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MORECA-2: Interactive Simulator for Modular High-Temperature Gas-Cooled Reactor Core Transients and Heatup Accidents with ATWS Options

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This is a follow-up to an earlier report documenting the MORECA code, an interactive simulation tool for performing independent analyses of postulated modular high-temperature gas-cooled reactor (MHTGR) core transients and heatup accidents. This research was performed at Oak Ridge National Laboratory to assist the Nuclear Regulatory Commission in preliminary determinations of licensability of the U.S. Department of Energy reference design of a standard MHTGR. The additional features of MORECA documented in this report are the interactive workstation capabilities and the options for studying anticipated transients without scram events.

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1. REACTOR DESCRIPTION

The U.S. Department of Energy (DOE) standard MHTGR[1] consists of four tall cylindrical ceramic core reactor modules each with a thermal power rating of 350 MW and a single once-through steam generator with a superheater to provide high-temperature [538°C (1000°F)] steam (Fig. 1). High-pressure helium is driven downward through cooling channels in the annular core by a single motor-driven main circulator. A smaller capacity circulator/heat exchanger loop, the shutdown cooling system (SCS), is located within the steel reactor vessel. In cases for which neither the main nor the SCS loop is available, afterheat is removed by a passive, safety-grade air-cooled reactor cavity cooling system (RCCS) surrounding the reactor vessel. RCCS is in operation at all times and does not require operator or automatic actuation in the event of an accident.

Instead of using a conventional sealed containment building, the reactor is housed in an underground silo with a vented, moderate-leakage reactor building above. The overall "containment" design is centered on silicon carbide and pyrolytic carbon coatings on the microscopic fuel kernels which, together with the primary pressure boundary, are considered by DOE to be an adequate containment barrier.

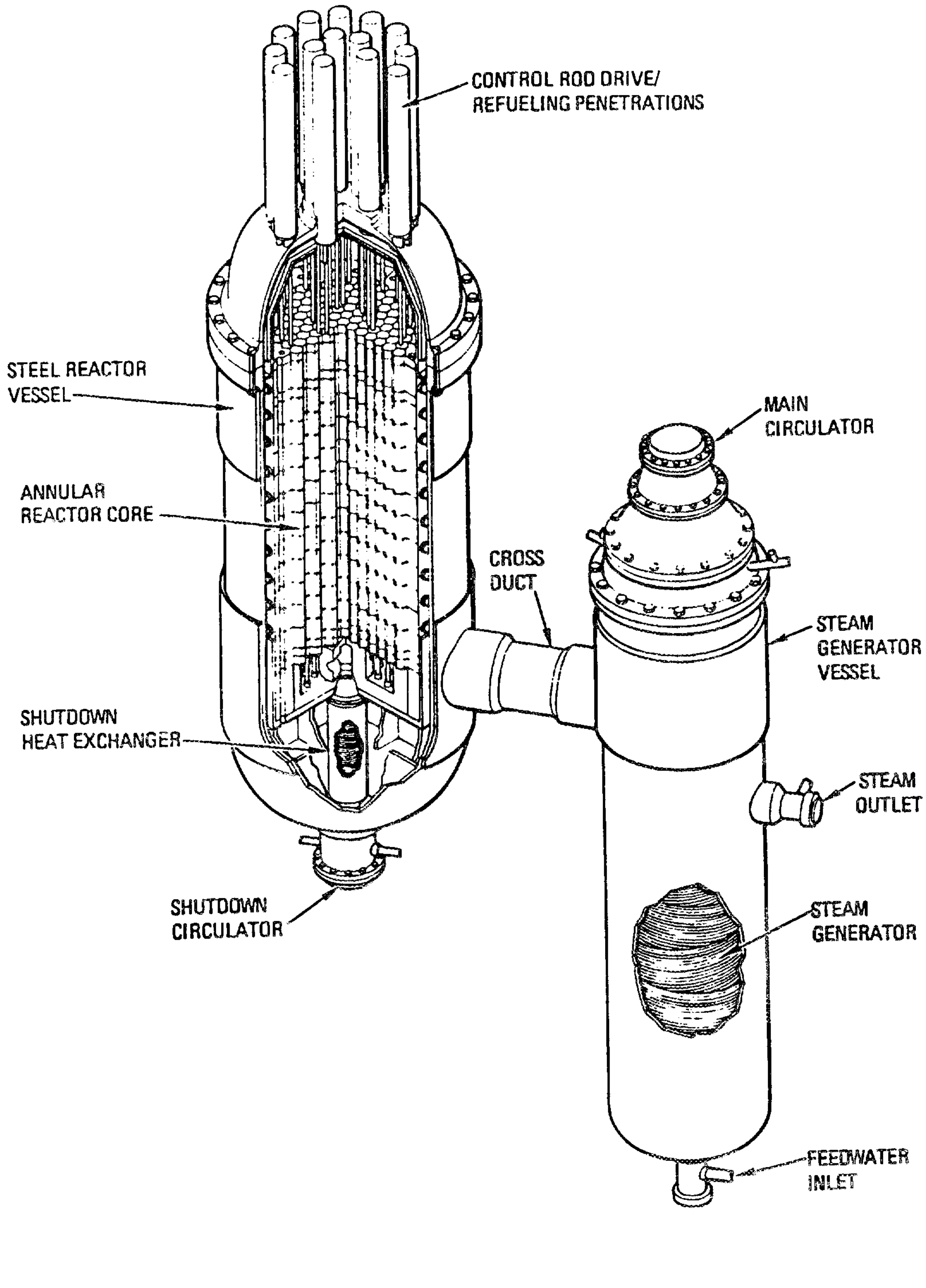

Fig. 1. The 350-MW(t) modular high-temperature gas-cooled reactor module. *Source:* U.S. Department of Energy, *Licensing Plan for the Standard MHTGR*, HTGR-85-001, Rev. 3, 1986 (this document is classified as "Applied Technology" and is not in the public domain; requests for this document should be made through the U.S. Department of Energy, Washington, D.C.).

2. MODEL DESCRIPTION

The ORNL MORECA code[2] was developed to study core transient and heatup accident scenarios and, thus, includes detailed thermalhydraulic models for the core, vessel, SCS, and RCCS. MORECA-2 includes the recent addition of core point kinetics. The interactive workstation and graphics features of the upgraded code are also documented here. The steam generator and balance of plant are currently not modeled.

Details of the models used in MORECA are given in reference 2. Model equations, with the exception of those used for the anticipated transient without scram (ATWS) capabilities, are given in the appendix of reference 2. The three-dimensional core model uses one node each for the 66 fuel and 139 reflector elements in each of 14 axial regions. The core representation ($205 \times 14 = 2{,}870$ nodes) allows for detailed investigations of azimuthal temperature asymmetries in addition to axial and radial profiles. Variable core thermal properties are computed functions of temperature and are dependent on orientation and radiation damage.

The primary coolant flow models cover the full ranges expected in both normal operation and accidents including pressurized and depressurized (and in between) for forced and natural circulation (upflow and downflow) and for turbulent, laminar, and transition flow regimes. The primary loop pressure calculation considers variable inventory (due to depressurization actions) and loop temperature changes and uses a simplified model of the steam generator cavity gas temperatures.

3. ATWS CAPABILITIES

The recent improvements to MORECA include interactive and graphic capabilities (for the workstation version) as well as the ability to treat ATWS events. With the use of the ATWS option, the expected scram does not occur at the time of a loss of forced circulation (LOFC) but, instead, is assumed to occur at an arbitrary later time or not at all. Slow rod-withdrawal accidents can also be simulated if they are in conjunction with an LOFC accident. The model for fuel (as distinct from moderator) temperature is a quasi-steady state approximation valid for only slow transients characteristic of LOFC accidents. The point kinetics approximation for the neutronics is a prompt-jump, single-precursor group model which compares favorably, for transients of the appropriate rate and magnitude, with calculations using a "full" model with prompt-neutron generation time and six delayed-neutron precursor groups included. Temperature-reactivity feedback from the three-dimensional modeling of fuel, moderator, and reflectors uses nuclear importance weighting. Models for xenon and samarium poisoning are included. Details of the ATWS model are given in Appendix A.

4. MORECA CODE DOCUMENTATION FEATURES

Documentation for MORECA is aided by the use of databases (dBase-3+ or dBase-4) that were developed to assist MORECA code users with program interpretation, verification, and modification. The major database file consists of a listing of program variables along with other essential information including a brief description, dimensions, what common block (if any) in which it appears, where it is defined, and where (else) it is later modified. A flag symbol in the description field is used to denote variables read in as part of the input file. The advantage of storing all this information in a large relational database is that cross-checking and special listings or searches can easily be done by simple dBase commands or programs. Two other smaller database files exist — one containing common block variable lists and the other containing subprogram information with data such as what common blocks are used, what other programs are called, and what variables are in its argument list. The latter database also has details (in the memo field) on the function of each subprogram. An updated listing of the database documentation is given in Appendix B.

5. MODEL VALIDATION

Considerable effort has gone into validation studies, making use of applicable data from the Fort St. Vrain (FSV) reactor and other special tests of high-temperature gas-cooled (HTGR) dynamic characteristics. The comparisons with FSV scram test data[3] were made by using the ORECA code[4], the forerunner of MORECA. The ORECA model of the prismatic core used coarser noding, with seven fuel elements per node vs one per node in MORECA, so the finer structure modeling should be at least as good as that used in the FSV comparisons. A special workstation program was also written to make use of FSV plant data logger tapes in code validation studies. All FSV data from 1984 has been processed and can be accessed interactively with the help of graphical displays.

Other comparisons with rod-jog tests validated the methodology and assumptions for point kinetics for the relatively slow reactivity transients modeled here in ATWS accidents.[5] ATWS tests run on the German Arbeitsge-meinschaft Versuchs Reaktor (AVR), a pebble-bed core HTGR, also confirmed (qualitatively) the features of the response predicted by MORECA.[6,7]

6. MORECA WORKSTATION VERSION

With the workstation version of MORECA, the operator/analyst is allowed direct on-line involvement with the postulated scenarios. Wide varieties of transients and LOFC accidents can be studied interactively, including long-term core heatup scenarios for which active cooling systems are either available or available only intermittently in degraded states. LOFC accidents can be simulated both with and without total or partial depressurization of the primary coolant and with or without scram.

The workstation display screen for accident analyses (Fig. 2) presents a summary status of the simulation for RCCS, vessel, core, and SCS. Along the bottom of the screen are the "buttons" (accessed by a mouse) allowing operator intervention and keyboard input for some control values. The controls include the simulation speed, plot activation, control of the SCS operating parameters, allowance for degrading the effectiveness of RCCS, and control of partial or total depressurization transients. The maximum vessel and core temperatures are displayed at elevations corresponding to their axial locations.

Dynamic time-history plots of selected variables can be displayed. A plot point interval of 10 min is used with the dynamic graphs to prevent having to change the time scale when the program's time step changes and to avoid confusion when looking at the graphs to determine the current scale. The plots automatically rescale when the dependent variable exceeds its upper or lower limits, and the graphs scroll when the plots reach the end of their current time scale. The variables to be plotted, the number of points to display on the graph, the number of points to scroll, and the (initial) upper and lower limits of each variable can be selected by the user. The dynamic plots and the workstation display use X-Windows; therefore, the operator/analyst can use one screen to view the workstation display screen and specify a second machine to display the dynamic plots. This convention enables the user to watch both displays simultaneously. Details of the display and graphics programs are given in Appendix C.

These display features are expected to be useful for review and confirmation studies of the safety system design, operator emergency procedures, operator training procedures, and postaccident monitoring systems. Because computations are fast (up to 2,100 times faster than real time on a Sun SPARC Station-2 for non-ATWS transients), sensitivity studies can be run readily.

Another interactive display feature of MORECA-2 is the screen that displays "core map" parameters (Fig. 3). This screen is used for studying effects of operational parameters (such as core flow, inlet temperature, power, and pressure) on the three-dimensional distributions of fuel temperatures and the core flow redistributions. Operator control of these parameters is by mouse access to buttons along the bottom of the screen and keyboard input. The maps display the 66 fuel (or upper/lower reflector) temperatures in any of the 14 axial regions, the outlet gas temperatures from each region, or the individual element flows, as selected. Another important variable affecting the fuel-element flow and temperature distributions is the assumed fraction of total core flow bypassing the fuel-element cooling channels. In the MORECA model, the "hot" bypass flow is assumed to be distributed uniformly in the spaces between reflector elements, where it is heated and eventually mixes with the cooling channel flows in the lower plenum. The "cold" bypass flow bypasses the core entirely and is not heated before mixing with the other core-outlet flows. The total bypass-flow fraction is a parameter under operator/analyst control.

