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STATION BLACKOUT CALCULATIONS FOR BROWNS FERRY*

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Abstract

This paper presents the results of calculations performed with the ORNL SASA code suite for the Station Blackout Severe Accident Sequence at Browns Ferry. The accident is initiated by a loss of offsite power combined with failure of all onsite emergency diesel generators to start and load. The Station Blackout is assumed to persist beyond the point of battery exhaustion (at six hours) and without DC power, cooling water could no longer be injected into the reactor vessel. Calculations are continued through the period of core degradation and melting, reactor vessel failure, and the subsequent containment failure. An estimate of the magnitude and timing of the concomitant fission product releases is also provided.

Background

The Oak Ridge National Laboratory (ORNL) has participated in the Severe Accident Sequence Analysis (SASA) program since it was established in 1980 by the Containment Systems Research Branch of the Nuclear Regulatory Commission. The SASA program at ORNL has examined potentially severe accidents at Boiling Water Reactors (BWRs), with the objective of establishing as realistically as possible the sequence of events and consequences of each accident. The Browns Ferry Unit 1 BWR has been utilized, with the full cooperation of the Tennessee Valley Authority, as the example plant for the accident studies.

The ORNL SASA program has performed detailed studies of five BWR accident sequences: Station Blackout (Ref. 1), Small Break LOCA Outside Primary Containment (Ref. 2), Loss of Decay Heat Removal (Ref. 3), Loss of Injection (Ref. 4), and Anticipated Transient without Scram (ATWS) (Ref. 5). An estimate of the magnitude and timing of fission product releases was published for Station Blackout (Ref. 6), Small Break LOCA Outside Primary Containment (Ref. 7) and Loss of Decay Heat Removal (Ref. 8).

Station Blackout (Refs. 1 and 6) was the first ORNL SASA study (completed in 1981). During the interviewing four years, significant improvements in analytical modeling capabilities (i.e., computer codes) have occurred. In light of these code improvements, the decision was made to repeat the Station Blackout calculations.

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ORNL SASA Code Suite

Determination and analysis of the events in an accident sequence that would occur prior to core uncovery is made by the simulation program BWR-LTAS (Ref. 9, developed at ORNL by R. M. Harrington). The basic assumption of the BWR-LTAS code is that the reactor vessel, internals, and fuel are undamaged. The thermal-hydraulic conditions are calculated for both reactor vessel and primary containment. The programming provides flexibility to model the effect of operator actions. The code simulates all interacting plant systems that determine accident sequence development. The predecessor of the current BWR-LTAS code was developed for the original Station Blackout calculations; but the code has been upgraded and expanded to meet the needs of each subsequent accident sequence studied at ORNL.

The original BWR-LTAS calculations (Ref. 1) for the Station Blackout study demonstrated the importance of depressurizing the reactor vessel before battery failure. The capability to calculate both reactor vessel and primary containment response over the period before core uncovery allowed investigators to fully understand that depressurizing the reactor vessel limits drywell temperature, reduces the total number of required SRV actuations, and extends the time to core uncovery after battery failure. The conservative approach, however, is to assume that the reactor vessel is not depressurized (i.e., the core would uncover at ~8 hours after scram instead of ~10 hours for the case with depressurization). The calculations for the sequence without depressurization are presented in this paper.

The response of the primary system and primary containment during the period of the accident sequence following core uncovery is determined by the MARCON 2.1B and MELRPI codes. MARCON 2.1B is based upon MARCH 2.0 (Ref. 10) but utilizes CORCON (Ref. 11) instead of INTER for the corium-concrete interactions and employs the ORNL SASA program BWR models. These include representation of all special BWR features such as channel boxes, control blades, and safety relief valves and incorporation of properties routines that are correct for the saturated conditions within a BWR vessel. The code also includes models for the reaction of the B_4C control rod powder with steam. Additional major model changes in MARCON for both in-vessel and ex-vessel phases of the simulation are presented in Table 1. The original ORNL Station Blackout calculations (Ref. 1) used MARCH 1.1 (Ref. 12) for the period of the accident following core uncovery. The deficiencies in early versions of MARCH with regard to BWR modeling have been extensively identified (Refs. 2, 4, 13).

MARCON 2.1B represents a major step forward in user ability to model BWRs; however, additional calculations are still required from a detailed BWR core degradation code MELRPI (Refs. 14, 15). The core melt and melt relocation models in MARCON are simplified and non-mechanistic; so, the MELRPI code results provide guidance in selecting MARCON input which results in the 'proper' progression in MARCON calculations of the core melt relocation. The application of MELRPI to the Station Blackout calculations also resulted in new model changes in MARCON. This represents the first application of MELRPI in a SASA accident study. The use of MELRPI is discussed by Ahmet Sozer in another paper presented at this WRSR meeting entitled "MELRPI Development and Use."

The BWR secondary containment model (SCM) was developed and first used in the fission product transport analysis of the small break LOCA outside containment sequence (Refs. 2 and 7). The SCM was originally developed because the containment model provided in MARCH subroutine MACE does not permit analyses that includes adequate representation of the response of BWR secondary containment structures.

The purpose of the secondary containment model is to calculate the response of the reactor building to in-leakage from the drywell under accident conditions. The model also calculates the response of the refueling floor to in-leakage from the reactor building and in-leakage (or exfiltration) from the atmosphere. Other factors such as heat sinks, condensation of steam, fire protection sprays and the standby gas treatment system (SGTS) are included in the SCM. However, the SGTS would not be operational during Station Blackout.

