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BWR SEVERE ACCIDENT SEQUENCE ANALYSES AT ORNL — SOME LESSONS LEARNED*

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The Severe Accident Sequence Analysis (SASA) Program was established in October 1980 by the Division of Accident Evaluation of the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (NRC) to study the possible effects of potential nuclear power plant accidents. Under the auspices of the program, boiling water reactor (BWR) studies are being conducted at the Oak Ridge National Laboratory (ORNL) using Browns Ferry Unit 1 as the model plant. Assistance and complete cooperation is provided by the plant owners and operators, the Tennessee Valley Authority (TVA).

The function of the SASA program at ORNL is to conduct detailed analyses of the dominant (most probable) BWR accident sequences, which have been identified by probabilistic risk assessments (PRAs). The SASA studies complement the PRA studies, which use fault-tree and event-tree analytical methods to identify the candidate dominant sequences in an effort to consider all possible accident sequences that might occur at a particular plant. The SASA study objectives include determination of the sequence of events and the magnitude and timing of the associated fission product releases; the results of this more detailed study can subsequently be fed back into an improved PRA.

Experience has shown that both PRA and SASA studies must be plant-specific to induce confidence in the applicability of the results. It has been generally accepted that separate studies are required to cover the broad plant classifications such as the differing PWR designs furnished by the various vendors and for the three basic BWR containment concepts. Beyond this, the PRA and SASA studies show that relatively small differences in plant design can have a large effect on the accident progression and its consequences; some examples for the BWR design are given in this paper.

All ORNL studies to date have concerned Unit 1 of the Browns Ferry Nuclear Plant, a BWR-4 with MK-1 containment design, which is typical of

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most BWR plants in operation today. Four accident studies have been completed, resulting in recommendations for improvements in system design, emergency procedures, and operator training. Necessary severe accident analysis computer code improvements and modifications are an important by-product. The fission product transport work has indicated areas where further basic research is needed. The published reports concern Station Blackout (NUREG/CR-2182), Scram Discharge Volume Break (NUREG/CR-2672), Loss of Decay Heat Removal (NUREG/CR-2973), and Loss of Injection (NUREG/CR-3179).

The ORNL SASA studies have shown that several BWR modeling needs for Severe Accident analyses are unique and very important. The channel boxes and control blades in the core (Fig. 1) and the operation of the safety/relief valves (Fig. 2) must be represented if adequate analyses of the period of core uncovery and fuel heatup are to be performed. After core degradation, the presence of the control rod drive guide tubes and mechanism assemblies in the reactor vessel lower plenum and the numerous penetrations in the reactor vessel bottom head must be considered in the analyses.

In the course of performing the Browns Ferry accident analyses, many lessons have been learned regarding the response of Browns Ferry Unit 1 to the conditions of a Severe Accident. It is the purpose of the paper to discuss some of the more important points, which are of general applicability to plants of the BWR-4 MK-1 design.

A major consideration in some BWR accident sequences is thermal stratification in the pressure suppression pool, which can be quite severe if the pool cooling system is not operational. Thermal stratification causes the upper layer of pool water to be much hotter than the pool bulk average temperature, advancing containment failure by overpressurization as much as eight hours in the Browns Ferry Loss of Decay Heat Removal (TW)* accident sequences.

The BWR-4 design provides two turbine-driven high-pressure injection systems for the purpose of using decay heat to maintain the core covered during accident sequences in which the reactor vessel remains pressurized (Fig. 3). The systems are automatically initiated when the reactor vessel water level falls to a certain value and are automatically terminated when the level is restored. Water is pumped from the condensate storage tank into the reactor vessel in an open cycle. The larger of these systems, the High Pressure Coolant Injection (HPCI) system would continue to automatically come on and go off as necessary to maintain adequate coverage of the core. Nevertheless, the HPCI system automatic controls can threaten the viability of the system.

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*This and other accident sequence alphabetic designators used in this paper are taken from the Reactor Safety Study, WASH-1400.

