Nuclear Science and Techniques **22** (2011) 60–64

Core cooling in pressurized-water reactor during water injection

TAO Jun LI Jingxi TONG Lili CAO Xuewu*

School of Mechanical Engineering, Shanghai Jiao Tong University, Shanghai 200240, China

Abstract In this paper, the reactor core cooling and its melt progression terminating is evaluated, and the initiation criterion for reactor cavity flooding during water injection is determined. The core cooling in pressurized-water reactor of severe accident is simulated with the thermal hydraulic and severe accident code of SCDAP/RELAP5. The results show that the core melt progression is terminated by water injection, before the core debris has formed at bottom of core, and the initiation of reactor cavity flooding is indicated by the core exit temperature.

Key words Water injection, Severe accident, Severe accident management guideline, Core damage state, Cavity flooding, PWR

1 Introduction

The core cooling of a pressurized-water reactor (PWR) can mitigate more than 80% core damage in a severe accident by implementing additional or innovative core cooling[1]. The water injection into the reactor coolant system (RCS) may cool the core of a PWR, but the cooling effects on different core damage states shall be analyzed in details, such as the in-vessel corium retention, kinetics of hydrogen production, RCS re-pressurization, and source term[2], because the severe accident progression depends strongly upon characteristics of accident sequences and core damage states. Also, a design against a PWR accident is evaluated with the capabilities and effectiveness of components, equipments and systems to prevent or mitigate severe accidents.

In this paper, SCDAP/RELAP5, the thermal hydraulic and severe accident code, is used to evaluate the reactor core cooling and its melt progression terminating, and to determine the initiation criterion for reactor cavity flooding during water injection for core cooling of a PWR in a severe accident. The analysis results are of help for developing and implementing plant-specific severe accident management guideline (SAMG).

2 Accident sequence selection and analysis

2.1 Accident sequences

According to the screening criteria of selecting severe accident sequences by the U.S. Nuclear Regulatory Commission (NRC), the potential functional sequences that may cause core damage, or poor core cooling, are determined considering in-vessel phenomena[3]: any functional sequence and failure contributing 1.0×10^{-6} or more per reactor year to core damage, and above 5% to the total core damage frequency. These potential sequences of causing core damage or poor core cooling, as described in Refs.[4,5], are station black out (SBO), small break loss of coolant accident (SBLOCA), steam generator tube rupture (SGTR), loss of feed water, and middle break loss of coolant accident (MBLOCA).

2.2 Analysis condition

As a typical three-loop PWR with a rated thermal power of 2895 MW, the core consists of 157 fuel assemblies in a 17×17 grid with an active fuel height of 3.66 m. Each of the three loops contains a U-tube steam generator, a reactor coolant pump, and associated piping. A single pressurizer is attached to the hot leg of one primary coolant loop. The flow rates

---

Supported by National Basic Research Program of China (No.2009CB 724301)

* Corresponding author. *E-mail address:* caoxuewu@sjtu.edu.cn

Received date: 2010-05-20

of total primary coolant mass and total secondary steam mass are 68520 $m^3 \cdot h^{-1}$, and 1614 $kg \cdot s^{-1}$, respectively. The primary loop, with the reactor-pressure-vessel inlet temperature at 292.4ºC and the outlet temperature at 327.6ºC, operates at 100% full power under 157 MPa. The emergency core cooling system, depressurization system, and feed water system are simulated.

While the PWR operates at steady state and 100% full power, all transients begin. All the LOCAs occur at the RCS cold leg between the emergency core cooling system and reactor-pressure-vessel inlet. The emergency core cooling system pumps and emergency diesel generators are unavailable in all base cases. The core degradation in different sequences and PWRs has different timing, but the core melt progressions are similar. The core melt progression can be characterized by the following core damage states (CDS):

CDS1, core uncovery. When the peak core temperature is equal to the saturation temperature of coolant, the core temperature begins to rise rapidly.

CDS2, fuel rod ballooning at peak core temperature of about 1100 K. The heat transfer of the core to coolant is deteriorated *via* coolant flow path decreasing.

CDS3, rapid zircaloy oxidation at peak core temperature of about 1500 K. Hydrogen and a substantial amount of heat are generated by the oxidation.

