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MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR SIMULATION USING PARALLEL PROCESSORS*

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ABSTRACT

The MHPP (Modular HTGR Parallel Processor) code has been developed to simulate modular high-temperature gas-cooled reactor (MHTGR) transients and accidents. MHPP incorporates a very detailed model for predicting the dynamics of the reactor core, vessel, and cooling systems over a wide variety of scenarios ranging from expected transients to very-low-probability severe accidents. The simulation routines, which had originally been developed entirely as serial code, were readily adapted to parallel processing Fortran. The resulting parallelized simulation speed was enhanced significantly. Workstation interfaces are being developed to provide for user ("operator") interaction. The benefits realized by adapting previous MHTGR codes to run on a parallel processor are discussed, along with results of typical accident analyses.

MODULAR HTGR DESCRIPTION

The U.S. Department of Energy (DOE) Standard MHTGR is shown schematically in Fig. 1. Each reactor module consists of a tall cylindrical ceramic core with a thermal power rating of 350 MW and a once-through steam generator with a superheater to provide high-temperature (538°C, 1000°F) steam to a steam header and turbine plant common to four modules. The rated electrical output of a standard four-module plant is 540 MW, with a net thermal efficiency of 39%. The high-pressure helium coolant is driven downward through the core by a single electric motor-driven circulator. The shutdown cooling system (SCS), located within the reactor vessel, is used for decay heat removal during maintenance. In cases for which neither the main loop nor the SCS loop is available, afterheat is removed by the passive, safety-grade air-cooled reactor cavity cooling system (RCCS). The RCCS is in operation at all times and does not require any operator or automatic actuation. There is no conventional containment building, since the multilayered refractory coatings on the microscopic fuel particles are claimed by DOE to be a sufficient containment barrier.

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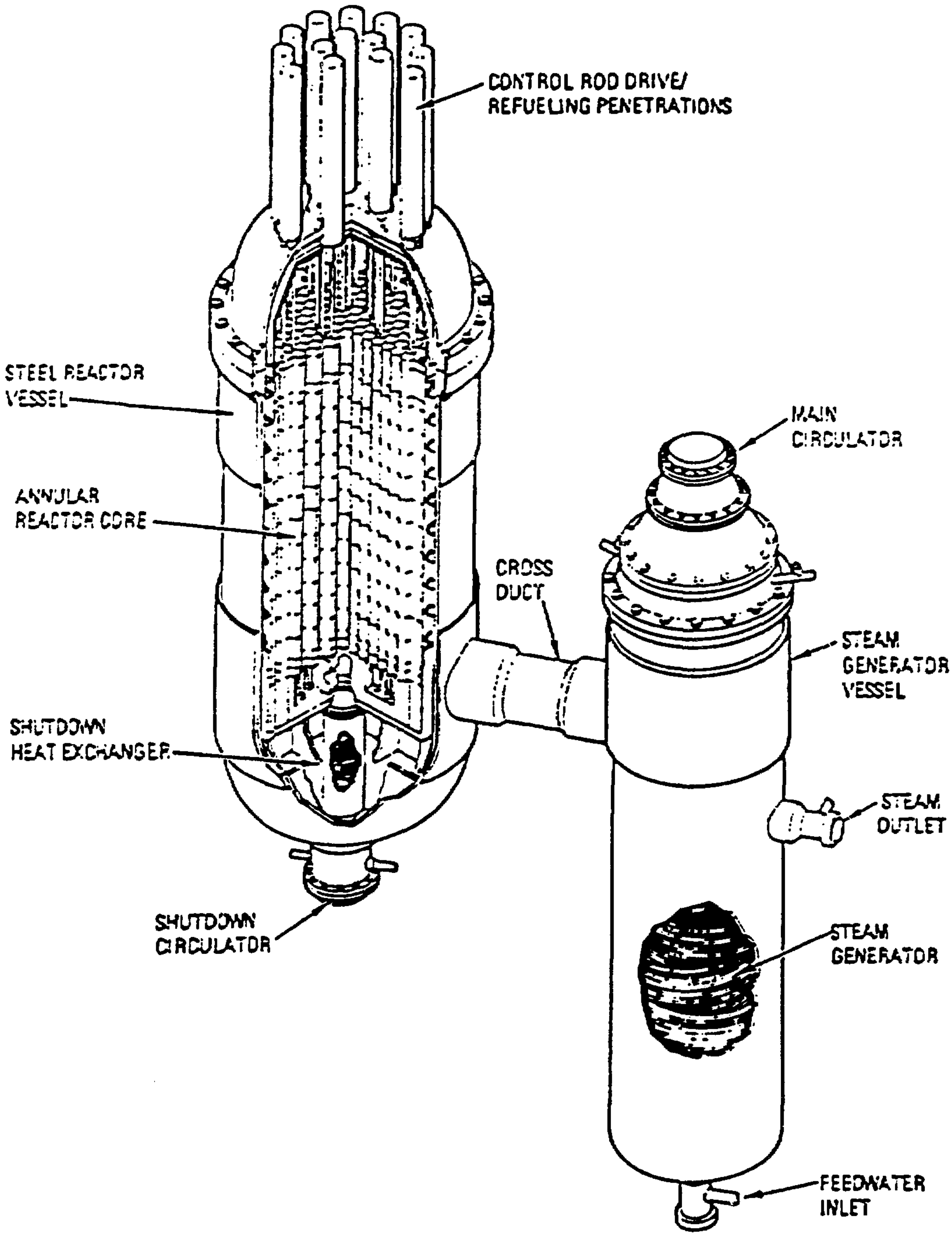