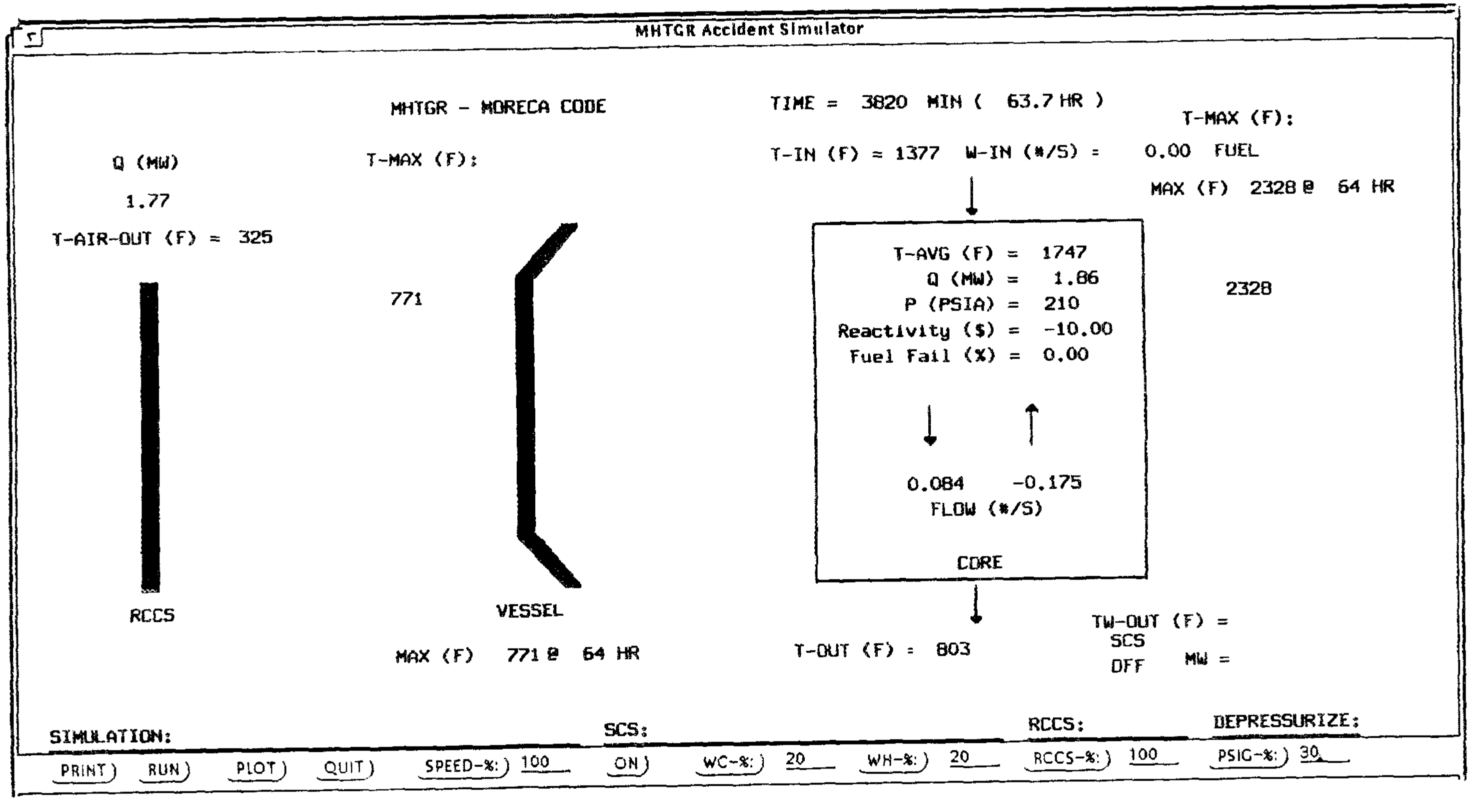

Fig. 2. MORECA interactive workstation accident scenario display screen.

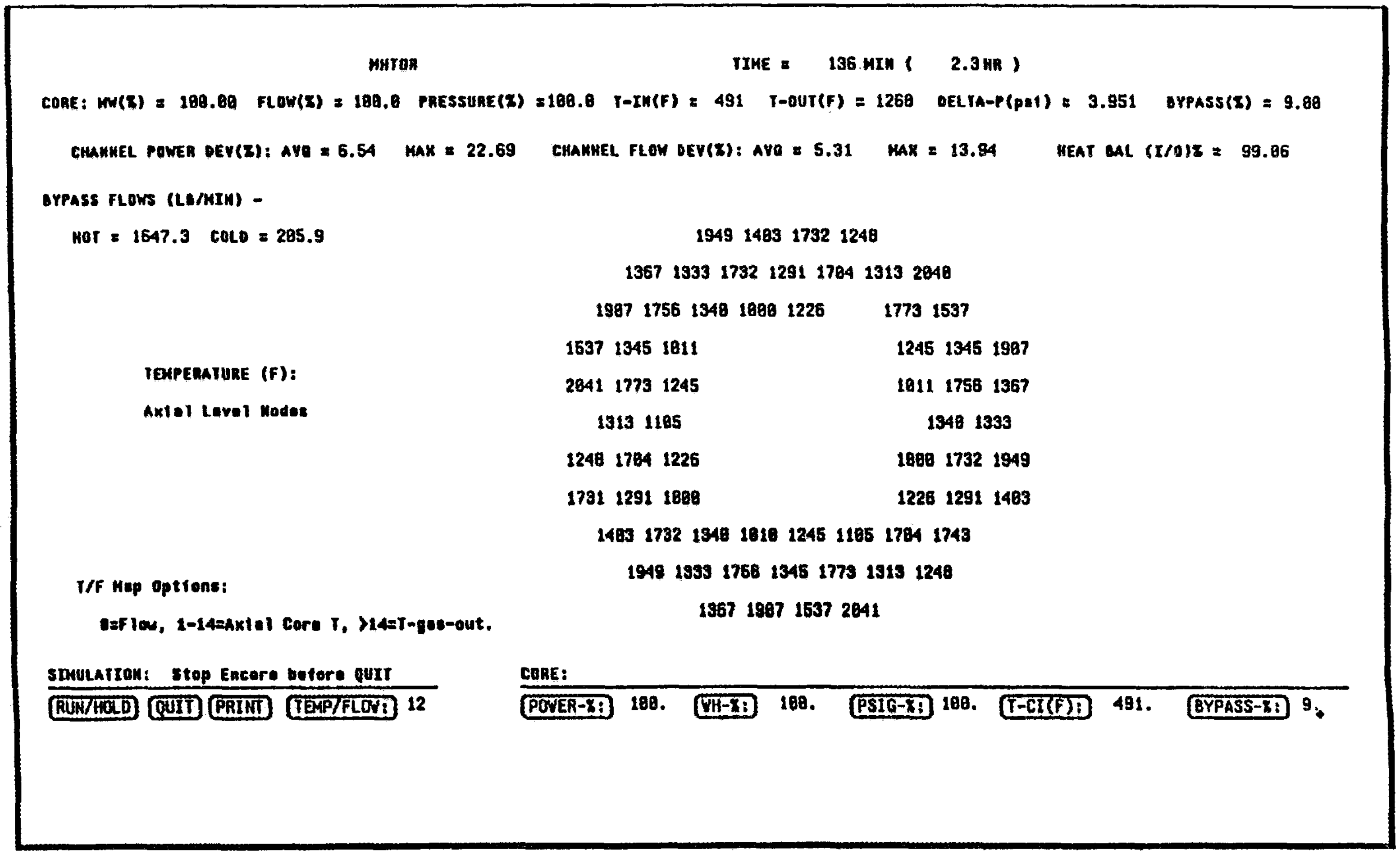

Fig. 3. MORECA interactive workstation core map display screen.

7. MORECA ANALYSIS RESULTS

The analyses of long-term LOFC accident scenarios generally show maximum fuel temperatures closely approaching or slightly exceeding the nominal failure onset limit (1,600°C) for only some ATWS cases and for some other cases in which RCCS is assumed to fail catastrophically. In certain scenarios, predicted maximum vessel temperatures exceed slightly the extended American Society of Mechanical Engineers (ASME) pressure vessel code's upper temperature limits. All of the transients are characterized by very slow heatups due to the small power densities and large heat capacities associated with the core. Results of non-ATWS accidents and other transients have been presented in previous MORECA documentation.[2]

For LOFC events accompanied by an ATWS and with no automatic or operator corrective action, recriticality following xenon poisoning decay may occur after about 40 h. Depending on the scenario and prior operating conditions, minor fuel damage would be expected (i.e., peak temperatures exceeding 1,600°C for significant periods) after 3 or 4 d. It should be emphasized that with three independent shutdown systems (inner reflector rods, outer reflector rods, and reserve shutdown system boron balls), the probability of ATWS events is extremely low, and they are considered only in bounding event sequence (BES) category scenarios.

A variety of ATWS scenarios including several in the BES category were studied. Again, these scenarios involve the assumed independent failure of all three reactivity shutdown systems and are thus of very low probability. Typically, no reactivity control action by either the control or safety systems or by the operator is assumed.

In the depressurized ATWS-LOFC reference case (Figs. 4 through 6), it is assumed that at the start of the accident, all forced circulation is stopped, the primary system undergoes a rapid depressurization, and no control or scram rod action occurs. Recriticality occurs after about 44 h because of the decay of the xenon, and the peak fuel temperature exceeds the 1,600°C "limit" in somewhat less than 3 d. With a sustained temperature of >1,700°C for several days, some fuel failure would be expected; however, with the system depressurized, minimal driving force for transport of the radionuclides to the environment would exist. The predicted peak vessel temperature is only a few degrees higher than the ASME code limit (depressurized) of 538°C.

For the pressurized ATWS-LOFC reference case (Figs. 7 through 10), the same initiating condition assumptions apply as above, except that the primary system does not depressurize. Both the peak fuel and vessel temperatures stay below their respective limits for the first 2 d. After recriticality, however, both limits are exceeded. An additional complicating factor is the potential for a depressurization, which occurs in this simulation after 4 d because the pressure exceeds the 7.18-MPa (1,041-psia) setting for the pressure relief valves (and with the assumption that the valves stick open thereafter). This depressurization would provide a significant driving force for any radionuclides present in the primary system as a result of fuel failures that may have occurred because of the sustained high fuel temperatures. Any filtering capabilities added to the pressure relief valve discharge path clearly would be of benefit in reducing the exclusion area boundary (EAB) dose. As noted before, the current simulation has only a crude simulation of the steam generator cavity gas temperature behavior (and this temperature would be very dependent on operational sequences), so the predictions of depressurizations may vary widely.

Recent studies of ATWS-LOFC scenarios have concentrated mainly on various operator-action scenarios, especially those in which the operator action is incorrect. It is clear that the peak fuel temperatures attained following recriticality are sensitive to long-term reactivity contributions from xenon and samarium and to whatever mitigation can be accomplished by poison insertion.

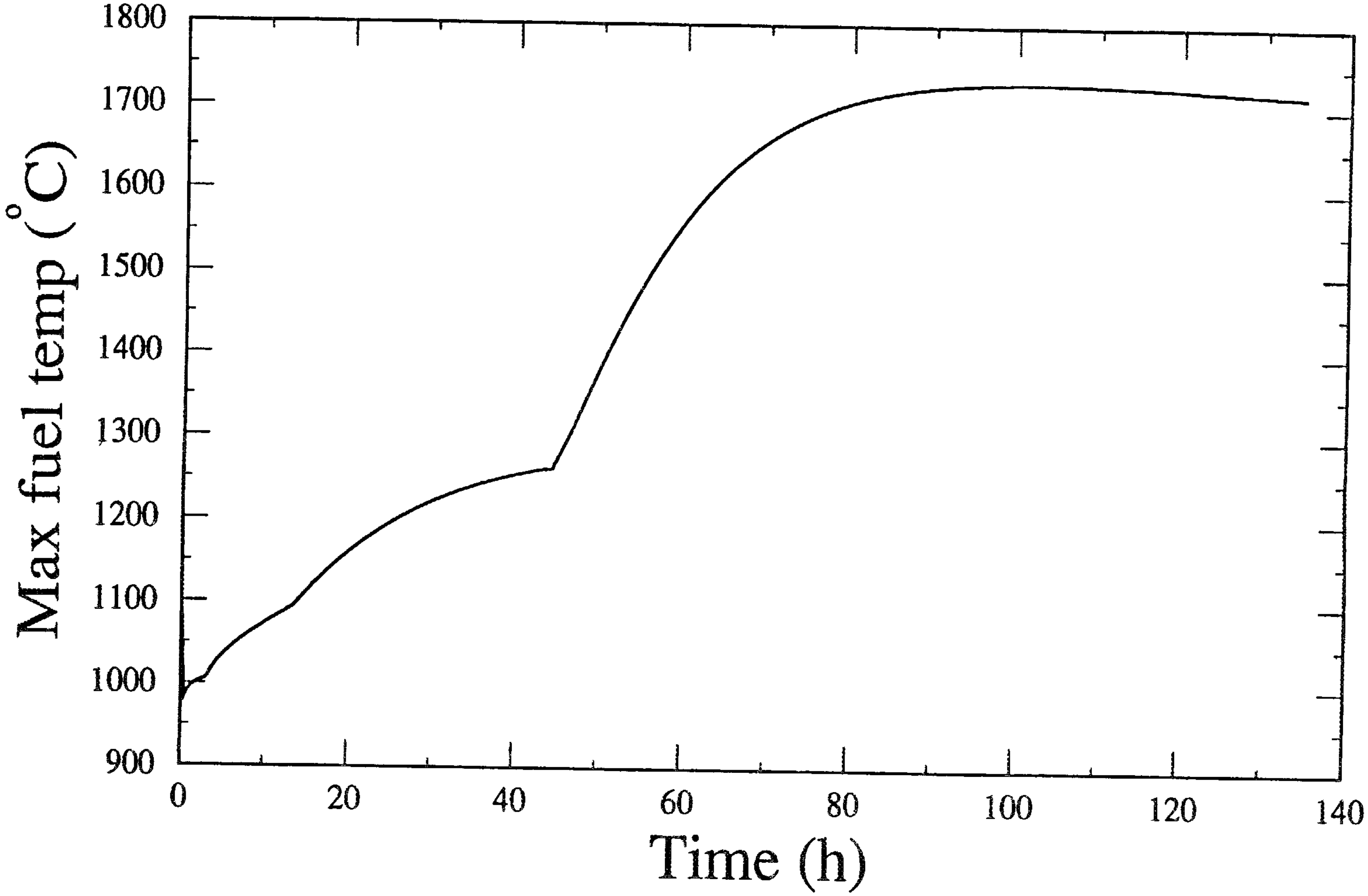

Fig. 4. Depressurized loss of forced circulation accident accompanied by anticipated transient without scram, maximum fuel temperature vs time.

