSYS based on the MMS code. The available safety margin has been estimated in terms of the incipient boiling heat flux ratio. Conservative values for the incipient boiling heat flux ratio have been encouraged by using less than available primary and secondary flows and high depressurization rate. The ratio indicates no incipient boiling conditions at the hot spot. The minimum value of the ratio is as high as its normal operational value indicating that there is at least as much safety margin as there is during normal operation at the hot spot. Thus no fuel damage is likely to occur during this transient.

REFERENCES

1. Modular Modeling System (MMS): A Code for the Dynamic Simulation of Fossil and Nuclear Power Plants, Volume 1: Theory Manual, Babcock & Wilcox, March 1985.

2. A. Sozer, A system Analysis Computer Model for the High Flux Isotope Reactor (HFIRSYS Version 1), ORNL/TM-11611 (in publication).

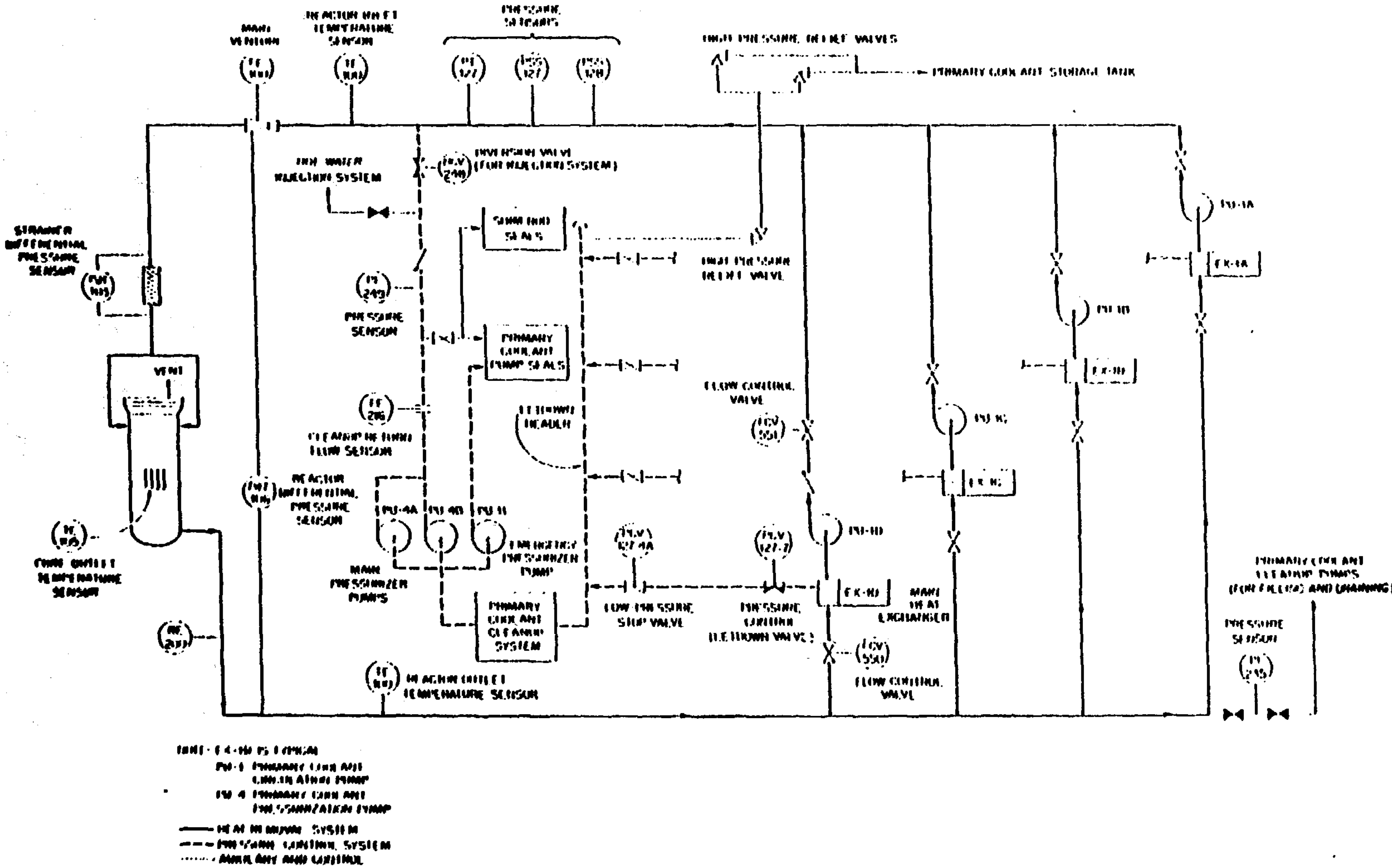

Figure 1. The HFIR primary coolant system.