Fig. 1. MHTGR nuclear steam supply module

MHPP CODE DESCRIPTION

MHPP is a parallel Fortran extension of several different (serial) codes developed previously by Oak Ridge National Laboratory (ORNL) under Nuclear Regulatory Commission (NRC) research program sponsorship. The most recent of these is the MORECA code.[1] MORECA was derived from earlier ORNL codes having a long history of use, validation, and verification.[2,3]

The model for the reactor core consists of a node with variable thermal properties for each of the 66 fuel and 139 reflector elements in all 14 axial regions. This fine structure (2870 nodes) permits investigations of azimuthal asymmetry, and allows investigations of worst-case postulated accidents where the fuel failure rate is highly sensitive to the time the fuel spends at high temperatures. Coolant flow is modeled explicitly for each fuel element over the full ranges expected in both normal operation and accidents, including pressurized and depressurized conditions for either forced or natural convection, upflow and downflow.

Other features of the current MHPP model are summarized as follows:

1) The core barrel and vessel are each represented by 7 axial and 4 radial nodes (quadrants), plus "roof" and "floor" heat shield nodes corresponding to regions opposite the inlet and outlet plenums.

2) The reactor cavity cooling system model computes the natural convection air flow, using independent flow and heat transfer (radiation and convection) equations for each of four RCCS quadrant panels. This allows for study of the full range of expected performance and degraded states, including partial and total failures.

3) The shutdown cooling system is for use during shutdowns when main loop cooling is not available. The SCS consists of a single electrical motor-driven circulator and a pressurized-water tube-in-shell heat exchanger. It is not currently considered to be a safety-grade system. The SCS model is used to investigate scenarios where forced circulation flow is restored following long heatup periods during which no forced circulation is available. In some HTGR designs this can become an operation limiting situation for fear of damage to components downstream of the hot core outlet gases.

The balance of plant and the module control and safety system will also be modeled to the extent that they can be used effectively to provide for simulated operator interaction. Future plans also call for the incorporation of all four plant reactor modules in the simulation.

NOTES ON PARALLEL PROCESSING

The MHPP code simulation models were implemented on the new ORNL Instrumentation and Controls Division Advanced Controls Laboratory's parallel processor, a 10-processor Encore Multimax 320 computer installed in the Fall of 1988. The Multimax has parallel Fortran capabilities and constructs that are similar to those of most of the shared memory machines now on the market, and utilizes an extension of the UNIX BSD 4.2 operating system. It can also be used as a conventional time-sharing machine, and is linked via Ethernet to an array of UNIX-based workstations in the Advanced Controls Lab.

The development of a parallel code for the MHTGR simulation was undertaken for the following reasons:

1. MHTGR plants consist of several primary systems (four in the standard DOE design) which operate in parallel and in conjunction with a common feedwater and steam turbine balance of plant system. The use of one or more parallel processor to simulate each primary system is therefore a logical way to allocate the computing chores, since these systems (and simulations) are loosely coupled. Enhanced efficiency can be realized both in execution speed and in programming.

2. There is a need to provide the user-operator with on-line interaction capabilities that are not feasible with the conventional, large batch processors available at ORNL (IBM 3033 and Cray XMP). Multimax links with workstations (also operating in parallel) will permit on-line graphics display, run control, and "plant operator action" capabilities for the simulator.

The current version of the MHPP code uses parallel algorithms for a single primary and afterheat-removal system simulation. This was accomplished by revising the serial code (MORECA) to share the computing chores among the processors, primarily by splitting up the most time-consuming DO-loop calculations having "independent" loops. Using 9 of the currently available 10 processors resulted in speedups by a factor of between 5 and 6 over the serial code. Simulation speeds of greater than 1000 times faster than real time have been achieved with the Multimax in "single user" mode, thus allowing week-long plant transients to be calculated and displayed in as little as 10 minutes.

