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SEVERE ACCIDENT INSIGHTS REPORT

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DEPARTMENT OF NUCLEAR ENERGY, BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973

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ABSTRACT

This report describes the conditions and events that nuclear power plant personnel may encounter during the latter stages of a severe core damage accident and what the consequences might be of actions they may take during these latter stages. The report also describes what can be expected of the performance of the key barriers to fission product release (primarily containment systems), what decisions the operating staff may face during the course of a severe accident, and what could result from these decisions based on our current state of knowledge of severe accident phenomena.

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This report summarizes information that has been developed by many different severe accident research programs over several years. As such, the report has benefitted from the efforts of many researchers who cannot all be acknowledged by name. Those who contributed directly to the summaries of the information are listed as contributors. In addition, the report has benefitted from review by the NRC staff in the Offices of RES and NRR.

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1. INTRODUCTION

1.1 Background

The Reactor Safety Study[1] was the first comprehensive evaluation of the risk due to all accidents that were thought at that time to be possible in nuclear power plants including consideration of accidents which are beyond the design basis (severe or degraded-core accidents). The Reactor Safety Study helped to define the "technique" of probabilistic risk assessment (PRA) and applied it to two plants (a pressurized water reactor, PWR and a boiling water reactor, BWR). The Reactor Safety Study found that most accidents would not lead to core damage. However, the study also found that offsite risk was dominated by low frequency accidents that lead to core damage and subsequent early containment failure or bypass. Thus this report deals with low frequency accidents which have the potential for high consequences. An additional insight from the Reactor Safety Study was that many types of accidents besides those initiated by large breaks in the reactor coolant system were important to risk. These sequences included intermediate and small pipe breaks as well as accidents initiated by various transient events.

The Three Mile Island-2 (TMI-2) accident of March 28, 1979 focused attention on accidents that could lead to severe core damage. As a result, PRA methodology received increased attention as an important tool for the estimation of accident frequencies and the probability of additional failures (either human or equipment) which could lead to a severe accident situation. Furthermore, the examination and understanding of beyond-design-basis accidents received increased attention. Several utilities performed comprehensive, plant-specific PRAs for plants that they owned. In addition, the NRC sponsored several studies that applied risk assessment concepts to other plant types and began using PRA techniques in a number of areas of the regulatory process. These uses included safety issue prioritization and resolution of the issues of station blackout and containment leak-tightness. PRA techniques and insights are also finding extensive usage in addressing the severe accident issue for nuclear power plants in terms of both prevention and mitigation. Included are the evaluation of the relative risk importance of containment failure modes from severe core melt accidents, and the development of mitigation strategies and potential plant modifications to enhance containment effectiveness against severe accidents.

The recent accident at Chernobyl[2] and the precursor studies[3] sponsored by the NRC continued to emphasize the likelihood and the importance of severe accidents. The most comprehensive severe accident research program has been sponsored by the NRC. In addition the nuclear industry has undertaken some severe accident research. The NRC programs have expanded the data base for source term analysis and have resulted in improved source term methods.[4] These new methods have been applied to five different U.S. light-water reactors to provide benchmarks to be used in individual plant examinations. These new perspectives on nuclear power plant risk are summarized in NUREG-1150.[5] The nuclear industry research was performed by EPRI and was utilized in the Industry Degraded Core Rulemaking Program (IDCOR) and published in an extensive series of technical reports which are outlined in a summary report.[6] The lessons learned from these studies were assembled by the NRC staff and their contractors at Brookhaven National Laboratory (BNL) into five reports, NUREG/CR-4920,[7] which identify the vulnerabilities and strengths of the various reactor and containment designs under severe accident conditions.

1.2 Objectives

The objective of this report is to summarize the current insights related to severe accidents at commercial nuclear power plants that have been derived from the various studies noted above. Particular emphasis is given to accident management of light-water reactors (LWRs) during severe accidents by highlighting actions that have the potential to either mitigate or aggravate the outcome of the accident.

Issues associated with multiple unit sites have not been addressed. For example, some units may share common systems whose failure could lead to problems at both units. Also the ability of one unit to shut down if another unit experiences a severe accident has not been discussed. Severe accidents caused by "external" evens such as earthquakes, internal floods, fires, windstorms or aircraft impacts have also not been addressed. In general such external events would still have to produce core melt by one of the paths described herein, although some of these external initiators could cause containment failure at the very beginning of the accident. Most of the symptom oriented approaches to mitigating severe accidents discussed in the present report would still apply. If containment failure occurs at the start of the accident the operator would be faced with a situation similar to one in which containment isolation has failed and is not recoverable.

The information in this report is presented in a concise format intended to highlight our current understanding of potential means of failing the barriers to fission product release that may occur during a severe accident. The report covers the five containment types commonly found in commercial nuclear power plants in the United States and emphasizes the mitigative capability inherent in each plant type. Specific options for severe accident management and design features to prevent and mitigate these accidents are identified, but no recommendations are made in this report. Various parts of this information base are described in detail in numerous reports and papers on this subject. The range of phenomena and conditions that may be encountered during severe accidents are identified and discussed.

2. BEYOND DESIGN BASIS ACCIDENTS AND EMERGENCY OPERATING PROCEDURES

2.1 Design Basis Accidents and Beyond

Design basis events (both transients and accidents) have been defined over the years and used to test the overall adequacy of each nuclear power plant design. These design basis events were intended to represent sound judgement regarding the reasonable range of events which might occur, and were thought to define a reasonable envelope of all credible events. Thus, the design of each plant was required to be capable of mitigating the consequences of those events considered in the design basis. The most severe of this set of design basis events in terms of challenging the containment and its associated systems is the spectrum of loss of coolant accidents (LOCA). These accidents serve to set the requirements for a number of safety systems, including the emergency core cooling system (ECCS) and design of the containment building. In conjunction with these accidents, a coolant and fission product release into containment is assumed to occur, and these assumptions are used to set the leak-tightness of the containment and the capabilities of other engineered safety features. The design criteria also consider so-called "external events" such as earthquakes, floods, and tornados.

The Chernobyl accident has focused attention on whether containments for U.S. light-water reactors that were built using criteria based on design basis accidents have adequate margins available to prevent the release of large quantities of fission products during severe accidents. The margins in safety provided through U.S. practice have been the subject of considerable research and evaluation, and these studies have indicated the ability of containment systems to survive pressure challenges of 2.5 to 3 times the design levels. Because of these margins, the various containment types presently utilized in U.S. nuclear power plants have the capability to cope, to varying degrees, with many of the challenges presented by severe accidents. For each type of containment, however, plausible failure mechanisms have been identified which could lead to containment failure. Therefore, the key question is the capability of containments to prevent the release of large quantities of fission products for the most likely severe accident sequences.

