IDENTIFICATION AND ASSESSMENT OF CONTAINMENT AND RELEASE
MANAGEMENT STRATEGIES

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J. R. Lehner, C. C. Lin, P. Neogy

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Department of Nuclear Energy
Brookhaven National Laboratory
Upton, NY 11973

Abstract

Brookhaven National Laboratory, under the auspices of the U.S. Nuclear Regulatory Commission, is investigating accident management strategies which could help preserve containment integrity or minimize releases during a severe accident. The objective is to make use of existing plant systems and equipment in innovative ways to reduce the likelihood of containment failure or to mitigate the release of fission products to the environment if failure cannot be prevented. Many of these strategies would be implemented during the later stages of a severe accident. The identification and assessment process for containment and release strategies is described, and some insights derived from its application to a BWR Mark I plant are presented here.

1. Background

The experience accumulated from Probabilistic Risk Assessment (PRA) analyses indicates that a cost effective means of reducing severe accident risk even beyond its current low levels can be effected by licensees supplementing plant operating procedures with additional guidance for severe accidents, that is, by planned management of severe accidents. While minor hardware modifications may in some cases be necessary to implement the resulting procedural changes or additions, much can be accomplished through innovative use of already existing plant systems. Such an approach to risk reduction is preferable to one which relies on significant, and therefore costly, hardware modifications or additions.

In March 1990 Brookhaven National Laboratory published NUREG/CR-5474, "Assessment of Candidate Accident Management Strategies" [1]. In this document a set of candidate accident management strategies, previously identified from various Nuclear Regulatory Commission (NRC) and industry reports, such as NUREG-1150 [2], were assessed to provide information to individual licensees for consideration when performing their Individual Plant Examinations. The emphasis of the strategies assessed in NUREG/CR-5474 was on preventing or mitigating core damage, i.e., to arrest the accident progression in-vessel. The effects of the strategies considered were generally well understood and many of the strategies

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were found to be already implemented at some plants.

Currently, Brookhaven National Laboratory (BNL), under the auspices of the U.S. Nuclear Regulatory Commission, is investigating accident management strategies which could help preserve containment integrity or minimize releases during a severe accident. The objective of this BNL program is to make use of existing plant systems and equipment in innovative ways to reduce the likelihood of containment failure or to mitigate the release of fission products to the environment if failure cannot be prevented. Many of these strategies would be implemented during the later stages of a severe accident, i.e. after vessel breach, and sizeable uncertainties exist regarding some of the phenomena involved. A majority of the strategies identified go well beyond existing procedures and often depend on the specific containment type. In the BNL program strategies for all of the five different containments used in the U. S. are being considered: BWR Mark I, Mark II, and Mark III, as well as PWR ice condenser and large dry containments. The two containments reviewed to date are the BWR Mark I [3] and the PWR ice condenser [4] containments.

The remainder of this paper presents, in an abbreviated form, a description of strategy identification and assessment as well as some of the insights from the BNL investigation of containment and release management strategies for a Mark I plant, documented in Reference 3. In Section 2 the strategy identification and assessment process is described. In Section 3 a station blackout sequence for a BWR Mark I plant is outlined and the application of various management strategies at different stages of the sequence are discussed. Section 4 provides a more detailed discussion of a subgroup of the strategies considered in Section 3, i. e. containment venting strategies. Section 5 contains a brief summary.

2. Identification and Assessment

For each containment type the most likely challenges are identified and existing emergency guidelines and procedures are reviewed to see how they address these challenges. Where needed and when possible, new strategies are devised. The feasibility and effectiveness of these new strategies are assessed, allowing for the complicated phenomena and associated uncertainties involved. Both beneficial and adverse effects of the suggested strategies are considered. The additional strategies are evaluated for consistency with existing procedures and for practicality in terms of available resources of personnel and equipment. For both the identification and assessment of strategies maximum use is made of available existing information from such sources as NUREG-1150, other PRA's, as well as industry and NRC reactor safety reports. The assessments also describe the indicators regarding plant status that the operators can realistically expect to have during a severe accident and which would enable them to implement a strategy and monitor its results.