EXAMPLE RESULTS

Severe accident studies for the MHTGR are somewhat different in nature from those normally considered for commercial power reactors. The MHTGR design is such that even for the very-low-probability accidents ("less than one per million plant years") there is little if any core damage and fission product release. Consequently, the DOE design does not have a conventional containment building. The MHTGR inherent safety features are of course quite beneficial in terms of both safety and investment protection, but the omission of a containment building does make it necessary to ensure that the safety analyses are complete, correct, and conservative.

One crucial type of severe accident for the MHTGR is the unrestricted core heatup accident (which means that no measures are assumed to be taken to restrict the core heatup following an accident). A major factor contributing to its low probability is the very long time (days) available in which corrective action could be taken to terminate the accident. In two different classes of heatup accidents, the passive, air-cooled RCCS was assumed to be operational. In the first, a rapid primary system depressurization and immediate loss of forced circulation (LOFC) with scram was assumed, with no subsequent primary coolant system forced cooling. In the reference case calculation (Fig. 2), peak temperatures are reached after 4 to 5 days. There is no fuel failure, as the maximum peak fuel temperature (1482°C, 2700°F) is well below the 1600°C nominal "limit," below which essentially no fuel particle failures are expected. The maximum temperature of the steel pressure vessel (479°C, 895°F) is below the 1000°F extended ASME code limit for the depressurized vessel. These results are generally in good agreement with calculations by DOE except for the vessel temperatures, where their predicted maximum was less than 427°C (800°F). Reasons for this discrepancy are still being investigated.

In the second class of heatup accidents with the RCCS operational, the pressurized LOFC with scram, the maximum fuel temperatures predicted are even lower than those in the depressurized LOFC case, so there is no concern about fuel damage. The primary concern here is that the predicted vessel temperature (maximum 468°C, 875°F) exceeds the 800°F extended ASME code limit for a pressurized vessel. The corresponding DOE prediction, using the General Atomics PANTHER code, was 400°C (750°F). Some of the discrepancies are due to simplifications in the PANTHER code that DOE plans to address in the next stages of the design; however, others have not yet been resolved.

Variations of these two classes of accidents were studied to observe sensitivities of the severity of the predicted results to both parametric (modeling) and operational assumptions. Three parametric variations were found to be of major significance to the outcome of the predictions: (1) fuel and reflector thermal conductivities; (2) use of a conservative afterheat relationship vs best-estimate values; and (3) variations in assumed RCCS performance, including effects of thermal emissivity values, which have a direct effect on transfer of heat from the core blocks to the RCCS panels.

The sensitivity studies indicated that while several-hundred-degree variations in peak fuel temperatures were possible due to reasonable variations in these three assumed parameter categories, "worst-case" values still gave acceptable results. While some predictions gave higher-than-acceptable vessel temperatures, the maximum temperatures and time-at-temperature transients were not severe enough to cause vessel failure. Studies were made of restarting cooling flow using the SCS after arbitrary heatup periods. The preliminary analysis has shown that in no case is the resulting core outlet temperature high enough to damage the metallic structures or components in the primary system.

A "complete" long-term failure of the RCCS is currently considered as a nonmechanistic failure, since no reasonable mechanisms have been postulated to cause such failures, assuming that the RCCS is built to the proposed quality specifications. Even so, a total RCCS failure was calculated in which the structure with its insulation between the riser and downcomer is assumed to be in place, but with no air flow. Conduction and thermal radiation to the concrete underground silo are modeled simplistically, with credit taken for the concrete heat capacity but no credit taken for heat losses to the upper and lower heads. The results are shown in Fig. 3. Although the peak fuel temperature slightly exceeds 1600°C, fuel failure would be insignificant. The vessel temperature, however, would exceed ASME code values in about 1 day, and in 2 to 4 days, temperatures could reach the point at which possible concrete degradation and vessel support failures would make it difficult to define recovery action alternatives.

