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Severe accident code-to-code comparison for two accident scenarios in a spent fuel pool

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ABSTRACT

Spent fuel pools (SFPs) are large structures equipped with storage racks designed to temporarily store irradiated nuclear fuel removed from the reactor. SFP severe accidents have long been considered as highly improbable since the accident progression is slow (in comparison with reactor core accidents) and let time to corrective operator actions. However, the accident at the Fukushima Dai-ichi Nuclear Power Plants has highlighted the vulnerability of nuclear fuels that are stored in SFPs in case of prolonged loss-of-cooling accidents and consequently renewed international interest in the safety of SFPs. In this context, the AIR-SFP project, funded by the Euratom 7th FP in the frame of the NUGENIA+ project, was launched in May 2015 with 15 participants. One of the objectives was to assess the applicability of Severe Accident (SA) codes, which were initially developed for reactor applications, to the calculation of transients in SFPs. To reach this objective, a benchmark, including a criticality risk assessment, was carried out. The degradation progression was computed by 14 participants with 6 different SA codes and 5 have participated to the criticality risk assessment. Main results are presented as well as conclusions that have been drawn concerning SA codes readiness to address these "beyond-scope" scenarios.

1. Introduction

Spent fuel pools (SFPs) are large structures equipped with storage racks designed to temporarily store irradiated nuclear fuel removed from the reactor either because it reached the final burnup or because some maintenance operation in the reactor requires it. SFP severe accidents have long been considered as highly improbable since the accident progression is slow (in comparison with reactor core accidents) and let time to corrective operator actions. However, the accident at the Fukushima Dai-ichi Nuclear Power Plants has highlighted the vulnerability of nuclear fuels that are stored in SFPs in case of prolonged loss-of-cooling accidents. Three of the SFPs were located in buildings that lost their roof and there was concern that the pool at unit 4, which had the highest residual heat, might boil dry (but it did not). The

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Fukushima Dai-ichi nuclear accident has consequently renewed international interest in the safety of SFPs and has given rise to the computation of loss of pool cooling (Wu et al., 2014) or coolant (Wu et al., 2015) accidents. In the frame of the SARNET2 FP7 project, several partners performed simulations of accident scenarios in SFP using different Severe Accident (SA) codes (ASTEC, MELCOR, ATHLET-CD, RELAP/SCDAPSIM, ICARE/CATHARE) (Fleurot et al., 2014). The studies raised questions about the reliability of the results obtained since these codes were initially developed for reactor applications. Moreover, the results were difficult to compare since the geometry and the accident scenarios were not identical.

One of the objectives of the AIR-SFP project, funded by the Euratom 7th FP and launched in May 2015 in the frame of the NUGENIA + project, was to assess more precisely the applicability of SA codes to the calculation of transients in SFPs through a benchmark, including a criticality risk assessment. It was the aim of the first Work Package (WP1) to address this issue. The WP1, led by PSI, was devoted to the analysis of SFP transients in the frame of a benchmark on SFP geometry similar to the unit 4 of the Fukushima Dai-ichi Nuclear Power Plant using different SA codes: MELCOR (NUBIKI, PSI, CIEMAT, REL, UJV, SSTC NRS, ENGIE, NRG), ASTEC (ENEA, IVS, LEI, IRSN), ATHLET-CD (GRS), RELAP/SCDAPSIM (LEI) and SPECTRA (NRG). The objective was to identify the discrepancies between the codes and to assess the level of uncertainty of SFP computations carried out with SA codes. For the benchmark, two types of transient were computed: loss-of-cooling and loss-of-coolant accident (LOCA). The geometry, the initial total power and repartition, the boundary conditions were specified by PSI. The conditions and accident scenarios are given in paragraph 2 and the participants and codes used for the benchmark in paragraph 3. The main results obtained for the two scenarios are then described in paragraphs 4.1 and 4.2. The criticality phenomena that are usually not taken into account by SA codes have been investigated in parallel. Some participants (ENEA, KIT, GRS, NRG, LEI) carried out analyses in order to determine under which accidental thermal-hydraulic conditions the criticality limit could be reached. These results are presented in Section 4.3. The conclusions of this benchmark exercise and recommendations of the project's participants are summarized in paragraph 5.

