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FISSION PRODUCT TRANSPORT ANALYSIS IN A LOSS OF DECAY HEAT REMOVAL ACCIDENT AT BROWNS FERRY*

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DISCLAIMER

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†Engineering Technology Division.

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1. INTRODUCTION AND SUMMARY

This paper summarizes an analysis of the movement of noble gases, iodine, and cesium fission products within the Mark-I containment BWR reactor system represented by Browns Ferry Unit 1 during a postulated accident sequence initiated by a loss of decay heat removal (DHR) capability following a scram. The complete report is provided in ref. 1, which is based on an earlier analysis of the event sequence provided in ref. 2. The event analysis showed that this accident could be brought under control by various means, but the sequence with no operator action ultimately leads to containment (drywell) failure followed by loss of water from the reactor vessel, core degradation due to overheating, and reactor vessel failure with attendant movement of core debris onto the drywell floor.

The analysis of fission product transport presented in this paper is based on the no-operator-action sequence and provides an estimate of fission product inventories as a function of time within 17 control volumes outside the core, with the atmosphere considered as the final control volume in the transport sequence. As in the case of accident sequences previously studied, we find small barrier for noble gas ejection to air, these gases being effectively purged from the drywell and reactor building by steam and concrete degradation gases. However, significant decay of krypton isotopes occurs during the long delay times involved in this sequence. In contrast, large degrees of holdup for iodine and cesium are projected due to the chemical reactivity of these elements. Only about 1.1×10^{-4}% of the initial iodine activity and 9.3×10^{-6}% of the cesium activity† are predicted to be released to the atmosphere. This is equivalent to about 4.3×10^{-4}% and 2.9×10^{-4}% of the current activities of iodine and cesium, respectively, at time equal to 3100 min following scram. Principal barriers for release are deposition on reactor vessel and containment walls. A significant amount of iodine is captured in the water pool formed in the reactor building basement after actuation of the fire protection system.

A listing of studies in this series performed for the Containment Research Branch is provided in Table 1.

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*Computer Sciences Division.
**Engineering Technology Division.
†Counting fission product nuclides of half-lives 30 min and longer.

Table 1. Studies of accident consequences in this series

Postulated accident	Fission products	Status
Complete station blackout	NG's,[a] I	Complete
Small break LOCA outside of containment	NG's, I, Cs	Complete
Loss of DHR	NG's, I, Cs	Complete
ATWS[b]	NG's, I, Cs, Te	(In process)
Reports		
NUREG/CR-3617		
-2973		
-2182		
-2672		

[a] Noble gases.
[b] Anticipated transient without scram.

2. FISSION PRODUCT TRANSPORT CALCULATION FEATURES

Some features of the fission product transport calculation are listed in Table 2, with more complete details provided in ref. 1 and other reports listed in Table 1. The initial step in the transport calculation involves estimation of the rates of release from overheated fuel elements by (1) estimation of the time of initial cladding failure and (2) determination of the fractional release rates using basically methods described in NUREG-0772.[3] Such calculations are performed in each of 100 core control volumes using temperature histories provided by MARCH and nuclide inventories provided by ORIGEN. Vaporized fission products are convected to the downstream control volume at rates determined from core gas velocities that depend on the predicted boiloff rate.

Provision is made to allow for chemical change along the transport pathway, although many aspects of this area are incompletely understood. The following chemical change models have been included in the loss of DHR sequence analysis: (1) equilibration of species in the gaseous system Cs/I/H/O in the reactor vessel, (2) chemisorption of I-vapors with aerosols and structural solids, (3) organic iodide formation in the containment vessel and reactor building, and (4) solubility of I-species in water. At present, the area of chemical change along the transport pathway is undergoing major improvements (see Sect. 5).

