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Magnitude and Reactivity Consequences of Moisture Ingress into the Modular High-Temperature Gas-Cooled Reactor Core

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

Inadvertent admission of moisture into the primary system of a modular high-temperature gas-cooled reactor has been identified in U.S. Department of Energy-sponsored studies as an important safety concern. The work described here develops an analytical methodology to quantify the pressure and reactivity consequences of steam-generator tube rupture and other moisture-ingress-related incidents. Important neutronic and thermohydraulic processes are coupled with reactivity feedback and safety and control system responses. The rate and magnitude of steam buildup are found to be dominated by major system features such as break size compared with safety valve capacity and reliability and less sensitive to factors such as heat transfer coefficients. The results indicate that ingress transients progress at a slower pace than previously predicted by bounding analyses, with milder power overshoots and more time for operator or automatic corrective actions.

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1. INTRODUCTION

Preliminary safety analysis of modular high-temperature gas-cooled reactors (MHTGRs) (ref. 1) has identified the inadvertent admission of secondary-side steam and/or water into the primary circuit as a safety concern because of (1) the resulting positive reactivity insertion and consequent power excursion and (2) the hydrolytic enhancement of the release of fission products from the graphite-fuel matrix, which creates the potential for the relatively prompt increased release of fission products through the safety valves if sufficient overpressure occurs.

This paper describes the development and application of a mathematical simulation to quantify the rate and magnitude of steam ingress and the resulting power excursion under postulated conditions. Included are neutronics, thermohydraulics, and control system formulations that permit the coupling of ingress with important reactivity and control feedback effects. The overall approach attempts to treat processes important to ingress in appropriate detail while suppressing phenomena judged to be of less significance, thereby yielding a fast, efficient analytical tool that avoids the long and costly setup and running times associated with conventional broad-spectrum production codes.

The precipitating event in the transients of this study is the failure of one to three steam generator tubes. This range of breaks is intended to represent credible common-cause events in which a localized thermal stress/strain reaction ruptures a tube and possibly nearest neighbors. The transient scenarios postulate that the conventional control system functions, but in some cases the safety system is partially impaired and the transients are in the anticipated transient without scram (ATWS) category. It is expected that these are among the credible ingress events that present the greatest safety challenge to the plant.

2. MODEL DESCRIPTION

The thermohydraulic processes of the injection of steam into and its subsequent transport around the primary system may be complicated by such factors as condensation on surfaces at temperatures below saturation, the entrainment and redistribution of condensate, the unrated and uncertain ability of the circulator to pump steam, the damaging effects of steam or a steam-water mixture on the safety relief valves, and the heat transfer coefficient of the helium-steam mixture. Because of the shortage of data required to treat such factors from first principles, a simplified approach is used here which models reasonably well known processes from first principles and treats others parametrically. A sensitivity analysis of parameter uncertainties may then be used to establish bounds on the effects of these uncertainties.

An overview of the model is shown in Fig. 1. Principal parameters (Table 1) are based in part on data in references 1 through 4. The primary circuit is mathematically replicated by four major regions—reactor, steam generator, hot leg, and cold leg. A solids node (fuel-graphite or metal) and a fluids node (helium or helium-steam mixture) are defined for each region. The secondary side of the steam generator is represented by a fluids node with boundary conditions for the feedwater temperature and pressure. The main steam temperature and pressure are initially set at 1000°F and 2500 psia. They subsequently are regulated by the control system, as is feedwater flow.

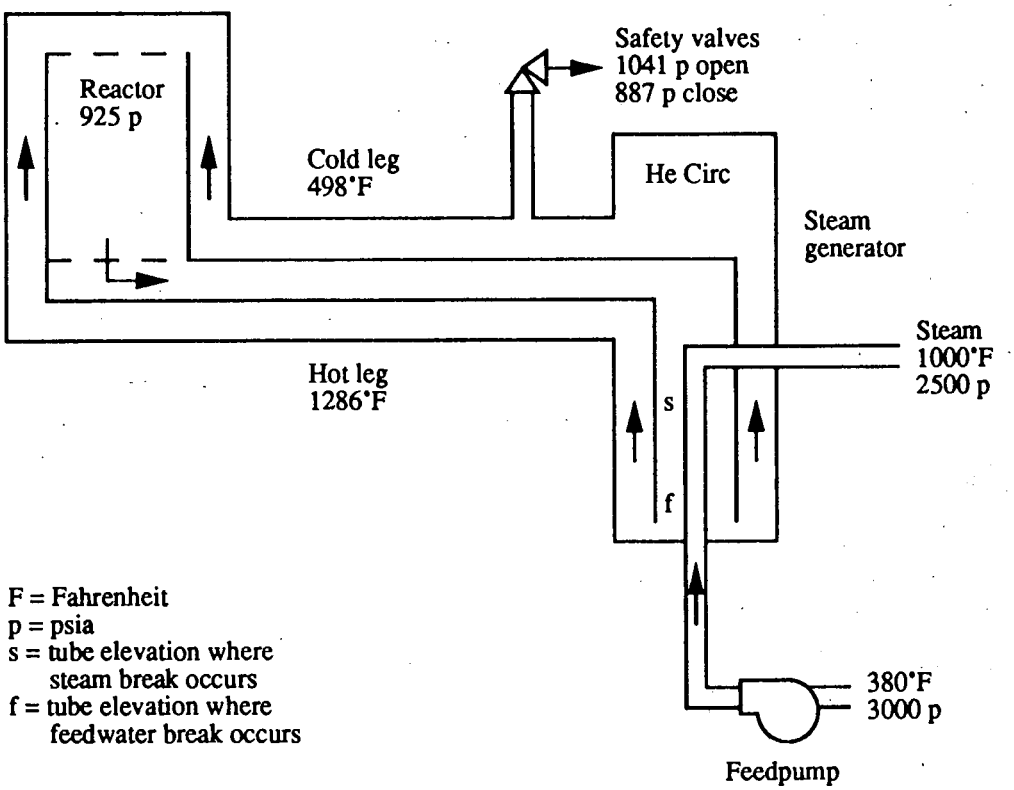
There are three principal submodels: thermohydraulics, neutronics, and controls. In the thermohydraulics submodel, the time-dependent energy and mass balance equations are solved to obtain system status (i.e., the solids or fluid temperature of each node and the partial pressures of helium and steam). In the fuel-graphite node, a neutronics submodel determines reactor power and includes reactivity feedback from the temperature and moderator density coefficients and the control rods. The controls submodel regulates rods, circulator speed, and feedwater flow to maintain the plant at setpoint. These submodels are described in the following sections.

2.1 THERMOHYDRAULICS

In each gas node the fluid(s) is assumed to be well mixed and to be at temperature T. Spatial variations of pressure around the primary loop are small compared with loop average pressures, and for this study loop average pressures are adequate to predict the magnitude of moisture ingress; therefore, the thermohydraulics may be mathematically simplified by assuming pressure and velocity are spatially uniform around the loop.

The conservation of energy equation in a gas node is

$$\frac{\partial}{\partial t}(\rho U) + \frac{1}{A}\frac{\partial}{\partial z}(\rho\ U\ v\ A) - q_S - q_{TR} + q_{SV} - q_B = 0\ , \qquad (1)$$



Controls

At power:

Rod control maintains steam temperature
Power demand sets feedwater flow and primary circulation rate

Tripped:

Feedwater flow maintains steam temperature
Minimum primary circulation 10%

Fig. 1. The modeled system.

