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STRATEGIES FOR THE PREVENTION AND MITIGATION OF SEVERE ACCIDENTS

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Abstract

The currently operating nuclear power plants have, in general, achieved a high level of safety, as a result of design philosophies that have emphasized concepts such as defense-in-depth. This type of an approach has resulted in plants that have robust designs and strong containments. These designs were later found to have capabilities to protect the public from severe accidents (accidents more severe than traditional design basis in which substantial damage is done to the reactor core). In spite of this high level of safety, it has also been recognized that future plants need to be designed to achieve an enhanced level of safety, in particular with respect to severe accidents. This has led both regulatory authorities and utilities to develop guidance and/or requirements to guide plant designers in achieving improved severe accident performance through prevention and mitigation. The considerable research programs initiated after the TMI-2 accident have provided a large body of technical data, analytical methods, and the expertise necessary to provide for an understanding of a range of severe accident phenomena. This understanding of the ways severe accidents can progress and challenge containments, combined with the wide use of probabilistic safety assessments, have provided designers of evolutionary water cooled reactors opportunities to develop designs that minimize the challenges to the plant and to the public from severe accidents, including the development of accident management strategies intended to further reduce the risk of severe accidents. This paper describes some of the recent progress made in the understanding of severe accidents and related safety assessment methodology and how this knowledge has supported the incorporation of features into representative evolutionary designs that will prevent or mitigate many of the severe accident challenges present in current plants.

1. INTRODUCTION

From the early days of nuclear power plant development, the possibility of a reactor accident has been recognized. Accordingly, nuclear power plants traditionally have been designed following a concept of defense-in-depth, utilizing multiple barriers to both prevent and/or mitigate the consequences of a reactor accident. This approach typically involves the specification, by national safety authorities, of "design-basis" accidents (accidents that are postulated to occur), which the plants have to be designed to withstand. Nuclear plant designers then engineered plant systems, often assuming a single failure in the system, to prevent and/or mitigate these accidents to minimize the fission product releases to the environment. This approach has, in general, served the nuclear industry well to ensure a low level of risk to the public. Although accidents beyond the design basis accidents, which could result in large releases of fission products, were recognized as conceivable, they were considered highly improbable. Reliance was placed upon the defense-in-depth concept to minimize the likelihood and consequences of such accidents.

With the completion of early probabilistic safety assessment (PSA) studies, such as WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," in 1975, there became a growing awareness that severe accident prevention and mitigation merited further consideration. WASH-1400 also represented a significant advance in the use of PSA methods in reactor safety to identify the more likely accident sequences, and also potential vulnerabilities in plant design that could lead to failures or bypass of the multiple safety barriers. The accident at the Three Mile Island Nuclear Plant, Unit 2 (TMI-2) in 1979, served to crystallize the recognition that severe accidents needed further attention. Initially, this involved an evaluation of the capability of existing plants to tolerate a severe accident. For a number of nuclear power plants, it was found that the design-basis approach resulted in significant safety margins for the analyzed events and that these margins permitted operating plants to accommodate a large spectrum of severe accidents. However, most national authorities responsible for the safety of nuclear installations also undertook a revision of their positions with respect to "beyond design accidents". Regulatory bodies, in general, expected that evolutionary plants would achieve a higher standard of severe accident safety performance than previous designs, and have generated regulatory guidance or requirements to guide the design of future plants. For example, in the United States, the USNRC issued a policy statement on the resolution of severe accident issues in which these expectations for new designs were initially articulated. This policy statement was later followed by more detailed review standards for severe accident issues that went beyond existing requirements. At the same time, nuclear utilities recognized the need to establish a higher standard for future plants to gain acceptance of both regulatory bodies and the public. As a result, additional guidance was developed to be used by designers to address severe accident issues.[1] In Germany and France, the advisory committees Gesellschaft für Anlagen- und Reacktorsicherheit (GRS) and Institut de Protection et de Sûreté Nucléaire (IPSN) developed in a coordinated way a joint safety approach for future pressurized water reactors, which was proposed to and adopted by the national safety authorities Bundesministerium für Naturschutz, Umwelt und Reaktorsicherheit (BMU) in Germany and Direction de la Sûreté des Installations Nucléaires (DSIN) in France in 1993.[2]

