Evaluations of Fission Product Reduction Strategies for Severe Accident Management in CANDU6 (Postprint)

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Abstract

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Full Text

Preamble

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Evaluations of Fission Product Reduction Strategies for Severe Accident Management in CANDU6

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The present analysis is mainly focused on determining an estimate of the fission products in the release and transport processes, i.e., (1) their release from the reactor core, (2) their distribution in a primary heat transport system, calandria vessel, molten corium, and reactor building, and (3) their release from the reactor building into the environment during a severe accident.

Keywords: CANDU6 Accident Management, Fission Product Reduction Strategy, Containment Filtered Vent System, ISAAC Program
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Introduction

Severe accident management guidance (SAMG) for a CANDU6 type reactor was developed by KAERI (Korea Atomic Energy Research Institute) [1, 2]. This SAMG comprises six severe accident guidelines (SAGs) that describe operator actions that may mitigate or terminate the progression of a severe accident and the release of fission products out of the reactor building. One of these six SAGs addresses the strategy of "reduction of a fission product release." Wolsong unit 1, a typical CANDU6 plant, was recently chosen to be back-fitted with a wet-style containment filtered vent system (CFVS). If the operational strategy of the CFVS is well established, it will greatly reduce the threat from steam overpressurization failure of the reactor building (or containment). In addition, it will minimize the uncontrolled releases of airborne particulate radionuclides and radio-iodine isotopes into the environment through controlled filtering.

During a severe nuclear accident, analysis of the source term and relevant fission product (FP) behavior is essential for accident management of nuclear power plants and emergency planning. This paper presents the mitigation effects of fission product reduction strategies that can be applied to severe accident management for CANDU6 plants. Strategy effects for "control of the reactor building condition" and "reduction of the fission product release" have been evaluated by simulating the operations of a dousing spray system, local air cooler system, and containment filtered vent system as severe accident management strategies. Severe accident phenomenological analyses for the evaluation have been performed using the ISAAC (Integrated Severe Accident Analysis Code for CANDU Plants) 4.03 [3].

The ISAAC code is "a system-level computer code capable of performing integrated analyses of severe accident progression, supporting level 2 probabilistic safety assessment studies or accident management strategy developments" [4]. It was developed based on MAAP4 [4] and predicts accident progression by modeling CANDU6-specific systems and expected physical phenomena based on the current understanding of unique accident progressions. According to a previous analysis [5], the accident progression of a CANDU6 type reactor until corium relocation into the calandria vessel bottom occurs considerably earlier than in typical PWRs or BWRs, resulting from design differences. The OPR-1000, a typical PWR, is equipped with turbine-driven auxiliary feed water systems and main steam safety valves, which can provide a secondary feed and bleed function for the station blackout (SBO) scenario using battery power. The BWR4/MARK1 plant has mitigation systems of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) against an SBO accident, which are operable with no electric power other than battery power. In contrast, the failure time of the calandria vessel in CANDU6 is greatly delayed compared to the others due to additional cooling water in the calandria vessel and calandria vault (CV), and the lower volumetric decay heat power of the molten corium on the calandria vessel bottom. However, containment failure of a CANDU6 plant occurs considerably earlier than that of a PWR owing to lower failure pressure. Thus, CANDU6 has an advantage in maintaining calandria integrity, while an OPR-1000 PWR has an advantage in containment integrity during an SBO accident. Nevertheless, proper mitigative operation of vent systems has the potential to delay or prevent containment failure in CANDU6 and BWR4/MARK1 plants. Therefore, plant responses against severe core damage scenarios like the Fukushima accident are expected to be very different.

The generic models of ISAAC evolved from MAAP4. While some of these models required minor modifications to adapt them to CANDU6 design features and integration with the rest of the code, they remained fundamentally unchanged from the generic MAAP4 models. As a CANDU6 type reactor differs from typical PWR/BWRs, CANDU6-specific features were newly modeled and added to the ISAAC code. These CANDU6-specific models include the calandria vessel, end shields, reactor vault, pressure and inventory control, and engineered safety systems (e.g., dousing, emergency core cooling).