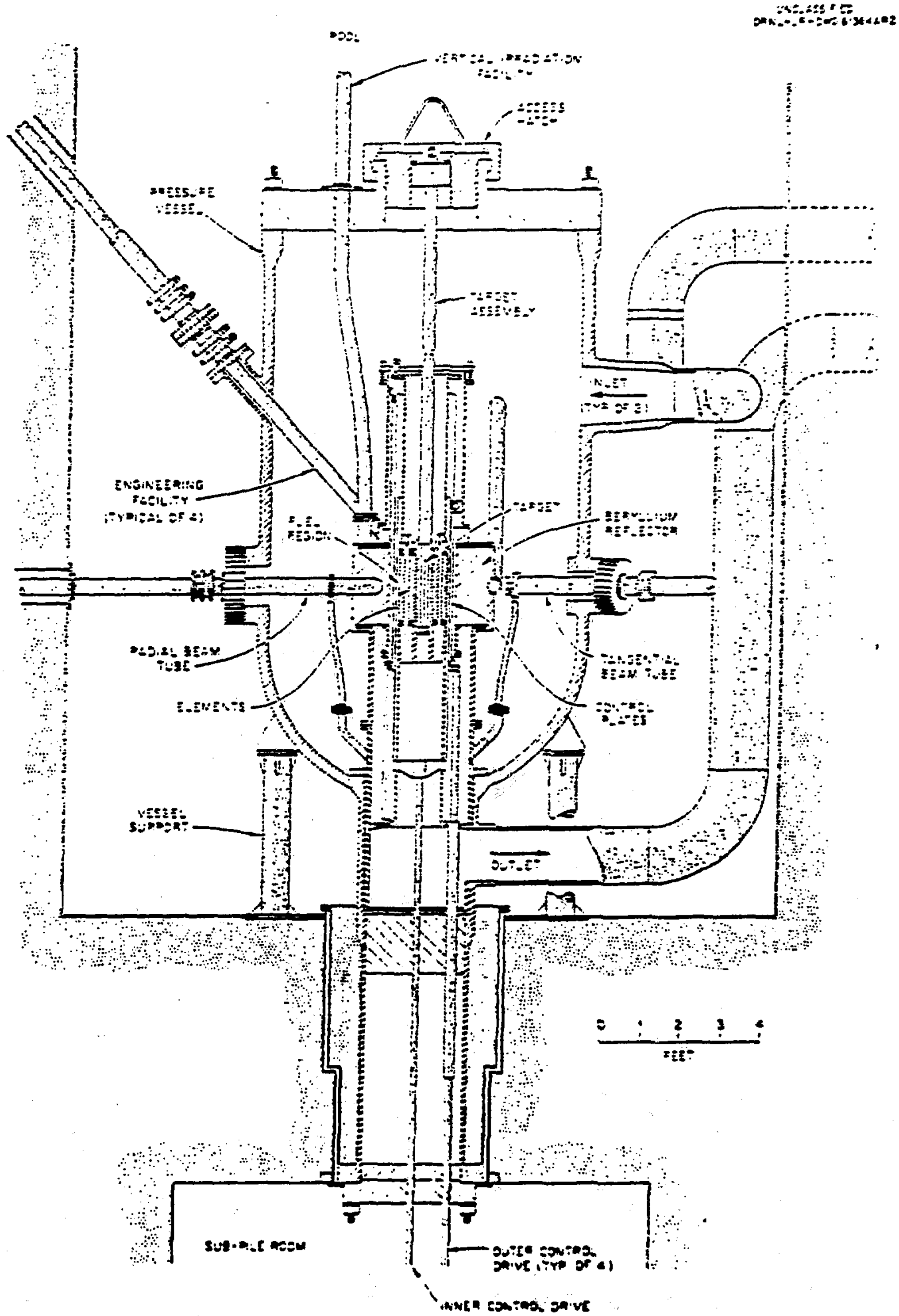

Figure 2. The HFIR vessel cross sectional view.