A further characteristic of the estimate is that the transport models are applied to individual isotopes of the fission products rather than to the element as a whole. The advantage of this procedure is that the radioactivity level is computed directly rather than inferred from the total mass

Table 2. Transport calculation features

- Release from fuel model
- Well-mixed control volumes
- Convection between control volumes
- Chemical change along pathway
- Simple I/water solubility estimate
- Fission product vapor/aerosol interaction
- Organic iodide production rate model
- Vapor deposition by condensation and chemisorption
- Transport of individual nuclides

Krypton	Stable nuclides plus
	^{85m}Kr
	^{85}Kr
	^{87}Kr
	^{88}Kr
Iodine (+Br)	Stable nuclides plus
	^{130}I
	^{131}I
	^{132}I
	^{133}I
	^{134}I
	^{135}I
Xenon	Stable nuclides plus
	^{133}Xe
	^{135}Xe
Cesium (+Rb)	Stable nuclides plus
	^{134}Cs
	^{136}Cs
	^{137}Cs
	^{138}Cs

- Calculates radioactivity transport directly
- Automatically takes precursor effects into account

transport, and radioactive precursor effects are more-or-less automatically accounted for. For the relatively slow-developing sequences dealt with through the loss of DHR accident, it was deemed satisfactory to follow only those nuclides with half-lives of 30 min and longer for each of the fission products. More rapid sequences, such as ATWS, require the inclusion of shorter lived nuclides, which is currently being done.

3. CONTROL VOLUMES

Transport of fission products is assumed to occur by convection through the series of selected control volumes shown in Table 3, which may become interconnected in several different ways depending on the particular sequence. Note that the core itself is subdivided into 100 control volumes, the balance of the reactor vessel volume into seven control volumes, the containment vessel (wetwell and drywell) into five, and the reactor building into four control volumes; the outside atmosphere is treated as the final control volume in the transport pathway.

Table 3. Control volumes

Reactor vessel	Core (100 control volumes)
	Lower plenum
	Upper plenum
	Steam separator
	Downcomer
	Steam drier
	Upper head
	Steam lines (to MSIV[a])
Containment vessel	Steam lines (from MSIV)
	Tail pipes
	Wetwell water
	Wetwell air
	Drywell
Building	Reactor building
	Refueling floor
	SGTS[b]
	Condenser
Atmosphere	

[a] Main stream isolation valve.
[b] Standby gas treatment system.

Figure 1 illustrates the seven reactor vessel control volumes in addition to the core, while Fig. 2 illustrates the flow relationships of the various containment vessel and reactor building control volumes.