As part of the identification process challenges and relevant strategies are systematically identified and grouped via the development of safety objective trees for the containment under analysis. These trees start from a safety objective which is maintained by certain safety functions. The functions are threatened by particular challenges via a number of mechanisms. Each mechanism can be addressed by one or more strategies. The tree developed by this process for the Mark I containment is illustrated in Figure 1. For example, one safety

objective for all containments is preventing containment failure. This objective is met by the safety functions of containment integrity control, containment pressure control, and containment temperature and thermal attack control. Containment pressure control can be challenged by rapid pressurization or slow pressurization. In a Mark I containment, slow pressurization can result via the mechanisms of noncombustible gas buildup, mass and energy addition, gas burning, or loss of heat rejection. Strategies which address gas burning, for instance, are: maintaining an inert atmosphere, venting to control gas concentration, spraying, or diluting of the atmosphere.

Once a comprehensive list of challenges and strategies has been established the strategy assessment must take into account the effect of accident phase and previous history on strategy implementation. Strategies for similar challenges may vary depending on the accident stage during which the challenge occurs, or different factors may become important for a strategy during different accident phases. For instance, different considerations may apply depending on whether the strategy is implemented before core melt, during core melt, in preparation for vessel breach, during the ex-vessel phase, or during the radiological release phase of the accident. For example, venting is an important containment and release strategy for BWR's. However the implications for venting are very different for "clean" venting than for post core-melt venting. Under some circumstances clean venting can be used to control containment conditions prior to fuel damage to prevent emergency core cooling system (ECCS) failure. For post core-melt venting at least three different "strategies" can be identified with different concerns regarding diagnostics, downside risks, human factors, etc. These depend on whether (1) venting is done early to reduce containment pressure in anticipation of a large pressure increase resulting from vessel breach, (2) venting is required later in the accident due to slow pressurization from core-concrete interaction, or (3) venting via the wetwell is used to provide fission product scrubbing to reduce emissions from a containment leaking under pressure.

To account for the factors alluded to above, accident management strategies are best assessed in the context of the sequences (or sequence categories) during which they may occur. Engineering judgement must be employed in selecting the type and number of sequences used for strategy assessment. Assessing every strategy under all credible circumstances (even if these are assumed to be known) would require the investigation of a very large number of sequences. There are obviously practical limitations on the number of sequences to be considered and the addition of new sequences after the first several will yield quickly diminishing returns.

To assess the containment and release strategies for each containment type a set of sequence categories was selected that would assure: (1) all identified challenges to safety functions were covered, (2) potential failures of safety systems were considered, and (3) sequences with the potential for high consequences were included. For the Mark I containment, the sequence categories selected based on these objectives consisted of station blackout, ATWS, LOCA, loss of containment heat removal, and isolation failure. These provide a range of the accident characteristics which need to be considered such as: The initial condition of the reactor and the containment at the inception of the accident, the speed of accident progression, and the availability of major safety systems. Of course the identified strategies are not only applicable to the sequences used for assessment. The strategies may often be beneficial under other conditions as well, although the circumstances surrounding those conditions may need to be

accounted for in strategy implementation.

The actual strategy assessment process consisted of the following steps:

- 1) Accident sequences were selected during which the identified challenges and associated mechanisms could arise.
- 2) How the existing Emergency Procedure Guidelines address these challenges was considered.
- 3) Where possible, and if necessary, appropriate additional strategies were suggested to address the challenges.
- 4) The feasibility and effectiveness of these strategies was discussed, taking into consideration the complicated phenomena and the uncertainties involved.
- 5) The added strategies were evaluated for consistency with existing guidelines and for practicality in terms of human factors and available resources.

To illustrate the strategy assessment process the following section discusses the application of some potentially beneficial containment and release strategies during a station blackout sequence in a BWR Mark I plant.

3. Application of Strategies During Station Blackout in a Mark I Plant

Station blackout sequences are initiated by a loss of all off-site and on-site ac power. This leads in a Mark I BWR to the loss of all active engineered safety features except the steam powered High Pressure Core Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system. Both high pressure core cooling systems require dc power for operation. Core damage develops when the high pressure core cooling systems cease to work, either from depletion of the station batteries or due to a saturated pool if the suppression pool is the water source for the systems. The latter failure possibility can be avoided if the operator uses the Condensate Storage Tank (CST), or an alternate water supply, as water source for the systems.

Figure 2 is a schematic of the containment pressure history for a particular station blackout sequence in a Mark I plant. The figure is useful as a reference for the following discussion.

