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CONTAINMENT VENTING AS A MITIGATION TECHNIQUE
FOR BWR MARK I PLANT ATWS

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ABSTRACT

Containment venting is studied as a mitigation strategy for preventing or delaying severe fuel damage following hypothetical BWR Anticipated Transient Without SCRAM (ATWS) accidents initiated by MSIV-closure, and compounded by failure of the Standby Liquid Control (SLC) system injection of sodium pentaborate solution and by the failure of manually initiated control rod insertion. The venting of primary containment after reaching 75 psia (0.52 MPa) is found to result in the release of the vented steam inside the reactor building, and to result in inadequate Net Positive Suction Head (NPSH) for any system pumping from the pressure suppression pool. CONTAIN code calculations show that personnel access to large portions of the reactor building would be lost soon after the initiation of venting and that the temperatures reached would be likely to result in independent equipment failures. It is concluded that containment venting would be more likely to cause or to hasten the onset of severe fuel damage than to prevent or to delay it.

Two alternative strategies that do not require containment venting, but that could delay or prevent severe fuel damage, are analyzed. BWR-LTAS code results are presented for a successful mitigation strategy in which the reactor vessel is depressurized, and for one in which the reactor vessel remains at pressure. For both cases the operators are assumed to take action to intentionally restrict injected flow such that fuel in the upper part of the core would be steam cooled. Resulting fuel temperatures are estimated with an offline calculation and found to be acceptable.

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INTRODUCTION

The calculations reported here were undertaken to determine if containment venting has a role in preventing or delaying severe fuel damage following MSIV closure initiated ATWS. Only ATWS from full power, with total failure to scram, is considered. To make containment venting a relevant issue, it is necessary to assume that the control rods cannot be scrambled or driven into the core by operator action, and that the SLC system injection of sodium pentaborate solution fails upon demand. If either manual control rod insertion or SLC initiation were successful, the core would be shut down before threatening pressures could build in the primary containment. Calculations previously performed at ORNL show that the only operator action necessary for successful ATWS mitigation would be to initiate the SLC system, as presently required by Browns Ferry and Peach Bottom plant emergency procedures (Ref. 1).

Both automatic and operator initiated protective features are provided in the BWR-4 design to combat the consequences of failure to scram. In the case of sudden closure of all MSIVs, an automatic trip of the reactor vessel recirculation pumps on detected high reactor vessel pressure would occur within seconds. This would invoke the negative void feedback characteristics of U.S. domestic BWRs to reduce the power level from 100% to about 30% without the insertion of any control rods. Flow through the safety relief valves (SRVs) would rapidly decrease vessel water level until automatic initiation of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) high pressure injection systems stabilizes downcomer water level at about five ft (1.5 meters) above the level of the top of active fuel. Reactor vessel pressure would be controlled in the neighborhood of 1100 psia (7.6 MPa) by automatic actuation of the SRVs.

Figure 1 shows schematically what would happen shortly after the initiation of the HPCI and RCIC systems, about two minutes after the beginning of the accident. With both the HPCI and RCIC systems injecting at their full flow, and with the small injection of the control rod drive (CRD) hydraulic system continuing, the core would be covered and adequately cooled. The steam generated in the reactor vessel, blocked by the shut MSIVs from reaching the main condenser, flows through the SRV tailpipes to the T-quencher distribution devices located about 10 ft (3 m) under water in the pressure suppression pool. The large quantity of water in the suppression pool condenses the steam escaping into the pool through the many small exit holes of the T-quenchers. The situation is, for the time being, not extremely threatening, except for the steady buildup of suppression pool temperature. When the suppression pool temperature reaches

110 F (317 K), the operators would, by procedure, initiate SLC system injection of sodium pentaborate solution into the reactor vessel; the reactor would be shut down within about twenty minutes, and the operators could then provide for final resolution of the problem by establishing decay heat removal, either by reopening the MSIVs and routing steam to the main condenser, or by utilizing the Residual Heat Removal (RHR) system in the suppression pool cooling mode.

By defining assumption, we are concerned with what happens if the SLC system fails and the operators cannot obtain an alternative scram of the control rods. In this case, the suppression pool temperature would continue to increase steadily. As the vapor pressure of the suppression pool water increases, steaming from the surface of the pool begins to increase the pressure of the wetwell atmosphere; communication by the wetwell to-drywell vacuum breakers allows the parallel pressurization of the drywell atmosphere. If continued indefinitely, the heatup of the pool and concomitant containment pressurization would eventually lead to failure of the primary containment boundary. If the containment failed in a catastrophic mode, the release of steam accompanying the depressurization might result in the failure of the reactor vessel injection systems, leading to the ultimate uncovering and severe damage of the reactor fuel.