The individual inventories of the six constituents that make up the atmospheres in the reactor building and refueling bay flow control volumes are determined at each time step; these are steam, CO_2, CO, H_2, N_2, and O_2. The leakage rates of each constituent from the drywell into the reactor building are taken directly from the MARCON code output, as is the temperature at the leakage source (reactor vessel or drywell).

The ORNL code TRENDS was used to estimate transport and retention characteristics for the following groups of volatile elements: xenon-krypton, iodine-bromine, cesium-rubidium, and tellurium-selenium. The TRENDS code analyzes fission product behavior in the primary coolant system as well as in both the primary and secondary containments, and includes models of the following processes:

1. releases from fuel (failed or intact)
2. convective transport in liquid or gas flows
3. chemical interactions
4. radioactive decay

Fission product inventories at shutdown are obtained from the ORIGEN2 code [16] and are calculated for each individual cell in the 5 × 5 core nodalization. Releases are determined from core fuel using a modification of the NUREG-0772 [17] model and from the drywell rubble by the VANESA code [18]. Convective transport is determined using thermal hydraulic information from MARCON and SCM, and the assumption that each control volume is instantly well-mixed. Aerosol behavior for the current analysis was done using the QUICK code [19], although current plans are to use a more comprehensive model in future work.

Chemical interactions in the reactor vessel include dissolution in water, as well as condensation (CsI, CsOH, $CsBO_2$) and adsorption (HI, I, I_2, Te_2) onto both steel surfaces and airborn aerosols (which may subsequently deposit). The species distribution is recalculated at each time step by the SOLGASMIX code [20], which solves for the equilibrium distribution by minimizing free energy. Currently included are 18 gas phase and 8 condensed species, including CsI, CsOH, HI, and $CsBO_2$. In the primary and secondary containment volumes, models are again included for deposition of various species onto surfaces and aerosols; dissolution in water pools is modelled using partition coefficients. Species redistribution currently involves the two iodine reactions

$$I^-(aq) \rightarrow I_2(aq)$$

$$I_2(g) \rightarrow CH_3I(g)$$

Another important characteristic of the TRENDS code is its inclusion of decay chain modelling. Decay equations that include all relevant parent-daughter relationships are solved numerically by the fourth order Adams predictor-corrector scheme. Decrement due to decay and increment due to precursor decay occurs for each nuclide, in each control volume, at each time step. Thus, precursor transport and contributions to daughter transport are automatically included.

Station Blackout Sequence

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a hypothetical Station Blackout. This accident would be initiated by a loss of offsite power concurrent with a failure of all eight of the onsite diesel-generators to start and load; the only remaining electrical power at this three-unit plant would be that derived from the station batteries. It is assumed that the Station Blackout occurs at a time when Unit 1 is operating at 100% power, and only Unit 1 is assumed to be affected. The 250 volt DC battery system at Browns Ferry could remain operational for a significant period of time. In response to AEC inquiry in 1971, during the period of plant construction, TVA estimated that the steam-driven High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, which use DC power for turbine control and valve operation, could remain operational for a period of four to six hours. Subsequently, in 1981, TVA performed a battery capacity calculation which shows that the unit batteries can be expected to last as long as seven hours under blackout conditions. A period of six hours has been assumed for this study.

The initial ORNL Station Blackout study (Ref. 1) demonstrated that it would be beneficial for operator action to depressurize the reactor vessel early in the initial phase of a Station Blackout. This depressurization removes a great deal of steam and the associated stored energy from the reactor vessel at a time when the RCIC system is available to inject replacement water from the condensate storage tank and thereby maintain the reactor vessel level. Subsequently, when water injection capability is lost for any reason, remote-manual relief valve operation would be terminated and there would be no further water loss from the reactor vessel until the pressure has been restored to the setpoint [7.72 MPa (1105 psig)] for automatic relief valve actuation. Because of the large amount of water to be reheated and the reduced level of decay heat, this repressurization would require a significant period of time. In addition, the subsequent boiloff would begin from a very high vessel level because of the increase in the specific volume of the water as it is heated and repressurized. Thus, an early depressurization will provide a significant period of valuable additional time for preparative and possible corrective action before core uncovery after injection capability is lost. This study conservatively assumes that there is no depressurization.

Results

Thermal-hydraulic behavior of the primary and secondary containments:

Within 30 seconds following the inception of a Station Blackout, the reactor would have scrammed and the reactor vessel would be isolated behind the closed main steam isolation valves (MSIV's). The initial phase of the Station Blackout extends from the time of reactor vessel isolation until the time at which the 250 volt DC system fails due to battery exhaustion. During this period, the operator would maintain reactor vessel water level in the normal operating range (Fig. 1) by intermittent operation of the RCIC system, with the HPCI system available as a backup (Fig. 2). Each of these water-injection systems is normally aligned to pump water from the condensate storage tank into the reactor vessel via a feedwater line.

The Control Room instrumentation necessary to monitor reactor vessel level and pressure and for operation of the RCIC and HPCI systems would remain available during this period.

