CONF-890546--1

DE89 005338



# THERMALHYDRAULIC PROCESSES IN THE REACTOR COOLANT SYSTEM OF A BWR UNDER SEVERE ACCIDENT CONDITIONS

Stephen A. Hodge
Boiling Water Reactor Severe Accident Technology (BWRSAT) Program[1]
Oak Ridge National Laboratory

## INTRODUCTION

Boiling water reactors (BWRs) incorporate many unique structural features that make their expected response under severe accident conditions very different from that predicted in the case of pressurized water reactor accident sequences [1]. Automatic main steam isolation valve (MSIV) closure as the vessel water level approaches the top of the core would cause reactor vessel isolation while automatic recirculation pump trip would limit the in-vessel flows to those characteristic of natural circulation (as disturbed by vessel relief valve actuation). This paper provides a brief discussion of the BWR control blade, channel box, core plate, control rod guide tube, and reactor vessel safety relief valve (SRV) configuration and the effects of these structural components upon thermalhydraulic processes within the reactor vessel under severe accident conditions. The dominant BWR severe accident sequences as determined by probabilistic risk assessment are briefly described and the expected timing of events for the unmitigated short-term station blackout severe accident sequence at the Peach Bottom Atomic Power Station is presented.

## IN-VESSEL STRUCTURES

The internal structure of the BWR reactor vessel is shown schematically in Figure 1. Forced circulation flow during power operation is downward through the jet pumps into the lower plenum surrounding the control rod guide tubes and then upward through the core. As indicated, the normal reactor vessel water level is near the top of the steam separators. Steam generated in the core region passes from the separators through the dryers into the upper vessel head, where it enters one of the four main steam lines. It should be noted, however, that a bypass path around the dryers is opened whenever the vessel water level falls below the bottom of the dryer skirt. Thus, approximately 10% of the flow rising from the damaged core region under severe accident conditions would be expected to bypass the dryers.



---

[1]Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement DOE 1886-8045-2B with the U.S. Department of Energy under contract DE-AC05-84OR21400 with the Martin Marietta Energy Systems, Inc.

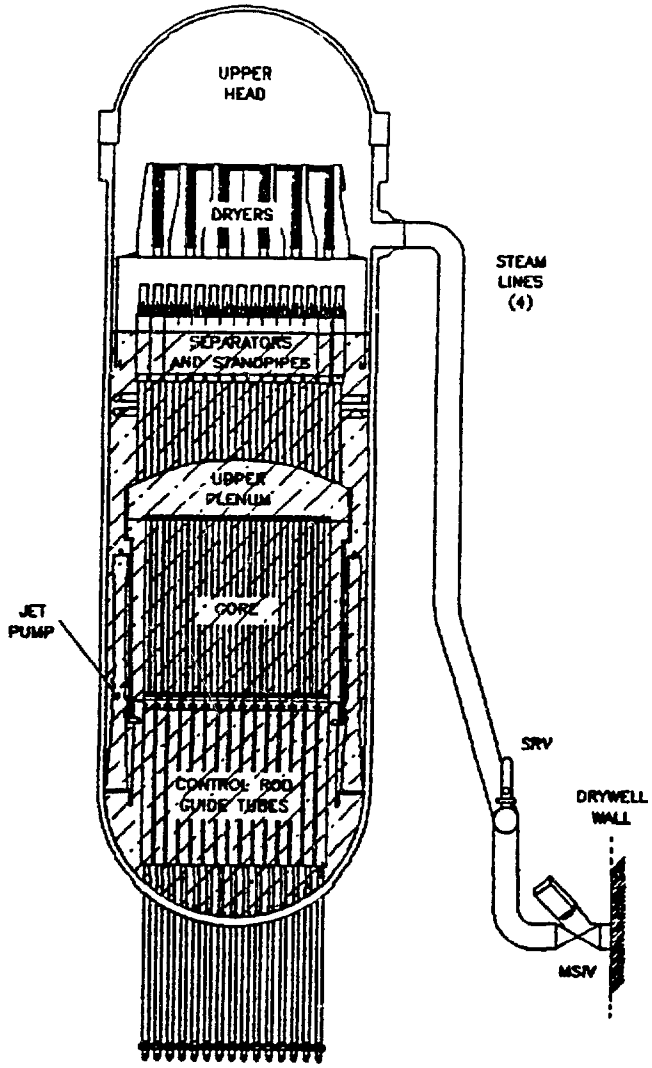


*FIGURE 1.* The major internal components of the BWR reactor vessel are the control rod guide tubes, the core, the upper plenum, the standpipes and steam separators, and the dryers. Shading indicates the volume of liquid water under normal operating conditions.