In the calculation of this sequence it is assumed that the high pressure core cooling systems fail because of battery depletion six hours after the initiation of the accident. (Battery life could be extended, to more than ten hours, by shedding nonessential loads. This will increase the probability of recovery but also results in higher containment pressure and temperature at battery depletion due to increased decay heat over the additional time period.)

The timing of key accident events is important to assess the probability of successful strategy implementation involving operator actions. As shown by the typical values of Figure 1, core melt starts at about 4.5 hours after the termination of core injection; vessel breach occurs approximately 1.5 hours after core melt; and containment failure occurs at about 3 hours after vessel breach. This timing depends on the assumptions used in the calculation, such as the cladding oxidation rate, and variations in reactor conditions, such as whether the reactor is depressurized or whether there is a stuck-open safety relief valve

(SRV). Results obtained from a number of calculations with a variety of assumptions and reactor conditions [5,6,7,8,9] give the following time window ranges: (1) core melt starts approximately 2 to 5 hours after battery depletion, (2) vessel breach occurs at about 1 to 2 hours after core melt, and (3) containment failure, from high temperature or high pressure, occurs at about 3 to 12 hours after vessel breach. Early containment failure from direct containment heating (DCH) or drywell liner melt-through (DLMT) may occur at the time of vessel breach.

For convenience the accident time history in Figure 1 has been divided into a number of phases. It starts with a "very early" phase which begins at accident initiation and extends to the beginning of core melt. This is followed by an "early" phase which covers the period from the start of core melt until just after vessel breach. The "late" phase lasts from shortly after vessel breach until several hours to a day later. Releases due to containment failure may occur near the end of the late phase but could occur as soon as the vessel fails with an early containment failure.

3.1 Very Early Phase

After the loss of all plant ac power the decay heat is removed by the steam turbine driven high pressure core cooling systems and discharged into the suppression pool. This will result in a temperature rise in the suppression pool due to the loss of pool cooling capability. A slow pressure and temperature rise in the containment atmosphere will also occur due to the evaporation of the water in the suppression pool and the heat transfer from the high temperature primary system. The Primary Containment Control Procedure of the plant will be entered early in the accident, probably due to high suppression pool temperature since torus temperature is the only containment instrument available in many Mark I plants during station blackout conditions. (A typical entry condition is a torus temperature of 95°F). However, containment related actions by the operators are unlikely at this time since there is a lack of containment pressure and temperature data and containment safety equipment such as the fan coolers and the Residual Heat Removal System are unavailable. When the suppression pool temperature exceeds the heat capacity temperature limit (HCTL) for high reactor vessel pressure (approximately 160°F at 1000 psi), the operator will depressurize the reactor pressure vessel (RPV) according to the HCTL curve to a pressure level that still allows the operation of the high pressure core cooling systems (approximately 150 psi). There is no challenge to containment integrity before battery depletion and a full recovery of plant systems is possible if ac power can be regained.

After the depletion of plant batteries, control of the SRV and the turbine driven high pressure core cooling systems will be lost, and the RPV pressure will again increase until the SRV setpoint is reached and an SRV blowdown occurs. The reactor pressure vessel continuously loses coolant through SRV operation and eventually the core is uncovered.

3.2 Early Phase

After core uncover core degradation begins followed by increasing core damage as more coolant inventory is lost through the SRV's. The hydrogen and other non-condensable gases being generated in the overheated core also pass into the suppression pool at the SRV set-point pressure, possibly causing significant suppression pool loads. Eventually, as core melt continues, molten core materials will reach and attack the reactor vessel boundary, resulting finally in vessel breach.

The blowdown of high pressure steam and non-condensable gases at vessel breach will result in a large containment pressure increase. Although this pressure increase alone may not always cause early containment failure, there are mechanisms that may cause containment failure at or shortly after vessel breach. They are direct containment heating (DCH), which may fail the containment by rapid containment pressurization, and drywell liner melt-through (DLMT), in which the spreading hot corium melts through the containment steel shell. Once initiated, these mechanisms can rapidly result in containment failure and, therefore, any strategy that is intended to mitigate their effects should be carried out well before vessel breach, especially since the timing of vessel breach is so uncertain.