Procedures currently in place at the Peach Bottom BWR require containment venting if containment pressure reaches 75 psia (0.52 MPa). This degree of pressurization is consistent with a suppression pool temperature of about 302 F (423 K). Condensation of the steam equivalent of about 11 full power minutes of steam dump to the suppression pool is required to reach this temperature. The actual time required to deposit this much steam in the suppression pool would, of course, depend on operator actions, the most important of which would be control of vessel injection. In a worst-case ATWS without sodium pentaborate injection, the reactor power is very directly tied to the rate of injection flow by the inherent mechanism of negative void reactivity feedback. If containment venting could stem the increase in containment pressure without causing the loss of all vessel injection, severe fuel damage would be prevented and the accident would be successfully mitigated.

Since MSIV closure initiated ATWS is an improbable accident, it might be argued that it is a waste of time to consider potentially beneficial operator actions for the even more improbable case of an ATWS that is further compounded by additional failures, e.g., failure of the SLC system. However, it is just such very improbable accident sequences that dominate the risk profile generated by recent BWR PRAs and by the Accident Sequences Evaluation Program (ASEP) (Ref. 2). Therefore, any mitigation strategy that would enable plant operators to delay or

avert severe fuel damage in these cases is important to the safety of the public, and should be considered.

Most of the reported calculations were performed for a study completed last spring at ORNL of the general utility of containment venting with respect to ATWS (Ref. 2). Both the Peach Bottom and Browns Ferry BWRs were used as example plants. The previous study identifies specific differences among these plants that can affect plant response. The ground rule is: if containment venting can be shown to delay containment failure until three hours or more after ATWS initiation, the venting operation can be termed successful. The rationale behind this success criterion is that this much delay would allow sufficient time for repair of the SLC system and for the operators to achieve some other alternative shutdown of the reactor. Even with such a loose success criterion, it was found that containment venting would be of use only for a few very narrowly defined instances. The overall conclusion of the previous study is that containment venting for ATWS sequences prior to fuel damage would, in general, do more harm than good. The present work expands on the previous study to offer two alternative mitigation strategies that would avoid the need for containment venting in any case.

Several different calculational techniques are used to estimate the effects of containment venting and to examine the efficacy of the two alternative mitigation strategies. Venting flow and ductwork pressures are calculated by simple computer assisted hand calculations. The effect of steam release on the reactor building environment is calculated by the CONTAIN code (Ref. 3). The overall sequence development is simulated by the BWR-LTAS code (Ref. 4). The effect of partial core uncover, necessitated by the alternative (non-venting) mitigation strategies, is shown to be acceptable by a simple computer program developed to estimate the fuel temperatures under steam cooling conditions.

CONSEQUENCES OF CONTAINMENT VENTING

The BWR Mark I primary containment is located within the reactor building, as illustrated on Fig. 2. The reactor building provides secondary containment protecting against the accidental release of radioactivity to the environment. Many of the safety related systems, instrumentation, power supplies and controls are located in the reactor building. The reactor building is protected from overpressure by blow out panels installed in the walls of the refueling bay to provide a relief path directly to the environment. At Browns Ferry, there are additional blowout panels separating the reactor building and the refueling bay. At Peach Bottom, the refueling bay of each unit is always in

communication with its reactor building by the large, normally open, fuel cask hatchway. A secondary relief path from the reactor building is provided at both plants by blow-out panels in the steam vault; these panels relieve to the turbine building.

In this section it is shown that containment venting under ATWS conditions would release essentially all the vented discharge into the reactor building atmosphere. The results of CONTAIN code calculations are utilized to assess the effects of the steam release on the composition and temperature of the atmosphere within the reactor building. In addition, an analysis is provided of the ability of the RHR or core spray systems to pump from the suppression pool after the initiation of venting.