Table 1. Parameters and values used in the modular high-temperature gas-cooled reactor model

Parameter	Value
Reactor	
Heat transfer surface	27,000 ft^2
Mass	125,000 lb
Specific heat capacity	0.35 Btu/lb-°F
Outlet helium temperature	1,287°F
Average fuel-graphite temperature	1,150°F
Fraction of total helium in core	0.05
Power	350 MWt
Prompt neutron lifetime	0.0004 s
Delayed neutron	
Fraction	0.0048
Decay constant	0.3/s
Fuel-moderator temperature coefficient	$-0.00002\ \delta\rho$/°F
Steam density coefficient	0.000036 $\delta\rho$/lb
Hot leg	
Heat transfer surface	1,000 ft^2
Mass	50,000 lb
Specific heat capacity	0.11 Btu/lb-°F
Outlet helium temperature	1,287°F
Fraction of total helium in hot leg	0.05
Steam generator	
Heat transfer surface	43,000 ft^2
Mass	180,000 lb
Specific heat capacity	0.11 Btu/lb-°F
Outlet helium temperature	498°F
Fraction of total helium in steam generator	0.18
Steam ingress velocity at rated steam conditions	2,000 ft/s
Feedwater	
Pressure	3,000 psia
Temperature	380°F
Main stream	
Pressure	2,500 psia
Temperature	1,000°F
Cold leg	
Heat transfer surface	15,000 ft^2
Mass	720,000 lb
Specific heat capacity	0.11 Btu/lb-°F
Outlet helium temperature	498°F
Fraction of total helium in cold leg	0.72

Table 1 (continued)

Parameter	Value
Helium	
Pressure	925 psia
Average velocity circulating	100 ft/s
Total volume in primary	11,653 ft^3
Critical pressure ratio (also used for steam)	0.5
Safety relief valves	
Area	0.087 ft^2
Open pressure	1,041 psia
Close pressure	887 psia
Helium velocity at rated conditions	4,445 ft/s
Control system	
Operating control rods	
Proportional gain setting	0.0003
Maximum rate of motion (% of rod length)	3%/s
Total worth	0.2 δk
Nominal fractional insertion	0.2
Main pump	
Proportional gain setting, reactor tripped	10.0
Maximum rate of speed change	3%/s
Minimum flow (% of full)	0.3%
Flow decay constant on trip	0.008/s
Booster pump pressure, assumed	500 psia
Helium circulator	
Maximum rate of speed change	3%/s
Minimum flow (% of full)	10%

where

$$\rho\, U\, v = \rho_{H2O}\, U_{H2O}\, v_{H2O} + \rho_{He}\, U_{He}\, v_{He}$$

and

A = flow cross-sectional area,
q_B = rate of energy input by break flow,
q_S = solids (metal or fuel-graphite) surface heat source/sink,
q_{SV} = rate of energy loss through safety valves,
q_{TR} = rate of entrained-moisture energy transport into node,
ρ = gas density,
U = gas internal energy,
v = circulator-induced gas flow velocity.

The terms in Eq. (1) will be discussed in the order that they appear. The first is the time rate of change of total gas internal energy. Using state functions that relate internal energy to temperature, the second term is conveniently expressed in terms of node temperature as

$$\frac{1}{V} (F_{He}\, c_{vHe} + F_{H2O}\, c_{vH2O}) \frac{\partial T}{\partial z} ,$$

where

c_v = constant-volume specific heat capacity,
F = flow rate = ρ v A,
T = node fluid temperature,
V = node volume.

The solids surface heat source/sink is

$$q_S = A_S[(h_{He} + h_{H2O})(T_S - T)] /V ,$$

where

A_S = solids heat transfer surface,
h_{He} = helium heat transfer coefficient,
h_{H2O} = effective steam heat transfer coefficient (modified by He),
T_S = solids temperature,

and where h_{H2O} is the condensation heat transfer coefficient when the surface temperature is below the steam saturation temperature.

The next term in Eq. (1), q_{TR}, is the net energy transfer from condensation in the node and from moisture entrainment originating in other nodes. Because this study treats the presence of substantial quantities of steam in the primary circuit, and because some surface temperatures may fall below the steam saturation temperature during simulated transients, the model includes a treatment of condensation and entrainment that attempts to quantify pressure and reactivity effects. Although these effects are anticipated to be second order compared with a large break flow, a bounding estimate of their importance is provided for completeness. Since its difficult if not impossible to predict the mechanistic details of entrainment and redistribution of condensate from first principles, a semiparametric approach is used. The model calculates the total condensation at a solids surface, based on solids and steam temperatures and the condensation heat transfer coefficient. It is assumed that condensate is removed from the node either by flow down surfaces or by entrainment and transport to other gas nodes. A matrix of transfer parameters specifies the redistribution of condensate to other parts of the system. Then first principles are again used to mix the transported condensate with the helium-steam in the target node or nodes. In this way, the analysis can investigate an envelope of redistribution of condensate throughout the system and bound the effects.

From conservation of energy, the condensation rate W_{CN} in a node is

$$W_{CN} = A_S h_{CN} (T - T_S) / (U_{H2O} - U_{CN}) ,$$

where

h_{CN} = condensation heat transfer coefficient,
U_{H2O} = steam internal energy,
U_{CN} = condensate internal energy.

Energy removed from the node by condensation is then $W_{CN} U_{CN}$. Energy transported from the node (say, node i) to other nodes (j) is by definition

$$f_{ij} W_{CN} U_{CN} ,$$

where the entrainment fractions f_{ij} are a stipulated distribution matrix. The source in the node from condensation and entrainment in other nodes is

$$\sum_j f_{ji}(W_{CN} U_{CN})_j .$$

So in the node, the net energy flow from condensation effects is

$$q_{TR} = [-W_{CN} U_{CN} + \sum_j f_{ji}(W_{CN} U_{CN})_j] / V .$$

The next term in Eq. (1) is energy loss through actuated safety valves

$$q_{SV} = (WV_{He} U_{He} + WV_{H2O} U_{H2O}) / V_T ,$$

where

V_T = coolant total volume,
WV = valve mass flow rate.

Because uniform pressure around the loop is assumed, gas loss is uniformly distributed over the coolant nodes. The valves have setpoints to open at 1041 psia and close at 887 psia (He plus H_2O pressure).

Energy ingress from a steam generator tube break is

$$q_B = W_B U_{H2O} / V_T ,$$

where W_B is mass flow rate through the break, and U_{H2O} is break steam internal energy. The flow rates WV_{He}, WV_{H2O}, and W_B are determined by conventional critical flow relationships. Since the break flow (sonic) velocity is near 2000 ft/s, break flow is assumed to distribute essentially instantaneously throughout the primary circuit and thereafter to be impelled by the circulator. Because of the uncertain ability of the circulator to pump

helium and steam under mixed flow conditions, the nominal flow velocities of helium and steam are specified as parameters that may be varied as postulated by the analysis.

Consistent with the assumption of spatially uniform pressure, the conservation of mass equations are written in terms of loop average density, which means density is evaluated at the loop average temperature. The equations for He and H_2O are

$$V_T \frac{d\rho_{He}}{dt} + WV_{He} = 0 , \tag{2a}$$

$$V_T \frac{d\rho_{H2O}}{dt} + WV_{H2O} - W_B + \sum_i [(1 - f)W_{CN}]_i = 0 , \tag{2b}$$

where $f_i \equiv \sum_j f_{ij}$ is the total fraction of condensate produced in node i that is entrained and redistributed to other nodes j. Thus, the last term in Eq. (2b) is the total condensate produced in all nodes that is effectively removed by puddling in hydraulically isolated regions of the circuit.

When combined with state relations for the density and internal energy of He and H_2O as functions of temperature and pressure, the coupled Eqs. (1) and (2) determine the gas temperature profile around the primary loop and the loop average gas partial pressures. Steam and water properties are based on functions reported in reference 5; helium pressure, temperature, and density are related by the perfect gas law. The helium heat transfer coefficient, thermal conductivity, viscosity, and specific heat correlations are from reference 6.