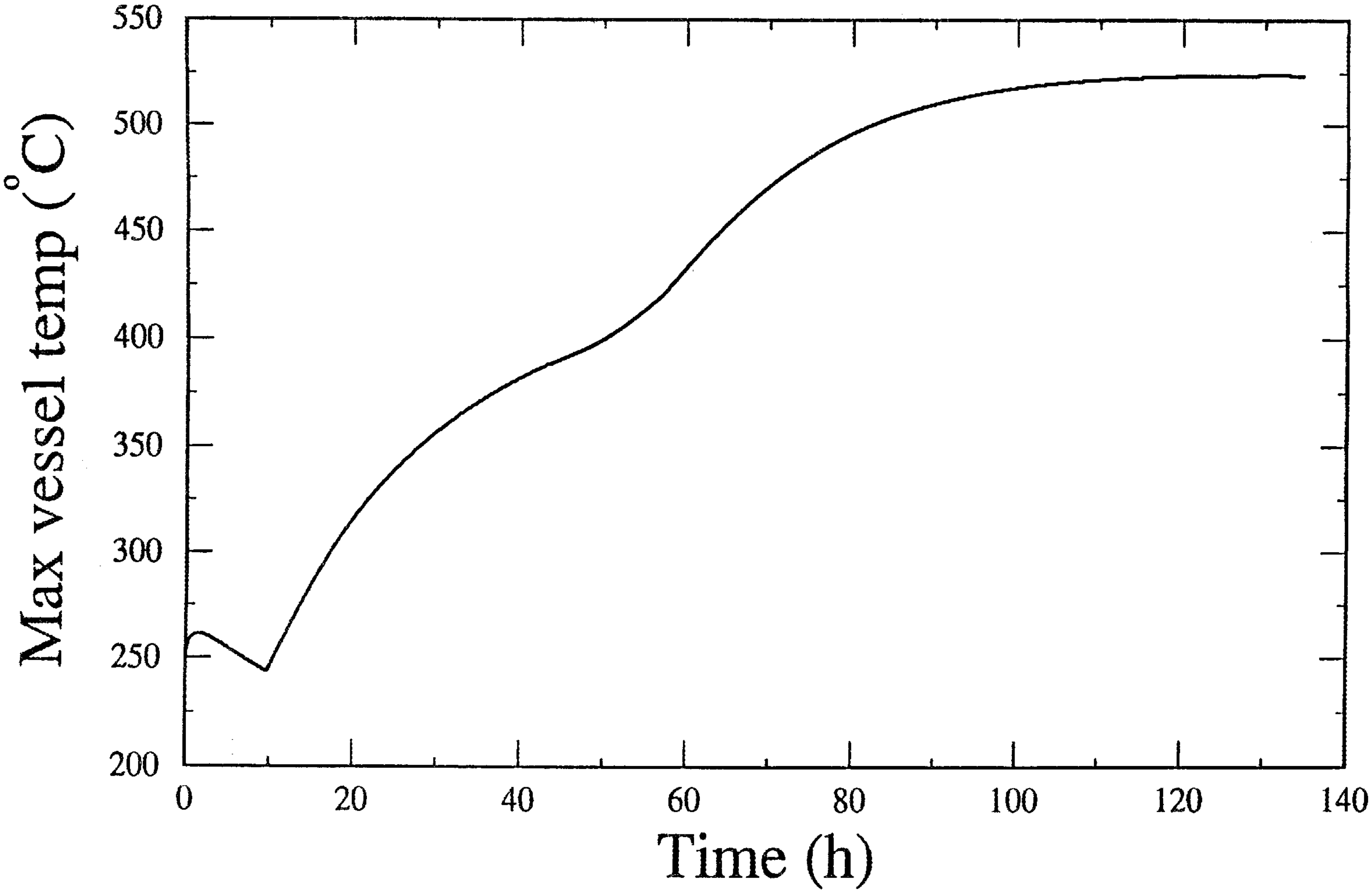

Fig. 5. Depressurized loss of forced circulation accident accompanied by anticipated transient without scram, maximum vessel temperature vs time.

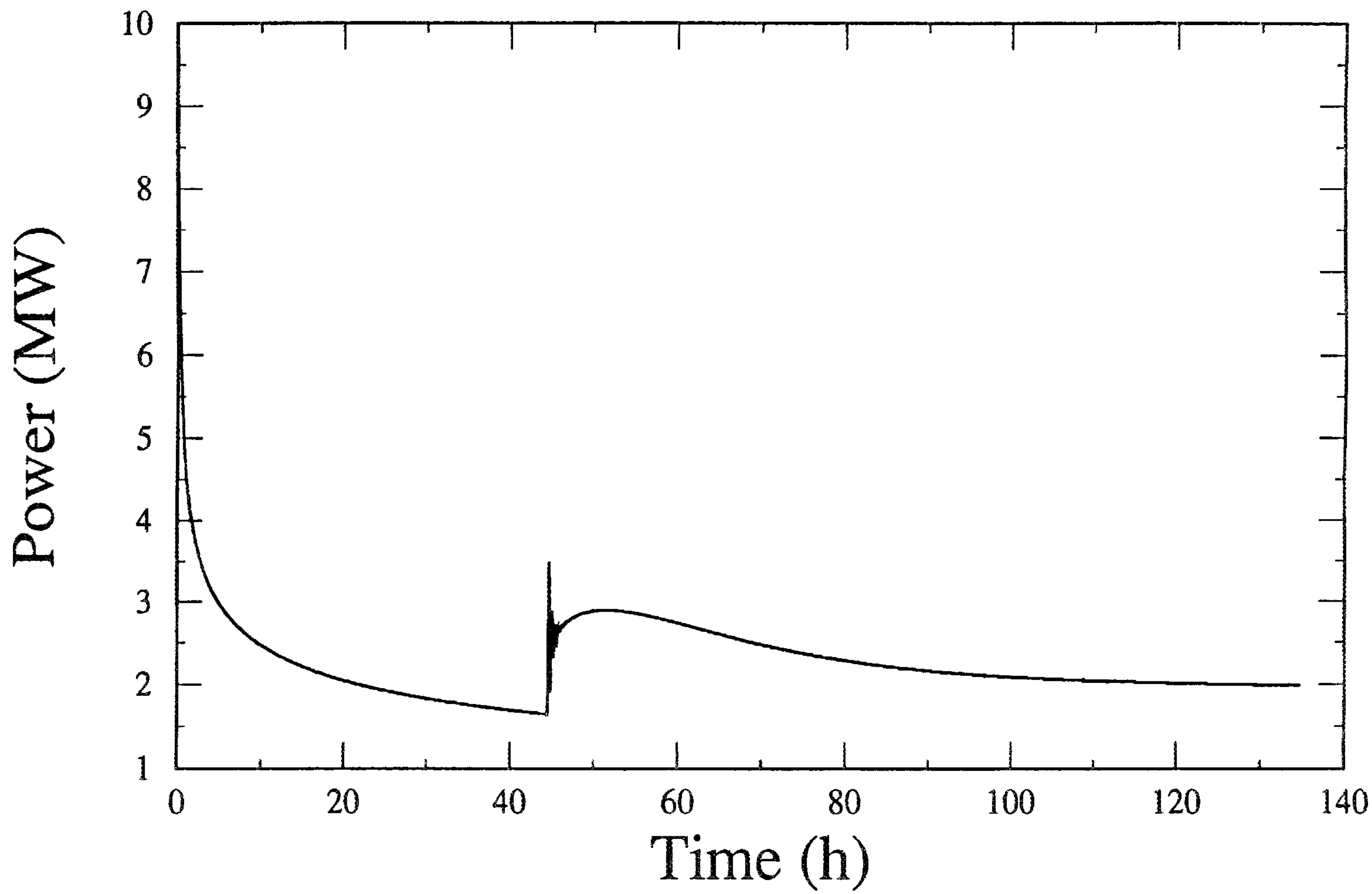

Fig. 6. Depressurized loss of forced circulation accident accompanied by anticipated transient without scram, power vs time.

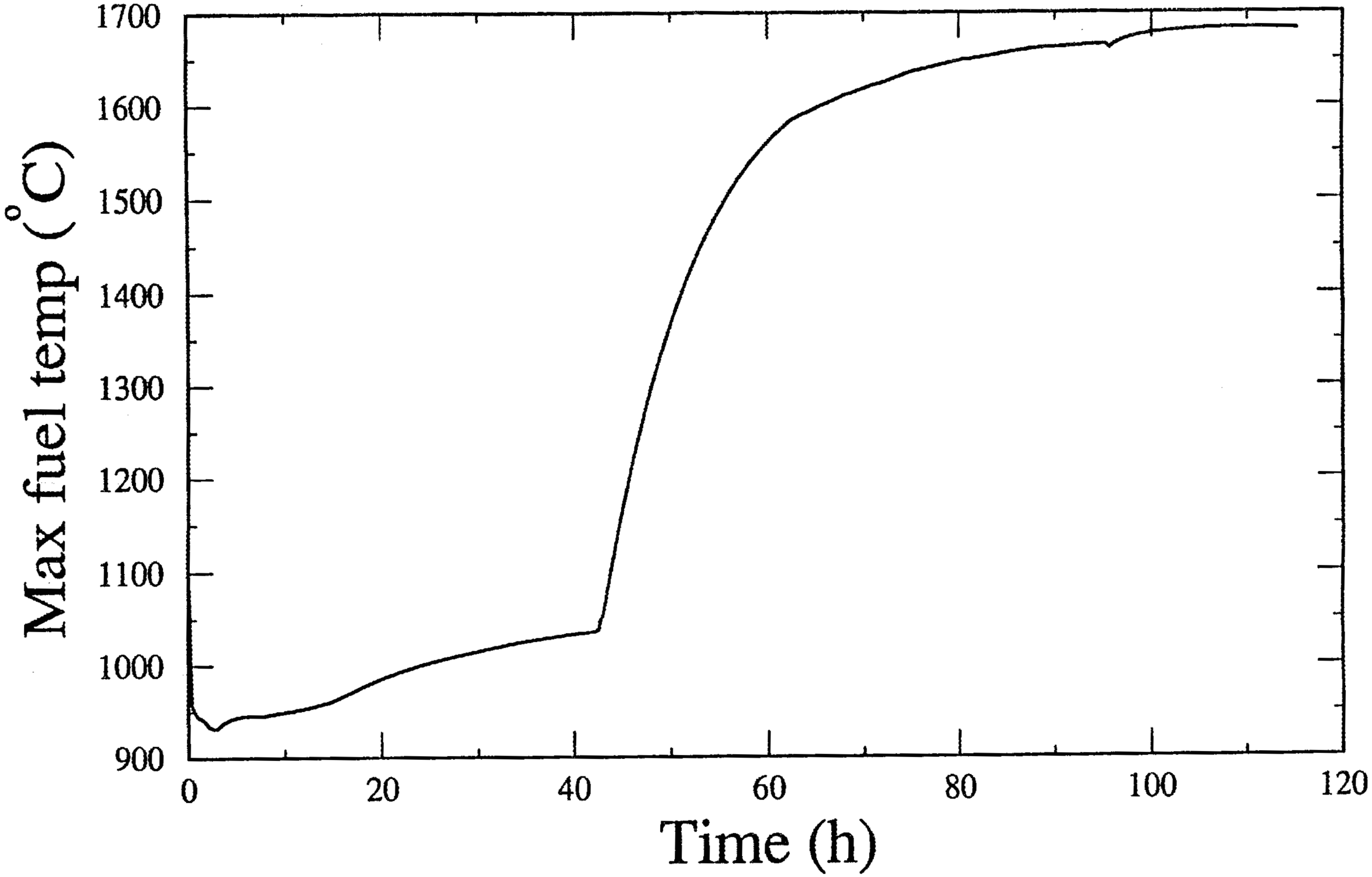

Fig. 7. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram, maximum fuel temperature vs time.

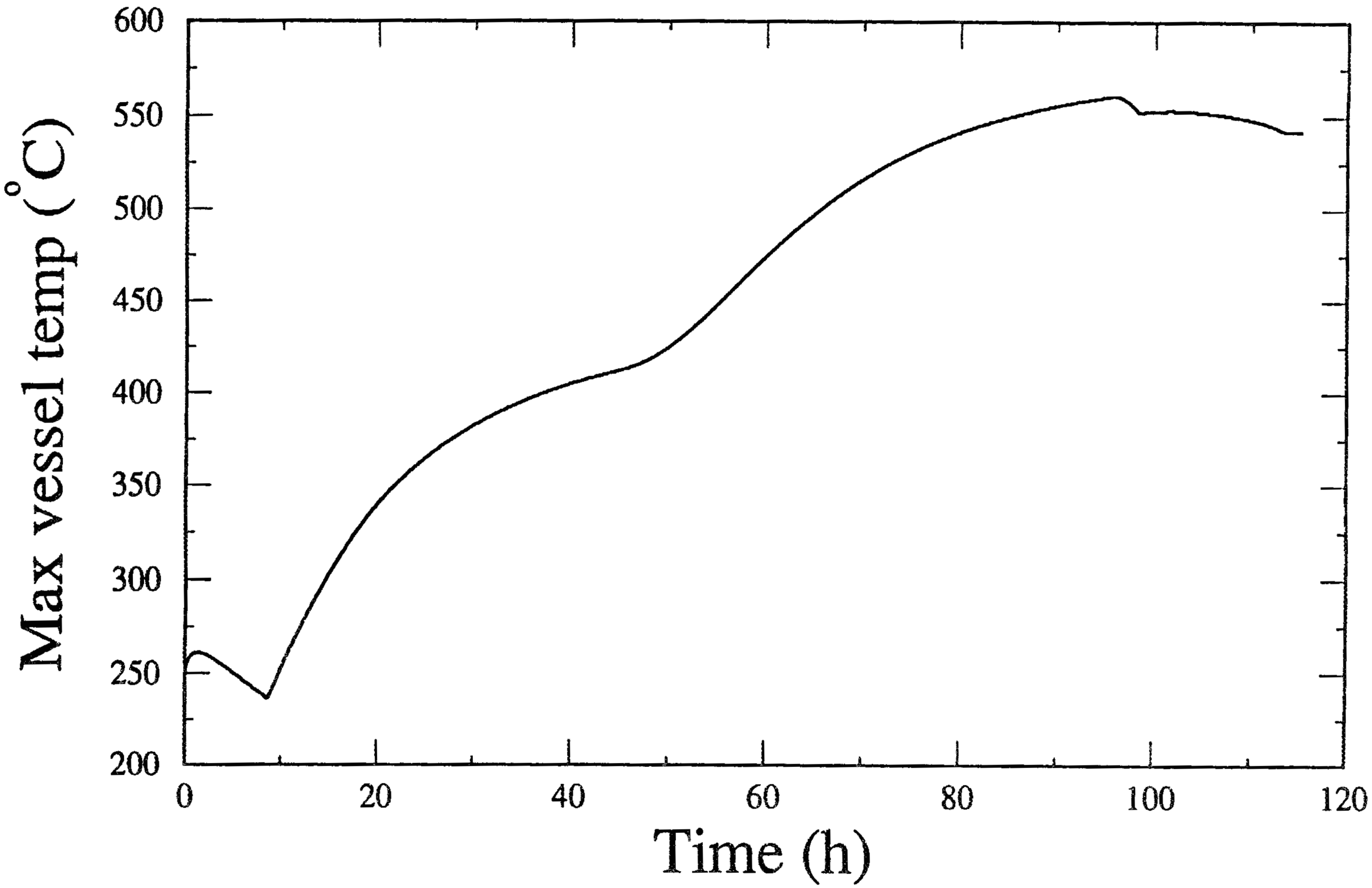

Fig. 8. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram, maximum vessel temperature vs time.