## SAFETY RELIEF VALVES

The BWR reactor vessel is protected from overpressure by multiple (eleven at Peach Bottom) safety relief valves (SRVs) mounted on the horizontal piping runs of the main steam lines between the vessel and the inboard MSIVs as indicated on Fig. 1. The SRV tailpipes (not shown) pass from the lower drywell into the wetwell and terminate in discharge devices (T-quenchers) well underwater in the pressure suppression pool. The SRVs actuate automatically to release steam from the reactor vessel whenever the vessel pressure reaches their setpoint (about $7.58 \times 10^6$ N/m$^2$), then reseat after a blowdown of about $52 \times 10^4$ N/m$^2$. Alternatively, as long as normal or battery power remains available, the SRVs can be remote-manually actuated by the control room operators as necessary to control the reactor vessel pressure within any desired lower band. The steam released from the reactor vessel via the SRVs is condensed in the pressure suppression pool and any fission products carried with the steam under severe accident conditions are subject to retention by the pool.

BWRs have an Automatic Depressurization System (ADS) designed to rapidly depressurize the reactor vessel under accident conditions should the vessel water level approach the top of the core (due to failure of the high-pressure injection systems) while the low-pressure injection systems are running and capable of injection into a depressurized vessel. ADS actuation under such conditions would rapidly permit the low-pressure injection systems to flood the vessel and terminate the accident sequence without core damage. However, for accident sequences that involve loss of the low pressure injection systems as well as the high pressure systems (such as Short-Term Station Blackout), automatic vessel depressurization would not occur. Under these circumstances, the BWR Owners Group Emergency Procedures Guidelines (EPGs) direct the control room operators to manually actuate the ADS when the reactor vessel water level has fallen to about one-third core height [2].

The effect of manual ADS actuation in accordance with the EPGs is shown in Figure 2. It is important to recognize that for the case without reactor vessel injection available, the EPG instructions with regard to manual ADS actuation have been carefully crafted by the BWR Owners Group to cause the control room operators to depressurize the reactor vessel precisely at a time when most of the core is uncovered and at elevated temperatures, but none of the core is at temperatures sufficient for runaway zirconium oxidation. Thus, the effect of the sudden depressurization is to cool the previously uncovered regions of the core. For Short-Term Station Blackout, calculations indicate that the hottest