After core melt but before vessel breach, containment pressure may reach the setpoint for containment sprays but operator action is again not expected because containment status indications are still lacking and the containment spray system is still inoperable. Even with a containment pressure indication (attainable under station blackout (SBO) conditions in some Mark I plants, e.g., Browns Ferry) and an alternate water supply available, containment sprays are not likely to be used at this stage under present guidelines since the immediate challenge to containment integrity is not likely to be very severe, and the emphasis is still on core cooling. According to current containment control guidelines an alternate water supply should be used for containment spray only if the primary containment pressure limit (PCPL) is reached otherwise it can be used for spray only if it is not needed or useable for core cooling. An example of an alternate supply not useable for core cooling might be a fire water system, cross-connected to the residual heat removal (RHR) system, which has an insufficient pressure head for core injection after battery depletion and the associated RPV pressure increase. This fire water could therefore be used for the containment spray system, even under present guidelines, to flood the reactor pedestal region and the drywell floor.

Such prior flooding could be of benefit in mitigating some of the challenges presented by the mechanisms described above which occur at vessel breach.

The probability of drywell liner melt-through may be reduced by flooding the reactor pedestal region and drywell floor before vessel breach, and continuing water addition to the corium after vessel breach. In cases where containment liner melt-through does occur, fission product release from the containment can be reduced by wetwell venting to reduce containment pressure, and thus the driving force for fission product release at the time of melt-through. The total amount of radioactive materials released will be reduced due to the fission product scrubbing capability of the suppression pool and fission product deposition along the vent path.

While a DCH pressure failure can best be avoided by maintaining the RPV at low pressure before vessel breach, if this is not possible the effect of DCH may also be moderated by reducing the initial containment pressure via containment spray. Sprays may also provide additional heat sinks for the dispersing core debris although at present a pressure increase due to droplet vaporization cannot be ruled out. In addition early wetwell venting can also reduce containment pressure by releasing a part of the containment atmosphere through the suppression pool. This has the added benefit of retaining some of the fission products in the suppression pool.

3.3 Late Phase

Assuming the containment has survived the threats associated with vessel breach, a number of mechanisms will produce challenges to the containment subsequently. The energy and non-condensable gases generated from core concrete interaction (CCI) produce significant thermal and pressure loads on the containment. The containment may fail either by thermal degradation of penetration seals or containment structural materials, from the pressure load, or a combination of both. It is therefore important to control both the temperature and pressure of the containment atmosphere.

The most effective way to control the loads from CCI is to succeed in quenching the corium. Again assuming an alternate water supply has been made available (or AC power has been restored) this may be achieved by keeping the corium submerged with water, supplied either via the drywell spray or through core injection. Although there are questions about whether CCI can be terminated by submerging the corium in a pool of water, adding water should at least lessen CCI. In addition a pool of water on top of the corium will provide aerosol scrubbing and reduce the amount of fission products released from the CCI by aerosol scrubbing.

The containment will fail from excessive pressure if containment cooling cannot be restored in time or if the amount of non-condensable gases generated in CCI exceeds the containment capacity. To prevent containment pressure failure, the current BWR emergency procedure guidelines call for containment venting when the containment pressure reaches the primary containment pressure limit (PCPL). Since the batteries are depleted, the required Control Room pressure indication will be unavailable and successful implementation of this strategy must rely on alternate pressure indications, such as readings from a mechanical pressure gauge. Manual operation of the vent path valves is also required because all electric power sources have been lost. The environmental conditions in the vicinity of the primary containment, where manual operation of the vent valves is performed, are likely to be very harsh at this time. The discharge of the hot and highly radioactive containment atmosphere to the reactor building after valve opening presents a severe threat to the plant personnel performing this task, and may drastically reduce the chance for successful completion. An additional safety risk arises from the possibility of a hydrogen burn in the reactor building.

In order to make containment venting a practical severe accident management strategy for SBO sequences, the above problems of hazardous environmental conditions associated with carrying out the venting procedures manually should be addressed and resolved, perhaps by using a dedicated power supply to operate some valves remotely.

3.4 Release Mitigation

If containment failure cannot be prevented there may still be opportunities to mitigate the resulting fission product release. If the containment failure occurred early, i.e. at vessel breach, due to rapid containment pressurization from DCH or mass and energy addition, or by DLMT this will result in the most severe fission product release: The containment is likely to fail in the drywell at high pressure, the break size is likely to be large, and CCI will be in progress. Without any mitigating measures the amount of release of radioactive materials from the containment will be significant. The only opportunities left for preventing or reducing fission product release to the environment occur in the reactor building, via its natural retention capability and in some cases via the actuation of the reactor building fire sprays. The fission product release from the containment continues after the initial discharge as a result of energy and mass addition to the containment atmosphere from decay heat and CCI.