A beyond design basis accident may occur when one or more safety systems fail to perform their function in response to an initial challenge or failure. The final outcome of such an event could range from limited fuel damage to a complete core melt, a subsequent severe challenge to the containment and possible releases of radioactivity to the environment. In considering these beyond-design-basis accidents (also called severe accidents), it is important to realistically evaluate the behavior of reactor systems and operating staff performance. This includes understanding the ability of both safety and auxiliary systems to terminate or mitigate the accident sequence in a severe accident environment and to keep fission products from escaping the containment.

2.2 Emergency Operating Procedures

Historically, emergency operating procedures and operator training were based on transients and accidents analyzed and presented in the safety analysis reports and reviewed by the NRC as part of the licensing process. Random and common-mode, multi-failure events, which may occur, were not explicitly analyzed in safety analyses reports. The TMI-2 accident, among other things, focused attention on the importance of managing transients/accidents which could evolve to more complex situations than previously had been analyzed. Plant personnel who attempted to control the TMI-2 accident had to operate beyond their emergency operating procedures and beyond the principles covered in their training program. In the years following the accident, the NRC developed substantial new requirements to address many of the specific weaknesses that had been identified at TMI-2.

Reactor vendors revised their emergency procedure guidelines by considering not only the "traditional" events analyzed in their safety analysis reports, but by considering failure sequences well beyond failure criteria. The emergency procedures approach changed from event oriented to symptom oriented, or a combination of event and symptom oriented. The guidelines were reviewed and approved by the NRC before they were given to utilities to develop and implement plant-specific emergency operating procedures for their plants.

The new emergency operating procedures represent a significant improvement over those used during the pre-TMI-2 years. However, these procedures may fall short of addressing severe accidents (core degradation/melting and containment failure) in an appropriate manner. Operators are expected to respond to <u>any</u> accident situation in accordance with their training and procedural guidance. If an accident degrades to the point that significant core damage occurs and/or severe challenges to containment integrity arise, the operating staff would be faced with a situation for which their training and procedures were not intended. Assessments are currently underway to determine if, and in what form, specific recommendations for action can be provided to operators for managing severe accident situations.

3. SEVERE ACCIDENT PHENOMENA AND CONTAINMENT RESPONSE

To obtain insights into the likely response of a nuclear power plant to a severe accident, it is essential to have an understanding of the phenomena which could occur. This chapter gives a brief generic description of the physical and chemical processes which could take place during the progression of an accident and describes how these phenomena affect containment performance. More detailed descriptions can be found in NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms;"[4] NUREG-1079, "Estimates of Early Containment Loads from Core Melt Accidents;"[8] NUREG-1150, "Reactor Risk Reference Document;"[5] the IDCOR Technical Summary Report "Nuclear Power Plant Response to Severe Accidents;"[6] and the report to the American Physical Society in "Reviews of Modern Physics."[9]

While there are some aspects of severe accident phenomena which depend on the specific reactor type and on unique containment design features, much of the principal phenomena affecting containment behavior can be described in a generic manner.

3.1 Severe Accident Phenomena

The progress of a severe accident can be divided into two phases. The first is the in-vessel stage during which the core heats up, melts and undergoes gross geometry changes while remaining in the reactor pressure vessel. The second or ex-vessel phase occurs after the core materials penetrate the bottom of the reactor vessel or other parts of the reactor coolant system.

A severe accident occurs because the reactor core does not receive adequate cooling and overheats. A decreasing level of coolant in the reactor vessel can be due to a break in the reactor coolant system causing loss of water and/or steam, or a gradual escape of steam from the reactor coolant system due to the heat generated in the reactor core if the decay heat removal system is interrupted.

Above the core mixture level, the fuel rods would be cooled only by the rising steam which may not be sufficient to prevent their temperature from increasing. Significant oxidation of the cladding by the steam eventually occurs. This chemical reaction generates hydrogen and releases energy. For small-break loss of coolant accidents and transients with loss of heat removal, the bottom of the core and the lower plenum can remain covered by liquid for a much longer time than the upper core, and substantial steaming and oxidation can take place.

As the core heats up, the radiative, conductive, and convective heat transfer between the fuel rods, steam, and other core materials will raise the temperature of the non-fuel materials along with that of the fuel. Since the material used for the control rods has a lower melting point than that of the fuel and cladding, the control rods are likely to melt first. This will be followed by local melting of the fuel rods, causing changes in core geometry, altering steam flow paths and changing heating and melting patterns. If the reactor coolant system is at high pressure, strong natural convection flow patterns may develop.

As the core continues to heat up, the fission products in vapor form and the vaporized core materials would be released. Various chemical reactions would take place between the steam, fission products and vaporized core materials. The transport of the fission products in the reactor coolant system would depend primarily on the flow rate of the mixture of steam and hydrogen gas and its interaction with the solid surfaces in the core. The upper internal structures of the reactor vessel may act as a filter where microscopic fission product aerosols suspended in the gas can settle out on comparatively cool surfaces via thermophorisis and diffusiophoresis. Gravitational settling may also aid in retaining fission products in the reactor coolant system.

As the core continues to degrade it becomes increasingly difficult to predict melting and relocation of the core materials and the release of fission products. Eventually, some core debris would fall into the water in the bottom of the vessel. Steam explosions could occur in which molten fuel rapidly fragments and transfers its energy to the water causing rapid steam generation and shock waves. Small steam explosions are probable but explosions sufficiently energetic to lift the reactor vessel head and thus fail containment are considered unlikely. As the core slumps into the bottom of the reactor vessel, core debris may quickly attack and penetrate the vessel or may first boil off the remaining water before melting through the bottom of the vessel.

The whole process of core degradation, relocation, and failure of the reactor coolant system has a considerable degree of uncertainty attached to it. This uncertainty involves the possibility of an in-vessel steam explosion, the nature of the thermal attack of the core debris on internal vessel structures and on the vessel boundary, the time and mode of vessel failure, as well as the characteristics of the core released at failure. Other uncertainties include the amount of in-vessel hydrogen generation, the in-vessel fission product release and the transport, and retention of fission products and other core materials in the reactor coolant system. The uncertainty in the amount of natural circulation in the reactor coolant system is an important factor. There is some indication, especially in a pressurized water reactor with the primary system at high pressure, that a large amount of natural circulation could occur. This circulation might heat up the reactor coolant system and cause it to fail. The location of this induced failure is uncertain and could occur in the steam generator tubes. If the secondary system relief valves were open, a direct path would exist from the damaged reactor core to the environment.

Fission products from the degraded core can enter the containment via pressure relief valves or breaks in the reactor coolant system prior to vessel penetration. Therefore, the status of the containment prior to vessel failure has important implications for the consequences of a severe accident. Containment isolation failure arising from inadvertent openings or pre-existing leaks, as well as containment bypass via an interfacing systems LOCA are possibilities that require attention. Less likely, but not completely ignorable for some containments, is the possibility of containment leakage or even failure at this stage because of high temperatures and pressures caused by steam, noncondensibles and combustible gases in the containment atmosphere.