The HPCI system controls were designed with the Large-Break LOCA Design Basis Accident (DBA) in mind. In consideration of the desirability of converting the open injection cycle into a closed injection cycle for recirculation of coolant through a failed primary coolant system, an automatic and irreversible shift of the HPCI pump suction from the condensate storage tank to the pressure suppression pool upon high sensed pressure suppression pool level has been provided. For accident sequences other than the DBA, this is most undesirable since a high pressure suppression pool level is synonymous with high pool temperature; the system lubricating oil is cooled by the water being pumped and a high level would be caused by the condensation of large amounts of steam. Thus the automatic and irreversible shift of the HPCI pump suction from the condensate storage tank to the pressure suppression pool upon high sensed pressure suppression pool level threatens the viability of the system in most Severe Accident sequences.

In all of the dominant BWR Severe Accident sequences, the preferred source for the available injection systems is the cool water source available from the condensate storage tank. This source is independent from any accident conditions in the Reactor Building and does not threaten the net positive suction head (NPSH) requirements of any Emergency Core Cooling System. The ample condensate storage tank facilities at Browns Ferry have played a large part in the demonstration of the low risk of the operation of this facility. However, since survival of a DBA does not require a large condensate storage tank and economics have a tendency to prevail, condensate storage facilities are smaller at the newly licensed facilities. The ability to inject water into the reactor vessel from the condensate storage tank is important to the prevention and mitigation of Severe Accidents; the condensate storage tank capacity should be large.

The Control Rod Drive (CRD) hydraulic system provides a cooling water flow of 60 gpm under normal reactor operating conditions. As shown in Fig. 4, this flow is automatically maintained by a flow control valve — venturi combination, passing through the cooling water header into the below-piston volume of each of the 185 CRD mechanisms. This cooling flow of about 0.32 gpm per mechanism is not sufficient to cause piston movement and passes into the reactor vessel. The scram accumulators (one for each CRD mechanism) float on the CRD hydraulic pump discharge; there is normally no flow in the scram header, which connects the scram inlet valve to the pump discharge piping between the venturi and the flow control valve.

The CRD hydraulic system alignment during the period when a scram is in effect is shown in Fig. 5. The scram outlet valves are open, providing a path from the above-piston volume of each CRD mechanism to the scram discharge volume. Each scram inlet valve is open, providing a path from the associated scram accumulator and from the CRD hydraulic pump discharge to the below-piston volume of each CRD mechanism. There is an increased flow through the venturi, and the flow control valve, in a vain attempt to reduce the venturi flow, is fully shut. Thus all pumped CRD hydraulic system flow passes through the scram inlet valves.

After the scram strokes are completed, the scram accumulators are fully discharged and the scram discharge volume is full. All of the flow from the CRD hydraulic pump flows through the scram inlet valves into the below-piston volumes of the CRD mechanisms and from there leaks into the reactor vessel past the CRD mechanism seals. It is important to note that the effect is to increase the reactor vessel injection via the CRD hydraulic system from 60 gpm to about 112 gpm and that this increase occurs without any operator action. The CRD hydraulic system centrifugal pump will of course inject through the scram inlet valves at an increased rate if the reactor vessel is depressurized, as much as 184 gpm with complete depressurization. Thus, the BWR scram system automatically provides increased reactor vessel injection when a scram is in effect.

The CRD mechanism seal leakage is such that the scram accumulators will not recharge unless the scram inlet valves are shut. Accordingly, a manual scram follow-up to an ineffective scram cannot be accomplished unless the initial scram can be reset. It should also be noted that scram reset also serves to terminate injection through the scram inlet valves and to restore the reactor vessel injection path through the flow control valve; this reduces the rate of reactor vessel injection back to 60 gpm.

SASA studies have shown that the CRD hydraulic system plays an important role in determining the progression of both TW and Loss of Injection (TQUV) accident sequences for a BWR. This important system should not be neglected in either PRA or SASA accident analyses.

The BWR MK-I primary containment (drywell and wetwell) spray systems are virtually useless for accidents other than the Large-Break LOCA because, by system design, if flow capability exists for primary containment spray, it also exists for reactor vessel injection and there would be no Severe Accident. However, the secondary containment sprays are completely independent and might be useful.