The operator would also take action during the initial phase to control reactor vessel pressure by means of remote-manual operation of the primary relief valves. The primary relief valves would actuate automatically to prevent vessel overpressurization if the operator did not act; the purpose of pressure control by remote-manual operation is to reduce the total number of valve actuations by means of an increased pressure reduction per valve operation and to permit the steam entering the pressure suppression pool to be passed by different relief valves in succession. This provides a more even spacial distribution of the transferred energy around the circumference of the pressure suppression pool.

The plant response during the initial phase of a Station Blackout can be summarized as an open cycle. Water would be pumped from the condensate storage tank into the reactor vessel by the RCIC system as necessary to maintain level in the normal operating range. The injected water would be heated by the reactor decay heat and subsequently passed to the pressure suppression pool as steam when the operator remote-manually opens the relief valves as necessary to maintain the desired reactor vessel pressure. Stable reactor vessel level and pressure control is maintained during this period, but the condensate storage tank is being depleted and both the level and temperature of the pressure suppression pool are increasing. However, without question, the limiting factor for continued removal of decay heat and the prevention of core uncovery is the availability of DC power.

The station batteries fail after six hours operation; subsequently the operator can no longer manually actuate the SRVs or inject water into the vessel. Thus begins a monotonic decrease (boiloff) in the reactor vessel water level (Figs. 1 and 3) due to intermittent loss of fluid (steam) through the primary relief valves which actuate automatically.

Without restoration of power, the operator can do nothing to impede the progression of the accident. The core uncovers at 479 minutes and the core structures begin heating up, oxidizing and melting. Significant core structural relocation (molten control blades and canisters) begins at 572 mins, this downward relocation immediately increases steaming which decreases water level (Fig. 3) and increases SRV actuation (Fig. 4) until core plate dryout occurs (at 630 mins) at which time steaming ceases. The pressure (Fig. 4) decreases after core plate dryout due to leakage through the MSIVs to the condenser. Fuel melting starts at 604 mins and structural relocation continues (dropping onto a dry core plate) until the core plate fails (on temperature) at 682 mins. By user input, core plate failure occurs when the combined core plate and debris are at 964 K (1275°F); at this time, there is approximately 33500 kg (73853 lbs) of solidified debris resting on the core plate. The core, however, is supported by the control blade guide tubes in the bottom head; and, although the core plate and debris fall into the bottom head and are quenched, the core remains in place until collapse (50% molten) at 695 mins. [The 50% criteria was used since it produced close agreement between MARCON and MELRPI.]

After core collapse, the core debris boils off the water in the vessel bottom head over a period of ~15 mins and, in the process, the core debris cools to 1580 K (2384°F). The debris then reheats, eventually failing a bottom head penetration at 734 mins. This causes the vessel to depressurize (Fig. 4) until the vessel pressure equalizes with the drywell pressure (Fig. 5). At this point, the corium is still solid; it is assumed to leave the vessel when it reaches a liquid state at 2200 K (3500°F).

The liquid corium leaves the vessel at 797 mins, falling into two 1900 liter (500 gal) sumps. After vessel failure, the containment pressure increases due to boiling of the water initially in the drywell sumps. The containment fails shortly after the corium/concrete reaction starts (~805 mins). For the remainder of the accident, the drywell thermal/hydraulics are dictated by the corium/concrete reaction with very high drywell atmospheric temperatures being predicted (Fig. 6).

A synopsis of the major events in the accident and the event timing is presented in Table 2. These events are clearly reflected in the in-vessel water level (Fig. 3) and pressure (Fig. 4) responses and in the drywell pressure response (Fig. 5). It should be noted that there is a significant period of time between reactor scram and core uncovery (8 hrs) and that the subsequent vessel failure dues not occur until 13.4 hours into the accident. The sequence timing also reflects the approach developed at ORNL to represent the events between onset of core degradation and vessel failure for BWRs (summarized in Table 3).

The blowdown from the drywell into the secondary containment after containment failure (Fig. 5) initially fails the blowout panels between the reactor building and the refueling bay and between the refueling bay and the environment which are depicted in Fig. 7. The conditions in the secondary containment (Fig. 8) are determined by the inleakage from the drywell and the fire protection system sprays which actuate automatically. The reactor building sprays are assumed to be available in Station Blackout since they have a dedicated diesel-generator.

Fission Product Transport Analysis

Pathways for the transport of fission products correspond closely with convective flow patterns between inner reactor volumes and the outside atmosphere, as illustrated in Figure 9. In the early stages of the station blackout accident (with the reactor vessel still intact), large amounts of volatile fission products are released from over-heated or melted fuel and flow into the upper regions of the reactor vessel. Those that are not deposited flow into the wetwell with SRV actuation or are carried into the main condensers by leakage flow past the MSIVs. The latter constitutes a significant pathway for the transfer of fission products out of primary containment, although not necessarily to the atmosphere. Because small venting occurs routinely between the primary and secondary containments, small amounts of wetwell inventories may leak into the reactor building during the early stages of the accident. However, these releases are dwarfed by those from the corium-concrete reaction after drywell failure.

The release pathways after reactor vessel bottom head failure and containment failure are illustrated in the lower portion of Fig. 9. The dominant pathway transports fission products out of the drywell and into the reactor building, from which small amounts leak directly to the atmosphere. Considerably larger amounts are retained in the reactor building by dissolution in water (arising from both condensing steam and fire protection system sprays), deposition onto wetted walls, or deposition on aerosols which subsequently settle. Significant leakage from the reactor building to the refueling bay also occurs, after which the fission products either deposit in the refueling bay or are carried to the atmosphere by leakage through the blowout panels. Thus, the primary atmospheric releases are due to leakage from the reactor building and refueling bay after drywell failure.