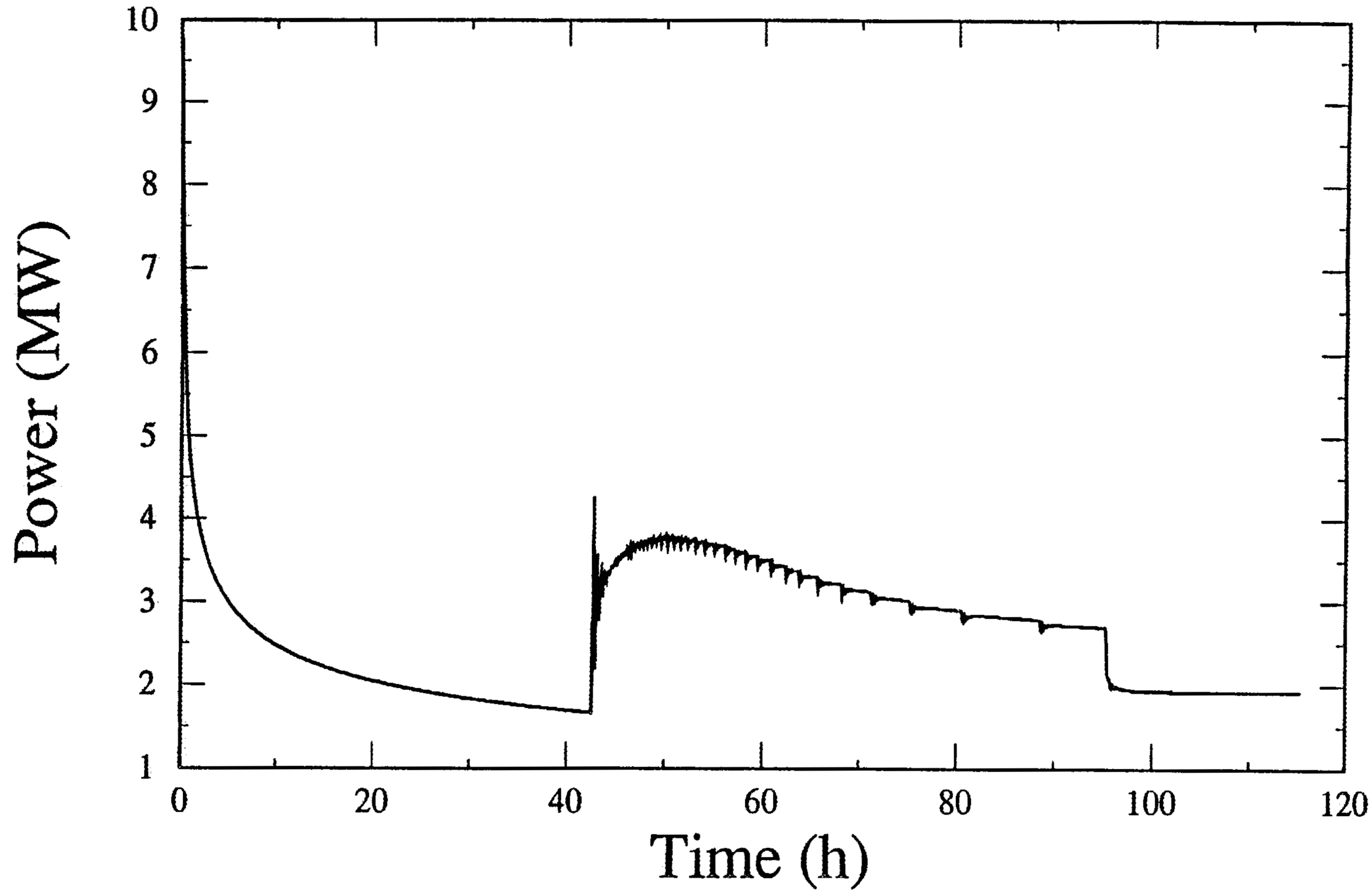

Fig. 9. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram, power vs time.

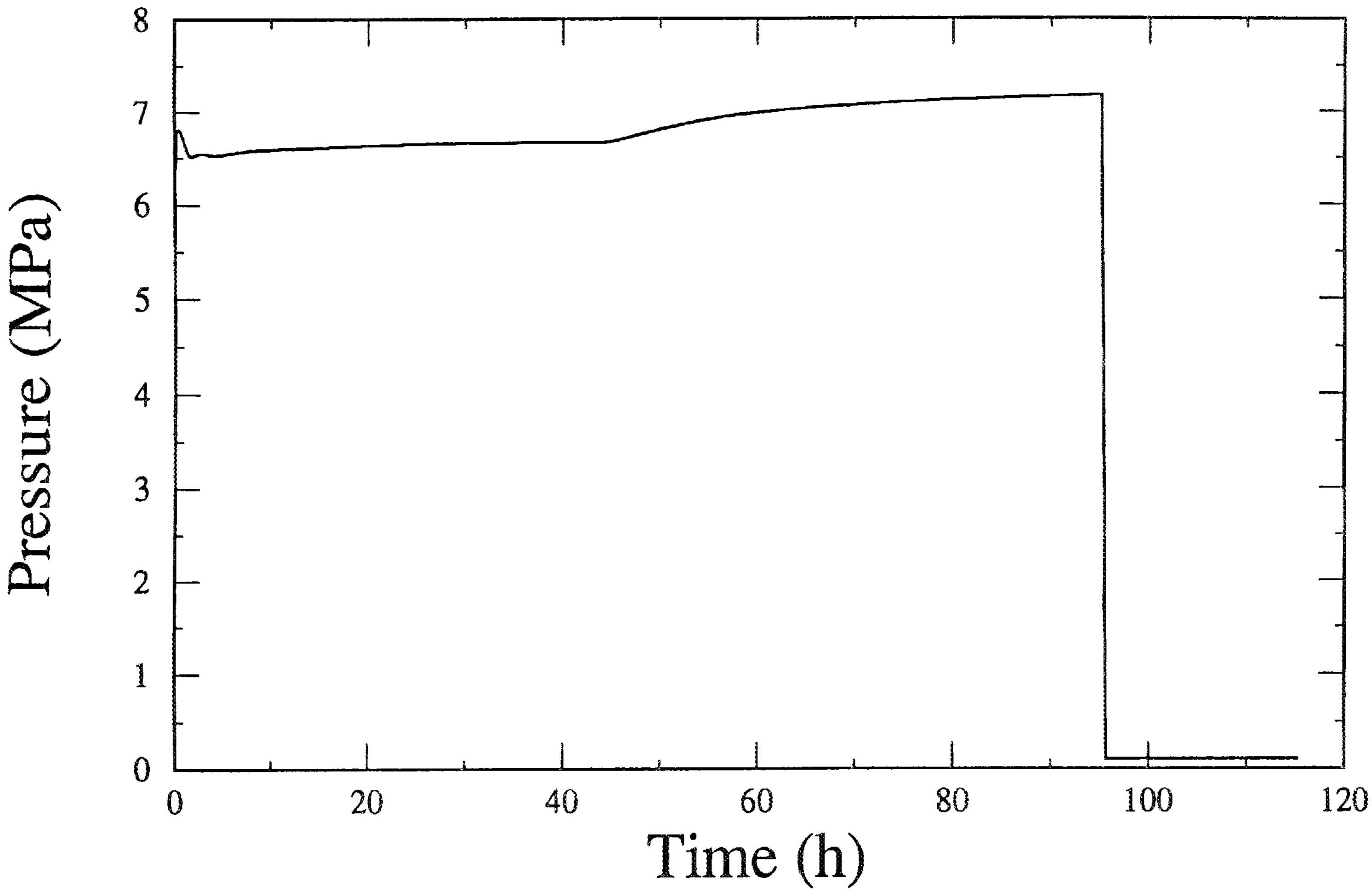

Fig. 10. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram, primary system pressure vs time.

Figures 11 through 14 show results of a simulation of the BES-1B accident, a pressurized ATWS-LOFC in which the control rods are also withdrawn, contributing an additional 3% positive reactivity. In this simulation, the primary system depressurizes because the primary pressure exceeds the relief valve trip pressure within about 15 min. Hence, this scenario is very much like the depressurized case (BES-1C). Recriticality occurs in about 33 h, compared to 44 h for the cases where no additional positive reactivity is introduced. Peak fuel temperatures exceed 1,600°C after about 40 h, but the peak vessel temperature does not exceed code limits (depressurized) until after about 2 d. If no depressurization is assumed, the peak vessel temperature exceeds code limits in about 36 h.

Operator interaction scenarios of particular interest include those where in certain later stages of ATWS-LOFCs, operator use of a degraded SCS for core cooldown could actually cause significant increases in maximum fuel temperature. This is due to the fact that the additional core cooling, which causes an increase in reactivity (and power), does not cool the hotter parts of the core in proportion to the power increase. Although the SCS cooling is effective in reducing vessel temperatures and lowering the primary system pressure, the effect on maximum fuel temperature is counterproductive.

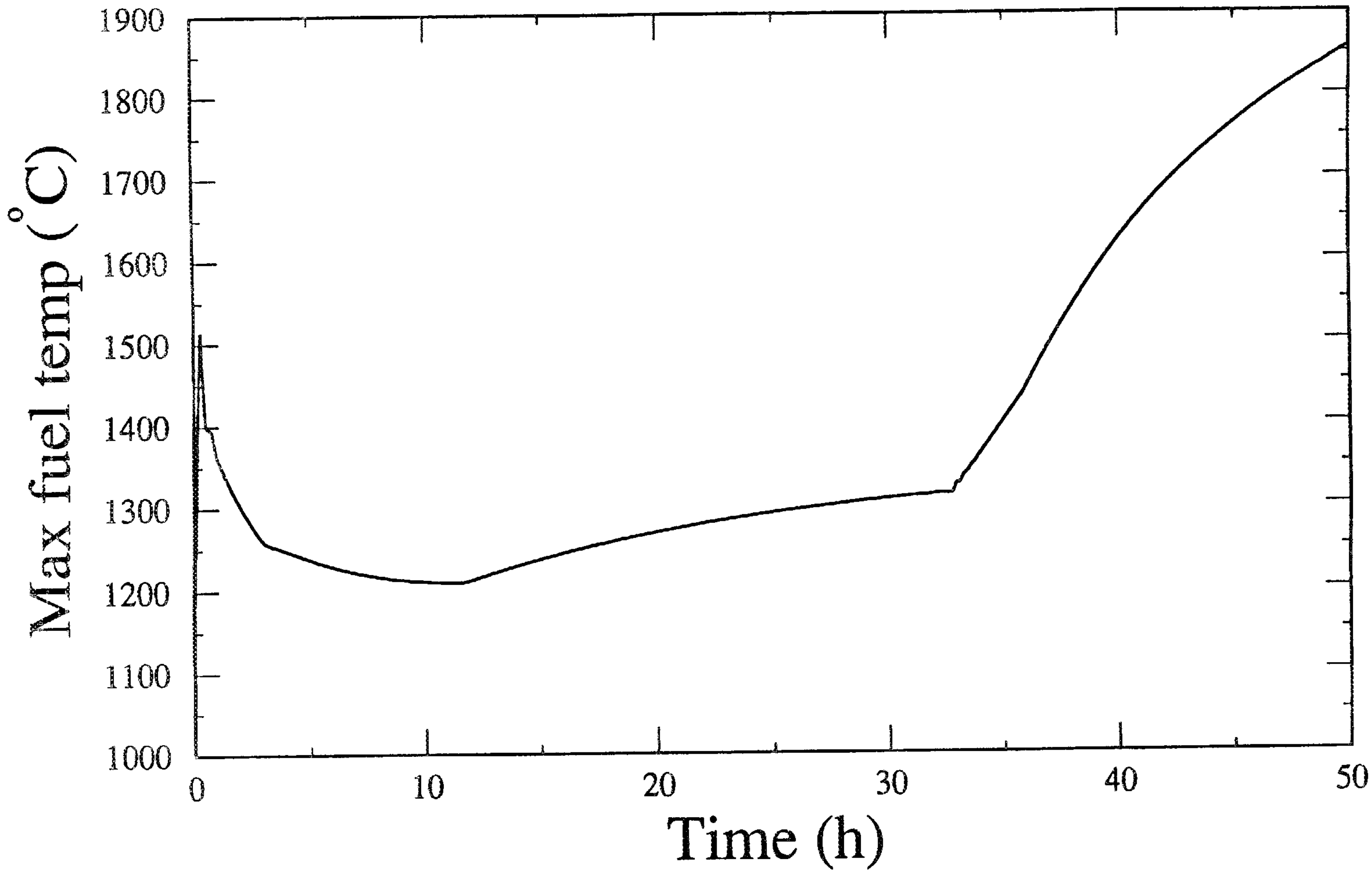

Fig. 11. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram and control rod withdrawal (3%), maximum fuel temperature vs time.