In Canada, the regulatory requirements set by the Atomic Energy Control Board (AECB) had always included certain severe accidents within the design basis. The first commercial CANDU, Douglas Point, used a probabilistic approach to licensing, which *ipso facto* included some severe accidents. By the time the Pickering-A plant was licenced in the early 1970s, the AECB requirements for safety design were more deterministic, but still included addressing "dual" failures such as loss-of-coolant plus loss of emergency core cooling (LOCA+LOECC). [3] This concept was further generalized in 1980 during the licensing of the Darlington Nuclear Generating Station near Toronto: a large number of accidents combined with failures of the mitigating systems, had to be analyzed, including some severe accidents. [4] The reason such combined failures could meet design-basis accident dose limits was because of the moderator surrounding the fuel channels: it could accept and remove the fuel decay heat and prevent gross UO_2 melting even with no fluid in the fuel channels. Thus this class of severe accident was "contained" within the fuel channels. However further equipment failures, such as loss of moderator heat removal, which could lead to severe core damage (collapse of the fuel channels in the calandria) were not required to be analyzed at that time. The licensing approach in Canada is largely non-prescriptive (the model is "applicant proposes/regulator disposes"), so the AECB has not yet issued requirements for severe core damage accidents. However AECB stated clearly their expectations that the design of evolutionary CANDU designs would explicitly include both preventative and mitigative measures; at the same time AECL was responding to utility requirements for severe accidents by incorporating appropriate features (discussed below) in the CANDU 9 design. Similarly, in keeping with the philosophy used by these countries, a recent proposal by the International Atomic Energy Agency (IAEA) recommends that severe accidents beyond the existing design basis should be systematically considered and explicitly addressed during the design process [5].

In support of this changing regulatory environment, a large worldwide effort in the area of severe accident research began after the TMI-2 accident to provide the nuclear community with the technical data, analytical methods, and expertise necessary for assessing plant response to a range of severe accident scenarios, assessing the efficacy of various strategies to prevent or mitigate the consequences of severe accidents, including improved design features or accident management strategies, and assessing the consequences of severe accidents. Early containment failure, or avoidance of it, became the focal point of much of this research and is key to many of the safety requirements for new designs, while at the same time research continued to make progress in an understanding of the basic phenomenological issues associated with severe accident progression (e.g., source term, hydrogen generation, fuel-coolant interactions, debris coolability) and in improving the complex computer codes necessary to analyze severe accident issues. This research provides the foundation of information and understanding of severe accident behavior to allow plant designers to minimize or eliminate the challenges to the plant from severe accidents. As a result, for evolutionary nuclear plants, designers are able to take full advantage of the insights gained from probabilistic safety assessments, operating experience, severe accident research and accidents analyzed by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. A recent review of trends in the development of water cooled reactors has resulted in a set of severe accident challenges that are commonly being considered in new plant designs[6]. Among these are challenges from high pressure melt ejection and direct containment heating, hydrogen production and combustion, steam explosions, both in-vessel and in the containment, and core-concrete interactions.

2. DISCUSSION

The large body of research into severe accidents initiated world-wide after the TMI-2 accident has provided not only the basis to address severe accident issues for operating plants but has also provided the knowledge and analytical tools to improve the designs of evolutionary plants to allow them to achieve a higher level of severe accident performance. The progress in understanding the challenges to containment from severe accidents and design approaches being considered to prevent or mitigate these challenges in selected evolutionary designs are presented in the following areas: high pressure melt ejection and direct containment heating; fuel-coolant interaction; debris coolability; and hydrogen combustion. In addition, progress in understanding the potential for in-vessel retention is also discussed as it provides the potential to eliminate several of the challenges to the containment from severe accidents. (The discussion in this paper will focus on the strategies utilized in the evolutionary plants in the US (CE System 80+, ABWR, AP-600), Canada (CANDU 9) and the European Pressurized Water Reactor). In many of these areas, the advances in the understanding of some of the more important severe accident challenges are often leading to common approaches being employed by designers to deal with these challenges.