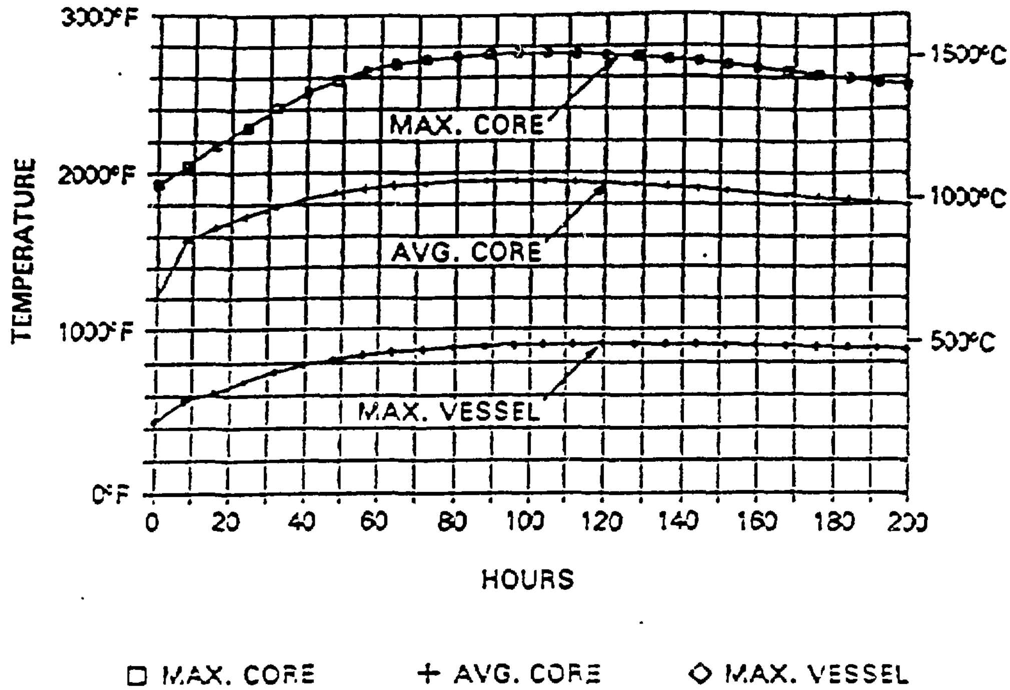

Fig. 2. Reference case depressurized loss-of-forced-convection (LOFC) accident

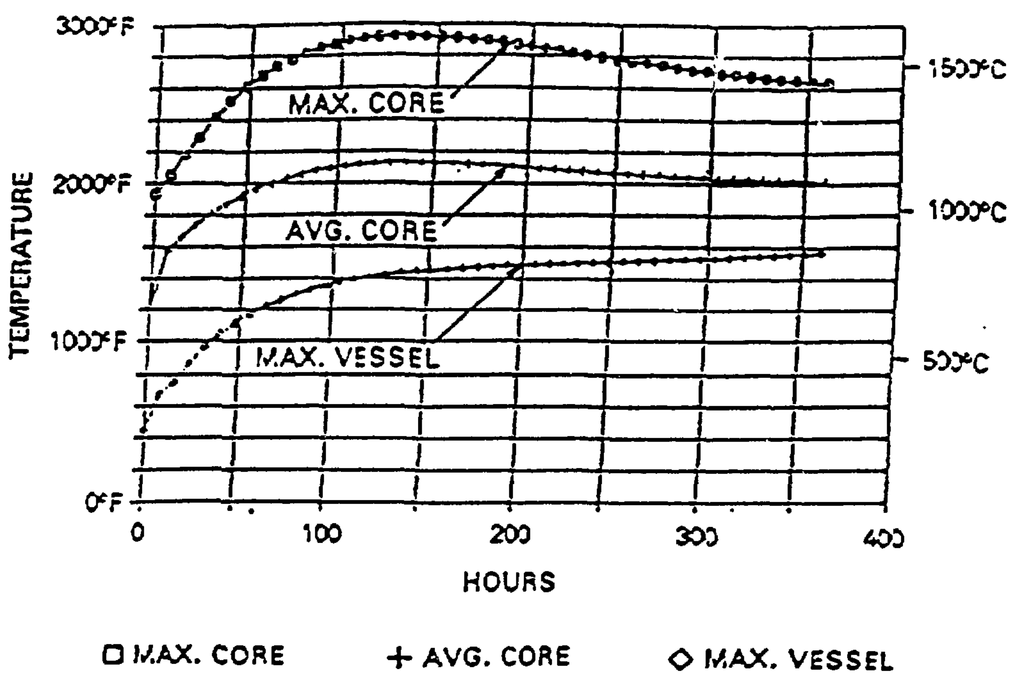

Fig. 3. Response to complete RCCS failure (depressurized LOFC).

CONCLUSIONS

MHPP is a parallel-processor simulation of the DOE Standard MHTGR which was developed to simulate a wide spectrum of transients and accidents efficiently, and to provide the user/operator with the capabilities of controlling the progress of the simulation and influencing plant operational parameters.

The LOFC heatup accident analyses show that the current MHTGR design is not susceptible to significant fuel failure from wide ranges of postulated design basis accidents, even those with very low probabilities. The same is true even for certain drastic, nonmechanistic events. This conclusion is based on the assumption that the R&D work planned by DOE is successful in confirming certain key design and inherent safety characteristics assumed in our models. These ORNL results generally correspond very well with independent calculations by DOE contractors and by Brookhaven National Laboratory. Considering that these are calculations of some of the most serious types of accidents that can be reasonably postulated, such good agreement indicates that the analyses are relatively straightforward and credible.

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