At Browns Ferry, the secondary containment is equipped with a reliable fire protection system that can play an important role in the mitigation of a Severe Accident sequence. The middle floors of the reactor building are protected by a ceiling sprinkler system that is automatically actuated by a combination of clouding of the building atmosphere (steam/ smoke) and high temperature (175°F) in the vicinity of the fusible link sprinkler heads. Three electric-motor driven pumps, powered from the emergency diesel-generator boards, and one diesel-driven fire pump take suction on the Tennessee River. As suggested by Figure 6, the action of the reactor building fire protection sprays is analogous to that of the primary containment sprays in a PWR. The effect is important in BWR accident sequences in which the pressure suppression pool is bypassed.

The major effect of the reactor building fire protection system sprays is to lower the temperature and pressure in the reactor building after the drywell has failed in a Severe Accident sequence. This virtually eliminates direct leakage to the surrounding atmosphere by exfiltration through the secondary containment walls and ensures that the flow from the drywell to the atmosphere passes through the Standby Gas Treatment System (SGTS).

The SGTS filters the exhaust from the BWR secondary containment under accident conditions. A large-capacity system such as that at Browns Ferry (Fig. 7) can provide filtered exhaust of virtually all of the release from a failed drywell even if the reactor building and refueling floor blowout panels are lifted by the brief positive building pressure pulse generated under Severe Accident conditions at the time the drywell fails by overpressurization. As illustrated in Fig. 8, the SGTS blowers have sufficient capacity to maintain a vacuum in the secondary containment even with the blowout panels ruptured. In the reactor building, the effluent from the failed drywell would mix with the inflow from the refueling floor before entering the SGTS.

The harsh environmental conditions that would be generated during an actual Severe Accident might threaten the viability of the SGTS. This has been recognized in the Browns Ferry design, in which the SGTS equipment is located underground in a concrete bunker remote from the reactor building. In assessing the threat to the HEPA filters and charcoal filters of the SGTS, it should be recognized that the atmosphere at the inlet to the filter trains is a mixture of the reactor building and refueling floor atmospheres and is therefore much less severe than that of the reactor building alone.

Under Severe Accident conditions, large amounts of aerosols would be generated in the primary system and, subsequently, by the corium-concrete reaction on the drywell floor of a BWR. Analyses indicate that the HEPA filters of the SGTS would eventually plug with aerosols, but would then immediately tear. Thus the charcoal filters would continue to remove iodine. The magnitude of the fission product releases to the atmosphere during a BWR Severe Accident would depend heavily on the timing of the releases. Releases subsequent to reactor vessel failure have a greater potential for bypassing the pressure suppression pool and ultimately reaching the environment.

A study of Anticipated Transient Without Scram (ATWS) accident sequences for Browns Ferry Unit 1 is currently underway at ORNL. This study involves a cooperative effort between the SASA and Human Factors programs at ORNL and the SASA program at Idaho National Engineering Laboratory as well as pre-publication review of the results by both TVA and General Electric Nuclear Division representatives. A draft report is scheduled for March 1984.

Published ORNL SASA Program Reports

1. Station Blackout at Browns Ferry Unit One — Accident Sequence Analysis, NUREG/CR-2182, Volume, 1, November 1981.

2. Station Blackout at Browns Ferry Unit One — Iodine and Noble Gas Distribution and Release, NUREG/CR-2182, Volume 2, August 1982.

3. SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis, NUREG/CR-2672, Volume 1, November 1982.

4. SBLOCA Outside Containment at Browns Ferry Unit One — Volume 2. Iodine, Cesium, and Noble Gas Distribution and Release, NUREG/CR-2672, Volume 2, September 1983.

5. Loss of DHR Sequences at Browns Ferry Unit One — Accident Sequence Analysis, NUREG/CR-2973, May 1983.

6. The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One, NUREG/CR-3179, September 1983.

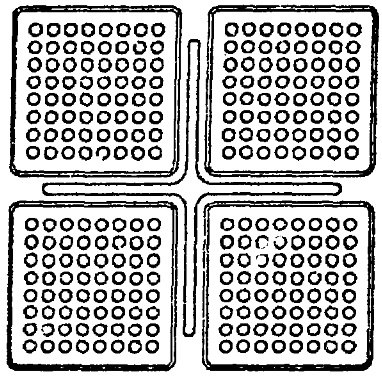

Fig. 1. The BWR core is comprised of repeated sets of four assemblies and a control blade, each individually supported from the reactor vessel bottom head. Each assembly is enclosed in a zircaloy channel box so there is no lateral escape of fluid from an assembly within the core.