As the above discussion indicates, the exact manner, mode and timing of vessel penetration are difficult to predict. The key factors affecting the progress of the severe accident after vessel penetration are the pressure of the reactor coolant system at vessel failure, the composition, amount and character of the molten core debris and the amount of water, if any, in the reactor cavity or pedestal region. If the reactor coolant system were still at elevated pressures when failure occurred, the molten core would likely be ejected through initially small breaches in the reactor vessel or other reactor coolant system location as a jet, breaking up into small particles. In some containment designs where suitable pathways exist, elevated pressures in the reactor coolant system may also cause core debris ejected from the vessel to be swept out of the pedestal region or reactor cavity and into the main containment. If the core debris becomes finely fragmented, the pressurized dispersal could suddenly heat the atmosphere of the containment and contribute to rapid, large pressure loads. Chemical reactions of the particulate with oxygen and steam would add to these loads. This phenomenon is termed direct containment heating.

If depressurization occurred prior to reactor pressure vessel failure, the molten core might flow through the initial opening, erode additional steel, and then relocate below the vessel without dispersal into the containment atmosphere. If there is water below the reactor into which the core materials drop, then steam would be generated which would raise pressure in the containment. Small steam explosions may also occur. These explosions are unlikely to directly threaten the containment integrity but may cause fragmentation and dispersal of core debris into the main containment and increase the containment's pressure.

Contact of the molten core debris with the concrete in the reactor cavity, pedestal, drywell floor, or basemat will lead to core-concrete interaction. The phenomenon of core-concrete interaction and the possibility of cooling the core debris are affected by many factors, including the amount of water available, the containment geometry, and the type of concrete involved. Containment integrity can be challenged by core-concrete interactions in several ways. The high temperatures and pressures resulting from the core-concrete interaction may cause leaks at containment seals or penetrations or even result in structural failure. Core-concrete interactions release additional steam, hydrogen, carbon monoxide, and other noncondensable gases, all of which could contribute to increases in containment pressure. Combustion of hydrogen and carbon monoxide could further increase the pressure and temperature in containment. The hydrogen released at this stage could augment the hydrogen generated during the in-vessel zirconium oxidation. Depending on the containment size and atmosphere, the large amount of energy released by burning or detonation of hydrogen (and carbon monoxide) could challenge containment integrity. As indicated in the next section, in some containments, i.e., BWR Mark I, direct attack of the containment boundary by the core debris is possible. Prolonged debris attack on the containment basemat may result in complete penetration of the concrete mat and the escape of core material into the underlying soil.

3.2 Containment Response

The containment represents the final barrier, emphasized in the 'defense-in-depth' strategy, between the fission products in a nuclear reactor and the environment. The timing of the containment failure is important to the consequences of an accident. Early failure or bypass of the containment could result in a large release of fission products to the environment. Late containment failure generally has less severe health consequences. If containment failure is delayed more than a few hours after the start of core damage, several particulate and vapor removal mechanisms have time to greatly reduce the concentration of fission products in the containment atmosphere and the volume of the release. The volume of the release of fission products is also determined by the size and location of the break (both highly uncertain) and the pressure in the containment.

3.2.1 Containment Structural Failures

Pressurized water reactors have primary systems which normally operate at very high pressures. Most pressurized water reactors have large-dry containments. These containments rely on structural strength and large internal volume to maintain containment integrity during an accident. In order to structurally fail these containments early in an accident sequence they must be subjected to very severe and rapid pressure loads. Such loads can be produced in the absence of containment heat removal systems and if direct containment heating occurs. If the primary system is at low pressure and with the containment heat removal systems operating, the likelihood of early containment failure is much lower.

Some pressurized water reactors have ice-condenser containments. These are smaller in volume than typical large-dry containments and are equipped with ice beds to condense steam during an accident. Ice condensers are required to have hydrogen igniters because their smaller volumes are more likely to develop higher hydrogen concentrations during a severe accident and are less able to accommodate the loads associated with hydrogen combustion events. Therefore station blackout accidents are important for ice-condenser plants because the hydrogen igniters and the air return fan might not operate. Similar to large-dry containments, the pressure of the primary system at vessel failure and the availability of containment heat removal systems influence the likelihood of early failure of an ice-condenser containment.

Boiling water reactors normally operate at approximately half the pressure of PWRs and have generally smaller containments. All boiling water reactors have automatic depressurization systems which can lower the reactor coolant system pressure and therefore could reduce the probability of a high pressure core meltdown event. In addition to their structural strength, BWRs rely on water pools to promptly condense steam to prevent overpressure. Most existing BWRs have either the Mark I or Mark II type of containments, which are completely enclosed in a reactor building. This building may also serve to trap fission products released from the primary containment during a severe accident. BWR Mark III containments have pressure capacities roughly similar to ice-condenser containments.

The most common BWR containments are the Mark I containments, which have an inerted atmosphere. Accidents in which effective containment heat removal cannot be achieved could cause containment failure prior to core melt in these containments because of their small volume. Early failure of the Mark I containment by direct containment heating, referred to in Section 3.1, is possible if the core were to melt through the reactor pressure vessel while the reactor coolant system was at high pressure. If the reactor coolant system is depressurized, this threat is diminished. Another potential type of early containment failure could occur if molten core materials flow across the drywell floor and contact the steel containment. If this were to occur, the steel could melt and fission products in the Mark I drywell would pass to the reactor building without pool scrubbing. However, some fission products will still be retained in the reactor building especially if fire sprays are operating. The suppression pool in a Mark I containment could prevent failure due to steam pressure (by condensing the steam) but could not prevent failure due to pressure from noncondensable gases. Because of their small volume Mark I containments are susceptible to overpressure failure within a few hours of the start of core-concrete interactions.

Mark II containments are very similar to Mark I containments in their essential features and potential failure modes. However, containment wall melt-through is much less likely for Mark II containments because of the large drywell floor area and the fact that core material would likely flow down the downcomers into the suppression pool before reaching the containment wall. Direct attack of the downcomers by the molten core is a possibility in most Mark II containments. Damage to the downcomers in a Mark II via debris interactions with the water in the downcomers is another possibility.