The means of venting the primary containment is provided by four 18-in. lines (two for the wetwell and two for the drywell) that connect the primary containment to the reactor building ventilation system. (The wetwell and drywell are also each supplied with a 3 in. venting line, but their capacities are too low to be of interest for ATWS accident sequences for which relatively large flows of steam would have to be relieved.) During power operation, the inert nitrogen atmosphere of the primary containment is prevented from escaping by a series combination of two closed butterfly isolation valves in each vent line, which can be opened from the main control room. The design pressure of the vent lines is 70.7 psia (0.49 MPa) from the point of connection to the drywell or wetwell to the downstream side of the outer of the two butterfly isolation valves.

Conventional sheet metal ductwork is utilized downstream of the butterfly valves. If a high drywell pressure signal were in effect, as it would be in the type of ATWS scenario considered here, automatic interlocks would prevent the 18-inch valves from opening on command. Before these valves could be opened, it would be necessary for an auxiliary operator to connect two specific sets of terminals in a cabinet in the cable spreading room (Ref. 5) to defeat the automatic interlocks. The valve discs are prevented from opening more than 55 degrees by mechanical stops.

In order to assess the effect of venting, an example case will be considered. Presume that an ATWS has taken place, and that the operators have not been able to initiate SLC system injection of sodium pentaborate, insert the control rods, open the closed MSIVs, or establish suppression pool cooling. Consequently, the continued SRV flow from the reactor vessel has increased the suppression pool temperature to about 302 F (423 K); the concurrent increase of primary containment pressure would be to the 75 psia (0.52 MPa) procedural limit requirement for containment venting. Suppose that the operators have decided to vent by opening the two 18 in. vents attached to

the wetwell. The BWR LTAS calculation used to set the stage for this example predicts that it would take about 57 minutes for the accident to reach this juncture. At the time of venting, the core spray low pressure injection system is pumping from the suppression pool to the reactor vessel.

The rate of venting would be limited because sonic flow conditions would develop at the point of minimum flow area of the 18-inch butterfly valves, which would be opened to the 55 degrees maximum-open position. The discharge would be mostly steam; initially, some non-condensable gases (primarily nitrogen) would be present, but continued steam discharge from flashing of the suppression pool would soon flush the non-condensables from the primary containment.

Figure 3 shows the situation at the beginning of the venting process, with 82 lb/s (37 kg/s) of steam discharge through each of the two opened 18-in. vent lines. Even though choking occurs at the minimum flow area of the butterfly valves, the discharge flow is sufficient to elevate the internal pressure of the vent line to 30 psia (0.21 MPa) at the section of duct immediately downstream from the outer valve, where the high pressure 18-in. pipe discharges into the conventional low-pressure ductwork of the reactor building ventilation system. The low-pressure ductwork would fail at this point. All the steam discharged by opening the 18-in. containment vents would then escape into the reactor building atmosphere.

The primary containment pressure would not rapidly decrease after the two wetwell vent lines are opened. There are two reasons for this. First, a large quantity of energy (equivalent to about 11 full power minutes) must have been deposited in the suppression pool to heat it to 302 F (423 K). For primary containment pressure to decrease, this stored energy must be released by the mechanism of flashing of the pool water, and the steam thus produced must be released via the open vents. This is a relatively slow process.

The second reason that the containment pressure does not decrease rapidly after venting has to do with the reactor power level and its corresponding steaming rate. Immediately before venting, the reactor is operating at about 12% of full power, and injection is being provided by two core spray pumps. The resulting steam production of about 425 lb/s (193 kg/s) is discharged to the suppression pool via the SRV T quenchers. Steaming from the suppression pool actually exceeds the capability of the two open 18 in. vents; thus, containment pressure continues to increase after venting, but at a slower rate. Before venting, the suppression pool is only slightly subcooled; after venting, the small amount of subcooling is lost as the rate of pool temperature increase is the same, while the

containment pressure increases at a lower rate than before. As a result, the NPSH at the core spray pump inlets becomes inadequate, and the pumps can no longer function. With only the CRD hydraulic system injecting at about 105 gpm ($0.00662 \text{ m}^3/\text{s}$), the reactor vessel water level decreases rapidly. The core becomes almost completely uncovered, and the power level of the mostly unmoderated fuel collapses to decay heat levels. At this point, the containment pressure begins decreasing. Thus, core cooling would become inadequate as a result of the venting; this must be considered as a penalty of containment venting for ATWS sequences.

