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Study of debris-generated core blockage scenarios during loss of coolant accidents using RELAP5-3D

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ABSTRACT

Two RELAP5-3D models of a typical four-loop pressurized water reactor were prepared to simulate the reactor system response during loss of coolant accident (LOCA) scenarios of different break sizes and locations, under hypothesized debris-generated core blockage conditions. Three break sizes consisting of 2-in., 6-in., and double-ended guillotine (DEG) were selected as representative cases for small, medium and large break sizes, respectively. Simulations were performed to analyze the behavior of the system during a cold leg break and a hot leg break, assuming that all safety systems were available during the phases of the accident. A simpler model was used to perform the simulations up to the long-term cooling phase of the accident, under a full core and core bypass blockage condition. The simulation results help in identifying critical scenarios which, under such circumstances, may lead to core damage. One critical scenario was selected and analyzed with a more detailed core nodalization using RELAP5-3D multi-dimensional components, under different core blockage schemes, including partial core blockage, showing the ability of the cooling water to remove the decay heat from the core under such conditions.

1. Introduction

The emergency core cooling system (ECCS) in a pressurized water reactor (PWR) is designed to provide the required coolant flow to removes the decay heat from the reactor core during a postulated loss of coolant accident (LOCA) scenario, bringing the system to the cold shutdown condition. During the first phase of the accident (blowdown), the ECCS uses the cold water contained in the refueling water storage tank (RWST) which is located inside the reactor containment. The water from this tank is partially injected into the primary system by the injection pumps (low head and high head pumps) and discharged directly into the containment via containment sprays to keep its pressure beneath desired limits. In a subsequent phase of the accident (long-term cooling) when the water of the RWST is depleted, the cooling process continues using the water discharged from the break into the reactor containment and collected in the sump. A set of screens are typically installed for each ECCS train to protect the ECCS components from possible damage induced by the LOCA-generated debris, created during the blowdown phase and transported through the containment floor into the sump by the water flow. A typical strainer contains several modules made of perforated stainless steel plates with hole diameters on the order of a few millimeters. The strainer is designed to reduce the amount of debris that could be injected into the primary system and its impact on the required core cooling (*downstream effects*) as described in NUREG/CR-6808 (2003). Such debris is usually fibers and/or particulates and its characteristics (length, diameters, etc.) depend on the type of insulation and other materials used in the reactor system and containment. The generic safety issue (GSI) 191 addressed the concerns associated with the generation of the debris during a LOCA in light water reactors (LWR), its transport in the containment from the generation site to the sump strainers, and the potential effects that such debris cause to the safety injection performances (in particular to the injection pumps) and to the core cooling capabilities that may be altered by the amount of debris that may bypass the sump strainers, as mentioned in the workshop proceedings of NEA/USNRC (2004). In particular, it is of paramount importance the understanding of the potential degradation of the core cooling capabilities produced by the transport and subsequent accumulation of the debris bypass into the fuel channels in the reactor core and the effects of the coolant flow through the reactor vessel and other parts of the primary systems during such events.

Abbreviations: AFW, auxiliary feedwater; CCW, component cooling water; CL, cold leg; DEG, double-ended guillotine; ECCS, emergency core cooling system; GSI, generic safety issue; HL, hot leg; HPSI, high pressure safety injection; LBLOCA, large break loss of coolant accident; LOCA, loss of coolant accident; LPSI, low pressure safety injection pump; LWR, light water reactor; MBLOCA, medium break loss of coolant accident; MFW, main feedwater; MSIV, main steam isolation valve; PWR, pressurized water reactor; RCP, reactor coolant pump; RHR, residual heat removal exchanger; RWST, refueling water storage tank; SBLOCA, small break loss of coolant accident; SG, steam generator; SI, safety injection; SJ, single junction; TDJ, time-dependent junction; TDV, time-dependent volume.