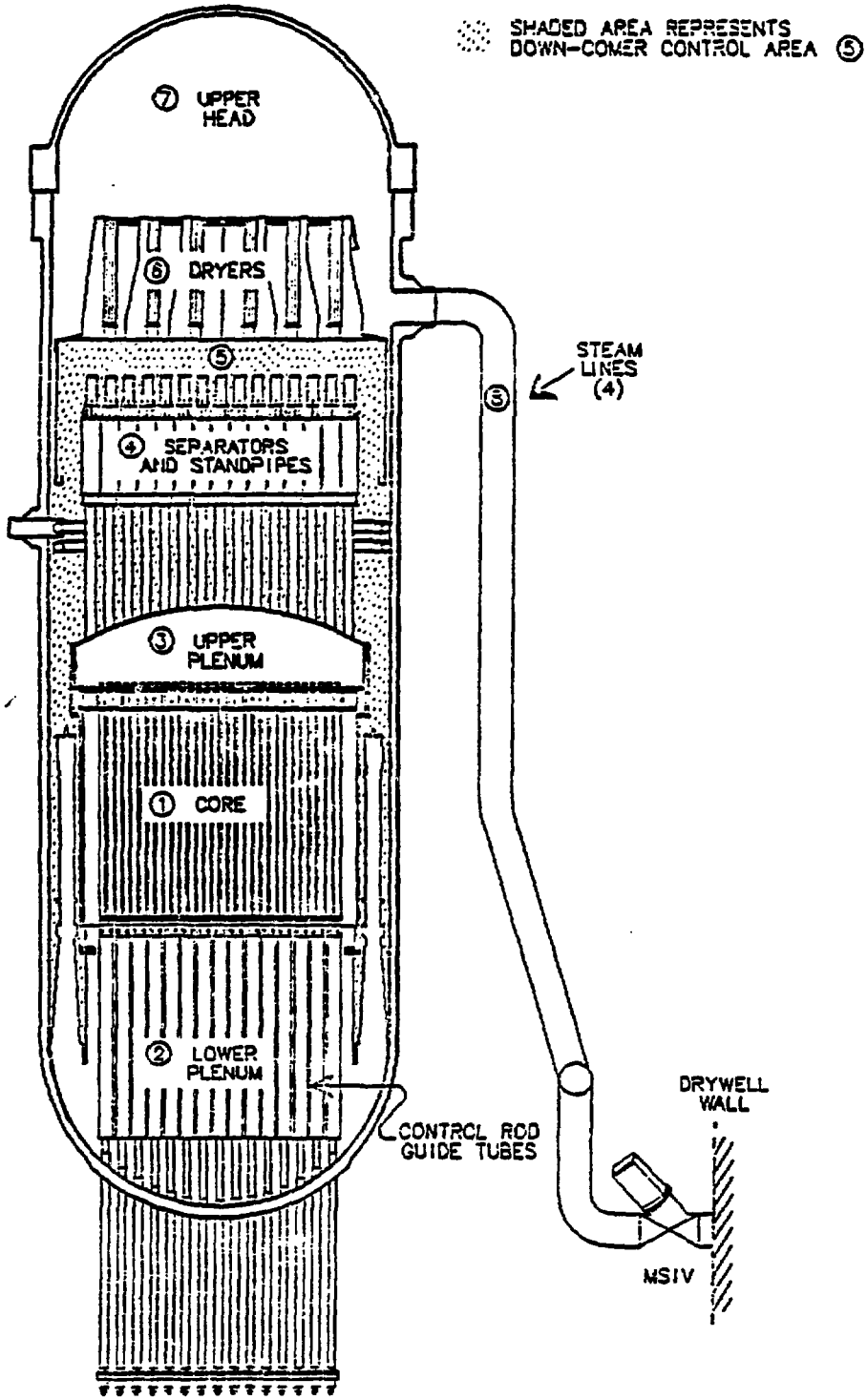

Fig. 1. Reactor vessel control volumes.