Mark III containments, like Mark I and II containments, rely on a water pool to condense steam during an accident. Since Mark III containments have a relatively large volume, their atmosphere is not inerted. Mark III containments have igniters to prevent the buildup of large hydrogen concentrations. Station blackout accidents have been found to be important for BWRs with Mark III containments since in these accidents the hydrogen igniters would not be available to provide early controlled hydrogen burning. Uncontrolled burning could lead to containment failure. A potential for early containment failure due to hydrogen combustion exists even with the thermal igniters operating since there is uncertainty associated with the performance of thermal igniters during core-melt accidents. Failure of Mark III containments could also be caused by direct containment heating if the core were to melt through the reactor pressure vessel while the reactor coolant system was at high pressure. However, early containment failure in Mark III plants would not necessarily result in a large fission product release. If drywell integrity were maintained, the aerosol fission products would be scrubbed in the suppression pool and the fraction released to the environment would be significantly reduced.

Finally, all the containments are also potentially susceptible to some form of late failure. Structural failure due to long-term pressure and temperature buildup or the penetration of the containment basemat by core debris are both possibilities. The likelihood of these failure modes depends on the individual containment type and the absence or presence of decay heat removal systems. In some large-dry containments, even with decay heat removal systems inoperable, structural failure may never occur.

3.2.2 Containment Isolation Failures

Loss of containment isolation during a severe accident may have consequences as severe as a large structural failure. Past history shows that isolation failures have occurred under normal operating conditions and therefore must be taken into account when discussing severe accidents. Containment isolation failure can result from inadvertent pre-existing openings in the containment boundary or from the failure of valves used to isolate the major process lines and other boundary penetrations. These valves together with the associated sensors and power supplies comprise the containment isolation system. In BWRs, openings between the drywell and wetwell, other than those submerged in the suppression pool, can more rapidly jeopardize proper functioning of the containment.

Before the operator can respond to an isolation failure, some symptoms indicated by plant instrumentation must make him aware of the failure. Incorrect instrument indications of isolation status would be difficult to detect except in special cases. In BWRs the closing of the main steam isolation valves (MSIVs) would cause a sudden increase in reactor coolant system pressure and activate the safety relief valves to relieve that pressure. The absence of such a reactor coolant system pressure increase would alert the operator that the MSIVs have not shut despite control panel indications to the contrary. However, partial isolation failures may be more difficult to detect.

In BWRs with Mark I and Mark II containments, the atmosphere inerting systems may alert the operator to an open access hatch or other inadvertent opening in the containment boundary by showing higher than normal nitrogen flows. However, nitrogen monitoring may not be performed on a continuous basis.

If a severe accident is in progress in a BWR with inadequate isolation, radiation detectors in the reactor building located on the refueling deck or in the ventilation exhaust would aid the operator in recognizing that an isolation problem exists. The reactor building or secondary containment of BWRs can play a significant role in mitigating the consequences of inadequate isolation by providing an additional obstacle to fission product release.

Regardless of whether the isolation problem is detected during normal or accident conditions, the operators would be expected to attempt to restore containment isolation. If a severe accident is in progress in a BWR and restoration of isolation cannot be achieved, the operator can still mitigate the consequences of isolation failure by prudent use of the various spray systems. For instance, if the isolation problem resides in the MSIVs and fission products are released into the reactor building atmosphere instead of the turbine or condenser, it may be possible to use the reactor building fire sprays to decontaminate the atmosphere and reduce temperature and pressure. However, the water from sprays may cause short circuits in electrical equipment. If the isolation failure is an opening in the drywell itself and there are fission products in the drywell atmosphere, use of the drywell spray will reduce the amount of radioactivity escaping through the opening into the secondary containment. In some cases, depending on the adequacy of water supplies, using sprays in both the primary and secondary containment may be helpful.

In PWRs detection of isolation problems, other than those identified by the isolation valve status indicators in the control room, is very difficult. Loss of vacuum in PWRs with subatmospheric containments would be one indication of inadequate isolation. Some PWRs use an enclosure building which is kept at less than atmospheric pressure. A pressure increase in this enclosure building would again be evidence of an isolation problem. However, the majority of PWRs have neither a subatmospheric containment nor an enclosure building.

If an isolation problem is detected in a PWR the operator would attempt to correct the problem and restore isolation. Should this prove impossible and a severe accident is in progress, the use of the containment spray system could lower the concentration of fission products in the containment atmosphere and reduce the amount of radioactivity escaping through the isolation breach.

3.2.3 Interfacing Systems LOCA

Failure of the barriers between the high-pressure reactor coolant system and connected low-pressure systems, with some components outside of primary containment, represent another way the containment function can be bypassed in both PWRs and BWRs. Although such interfacing systems LOCA sequences have been found to be relatively low frequency events, they may lead to potentially high radiological releases because these events provide a direct path for release of fission products to the atmosphere.

If an interfacing systems LOCA is in progress the operator may be able to arrest it by isolating the component or section of piping where the failure occurred. In many cases adequate additional valves exist to isolate the affected section of the low-pressure system if it can be located precisely.

The breach of the low-pressure system outside of the primary containment boundary will occur in the reactor building of BWRs or the auxiliary building or safeguards building in PWRs. If isolation of the failed component is not feasible it may be possible to flood the location of the break with water (provided such flooding does not adversely affect other essential components) and mitigate the consequences of the LOCA by scrubbing any fission products which are being released. If flooding is not possible, the operator may be able to turn on fire sprays in the BWR reactor building or the PWR auxiliary building to reduce the pressures and temperatures and decontaminate the atmosphere. In some situations it may be feasible to depressurize the high-pressure system to reduce the flow bypassing the containment.

3.2.4 Induced Steam Generator Tube Rupture (SGTR)

As mentioned in Section 3.1, one of the possible failure sites of the PWR primary system during a severe accident is in the steam generator tubes. Although SGTR has many of the characteristics of a small LOCA, it is unique in the sense that it is also a potential containment bypass LOCA, releasing fission products in the primary reactor coolant into the secondary-side of steam generators. This could provide several potential paths for the release of fission products to the environment outside the containment (e.g., via the main steamline, turbine, turbine bypass, condenser, condenser exhaust, steam generator atmospheric relief or safety valves, and the steam generator blowdown line). Thus, despite the low frequency of SGTR as an event which may lead to core damage, it can potentially result in a large fission product release if it occurs before or during a severe accident.

The principal objectives of recovery actions and actions to mitigate the consequences of SGTR are (1) to stop the primary-to-secondary leakage, (2) to restore reactor coolant inventory, (3) to minimize the release of fission products from the ruptured steam generators to the surrounding environment, and (4) to regain plant control.

Several operator actions are important in achieving these objectives, including early diagnosis of SGTR, identification and isolation of the faulty steam generator, manual depressurization of the primary system to stop the leakage flow, and prevention of main steam line flooding. Some of the key actions to be taken will differ substantially depending on whether offsite power is available or not.

3.3 Summary

The phenomena accompanying the core melt progression and the containment failure modes discussed above illustrate some of the ways that containment integrity can be lost or the containment function bypassed during a severe accident. Because of the wide range of possible severe accident sequences, a realistic assessment of the loads the containment is experiencing is essential in trying to maintain the containment function. Operator responses are necessarily dependent on reactor and containment type as indicated in the following two chapters.