If the containment survives the challenges at vessel breach, containment failure may still result from the longer term thermal or pressure loads or a combination of both. A containment rupture will result in a significant initial puff release of fission products. If a containment pressure failure is prevented, without containment temperature control containment leaks may develop due to the thermal degradation of the containment penetration seals. The fission product release will depend on the containment failure mode, and so will the severe accident management strategies that are employed.

If all plant electric power remains unavailable at this stage of an SBO accident, there is not much the operator can do to manage the accident. Most of the actions that could be used to mitigate the consequences of the accident should have already been performed, as discussed previously. The only system the operator can perhaps control is the reactor building fire protection system if manual control is available and if the fire spray has not been activated automatically by the high temperature discharge from the break. However if ac power is restored, a not unreasonable assumption since approximately 15 hours have passed since the initial blackout, other strategies are also feasible.

With power restored, wetwell venting and water addition can still play important roles in reducing radiological releases. Other systems that have the potential to reduce the release of radioactive materials to the environment are the reactor building (RB) itself, the reactor building fire protection system (FPS), and the standby gas treatment system (SGTS). If the containment has been flooded previously, this would also achieve some fission product control.

If the containment failure takes the form of a drywell leak and the leakage area is small, the containment will remain at pressure and the fission products will be released slowly, but continuously, from the drywell. Wetwell venting can reduce the driving force for fission product release, i.e., containment pressure, and via fission product scrubbing in the suppression pool, reduce the total amount of radioactive materials released to the environment.

Should the containment failure mode be a leak, fission product release can also be reduced if the leak area can be identified and flooded. The leakage flow from the containment atmosphere will then pass through a pool of water where some of the fission products will be retained in the pool.

At this stage operators will try to add water to the core debris once a water source is available. However, adding water to hot corium will produce a pressure increase in the containment from corium-water interaction and could result in a puff release of fission products to the environment or further aggravate containment leaks. The decision on when and how to deliver water to the corium may depend on the status of off-site evacuation, plant information and analytical predictions regarding the effects of water addition. The availability of plant information and the safety equipment required for water addition depends on the previous course of the accident at the time of ac recovery.

Water addition via containment sprays can be used to scrub fission products from the containment atmosphere and thus reduce fission product releases. The sprays are most necessary and useful when a containment leak or rupture exists and wetwell scrubbing is ineffective, either because of insufficient drywell to wetwell pressure differential or a suppression pool bypass.

Whether there is sufficient information for appropriate spray management is uncertain and may depend on the previous accident progression to the time of recovery. The operators may need additional instructions to initiate the drywell spray because of uncertainty in the conditions limiting the operation of drywell spray. The drywell spray initiation criteria and limiting conditions specified in the existing BWR guidelines may not be sufficient to make spray decisions at this time.

Since the containment is already breached, it is important to avoid the potential adverse effects of sprays such as containment de-inerting. A de-inerted atmosphere can result in hydrogen burns in the containment which may further damage the containment, or cause a puff release of fission products to the environment. Spray management will depend on plant status information and predictions of containment response to containment spray actions.

Fission product retention in the primary system, the suppression pool, and on containment structures is more effective if these elements can be maintained at a lower temperature. Containment sprays using cool water, either from the use of RHR heat exchangers or from alternate cool water sources, can be utilized to achieve this purpose.

The Standby Gas Treatment System (SGTS) may be available after ac power recovery. It can be used to remove fission products from the reactor building atmosphere by HEPA and charcoal filters. It may also be desirable to discharge the release from a higher location, such as the off-gas stack if possible.

Although the HEPA filters of the SGTS are likely to fail in a severe accident from aerosol plugging, the charcoal bed adsorption efficiency may still be maintained. The operation of the SGTS may be desirable even in the case where both HEPA and charcoal filters fail because of the paths and release point associated with the SGTS.

The SGTS may become inoperable if the fire dampers are closed in response to the high temperature of the discharge from the containment break. It may be desirable to set the fire damper closing temperature at a point such that the SGTS will continue to work under severe accident conditions but is still protected from possible fire damage.

On the negative side the operation of the SGTS may reduce the residence time of fission products in the reactor building and, in the event of a loss of system filters, accelerate the fission product release.