2. Conditions and accident scenarios

The geometry for the code benchmark calculations was selected to represent the Fukushima Dai-ichi unit 4 SFP, which is the most accurately described one due to the strong interest from the research community after the accident and the explosion in March 2011 (Wang et al., 2012; Nuclear Safety NEA/CSNI/R, 2015; SAND2012-6173, 2012; Burnup and Storage data of Fukushima spent fuel assemblies). The SFP is 12.2 m long and 9.9 m wide. The total number of fuel assemblies (FA's) in the pool is 1535 with the average assembly decay heat pattern as displayed in Fig. 1a. For the benchmark, the fuel loading was simplified and FA's were divided in three groups: recently unloaded (548 FA's named "hot" FA's), longer stored (783 FA's named "cold" FA's) and fresh fuels (204 FA's). The total decay power is assumed to be constant during the code calculations and corresponds to 1.9 MW for recently unloaded FA's and 0.5 MW for longer stored FA's. Moreover, the axial power profile in FA's is assumed to be flat. For reasons of simplification, and due to code limitation, the fission product inventory must be considered as the same for low and high decay heat and the inventory for the high decay heat is used.

For the SA benchmark calculation, only one type of fuel assemblies is used (9x9-9 assembly with a central squared water channel. This assembly design is called STEP3B, see Fig. 1c). It is assumed that there are 7 spacers equally distributed. Each FA is surrounded by a steel rack cell and the spent fuel assemblies are stored in 3x10 spent fuel racks (see Fig. 1b), with 53 racks placed in the SFP. The steel walls of the racks are double walls with some space for water in between. For criticality calculations, the burnup after 4 to 5 cycles is assumed to be about 42 MWd/kg with a remaining enrichment of about 0.7%. The mean enrichment of the fresh fuel is about 4%. The distribution of the enrichment and the presence of burnable poison in different rods have been taken from (Suyama et al., 2015).

The two scenarios considered for the benchmark are loss-of-cooling and loss-of-coolant ones. Initially, the water level is at 11.5 m, the air temperature is 30 °C and the water temperature is 40 °C. Another benchmark (Adorni et al., 2016) was carried out some years ago for a total loss-of-coolant accident with experimental data for comparison. Based on this it was decided for this benchmark to select a scenario where the water is allowed to reach boiling condition during the draining of the spent fuel pool. A 2.5 cm diameter and 2 m long vertical pipe is selected to drain the water out of the pool while all cooling mechanisms have failed.

3. Participants and code models

14 participants from 14 countries have participated in the SA benchmark with 6 different computer codes (see Table 1), either

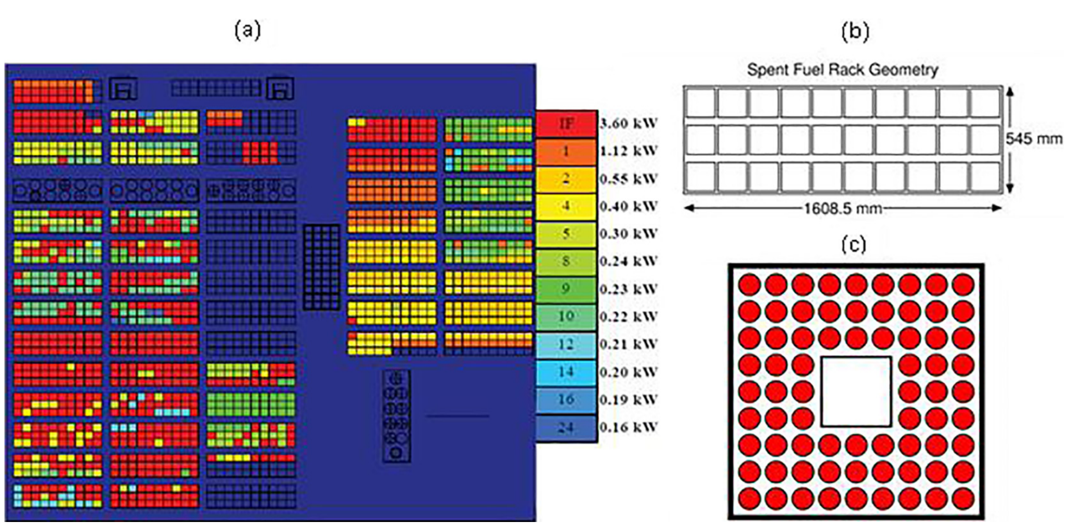

Fig. 1. Layout of Fukushima unit 4 SFP (a), scheme of a spent fuel rack (b) and of a STEP3B fuel assembly (c).

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