CDS4, failure and relocation of control rod in the conel grid spacer and other in-core structures at peak core temperature of about 1700 K. The materials are re-solidified in lower core region at lower temperatures to form initial debris. Flow paths of coolant are blocked.

CDS5, formation of large core debris bed at peak core temperature of about 2500 K. The dissolved UO_2 increases with the core degradation. The formed Zr-U-O drops flow downward along fuel rods after the failure of protective ZrO_2 layer. The materials are re-solidified in lower core region. Large debris bed and molten pool is gradually formed.

CDS6, debris bed growth and molten pool crust rupture at peak core temperature of about 2800 K. Upper parts of the core sink into the molten pool located in the lower core region.

CDS7, relocation of core debris and molten pool into the PRV's (reactor-pressure-vessel) lower head, melting at peak core temperature of about 3200 K.

3 Base case analyses

The base cases of peak core temperature, reactor coolant system (RCS) pressure, and reactor pressure vessel (RPV) water level during severe accident progression are analyzed without any mitigation measure, as shown in Fig.1.

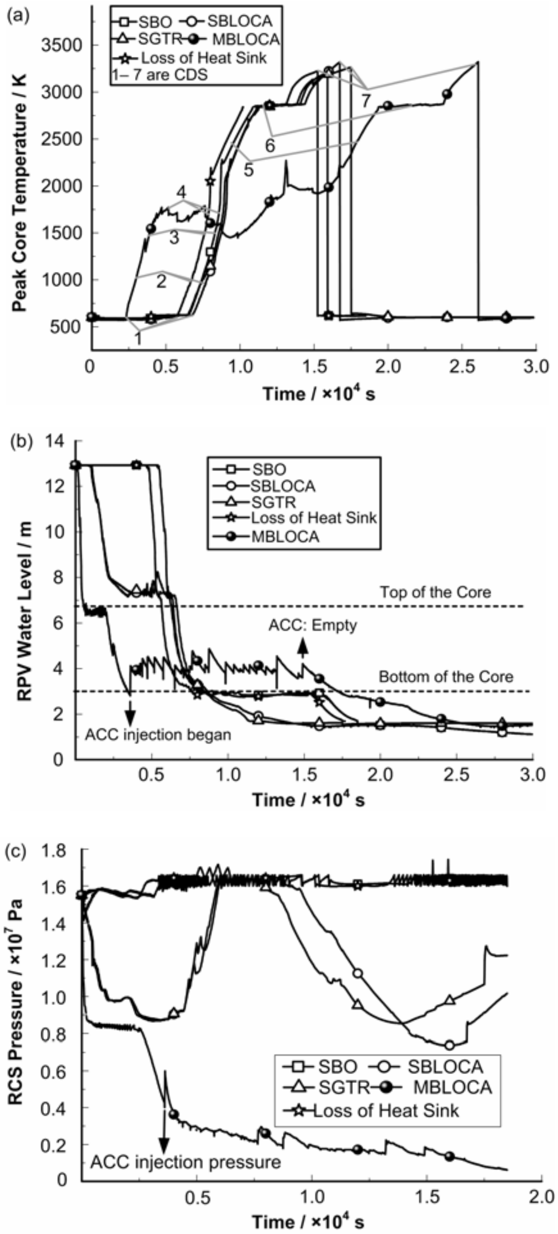

Fig.1 Base cases analyses. (a) Peak core temperature, (b) RPV water level, (c) RCS pressure.

The results indicate that the RPV lower head is damaged by high thermal and structural loads in all sequences, and fails at the primary system pressure of over 10 MPa except for MBLOCA. But high pressure melt ejection or direct containment heating is not predicted by this analysis code, and an alternative tool should be used instead. The RPV lower head fails earlier in high pressure sequences than in low pressure sequences. The RCS pressure in MBLOCA decreases soon to the set pressure of accumulator injection, and the core is re-flooded until water in the accumulator is used up, while the core melt progression is significantly delayed. Hence RCS depressurization has important effect on core melt progression.

4 Water injection for core cooling

4.1 Water injection methods

Water is usually injected into the RCS in a PWR severe accident by restarting reactor coolant pumps, emergency core cooling system, residual heat removal system and accumulator.

The emergency core cooling system pumps are proposed considering the danger of cavitation for restarting reactor coolant pumps and the limited flow rate of residual heat removal system pumps. When the primary system pressure decreases to set-point, water in the accumulator is automatically injected into RCS until drained in the refueling water storage tank, and recirculation water from containment sump is used.