To illustrate the effect of containment venting on pump NPSH, the pump inlet NPSH calculated by BWR-LTAS for the Browns Ferry RHR system after the opening of the two 18-in. vents is plotted in Fig. 4. The illustration is made for the RHR pumps because the Tennessee Valley Authority (TVA) has performed actual plant tests at Browns Ferry to determine how far below the manufacturer's minimum recommended value the NPSH can be degraded without harming the pump through excessive vibration or by experiencing gross degradation of pump output (Ref. 6). The figure clearly shows that pump performance cannot be assured during venting. Any pump taking suction on the suppression pool during venting would experience a similar problem, its magnitude depending on the design of the pump.

The conditions within the Browns Ferry Unit 1 and the Peach Bottom Unit 2 reactor buildings for a period of one hour following the initiation of wetwell venting were calculated using the CONTAIN code. The rate of steam escape through the two open 18-in. vent lines, calculated using the BWR-LTAS code, was input to CONTAIN, along with geometric and other information about each reactor building. The total steam flow through the two 18-in. vent lines begins at about 164 lb/s (74 kg/s), and decreases slowly to about 110 lb/s (50 kg/s) at the end of the one hour period. In order to fully depressurize the containment, the vents would have to remain open much longer.

Selected CONTAIN code results for the Browns Ferry Unit 1 reactor building response during the first hour of containment venting are exhibited on Fig. 5. The atmosphere temperature is plotted for the four control volumes that represent the reactor building rooms with floor levels at the 565 ft, 593 ft, 621 ft, and 639 ft elevations. Very soon after initiation of venting, the vent lines would fail at or near the transition from high pressure pipe to low pressure ductwork, allowing all of the 164 lbs/s (74 kg/s) of steam discharge to escape into the atmosphere of the 565 ft elevation floor level. The temperature of this control volume would rapidly increase beyond the 165 F (348 K) temperature at which melting of fusible links would initiate the fire protection system sprays.

Initiation of the sprays slows, but does not reverse, the temperature increase of the atmosphere at the 565 ft elevation. Even if the spray efficiency were 100%, the very substantial 396 lbs/s (180 kg/s) spray flow into the 565 ft elevation would be thermodynamically able to condense only about half of the total steam escaping into the reactor building from the ruptured vent ductwork. Therefore, the steam release is not contained and reactor building internal pressure increases rapidly, rupturing blowout panels at the 565, 593, and 621 ft elevations in the fuel cask shaft that leads to the refueling bay.

Fire protection sprays are also provided for the 593 ft elevation, but these sprays, like those on the floor below, are overwhelmed by the steam flowing into their coverage area. The temperature of the 593 ft control volume atmosphere exceeds 200 F (367 K) after about 25 min.

Although no sprays are present, the temperature of the atmosphere at the 621 ft elevation increases slowly because this control volume benefits from the steam condensation taking place in the two floor areas directly below. Nevertheless, the temperature of the atmosphere at this floor level increases to above 150 F (339 K) within about 23 min.

The atmosphere temperature at the 639 ft elevation does not increase significantly throughout the venting. There are no blow-out panels at this elevation, so it is not in the direct flow path for steam released by the ruptured ventilation ductwork. (The atmosphere temperature of the basement corner rooms would be similarly low for basically the same reasons.) Atmospheric compositions in the reactor building, after one hour of containment venting, would range from almost pure water vapor at the 565 ft elevation to about 5% water vapor (95% air) at the 639 ft elevation.

The results discussed above and illustrated by Fig. 5 provide evidence that containment venting would prevent personnel access to most of the reactor building. A good rule of thumb is that unprotected personnel access would not be practical in a steam-air environment in which temperature exceeds 150 F (339 K). Accordingly, access to the 565 ft and 593 ft elevations would be denied almost immediately. Temperature at the 621 ft elevation would exceed 150 F (339 K) after about 23 minutes. Access to the 639 ft elevation should be feasible throughout the venting.

The loss of personnel access to almost all of the reactor building would severely hamper efforts to investigate or repair the failures of the reactor protection system or of the SLC system. For example, the CRD scram discharge valve and the CRD hydraulic control units, both of which are necessary for reactor scram, are located at the 565 ft elevation. Besides denying

personnel access, the escape of steam accompanying containment venting would be likely to cause additional, independent, equipment failures. The reactor building floor elevations that receive the most severe steam environment have vital equipment associated with many essential systems, including the plant protective system, and the emergency core cooling systems. Not all of this equipment is qualified for a hot steam environment. Electrical equipment qualified for a steam environment must be sealed against steam ingress, and must be capable of withstanding the accompanying high temperatures. Actuation of the building fire protection sprays would pose a moisture threat to any system located in the active zone of the spray heads on the 565 ft or 593 ft elevations.