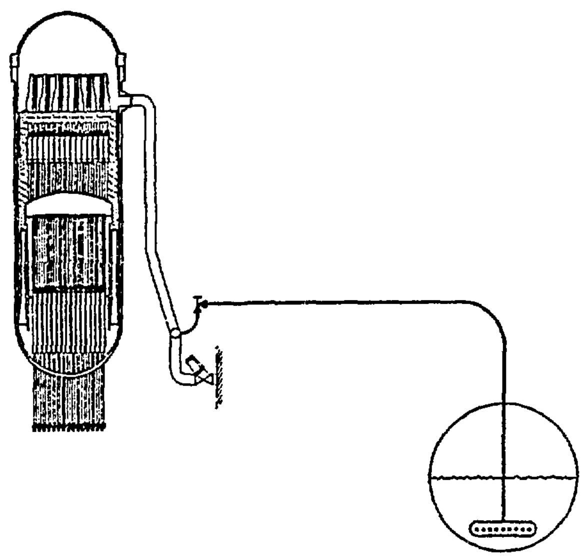

Fig. 2. Under Severe Accident conditions, periodic safety/relief valve discharge would quench the hot fuel in the uncovered portion of the core.

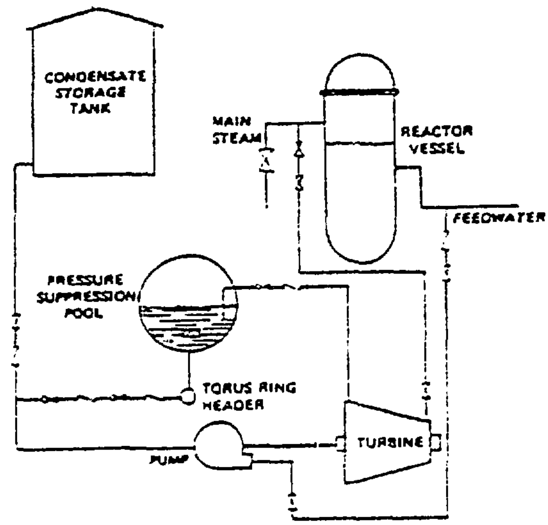

Fig. 3. The High-Pressure Injection Systems require only DC power from the plant batteries and reactor decay heat to provide adequate reactor vessel water level control.

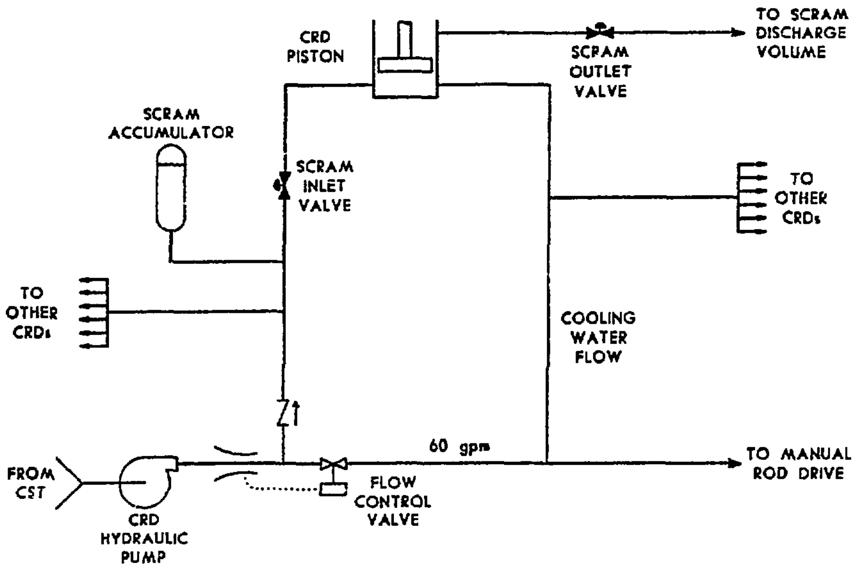

Fig. 4. Reactor vessel injection via the CRD hydraulic system is maintained at 60 gpm during normal reactor operation.