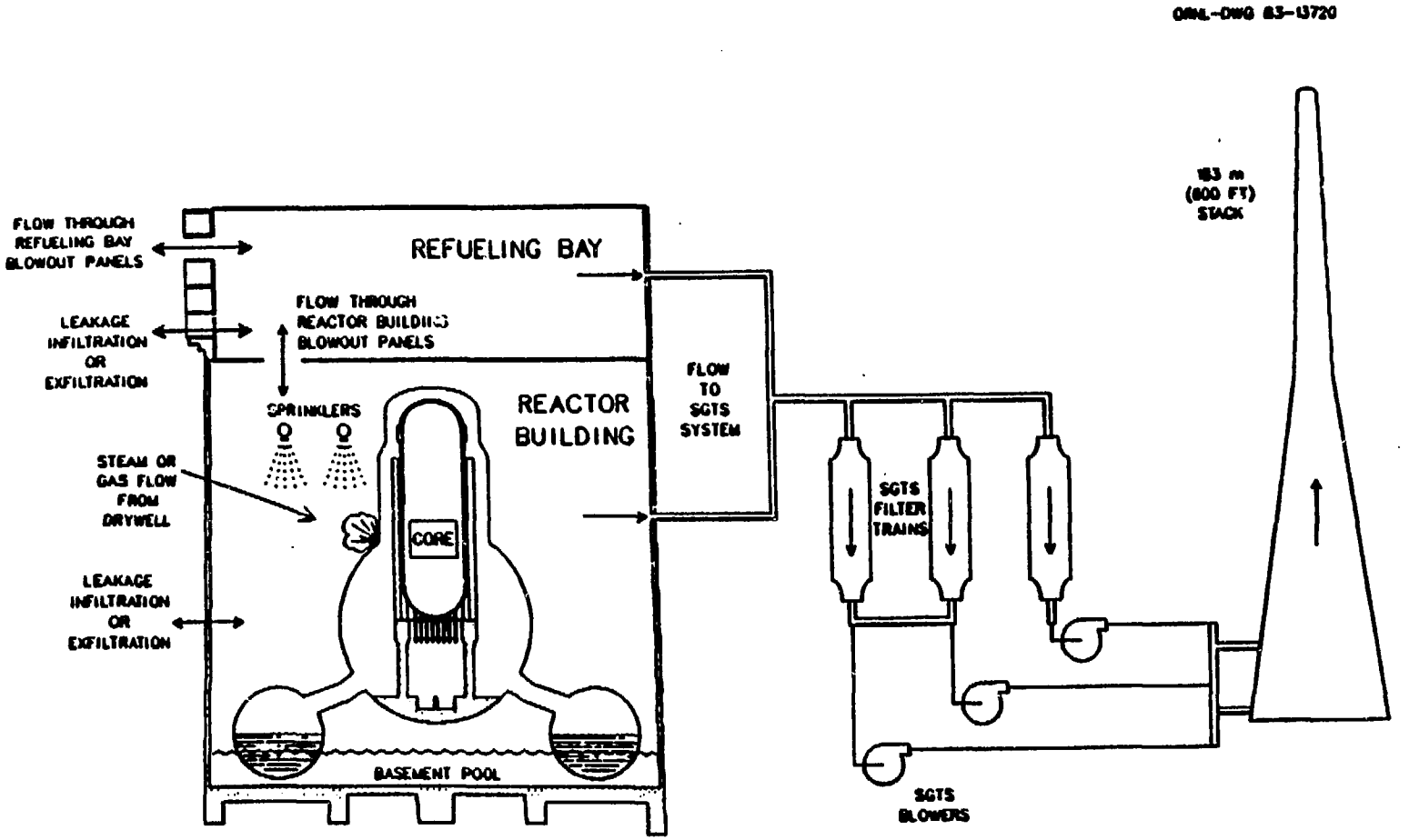

Fig. 2. Reactor containment systems schematic.

4. LOSS OF DHR ACCIDENT SEQUENCE

The loss of DHR accident is a slowly developing sequence initiated by a reactor trip and MSIV closure followed by a failure of both the suppression pool and shutdown cooling modes of the residual heat removal system (RHR). If no operator action is assumed, the steadily increasing vapor pressure of water in the suppression pool ultimately causes a loss of ability to supply water to the core which, in turn, results in core uncovery and overheating.

A detailed analysis of events leading to containment failure (by overpressure due to high steam pressure) for the no-operator-action case is provided in ref. 2; a summary of major events relative to fission product transport is listed in Table 4. Note, containment (drywell) failure is projected to occur 2064 min (34.4 h) following the initiating event, while initial cladding failure occurs later at 2330 min (38.8 h). The earlier containment failure is a key feature of this event sequence. One should also note that the SGTS is actuated early and is presumed to remain on for the entire sequence. Although calculations were terminated at 3100 min (51.7 h), a low rate of fission product transport and release to air was found to still occur at this time.

Table 4. Loss of DHR sequence of events

Event	Time	
	min	hour
Reactor trip, MSIV closure, RHR fails	0	0
SGTS ON	60	1
.	.	.
.	.	.
.	.	.
Suppression pool temperature = 338 F (~170°C)	2060	34
Drywell fails and water boiloff begins	2064	34.4
Reactor building sprays on	~2080	34.7
Core uncovery begins	2200	36.7
First cladding failure	2330	38.8
Core melting begins	2407	40.1
Core slump assumed at 75% of fuel melt	2492	41.5
Reactor vessel bottom head failure	2590	43.2
Core/concrete reaction, drywell aerosol production	~2600	~43.3
Downstream HEPA filter in SGTS tears	2917	48.6
Calculations terminated	3100	51.7

Tables 5 and 6 and Figs. 3 and 4 illustrate some of the key features of this accident. Table 5 shows the core temperature map predicted by MARCH at the time of projected core collapse. Note that both the bottom axial and outer radial core zones remain relatively cool. Based on these temperature projections, we have assumed that the outer radial zone remains in place (with failed cladding) while the lower axial zone falls with the remainder of the core to the bottom of the reactor vessel at core collapse. By this means, about 10% of the volatile fission products are ultimately transported with the core rubble to the concrete floor of the drywell.

Figure 3 illustrates the projected volume flowrates from the drywell into the reactor building. For times up to reactor vessel failure (2590 min), flows are initially caused by suppression pool boiling. When core degradation begins, flows are caused by a combination of steam from the suppression pool and both steam and H_2 from the reactor vessel via the safety relief valves, tailpipes, vacuum break lines, and the prior failure of the drywell containment. Flows for these times are computed by MARCH. Following reactor vessel failure, flows emanating from degrading concrete join the steam and H_2 from the vessel and wetwell. Values predicted by both the INTER subroutine of MARCH and by CORCON MOD1 are compared in Fig. 3, showing the INTER-predicted flows to be approximately three times that for CORCON MOD1.