The actual calculated fission product releases for this accident are fairly low, indicating extensive mitigation by various reactor systems and containment volumes. Considerable holdup of xenon and krypton is indicated, due primarily to the slow leakage rates from secondary containment to the atmosphere. At the end of the transient analysis (1500 min. after shutdown), only about 18% of the shutdown inventory of noble gases had reached the atmosphere, with 33% still in the secondary containment and almost 50% decayed. (In calculating shutdown inventories, only those isotopes are included which are important during the release phase of the accident, i.e., the middle and longer lived isotopes that are actually released during the sequence.)

Atmospheric releases of iodine, cesium, and tellurium constitute only 0.16%, 0.058%, and 0.40%, respectively, of the shutdown fuel inventories for these elements. It is interesting to note that these all are predicted to occur as gas phase releases, with aerosol contributions being several orders of magnitude lower. A principal repository of these elements is the pressure suppression pool, for which transient activity is shown in Figure 10. As seen in the figure, major events in the accident sequence can be noted by their effects on wetwell fission product inventories. The pressure vessel releases due to SRV actuation can be seen as stepwise increases for both iodine and cesium between 600 and 700 min. During this time, tellurium is largely retained in the fuel by reaction with zirconium, but is released in large quantities from the drywell rubble after 806 min. Subsequent drywell venting into the wetwell produces the gradual rise in the wetwell tellurium inventory between 1000 and 1100 min.

Other locations containing high levels of iodine are shown in Figure 11, where it is seen that retention by water pools and surfaces play a very important role. It should be noted in Figure 11 that 44% of the shutdown inventory of iodine is accounted for — the remainder has decayed.

Summary

The ORNL SASA code suite is capable of comprehensive analyses of BWR severe accident sequences:

1. greatly improved in-vessel and ex-vessel thermal hydraulic modeling (BWR-LTAS and MARCON 2.1B)

2. secondary containment model (SCM) complements MARCON 2.1B by calculating the thermal-hydraulic response of secondary containment volumes and the effects of additional systems such as SGTS and fire protection sprays
3. comprehensive fission product transport code (TRENDS) with "state-of-the-art" chemistry, radioactive decay for nuclides, and all major transport and retention mechanisms for important volatile elements.

In the Station Blackout accident sequence at Browns Ferry Unit 1, there is significant time available for corrective action, for instance:

1. the core uncovers 8 hours after scram
2. fuel starts to melt 10 hours after scram
3. reactor vessel fails 13.4 hours after scram.

Even without corrective action, the fission product releases are small:

1. less than 1% of I, Cs and Te are released to the atmosphere
2. there is also significant holdup of noble gases in the primary and secondary containments.

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Table 1

MARCON 2.1B INCORPORATES MAJOR IMPROVEMENTS IN BWR
UNCOVERED CORE AND PRIMARY CONTAINMENT RESPONSE
CALCULATIONAL METHODOLOGY

- In-Vessel
  - Decay Heat by Rigorous ANS Standard
  - Canisters, Control Blades, and SRVs
  - Separation of Fuel and Cladding
  - Multi-Region Water Inventory and True Algorithm for Water Level
  - Physical Properties, Steam/Gas Equation of State
  - Accurate Pressure Calculation, Core Quench Models, Boiling and Flashing Algorithm
  - Heat Transfer Correlations for Uncovered Core
  - B_4C/Steam Reaction Models
  - Limited Melt Relocation Models
  - Bases for Vessel Failure
- Ex-Vessel
  - Reactor Vessel Heat Source To Drywell
  - Drywell Sump Models
  - Continuity of Mass and Volume at Vessel Failure
  - CORCON MOD 2
  - Temperature Dependent Specific Heat
  - Pressure Dependent Correlation for Superheat Temperatures
  - Degassing of Concrete in Drywell

Table 2

MAJOR ACCIDENT EVENTS AFTER CORE UNCOVERY

Event	Time after scram (mins)
Core uncovery	479
Structural relocation starts	572
Fuel melting starts	604
Core plate dryout	630
Core plate failure	682
Core collapse	695
Bottom head dryout	709
Penetration failure	734
Vessel pressure equalizes with containment	743
Corium leaves vessel	797
End Hotdrop/start Corcon	800
Containment failure	805

Table 3

ORNL METHODOLOGY EMPLOYED TO REPRESENT THE EVENTS BETWEEN ONSET OF CORE DEGRADATION AND VESSEL FAILURE FOR BWRS

- Molten canisters and control blades relocate onto core plate which causes
  - dryout of core above core plate, and
  - steaming increased before dryout, stopped until core plate failure
- Core plate fails at 1275°F (964 K) subsequently
  - Debris falls into bottom head
- Remaining intact core collapses when molten fraction exceeds specified amount (currently 50%)
- Bottom head dryout
- Penetration failure at debris temperature of 2800°F (1811 K), thus
  - vessel depressurizes, until
  - vessel pressure equilizes with containment
- corium liquidus [after heatup to ~3500°F (2200 K)] leaves vessel