It must be concluded that the ability of the plant operators to maintain reactor injection and to monitor important reactor parameters would be compromised as a result of containment venting. This applies to both the Browns Ferry and the Peach Bottom plants, although the steam escape would be into the torus room at Peach Bottom, and there are other significant differences in the two reactor buildings (discussed in Ref. 2). For example, there is no floor-wide system of fire protection sprays on any floor level at Peach Bottom. The Peach Bottom atmosphere heats up more rapidly than would that of Browns Ferry. However, there is an open fuel cask hatchway at Peach Bottom that would allow the steam released by the containment venting to draft upward, from the 135 ft elevation (the Peach Bottom equivalent of the 565 ft elevation at Browns Ferry) to the refueling bay. Therefore, the parts of the floor areas in the upper stories of the Peach Bottom reactor building that are farthest away from the equipment shaft would see lower temperatures. However, the overall conclusion with regard to the effect of the steam environment is the same: personnel access would be denied in large areas of the reactor building and additional equipment failures would be probable due to overheating or due to the condensation of steam inside electrical or electronic equipment.

MITIGATION WITHOUT CONTAINMENT VENTING

In the unlikely event of an MSIV-closure-initiated ATWS with failure of control rod insertion and failure of the SLC system, the operators would need to take action to slow or prevent the buildup of primary containment pressure if failure of the primary containment pressure boundary and the associated threat of severe core damage is to be averted. Containment venting would certainly provide some control over containment pressure, but the penalty, as demonstrated above and by previous ORNL work, would likely be the loss of reactor vessel injection and the interruption of ongoing recovery activities in the reactor.

building. Alternative mitigation schemes not involving containment venting, therefore, appear to have a better chance of achieving the goal of preventing severe fuel damage.

Two non-venting strategies are discussed in this section, a low reactor vessel pressure case and a high reactor vessel pressure case. Although these two example cases seem outwardly very different, they succeed for the same reasons. Reactor power level is reduced to the neighborhood of 6% of full power (about 200 MW) by limiting injection flow and the suppression pool cooling mode of the RHR system is used to remove energy from containment. Primary containment pressure is controlled, and this gives the operators more time in which to achieve an alternative shutdown.

Both cases involve some degree of uncovering of fuel in the upper part of the core, but the resulting heatup of the fuel is shown to be acceptable for the high pressure case, which experiences the more severe fuel heatup. Therefore, it can be assumed that partial core uncover under these conditions would be acceptable for the low pressure case as well. For both cases, the recirculation pumps trip shortly after the MSIVs shut, and the HPCI system actuates automatically. The reactor power stabilizes at about 30% of full power within several minutes after accident initiation with the HPCI system injecting at its full 5000 gpm (0.32 m³/s) capacity. Reactor vessel downcomer water level is below the setpoint for automatic HPCI injection, about 5 ft (1.5 m) above the top of the active fuel.

Low Pressure No-Venting Case

The low reactor vessel pressure case is patterned after an ATWS success path first noticed by Richard Deem of the New York Power Authority in calculations performed for the Fitzpatrick BWR. Minimal operator control is required for this success path: operators must establish pressure suppression pool cooling, prevent the RHR system suppression pool cooling flow from being diverted to reactor vessel injection, and must turn off two of the four core spray pumps after the pumps automatically start on low vessel water level. Turning off two of four core spray pumps is necessary to limit injected flow, although it is not a specific part of present EPGs for ATWS. The BWR Emergency Procedure Guidelines (EPGs) specify that almost all other injection systems of adequate flow capacity would be preferable to core spray for use in ATWS mitigation. Therefore, for this case, we must implicitly assume that the preferred injection systems have failed or are otherwise not available.

The BWR-LTAS results for the low pressure case based upon Browns Ferry Unit 1 are plotted in Figs. 6 - 11. The operators

begin to control (reduce) HPCI system injected flow (Fig. 7) after 5 min in an attempt to lower downcomer water level (Fig. 8) to near the top of active fuel, as specified in the current BWR Owners Group EPGs for ATWS. This results in the reduction of reactor power (Fig. 6) to about 20% of full power. Suppression pool temperature (Fig. 10) increases steadily; when it exceeds 200 F (367 K) at 19 min, the HPCI system, which is taking suction on the suppression pool at this point, is assumed to fail. [The HPCI turbine lubricating oil is cooled by the pumped water, which, by manufacturer's recommendation, should not be allowed to exceed 140 F (333 K).] Reactor vessel pressure (Fig. 9) is controlled by automatic cycling of the SRVs throughout the accident sequence.