ISAAC evaluates a wide spectrum of phenomena including steam formation, core heat-up, cladding oxidation and hydrogen evolution, vessel failure, corium-concrete interactions, ignition of combustible gases, fluid entrainment by high-velocity gases, and fission-product release, transport, and deposition. The code also addresses important engineering safety systems and allows users to model operator interventions. Furthermore, models are added to characterize actions that could stop an accident, i.e., in-vessel cooling, external cooling of the reactor vessel, and ex-vessel cooling. Mathematical solution techniques are implemented to maintain a quick-running code suitable for extensive accident screening and parameter sensitivity analysis applications.

As described in reference [6], the FP model of ISAAC "contains FP behavior models that calculate the release of FPs from the core and the transport and deposition in the primary heat transport system (PHTS) and in the reactor building (R/B) as well as the release of FPs into the environment. The code sorts the initial masses of 22 FP elements into twelve groups and then tracks the mass of each group in each of four physical states (vapor, aerosol, surfaces-deposited, and corium-contained) in the various components of the PHTS and R/B."

Table 1 shows the twelve fission product groups for the 22 elements. The elements are grouped according to their chemical and transport characteristics. Group 12 is reserved for UO_2 fuel. Some elements such as cesium, rubidium, and tellurium are present in more than one group: Cs and Rb are in groups 2 and 6, and Te is in groups 3 and 11.

A station blackout scenario (SBO) is simulated as the initiating event sequence. The definition of an SBO for CANDU6, quoted from reference [6], states: "In SBO, all off-site power is lost and the diesel generators (DGs) fail. The scenario is considered a very low-frequency, but high-risk, accident event. All current generation reactors are only partially designed to cope with an SBO. During an SBO event, the initiating event is a loss of Class IV and Class III power, causing a loss of pumps used in systems such as the primary heat transport system, moderator cooling, shield cooling, steam generator feed water, and recirculating cooling water." Class IV power supplies AC loads that can be interrupted indefinitely without affecting personnel or plant safety, while Class III power supplies AC loads that can tolerate a short interruption (1 to 3 minutes) required to start standby generators without affecting personnel or plant safety but are required for safe plant shutdown. "The SBO base case does not credit any of these active heat sinks, but relies only on passive heat sinks, particularly the initial water inventories of the PHTS, moderator, steam generator (SG) secondary side, end shields, and calandria vault (CV)" [6].

Brief Outlines of CANDU6 Design Features

The application was performed for Wolsong-1 plant as a reference plant for a typical CANDU6. A CANDU6 is a heavy water-moderated, natural uranium-fueled, pressurized heavy water reactor with a thermal output of about 2,140 MWth. Brief design features of CANDU6 are well described in a previous study [7]: "A PHTS consists of two loops and each loop has 190 horizontal fuel channels that are surrounded by a heavy water moderator inside a horizontal calandria vessel."

The calandria vessel is housed in and supported by a light water-filled steel-lined concrete vault, called a calandria vault (CV), which provides thermal shielding. A significant quantity of heavy water surrounding the fuel acts as a heat sink to remove decay heat after reactor shutdown. The heavy water moderator (2.16×10^{5} kg) in the calandria is a unique CANDU feature that provides a passive heat sink for certain accident scenarios. "The CV is built of ordinary concrete and is filled with about 5.0×10^{5} kg of light water which functions as a biological shield under normal operating conditions and as a passive heat sink under certain severe accident scenarios. For example, if hot dry debris is collected in the calandria bottom after core relocation, the CV water will remove decay heat from the calandria through the calandria wall. In-vessel retention through external vessel cooling, which is considered an important accident management program in a PWR, is inherent in the CANDU design" [7].

As CANDU6 has this additional water inventory, the role of a containment heat removal system such as a dousing spray or local air coolers becomes important to control containment pressure below the failure pressure. "The dousing spray system is designed to limit the magnitude and duration of containment over-pressure. The system is automatically initiated when containment pressure exceeds 14 kPa(g)" [7]. However, the effect of dousing sprays on long-term heat removal is not significant because the sprays are designed only for short-term use. The containment design pressure of the modeled CANDU6 plant is 0.124 MPa(g), the median value of containment failure pressure is 0.426 MPa(g), and the total net volume is approximately 48,000 m^3. At this failure pressure, concrete on the wall starts cracking and pressure can be relieved. Design parameters of the Wolsong-1 plant are summarized in Table 2 .