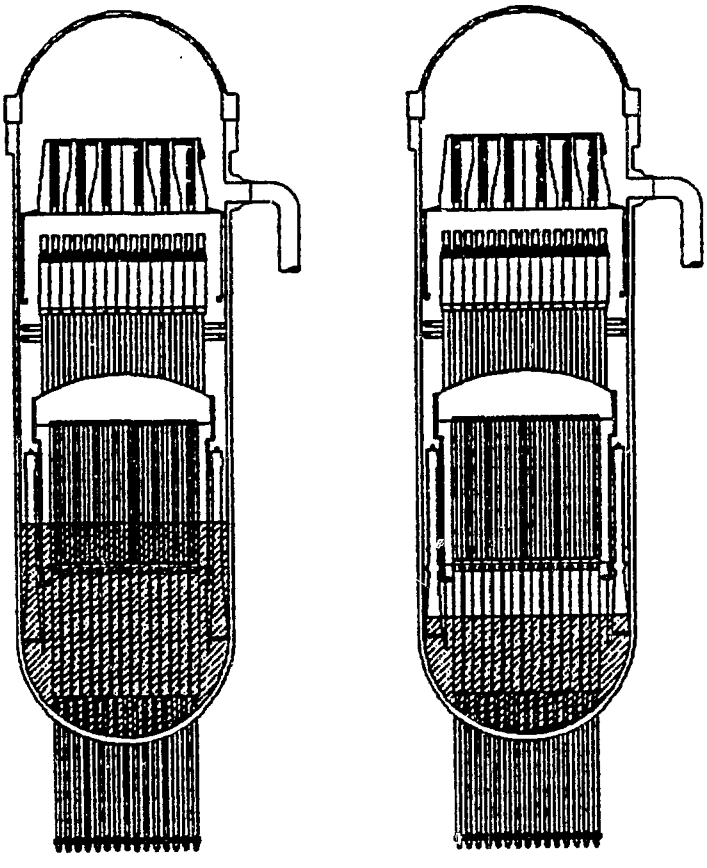

FIGURE 2. Flashing attendant to manual ADS actuation with vessel water level at 1/3 core height lowers the water level into the lower plenum, but does not uncover the jet pump exits.

regions of the uncovered core are cooled by about 300 K and that about 30 minutes of time are gained by this maneuver, before core temperatures again approach runaway zirconium oxidation levels [3].

## CONTROL BLADES

Multiple control blades (185 at Peach Bottom) are employed in the BWR design. The control blades are driven hydraulically upward into the core region from the reactor vessel lower plenum, where they are housed in control rod guide tubes that occupy approximately two-thirds of the lower plenum volume beneath the core. Each control rod guide tube supports four fuel assemblies in the core region. Each fuel assembly is comprised of 62 fuel rods and two water rods, and is cooled by an upward channeled flow that is guided by a square canister or channel box as shown on the right side of Figure 3. When the control blades are inserted into the core, they occupy the interstitial region between the channel boxes, as shown in the center of Figure 3.

The control blade neutron poison is $B_4C$ powder, stored within the neutron absorber rods located within the control blade sheaths as indicated on the left side of Figure 3. The presence of $B_4C$ powder within the core region has two important ramifications with respect to the BWR response under severe accident conditions. First, the stainless steel sheath and absorber rod walls have a lower melting temperature than the Zircaloy of the channel box walls and fuel cladding. Thus, severe accident calculations predict control blade structural failure and relocation while the fuel assemblies remain standing in the uncovered region of the core [3]. This early relocation of the control blade was also observed in the DF-4 experiment in the ACRR at Sandia National Laboratories [4] and clearly raises the question of recriticality should reactor vessel water injection capability be restored after partial core degradation has occurred. The effect is aggravated by the tendency of the $B_4C$ powder to form a lower-melting-temperature mixture with the surrounding stainless steel, causing an even earlier blade relocation than that which is predicted by consideration of the melting temperature of stainless steel (1672 K).