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System codes have been largely used to simulate the thermal-hydraulic response of a nuclear reactor system under postulated LOCA scenarios. Such codes can provide the user much flexibility in modeling the entire reactor system, including ECCS and other safety features. RELAP5-3D (INEEL-EXT-98-00834, 2005), a thermal-hydraulic system code used for best estimate simulations of normal operation and postulated transients (including LOCA) in LWRs, has been selected to perform the analysis of the reactor systems during selected LOCA scenarios under hypothesized core blockage. RELAP5-3D, developed by Idaho National Laboratory under sponsorship of the U.S. Department of Energy, has been originated from the U.S. Nuclear Regulatory Commission code family RELAP5. It includes all the features of the RELAP5 code family plus the capability to use multi-dimensional components in the model, allowing the user to more accurately model the particular components or regions of a LWR. This includes the lower plenum, core, and upper plenum and downcomer regions of a PWR. An input deck of a typical four-loop Westinghouse PWR plant was developed to perform the analysis of the reactor systems under postulated LOCA scenarios of different break sizes and locations. In particular, the present work provides the simulation result performed for LOCA of three different break sizes representative of the small, medium and large break categories in two selected location of the primary system, cold and hot leg, as suggested in NUREG/CR-6770 (2002), NEI 04-07 (2004), and IAEA Safety Reports (2003). Different core blockage scenarios were assumed to perform the simulations of the reactor response during the long-term cooling phase of the accident to study the possible effects of debris-generated core cooling degradation. Different nodalizations were proposed to perform the analysis. In particular, a more detailed nodalization of the core was assumed in order to study different configurations of partial core blockage.

2. Purpose

The purpose of these simulations was to study the thermal-hydraulic response of the reactor system under hypothesized debris-generated core blockage. In particular, the authors proposed to:

- Analyze LOCA scenarios of different break sizes and locations under the highest conservative hypothesis of full core and core bypass blockage at the bottom of the core;
- Identify the critical scenarios which may produce an increase in the peak cladding temperature and, subsequently, may lead to a potential core damage;
- Select one of the critical scenarios and perform further analysis using a more detailed nodalization of the core.
- Conduct for the selected case additional simulations hypothesizing a partial core blockage and study its effects on the core coolability.

3. General plant description

The power plant under consideration is a typical four-loop Westinghouse PWR. The main features of such system can be summarized as follows:

- Four independent primary loops, identified as A (loop with the pressurizer), B–D;
- Three independent safety injection (SI) trains discharging in loops B–D. Each train is equipped with one high pressure safety injection (HPSI), one low pressure safety injection (LPSI) pump, one accumulator and one sump strainer;
- Safety injection in cold legs, downstream the reactor coolant pumps (RCP) with possible manual switchover to hot leg (the switchover usually occurs after approximately 5.5 h after the break event);
- One residual heat removal (RHR) exchanger connected to each LPSI (downstream) activated during the long-term cooling phase.

4. RELAP5-3D models description

Two RELAP5-3D models were developed and used to conduct the simulations. The *3D Vessel–1D Core model* was used to run the simulations of the LOCA scenarios under a hypothesized full core and core bypass blockage. This model was selected because it combines the detailed nodalization of some regions of the vessel (using multi-dimensional components available in RELAP5-3D), accounting for more realistic flow paths, with the one-dimensional core and core bypass to minimize the simulation time. The following scenarios were simulated using the *3D Vessel–1D Core* model:

- Small break (2″) in cold leg.
- Small break (2″) in hot leg.
- Medium break (6″) in cold leg.
- Medium break (6″) in hot leg.
- Double-ended guillotine (DEG) break in cold leg.
- Double-ended guillotine (DEG) break in hot leg.

Additional simulations were conducted to study the thermal-hydraulic behavior of the core under partial core blockage for a selected case, using the *3D Vessel–3D Core* model. This model simulates the reactor core with multi-dimensional components, allowing partial core blockage (by fuel channel) with a larger simulation time. Both models used for these simulation sets originated from a *Full 1D* model, which will be described in detail.

4.1. Full 1D model

An overview of the full plant RELAP5-3D nodalization adopted for the simulations is provided in Fig. 1.

The primary cooling loops were simulated independently to account for the expected flow asymmetry during the phases of the injection (only three loops are equipped with SI trains). For convenience, the loops were identified with number 1 (loop D), 2 (loop A), 3 (loop B), and 4 (loop C). All the hydrodynamic components (single volumes, pipes, pumps, junctions, etc.) belonging to each loop were identified with a 3-digit number starting with the digit of the loop. In Fig. 1, the SI trains and other components of the reactor vessel are shown with a simple view. A detailed description of the nodalization of the main components of the reactor system and the techniques adapted to model specific parts of the plant are provided below.

4.1.1. Steam generators

The primary (U-tubes) and secondary sides of the steam generators were simulated using one-dimensional components as shown in Fig. 2. The inlet and outlet lower head compartments were modeled with two single volumes (X06 and X10 where X is the loop number) connected with the hot and cold leg of the related loop, respectively. The steam generator tubes were lumped together and simulated using one pipe component (X08). The vertical sections (ascendant and descendant) and the u-shaped top section of the tubes were modeled setting appropriate elevation changes and vertical angles for the subvolumes of the pipe.

The secondary side of the steam generators contains the shell side of the cylindrical section (pipe X70), the separator (X71), and the steam dome (X80). Each steam generator is equipped with main