For the thermodynamics of the solids nodes that interface with the gas nodes, the temperatures of the hot- and cold-leg metal are given by the heat transfer equation

$$M_S c_{pS} \frac{dT_S}{dt} + A_S(h_{He} + h_{H2O})(T_S - T) = 0 ,$$

where

c_{pS} = metal constant-pressure specific heat,
M_S = metal mass,
S = hotleg or coldleg index,

and other symbols are as defined previously.

In the reactor, the heat transfer equation for fuel-graphite takes the form

$$M_R\, c_{pR} \frac{dT_R}{dt} + A_R(h_{He} + h_{H2O})(T_R - T) - P = 0 , \tag{3}$$

where P is the core-average fuel heat source density.

For the steam generator the heat transfer equation is

$$M_{SG}\, c_{pSG} \frac{dT_{SG}}{dt} + A_{SG}\,(h_{He} + h_{H2O})(T_{SG} - T) + A_{SG}\, h_{SE}(T_{SG} - T_{SE}) = 0 ,$$

where

h_{SE} = secondary side heat transfer coefficient,
T_{SE} = secondary side fluid mean temperature.

The second term on the left-hand side is the transfer of heat from the primary steam-helium mixture to the tube metal, and the last term is transfer from tube metal to secondary fluid.

The secondary side is simulated as a quasi–steady–state fluid node with a specified value of feedwater temperature and with the feedwater pressure and flow determined by the control system. The conservation of energy equation is

$$F_F(U_M - U_F) = A_{SG}\, h_{SE}(T_{SG} - T_{SE}) ,$$

where

F_F = feed flow,
T_F = feedwater temperature,
T_M = main steam temperature,
U_F = feedwater internal energy,
U_M = main steam internal energy.

The secondary-side temperature T_{SE} is taken to be the average of the feedwater and main steam temperatures,

$$T_{SE} = (T_F + T_M)/2 .$$

Main steam pressure is initially at rated conditions and is subsequently regulated by the control system.

The feedpump is simulated by allowing the control system to vary feedflow with a maximum rate of change of 3%/s, typical of the response of many feedpumps. Following

main pump trip, the pressure and flow rate decay exponentially with a specified time constant.

2.2 NEUTRONICS

The neutronics are treated by the lumped approximation with a prompt neutron group and one effective delayed neutron group,

$$\tau \frac{dP_P}{dt} = \frac{\delta k}{k} P_P - \beta P_P + \lambda P_D$$

$$\frac{dP_D}{dt} = \beta P_P - \lambda P_D ,$$

where

k = neutron multiplication factor,
P_D = delayed neutron power density,
P_P = prompt neutron power density,
β = delayed neutron fraction,
λ = delayed neutron decay constant,
τ = prompt neutron lifetime.

The total power density in Eq. (3) is therefore

$$P = P_P + P_D .$$

Reactivity coefficients are specified for fuel-graphite temperature and steam/water density and, together with the control rod worth, are components of the total reactivity $\delta k/k$. Nominal values of the reactivity coefficients are used. It is expected that the fuel component of the temperature coefficient will be comparatively insensitive to moisture ingress. However, the moderator component may be affected. Detailed calculations of the neutron spectral shift are needed to quantify this effect. Here the temperature coefficient is taken independent of moisture buildup.

Following reactor trip the decay power is given by (ref. 5)

$$P_{DE} = [0.11522115t^{-0.202428} - 0.003557102 \exp(4.867016 \times 10^{-10} t)] P_{P,0} ,$$

where

P_{DE} = decay heat,
$P_{P,0}$ = power at time of shutdown,
t = time after shutdown (s).

2.3 CONTROLS

The proportional control system is taken to be in turbine-following mode, and nominally the reactor control rods are manipulated to maintain the steam generator steam outlet temperature at setpoint. An ideal controller on the outlet pressure is assumed. Under reactor trip conditions, the control system modulates feedflow to maintain a constant steam generator outlet steam temperature. The control system also varies circulator speed (simulated here by flow velocity) to maintain helium flow proportional to feedwater flow down to a minimum helium flow of 10% of full flow. This limit prevents the controller from becoming unstable at low flowrates with long circulation times.

2.4 EQUATIONS SOLUTION METHODOLOGY

A fully implicit finite difference scheme is used to solve the model equations, circumventing the common stiffness problem of coupled neutronics and thermodynamics systems. A suitable timestep was found by starting with a comparatively large value and reducing it until no significant change in results was observed. With a timestep of 1 s chosen in this manner, a 1-h transient runs on a SUN4 computer in less than 1 min. Therefore, many cases may be run in a short time.

3. VALIDATION

For partial validation, the model was used to simulate two transients reported by Woodworth et al. (ref. 7). In the first case, the helium flow rate was reduced to 50% and then restored to full flow over a period of 3 min. The results are compared in Fig. 2; the model shows good agreement with reference 7.

In the second test, the power level was ramped back to 20% in 1 min. This was achieved in the simulation by reducing feedwater flow, with the reactor tracking. The simulated variables again compared well with their counterparts in reference 7.

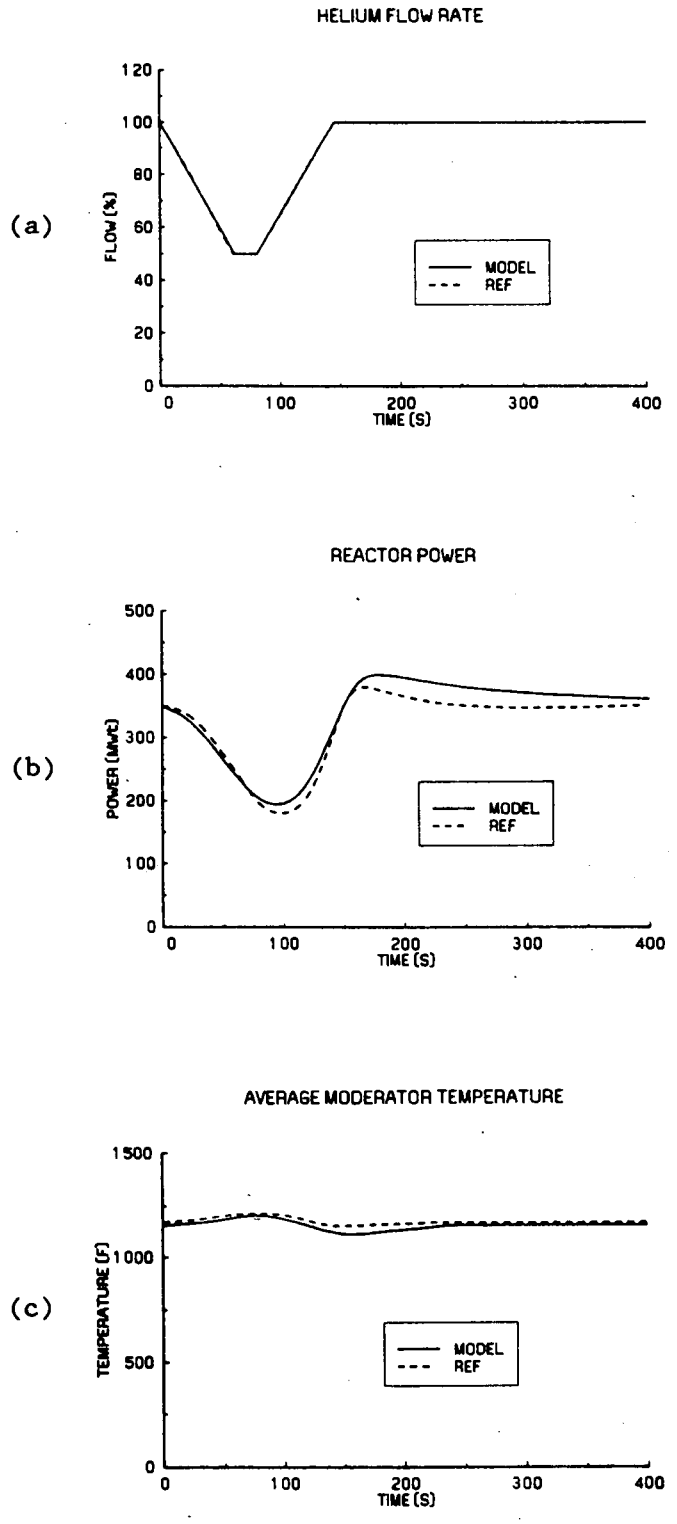


Fig. 2. Comparison of model with reference 7 transient in which helium flow is varied as shown in panel (a).