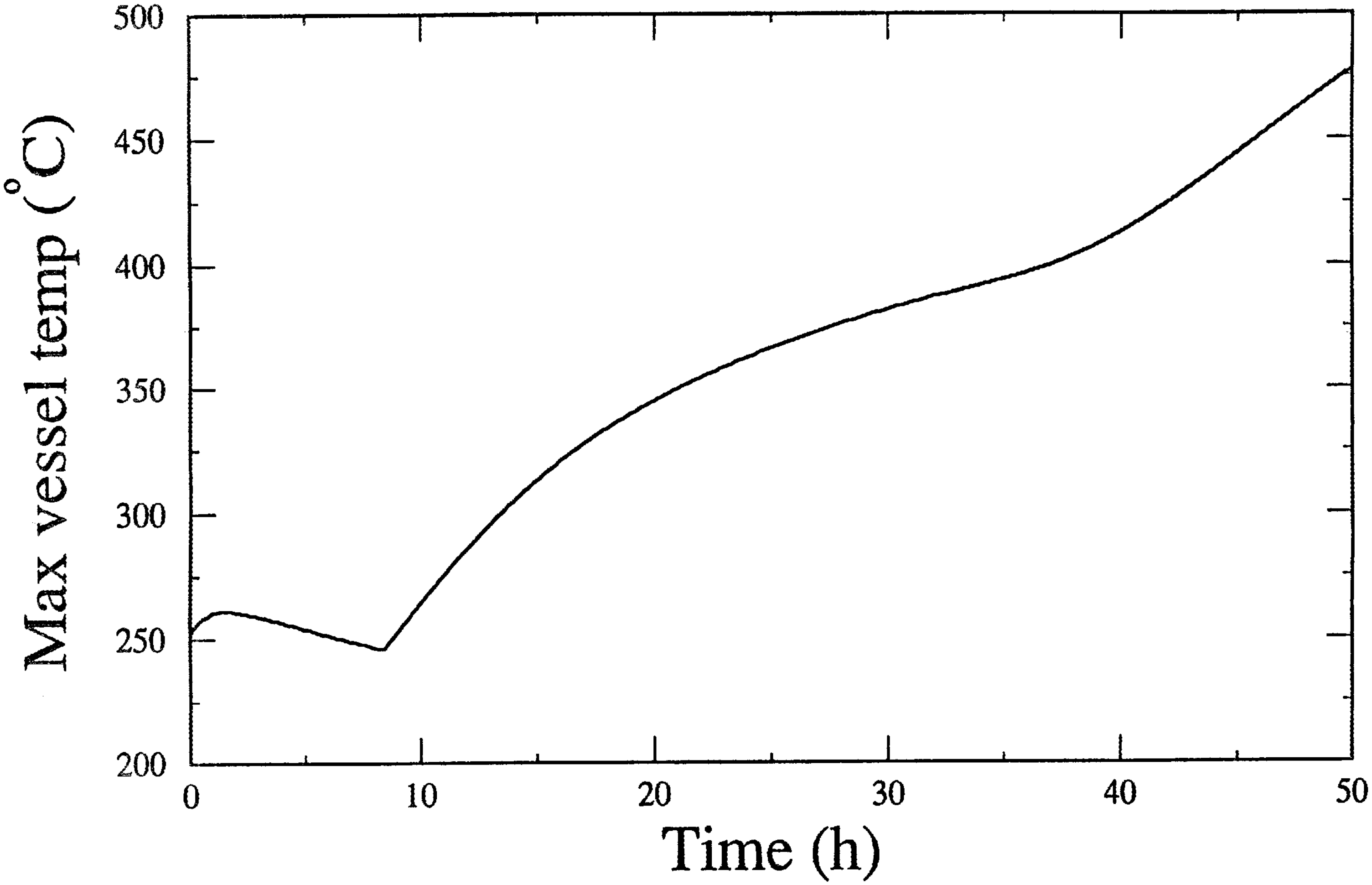

Fig. 12. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram and control rod withdrawal (3%), maximum vessel temperature vs time.

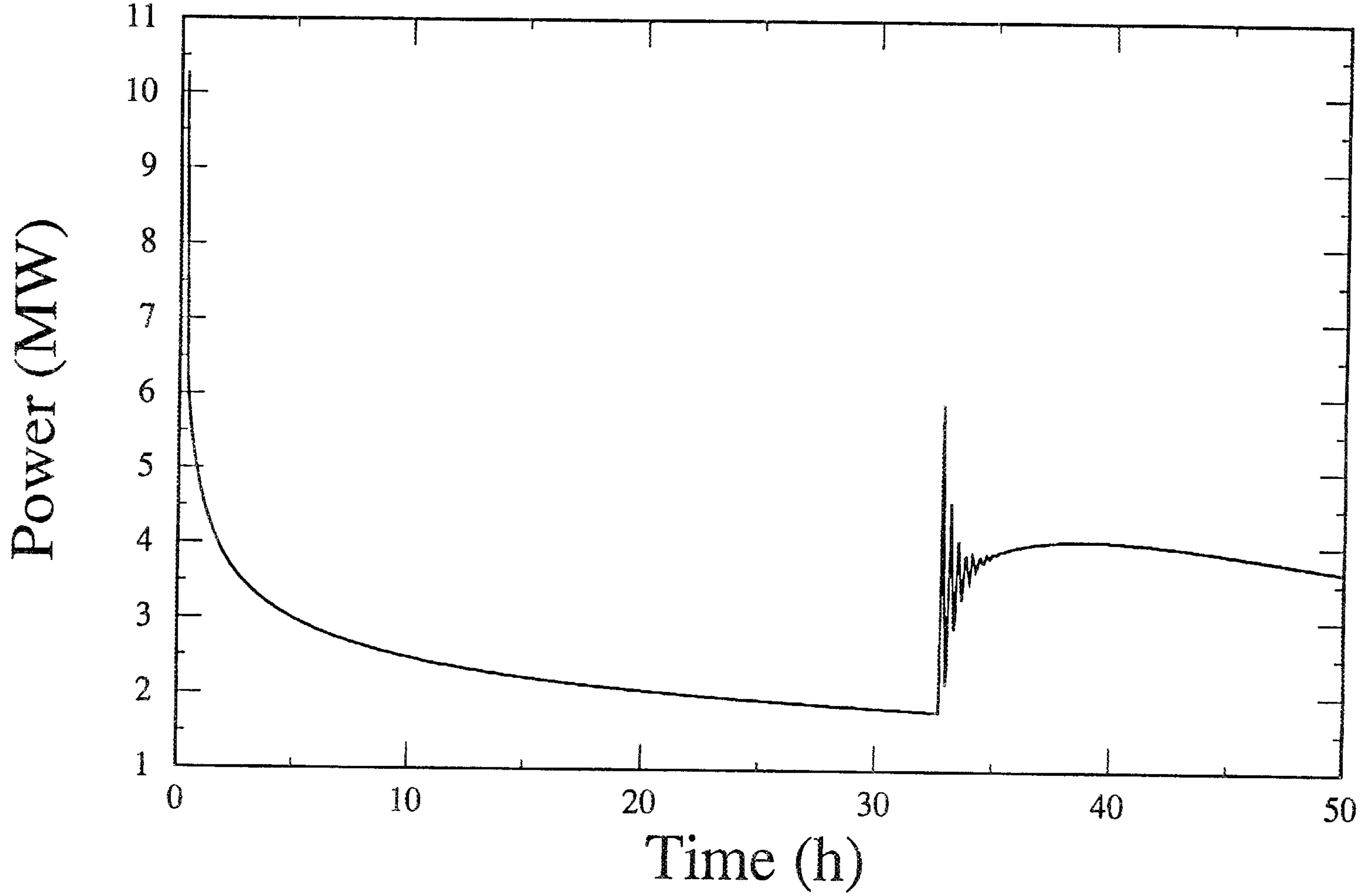

Fig. 13. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram and control rod withdrawal (3%), power vs time.

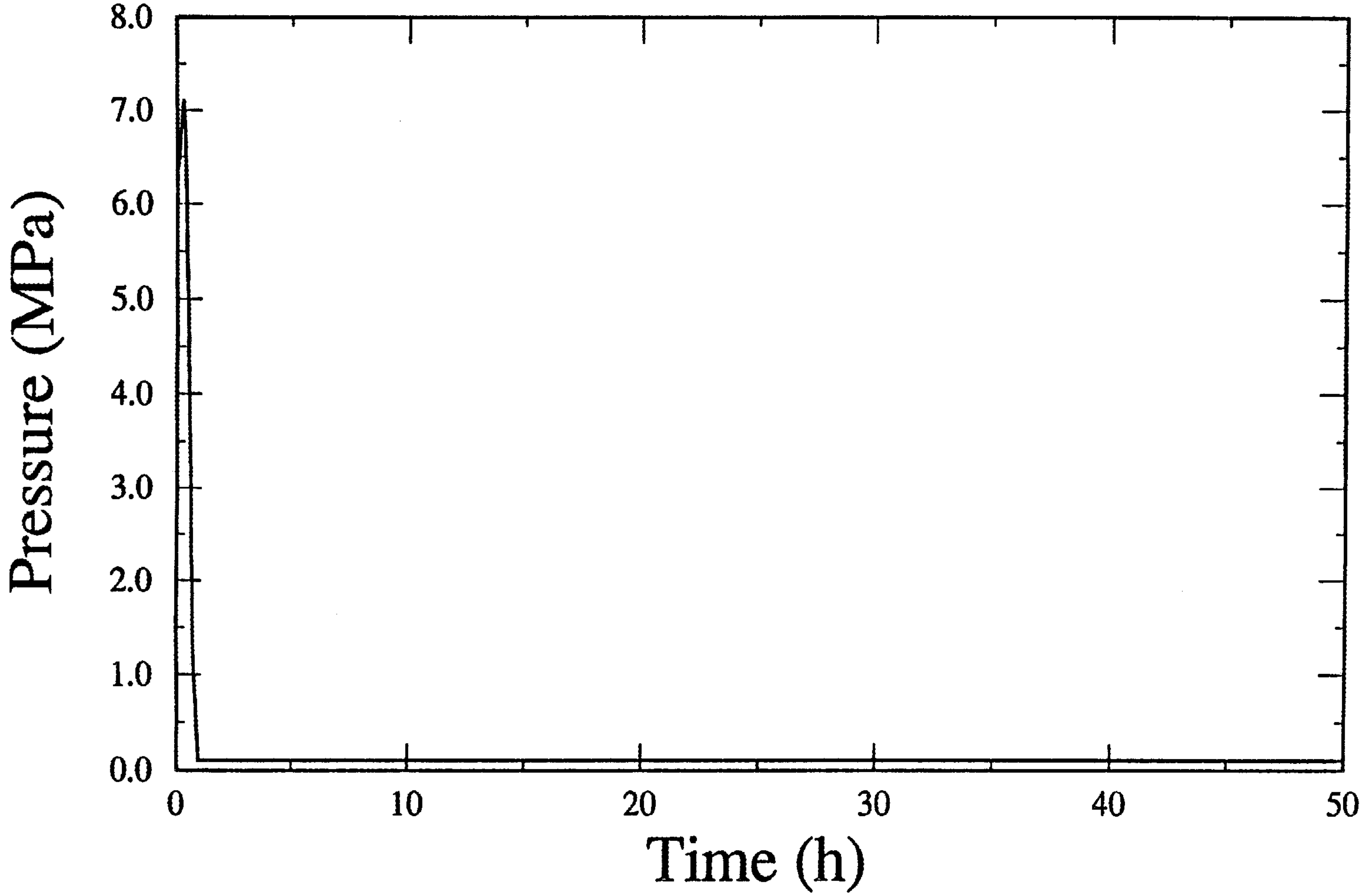

Fig. 14. Pressurized loss of forced circulation accident accompanied by anticipated transient without scram and control rod withdrawal (3%), primary system pressure vs time.

8. TRANSIENT AND ACCIDENT SCENARIO SENSITIVITY STUDIES

Many variations of transient and LOFC accident scenarios have been studied to observe sensitivities of predictions to parametric and operational assumptions. These scenarios are described in detail in the earlier report documenting MORECA.[2]

Sensitivity studies of ATWS-LOFC scenarios to date have concentrated mainly on various operator-action scenarios. Clearly, however, the peak fuel temperatures attained following recriticality are sensitive to long-term reactivity contributions from xenon and samarium and to whatever mitigation could be done via poison insertion.

9. CONCLUSIONS

The workstation version of MORECA is a useful tool for studying MHTGR transient and accident scenarios and the sensitivity of the results to model variations, operational characteristics, and operator actions.

The LOFC heatup accident analyses and sensitivity studies have shown that the current MHTGR design does not appear to be susceptible to significant fuel failure from postulated LOFC accidents, even from those of extremely low probability. Several days would elapse before worst-case ATWS-LOFC combination events lead to fuel temperatures in excess of 1,600°C, the point at which initiation of fuel failures is expected.

The Oak Ridge National Laboratory results generally corresponded well with independent calculations by DOE contractors and by Brookhaven National Laboratory.[1] It should be emphasized that these calculations include predictions of some of the most serious types of accidents that can be reasonably postulated. The fact that there is such good agreement between independent predictions is strong evidence that the analyses are relatively straightforward, therefore lending credibility to the results.

The one major area of concern for non-ATWS accidents was with possible vessel overheating, and that would not be considered an immediate safety concern unless long-term failures or partial failures of RCCS were to occur.

10. REFERENCES

1. P. M. Williams et al., *Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor*, NUREG-1338, U.S. Nuclear Regulatory Commission, 1989.

2. S. J. Ball, *MORECA: A Computer Code for Simulating Modular High-Temperature Gas-Cooled Reactor Core Heatup Accidents*, NUREG/CR-5712, ORNL/TM-11823, Oak Ridge National Laboratory, Oak Ridge, Tenn., October 1991.

3. S. J. Ball, "Dynamic Model Verification Studies for the Thermal Response of the Fort St. Vrain HTGR Core," *Proceedings of the Fourth Power Plant Dynamics, Control and Testing Symposium, March 17–19, 1980*, The University of Tennessee, Nuclear Engineering Department, Knoxville, 1980.

4. S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159, Oak Ridge National Laboratory, Oak Ridge, Tenn., April 1976.

5. S. J. Ball et al., *Summary of ORNL Work on NRC-Sponsored HTGR Safety Research, July 1974–September 1980*, NUREG/CR-2392, ORNL/TM-8073, Oak Ridge National Laboratory, Oak Ridge, Tenn., March 1982.

6. R. M. Harrington et al., "Hypothetical Accident Scenario Analyses for a 250-MW(T) Modular High Temperature Gas-Cooled Reactor," ASME Winter Annual Meeting, November 17–21, 1985, Miami, Paper 85-WA/NE-16.

7. G. Ivens and K. Krueger, "Safety-Related Experiments with the AVR-Reactor," IAEA Specialists' Meeting on Safety and Accident Analysis for Gas-Cooled Reactors, Oak Ridge, Tenn., May 13–15, 1985.

Appendix A

INCORPORATION OF ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) CAPABILITIES

A.1. INTRODUCTION

MORECA-2 code capabilities now include ATWS events. With the use of the ATWS option, the expected scram does not occur at the time of a loss of forced circulation (LOFC) but, instead, is assumed to occur at an arbitrary time later or not at all. With certain limitations, rod withdrawal accidents can also be simulated.

In the ATWS calculation, the model for fuel (as distinct from moderator) temperature is a quasi-steady state approximation valid for only slow transients characteristic of LOFC accidents. The point kinetics approximation used for the neutronics calculation is a prompt-jump, single-precursor group model. This model was tested by comparing (for transients of the appropriate rate and magnitude) results with calculations using a "full" model with prompt-neutron generation time and six delayed-neutron precursor groups. The simplified model response was judged satisfactory. Temperature-reactivity feedback from the three-dimensional modeling of fuel, moderator, and reflectors uses nuclear importance weighting using flux shapes for various times in the fuel cycle as given in reference A.1. Models for xenon and samarium poisoning are included.