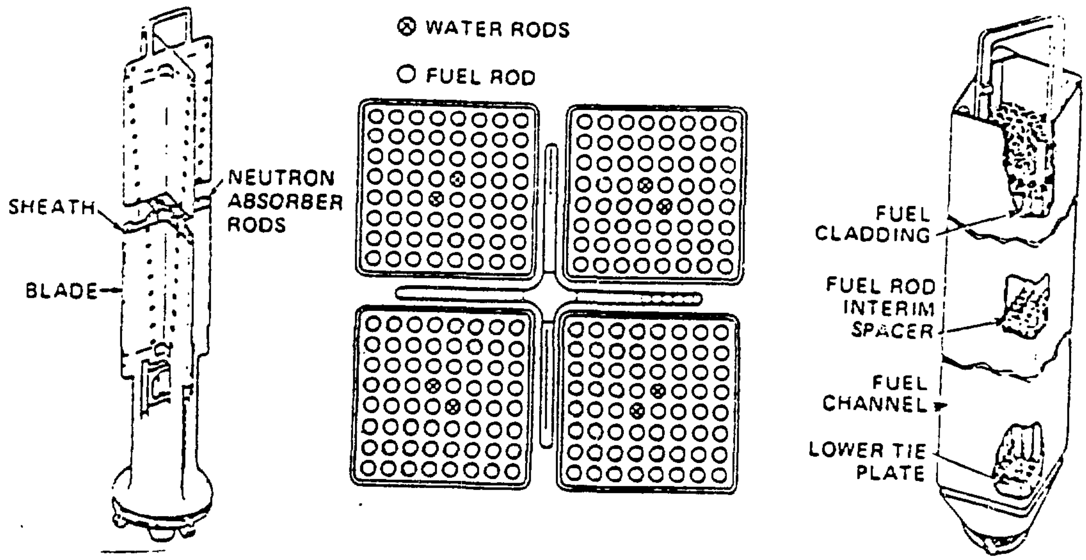


FIGURE 3. Arrangement of control blades and channel box fuel assemblies within the BWR core.

The second important ramification of the presence of $B_4C$ powder in the BWR core is that the reaction of $B_4C$ with steam produces, among other gases, methane [5]. To recognize the significance of this, it should be recalled that under accident conditions, reactor vessel gases would be passed to the pressure suppression pool via the SRVs. By this means, volatile fission products in particulate form released from the fuel after cladding failure would be carried with the flow of steam into an underwater release in the pressure suppression pool, where the steam would be condensed. The effectiveness of the pool in retaining CsI is well known [6], but the addition of methane from the $B_4C$-steam reaction would induce transformation of some of the iodine into the much more volatile methyl iodine form. Therefore, it is most important that the reaction of $B_4C$ powder with steam be considered in any BWR severe accident thermal hydraulic calculation that is to provide boundary conditions for a fission product transport analysis.

## CORE DEGRADATION AND RELOCATION INTO THE LOWER PLENUM

The chief result of the manual ADS actuation as directed by the EPGs for BWR accident sequences without reactor vessel injection available is that structural deformation and downward relocation of molten control blade, channel box, candling clad, and fuel (in that order) would occur within a totally dry region above the core plate [3]. What subsequently happens to the core plate is a matter of great importance in predicting the progression of a BWR severe accident.

As indicated in Figure 4, the BWR core plate separates the core region from the very large lower plenum, where the control rod guide tubes that hold the withdrawn control blades during power operation are located. It should be noted that the core plate does not support the core, but merely ensures proper lateral alignment of the upper portions of the control rod guide tubes. Clearly, if the relocating molten material falling on the dry core plate causes local core plate failure, then the molten debris would fall into the lower plenum and be quenched, forming an underwater debris bed (there is more than enough water in the BWR lower plenum to quench an entire molten core). On the other hand, if the core plate remains intact, then the BWR debris bed would be expected to form above the core plate, in a fashion similar to that observed in the Three Mile Island accident (PWR). Thus, it is most important to consider the fate of the core plate in BWR severe accident analyses.

## DOMINANT BWR SEVERE ACCIDENT SEQUENCES

The dominant BWR severe accident sequences leading to core melt are Station Blackout and Anticipated Transient Without Scram (ATWS). For Peach Bottom, the recently completed Probabilistic Risk Assessment (PRA) performed for the U.S. Nuclear Regulatory Commission (NRC) by the Accident Sequences Evaluation Program (ASEP) assigns 60% of the risk to Station Blackout, 31% to ATWS, and 9% collectively to all other possible accident sequences [7]. However, there are two forms of Station Blackout [8]. Long-Term Station Blackout implies maintenance of reactor vessel injection capability by means of steam turbine-driven systems until exhaustion of the plant batteries (6-8 hours after scram), whereas Short-Term Station Blackout implies independent failure of the steam turbine-driven injection systems and provides the most rapid progression of events to core degradation, reactor vessel failure, and fission product escape from