4. APPLICATION

The model was used to investigate the consequences of a range of break sizes with and without reactor trip and with and without steam generator isolation (Table 2). The model was also used to examine the sensitivity of results to key parameters including the steam-helium heat transfer coefficient, the density and temperature coefficients of

Table 2. Cases investigated with the model

Case	Type break	Break flow (lb/s)	Maximum Power [MW(t)]	Maximum F/G temp[a] (F)	Time to ρ_{H2O} = 1 lb/ft^3 (min)
1A	Steam, 1 tube, trip in 60 s	15	500	1,225	17
1B	2 tubes, trip in 60 s	30	675	1,300	7.5
1C	3 tubes, trip in 60 s	45	825	1,350	5
2	Feedwater, 1 tube, trip in 60 s	8.3	440	1,220	30
3	Steam, 1 tube, no trip	15	510	1,250	17
4	Steam, 1 tube, reactor trip in 60 s main pump trip in 75 s	15	500	1,225	∞ (0.3)[b]
5A	Steam, 1 tube, reactor trip in 60 s start SG[c] isolation at 90 s	15	500	1,225	∞ (0.15)
5B	reactor trip in 60 s start SG isolation at 360 s	15	500	1,225	∞ (0.45)
5C	reactor trip in 300 s start SG isolation at 600 s	15	500	1,250	∞ (0.7)
6A	Steam, 1 tube, trip in 60 s, H_2O heat transfer coefficient divided by 2	15	500	1,225	17
6B	Temperature coefficient of reactivity multiplied by 2	15	435	1,200	17
6C	Steam density coefficient of reactivity multiplied by 2	15	650	1,300	17
6D	Circulating steam velocity divided by 5	15	495	1,230	16
6E	Main steam pressure reduced 20%, SG isolation at 90 s	15	500	1,225	∞ (0.106)

[a]F/G temp = maximum core-average fuel/graphite temperature, starting value = 1,150°F. Starting power = 350 MW(t).

[b]Number in parentheses is maximum ρ_{H2O}, lb/ft^3.

[c]SG = steam generator.

reactivity, the extent of moisture sorption in core graphite, the capability of the circulator to pump steam, and the main steam pressure.

Although pinholes and slit-shaped fractures are the more common types of steam generator tube breaks, larger and rarer but more serious faults are considered here. The precipitating event in each case is the double-ended guillotine break of 1, 2, or 3 steam generator tubes. The nominal flow area of a tube is 0.3 in^2, so one double-ended tube break opens an area of 0.6 in^2. Critical flow conditions prevail at the break; and the vena contract, where flow is sonic, is taken to be 50% of the nominal area, making the effective flow area per broken tube 0.3 in^2. The resulting break flows are shown in Table 2.

4.1 STEAM INGRESS: ONE TO THREE STEAM GENERATOR TUBES FAIL AND REACTOR TRIPS IN 60 SECONDS

Cases 1A through 1C, with key variables shown in Figs. 3 through 5, investigate the consequences of various sizes of breaks in the upper portion of the steam generator, where the discharge from both sides of the break is steam. These cases postulate, respectively, that 1, 2, or 3 tubes break. The reactor is postulated to trip 60 s after the break occurs. This length delay was chosen to allow the power transient to exhibit its intrinsic maximum as determined by the negative temperature coefficient. The power rise is comparatively slow, taking nearly 30 s to reach a peak that increases with break size. Nonsafety-grade sensors are intended to detect moisture levels above 1200 ppm and trip the reactor; if these or the operator trip the reactor in less than the 60 s postulated here, the peak may be truncated. To show the intrinsic rate of moisture buildup, the steam generator is assumed not to be isolated during the first 15 min of the event. This would correspond to a 15-min interval without operator intervention.

Case 1A, involving the rupture of one tube, is used here as a reference case, with which others are compared. Because of the positive reactivity of steam, power initially increases 43% above nominal 350 MW(t) (Fig. 3a). The steam density in the primary (Fig. 3b) increases relatively slowly and uniformly with slight disturbances when the safety valves cycle every 3 to 4 min (Fig. 3c). A moisture equilibrium (not shown) is achieved when steam input from the break matches ejection through the safety valves and steam density asymptotically approaches 1.5 lb/ft^3. Helium is steadily lost from the primary system through the safety valves (Fig. 3d), and cooling is achieved by circulating steam. Here the circulator is assumed to impart the same velocity to steam as it does to helium, without attenuation of efficiency. The helium/steam average temperature, the main steam temperature, and the reactor core average temperature are shown in Fig. 3e. The core-average temperature peaks 75°F above the starting value of 1150°F.

Cases 1A through 1C show that for breaks ranging from one to three tubes, the moisture ingress rate varies proportionally from 15 to 45 lb/s. The peak power, which occurs approximately 30 s after the break and which could be truncated by earlier reactor trip, varies from 500 MW(t) (43% overshoot) for the one-tube case to 825 MW(t) (136% overshoot) for three tubes. The moisture level in the primary peaks near 1.5 lb/ft^3, and the length of time to reach the maximum value varies inversely with break flow. Two thirds of the buildup is nearly linear with time, and the time interval needed to reach this