4. PWR SEVERE ACCIDENT SEQUENCE PROGRESSION AND EFFECTS ON CONTAINMENT

This chapter describes severe accident progression for PWRs and identifies possible operator actions which may reduce the severity of a severe accident. It also describes possible risks that may accompany operator actions.

Many studies have predicted that most severe accidents would be started by transients or small breaks in the primary system pressure boundary. Under these circumstances, the primary system would remain at high pressure unless the plant operators take actions to reduce the pressure. Actions taken to depressurize the primary system may be helpful since the likelihood of early containment failure is higher for a severe accident in which the primary system remains pressurized during core meltdown than if the primary system is depressurized. Thus, differences between accidents with the primary system at high and low pressures warrant separate discussions up to and including penetration of the reactor pressure vessel by the molten core and its subsequent blowdown. However, one discussion covering the period of time after the core debris has been released into containment is sufficient to cover the expected phenomena.

4.1 High-Pressure Sequences

If an accident occurs in which the primary system is at high pressure, high-pressure injection is not available, and heat removal through the steam generators is not effective, then core damage will occur unless the plant operators restore high-pressure injection, reestablish heat removal through the steam generators or depressurize the primary or secondary systems to take advantage of any available low-pressure injection systems.

One potentially effective way of making up core inventory if the high-pressure injection systems are unavailable (either for injection or recirculation) is to depressurize the primary system using the steam generators. After the primary system has been depressurized via secondary heat removal, the accumulators will inject and the low-pressure injection systems can be actuated. This emergency procedure is being investigated by both the nuclear industry and the NRC. To attain success in this procedure, the operator must open the steam generator atmospheric steam dump valves, maintain auxiliary feedwater, main feedwater, or special makeup (e.g., fire pumps) to the steam generators, and have one of the low-pressure injection systems available. Although these procedures may improve cooling capabilities, the secondary side depressurization and reflood may induce thermal shock and hence increase the possibility of steam generator tube rupture, which may aggravate the accident sequence. In addition, depressurization of the steam generators will cause loss of steam that could otherwise be used for the steam-driven pumps.

If primary system depressurization cannot be achieved by heat removal through the steam generators, direct depressurization may be possible by opening relief valves in the primary system. This procedure is also being investigated by both the nuclear industry and the NRC. If successful, the procedure will allow the accumulators to inject, and available low-pressure injection systems can be actuated. Even if no low-pressure injection systems can be made available and core damage eventually occurs, depressurization may still be advantageous because vessel failure with the primary system depressurized represents much less of a challenge to containment integrity than if the primary system is allowed to remain at high pressure. Although this procedure has obvious advantages, it is not clear to what extent existing systems in various reactors are capable of meeting such an objective. Depending on the particular strategy, it is also possible that for some accidents the time to core damage could be reduced relative to time that it would have taken with the primary system at high pressure. In addition, pressure relief will also vent hydrogen to the containment atmosphere early in the accident, with the possibility of a burn or detonation. Therefore before implementing this procedure the advantages and disadvantages have to be carefully assessed.

If the primary system cannot be depressurized and the core cannot be adequately cooled, then core damage will eventually occur. The initial stages of core degradation involve coolant boiloff and core heatup in a steam environment. At such high pressures the volumetric heat capacity of steam is a significant fraction of that of water (about one-third) and one should expect significant core (decay) energy redistribution due to natural circulation loops set up between the core and the remaining cooler components of the primary system. As a result of this energy redistribution, the primary system pressure boundary could fail prior to the occurrence of large-scale core melt. The location and the size of failure, however, remain uncertain. In particular, concerns have been raised about the possibility of steam generator tube failures and associated containment bypass.

It is also possible that plant operators may restore the coolant injection systems after the start of core damage. Restoring water flow to a damaged core is very important. The water will cool the reactor core, prevent further degradation, and could prevent the core from melting through the reactor pressure vessel. Although supplying water to the damaged core is a major objective of the plant operators, there are phenomena associated with mixing water and high temperature molten core materials that are difficult to predict. It is possible that by adding water additional steam will be produced which in turn could generate more hydrogen before the core materials are cooled. It is also possible that more violent interactions may occur between the hot core debris and water and additional fission products may be released. In addition, adding relatively cool water to hot fuel could shatter the fuel into rubble, impeding coolant flow through the core. This was speculated to have occurred at TMI-2 when the coolant pumps were turned back on.

If water flow to the core cannot be restored and the primary system is not depressurized, then core relocation (into the lower plenum) and subsequent lower vessel head failure at high pressure will follow. Upon vessel failure, violent melt ejection could produce large-scale dispersal and the direct containment heating phenomenon mentioned previously in Chapter 3. Hydrogen can also play a potentially energetic role during the blowdown process. The presence of hydrogen arises from two complementary mechanisms: (1) the metal-water reaction occurs at an accelerated pace and generates hydrogen throughout the in-vessel core heatup/meltdown/relocation portion of the transient, and (2) the reaction between any remaining metallic components in the melt and the high-speed steam flow that partly overlaps and follows the melt expulsions from the reactor vessel will generate hydrogen. The combined result is the release of rather large quantities of hydrogen into the containment volume within a short time period (a few tens of seconds). In addition, for some metal containments, debris dispersal can result in containment failure by melt through.

In general, if the accident has progressed to this point there appears to be very little that can be done by the plant operators to mitigate the effects of a high-pressure failure of the reactor vessel during the blowdown phase. However, it is possible for the plant operators to manage containment-related safety features prior to reactor vessel penetration to minimize the effects of high pressure melt ejection. If fan coolers or sprays are available they will keep the containment pressure low so that any pressure increase generated at the time of reactor pressure vessel failure will have less impact than if these containment systems were not operating. In addition, if fan coolers are able to maintain a low containment pressure, spray operation could be saved until the plant operators have an indication of core damage. Spray operation after core damage and fission product release could aid in removing aerosol fission products from the containment atmosphere. However, it has also been suggested that it is an advantage to have water in the reactor cavity prior to the release of the core debris from the reactor pressure vessel. For some containment designs, water will only be available in the reactor cavity after the sprays have injected all of the water from the refueling water storage tank into containment. Thus, the advantages and disadvantages of early or late spray operation depend on the containment design.

After blowdown of the reactor pressure vessel, there are several possible operator actions, which are discussed in Section 4.3.