A.2. POINT KINETICS APPROXIMATION

The "full" model of the basic space-independent or "point" neutron kinetics model consists of population balance equations for the neutrons and the delayed-neutron precursor groups:

$$\frac{dn}{dt} = \frac{\rho - \beta_T}{\Lambda} n + \sum_{i=1}^{6} \lambda_i C_i + s_o \,, \tag{A.1}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} n - \lambda_i C_i \,, \; i = 1, 6 \,, \tag{A.2}$$

where

n = neutron population,

ρ = reactivity = $k - 1$,

k = reactor multiplication factor = neutrons released per neutron lost,

β_T = $\Sigma\beta_i$; β_i = fractional yield of delayed-neutron precursor group i,

λ_i = decay constant of precursor group i, s^{-1},

Λ = prompt-neutron generation time, s

C_i = population of precursor group i,

s_o = neutron source generation rate, n/s.

The "prompt-jump" approximation is appropriate for cases in which the reactivity changes are relatively slow and positive reactivity transients are both slow and limited (<<\$1.00). In this approximation, it is assumed that in the neutron population equation (Eq. A.1), n responds instantaneously to ρ. Solving Eq. (A.1), for $\Lambda \frac{dn}{dt} = 0$: $n = \frac{\Lambda(\Sigma\lambda_i C_i + s_o)}{\beta_T - \rho}$. (A.3)

Reduction of the six delayed-neutron precursor equations to a single group model is done as follows:

$$\bar{\beta}_{1-6} = \sum_{i=1}^{6} \beta_i \quad (A.4)$$

$$\bar{\lambda}_{1-6} = \frac{\bar{\beta}_{1-6}}{\sum_{i=1}^{6} \beta_i / \lambda_i} . \quad (A.5)$$

The reactivity is calculated as the sum of the temperature feedback terms from the fuel, moderator, and reflectors; the contributions (if any) from control or scram rod action; and poisoning from xenon and samarium buildup following shutdown.

The temperature feedback functions are derived from the curves given in reference A.1, Fig. 4.2-7. Nuclear importance factors are used to weight the fuel and moderator temperatures according to the square of the fission peaking factors for each element. Weighting factors for the center and side reflector blocks are assumed to be 1.0, with the top and bottom reflector and core support block temperatures not considered in the calculation. Because the nodalization scheme in MORECA considers each fuel element as a point mass, no distinction is normally made between the fuel and moderator temperatures. For LOFC heatup accident calculations, where the power generated in the elements is small (at decay heat levels), fuel and moderator temperatures differ by only a few degrees, so this is a reasonably good approximation. For ATWS/LOFC transients, however, the fission heating can be such that the difference between the fuel and moderator temperatures would be significant. Because the thermal response time constants for fuel pin temperature changes with respect to the surrounding graphite moderator are relatively small (~5 s), they can be ignored for the slow transients characteristic of LOFC accidents. Hence, the difference between the fuel and moderator temperatures is calculated for each element as an algebraic (instantaneous) function of the core power level.

The reactivity effects of xenon and samarium poisoning are both significant in ATWS/LOFC accidents. The xenon and iodine balance calculations use Cleveland's model and parameters (ref. A.2), with the exception of the initial (100% power) xenon reactivity estimate. The value of 3.7% (ref. A.1, Table 4.2-11) is used here (Cleveland's report assumed 3.2%).

The time-dependent xenon reactivity change, $\Delta\rho_{Xe}(t)$, is computed by

$$\Delta\rho_{Xe}(t) = \Delta\rho_{Xe}(t=0) * \left[\frac{X(t)}{X(0)} - 1\right] . \quad (A.6)$$

The time-dependent ^{135}Xe concentration X(t) is computed from differential equations for ^{135}I and ^{135}Xe concentrations:

$$\frac{dI}{dt} = -\lambda_I I - \sigma_I \phi I + \gamma_I \Sigma_f \phi \quad (A.7)$$

and

$$\frac{dX}{dt} = -\lambda_{Xe} X - \sigma_{Xe} \phi X + \lambda_I I + \gamma_{Xe} \Sigma_f \phi \,, \tag{A.8}$$

where

I	=	^{135}I concentration,
t	=	time,
X	=	^{135}Xe concentration,
$\Delta\rho_{Xe}$	=	change in reactivity due to change in xenon concentration,
γ_I	=	fission yield for ^{135}I,
γ_{Xe}	=	fission yield for ^{135}Xe,
λ_I	=	^{135}I decay constant,
λ_{Xe}	=	^{135}Xe decay constant,
ϕ	=	total neutron flux,
Σ_f	=	fuel macroscopic one-group fission cross section,
σ_I	=	one-group microscopic absorption cross section for ^{135}I,
σ_{Xe}	=	one-group microscopic absorption cross section for ^{135}Xe.

Samarium poisoning following shutdown is approximated by using the model described by Lamarsh (ref. A.3). After shutdown, the samarium-149 builds up as the accumulated promethium-149 decays. Because the samarium is stable, its poison effects are not removed from the system until restart, when the samarium is removed via neutron capture. In the current model, the low fission power levels achieved following recriticality are neglected with respect to their samarium removal. Hence, the postshutdown reactivity due to samarium is

$$\rho = \rho_{0S}\left[1 + \frac{\phi_0}{\phi_S}\left(1 - e^{-\lambda_P t}\right)\right], \tag{A.9}$$

where ϕ_s is defined as

$$\phi_s = \frac{\lambda_P}{\overline{\sigma}_{aS}} \,, \tag{A.10}$$

and

ϕ_0	=	initial neutron flux,
λ_P	=	^{149}Pm decay constant,
ρ_{0S}	=	equilibrium samarium reactivity = 0.463%,
$\overline{\sigma}_{aS}$	=	absorption cross section for samarium.

The new subroutine in MORECA that calculates the power for ATWS cases is subroutine POWX, which is called when the ATWS flag in the input deck is set = 1. On the initial call, POWX calculates initial reactivity contributions due to the temperature feedback terms and the xenon and samarium poisoning. The rod reactivity term is calculated to give an initial value of zero for the total reactivity. Provisions are made via a data statement in POWX to make subsequent ramp changes (positive or negative) in rod reactivity. In setting these values, however, the user should recall the limitations on reactivity swings noted previously.

When the point kinetics calculations are activated, it is necessary to use much smaller computation time intervals than for cases when thermalhydraulic considerations are controlling. For the initial part of the ATWS/LOFC transient, 1-s intervals are used for the neutronics and 12 s for the thermalhydraulics (as compared to 30-s time steps for the full-power core map calculations and 5- to 10-min intervals for the routine LOFC accidents). The time step is automatically increased later in the transient when the reactor is well into the subcritical range and the fission power is small compared to the decay power, and then the neutron kinetics computations are suspended. After the xenon poisoning decays such that recriticality is approaching, the kinetics calculations are restarted and the computation time step is reduced. Optional values for control of computation time step intervals and changeover times are set via data statements in POWX.

A.3. REFERENCES

A.1. *Preliminary Safety Information Document for the Standard MHTGR*, HTGR-86-024, U.S. Department of Energy, Amendment 12, March 1992.

A.2 J. C. Cleveland, "MHTGR Short-Term Thermal Response to Flow and Reactivity Transients," *Nucl. Technol.*, **91**, 247–58 (August 1990).

A.3 J. R. Lamarsh, *Introduction to Nuclear Reactor Theory*, Addison-Wesley Publishing Co., Inc., Reading, Mass., 1966.

Appendix B

MORECA-2 PROGRAM DESCRIPTION DATABASES

In the development of evolutionary codes such as MORECA, where several versions are created for different design variations and program needs, it is important to have structured means for keeping track of program changes to minimize the chances for errors. A database system for tracking and documenting program variables, common blocks, subroutine arguments, and other attributes was developed for the original MORECA code (ref. B.1). This database is available to all code users interested in understanding more about the code details or in making modifications to the code. Through the use of the relational data, cross referencing and checking can be done very efficiently by the use of simple database commands or programs. A more detailed description of the MORECA databases (using dBase-3+ or dBase-4) is given in Appendix C of reference B.1.

As before, the MORECA-2 database set consists of a major database (OR2VAR) that contains the names and other pertinent information about each variable, such as a brief description of its function, its dimensions, its common block location, and where it is created and modified. A second, much smaller database (OR2COM) contains the common blocks with their variables and locations and is used to cross-check data in the other databases. A third database (PR2CAL) lists the main program and all the subroutines, contains a brief description of the functions each performs, and lists its common blocks and arguments and what routines it calls.

The dBase programs provided with the database files can be used to obtain selective listings to be used for cross-checks. The program OR2P (Fig. B.1), for example, can list all the variables in the input data file, which are marked in the OR2VAR database description field (des) with an "@" symbol. The printout of input variables from this exercise is shown in Fig. B.2. A complete listing of all the variables (var) is obtained by using "!" as the search character.

Another example shown is the use of the OR2P program to list the variables in a given common block (com) (Fig. B.3), which can be used in a cross-check of the OR2COM database (Fig. B.4). Other programs and example database uses and outputs are given in Appendix C of reference B.1. An updated printout of the program descriptions is given in Fig. B.5.

REFERENCES

B.1. S. J. Ball, *MORECA: A Computer Code for Simulating Modular High-Temperature Gas-Cooled Reactor Core Heatup Accidents*, NUREG/CR-5712, ORNL/TM-11823, Oak Ridge National Laboratory, Oak Ridge, Tenn., October 1991.

B.2. J. C. Conklin, *Modeling and Performance of the MHTGR Reactor Cavity Cooling System*, ORNL/TM-11451, Oak Ridge National Laboratory, Oak Ridge, Tenn., 1990.

```
* OR2P.PRG - Program to output OR2VAR files - MORECA2
USE or2var INDEX ov2
_pquality=.T.
ACCEPT " Field name to be searched (var des dim com def mod) = " TO fn
* For a complete listing, enter "!" for phrase
ACCEPT " Phrase to search for = " TO phr
?
?" VAR       DESCRIPTION               DIMENS  COM    DEF    MOD"
?
GO TOP
lctr=0
* first page line limit = 55
lctrlim=55
DO WHILE .NOT. EOF()
IF phr $ &fn .OR. phr = "!"
 ? var, des, dim, com, def, mod
 lctr=lctr+1
 IF lctr = lctrlim
  lctr=0
  * line limit after first page = 58
  lctrlim=58
  * skip 8 lines for total lines/page = 66
  ?
  ?
  ?
  ?
  ?
  ?
  ?
  ?
 ENDIF
ENDIF
SKIP
ENDDO
CLOSE DATABASES
```

Fig. B.1. OR2P program.

. do or2p
Field name to be searched (var des dim com def mod) = des
Phrase to search for = @

VAR	DESCRIPTION	DIMENS	COM	DEF	MOD	
AL	@Alpha - reference diffusivity	1		MAIN		
CTOR	@Core outlet temps - initial	90		CFLOW		
DP	@Core pressure drop	1	Arg	MAIN	CFLOW	
DT	@Computation time step (min)	1	LCSR	MAIN		
FBPLP	@Cold bypass to lower plenum	1	PASS	INIT	CFLOW	
FCBP	@Unheated bypass fraction	1	PASS	INIT	CFLOW	
FHBP	@Heated bypass fract	1	PASS	INIT	CFLOW	
FSQ	@Channel flow distrib flag	1	Arg	MAIN		
FTOT0	@Initial value of total flow	1	Arg	INIT		
IATWS	@ATWS flag	1		MAIN		
IDEP	@Depressurization flag (1=dep)	1	IDEPP	MAIN	PRESS	
JPF	@Print control flag	1	Arg	MAIN		
JSEQ	@Run control flag	1	Arg	MAIN		
NTP	@No of time steps between prnt	1		MAIN		
PIN	@Core inlet pressure	1	Arg	MAIN		
QBY	@Fract. Q=hot bypass	1	PASS	INIT		
QR	@Radial peaking factors	85	TFSTIK	MAIN	CFLOW	QSET
QRID	@Label for radial peaking fact	20		MAIN		
QZ	@Full power F/min	1	PASS	MAIN	CFLOW	
TIP	@Inlet plenum temperature	1	LCSR	INIT	MAIN	
TM	@Max time for computation(min)	1		MAIN		
TSGO	@Steam gen outlet temp (avg)	1	Arg	INIT	MAIN,	CFLOW
XP	@Fuel/reflectr node temp-saved	206,14	Arg	INIT	MAIN	

Fig. B.2 MORECA-2 input variables.

```
. do or2p
Field name to be searched (var des dim com def mod) = com
Phrase to search for = CBVES
```

VAR	DESCRIPTION	DIMENS	COM	DEF	MOD
CB	Core barrel temps	4,7	CBVES	CONVEC	
TLVSW	Lower vessel sidewall temp	1	CBVES	CONVEC	BOTTEM
TTOPV	Vessel top temp	1	CBVES	CONVEC	TOPTEM
TUVSW	Upper vessel sidewall temp	1	CBVES	CONVEC	TOPTEM
VES	Vessel temps	4,7	CBVES	CONVEC	

Fig. B.3. Listing of variables in common block CBVES.

```
Record#   COMM     CVARS
PROGS
    1     CAHEV    COMF COMT TOPC COMCO TMMAX HG2 JCAHE
    MAIN FLOW TIN PRESS OUTNOS
    2     CBVES    CB VES TTOPV TUVSW TLVSW
    OUTNOS CONVEC RCCS BOTTEM TOPTEM VESCON
    3     GBYE     FVT NGFAIL JGOOD GOODFF
    MAIN OUTNOS
    4     IDEPP    IDEP
    MAIN PRESS
    5     LCS      QLUPG TCP TGP QLLPG TUPSW TLPSW QLUPT QLUPS QLLPS
    QLLPB
    CONVEC OUTNOS TOPTEM BOTTEM
    6     LCSR     DT PINC TIP TOP
    MAIN CONVEC POW INIT OUTNOS RCCS TOPTEM BOTTEM CAHE PRESS
    POWX
    7     ORIFIC   ORIOPN DPCOM
    MAIN CFLOW CONVEC OUTNOS
    8     PASS     T QC QZ QBY FHBP FCBP FBYP FBPLP CBFLP
    MAIN CFLOW CONVEC POW INIT RCCS TOPTEM BOTTEM OUTNOS FLOW TIN
    PRESS POWX
    9     RUMOTE   REMO REMI
    MAIN OUTNOS FLOW TIN PRESS RCCS RCCSD POWX
    10    TFSTIK   QA QR
    MAIN OUTNOS POWX CFLOW INIT QSET
    11    VESRC    QVLRC QVRC QVURC QVTRC TPANEL HEAT TOUTF AFOUTE
    CONVEC OUTNOS RCCS TOPTEM BOTTEM
    12    WSXTO    X TO
    MAIN OUTNOS CONVEC BOTTEM PRESS TOPTEM POWX CFLOW
```

Fig. B.4. OR2COM file of variables in common blocks and subroutines in which they appear.