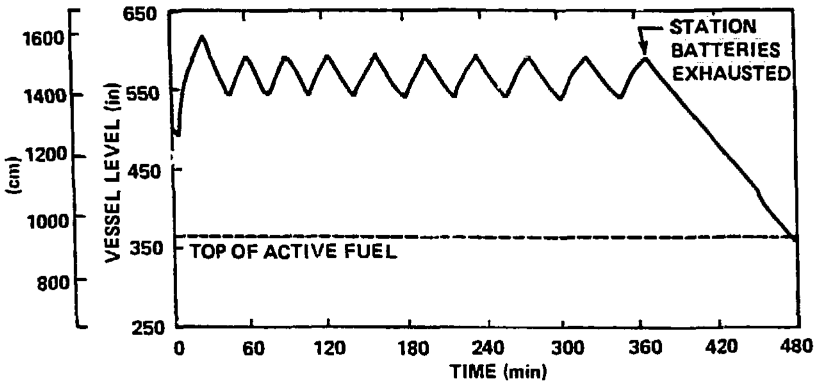

Fig. 1. Reactor vessel water level in the initial phase of the accident (prior to core uncovery).

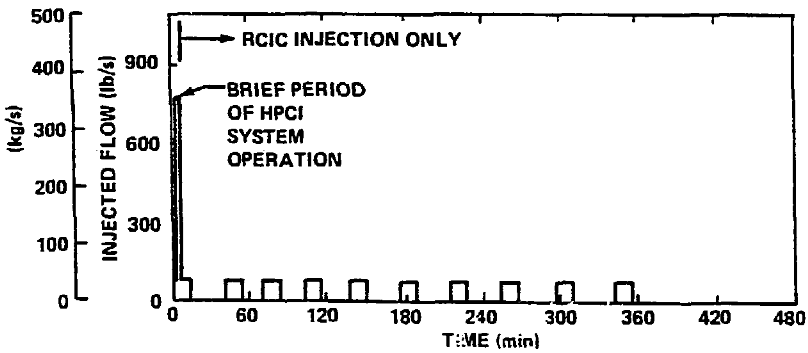

Fig. 2. Reactor vessel injection flow in the initial phase of the accident (prior to core uncovery).

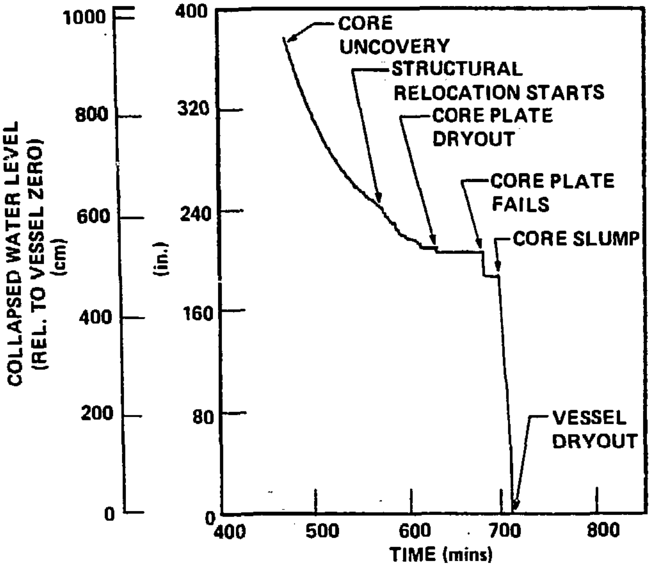

Fig. 3. Reactor vessel water level after core uncovery.

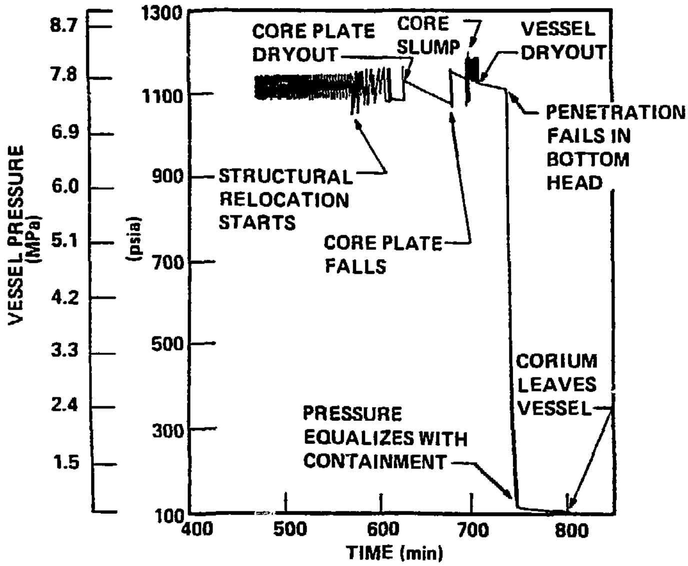

Fig. 4. Reactor vessel pressure after core uncovery.