Analysis Cases and Assumptions

The base case of an SBO accident sequence has been modeled with loss of Class IV and all backup power including loss of all on-site standby and emergency electric power supplies. Reactor shutdown is initiated immediately after accident initiation. The high/medium/low pressure emergency core cooling, crash-cooling function, shut-down cooling, moderator cooling, and shield cooling systems are unavailable. Operator intervention for local air coolers is not credited. The passive containment dousing spray system is working. Main steam isolation valves (MSIVs) are not modeled; instead, turbine bypass valves and turbine stop valves are closed, resulting in the same pressure behavior in the four SGs. Liquid relief valves (LRVs) and pressurizer relief valves are assumed to discharge PHTS inventory into the R/B through the degasser condenser tank.

There is an assumption that relocated molten corium on the calandria bottom will be coolable in the CANDU6 plant because so-called in-vessel corium retention by external vessel cooling is very feasible [8]. For molten core concrete interaction (MCCI) calculation, the molten corium has been assumed to spread across the entire floor of the CV (54.2 m^2).

The base case analysis is followed by three sensitivity cases assuming certain system availabilities to assess their mitigating effects.

Case 1 is similar to the base case, except that the dousing tank inventory is injected into the SGs and no water is available as a dousing spray. The SG crash cool-down is available to depressurize the SGs. The dousing tank inventory is assumed to be injected into four SGs when pressure decreases sufficiently after crash cool-down operation.

Case 2 involves operation of the CFVS. If R/B sprays or local air coolers are unavailable, depressurization by the CFVS can be considered an alternative mitigation strategy. For example, the CFVS is under construction in Wolsong-1. In the CFVS, it is assumed that the orifice diameter is 0.2548 m, the CsI removal rate is 99.99%, and it operates between 225 kPa(a) and 151 kPa(a) [9].

Case 3 is the recovery of a local air cooler system. This system consists of 35 units, of which sixteen local air coolers have a safety function and are capable of operating in the harsh environment caused by an accident. After loss of Class IV power, the sixteen local air coolers operate on Class III power. Twelve out of the sixteen local air coolers with safety function during an accident are assumed to be operable at 10 hours after accident initiation.

Analysis Results

In the base case analysis, the transient is initiated by loss of all AC power with all emergency core cooling systems unavailable, and the sequence progresses to a severe core damage accident. Water in the SG secondary side boils off as heat is transferred from PHTS to SGs. Consequently, SG secondary side pressure increases gradually and the main steam safety valve (MSSV) opens, discharging steam from the SG to the environment outside the R/B. As boil-off proceeds, water in the SGs is depleted at about 2.8 hours (Table 3) and the SGs are no longer a heat sink for the PHTS. Thus, PHTS pressure starts to increase until it reaches the set point of PHTS liquid relief valve (LRV) at 10.16 MPa. Fuel channel dryout occurs as PHTS inventory is gradually lost through the LRVs.

In parallel, as water level in the calandria vessel decreases gradually, the fuel channel with highest decay power ruptures at 4 hours, where moderator cooling is assumed unavailable. After fuel channel rupture and corium relocation, remaining water in the calandria vessel dries out and fission products are released from the calandria vessel into the R/B. The moderator of the calandria vessel is depleted at about 12.2 hours (Table 3). Water in the calandria vessel acts as a heat sink to cool the external calandria vessel wall. Steam generated in the calandria vault (CV) is released into the R/B. When water level in the CV reaches the calandria vessel bottom, the vessel bottom heats up rapidly from decay heat of corium inside, and the vessel fails due to creep. When the calandria vessel fails at 46.2 hours, debris relocates into the CV where it is cooled by water.

Figure 1 [Figure 1: see original paper] shows pressure behavior in the R/B for all cases. After accident initiation, R/B pressure increases gradually because water is discharged into the R/B through PHTS LRVs. Pressure peaks shown in the base case of Figure 1 at approximately 4.0 h, 12.2 h, 22.7 h, 46.2 h, and 130.6 h can be explained by processes occurring at these respective times: (1) fuel channel rupture, (2) water dryout in the calandria vessel, (3) R/B failure, (4) calandria vessel failure and corium relocation into the CV, and (5) corium relocation into the basement after CV bottom failure by MCCI. Water in the

CV reaches saturation temperature at about 17.0 h and begins to boil off, thus gradually increasing R/B pressure. At about 22.7 h, R/B pressure reaches the failure set point of 426 kPa(a), resulting in R/B failure.