4.2 Low-Pressure Sequences

At low system pressure, decay heat redistribution via steam flows due to internal natural circulation flow is negligible and core degradation occurs with very little heat loss to the remainder of the primary system. Limited availability of steam ensures low hydrogen generation rates. Steam boiloff together with any generated hydrogen is continuously released from the primary system to the containment atmosphere, where mixing is driven by natural convection currents coupled with condensation processes. The reactor pressure vessel upper internals remain relatively cool offering the possibility of trapping fission product vapor and aerosols before they are released to the containment atmosphere. Throughout this core heatup and meltdown process the potential to significantly pressurize the containment is small. In the very unlikely event of complete loss of all emergency cooling functions and a large break in the primary system, core damage could begin 30 minutes or sooner after the start of the accident. If core cooling is not restored, the core could relocate to lower regions of the reactor vessel and penetrate the vessel lower head within 2 hours of the start of the accident. Thus, the time frame for operator action for this type of accident is relatively short if core damage is to be prevented or the core is to be retained within the reactor vessel. However, with the primary system at low pressure, the plant operators have a number of additional systems that could be made available to inject water into the core.

If water flow to the core cannot be restored, the core materials will eventually penetrate the reactor vessel. Failure of the vessel at low pressure does not generally present an immediate threat to containment integrity. However, it has again been suggested that it would also be an advantage to have water in the cavity prior to vessel failure for low-pressure accidents to mitigate core-concrete interactions. It is also an advantage to have spray operation during core degradation and vessel failure. Thus, it is again necessary to optimize spray operation to ensure the most effective mitigation of the accident.

4.3 Ex-Vessel Sequences

The composition, temperature and mass of core materials (corium) released from the vessel are all uncertain. The previous sections discussed possible actions that operators might take prior to the release of the core debris from the vessel. In this section actions that might be taken after the vessel has failed are discussed.

Whether or not the containment fails early, the long-term objectives are the same, namely to try to flood the core debris with water and to maintain or restore containment heat removal systems. Even if the containment has failed, it is advantageous to flood and cool the core debris to prevent further fission product release from the damaged fuel and to keep the containment atmosphere at a low pressure to minimize the driving force for fission product release to the environment. Although these objectives are well defined, in practice they may be difficult to achieve and may have side effects that have to be carefully considered.

When trying to flood the core debris with water, the plant operators may have to decide to restore water flow to the primary system or directly to the containment atmosphere via the spray system. If the core debris has been released from the reactor vessel, there are advantages and disadvantages to using either system. However, more fundamentally, the plant operators will probably not be able to determine whether or not the core debris has actually been released from the vessel or know the extent to which it is dispersed within the containment. Thus, the first priority would be to restore water flow to the primary system in an attempt to retain the core debris in the vessel. If the core debris has already penetrated the bottom head, the water would flow through the break onto the top of the core debris in the reactor cavity. The subsequent interactions between the water and core debris depend on the containment design and the details of the accident prior to vessel failure.

If the core debris melts through the vessel with the primary system at high pressure, it is possible that a large fraction of the core debris could be blown out of the reactor cavity. It is therefore possible that limited quantities of core debris would be left in the cavity and thus minimum concrete attack would occur. However, at the other extreme, if the primary system is depressurized when the core debris melts through the vessel, most if not all the core debris would likely be retained in the reactor cavity. If the cavity is initially dry and the core debris forms a deep bed, it could remain hot for a relatively long time and extensive concrete attack would occur. Under these circumstances, pouring water on top of the core debris may have the effect of rapidly cooling the core and stopping concrete attack. However, experiments have shown that a crust can form on top of the molten core debris and effectively prevent the water from mixing with and cooling the core. These experiments were performed at small scale and the stability of the crusts under the conditions of a severe accident in a power plant have not been established. Thus, the effect of pouring water on top of the core debris is uncertain. Therefore, even though water flow is restored to the core debris continued concrete attack with the generation of more combustible gases and fission products is still possible. The presence of water above the debris does have the advantage, however, of trapping a fraction of the fission-product aerosols generated during the concrete attack, which would otherwise have reached the containment atmosphere.

If the cavity is flooded with water prior to the core debris melting through the vessel, then, as the molten core materials fall into the water, rapid cooling and fragmentation is possible. This process could result in the formation of a coolable debris bed in which all the core decay heat is removed by boiling water and concrete attack is prevented. This is why having water available in the cavity prior to vessel failure was considered advantageous in previous sections. Under these circumstances providing water flow to the cavity would replenish water loss due to bo ng and ensure that the core remains in a coolable and stable configuration.

If the containment heat removal systems are not working, it should be noted that pressurization of the containment due to core debris attacking concrete is a relatively slow process. However, concrete attack does generate large quantities of aerosols (nonradioactive and radioactive), combustible and noncondensable gases, all at relatively high temperatures. If water cools the core debris and all the decay heat goes to boiling water, containment pressurization is significantly faster. However the containment temperatures are relatively low and concrete attack is slowe. or prevented.

Also if the containment heat removal systems are not working, high steam (above 50 volume percent) and noncondensable gas concentrations in containment can prevent burning of combustible gases. If containment heat removal is restored under these circumstances, steam condensation could result in the formation of a combustible mixture of possibly detonable gases in containment and, if an ignition source is available, a potentially damaging combustion event could occur. Therefore, the decision to restore containment heat removal systems after extensive core damage has occurred must be made with caution. If containment heat removal becomes available after core damage, it should be applied for some containment designs in conjunction with hydrogen control (igniters).

If containment heat removal cannot be restored, containment pressurization might continue until structural failure. Under these circumstances, controlled containment venting has been suggested as a way of preventing the uncontrolled release of radioactivity that would accompany structural failure of the containment. However, there are questions regarding venting that must be answered prior to implementing such a procedure. Combustion could occur in the vent line, which in turn could produce an uncontrolled release. Venting at some predetermined pressure level significantly below the containment ultimate capacity means that radioactivity release is certain whereas if the accident had been allowed to proceed structural failure may not occur. Isolation valve performance during venting is also uncertain and, if the valves fail open, could result in an uncontrolled release. Finally, restoration of containment heat removal to a vented containment could condense residual steam, and produce a vacuum in containment, with the attendant possibility of implosion.

Shutdown/refueling conditions are also vulnerable to loss-of-cooling events. There have been a number of loss-of-cooling events in PWRs that have led to boiling in the reactor vessel. Failure to mitigate such events would lead to core damage and any mitigation could be hampered by necessary equipment being unavailable due to maintenance. When coupled with the possibilities of the reactor head being removed and/or the containment being open, this situation could lead to a severe accident. However, there is considerable time available to take mitigative actions under these circumstances.

In summary, accident management strategies can be devised that deal with the potential challenges to containment integrity and with the potential for release of fission products after reactor vessel penetration. After penetration of the reactor pressure vessel by molten core debris, emphasis should be placed on terminating or mitigating the effects of interactions between core debris and containment concrete. The introduction of water (from any source) to the containment floor or reactor cavity (via reactor vessel injection pathways or containment sprays) and filtered containment venting are two possible means of managing the containment integrity and fission product release. Each means can have a beneficial impact on the progression of the accident; but each has the potential for undesirable consequences that must be weighed when selecting an optimum strategy.