2.1 High Pressure Melt Ejection (HPME)/Direct Containment Heating (DCH)

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. A degraded core left uncooled will slump and relocate to the bottom of the reactor vessel. If the reactor vessel fails, the core melt can be ejected into the containment cavity under pressure. If the material subsequently should be ejected from the reactor cavity into the surrounding containment volumes in the form of fine particles, thermal energy can be quickly transferred to the containment atmosphere, pressurizing it. The metallic components of the ejected core debris could further oxidize in air or in steam and can generate hydrogen and chemical energy that would further pressurize the containment. This process is called direct containment heating (DCH).

A number of Probabilistic Safety Assessment (PSA) studies for existing reactors, such as NUREG-1150 in the U.S. have identified DCH as one of the major threats for early containment failure for pressurized water reactors. As a result of the concern of early containment failure arising from DCH, this issue has been discussed by the international nuclear safety community for a number of years and a significant experimental and analytical program had been undertaken to support resolution of the DCH issue for existing plants. A recent review of the this large body of work is contained in the recent State of the Art Report issued by CSNI in December 1996.[7] In particular, a large program was undertaken in the U.S. which consisted of (1) integral testing at different scales, (2) separate effects testing, and (3) analytical model development and validation. This research has provided the necessary information for a defense-in-depth approach for prevention or mitigation of DCH as a threat to early containment failure for evolutionary designs.

In select evolutionary designs currently being pursued, HPME has been considered and a combination of design features contribute to the prevention and/or mitigation of the consequences of high-pressure melt ejection. A common provision in minimizing the likelihood of HPME is to ensure that any significant core damage events will occur at low pressure. Other design features contributing to the mitigation of the consequences of HPME (the effectiveness of which has been confirmed by research results) are designs of the reactor cavity to limit debris dispersal and thereby prevent interaction of core debris with the containment atmosphere, and design of robust containments to withstand the predicted loads from DCH. For reactor designs, such as the Combustion Engineering (CE) System 80+, European Pressurized Water Reactor (EPR), Advanced Boiling Water Reactor and Westinghouse AP-600, a key design feature is to incorporate reliable systems for reactor system depressurization to minimize the likelihood of HPME and to reduce the threat of DCH. A second line of defense for the CE System 80+, ABWR, AP600 and EPR designs is to incorporate reactor cavity designs that limits the fraction of core debris dispersed to the upper containment atmosphere, thereby limiting the resultant loads to the containment from HPME and containment designs that will withstand the remaining loads from DCH.

CANDU reactors include forced depressurization through opening main steam relief valves on the secondary side, as well as incorporation of diverse and independent cooling systems to provide for a similarly low probability of a severe core damage accident occurring at high pressure. Further, the design of the CANDU is such that if a total loss of heat sink at high pressure occurred, a limited number of pressure tubes would eventually fail due to over-pressure and over-heating. These failures would depressurize the reactor prior to significant fuel melting, so that any further core degradation would occur at low pressures. The discharge from these failures would be to the liquid moderator, not directly to containment in any case. Thus DCH is not an issue.

2.2 Fuel-Coolant Interactions

Fuel-coolant interaction (FCI) is a process by which molten fuel transfers energy to the surrounding coolant, leading to breakup and quenching of melt with possible formation of a coolable debris bed or, alternately, the fuel-coolant interaction could take the form of an energetic steam explosions that could challenge reactor vessel and containment integrity.