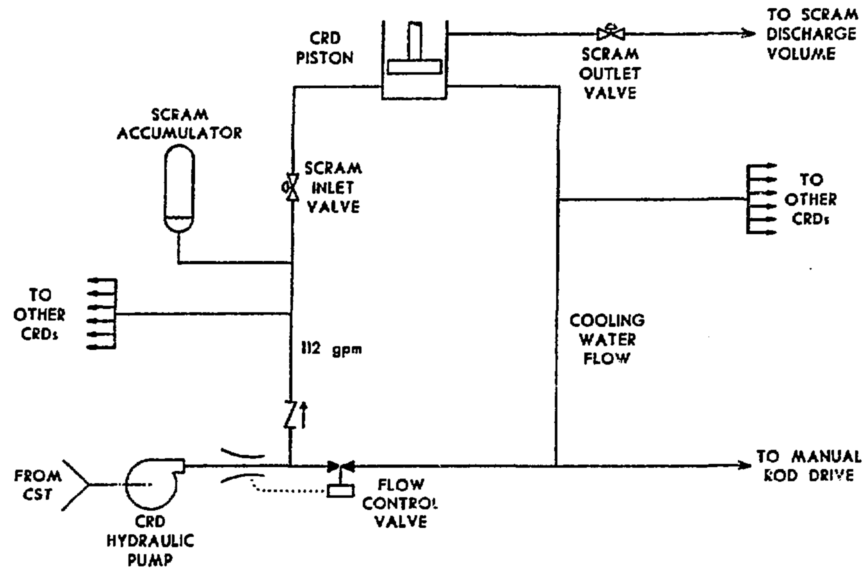

Fig. 5. CRD hydraulic system alignment when a reactor scram is in effect.