The reactor building (RB) fire spray can reduce the release of radioactive materials to the environment. If possible the spray should be actuated manually, after containment failure if it has not been actuated automatically by the high temperature of the break flow. The effectiveness of the fire spray in achieving fission product retention depends on the design of the system: An area spray design which covers an entire area of the RB is more effective than a curtain spray design which prevents the spreading of fire from one area to another.

The diesel driven part of the fire spray system does not depend on plant electric power and may be triggered automatically by the high temperature in the reactor building. However, remote manual actuation, if available, may need plant power. Local manual actuation may not be practical because of the hazardous condition in the reactor building when this strategy is needed. Manual operation of the fire spray system, when all plant power is lost, is further complicated by the loss of all normal reactor building radiation and temperature monitoring systems. Furthermore, the water source of the fire spray system may have been used as an alternate water supply for core cooling or containment spray.

There can be adverse effects associated with use of the RB fire spray: It will condense the steam in the reactor building and increase the possibility of an early hydrogen burn in the reactor building. Continuous operation of the reactor building fire spray may flood the areas where safety-related equipment is housed. This raises the possibility of a common-mode event disabling equipment in the flooded areas which could otherwise be useful subsequent to power recovery.

Finally the possibility of primary containment flooding should be mentioned. Primary containment flooding is part of the reactor pressure vessel (RPV) control guideline in the BWR EPGs (Contingency #6) [10]. The guideline calls for flooding the containment up to the top of the active fuel (TAF) level of the reactor core and thus provide cooling water to the core. While it may be first applied as an in-vessel strategy, its use also mitigates fission product release from the containment by keeping the core materials, both in and out of the vessel, submerged and providing a water pool for fission product scrubbing.

As a strategy for controlling fission product release, it can be used to maintain the plant in a long term stable condition even after significant plant damage from an accident. This flooding may be initiated early in an accident (as compared with most of the containment strategies discussed here) because it is part of the RPV control guidelines in the EPGs. To flood the containment, the suppression pool makeup system is initiated to rapidly add a large quantity of water to the containment, and all available systems that take suction from sources outside the containment are started to deliver water to the containment. Contingency #6 of the EPGs describes in detail the systems, water sources, and procedures to be used to fill the containment. Estimates by Oak Ridge National Laboratory of the amount of water needed to flood the drywell to cover the RPV bottom head in a Peach Bottom type of containment show approximately 1.5 million gallons are needed. Note that previous wetwell venting would increase the amount of water needed, since less gas would be trapped in the wetwell atmosphere space.

Plant systems and instrumentation in the containment may be submerged and damaged. The addition of a large quantity of water to the containment will increase the containment load by increasing containment pressure as a result of a reduction in containment airspace volume, and a hydrostatic load from added water level. The decay heat will also raise containment pressure steadily and the drywell vent paths needed to reject this energy may be flooded. Instrument taps in the containment may be submerged and the effect on instrument readings should be considered.

4. Mark I Containment Venting Strategies

Among the containment and release strategies discussed above the possibility of containment venting was mentioned several times during the description of the accident progression.

The current BWR Emergency Procedure Guidelines (EPGs) [10] call for containment venting as a means to prevent a containment pressure failure. The BWR EPGs prompt the operator to carry out containment venting before the containment pressure reaches the primary containment pressure limit (PCPL). However, the discussion of Section 3 suggested that during a severe accident containment venting could also be used to good effect for other reasons than to prevent exceeding the PCPL. These other uses are: (1) to prevent containment pressure failure at reactor vessel breach by reducing the base containment pressure before vessel breach, and (2) to reduce the total amount of fission products released to the environment by suppression pool scrubbing even after the loss of containment integrity.

In addition containment venting could be enhanced if criteria for initiation of venting were based not only on containment pressure, but also on the rate of containment pressure increase.

4.1 General Venting Considerations

Whenever the primary containment is intentionally opened during the course of an accident certain considerations apply regardless of the reasons for venting.

In all cases operator actions needed to carry out the containment venting strategy include (1) determining that the condition for venting initiation has been reached, (2) determining the vent paths to be opened, (3) defeating the containment isolation valve interlocks (This step needs the assistance of an auxiliary operator to obtain the necessary equipment and make the needed temporary terminal connections.), and (4) opening the valves from the control room. Step 4 requires ac power. In the case of an SBO event ac power is not available and the auxiliary operator must open the valves manually wherever the valves are actually located in the plant. Step 1, determining venting initiation conditions, requires availability of instrument indications, especially pressure readings. Typical pressures and temperatures for instrument qualification are approximately 70 psia and 350°F, respectively [11]. The actual environmental conditions in a severe accident when this strategy is called for may be harsher than the qualification conditions, particularly if CCI has been in progress for some time. In a station blackout (SBO) sequence, the control room indication of

containment pressure is lost after dc power is depleted. This is the most likely time period when containment venting would be needed. Therefore, for SBO alternate pressure indications, such as mechanical pressure gauges, are required for a successful implementation.