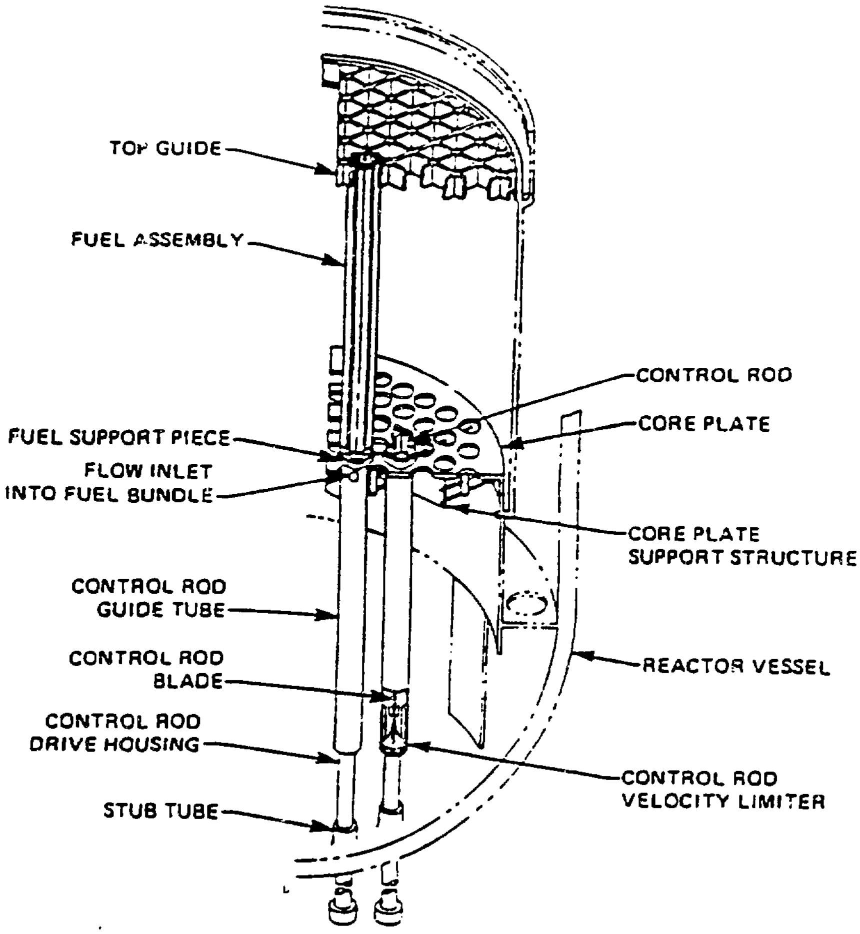


**FIGURE 4** The BWR core plate separates the core region from the reactor vessel lower plenum. The core fuel assemblies are supported by the fuel support pieces, the control rod guide tubes, the control rod drive housings, and the stub tube welds.

primary containment. It should also be noted that the reactor vessel would be depressurized during the periods of core degradation and bottom head failure in Short-Term Station Blackout whereas battery failure would preclude SRV actuation during these periods for the long-term case.

Severe accident analysis codes sponsored by the NRC for the purpose of providing specific models for representation of the sequence of events within a BWR reactor vessel under severe accident conditions include the APRIL code developed under joint sponsorship with ESEERCO at Rensselaer Polytechnic Institute [9], the BWRSAR code developed by the BWRSAT Program at Oak Ridge National Laboratory [10], and the BWR portion of the MELCOR code currently under development at Sandia National Laboratories. Close coordination has been maintained over the years between these BWR code development activities and it is intended that all of the capabilities of APRIL and BWRSAR will be available in MELCOR when it is released for general use and becomes the NRC flagship code for general severe accident applications (both BWR and PWR). In the meantime, the BWRSAR code is being extensively applied in BWR station blackout studies. In the next section, results of recent BWRSAR calculations of the short-term station blackout accident sequence are discussed.

## SHORT-TERM STATION BLACKOUT

The recent ASEP results assign 30% of the total risk of Peach Bottom core melt to the short-term station blackout accident sequence [7]. The estimated timing of events for this relatively fast-moving accident sequence as recently calculated with the BWRSAR code at Oak Ridge National Laboratory [3] is provided in Table 1. Since, by definition, all forms of reactor vessel injection are lost at the inception of the Short-Term Station Blackout, this calculation produces the shortest time to core uncovering of any accident sequence not involving ATWS or LOCA. Control room operator manual actuation of the ADS is assumed to be in accordance with the EPGs and occurs about 80 minutes after scram. It should be noted that core plate dryout is predicted to occur almost immediately after ADS actuation; however, because of the concomitant steam cooling of the uncovered regions of the core, the onset of debris relocation is delayed for some 3000 s. Since the core plate is dry when debris relocation does occur, the relocating molten material immediately induces local core plate failure.