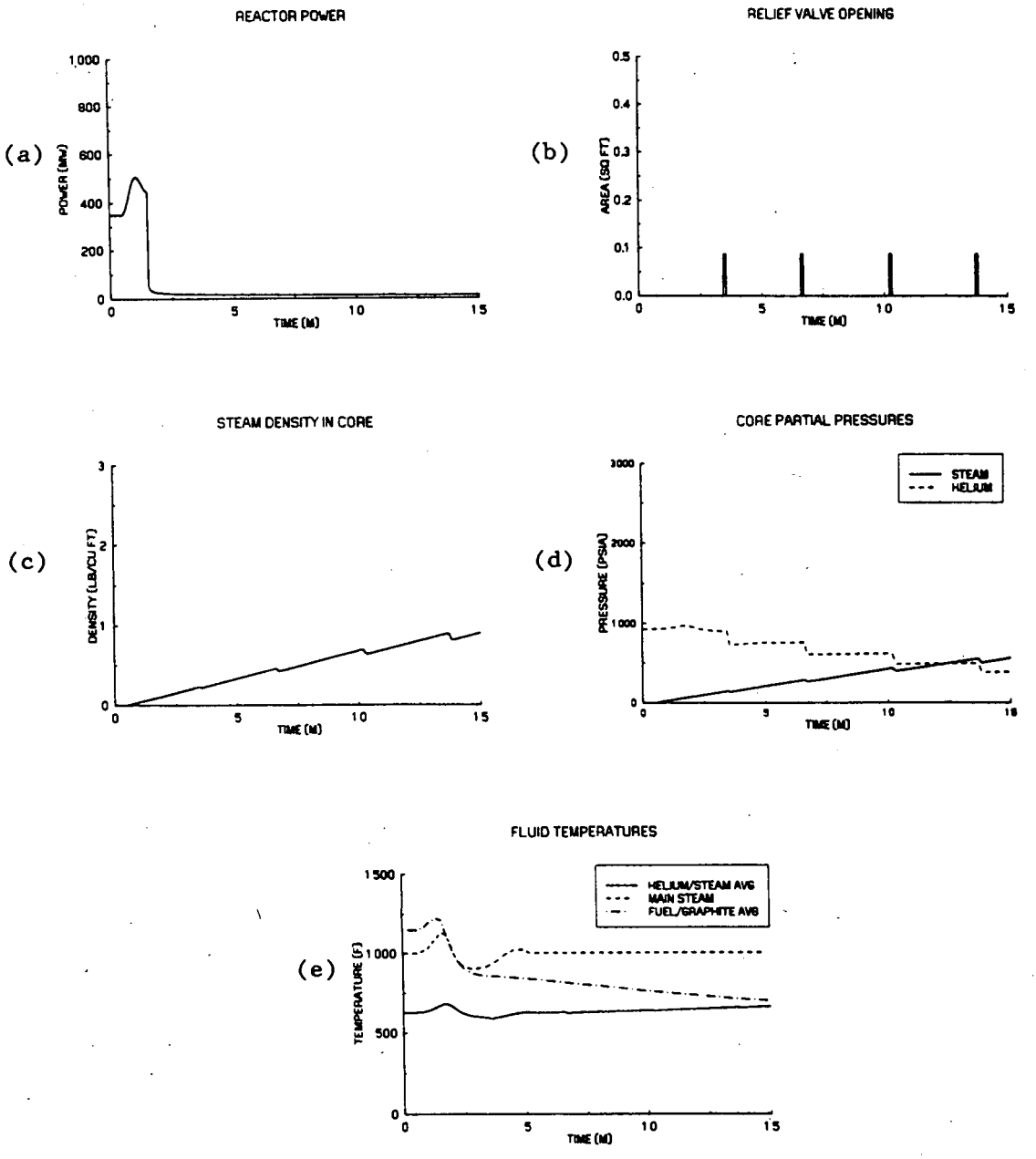


Fig. 3. Steam ingress: one steam generator tube fails and reactor trips in 60 seconds (reference case).

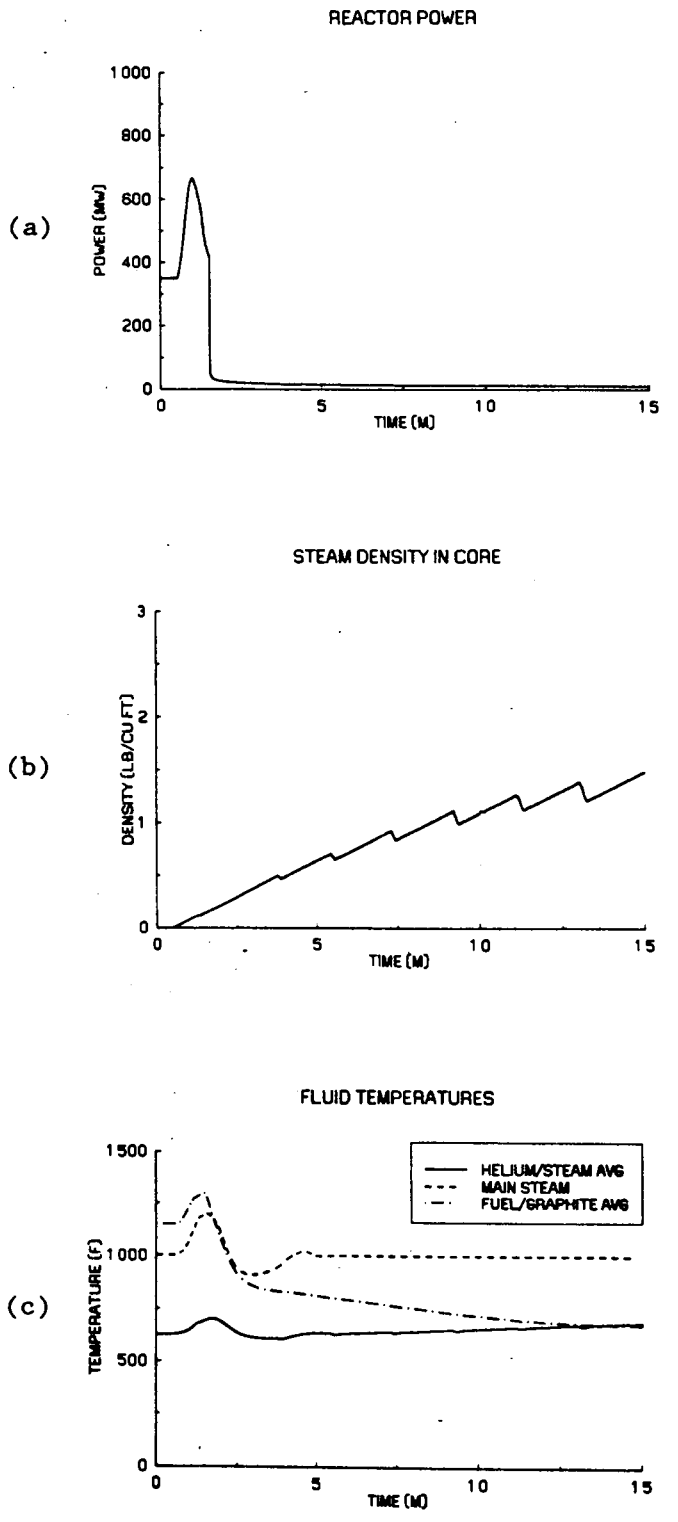


Fig. 4. Steam ingress: two steam generator tubes fail and reactor trips in 60 seconds.