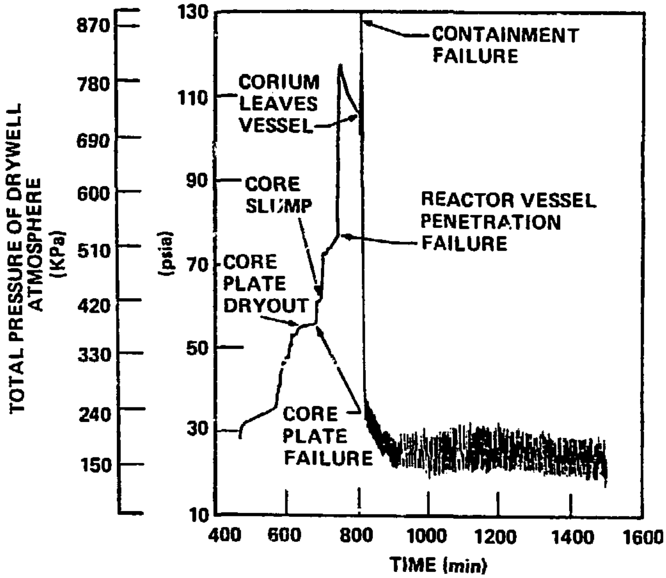

Fig. 5. Drywell pressure after battery exhaustion.

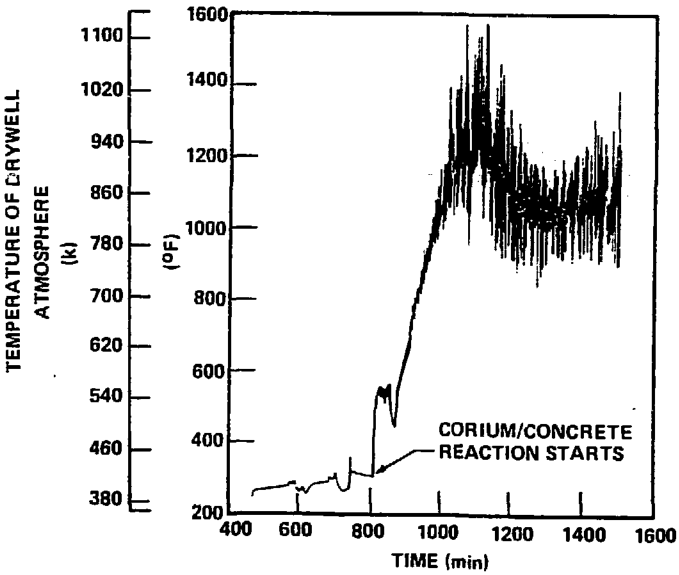

Fig. 6. Drywell atmospheric temperature after battery exhaustion.

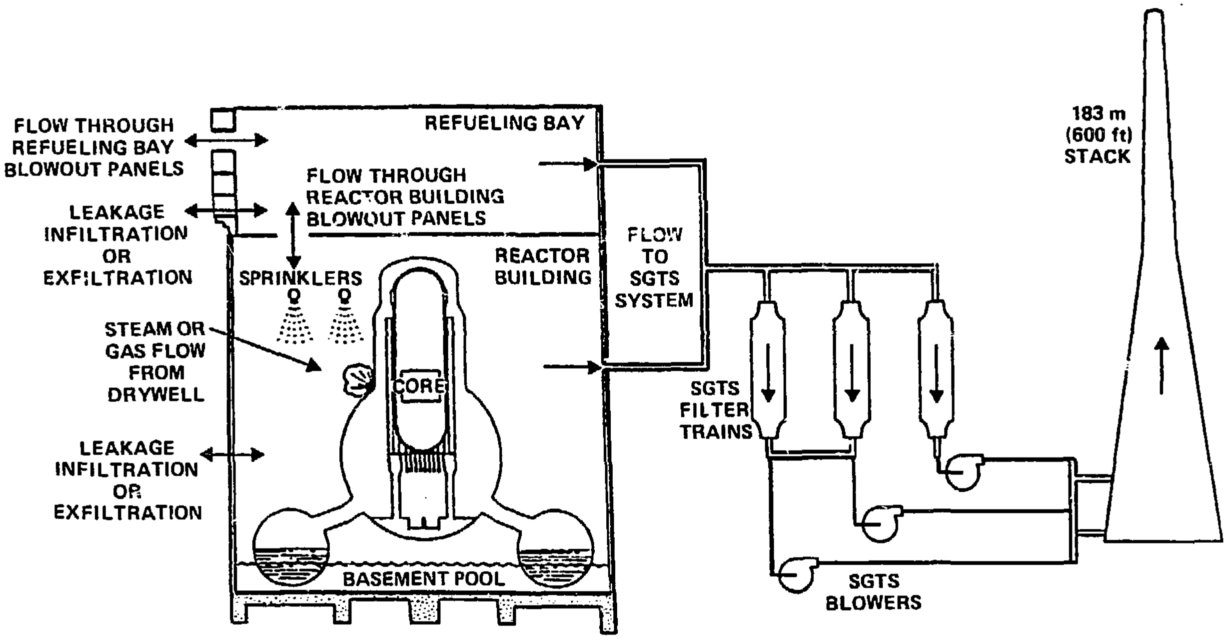

Fig. 7. The Mark 1 secondary containment design with indicated response after drywell failure (the SGTS does not operate in the Station Blackout Sequence).