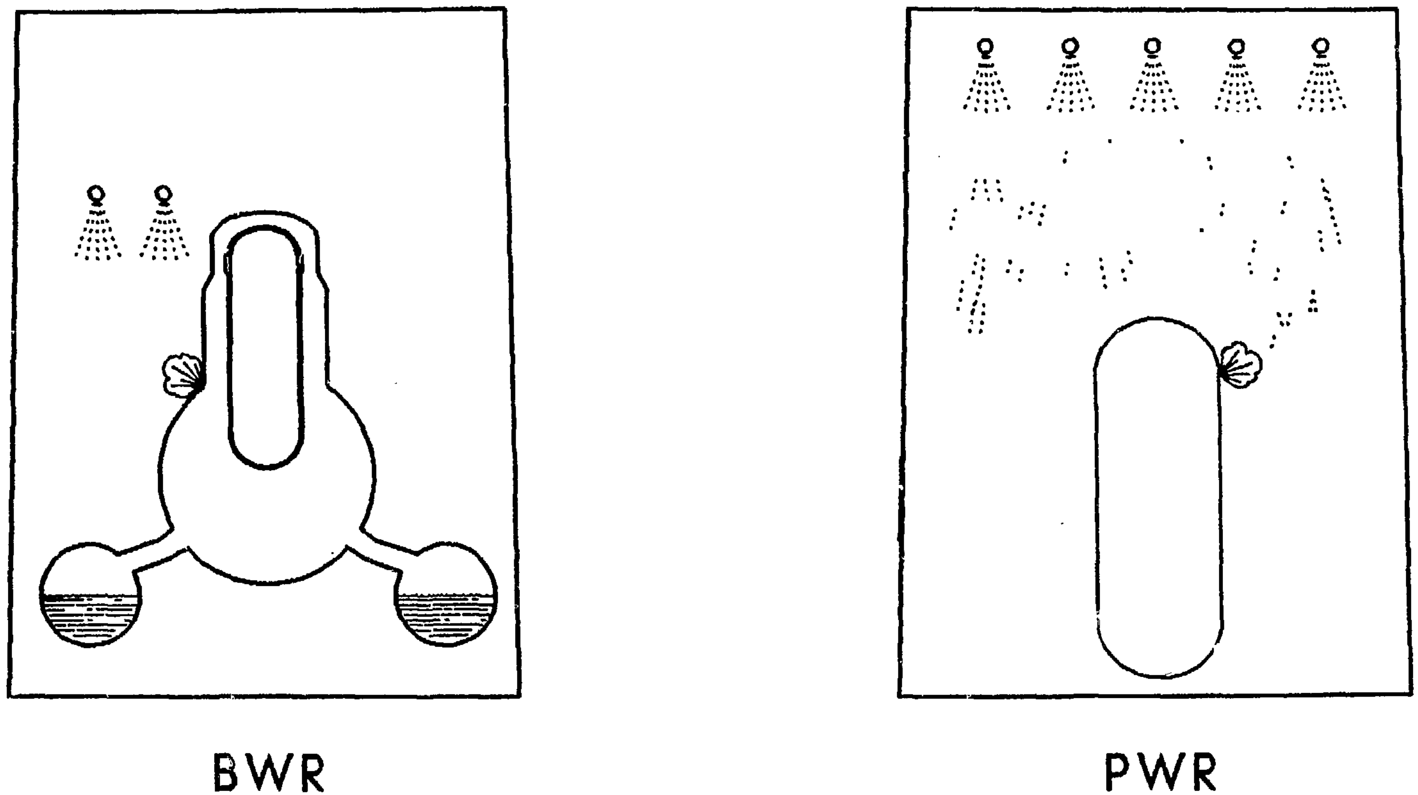

Fig. 6. Illustration of the similarity between operation of primary containment sprays in a PWR and reactor building sprays in a BWR.