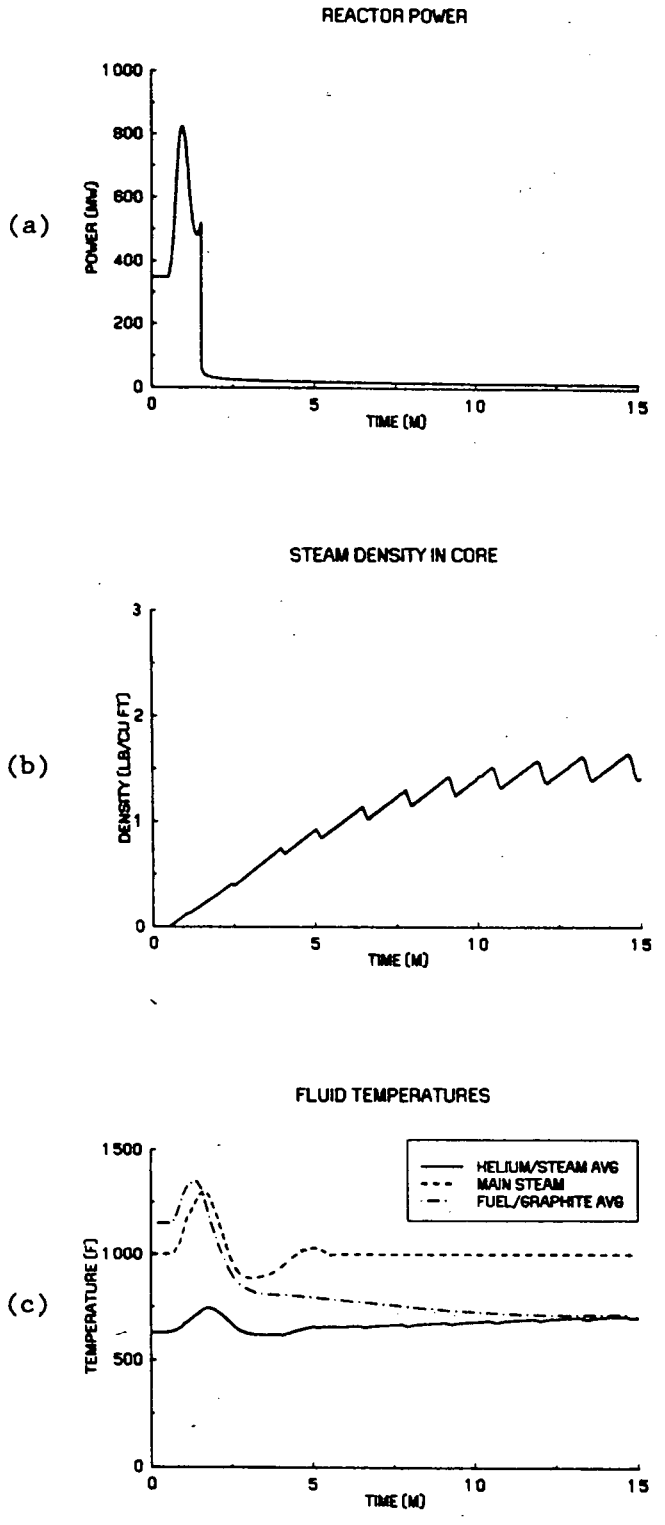


Fig. 5. Steam ingress: three steam generator tubes fail and reactor trips in 60 seconds.

level (i.e., ρ_{H2O} = 1 lb/ft^3) is a useful measure for comparing the rate of buildup in the various cases. As shown in the last column of Table 2, this time is 17 min for a break of one tube, 7.5 min for two tubes, and 5 min for three tubes.

These times can be confirmed by a simple calculation in which a tank the size of the primary coolant volume V is filled with steam through an orifice at the pressure and flow conditions of the break. Because flow F is choked and independent of primary side pressure, steam density ρ in the tank may be approximated by

$$V \frac{d\rho}{dt} = F ,$$

which integrates to

$$t = \rho V / F , \tag{4}$$

where t is the time required to accumulate density ρ. In the reference case, the flow rate is F = 15 lb/s. According to Eq. (4), the time required to accumulate a density of 1 lb/ft^3 is 13 min, consistent with the more accurate determination of 17 min in the reference case, which includes loss through the safety valves.

The sustained mass flow rate of the reference case is similar to that of design basis event (DBE) 6 in the Preliminary Safety Information Document (ref. 1). However, the resulting power transient in the reference case is seen to be milder: a 43% overshoot in 30 s versus an 86% overshoot in 10 s in DBE 6. The two power transients are compared in Fig. 6, with each starting at time = 4 s. The cause of the difference is unclear; it may be the result of differences in the rate of moisture buildup in the active core or in the value of the steam density coefficient of reactivity, or the result of other factors. The sensitivity of results to key parameters is examined further in Sect. 4.6.

4.2 FEEDWATER INGRESS: ONE STEAM GENERATOR TUBE FAILS AND REACTOR TRIPS IN 60 SECONDS

Case 2 is similar to the reference case—a double-ended break of one tube—but the location is at the bottom of the steam generator at feedwater inlet temperature and pressure conditions. Based on pressure differentials across the tube before and after the break, flow from the feedwater side of the break is taken to be twice the nominal feedwater flow rate (per tube). It is assumed that the inventory of feedwater in the tube above the break quickly discharges through the break, and that the sustained ingress from that side is steam backflow from boiling in other tubes. The total sustained flow from the break is 8.3 lb/s, consisting of 0.8 lb/s from the feedwater side and the rest from the steam side. The peak power before reactor trip is 440 MW(t), and the time to reach a core steam density of 1 lb/ft^3 is 30 min.

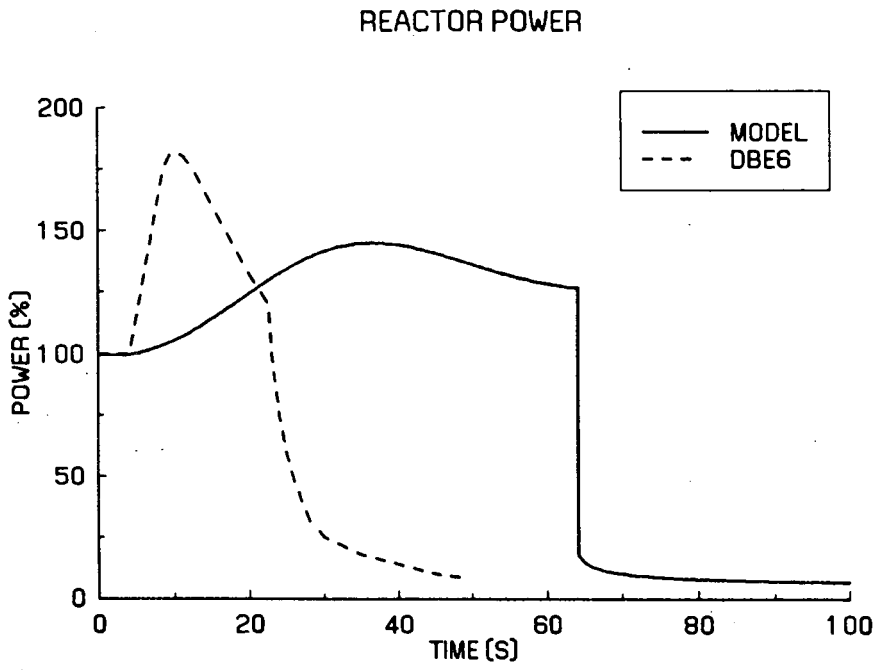


Fig. 6. Comparison of reference case (one-tube-failure) power transient with design basis event (DBE) 6.

4.3 STEAM INGRESS: ONE STEAM GENERATOR TUBE FAILS WITHOUT REACTOR TRIP

The ATWS event of case 3 is like the reference case except that the reactor does not trip (the results are shown in Fig. 7). After power peaks at 510 MW(t), the temperature coefficient turns the power down and the power then fluctuates with the cycling safety valves. With the controller gain used here, the power oscillation shows a gradual divergence during the first 15 min. A comparison of the key variables of this case with the reference case shows the principal effect of reactor trip is to truncate the power excursion. The level of moisture ingress changes little (Fig. 7b) because ingress is determined primarily by input pressure on the secondary side of the break and by the rate of loss through the safety valves.