Since the quantification of the containment failure mode induced by in-vessel steam explosion-generated missiles in the Reactor Safety Study, WASH-1400 (identified in the study as the alpha-mode failure), significant progress has been made in understanding the processes and parameters that effectively limit the potential of missile-induced failure by an in-vessel steam explosion. Most recently, in June 1995, a second Steam Explosion Review Group (SERG-2) workshop was convened by USNRC to review the current understanding of the complete spectrum of FCI issues by a panel of international experts. The first Steam Explosion Review Group (SERG-1) workshop took place in 1985. The SERG-2 experts generally concluded that the alpha-mode failure issue was resolved or "essentially" resolved meaning that this mode of failure is of very low probability and of little or no significance to the overall risk in a nuclear power plant. NUREG-1524 [8] was issued in August 1996 summarizing the deliberations of the experts.

The SERG-2 experts noted that with the essential resolution of the alpha-mode failure issue, the emphasis of FCI research shifted to other issues such as mild quenching of core melt during non-explosive FCI, and shock loading of ex-vessel structures arising from explosive localized FCI. These issues are relevant with regard to assessing certain accident management strategies for operating reactors and the adequacy of certain passive system design features of evolutionary water reactors.

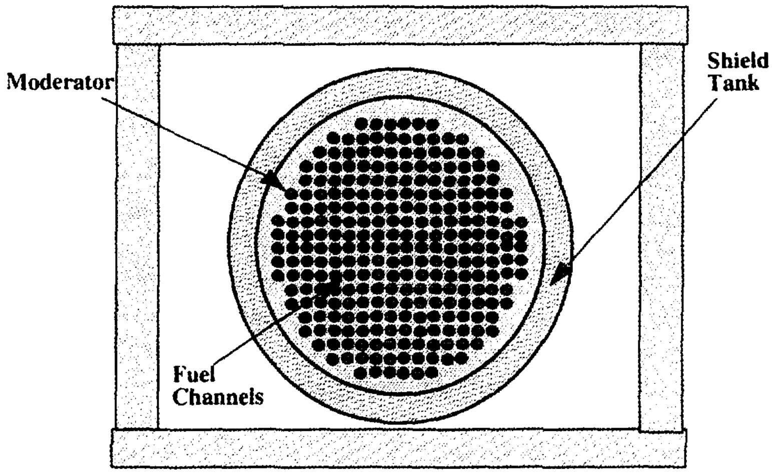

FIG. 1 Emergency heat removal in the CANDU 9

There continues a large body of experimental and analytical work ongoing on in-vessel and ex-vessel steam explosions [9], in programs such as the FARO/KROTOS program at the Safety Technology Institute of the Joint Research Center at Ispra, Italy. Although considerable progress has been made in the basic understanding of FCI, energetic steam explosions in reactor geometries and with reactor prototypic materials can not be excluded at present. Therefore, the energetics and consequential damage from localized FCI must continue to be considered and assessed in the design of evolutionary reactors. In the FARO facility, large masses (typically, up to 250 kg) of prototypic reactor melt are generated and poured into a water pool of varying depths at a range of system pressures. The FARO/KROTOS experiments tests that have been carried out show generally consistent melt quenching with either no steam explosion or very mild energetics. A complementary experimental program recently completed at Argonne National Laboratory has provided information on the energetics, including chemical augmentation, of Zr-water and Zr-ZrO_2 interactions.