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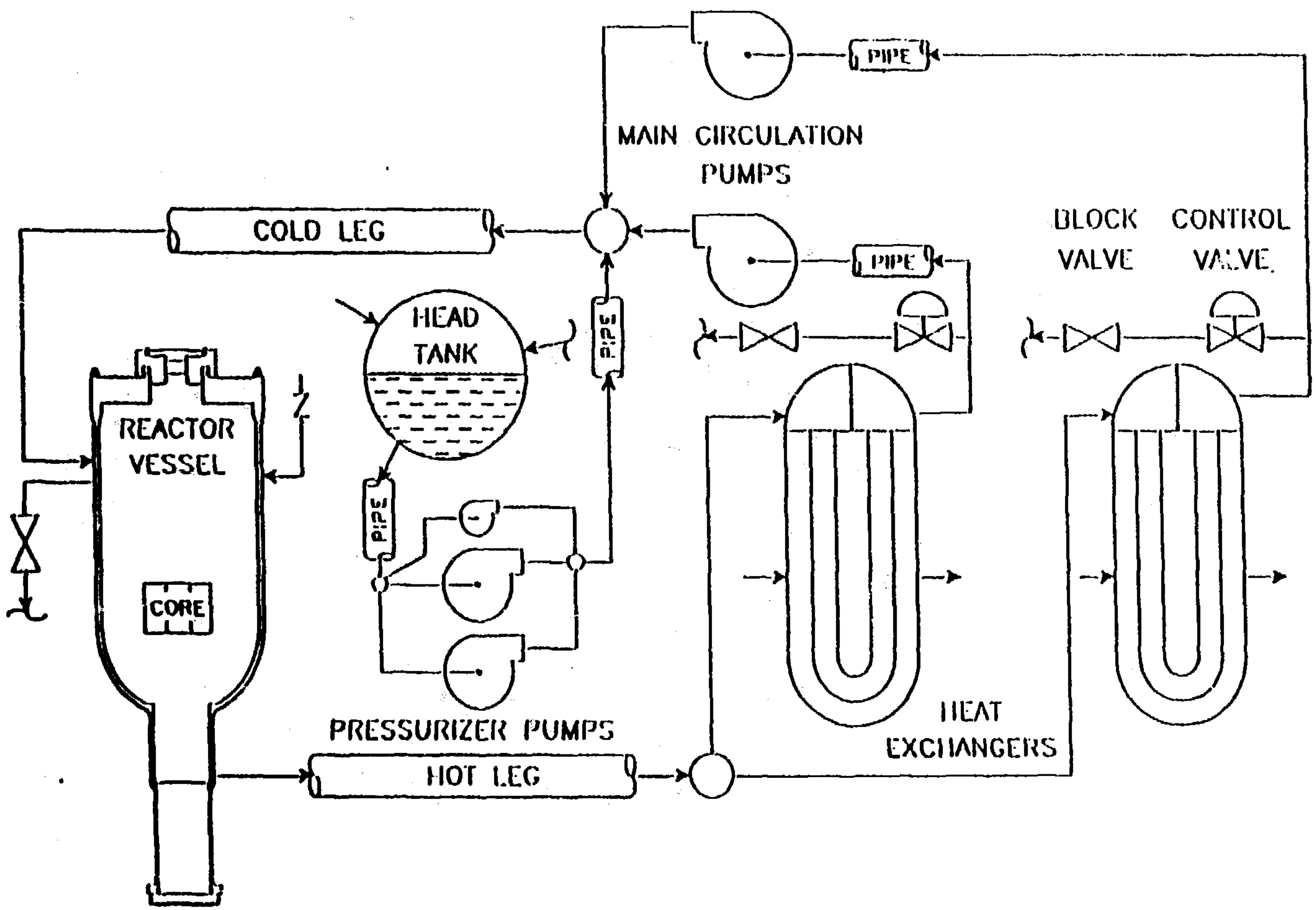