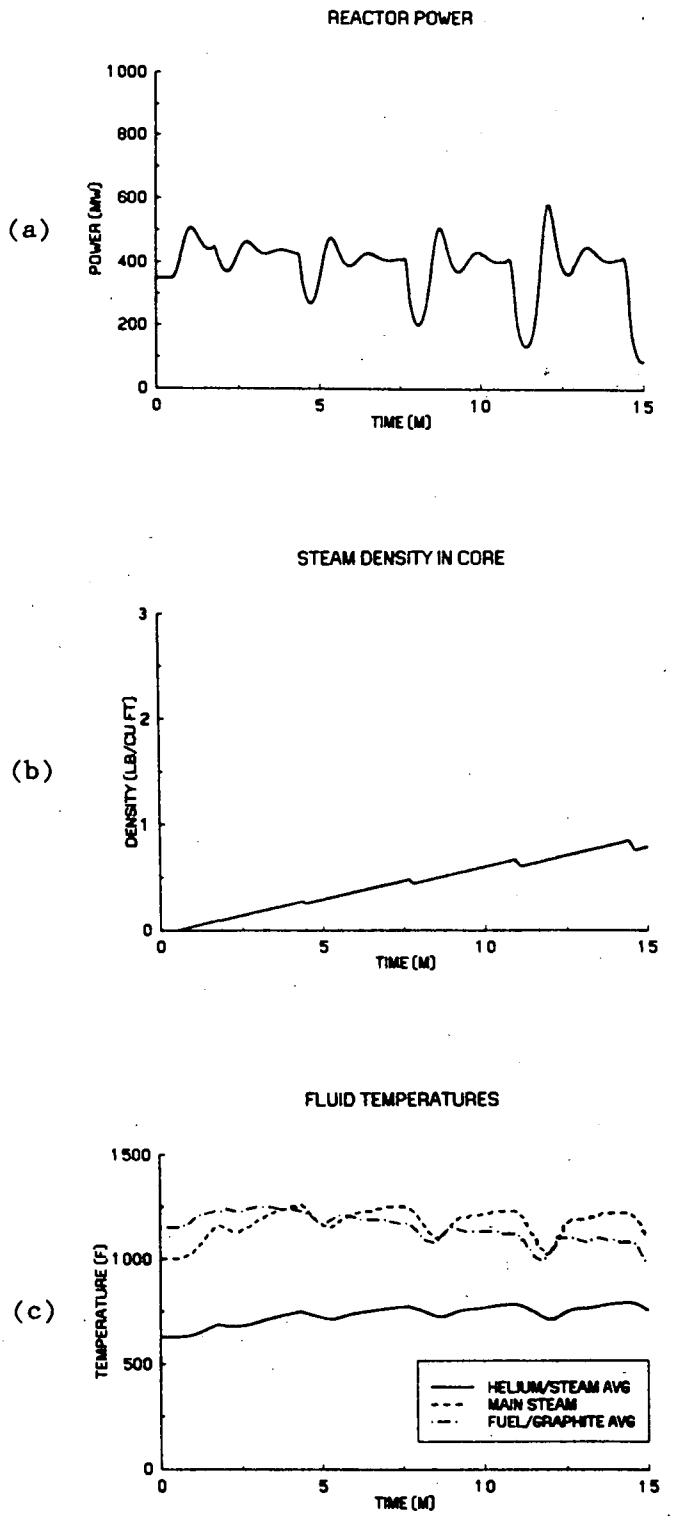


Fig. 7. Steam ingress: one steam generator tube fails and reactor does not trip.

4.4 STEAM INGRESS: ONE STEAM GENERATOR TUBE FAILS AND THE REACTOR AND PUMPS TRIP

In case 4, reactor trip 60 s after the break is followed by pump trip at 75 s after the break. The maximum moisture density in the core is 0.3 lb/ft^3, only 20% of that of the reference case. Earlier pump trip would reduce this value further.

Collectively, cases 1 through 4 suggest that for breaks of up to three tubes, the power excursion is relatively modest and the rate of ingress is slow enough to allow the plant operator to take corrective action (steam generator isolation and/or pump trip) if the normal safety functions do not automatically occur.

4.5 STEAM INGRESS: ONE STEAM GENERATOR TUBE FAILS, THE REACTOR TRIPS, AND STEAM GENERATOR ISOLATION OCCURS

In case 5, the reactor trips and then steam generator isolation occurs. During the comparatively short duration of these transients, the auxiliary heat removal system is not implemented upon steam generator isolation, and the predicted temperature overshoots are therefore conservative.

In case 5A, the reactor trips at 60 s (Fig. 8a), and steam generator isolation is postulated to begin 30 s later and require an additional 30 s for completion (loss of heat removal and throttling of the secondary pressure that drives break flow). The driving pressure at the break is shown with core pressures in Fig 8c. As summarized in Table 2, the peak power and maximum fuel/graphite average temperature remain unchanged from the reference case because these occur during the time prior to steam generator isolation. The principal effects of isolation are to (1) limit the buildup of moisture in the core to ~0.15 lb/ft^3, which is ~10% of the buildup when isolation does not occur (Fig. 8b), and (2) avoid actuation of the safety relief valves.

Case 5B postulates that steam generator isolation is, for whatever reason, delayed by 5 min after reactor trip at 60 s. Again, there is no change in peak power or core average temperature. The principal effects are (1) the steam density in the core increases to 0.45 lb/ft^3, and (2) the safety valves actuate twice during the first 7 min of the transient, at which time enough helium has been lost to compensate for the steam pressure and no further safety valve action occurs.

Case 5C postulates that reactor trip is, for whatever reason, delayed by 5 min and then isolation of the steam generator is delayed by another 5 min. The temperature coefficient of reactivity still limits the maximum overshoot to 500 MW(t), but above-normal power levels persist longer and the maximum average core temperature rises to 1250°F. The safety valves actuate three times during the first 10 min, and the maximum moisture level is 0.7 lb/ft^3, ~50% of the maximum without steam generator isolation.

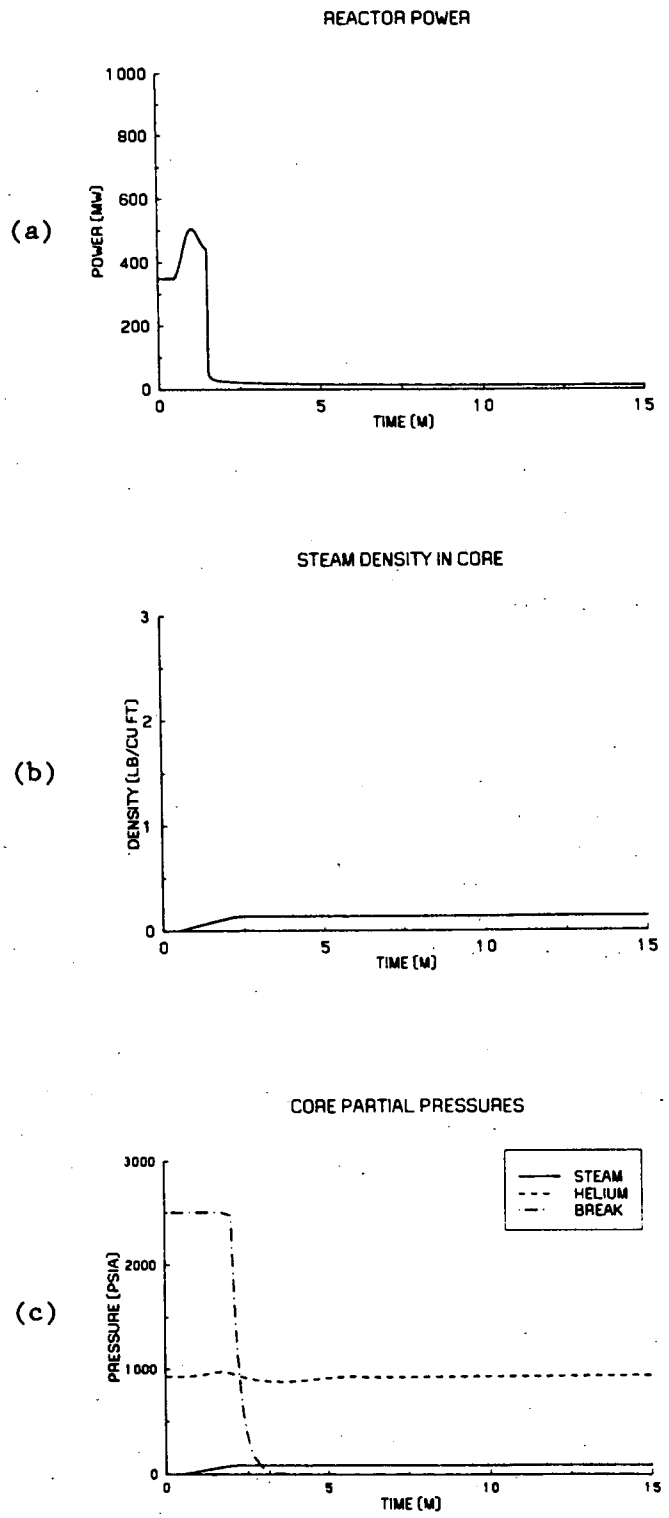


Fig. 8. Steam ingress: one steam generator tube fails, reactor trips in 60 seconds, and steam generator isolation starts at 90 seconds.