5. BWR SEVERE ACCIDENT SEQUENCE PROGRESSION AND EFFECTS ON CONTAINMENT

This chapter follows the same format as the previous chapter and describes severe accident progression for BWRs. Possible operator actions which may reduce the severity of a severe accident are again identified together with the risks that may accompany some operator actions.

Many BWR studies have indicated that accidents initiated by transients are more likely than LOCAs. The most recent BWR studies point to common-cause failures as important causes of severe accidents. The major subset of common-cause failures that were found to be important relates to station blackout. However, other contributors such as internal floods, loss of vital bus, instrument error, and loss of service water were also found to be important. Under most of these circumstances, the reactor coolant system would remain at high pressure unless actions are taken to reduce the pressure. All BWRs have automatic depressurization systems (ADS) which are designed to rapidly depressurize the reactor coolant system. Thus, the potential exists to reduce the likelihood of high pressure core meltdown events in BWRs. However, the ability of some existing ADS to keep the reactor coolant system depressurized for all potential core melt accidents is not fully established. Therefore, core melt with the reactor coolant system at high pressure will also be considered for BWRs.

5.1 High-Pressure Sequences

Station blackout accidents have been found to be potentially important high-pressure sequences. Station blackout refers to a loss of the offsite power supply with concurrent failure of the emergency ac power divisions.

For most BWR plants, the two systems designed to operate in the presence of a station blackout are the high-pressure coolant injection (HPCI) or high-pressure core spray (HPCS) and the reactor core isolation cooling (RCIC) systems. Therefore a primary objective of accident management during a high-pressure sequence is to prevent failure of these high-pressure pumps. If HPCI/HPCS and RCIC do become unavailable, activation of the ADS to lower reactor coolant system pressure, allowing the use of alternate injection from low-pressure systems, remains an option if ADS capability has been maintained. The control of both the high-pressure injection system as well as the control of ADS requires dc power. Therefore a third important consideration during a station blackout is the management of dc power to maintain control of the most important systems for the maximum length of time.

Station blackout accidents have been classified as long-term or short-term sequences. In long-term station blackout sequences the emergency high pressure coolant injection systems would operate until control power (batteries) is lost or the turbine driven pumps fail (because of high pool temperatures, high turbine exhaust pressure, or loss of net positive suction head). In a short-term station blackout early failure of HPCI/RCIC occurs due to common causes (such as dc bus failure).

For long-term station blackout sequences eventual failure of the pumps due to heating of the suppression pool can be avoided if the HPCI/RCIC suction can be switched from the suppression pool to another water source (e.g., condensate storage tank or the fire system). Even if HPCI/HPCS and RCIC become unavailable, alternative injection sources could be used for vessel makeup water. If appropriate plant procedures and connecting equipment were in place, water from fire trucks or diesel-driven pumps could be utilized in some plants. If the necessary high-pressure alignments are made, the high-pressure service water/residual heat removal systems may be used to keep the core cooled or at least delay the time to core degradation.

During a long-term blackout operators are estimated[5] to have 6 to 12 hours (depending upon dc loads, the heatup rate of the battery room, and the heatup of the pool) to recover ac power from either an onsite or offsite source before the high-pressure emergency coolant injection systems are lost because of battery failure. After coolant injection is lost, there is limited opportunity for core cooling without ac recovery.

If dc power is available, a possible [illegible]ing option is to use the ADS to lower reactor coolant system pressure. This may avoid a high-pressure sequence (although in some plants the ADS valves will reclose if the containment pressure increases above the pneumatic control system pressure), and it may facilitate alternative injection to the core (e.g., from the fire system). However, depressurization may also hasten the core degradation process and add more hydrogen to the containment atmosphere. If the fire pumps are operable, these pumps may be capable of providing injection to the core (depending on the plant configuration) or spraying the drywell and reducing the pressure and temperature in the containment.

The options for operator intervention in a short-term station blackout are much fewer due to the lack of control power. Early recovery of dc power could extend the sequence to a long-term blackout discussed above, but without dc power, ac power cannot be recovered. Without dc power the ADS cannot be operated and the reactor coolant system will remain at high pressure. Even without dc power, water from the fire protection system is still available but may not be at sufficient pressure to supply the reactor coolant system. However, in some BWRs water from the fire protection system could be supplied to the containment spray system.

If ac power is recovered after core damage, it is beneficial to restore injection into the reactor coolant system and start residual heat removal (RHR) cooling of the suppression pool. Some of the issues related to restoring coolant flow to a degraded core were discussed in Chapter 4. However, cooling the debris is the prime concern. Starting the recirculation pumps may enhance cooling of the debris, but there is some potential for causing pump seal failure and aggravating the accident sequence. Operation of the ADS at this stage would allow high flow from the low-pressure injection pumps which would cool the core debris more rapidly.

Besides equipment capability, operator actions and potential operator errors have been found to be important during station blackout. Operator training and procedures specifying the plant parameters indicative of the need for HPCI/HPCS and RCIC initiation can minimize the potential for operator errors. Clear statements in the appropriate procedures concerning the actions required to place these systems into operation and to ensure their continued operation under station blackout conditions are helpful.

If water flow to the core cannot be restored and the reactor coolant system is not depressurized, then core relocation (into the lower plenum) and subsequent vessel lower head failure at high pressure are possible. Under these circumstances the potential for direct containment heating discussed in Chapters 3 and 4 is also possible for BWRs. In the unlikely event that the accident has progressed to this point there appears to be little that can be done by the plant operators to mitigate the effects of a high pressure failure during the blowdown phase. It is again possible for the plant operators to manage containment-related safety features (principally sprays) to help lower the consequences of high pressure ejection. However, it may be more beneficial to eliminate high pressure meltdown through use of the ADS.

After failure of the reactor pressure vessel, there are several possible operator actions which are discussed in Section 5.3.

5.2 Low-Pressure Sequences

The discussion related to low-pressure sequences for PWRs in Chapter 4 is generally applicable to BWRs. Here too the objective is to restore water flow to the core and prevent core meltdown and penetration of the lower vessel head. The time frames for operator action can again be relatively short but the plant operators have an even wider range of systems available to inject water into the core in BWRs.

As noted above, in BWRs the ADS has the capability to convert accidents that are initially at high pressure into low pressure sequences. Even under station blackout conditions, operation of the ADS is possible if a source of dc power is available (as indicated in Section 5.1). Thus, it is possible to have a depressurized reactor coolant system without ac power available. The logic used to automatically activate the ADS in many BWRs is such that automatic actuation will not occur for a number of sequences with loss of high-pressure injection. Intervention by the operator to manually depressurize is required. Changes in the automatic activation logic to eliminate the need for operator action during sequences known to include loss of high-pressure injection could be beneficial. A dedicated dc power supply to ensure depressurization capability for a longer time during station blackout conditions would also improve the ADS.