Possible adverse effects of containment venting include an unnecessary fission product release in those cases where containment integrity could have been maintained without venting. The amount of fission product release from venting depends on its timing with respect to core damage and vessel breach.

Venting may also cause the loss of ECCS pumps due to insufficient net positive suction head (NPSH) from the reduction of containment pressure combined with a high suppression pool temperature. This problem may be avoided by switching the pump suction from the suppression pool to an alternate source before venting.

The vent valves, once opened, may not reclose successfully when required later. The performance of the vent valves is especially important if intermittent venting is employed. The vent valves may experience some plastic deformation or even fail during the opening and closing process because the valves involved may not have been designed for this particular purpose. This deformation or failure may result in a varying amount of containment leakage depending on whether the valves reseal improperly or suffer a complete inability to reclose.

Duct work which might be used for some of the vent paths outside the primary containment may fail and the vented flow will enter the reactor building directly. This may create a very harsh environmental condition in the reactor building and impair the functioning of important equipment in the reactor building, which is normally qualified for less severe environmental conditions. If hydrogen is released from the primary containment in this way it may burn in the reactor building, which is not inerted. This may damage reactor building structures and also adversely affect equipment performance.

Finally, venting may lead to containment de-inerting and should be terminated before de-inerting occurs.

The time available for the necessary venting actions plays an important role in the feasibility of the strategy, as do environmental conditions if manual opening of the valves is required. For the Peach Bottom plant the time required to successfully carry out the procedure is estimated to be about 30 minutes for both TC1 and TC2 ATWS sequences [9], where power is available but many vent paths must be opened, and about 2 hours for the station blackout sequences where power is unavailable [9]. The environmental conditions such as heat and radiation level may be too high for the operators to manually open the valves, although this is highly plant dependent. For instance, a human reliability analysis for the Peach Bottom plant [9] estimates the success probability for manual containment venting after vessel breach for the TBI station blackout sequence to be zero because the duct work in the reactor building is predicted to rupture and the radiation and temperature environment near the 18-inch wetwell exhaust valves will be lethal (The radiation is estimated to be about 12000 rads/hr) [9]. For ATWS, where time available is shorter and more vent paths need to be opened, the same analysis estimated that the success probability of venting is 0.7 because with power available venting can be accomplished quickly and remotely.

4.2 Early Wetwell Venting to Prevent Containment Pressure Failure at vessel Breach

This strategy differs from current BWR EPGs in that the PCPL is not reached, and not in danger of being imminently reached at the prevailing rate of pressure increase, when venting is initiated in this strategy. The purpose of venting at a lower pressure is to reduce the initial containment pressure, in anticipation of a sudden, large pressure increase associated with a high pressure vessel breach, and thus prevent catastrophic early containment failure. If containment failure is inevitable even with reduced initial containment pressure, such as may occur due to direct containment heating (DCH) or an early drywell liner melt through (DLMT), this strategy may still have some benefit. It can reduce the total amount of fission products released to the environment because that part of the fission product inventory accumulated before vessel breach will be passed through the suppression pool and scrubbed.

Since the timing of vessel breach remains a highly uncertain issue this strategy would have to be implemented based on time from injection failure and available indications from in-vessel instruments. The strategy may be helpful during sequences in which the containment reaches significant pressure well before vessel breach. The application of the strategy would also be based on previously established analytic models and predictions regarding the effect of vessel breach on containment loads under a variety of conditions.

Early venting, when the containment atmosphere is relatively clean, combined with the use of the Standby Gas Treatment System (SGTS), will reduce containment pressure and not increase fission product release to the environment significantly. However, its effectiveness is limited by the capacity of the SGTS.

For cases where the release of radioactive materials is more significant, either from a venting rate exceeding the SGTS capacity due to the required greater pressure reduction rate, or the loss of SGTS function due to the harsh environmental condition created by the vented gas, early venting is less desirable unless there is a strong indication that it is required to prevent early containment failure.