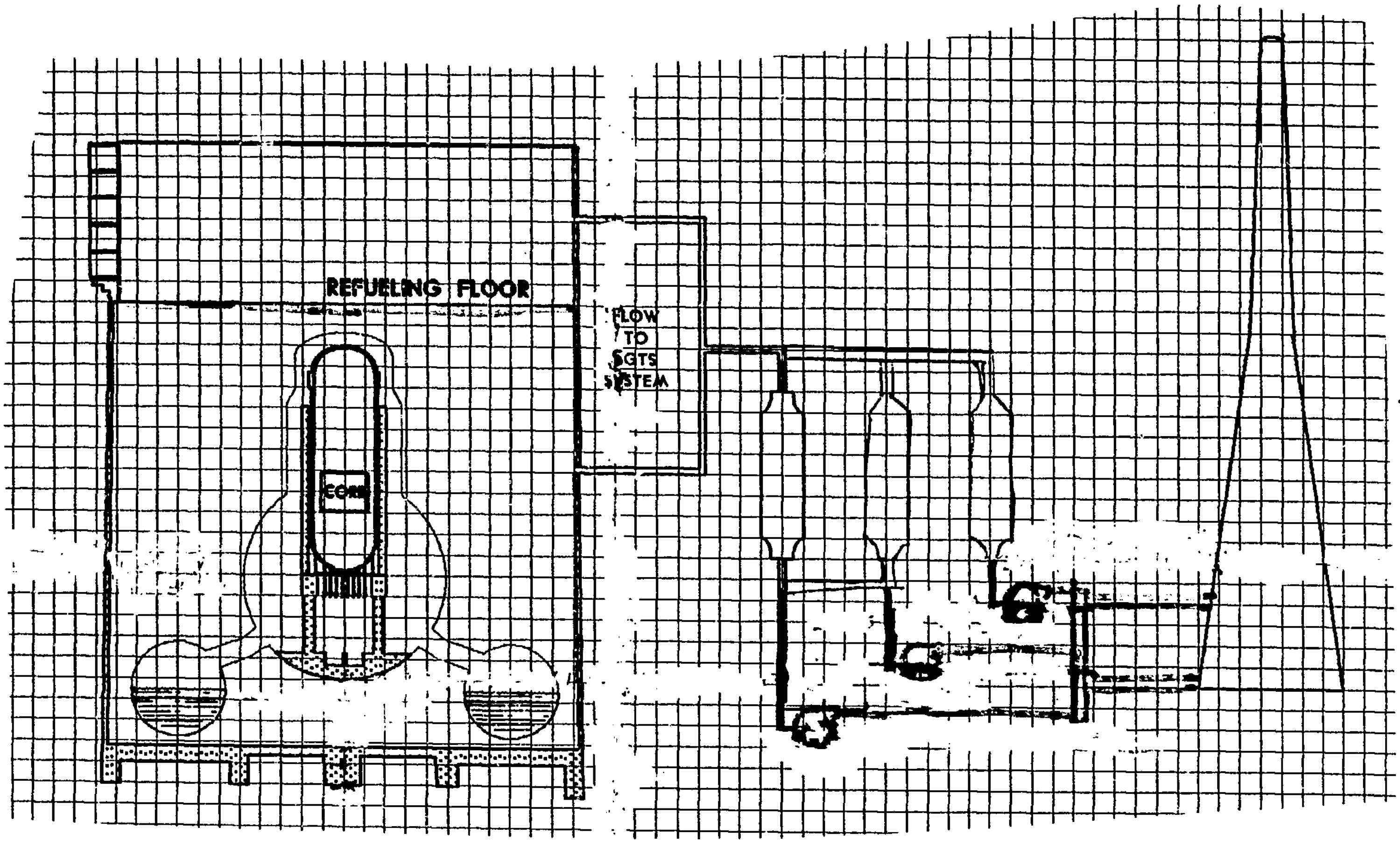

Fig. 7. Schematic drawing of the (25,000 cfm) Browns Ferry standby gas treatment system.

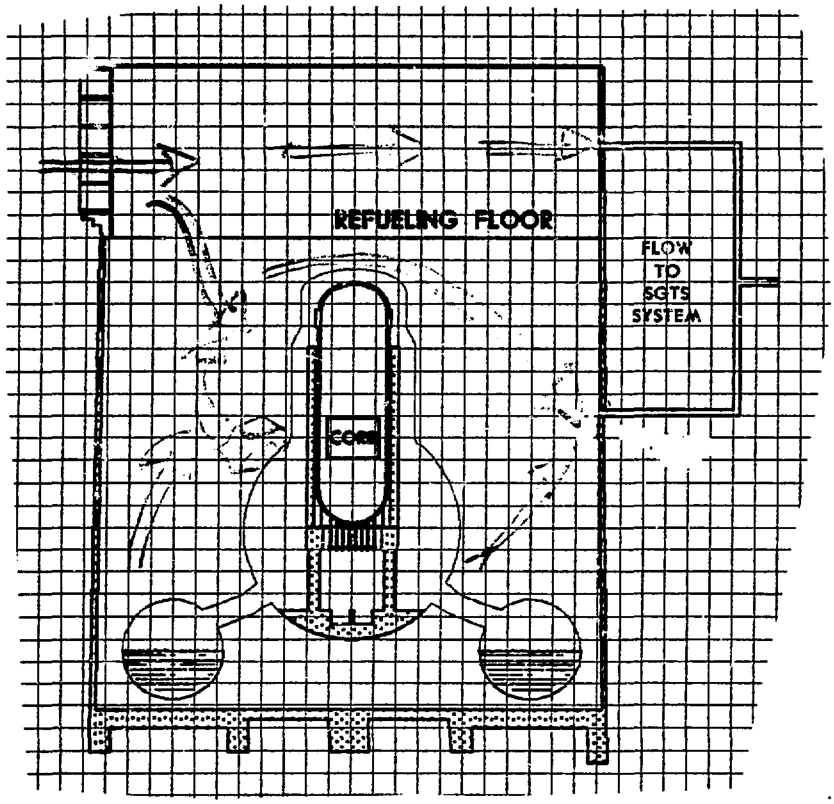

Fig. 8. Schematic representation of how the secondary containment flows would be distributed in the MK-I design. The reactor building-to-refueling floor and refueling floor-to-atmosphere blowout panels are assumed to have ruptured at the time of drywell failure.