The rate of fission product sparging from the rubble depends on both the rubble bed temperatures and sparging rates from the decomposing concrete. Figure 4 illustrates that the oxide phase of the rubble bed is projected by CORCON MOD1 to reach high, probably unrealistically high, temperatures. (A more recent version of CORCON predicts much lower oxide layer temperatures.)

The amount and deposition of aerosol material produced in the drywell as the consequence of the degrading concrete are illustrated in Table 6. These estimates use the concrete basemat temperatures and sparging rates predicted by CORCON coupled with concrete vaporization predictions using VANESA. Aerosol settling, agglomeration, and convection predictions were obtained using QUICK coupled with our predicted reactor building and SGTS flowrates. Table 6 shows that ~54% of the estimated 1035 kg of aerosol produced ultimately settles to the reactor building floor (the largest repository) while an estimated 7% ultimately passes through the SGTS following projected HEPA filter failure.

5. RESULTS

Table 7 lists the predicted radioactive inventories in the major reactor zones at $t = 3100$, the final time assumed for the sequence. The percentages listed are normalized to projected elemental activities at $t = 3100$. Note, most of the noble gases (~84%) are effectively sparged from the system by concrete degradation gases. About 10% remains in place in the relatively cool fuel, while ~4.9% follows a leakage path through the shut MSIVs to the condenser.

Table 5. Core temperature map (°C) at time of postulated collapse: Time 2491 min; % core melted = 74.0

Top									
10		2077		2249	2110	2066	2382	2093	2354	1032
9										1404
8										1410
7										1227
6					Molten					1160
5										1093
4										949
3										916
2										843
1	743	971	1127	1116	1093	1060	582	1910	649	372
Bottom	1	2	3	4	5	6	7	8	9	10
	Center									Edge

Table 6. Aerosol production and transport[a] (t = 3070 min)

Total mass produced in drywell = 1035 kg aerosol transport zones	% of mass produced
Airborne in drywell	5
Deposited in drywell	14
Airborne in reactor building	5
Settled on reactor building floor	54
On SGTS filters	15
Passed through SGTS filters	7
Passed through reactor building walls	0

[a]Computed using CORCON MOD1, VANESA, QUICK, and the Reactor Building Model.