In Case 1, the dousing tank inventory is injected into SGs and no water is available for dousing spray. The SG crash cool-down is available to depressurize the SGs, with dousing tank inventory assumed to be injected into four SGs when pressure decreases sufficiently after crash cool-down operation. Decay heat can be removed through the SGs, resulting in significant delay of accident progression as shown in Case 1 of Table 3 and Figure 1.

Analysis results of Case 2 show that R/B pressure is controlled between CFVS actuation and closure pressures during most accident periods, but two peak spikes appear uncontrolled. The first peak occurs at calandria vessel failure (44.2 h) when corium relocates into CV water. The second peak occurs at CV bottom (concrete floor) penetration failure (119.3 h) by MCCI when corium relocates into basement water, resulting in R/B failure.

In Case 3, twelve out of sixteen local air coolers are assumed to be recovered at 10 hours after accident initiation. There is no pressure buildup in R/B due to sufficient cooling capacity of the air coolers.

This analysis focuses on FP release from the core, fission product distribution inside the plant, and fission product release from the R/B into the environment. FP behavior during an accident is strongly dependent on the thermal hydraulic and severe accident phenomenological conditions previously discussed. CsI is selected as the representative FP because, as expressed in a previous study [10]: "Among the FP groups, the noble gas and cesium iodine groups are considered more volatile. Noble gas, which is a chemically inert material, poses considerably lesser danger to human health. Instead, iodine is one of the elements vital to proper functioning of the human body. Iodine exits from damaged fuel rods predominantly as cesium iodine (CsI) during severe accidents rather than as molecular iodine (I_2)."

Figure 2 [Figure 2: see original paper] illustrates CsI behavior as a representative FP for the base case. CsI release into the PHTS occurs after fuel damage. Since fuel channels rupture at 4.0 hours after accident initiation (Table 2), CsI can be transported into the containment. Moreover, when molten fuel relocation starts, CsI is included in the molten corium. CsI in the corium is released into the R/B only after MCCI starts, and water of the calandria vault dries out at about 57 hours. Approximately 5.8%, 91.5%, and 2.6% of the initial CsI mass are distributed in the PHTS, R/B, and environment, respectively. The release time of FPs into the environment is about 22.7 h.

Operation of the dousing spray in Case 1 delays R/B failure and FP release into the environment (Fig. 3 [Figure 3: see original paper]). About 6.0%, 86.7%, and 6.9% of the initial CsI inventory is distributed in the PHTS, R/B, and environment, respectively.

Figure 4 [Figure 4: see original paper] shows Case 2 results where CFVS operation prevents R/B failure for a very long time and reduces the FP release rate significantly. The release rate of CsI is reduced to 0.03%. However, FP release initiation occurs very early (6.5 h) due to CFVS operation. Meanwhile, control of the R/B condition using LACs (Case 3) maintains building integrity without any FP release into the environment due to sufficient cooling capacity (Fig. 5 [Figure 5: see original paper]).

Summary

In this paper, the mitigation effects of tentative FP reduction strategies that can be applied to severe accident management for CANDU6 plants were evaluated based on the ISAAC computer program. Alternate usage of a dousing spray system, recovery of a local air cooler system, and operation of a containment filtered vent system are considered as severe accident management strategies for "control of the reactor building condition" and "reduction of the fission product release." Decay heat can be removed through injection of dousing tank inventory into SGs with crash cool-down operation, resulting in significant delay of accident progression. Meanwhile, even though CFVS operation may not be an effective risk management policy because of early FP release, the strategy can be proposed to prevent catastrophic R/B failure and effectively reduce the amount of FP release. The most effective means for R/B integrity is LAC recovery as long as power can be recovered. Control of the R/B condition using LACs maintains building integrity without any FP release into the environment due to sufficient cooling capacity of the air coolers in CANDU6 plants.

The limitations of this analysis may result from uncertain severe accident phenomena, for example, the fuel melting process, calandria vessel failure mechanism, reactor building failure mode, and fission product release/transport. Uncertainties should be inherently very large for analysis of severe accident phenomena. In addition, an assessment that relies on one computer code is not necessarily reliable in an absolute manner, though a relative comparison of SAMG strategies on the same basis would be very useful.

Unlike a light water reactor, evaluation results for accident management strategies of a heavy water reactor are very limited. Therefore, this study aims at the development of feasible accident management strategies for CANDU6 plants that have yet to be addressed.

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Note: Figure translations are in progress. See original paper for figures.