Figure 3. The schematic drawing of the HFIRSYS computer model.

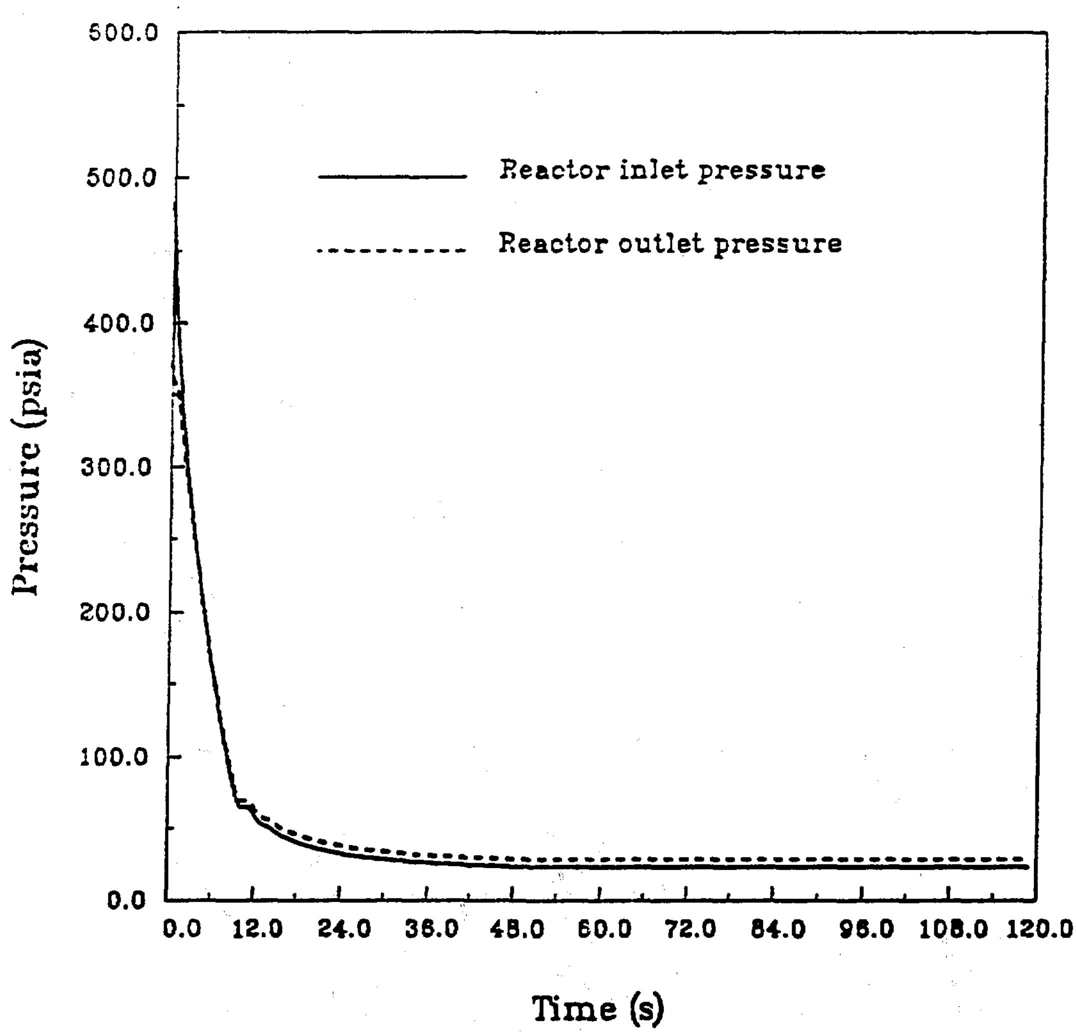

Figure 4. The reactor vessel inlet and outlet pressures.