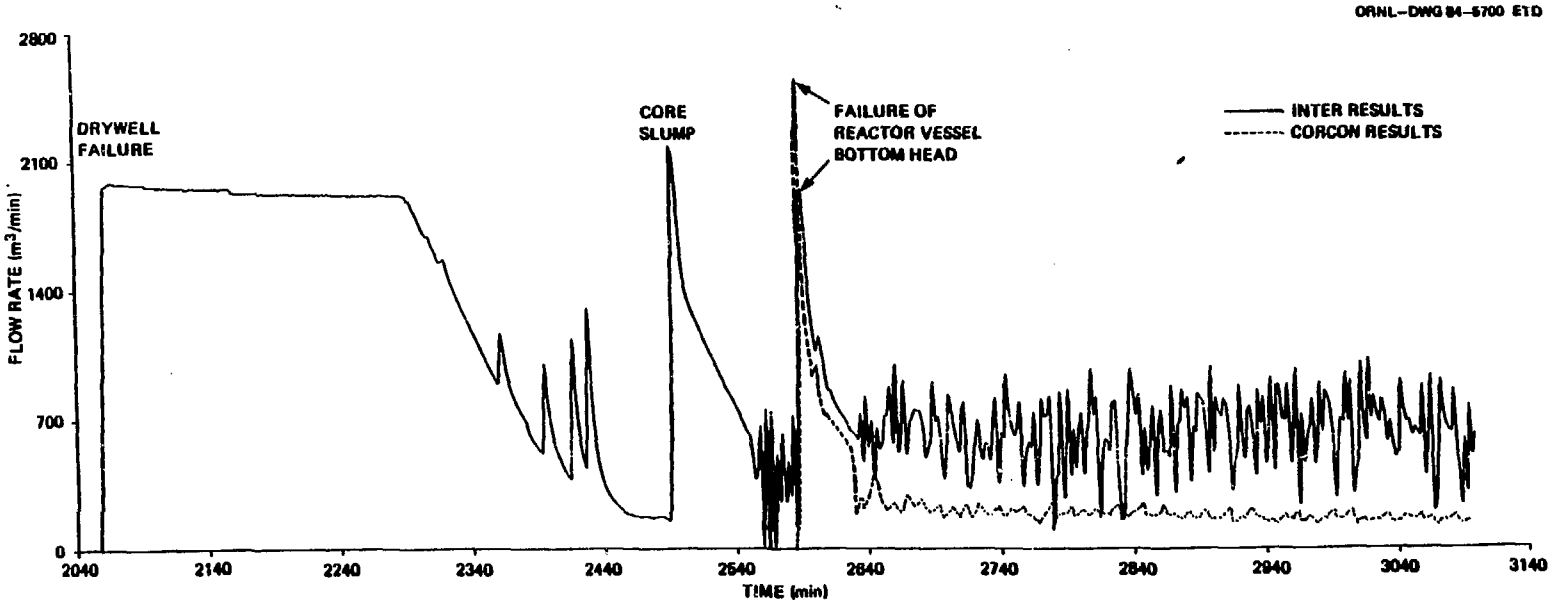

Fig. 3. Volumetric flowrates from the drywell to the reactor building.

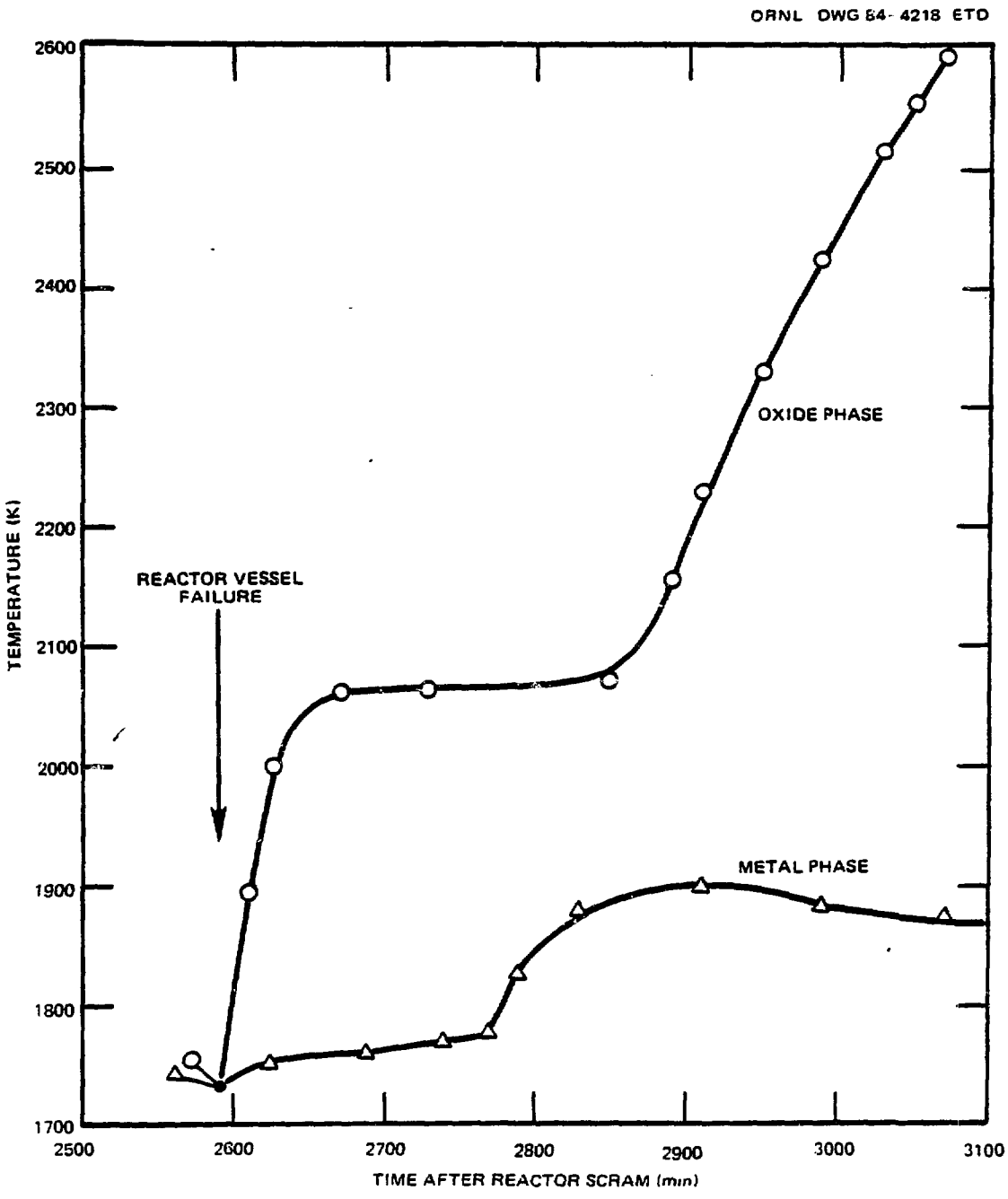

Fig. 4. Fuel/concrete rubble temperatures as predicted by CORCON MOD1.