. do p2cal 09/08/92
Include summary descriptions of each routine (Y/N)? y
Output to printer (P) or screen (S)? p

Program - ALGEN
Calls =
Coms =
Args = NRR I J I1 I2 I3 I4 I5 I6 ALR ALRU ALRD SALR XKR XKA

Subroutine ALGEN generates the average of the diffusivity ratios for core nodes neighboring the selected (i,j) node.

Program - AXIK
Calls =
Coms =
Args = T I J

Function AXIK computes core node axial conductance as a function of temperature and material. A flag (KCH) can be set to choose between the latest GA MHTGR correlations or the Fort St. Vrain FSAR correlations.

Program - BOTTEM
Calls = VFRING
Coms = LCS LCSR CBVES VESRC PASS WSXTO
Args =

Subroutine BOTTEM is used to calculate heat transfer in the lower plenum region, including radiant heat transfer from the core support blocks to the floor and side walls by using ring nodes (see VFRING). Node temperature averaging to obtain an effective ring temperature for radiant heat transfer is done on the basis of its 4th power. A simple model (with fixed h) is used for convection heat loss to the side wall coverplates. Heat transfer from the floor to the bottom vessel wall is neglected. Coverplate and vessel wall temperature updates are done via Euler approximations.

Fig. B.5. PR2CAL listing.

Program - CAHE
Calls =
Coms = LCSR
Args = THI THO TCI TCO WH WC TMMAX HG2

Subroutine CAHE is used to calculate the performance of the shutdown cooling system (SCS). It makes use of the analytical steady-state solution for single-phase counterflow heat exchanger behavior given both hot and cold side inlet temperatures and flow. The variable helium and coolant water properties are accounted for. A single average tube model is used. CAHE takes advantage of the fact that the response time of the SCS heat exchanger is much shorter than that of the core it is cooling, especially during low-flow "shutdown" conditions.

- -

Program - CFLOW
Calls = CONVEC SUMW
Coms = PASS ORIFIC WSXTO TFSTIK
Args = CF FTOT PIN DP IC TREV JSEQ RE FSQ JPF

Subroutine CFLOW computes the flows in each of the fuel elements individually. The flow effective resistance for an element is computed by using a weighted average accounting for the number and differences in the coolant hole sizes. Fuel element bypass flows are also computed on the basis of input values of initial bypass flow fractions and thereafter assuming fixed orifice characteristics. Flow resistances are based on viscosity calculated at the mean channel temperature and account for laminar, turbulent, and transition flow regions. Buoyancy forces allow for flow in some elements to be reversed (upward) while other are downward, with or without forced circulation. An iterative scheme is used to determine a net plenum-to-plenum pressure difference that satisfies the net total flow (input) requirement to within specified (input) error bounds.

- -

Program - CONVEC
Calls = BOTTEM RCCS TOPTEM VESCON
Coms = PASS ORIFIC LCS LCSR CBVES VESRC WSXTO
Args = CF TREV RE STI SCFN FTOT TSGO TINP

Subroutine CONVEC computes the convection heat transfer in each of the fuel elements in the core, accounting for variations in both flow regime and direction. An average reflector (or heated bypass) flow is used. Average plenum temperatures are calculated by assuming well-mixed flow-weighted averages of all contributing inputs. Approximate heat capacities of the core support posts are included in the lower plenum mixed-mean temperature calculation. Core barrel and coverplate to vessel nodes heat transfer is also calculated. The inlet plenum inlet temperature is dependent on a computed temperature rise across the circulator and heat transfer in the (upflow) channels adjacent to the core barrel. Calls to the subroutines for reactor cavity cooling system performance (RCCS), vessel conduction heat transfer (VESCON), and upper and lower plenum heat transfer (TOPTEM and BOTTEM) are made from CONVEC.

Fig. B.5 (continued)

Program - CP
Calls =
Coms =
Args = T

Function CP calculates air specific heat as a function of temperature for the RCCS model.

Program - FLOW
Calls =
Coms = CAHEV PASS RUMOTE
Args = I

Subroutine FLOW calculates the total primary system flow. It uses a flow vs time schedule (via a data statement for X [time, min] and Y [flow, lb/min]). If the SCS is operating (dependent on the JCAHE flag schedule), then the upper limit on primary flow is adjusted, if necessary, to prevent the SCS coolant water outlet temperature from exceeding its limit [TCOLIM].

Program - GOODVT
Calls =
Coms =
Args = TEMPF AP FBP SP FLO SLO DT FVT

Subroutine GOODVT implements the Goodin model for fuel failure as a function of time and temperature for each fuel element node.

Program - INIT
Calls =
Coms = PASS LCSR TFSTIK
Args = XP TSGO FTOT0

Subroutine INIT is called in MAIN to input the bulk of the initial condition data. In recent revisions, it now reads in initial temperatures for all of the core nodes and the vessel nodes.

Fig. B.5 (continued)

Program - MAIN
Calls = ALGEN AXIK CAHE CFLOW FLOW INIT OUTNOS POW PRESS RADK SUBS TIN TPROP CONVEC QSET POWX
Coms = PASS ORIFIC CAHEV LCSR GBYE WSXTO TFSTIK IDEPP RUMOTE
Args =

MORECA's MAIN program controls the action between subroutines, does variable initialization, data inputting, and calculations. Some initialization is done by calls to other subroutines, including FLOW, TIN, CAHE, INIT, etc. MAIN contains the loop controlling the progression of the simulation time steps. It also computes the 3-D core (solid) node temperatures. The temperature-dependent conductance between blocks is obtained from calls to functions RADK and AXIK, with effective conductance between individual blocks computed in subroutine ALGEN. For each element (node), the neighboring node identifiers are obtained from subroutine SUBS. Variable node physical properties are called from subroutine TPROP. Inlet temperature, flow, pressure, and afterheat information is obtained via calls to TIN, FLOW, PRESS, and POW respectively. For ATWS runs, power is calculated from POWX. Detailed and/or summary outputs and workstation outputs are generated via calls to subroutine OUTNOS.

- -

Program - OUTNOS
Calls = GOODVT
Coms = PASS LCS LCSR CBVES VESRC CAHEV ORIFIC GBYE WSXTO TFSTIK RUMOTE
Args = ALPH QT FT CF TREV RE PIN TSGO TAVGF

Subroutine OUTNOS provides the output for a variety of options at specified intervals, including some postprocessing to obtain variables that are not needed at each computation time interval. The calls to subroutine GOODVT and the accounting needed for fuel failure calculations are also done in OUTNOS. OUTNOS provides the interface between MORECA-2 and the interactive workstation programs. Variables are passed via arrays REMI (inputs from the workstation) and REMO (outputs to the workstation).

- -

Program - POW
Calls =
Coms = PASS LCSR
Args = I

Function POW calculates the afterheat as a fraction of initial power by using either the MHTGR PSID correlation, the MHTGR "best estimate" (HTGR-86-109) or the Fort St. Vrain FSAR correlation. When the ATWS option is chosen, power is computed in subroutine POWX.

Fig. B.5 (continued)

Program - POWX
Calls =
Coms = LCSR WSXTO TFSTIK PASS RUMOTE
Args = QT INCP

Subroutine POWX calculates reactivity and power during LOFC accidents for the ATWS option. Reactivity is calculated by using nuclear importance weighting from 3-D temperatures (fuel, moderator, and inner and outer reflectors). Xenon and samarium poisoning is included. Point neutron kinetics use a prompt-jump approximation with one delayed-neutron group (verified vs 6-group for slow LOFC transients). Point kinetics are not invoked when the neutron power is small vs afterheat and the reactor is sufficiently subcritical. Because thermalhydraulic (TH) responses are much slower than neutronics, shorter time steps are used to solve for fission power. The TH time step is reduced whenever neutronics are calculated. ATWS capability is limited to slow reactivity transients characteristic of LOFC/ATWS events. The fuel (vs moderator) temperature is calculated via a quasi-steady state relationship from the total fuel element (bulk) temperature. Provisions are made via DATA statement changes to insert reactivity ramps at preset times during the run.

- -

Program - PRESS
Calls =
Coms = CAHEV LCSR PASS IDEPP RUMOTE WSXTO
Args = FT P0 IC

Function PRESS provides a simplified primary system constant-inventory pressure calculation based on a detailed averaging of gas volume temperatures in the reactor vessel but only a cursory approximation in the steam generator. Programmed depressurization can be introduced, where the pressure is ramped downward at a specified rate. Depressurizations can be initiated via the interactive workstation inputs. Full depressurization is assumed to occur if the relief valve limit is reached. If depressurization is to an intermediate pressure, pressure is computed subsequently based on constant inventory at the end point; otherwise, it stays at atmospheric. Approximations to the steam generator average gas volume temperature are computed in PRESS.

- -

Program - PRR
Calls = CP THERMK VISC
Coms =
Args = T

Function PRR computes air Prandtl number from calls to VISC, CP, and THERMK for the RCCS model.

Fig. B.5 (continued)

Program - QSET
Calls =
Coms = TFSTIK
Args = QRSAV QAP QADH IPORS

Subroutine QSET sets the QA and QR arrays (axial and radial peaking factors) to the at-power or afterheat values per flag IPORS. Radial PFs are all set to 1.0 in the afterheat mode.

- -

Program - RADK
Calls =
Coms =
Args = T I J

Function RADK computes core node radial conductance as a function of temperature and material. A flag (KCH) can be set to choose between the latest GA MHTGR correlations or the Fort St. Vrain FSAR correlations.

- -

Program - RCCS
Calls = CP PRR RCCSD RHO RK4 THERMK VISC
Coms = CBVES VESRC LCSR PASS [JCC] RUMOTE
Args = TINF

Subroutine RCCS provides the heat loss terms to the passive, air-cooled reactor cavity cooling system (RCCS) from the corresponding vessel nodes in an array QLOSS. The RCCS model is divided into 4 quadrants with 9 axial nodes per quadrant. Details are given in J. C. Conklin's RCCS report ORNL/TM-11451 (ref. B.2.).

- -

Program - RCCSD
Calls = CP PRR RHO THERMK VISC
Coms = RUMOTE [JCC]
Args = T [TP] [TPDOT]

Subroutine RCCSD provides detailed calculations of RCCS heat transfer for its calling routine, RCCS.

- -

Program - RHO
Calls =
Coms =
Args = T

Function RHO computes air density as a function of temperature (assuming atmospheric pressure) for the RCCS model.

Fig. B.5 (continued)

Program - RK4
Calls = F?
Coms =
Args = F Y T DT

Subroutine RK4 provides a 4th-order Runge-Kutta solution for the RCCS model.

- -

Program - SUBS
Calls =
Coms =
Args = I NRR I1 I2 I3 I4 I5 I6

Subroutine SUBS provides the array indices (subscripts) for neighboring nodes of the reference (i,j) node.

- -

Program - SUMW
Calls =
Coms =
Args = A B RAC PS DP WTA

Subroutine SUMW is used to sum the individual fuel element flows as computed in CFLOW for each iteration in the solution for plenum-to-plenum pressure drop.

- -

Program - THERMK
Calls =
Coms =
Args = T

Function THERMK calculates air thermal conductivity as a function of temperature for the RCCS model.

- -

Program - TIN
Calls = CAHE
Coms = CAHEV PASS RUMOTE
Args = I

Subroutine TIN is used to calculate the reactor inlet temperature. The temperature is either read from a schedule of X [time, min] and Y [temp, F] input via data statements, or if the SCS operating flag [JCAHE, generated in subroutine FLOW] is set =1, then the reactor inlet temperature (or SCS helium outlet temperature) is calculated by using subroutine CAHE. Inlet temperatures can also be input from the workstation interface when in the core map mode.

Fig. B.5 (continued)

Program - TOPTEM
Calls = VFRING
Coms = LCS LCSR PASS CBVES VESRC WSXTO
Args =

Subroutine TOPTEM is used to calculate heat transfer in the upper plenum region, including radiant heat transfer from the core plenum element blocks to the top and sidewall coverplate ring nodes (see VFRING). Node temperature averaging to obtain an effective ring temperature for radiant heat transfer is done on the basis of its 4th power. A simple model (with fixed h) is used for convection heat loss to the upper and side wall coverplates. Coverplate and vessel wall temperature updates are done via Euler approximations.

- -

Program - TPROP
Calls =
Coms =
Args = T I J ALFT QFACT XKR

Subroutine TPROP calculates the temperature-dependent diffusivity for the core nodes, and accounts for the geometry and composition differences according to node position.

- -

Program - VESCON
Calls =
Coms = CBVES
Args = QCONV

Subroutine VESCON computes the negligible conduction heat transfer between vessel nodes.

- -

Program - VFRING
Calls =
Coms =
Args = HT RVFS VF

Subroutine VFRING calculates the view factors for radiant heat transfer (1) between the upper plenum elements and the upper plenum side walls and vessel head when called by subroutine TOPTEM and (2) between the core support blocks' lower surfaces and the lower plenum side walls and floor when called by subroutine BOTTEM. Heat transfer view factors for rings of elements are used instead of individual elements.

Fig. B.5 (continued)

Program - VISC
Calls =
Coms =
Args = T

Function VISC calculates air viscosity as a function of temperature for the RCCS model.

Fig. B.5 (continued)

Appendix C

WORKSTATION DISPLAY SCREEN FOR ACCIDENT AND OTHER TRANSIENT ANALYSES

Three programs are required to run the MORECA workstation display for accidents and other transients. One program is the FORTRAN-77 simulation, **morecxw** with **mhtsmaw.dat** as input. The second is the C program **pixmawxv**, which generates the Workstation Accident Analyses Display Screen. The third is the C-language program **gui2test**, which produces the dynamic plots. All of these programs run on SUN-4, UNIX-based workstations. The Workstation Accident Analyses Display Screen program incorporates SUN's X-Window-based graphical package **XView**, and the plotting program uses the X-Window version of VI Corporation's DataViews. The simulation and the Workstation Accident Analyses Display program communicate through shared memory.