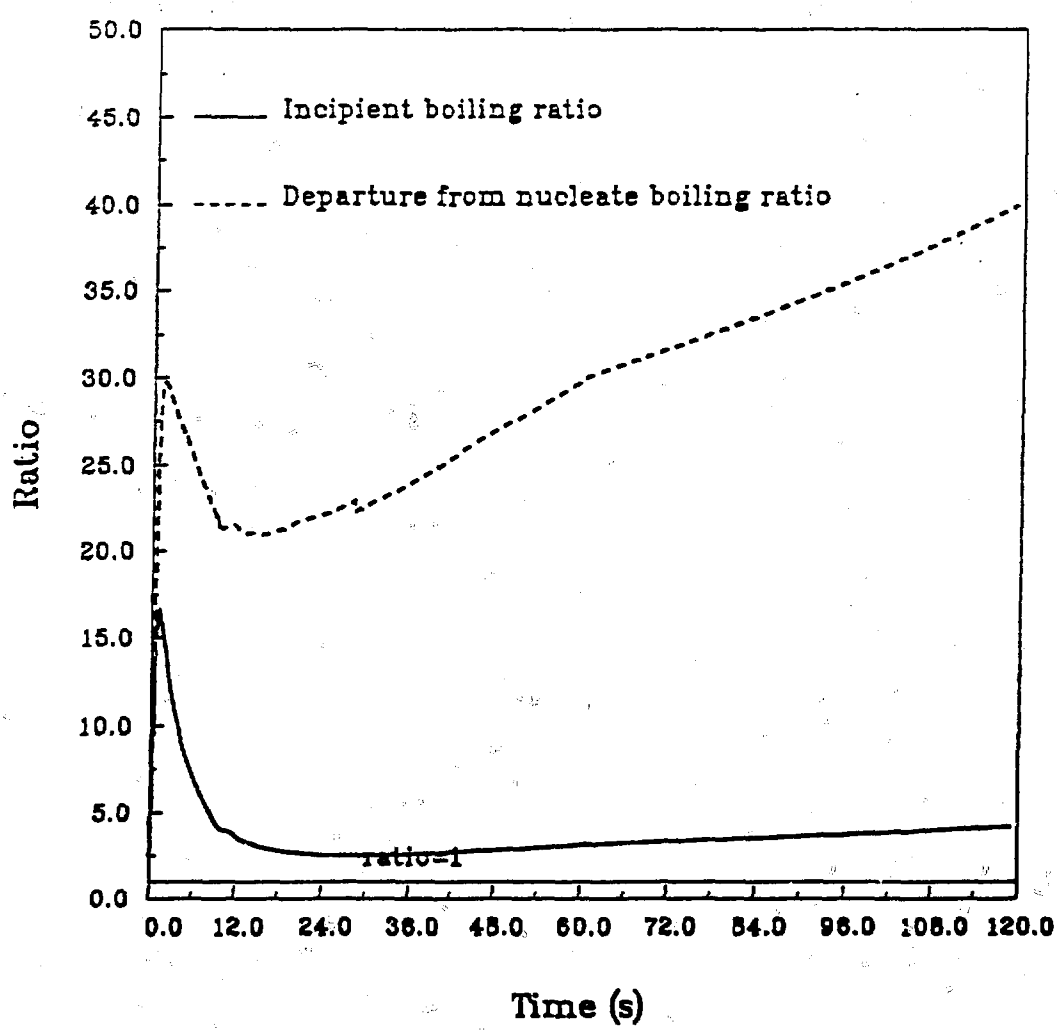

Figure 5. The incipient boiling and critical heat flux ratios.