Table 7. Radioactive inventories at t = 3100 min
(% activity at t = 3100 min)

	Fuel[a]	Reactor vessel	Drywell	Wetwell	Reactor building	Condenser	SGTS	Atmosphere
Kr	10	ε[b]	ε	ε	1.1	4.9	ε	84
Xe	10	ε	ε	ε	1.1	4.9	ε	84
I	11	81	3.5	0.1	2.8	ε	0.86	4.3×10^{-4}
Cs	12	81	7.3	0.083	0.4	0.01	3.1×10^{-3}	2.9×10^{-4}

[a]For fuel remaining in place.
[b]ε = <0.01%.

The major overall difference between noble gas and iodine and cesium transport is seen to be the degree of trapping on reactor vessel and drywell surfaces. Current transport estimates lead to the result that ~81% of the Cs and I released from the fuel is retained on the relatively cool reactor vessel surfaces either by direct condensation of CsOH and CsI or by association with aerosols which subsequently plate out.

The third largest downstream repository from the fuel for iodine at t = 3100 is seen to be the reactor building pool formed mainly by the opening of the fire protection spray system. Somewhat surprisingly, more iodine is trapped in this manner than by the suppression pool.

Similarly, about 3.5% of the iodine and 7.3% of the cesium are estimated to be captured on drywell surfaces by various plating mechanisms such as condensation or chemisorption on aerosols and subsequent aerosol deposition.

As shown in Table 7, an estimated 0.86% of the iodine at t = 3100 min is predicted to reside in the SGTS (primarily on the charcoal) while about 4×10^{-4}% of the current iodine activity at 3100 min is predicted to be in the atmosphere.

An estimated 3×10^{-3}% of the current cesium activity is predicted to exist in the SGTS (primarily on the HEPA filters) at t = 3100 min while ~2.9×10^{-4}% of the current cesium activity is predicted to be released to the air, the overwhelming portion of which occurs following projected HEPA filter rupture.

An example of the numerous time-dependent inventory values presented in ref. 1 is illustrated in Fig. 5, which refers to predicted iodine activities released to the atmosphere as either gaseous iodine, iodine associated aerosols by chemisorption, and iodine converted to organic iodides, primarily in the reactor building. (This latter estimate is currently under careful examination and is likely to be changed.) Note, that current model assumptions predict the predominant mode of iodine release for this accident to be gaseous and to occur through the charcoal beds of the SGTS.