4.6 SENSITIVITY ANALYSIS

Cases 6A through 6D in Table 2 are a sensitivity study of important parameters that are traditionally subject to uncertainty: the steam-helium heat transfer coefficient; the temperature and density coefficients of reactivity; the quantity of steam sorbed in the porous graphite moderator; and the ability of the circulator to pump steam, which, in the transients studied here, provides cooling when coolant is lost through the safety valves. The purpose of the analysis is to determine what, if any, effect parameter uncertainty has on key results such as power excursion and magnitude of moisture ingress.

Heat transfer coefficient. While it is well known that the heat transfer coefficient of steam is reduced when it is mixed with helium, data describing the magnitude of the reduction in an MHTGR environment are scarce. The model includes a steam coefficient that is basically the Dittus-Boelter relationship for turbulent flow, modified by a factor that is an exponential function of the weight fraction of steam in the helium-steam mixture. The shape of the modifier is based upon data reported for mixtures of steam with several organic gases (ref. 8). At 20% weight fraction steam, for example, the steam heat transfer coefficient is reduced to one-third the value for pure steam.

To examine the effects of uncertainty in this prescription, reference case 1A was repeated with the steam heat transfer coefficient reduced by a factor of two. As shown in Table 2, case 6A, the power peak, the maximum core-average temperature, and the steam-ingress rate and magnitude are nearly unchanged. The low sensitivity of moisture ingress to uncertainty in the steam heat transfer coefficient is, as noted earlier, a consequence of ingress being dominated by the secondary-side pressure and the primary pressure as modulated by the safety valves.

Reactivity coefficients. In case 6B the (negative) temperature coefficient of reactivity was increased in magnitude by a factor of two. Power peak and temperature rise were reduced significantly. The power overshoot was 24% versus 43% for the reference case, and the temperature peak was 1200°F versus 1225°F for the reference case. In case 6C the (positive) steam density coefficient of reactivity was increased by a factor of two. Power overshoot and temperature rise increased significantly. The power overshoot was 86% and the temperature peak was 1300°F. Again, in both of these cases moisture ingress was little changed. The time to reach a moisture density of 1 lb/ft^3 remained at 17 s.

Sorption. The core reactivity will be affected not only by steam in the coolant channels but also by any steam that is sorbed in the porous graphite moderator. Nuclear-grade graphite typically has a pore volume of about 20%, and in the MHTGR this volume approximately equals the normal core coolant volume. Diffusion filling of this space with steam therefore would nearly double the steam density coefficient. Thus case 6C, in which the density coefficient was doubled, may also be interpreted as an indication of the maximum effects of absorbed steam.

Adsorption is estimated to make only a small contribution to total sorption. Using the BET (refs. 9 and 10) surface of graphite as a measure of the total amount of moisture that can be adsorbed on active lattice sites, adsorption would contribute a maximum of

only 7×10^{-5} lb H_2O per pound of graphite, that is, about 1% of the amount in filled pores.

Circulating steam velocity. The ability of the circulator to pump steam is uncertain. In the reference case, it is assumed that the velocity of steam is the same as that of the helium. To test the sensitivity of results to this assumption, the circulator was postulated to pump steam at only 20% of the velocity of helium (case 6D). Although the lower heat-transfer coefficient resulting from the lower velocity produced some longer term variations in state variables (e.g., more slowly declining core temperatures), power overshoot and buildup of steam density during the important first minutes of the transient were not greatly affected.

Condensate entrainment. If steam should condense in any part of the primary circuit, complex hydraulics would make the fate of the condensate uncertain. Some of it might drain to isolated parts of the loop, while portions might be entrained, distributed to hotter areas, and turned to steam again. In the model, the redistribution is treated by a matrix of fractions that specify the amount of condensate that is transported to any other node. However, for the cases in Table 2, loop temperatures did not fall below the steam saturation temperature, and condensate production was not observed. Therefore the model's entrainment algorithm was not exploited in these transients.

Secondary-side pressure. In all scenarios discussed thus far, main steam pressure was the current design value of 2500 lb/ft^2. However, some design reviews have considered lowering the main steam pressure to improve the quality of steam entering the turbine generator. Lower pressure would also reduce the rate of moisture ingress into the primary during a break. In the final sensitivity study, the main steam pressure was reduced by 25% to 2000 lb/ft^2. Although the rate of ingress would be reduced in all cases, in longer term transients (greater than ~30 min) the maximum steam buildup in the primary would be unchanged because it depends upon the settings of the safety relief valves. It is principally the shorter transients that would experience lower moisture buildup. For example, in case 5A where the reactor tripped at 60 s and isolation of the steam generator began at 90 s, with nominal 2500 lb/ft^2 main steam pressure the maximum steam buildup was 0.15 lb/ft^3; whereas, with a main steam pressure of 2000 lb/ft^2 the maximum buildup is reduced by 30% to 0.106 lb/ft^3.

5. SUMMARY

The MHTGR moisture-ingress model described here includes important neutronic, thermohydraulic, and control system processes such that in a moisture ingress event, ingress is coupled with reactivity feedback and controls response. Some of the more serious moisture ingress scenarios are studied; the nominal controls are assumed to operate, but in certain ATWS-type events the safety system is partially defeated.

The studies suggest that for the range of breaks postulated (one to three steam generator tubes), the rate of ingress and accumulation of moisture in the primary is a comparatively slow process, requiring at least 5 min to approach peak values of around 1.5 lb/ft^3. This result is encouraging in that there appears to be time for appropriate operator action, such as system trip and isolation of the steam generator, in case automatic trips fail.

The comparatively slow buildup of moisture in the core leads to mild reactivity insertion rates with the result that power is limited by the fuel temperature rise and reactivity feedback. In previous scoping studies, evidently faster moisture buildups lead to much larger and faster power excursions. Here, ruptures of one to three steam generator tubes produced overshoots of 43 to 136% in 30 s.

The analyses indicate that the rate and magnitude of moisture buildup in the reactor core are dominated by major system features such as the break size compared with safety valve capacity and reliability. They are less sensitive to factors such as heat transfer and reactivity coefficients. Power and temperature overshoots are strongly dependent on the density and temperature coefficients of reactivity and comparatively insensitive to the other parameters.

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11. ABSTRACT (200 words or less)

Inadvertent admission of moisture into the primary system of a Modular High-Temperature Gas-Cooled Reactor (MHTGR) has been identified in U.S. Department of Energy sponsored studies as an important safety concern. The work described here develops an analytical methodology to quantify the pressure and reactivity consequences of steam-generator tube rupture and other moisture-ingress-related incidents. Important neutronic and thermohydraulic processes are coupled with reactivity feedback and safety and control system responses. Rate and magnitude of steam buildup are found to be dominated by major system features such as break size in comparison with safety valve capacity and reliability, while being less sensitive to factors such as heat transfer coefficients. The results indicate that ingress transients progress at a slower pace than previously predicted by bounding analyses, with milder power overshoots and more time for operator or automatic corrective actions.

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