TABLE 1. Timing of events for Peach Bottom short-term station blackout

| Event | Time after scram, s |
|---|---|
| Swollen level below top of core | 2 412 |
| ADS system actuation | 4 800 |
| Core plate dryout | 4 854 |
| Debris relocation begins | 7 944 |
| First local core plate failure | 7 962 |
| Central fuel column collapse | 13 368 |
| Lower plenum dryout | 15 294 |
| Bottom head penetration failure | 15 300 |

The establishment of natural circulation loops within the BWR reactor vessel after the vessel water level falls below the core plate is prevented by the action of the SRVs. In Short-Term Station Blackout, the valves associated with the ADS (5 at Peach Bottom) are opened when the system is manually actuated; subsequently the prominent streamlines within the reactor vessel are directed toward the main steam lines and release to the pressure suppression pool via the SRVs.

The very pronounced falloff of the radial power profile in the outer regions of the BWR core also has significant effect in differentiating the events of a BWR severe accident sequence from those that might occur in a PWR under similar circumstances. Indeed, in the current calculation, the outer 20% of the Peach Bottom core is predicted to be undamaged at the time that collapse of the central fuel column occurs. Because of the reduced power at the core periphery, events in the outer regions of the BWR core would lag far behind the situation in the inner core. This time lag in the onset of fuel degradation is sufficient that collapse of the outer regions is predicted by BWRSAR only after the central core has relocated into the vessel lower plenum, boiled off the water there, and caused thermally-induced failure of the control rod guide tubes that support the outer region fuel assemblies.

The initial release of fission products from BWR fuel under severe accident conditions would occur at higher cladding temperatures than for PWR fuel under similar conditions [12]. This is because the pressure differential across the cladding would be much less for the BWR. The internal BWR fuel rod pressure is generated by the heatup and expansion of the initial helium fill gas ($30.4 \times 10^4$ N/m$^2$ at room temperature for an $8 \times 8$ P fuel rod) plus the fission gas released to the rod plenum and void spaces during the previous periods of normal reactor operation. If the reactor vessel remains pressurized as in Long-Term Station Blackout, then the internal fuel rod pressure would never exceed the vessel pressure and cladding failure would not occur until cladding temperature had increased to the range of 1450-1550 K, so that temperature-induced embrittlement causes failure by means of cracking or fragmentation.

For Short-Term Station Blackout, however, the reactor vessel would be depressurized by actuation of the ADS and the internal fuel rod pressure would exceed the reactor vessel pressure by about $106 \times 10^4$ N/m$^2$ at the time of cladding failure. With this pressure differential, fission product release from fuel would be expected to begin within the core whenever the maximum cladding temperature reached 1280 K. This occurs at time 5 994 s for the accident sequence described by Table 1.

The pathway from the reactor vessel for fission products released from fuel involves passage through the upper plenum and the separators and standpipes, whose location within the reactor vessel is identified in Figure 1. Most of the escaping fission products would also pass through the dryer assembly, although there is a small bypass path around the dryers at the low water levels associated with severe accidents, and approximately 10 percent of the flow would follow this bypass to the main steam lines.

Since the upper reactor vessel structures have large surface areas, it is important to consider the potential that they offer for fission product deposition. This depends to a large extent on their temperatures during the periods that fission products are being carried through these structures as part of the gas streams flowing toward the main steam lines and the open SRVs. Experience with BWR severe accident calculations at Oak Ridge has shown that the general range of upper structure temperatures is accident sequence-dependent. For Short Term Station Blackout, these temperatures are predicted to remain relatively low, increasing from approximately 775 K to 1425 K during the period of rapid release of volatile fission products, which occurs between time 7 500 and 11 700 s for the accident sequence described by Table 1.