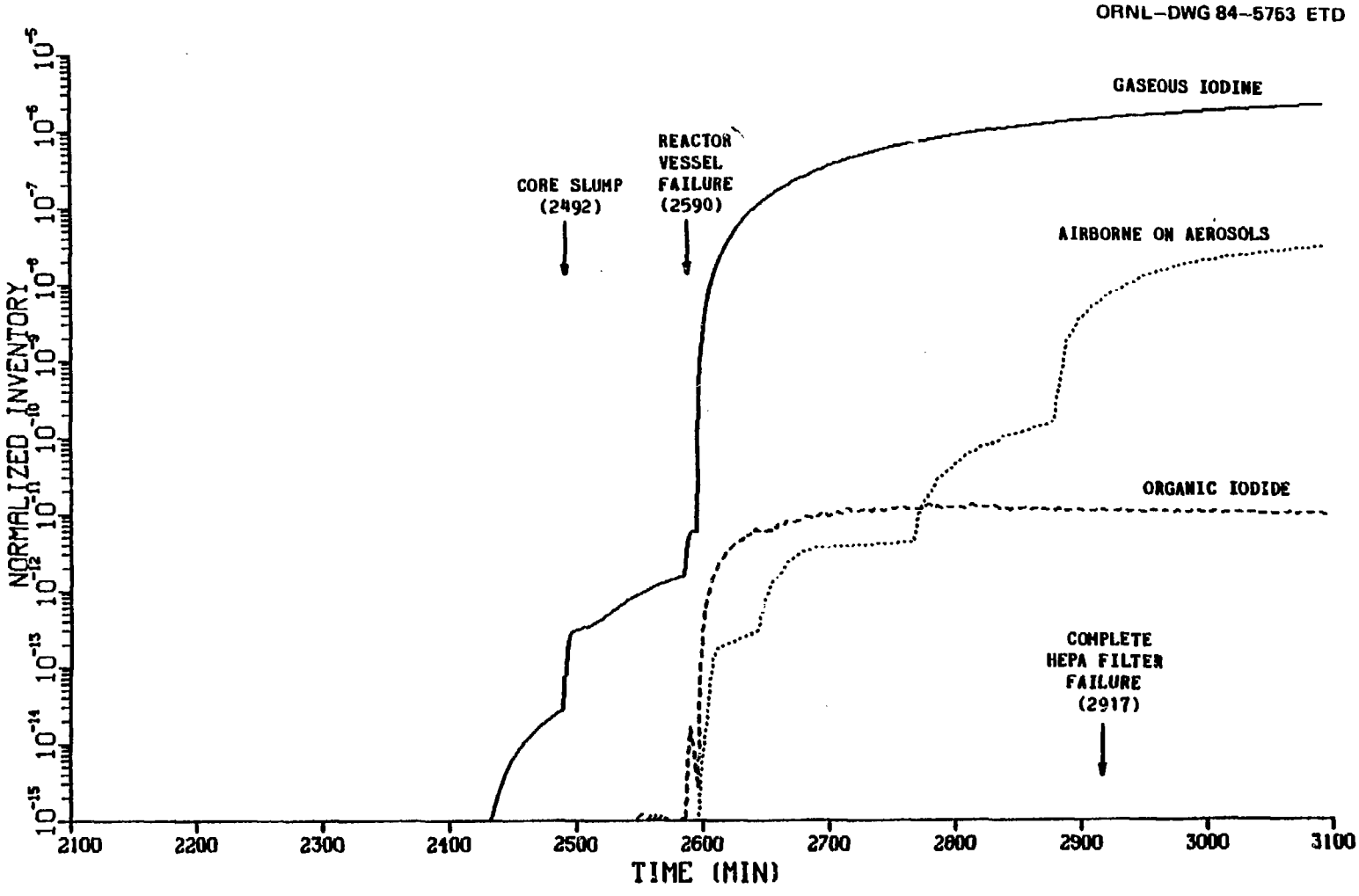

Fig. 5. Predicted iodine radioactivity in the atmosphere — normalized to initial activity of nuclide with half-lives >30 min.

6. FISSION PRODUCT TRANSPORT MODEL CHANGES

At present, a number of model changes are being made, motivated by the following considerations:

We are currently adding tellurium to the transport calculation, which entails providing estimates of tellurium species and deposition rates, and including appropriate tellurium nuclides.

The accident sequence currently being analyzed is the ATWS sequence, which progresses far more rapidly than the three earlier cases. Therefore, additional, shorter-lived nuclides are being added to the list shown in Table 2. An objective rule for the selection of nuclides, based on ORIGEN-predicted decay power levels, has been developed.

We have also replaced the method for predicting chemical forms, which had serious limitations, with a procedure using the SOLGASMIX program which allows consideration of more chemical forms, and more importantly, interactions between vapor and condensed materials.

In addition, we are reexamining the basic assumptions of the SGTS model, which is of critical importance as the last barrier to the atmosphere. Model revisions, relating particularly to the failure mode of the HEPA filters in the SGTS, will be based on tests being performed for the SASA program at New Mexico State University.

In the future, we hope to incorporate improved methods for organic iodide production rate and iodine volatility predictions that are now under study in Fuel Systems Research Branch programs.

7. REFERENCES

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2. S. A. Hodge et al., Loss of DHR Sequences at Browns Ferry Unit 1 — Accident Sequence Analysis, NUREG/CR-1973 (ORNL/TM-8532), Oak Ridge Natl. Lab., May 1983.

3. Technical Bases for Estimating Fission Product Behavior During LWR Accidents, Chapter 4, NUREG-0772, June 1981.