4.2 Depressurization of reactor coolant system

Except for the MBLOCA, the RCS depressurization for investigating core melt progression is conducted by opening the power-operated relief valves (PORV) of the pressurizer in the other sequences. All the three trains of PORV are opened at 650ºC of core exit temperature for sufficient depressurization of the RCS[6]. So the high pressure melt ejection and the containment failure from direct containment heating is prevented. Water is injected into RCS by emergency core cooling system pumps at high flow rate. The core decay heat is removed from primary loop in time. Water in accumulator can be injected into the RCS. And the structural loads imposed on the lower head of reactor pressure vessel are reduced.

4.3 Water injection to terminate core melt progression

The time the closest to core melt progression termination is determined by evaluating the effectiveness of water injection at different CDSs. The core at the CDS7 cannot effectively be cooled by water injection, because only the top steel layer of corium pool touches with the coolant. Fig.2 shows the peak core temperature, reactor pressure vessel water level and RCS pressure in CDS6 with water injection. The results show that molten debris relocates downward into the RPV lower head, and the lower head fails. This indicates that the water injection to terminate core melt progression is too late in CDS6. Similarly, the termination at CDS5 is not effective.

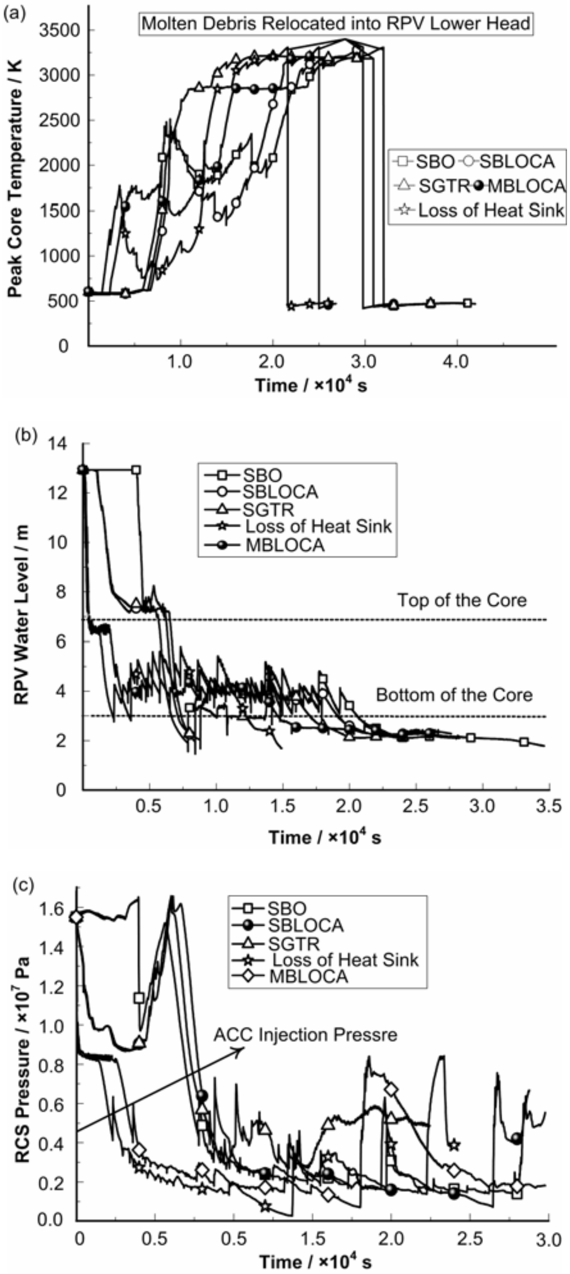

Fig.2 Water injection analyzing at CDS 6. (a) Peak core temperature, (b) RPV water level, (c) RCS pressure.

The peak core temperature, reactor pressure vessel water level and RCS pressure in the case of water injection at CDS4 are shown in Fig.3. When terminating the core relocation, the peak core temperature reaches 2460, 2483, 2467, 2457 and 2453 K in SBO, SBLOCA, SGTR, loss of heat sink, and MBLOCA, respectively. The RPV lower head remains intact, which indicate that the CDS4 for water injection to terminate core melt progression is effective. However, the time intervals between CDS3 and CDS4 in SBO, loss of heat sink, SGTR, SBLOCA, and MBLOCA are 4.5, 2.7, 5.3, 2.7, and 16.7 min (Fig.3a), respectively. CDS3 is closer to the termination of core melt progression with water injection.