Because of the uncertainties in the current understanding of fuel coolant interactions and the likelihood of steam explosions, there is no common approach among the various reactor designs to prevent this challenge. Furthermore, competing strategies for the mitigation of core-concrete interactions and retention of core debris in-vessel by ex-vessel flooding have resulted in different concerns regarding the likelihood of steam explosions. For the ABWR and EPR reactors, the designers have developed designs that attempt to minimize the likelihood of water in the reactor cavity prior to RPV failure and release of molten core debris, thereby limiting the likelihood of energetic fuel-coolant interactions. Balancing the competing interests differently, the designers of the CE System 80+ and the AP-600 do not attempt to minimize the water in the reactor cavities. In fact to attempt to minimize the core concrete interactions for the CE System 80+ and to promote in-vessel retention for the AP-600, both designers take steps to introduce water into the reactor cavities. For these designs, the reactor cavity and containment designs have been designed to be sufficiently robust to survive the calculated loading on the cavity or failure of the cavity walls. Again, the inherent features in the design of the CANDU result in less likelihood of significant melt formation. In a LOCA + LOECC in which the moderator heat removal also fails, the moderator will gradually (hours) boil away. As it does so, upper fuel channels will become uncovered, overheat, and collapse onto lower ones, potentially with some fuel melting inside the channel. The end result is a buildup of coarse debris at the bottom of the calandria, instead of formation of large suspended quantities of molten fuel. Experimental validation of this predicted behavior is required and is now underway.[10] The calandria shell is surrounded by a water-filled shield tank, which will preserve the calandria shell as a fuel debris container [11] until, after about a day, it too has boiled away (this is possible because of the thin (about 1 inch) calandria wall, the large calandria surface area and the low power density of the debris). AECL's approach in CANDU 9 has therefore been to preserve the calandria shell boundary in severe core damage accidents by adding gravity-driven makeup water, sufficient to remove decay heat, to the calandria and to the shield tank (Figure 1).

2.3 Core Concrete Interaction/Melt Debris Coolability

The eventual contact of molten core debris with the concrete in the lower containment will lead to core-concrete interaction (CCI), which can challenge the containment by (1) pressurization resulting from production of steam and non-condensible gases, and (2) base-mat melt-through. CCI is affected by the availability of water in the lower containment and the composition and amount of concrete in the basemat. Melt coolability is essential to prevent both the basemat meltthrough and the continued containment pressurization due to core concrete interaction. The common strategy being pursued by most designs is to provide a large spreading area for the core debris and means to provide water to cool the core. Unfortunately, at this time, the ability to cool the core ex-vessel has not been demonstrated. The currently active experimental research on debris coolability, called the Melt Attack and Coolability Experiments (MACE) program [12], was developed as an extension of the Advanced Containment Experiments (ACE) program under the sponsorship of a number of countries. The MACE program is intended to determine the ability of water to cool prototypic ex-vessel core debris of urania-zirconia composition. To date, this program has not be able to conclude that ex-vessel core debris can be cooled by an overlying pool of water.

The ABWR has incorporated several design features to mitigate the effects of CCI, including basaltic concrete and a drywell flooding system once debris enters the drywell. In addition, the ABWR design provides a large lower drywell floor area to provide for spreading of a molten core, thereby satisfying the EPRI design criterion of 0.02m^2/MWt, a thick reactor pedestal wall and a lower drywell flooding system. Likewise, the CE System 80+ and AP600 have incorporated many features to mitigate CCI. These include large reactor cavity floor areas for spreading core debris sufficient to meet EPRI design criterion of 0.02m^2/MWt and cavity flooding systems which are designed to ensure water is available in the cavity prior to core debris entering the cavity. Similarly, for the CANDU design, if core debris penetrates the shield tank, a large spreading area is provided (approx. O.04m^2/MWt), the location of the reactor just above the floor of the containment (and the fact that there is no basement) will provide for a water layer on any core debris. In addition concrete composition below the reactor is chosen so as to minimize production of hydrogen.

For the EPR design, considerable design and research efforts have been directed at providing an engineered core catcher design that relies on directing the molten debris to a dedicated spreading area that is covered with both sacrificial and protective layers and incorporates cooling from both above and below to prevent stabilized melt, thereby preventing base-mat melt-through. A number of experimental programs have supported this innovative approach, including spreading experiments at the COMAS facility utilizing prototypic material[13].