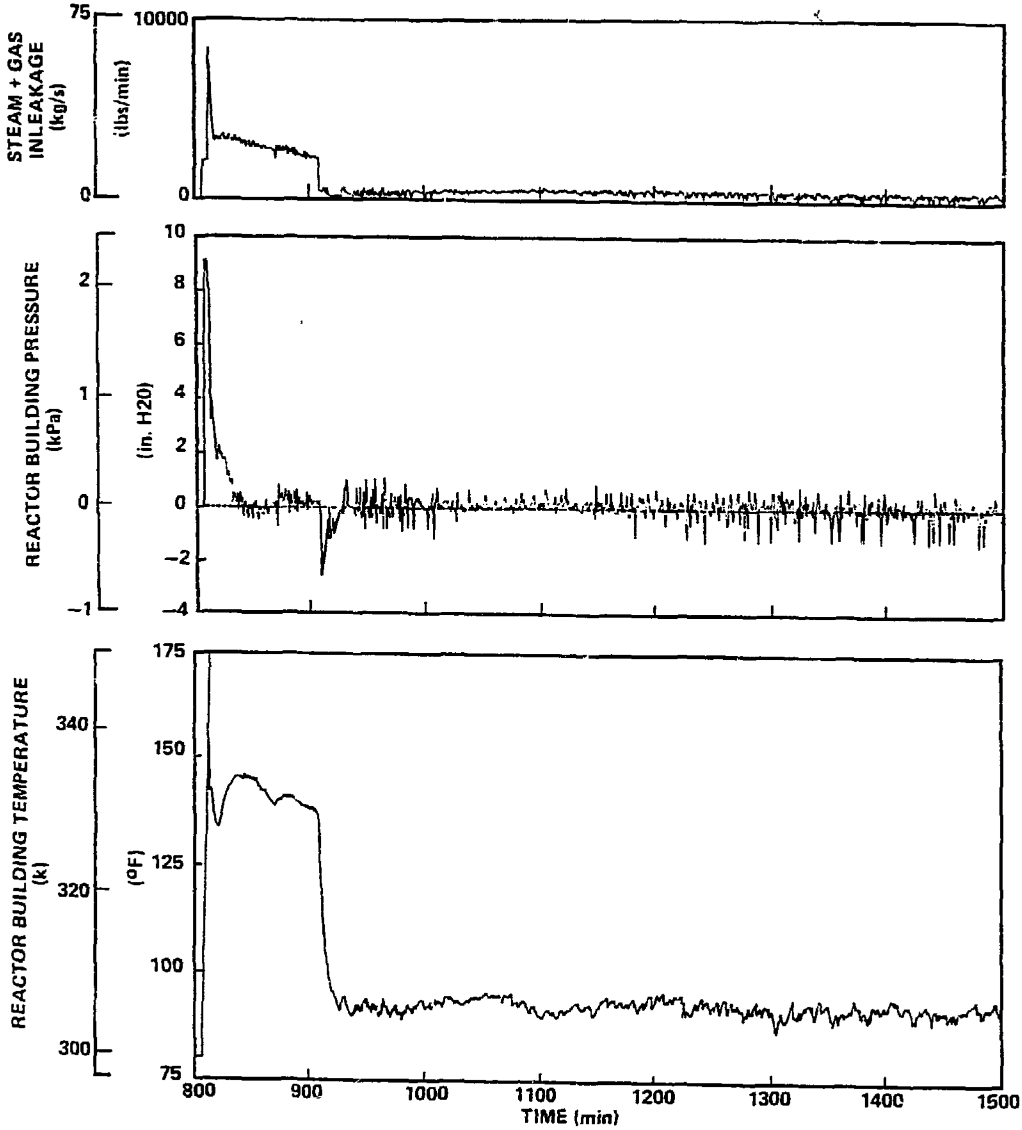

Fig. 8. Reactor building pressure and temperature response caused by steam and gas inleakage from drywell.