## SUMMARY

It has been possible in this short paper to identify and briefly discuss only the major unique characteristics of BWR severe accident response that would affect considerations of in-vessel fission product transport. BWRs are very different from PWRs and the important differences in structure, modes of pressure control, and emergency operating procedures must be considered in performing severe accident and fission product transport analyses. In particular, it is absolutely necessary that the effects of SRV actuation upon the heatup of the core and upon the establishment of flow patterns within the reactor vessel be included in any BWR severe accident analysis that is intended to be realistic. Furthermore, for fission product transport calculations, the reaction of the $B_4C$ control rod powder with steam and the attendant production of methane must also be considered.

Of the dominant BWR accident sequences leading to core melt, Short-Term Station Blackout involves the shortest period of time between reactor shutdown and the onset of core degradation and fission product release from fuel. For this reason, much of the NRC-sponsored severe accident analytical effort for BWRs has been focused upon this important accident sequence. The BWR-specific models developed at Rensselaer Polytechnic Institute [9] and Oak Ridge National Laboratory [10] as necessary for use in ongoing severe accident studies have all been provided to the formal code development programs at Sandia National Laboratories. In the meantime, the best available estimate of the sequence of events for Station Blackout is believed to be achieved by application of the BWRSAR code at Oak Ridge.

## REFERENCES


1. Greene, S. R., Realistic Simulation of Severe Accidents in BWRs — Computer Modeling Requirements, NUREG/CR-2940, ORNL/TM-8517, April 1984.

2. The General Electric Company and the BWR Owners Group, EMERGENCY PROCEDURES GUIDELINES, BWR 1 through 6, Revision 3, December 1982.

3. Hodge, S. A., Hyman, C. R., and Ott, L. J., Primary Containment Response Calculations for Unmitigated Short-Term Station Blackout at Peach Bottom, NUREG/CR-5317, ORNL/TM-11057, Forthcoming.

4. Gauntt, R. O., and Gasser, R. D., The DF-4 BWR Control Blade/Channel Box Experiment, NUREG/CR-4671, SAND86-1443, Forthcoming.

5. Beahm, E. C., Wichner, R. P., and Weber, C. F., Chemical Factors Affecting Fission Product Transport in BWR Severe Accidents, *Proc. Twelfth Water Reactor Safety Research Information Meeting, Gaithersburg*, Vol. 3, pp. 109-121, 1983.

6. Wichner, R. P., Weber, C. F., Hodge, S. A., Beahm, E. C., and Wright, A. L., Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry, NUREG/CR-3617, ORNL/TM-9028, August 1984.

7. Kolaczkowski, A. M., Cramond, W. R., Daniel, S. L., Maloney, K. J., Sype, P. T., and Wheeler, T. A., Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events, NUREG/CR-4550 Rev. 1, Vol. 4, SAND 86-2084, Forthcoming.

8. Greene, S. R., Harrington, R. M., Hodge, S. A., Hyman, C. R., Ott, L. J., Sozer, A., Station Blackout at Browns Ferry Unit One, NUREG/CR-4571, ORNL/TM-10011, Forthcoming.

9. Kim, S. H., Kim, D. H., Koh, B. R., Pessanha, J., SiAhmed, El-k, Podowski, M. Z., Lahey, R. T., The Development of APRIL.MOD2 — A Computer Code for Core Meltdown Accident Analysis of Boiling Water Nuclear Reactors, NUREG/CR-5157, ORNL/Sub/81-9089/3, July 1988.

10. Hodge, S. A., and Ott, L. J., Boiling Water Reactor Severe Accident Response (BWRSAR) Code Description and Assessment, NUREG/CR-5318, ORNL/TM 11058, Forthcoming.

11. Haskin, F. E. and Webb, S. W., Development and Status of MELCOR Code, *Proc. Fourteenth Water Reactor Safety Research Information Meeting, Gaithersburg*, 1986.

12. Wichner, R. P., Davis, W., Weber, C. F., Hodge, S. A., Lorenz, R. A., Mitchell, A. D., Station Blackout at Browns Ferry Unit One — Iodine and Noble Gas Distribution and Release, NUREG/CR-2182, Vol. 2, ORNL/NUREG/TM-455/V2, pp. 29-35, August 1982.