2.4 Hydrogen Combustion

The major concerns regarding hydrogen are that the static or dynamic pressure loads for hydrogen combustion and detonation may pose a challenge to containment integrity or to the survival or functioning of essential safety equipment. When hydrogen combustion alone is insufficient to threaten containment integrity, combustion may still represent a significant contribution to containment loadings when considered conjunctively with other phenomena.

Research conducted world-wide over the past 17 years has extensively investigated a number of issues related to hydrogen combustion and transport during severe reactor accidents. Much of the work, performed to experimentally investigate the design and evaluation basis for reactor containment performance, focused on global deflagrations of premixed volumes of hydrogen, air and steam. Significant information exists on hydrogen combustion to assess the possible threat to containment and safety-related equipment and to allow implementation of hydrogen mitigation techniques. However, there are still issues related to a better understanding of the hydrogen combustion, particularly as it affects prediction of localized combustion phenomena such as deflagration-to-detonation transition, and the ability of current analytical tools to predict hydrogen distribution in complex geometries. Further, combustion behavior in sub-compartments is uncertain and depends on geometry and hydrogen concentration. Uncertainties remain, but the uncertainties are not considered to be of a nature to prevent implementation of hydrogen mitigation measures in designs of evolutionary water reactors[14].

Again alternate approaches are utilized to limit the challenges from hydrogen combustion. For example, the ABWR utilizes a nitrogen-inerted atmosphere within the containment to prevent hydrogen combustion, while the remaining designs incorporate large containment volumes to limit hydrogen concentrations, use layouts that minimize or avoid internal areas where hydrogen can pocket, and promote natural circulation to provide mixing of the containment atmosphere. Furthermore, for the CE System 80+, AP-600, and CANDU 9 designs ignitors are employed to limit hydrogen concentrations. In addition, in the AP-600, CANDU 9 and EPR designs, passive autocatalytic recombiners are also employed to limit the hydrogen concentrations.

2.5 Reactor Vessel Integrity

During the late phase of a severe accident, a significant amount of core material may relocate downward into the lower head of the reactor vessel or calandria for the CANDU. When this core material is relocated into the lower head of the reactor pressure vessel, a molten pool forms without sufficient cooling and can impose a significant heat load on the reactor vessel lower head. Post-accident examinations of the TMI-2 reactor core and lower plenum found that approximately 19,000 kg of molten material had relocated onto the lower head of the reactor vessel. Both the AP-600 and CANDU designs incorporate features that can provide cooling to the exterior of the reactor vessel and potentially maintain the core material inside the vessel. Knowledge of in-vessel and ex-vessel heat transfer phenomena to the lower head is needed to assess the ability of the reactor pressure vessel to maintain its integrity during a severe accident. A major research project in this area, utilizing prototypic material, that is providing data to assess the likely success of this approach is the OECD RASPLAV project [15] on melt pool natural convection, crust formation and growth, and heat flux distribution on the RPV lower head.

The overall objective of the RASPLAV program is to provide analytical and experimental information that can be used to assess whether, and under what conditions, molten core materials can be cooled/retained inside a reactor pressure vessel. The experimental program includes several integral experiments utilizing ceramic UO_2/ZrO_2 and metallic Zr melt of varying compositions in a slice geometry representing the lower head of the RPV, and a number of separate effects experiments. Three RASPLAV integral experiments have been performed to date, along with a number of smaller scale experiments which are providing necessary data to reach a conclusion on this concept.

3. CONCLUSIONS

Because of the extensive severe accident research efforts over the past 17 years, there is now a large body of technical data and well developed analytical tools to allow for an understanding of severe accident phenomena and to address the severe accident issues on evolutionary reactors to minimize the risk to the public. A number of the major severe accident issues can now be effectively addressed in these designs, while in other areas, the challenges from severe accidents can be minimized. As a result of this effort, evolutionary reactor designs should be able to demonstrate that they have indeed resulted in an enhanced level of safety over existing plants.

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