Some BWRs have diesel-driven fire pumps which could be adapted to provide reactor vessel injection or containment spray. Obviously, the first priority would be to inject into the reactor coolant system but if this is not possible then directing the flow to the containment spray would reduce the containment pressure and temperature and flood the drywell floor. Flooding the drywell floor would help to cool the core debris after it penetrates the lower vessel head (refer to Section 5.3).

After core damage it is also possible that power may be restored and coolant injection flow restarted. The issues related to restoring water flow to a degraded core have been discussed in Section 4, but again the need to flood the vessel and cool the core debris is the main objective.

If water flow to the core cannot be restored, then the core materials will eventually penetrate the reactor vessel. Possible operator actions after vessel penetration are discussed below.

5.3 Ex-Vessel Sequences

If the core materials cannot be cooled in the reactor pressure vessel, they will eventually penetrate the vessel head. The long-term objectives after vessel penetration are the same for BWRs as for PWRs, namely to try to flood the core debris with water and to maintain or restore containment sprays. However, there are differences between the containment designs that are worth noting.

The composition temperature, and mass of the corium released from the vessel are equally as uncertain for BWRs as they are for PWRs. However, the corium presents more of a direct threat to containment integrity for some BWPs than it does for PWRs. Thus, uncertainties in the mass and temperature of the corium leaving the vessel can lead to very conservative results if bounding assumptions are made. For example, if it is assumed that a large mass of molten corium is released from the vessel in a BWR with a Mark I containment, then the corium could be postulated to flow across the drywell floor and melt through the primary containment resulting in early containment failure. This potential failure mode remains uncertain, but is possible. It is not know if water on the drywell floor (or in the reactor cavity) will preclude core debris from flowing across the floor and interacting with the drywell shell. Alternative strategies are being investigated which address the consequences of core debris reaching the containment pressure boundary.

If the drywell floor is not covered with water at the time the core debris penetrates the reactor vessel lower head, it is advantageous to attempt to get water to the core debris even after vessel failure. Because the operators will be unable to determine the extent of core damage, they would first attempt to restore water to the vessel. If the vessel is penetrated, the water will reach the corium through the breach. However, pouring water onto a small area of the corium under the vessel may not be as effective as spray operation. Sprays have the additional benefit of removing aerosol fission products from the drywell atmosphere and lowering the pressure and temperature in the drywell. Spray operation is very beneficial even if the containment has failed because it floods the core debris, scrubs the fission products, and reduces the driving force for fission product release.

It is important to note that large concentrations of hydrogen could accumulate in BWR Mark III containments under station blackout conditions because the igniters would not be operable. If power is restored and the igniters are activated then a damaging burn or detonation may occur. Thus, under station blackout conditions, it may help to switch the igniter system to the off position so that when power is restored the igniters do not cause a damaging combustion event. Under these circumstances, venting of the containment to reduce the hydrogen concentration may be helpful prior to activating the igniter system.

If the sprays or vessel injection cannot be restored, containment pressurization will continue until structural failure becomes a possibility. Under these circumstances, wetwell venting has been suggested as a way of preventing structural failure. The issues related to venting are discussed in Chapter 4. However, there is an additional issue for long-term station blackout sequences in BWRs, namely whether or not to vent containment prior to loss of dc control power. This would result in venting before core degradation. After dc power is lost, initiating venting may become difficult.

Finally, Mark I and Mark II containments are enclosed in a secondary containment structure (reactor building). This building is very large in volume and contains a large amount of structural surface area. Depending on the location at which material released from the primary containment enters the reactor building, fission products may have to be transported through a tortuous path and over a great distance to be released to the environment (typically via "blowout panels" in the refueling bay). As a result, the reactor building can serve as an effective barrier to fission product release. Events such as the combustion of released hydrogen in the reactor building can significantly reduce the effectiveness of the reactor building in filtering released fission products and, conversely, manual operation of building fire spray systems can significantly enhance fission product scrubbing.

Accident situations could also occur during reactor startup or shutdown. In accord with technical specifications for BWRs with Mark I and Mark II containments, the atmosphere should be inerted within 24 hours once the power level exceeds 15 percent during startup. Likewise, within 24 hours from de-inerting the containment, the power level should be reduced below 15 percent during shutdown. Obviously the danger of burning or detonating hydrogen would be increased if a severe accident were to occur during the period when the containment is in a de-inerted condition.

In summary, accident management strategies can be devised to deal with the potential consequences of severe accidents in BWRs as well as for PWRs. Prior to vessel failure the emphasis is on maintaining core cooling via either high-pressure systems or, if this is not possible to depressurize and use low-pressure sources. After reactor vessel penetration by molten core debris the emphasis is on maintaining containment integrity and mitigating fission product release. Limiting the interaction of molten core materials with concrete (and also with steel in the case of the Mark I) can be accomplished via water injected through the failed vessel or by drywell sprays. Sprays will also reduce fission product concentration in the atmosphere. Controlled venting is another means of maintaining containment integrity and controlling fission product release. Because of differences in response, the strategies have different emphasis for various containment designs.

6. SUMMARY

The information on severe accident phenomena and accident management contained in this report is by neccessity abbreviated and subject to uncertainty. Because of the significant severe accident research efforts underway, the understanding of severe accident phenomena and the knowledge of the efficacy of accident management strategies is expected to continually improve over the next several years.

However, accident management strategies can be devised now to deal with the potential consequences of severe accidents in light water reactors based on existing knowledge. Prior to vessel failure these strategies are aimed at providing adequate core cooling to prevent or delay further core damage. If high-pressure injection is not available depressurization of the reactor coolant system should be attempted. Such depressurization not only allows for the possibility of using a number of low-pressure systems to cool the core if they are available but also reduces the threat of direct containment heating by the dispersed core if core melting cannot be prevented.

If molten core debris penetrates the reactor pressure vessel, strategies are directed toward reducing or terminating core-concrete interactions, and preventing the release of fission products to the environment. In both PWRs and BWRs this involves using the available water sources to quench the debris and reduce containment atmosphere contamination. In addition controlled venting has been suggested as a way to avoid structural failure of containment due to overpressurization.

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This report describes the conditions and events that nuclear power plant personnel may encounter during the latter stages of a severe core damage accident and what the consequences might be of actions they may take during these latter stages. The report also describes what can be expected of the performance of the key barriers to fission product release (primarily containment systems), what decisions the operating staff may face during the course of a severe accident, and what could result from these decisions based on our current state of knowledge of severe accident phenomena.

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