After the HPCI injection fails, downcomer water level decreases rapidly, temporarily uncovering a large part of the core and introducing negative void reactivity to shut the core down to decay heat power. Upon expiration of its 2-min timer (21 min after the beginning of the accident), the Automatic Depressurization System (ADS) automatically opens the SRVs (six for Browns Ferry and five for Peach Bottom) associated with the ADS. All four pumps of the core spray system start automatically on detected low water level; two of the four pumps are allowed to continue running. The RHR system pumps are employed in pressure suppression pool cooling.

Core spray injection begins at 22 min as the ADS operation brings reactor vessel pressure to below the shutoff head of the core spray pumps. Due to the resulting rapid refilling of the reactor vessel, there are two initial power excursions to 60% and 44%, respectively. Reactor power soon stabilizes at an average of about 13% of full power. Reactor vessel pressure stabilizes near the shutoff head of the core spray pumps, thereby preventing these large capacity pumps from providing too much injected flow and initiating additional power excursions.

When drywell pressure (Fig. 11) reaches 78 psia (0.54 MPa), four of the six open SRVs go shut as a result of insufficient control air pressure. As illustrated by Fig. 12, control air pressure must work against drywell pressure to open or to hold open an SRV. The net effect of the SRV closings at 80 min is that reactor vessel pressure increases slightly and forces the core spray injection to equilibrate at a lower flow; this causes the reactor core to operate at a lower power.

The serendipitous balance between reactor power and pressure suppression pool cooling that automatically comes about in the low pressure case without venting is due to several complex interrelated system characteristics. As the increasing drywell pressure comes within a certain range of the available control air pressure, the SRVs, heretofore held open only by control air

pressure, begin to close. The steaming rate from the reactor vessel is correspondingly reduced, slightly increasing reactor vessel pressure. Increased reactor vessel pressure reduces the rate of reactor vessel water injection by the low-pressure systems, which in turn reduces reactor power. With a lower rate of steam discharge from the reactor vessel to the suppression pool through the SRVs, the continued heat removal by the RHR system heat exchangers reduces suppression pool temperature. Lower drywell pressure follows, and soon the available control air pressure is again sufficiently above the drywell pressure so that the SRVs reopen. The cycle repeats, core thermal power averages about 6%, and this energy is removed from the suppression pool, whose bulk-averaged temperature remains in the neighborhood of 300 F (422 K) by action of the RHR system heat exchangers.

The calculated peak drywell pressure for the low pressure no-venting case is lower for Browns Ferry (80 psia) than for Peach Bottom (103 psia). This is simply because the pressure of the isolated stored drywell control air volume at Browns Ferry would decrease with time, whereas the control air system at Peach Bottom would be vibrant throughout the ATWS accident sequence. [There is an automatic shutdown of the drywell control air compressors when drywell pressure exceeds 2.45 psig at Browns Ferry, such that after about 24 min into the ATWS, the control air pressure would begin decaying at a rate of about 10 psi/hour (0.69 MPa/s). There is no such failure of the drywell control air system at Peach Bottom; therefore, the Peach Bottom drywell control air pressure should remain approximately constant during the first three hours of the ATWS accident sequence.]

One of the most important plant-specific equipment differences with respect to impact upon plant response to ATWS is the installed reactor vessel pressure relief system. The Peach Bottom plant employs the three-stage Target Rock safety relief valves, which differ significantly from the two-stage Target Rock valves installed at Browns Ferry with respect to the ability of the valves to remain open under the impetus of control air pressure in the face of increasing drywell pressure. In the two stage Target Rock design (Browns Ferry), the reactor vessel-drywell pressure differential and the control air-drywell pressure differential are applied in tandem to reposition the pilot valve and cause the main valve to open. In the three-stage design (Peach Bottom), these two pressure differentials act in opposition. Therefore, these two valve designs respond differently in the face of steadily increasing drywell pressure.