To start the program, the user types **st350** at the command prompt. This shell program creates a window for the FORTRAN simulation and starts the display program. When the user has finished running the program, the shell script also cleans up and ensures that no unnecessary processes are still running in the background (the script removes any that are running).

After the user types **st350**, a command tool window pops up at the top of the screen. In this window, the user starts the simulation by typing **morecxw < mhtsmaw.dat**. After the FORTRAN simulation starts, selected simulation parameters are output in this window.

Displayed below the command tool window is the Workstation Accident Analyses Display Screen. This display allows the user to follow the progress of the simulation (Fig. 2 in the main part of this report). On the left side of the screen is a display of the reactor cavity cooling system (RCCS) showing the RCCS heat removal rate in megawatts and the temperature of the cooling air exiting the RCCS (all temperatures in the display are in Fahrenheit). In the center of the window is a graphical display of the VESSEL showing its maximum temperature at an elevation corresponding to its location. The maximum vessel temperature encountered in the course of the simulation is also shown along with its time of occurrence.

On the right side of the display is information about the CORE. The temperature and flow of the gas entering the core are displayed at the top, and the mean temperature of the exit gas is displayed at the bottom. Inside the CORE box, the average fuel temperature, power, pressure, reactivity, and percentage of failed fuel is displayed along with the summations of all up and down flows in the fuel element cooling channels. To the right of the CORE box, the maximum temperature of the fuel is displayed according to its axial location. As with the vessel, the peak fuel temperature is also captured along with its time of occurrence.

In the lower right corner of the window is the shutdown cooling system (SCS) information. The user can see whether the SCS is on; if it is, the SCS heat removal rate (in megawatts) and the temperature of the outlet cooling water is displayed.

At the bottom of the Workstation Accident Analyses Display Screen, various interactive buttons are displayed. In addition to the five buttons affecting simulator operation, there are three buttons for controlling the SCS, one for the RCCS, and one to control depressurizations.

The buttons that affect the operation are PRINT, HOLD/RUN, PLOT, QUIT, and SPEED-%. The PRINT button prints a screendump of the entire display and writes a snapshot of the three-dimensional core map fuel temperatures to the file **cprin**. The HOLD button toggles between HOLD and RUN and is used to pause and resume the simulation respectively. The PLOT button is used to actuate the plot display screen, where selected variables can be plotted dynamically. (For more information on the dynamic plots, see the Dynamic Plots section below.) The QUIT button is used to quit the simulation, Workstation Display, and dynamic plots (if active). The last simulation control button, SPEED-%, allows the user to reduce the simulation speed. To enter a simulation speed, the user clicks on the SPEED-% button to make the input field active, enters (via keyboard) the percentage simulation speed (which is displayed in the input field to the right of the button), and then presses the RETURN key to notify the display program to read the input field.

Three buttons affect SCS parameters: ON/OFF, WC-%, and WH-%. The ON button toggles between ON and OFF when activated, turning the SCS on and off. The WC-% button works like the SPEED-% button described above and allows the user to enter the percentage of rated cooling water flow through the SCS. The WH-% button is activated similarly, where the user enters the percentage of rated primary helium coolant flow through the SCS. Rated flows are dependent on primary system pressure. Unless the SCS is ON, the WH-% and WC-% entries have no effect on the simulation. As in the SCS design, a built-in automatic control system acts to reduce the helium flow if the water coolant temperature approaches the boiling point.

The RCCS-% button also works like the SPEED-% button and enables the user to degrade the RCCS performance. If 0% is entered, a model for heat conduction to the concrete silo is invoked.

The button on the far right of the display, PSIG-%, is activated like the other button/input field combinations and enables the user to DEPRESSURIZE the reactor either partially or fully, at a prespecified ramp rate. For the case of partial depressurizations, pressure is ramped down to the selected (%) value (pressure increases are not allowed), and then the depressurization path is "sealed off" (i.e., the gas inventory is maintained at that point). Subsequent pressures are computed on the basis of that new inventory.

In any of the cases in which the user inputs numeric values, inputs that are invalid are flagged by the message "Value out of range!" which is printed below the button display area, and the invalid input is ignored.

Dynamic Plots used with Accident Analyses

The PLOT button on the Workstation Accident Analyses Display Screen activates the program **gui2test**, which initially displays a scrolling list from which the user can choose the variables to be plotted. It then displays the plots and updates them each time step. The plots rescale if values go outside the minimum or maximum range and scroll when the plots reach the end of the current time scale. This plotting package is written in the X-Windows-based DataViews software package, which allows the user to specify another workstation to display the plots, enabling the user to monitor the Workstation Accident Analyses Display Screen and the plots simultaneously. The user must be logged on to the second machine and running an X-Windows-based window manager (OpenWindows or X-Windows, for example) for the plots to be displayed.

The user has many options for setting up the plots. The program reads the file **gui.dat** to obtain initial information about the plots. The default **gui.dat** file is shown in Fig. C.1. The framelabel is the label displayed above the plots. The display is the identifier for the machine displaying the plots. It could be the same one used for the Workstation Accident Analyses Display Screen, but use of a different machine is highly recommended.

```
framelabel "MHTGR Accident Simulator"
display "icacp20:0.0"
nscroll 200
nslots 400

fields 19

"Time (min)"                    0
"Core MW"                       1
"RCCS MW"                       2
"Position of T-F Max"           3
"T-F Max"                       4
"Inlet Plenum Temp"             5

"Outlet Plenum Temp"            6
"Core Inlet Flow, #/s"          7
"Core Downflow, #/s"            8
"Core Upflow, #/s"              9

"Pressure, psia"                10
"Avg Fuel Temp"                 11
"Position of T-V Max"           12
"T-V Max"                       13

"SCS CW Outlet Temp"            14
"SCS MW"                        15
"RCCS Air Outlet Temp (F)"      16
"Reactivity, $"                 17
"Fuel Failure, %"               18
$end

range 1        0. 10.
range 2        0. 2.
range 3        0. 15.
range 4        2000. 3000.
range 5        500. 1000.
range 6        1000. 1500.
range 7        0. .1
range 8        0. 1.
range 9        0. 1.
range 10       500. 1000.
range 11       1000. 1500.
range 12       0. 15.
range 13       500. 1000.
range 14       100. 300.
range 15       0. 15.
range 16       500. 1000.
range 17       100. 300.
range 18       0. .2
```

Fig. C.1. MORECA-2 default plot setup file, gui.dat.

The nscroll field tells the program how many slots to scroll. In the default file, nscroll is set to 200, which means that the plots scroll 2000 minutes (each slot represents 10 minutes). The nslots parameter tells the program the total number of slots on each plot, where again, each slot represents 10 minutes.

The fields keyword is followed by the number of variables to be listed in the scrolling list. Immediately following the fields parameter is the list of variables. The first field is the label of the variable that will appear in the scrolling list and on the plot. The second field on the line is the number or index of the variable in the list. This number sequence must start with 0 and be sequential. The fields list must end with the $end keyword to let the program know that this is the end of the fields input. Following the fields definition in the default file are the ranges. Note that the range keyword is followed by the corresponding field number or index used in the fields definition. The third field is the minimum range for the variable, and the fourth field is the maximum range. These ranges are not required, nor are the field definitions, but they can prevent excessive rescalings when the parameter values cover wide ranges.

As noted above, when the user activates the PLOT button on the display screen, a scrolling list of variables is displayed for the user to select for plotting. An example of what the list would look like if the default **gui.dat** file is used is shown in Fig. C.2. One can see how the field definitions are used in the scrolling list. To choose variables to be plotted, the mouse is pointed to the variable name. Clicking the left mouse button will cause the variable to be highlighted. If the variable is already highlighted, clicking it again will unhighlight it. Note that three buttons are at the bottom of the list: PLOT, RESET, and CANCEL. The PLOT button activates the dynamic plots of those variables the user has highlighted. The RESET button unhighlights all highlighted variables. The CANCEL button removes the dynamic plot windows and scrolling list and returns control to the Workstation Accident Analyses Display Screen.

At the bottom of the dynamic plotting screen are three buttons: PAUSE, COPY, and RETURN. The PAUSE button toggles between PAUSE and RESUME and is used to pause and resume the simulation and plots. The Copy button is used to obtain a screendump of the plotting screen. The Return button returns to the scrolling list for either selecting variables to plot or returning to the Workstation Accident Analyses Display Screen. An example of the dynamic plotting display is shown in Fig. C.3.

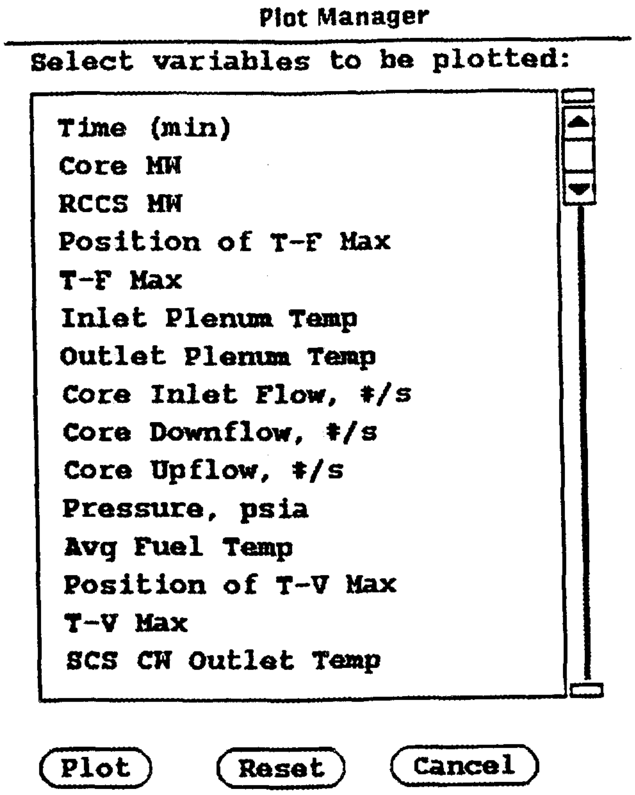

Fig. C.2. MORECA-2 sample listing of scrolling variables for plotting.

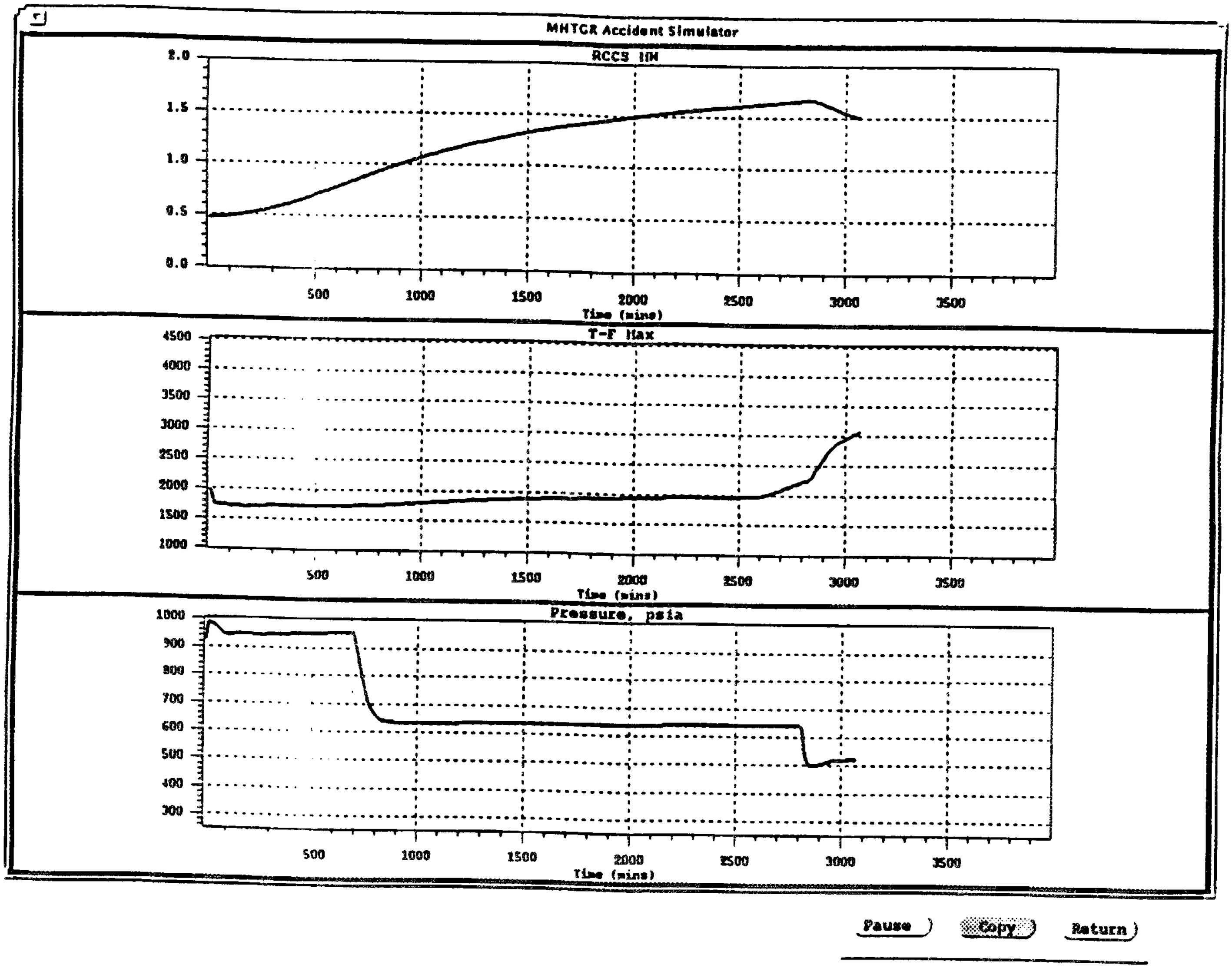

Fig. C.3. MORECA-2 sample on-line plots.

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This is a followup to an earlier report documenting the MORECA code, an interactive simulati tool for performing indepedent analyses of postulated MHTGR core transients and heatup accidents. This research was performed at Oak Ridge National Laboratory (ORNL) to assist tn Nuclear Regulatory Commission (NRC) in preliminary determinations of licensability of the U. Department of Energy (DOE) reference design of a standard modular high-temperature gas-coole reactor (MHTGR). The additional features of MORECA documented in this report are the interactive workstation capabilities and the options for studying anticipated transients without scram (ATWS) events.

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