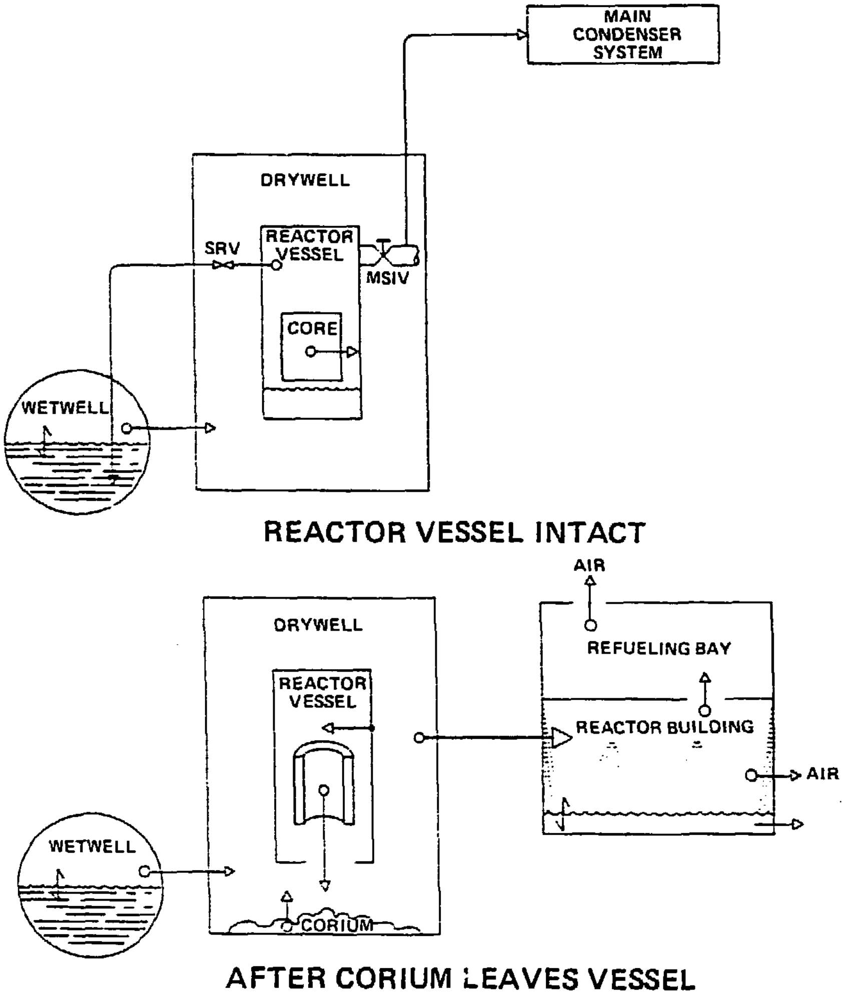

Fig. 9. Fission product pathways for Station Blackout.

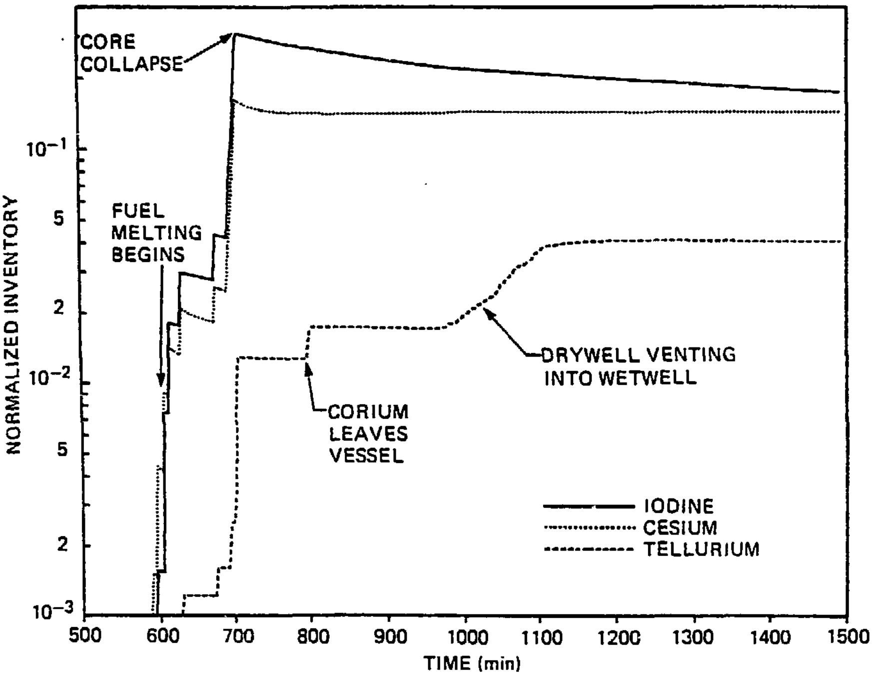

Fig. 10. Transient inventory of I, Cs, and Te in suppression pool.

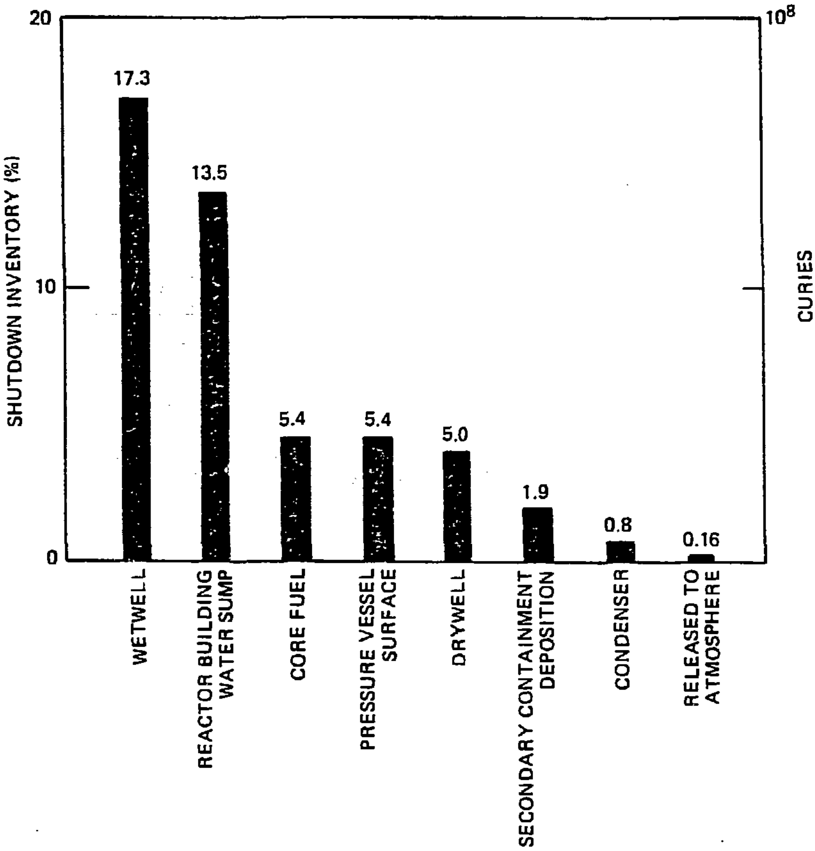

Fig. 11. Principal iodine repositories at end of transient (note that 56% of the shutdown inventory has decayed).