BWR-LTAS calculations reveal that the favorable outcome of the low pressure no-venting case at Peach Bottom depends upon the

details of how the SRVs are assumed to behave as the control air pressure becomes inadequate to hold them open. Control air pressure must exceed drywell pressure by at least 5 psi in order for an open three-stage Target Rock SRV to remain open. If all five of the open ADS SRVs are assumed to close at the same instant as soon as the 5-psid (0.034 MPa) necessary to hold open an open SRV is not met), then the reactor vessel will repressurize and the SRVs will remain closed until the setpoint for automatic actuation is reached. The reactor vessel repressurization would prevent low pressure injection and fuel damage would follow. If, however, it is assumed that there is a statistical variation among the individual SRVs of as little as 0.1 psi in the control air pressure required to hold an individual valve open, then all of the automatic depressurization system SRVs do not close simultaneously. While some of the valves close, others remain open so that the vessel does not repressurize, and low pressure injection is maintained. This model sensitivity does not occur for the Browns Ferry case, because the two-stage Target Rock SRVs behave differently. Even if all six of the Browns Ferry ADS SRVs closed simultaneously on inadequate control air pressure, they would soon reopen when the reactor vessel reached a slightly higher pressure. For the two-stage valves (Fig. 12), reactor pressure and control air pressure act in tandem, so increasing reactor pressure makes it easier for the control air pressure to open or hold open the valves.

High Pressure No-Venting Case

For this case it is assumed that the operators reduce HPCI flow from its full 5000 gpm (0.32 m³/s) output to about 3300 gpm (0.063 m³/s) at 2 min after accident initiation, the same time at which they decide to initiate SLC system injection of sodium pentaborate solution. Operation of HPCI in a flow control mode is convenient for the operator because the installed HPCI controller provides automatic control of the HPCI turbine to an operator-input flow setpoint. The intent of the BWR Owners Group EPGs is met because downcomer water level is reduced to near the top of active fuel and reactor power is reduced to about 17%. The operators allow the SRVs to actuate automatically, controlling reactor vessel pressure to the neighborhood of 1100 psia (7.6 MPa). This further frees the operators to devote more time to diagnosing the system response during the accident.

By 10 min, the operators diagnose that the SLC system has failed to actuate, and decide that they must initiate a more stringent form of injection control by reducing the flow injected by the HPCI system to 1000 gpm (0.063 m³/s). (No further adjustments are made to HPCI flow after the readjustment at 10 min). The CRD hydraulic system continues to inject at about 105 gpm (0.0066 m³/s). The operators initiate the suppression

pool cooling mode of the RHR system during the first 20 min.

The results of BWR-LTAS calculations of this sequence are shown on Figs. 13-17. Reactor power (Fig. 13) remains at about 6% after the final HPCI flow reduction (at 10 min). The SRVs actuate automatically (Fig. 14), discharging steam to the suppression pool. The pool temperature (Fig. 15) increases rapidly at first, and then less rapidly after the HPCI flow reduction at 10 min. Pressure suppression pool cooling becomes more effective as the pool temperature increases. However, at the end of 3 h, the pool temperature has only increased to 236 F (387 K).

The drywell pressure is only 24 psia (0.16 MPa) after 3 h - far from the 75 psia (0.52 MPa) containment venting pressure of the Peach Bottom plant emergency procedure. The drywell pressure exceeds 2.5 psig (0.12 MPa) at 34 min, and a high drywell pressure signal would remain in effect after this time. Periodic operator action to reset the ADS timer would be required during the remainder of the accident sequence in order to prevent automatic depressurization of the reactor vessel.

Downcomer water level (Fig. 17) is stable at about 8 ft (2.4 m) above the bottom of active fuel. (We cannot conclude that the level of two-phase coolant in the core would be higher than 8 ft (2.4 m) because of the extra flow resistance that would occur at the jet pump inlets when downcomer water level is above the elevation of the inlets.) In reality, the upper portion of the core would be uncovered and cooled by the substantial flow of steam created by boiling in the lower, covered portion of the core. Therefore, off-line calculations of clad heatup in the upper part of the core were undertaken.

The upper, unmoderated part of the core would be at, or near, decay heat power, and the lower, moderated part would generate all, or almost all, of the fission power. The off-line computer code used for the fuel heatup evaluation is described in Appendix D of Ref. 1. One small change was made for the present calculations: a small amount of fission power was assumed to be produced in the first (lower) one ft (0.3 m) of the uncovered region. Fission power in this region was assumed to decrease linearly from 50% of the average fission power density in the covered part of the core to zero at one ft (0.3 m) above the transition. The one ft (0.3 m) transition zone also receives decay heat.