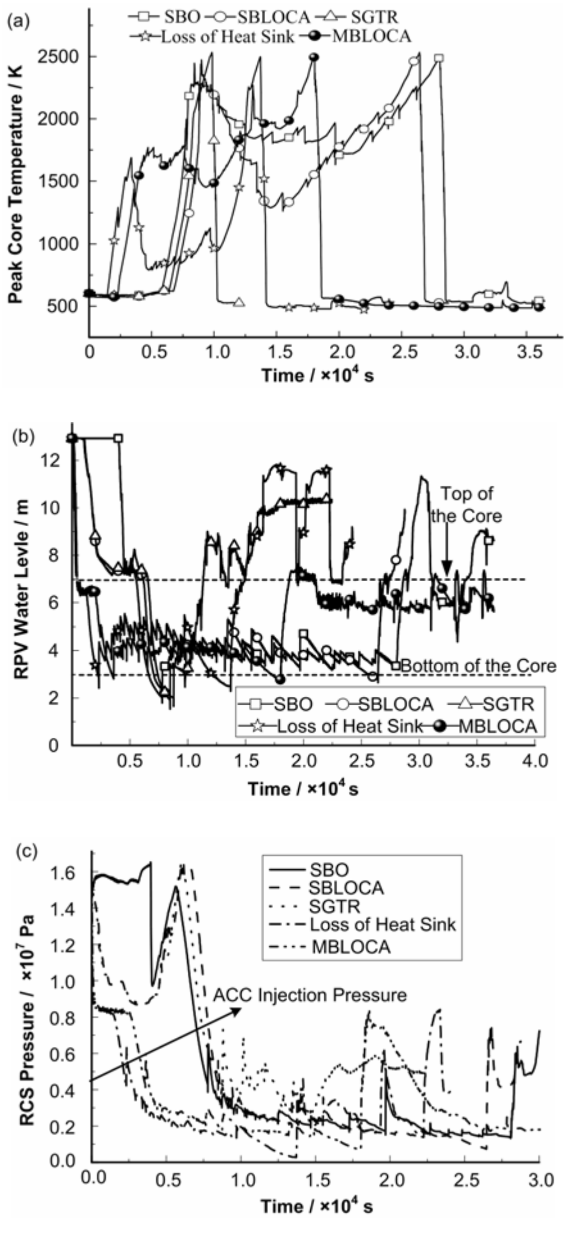

Fig.3 Water injection analyzing at CDS4. (a) Peak core temperature, (b) RPV water level, (c) RCS pressure.

4.4 Indication of the CDSs

From the analyses above, the CDS parameters for severe accident management can be indicated by the peak core temperature. However, computing the peak core temperatures in complicated and high stress environment during a severe accident is not an easy task. On the other hand, as stated in an IAEA safety standard[7], procedures and guidelines should be based on direct measurement of PWR parameters; but the parameter measurements are difficult, too. Therefore, an alternative way of obtaining the parameters indicating the CDSs is to estimate them by simplified computation or pre-calculated graph.