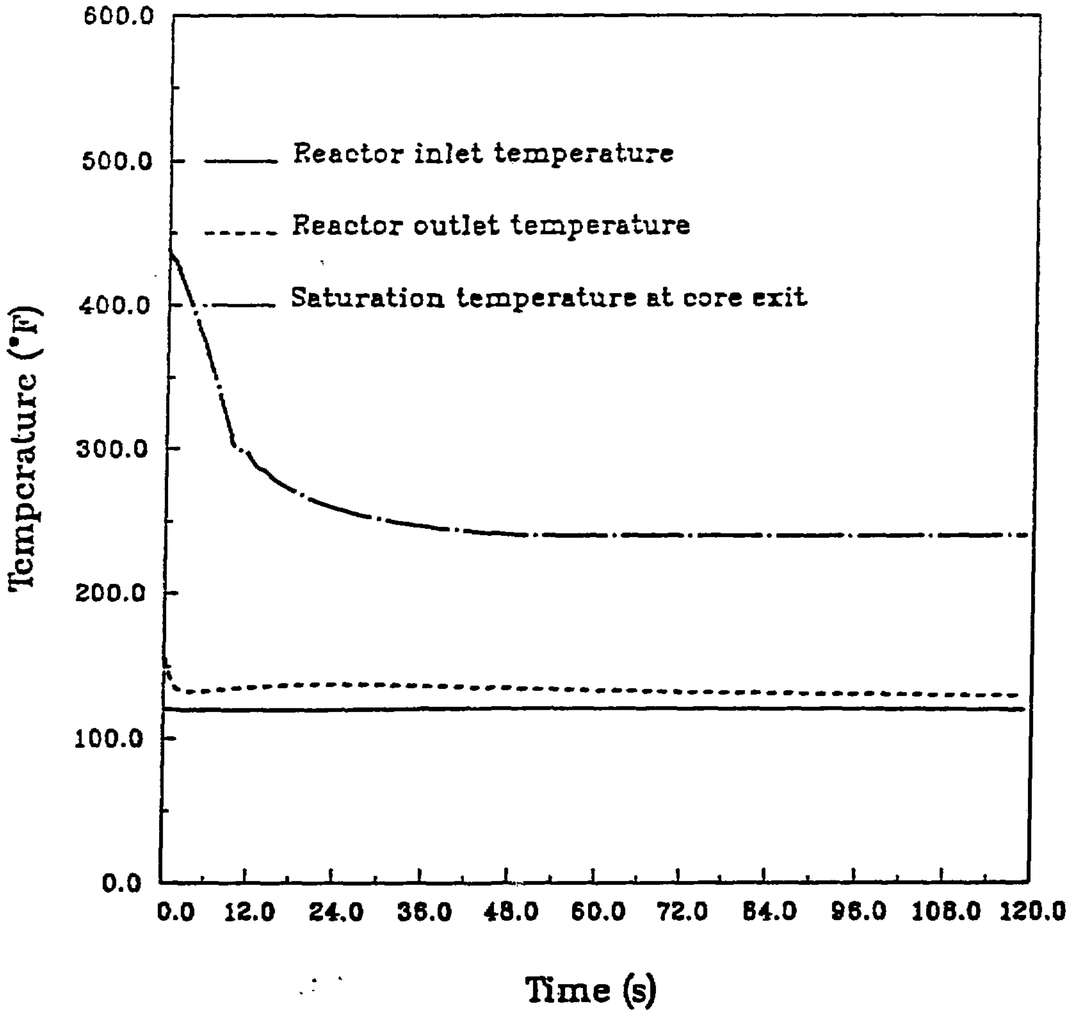

Figure 6. The reactor vessel inlet and outlet temperatures and the saturation temperature at the core exit.