The effect of the transition zone fission power tail is very evident in the average channel results plotted in Fig. 18. Results (not shown) for a hotter channel with a 1.25 radial peaking factor are similar, but exhibit a 70 F (39 K) higher clad temperature peak. The maximum temperatures calculated are well below the temperature range [1800 F (1256 K) and above] at which

the initiation of rapid zirconium-water reaction would begin.

The degree of fuel heatup experienced in the uncovered portion of the core for this case would be acceptable in view of the rather moderate maximum fuel temperatures predicted, recognizing the effective reduction in power level and the extra time gained for operator action to achieve recovery without severe fuel damage.

SUMMARY

The ORNL studies reported here are based on the Browns Ferry and Peach Bottom plants, but the results should have general applicability to BWRs with Mark I containments. The results show how to prevent severe fuel damage in the event of a worst case ATWS compounded by failure of the SLC system. The most successful techniques for delaying or preventing severe fuel damage are based on restricting (but not eliminating) injected flow.

Containment venting would be of very limited value in an ATWS accident sequence before the occurrence of fuel damage because venting would threaten reactor vessel injection, and possibly cause severe fuel damage. Personnel access to large areas of the reactor building would be sacrificed after the initiation of venting; the hot steamy environment in the building would be likely to cause additional, independent, equipment failures. Pumping systems would experience inadequate NPSH and be unable to continue pumping from the flashing suppression pool throughout the depressurization.

Alternative schemes that do not require containment venting are available for preventing fuel damage. Two such schemes are discussed in this report: a case in which the reactor vessel is depressurized (low pressure case), and a case in which the reactor vessel remains at pressure (high pressure case). Both strategies rely on limiting the flow injected to the reactor vessel and utilizing the pressure suppression pool cooling mode of the RHR system. In the low pressure case, the reactor vessel pressure cycles around the low pressure injection system shutoff head. Injection is limited via inherent mechanisms, and containment pressure exceeds the current 75 psia (0.52 MPa) Peach Bottom containment venting initiation requirement, but does not reach threatening levels.

The high pressure, non-venting case requires that the operators reduce the flow injected by the HPCI system to about 20% of its full capacity. This results in a partially uncovered core, but steam cooling of the uncovered portion of the core is shown to be adequate to prevent severe fuel damage. During the

first three hours after ATWS initiation, drywell pressure increases only to 24 psia (0.16 MPa).

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Fig. 1. Plant operation after failure to scram in the MSIV closure-initiated ATWS accident sequence.

Fig. 2. The BWR Mark 1 containment design employs a small primary containment with a pressure suppression pool; secondary containment is provided by the surrounding structure.

Fig. 3. Venting at elevated pressure would fail ventilation system ductwork in the torus room.

Fig. 4. Net positive suction head becomes inadequate after the initiation of wetwell venting.

Fig. 5. The Browns Ferry Reactor Building atmosphere temperature would increase rapidly after the initiation of wetwell venting.

Fig. 6. Operator action to trip two of four core spray pumps results in moderate power levels.

Fig. 7. Core spray injection decreases when four of six open SRVs close at 80 min.

Fig. 8. Lower injection flow after 80 min results in partial uncovering of fuel.

Fig. 9. Four of the six open SRVs close at 80 min due to increasing drywell pressure and decreasing control air pressure.

Fig. 10. Suppression pool temperature begins decreasing when reactor power is reduced to below the RHR system pool cooling capability.

Fig. 11. Suppression pool temperature turn-around results in drywell pressure decrease.

Fig. 12. For the two-stage target rock SRV, control air and system pressure act in concert.

Fig. 13. Power is reduced to 6% shortly after the HPCI flow reduction at 10 min.

Fig. 14. The reactor vessel is maintained at pressure by automatic SRV actuation.

Fig. 15. The suppression pool cooling dramatically inhibits pool temperature increase during the 3 h period.

Fig. 16. Containment pressure does not exceed values normally considered for initiation of venting.

Fig. 17. Low downcomer water level after the HPCI flow reduction results in partial uncovering of the fuel.

Fig. 18. The peak fuel temperatures occur about 5 min after the HPCI flow reduction to 1000 gal/min.