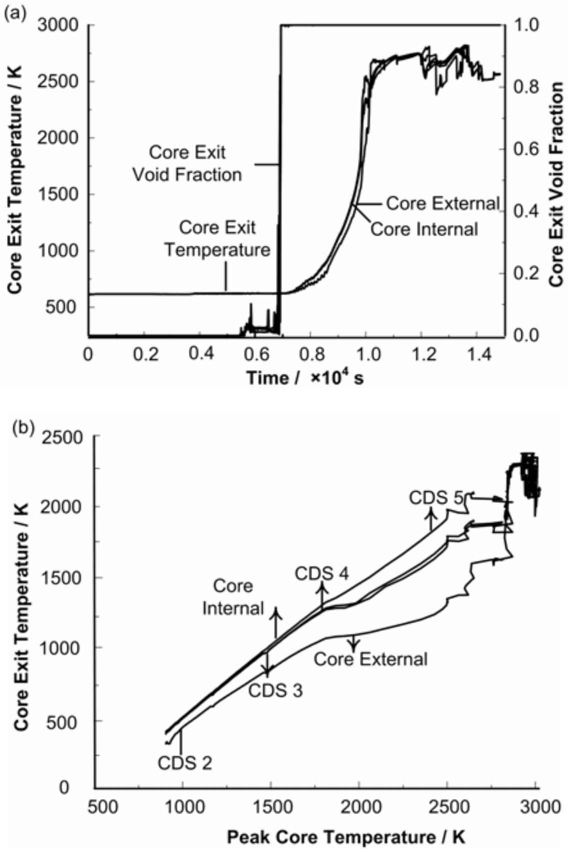

Fig.4 Core exit temperature at different CDSs.
(a) Core exit temperature and core exit void fraction,
(b) peak core temperature and core exit temperature.

A key parameter for developing and implementing the severe accident management guideline is the core exit temperature, at which the transition of emergency operating procedure is triggered. For different techniques of severe accident management, the core exit temperature indicates whether a core cooling is re-established[8]. Fig.4 shows the relationship between the core exit temperature and

CDS using the SBO scenario as a referenced accident. The measured core exit temperature up to the CDS5 well indicates the damage state of internal core channel, and a linear correlation and pre-calculated graph can be developed. Because temperatures of 1500 K and above are difficult to measure, and we propose that the reliable core exit temperature is up to CDS4. When accident progression develops to the formation of core debris and molten pool, such measurable parameters as radiation dose and hydrogen concentration in the containment are suggested to indicate the CDS. The core exit temperatures in CDS4 and CDS3 are about 1350 and 940 K, respectively. Considering the short time intervals between CDS3 and CDS4, 940 K is chosen for terminating core melt progression.

5 Initiation criterion for the cavity flooding measure

When water injection cannot terminate core melt progression, attention of accident management should be paid to delay or prevent the reactor pressure vessel failure and reduce the risk of containment failure. The in-vessel retention (IVR), in which the core is cooled by external rector vessel cooling (ERVC), is a promising strategy for preventing the RPV failure in some existing light water reactors.

Cavity flooding is effective for the IVR-ERVC strategy in PWRs. On developing and implementing the measurement, cavity flooding has a lower priority than that of water injection into the core. When the core melt progression cannot be terminated by water injection after CDS4 at the 1350 K of core exit temperature, and the molten core eventually can be relocated into the RPV lower head, the cavity flooding is implemented. Conservatively, considering the short time interval between CDS3 and CDS4, the initiation criterion is at core exit temperature of 940 K. Nevertheless, effectiveness of the cavity flooding shall be further investigated.

6 Conclusion

Water injection is usually used to cool the core and terminate the core melt progression in PWRs of severe accident. Conservatively, the time the closest to the core melt progression termination is at core exit temperature of 940 K because of the short time interval between CDS3 and CDS4, and as the cavity flooding has lower priority than that of water injection into the core, the initiation criterion is proposed at 940 K of core exit temperature. Considering the reliability of instrumentation, the CDS well indicates by the exit temperature of internal core channel up to CDS4, and its linear correlation with the core exit temperature is developed and pre-calculated graph can be developed before CDS4. When accident progression develops to the formation of core debris and molten pool, other measurable parameters may be suitable for indicating the CDS.

References

1 U S NRC, Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plant, NUREG-1150, Revision 1, 1989.

2 Dorsselaere J V, Fichot F. Nucl Eng Design. 2006, **236:** 1976–1990.

3 Idaho National Engineering Laboratory, Criteria for selecting important severe accident sequence, NRC 10CFR 5.0.54(f), Appendix 2, NUREG/CR-2300, 1998.

4 Southern California Edison. Individual Plant Examination Report for San Onofre Nuclear Generating Station Units 2 and 3, 1993, **1:** 40–51.

5 Omaha Public Power District Fort Calhoun Station. Individual Plant Examination Submittal, 1993, **1:** 37–51.

6 Knudson D L, Rempe J L. Nucl Eng Design, 2004, **230:** 133–150.

7 IAEA, Safety Standards: Severe Accident Management Programmes for Nuclear Power Plants. Safety Guide NO. N-S-G 2.15, 2009.

8 Allison, C M. Nucl Eng Design, 2008, **238:** 1547–1560.