Since this strategy involves engineering judgements and analytic predictions, the decision to implement this strategy should rely on the Technical Support Center (TSC) for technical recommendations. The operator will be reluctant to carry out this strategy once a considerable fission product inventory has accumulated in the containment unless there is explicit and unambiguous guidance. If fission product inventory is relatively low, it would be relatively easy to make a decision on small scale early venting through the SGTS which would keep the off-site radioactivity release rate within allowed limits.

4.3 Venting to Prevent Containment Pressure Failure

This strategy would be carried out when the primary containment pressure and/or the containment pressure rise rate reaches pre-determined limits, i.e. the primary containment pressure limit (PCPL). Current venting guidelines in the BWR EPGs state that the primary containment will be vented before its pressure

reaches the PCPL, even if the permitted off-site radioactivity release level is exceeded.

The determination of the PCPL needs careful consideration because too low a value may cause unnecessary release of fission products to the environment while a higher value increases the potential for containment failure or the inability to activate vital safety equipment [12]. The challenge to containment integrity by containment pressure is determined by the balance between the energy input rate to the containment and the energy removal rate via containment venting (and other containment heat removal mechanisms if they are available). Therefore the size of the vent path chosen (assuming this can be varied) and the time for the initiation of venting should be a function of both the rate of pressure increase as well as the actual pressure: a rapidly rising pressure may require venting at a lower pressure and with a larger total vent area than a slower rate of increase. Some individual plants already include pressure rate considerations in their venting procedures.

Presently there is no guideline in the BWR EPGs on when to reclose the vent path(s). It would be desirable that guidelines, based on pressure considerations, be provided for vent reclosing to minimize the release of fission products.

Depending on the accident sequence in progress, the PCPL may be reached in any phase of the accident. For instance containment venting may be required before core melt in some ATWS sequences. The venting area needed for successful mitigation of an ATWS sequence is likely to be very large. The total effective venting area available in the plant (which may be considerably less than the cross-sectional area of the pipes and ducts of the vent paths) should be evaluated to assure that it is sufficient for containment pressure control. Since the fission products in the containment atmosphere are relatively low before core melt begins, the major concern during such an early vent case is the effect of the high temperature and high flow rate discharge on the reactor building structural integrity and on reactor building equipment. The fission product release will be more severe if containment venting is used after vessel breach. Reactor building retention, including the use of the SGTS and the fire spray, becomes more important for these cases.

4.4 Wetwell Venting for Fission Product Scrubbing

Drywell leakage from a combination of pressure and temperature loading is very possible in the later stages of some severe accidents [13]. The driving force for fission product release is the pressure in the primary containment. Even though this pressure may be below the PCPL, and may not be increasing quickly, wetwell venting may be called for. Such venting will reduce the driving force and provide fission product scrubbing through the suppression pool and thus reduce the total release of radionuclides to the environment.

This strategy may be called for when the radioactivity in the reactor building or off-site is abnormally high (e.g., exceeding the maximum safe operating limit) but the containment remains at pressure. The need of the strategy is indicated by the history of the accident progression (e.g., whether there is a core melt or a vessel breach) and the numerous radiation monitoring instruments in the reactor building and off-site.

To avoid unnecessary venting there must be a clear indication that containment leakage from the drywell exists before this strategy is carried out. There are normally sufficient number of area radiation and temperature monitoring instruments in the reactor building to determine the approximate location of the leak. However, the instruments in the secondary containment are generally qualified for environmental conditions much less severe than those in the containment. Therefore the environmental conditions near the leak may be harsher than those of equipment qualification. In addition, the radiation and temperature monitoring systems will not be working during an SBO event.

5. Summary

In the previous sections an approach for the systematic identification and assessment of containment and release management strategies has been presented and illustrated by applying it to a BWR Mark I plant.

Clearly opportunities exist to apply containment and release strategies during the various phases of a severe accident to mitigate many of the challenges to the containment and to reduce fission product release.

The strategies considered involve only existing plant systems and require only small or no hardware modifications. Therefore they have the potential for providing a cost effective way of achieving additional risk reduction.

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Figure 1 - Safety/Objective Tree with Identified Strategies for a Mark I Containment

Figure 1 -Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 1 - Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 1 - Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 1 - Safety/Objective Tree with Identified Strategies for a Mark I Containment (Continued)

Figure 2 - Mark I Station Blackout

END

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