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Simulation of FIST Tests Using TRAC-BD1/MOD1

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Several FIST tests were simulated using the TRAC-BD1/MOD1 code at Brookhaven National Laboratory. This work was supported by the U.S. Nuclear Regulatory Commission (NRC) as a part of its independent code assessment program. The TRAC-BD1/MOD1 code is an advanced best-estimate system code developed at Idaho National Engineering Laboratory primarily to analyze postulated accidents and transients in BWR systems.

The FIST (Full Integral Simulation Test) program[1] is a joint project of the NRC, Electric Power Research Institute and General Electric. It was built to investigate small break LOCA and operational transients in BWRs and to complement earlier large break LOCA test results from TLTA (Two-Loop Test Apparatus). The facility is a BWR integral test facility with a full BWR height but volume scaled to 1/624 to the BWR/6 vessel. It has all the prototypical components of a BWR/6 but contains a single electrically heated full-size BWR fuel bundle. The flow areas and fluid volumes in all regions are closely scaled to 1/624. However, because of scaling difficulty, the test facility has a cylindrical external downcomer connected to the main vessel.

The FIST tests consist of two phases (Phase I and Phase II) and the Phase I consists of ten tests[2]. Among these, four tests were selected to be simulated in this study. They were: a BWR/4 MSIV closure ATWS (Test 4PMC1), a BWR/6 small break LOCA without HPCS (6SB2C), a BWR/6 large break LOCA (6DBA1B), and a BWR/6 main steam line break test (6MSB1).

The "VESSEL" component of TRAC-BD1/MOD1 was used to represent the FIST facility. The VESSEL was nodalized with 12 axial levels, 2 radial rings, and 2 azimuthal sectors.

Test 4PMC1 was a power transient simulation test for a BWR/4 with MSIV closure and without power scram. The transient calculation of this test was terminated at 400 seconds since all the significant events occurred during this period and the rest of the transient was predictable. The calculated results were generally in good agreement with the test data, particularly the pressure, steam line mass flow rate, downcomer collapsed water level, and the frequency of SRV opening and closing. However, the code predicted a void fraction higher than the test void fraction in the core. This may affect the core power if the power is calculated by neutron kinetics. The core was always covered and no rod heatup was observed in either the calculation or in the test.

Test 6SB2C was a small break test, simulating a BWR/6 recirculation line break of 0.05 ft^2. The High Pressure Core Spray (HPCS) was assumed to be unavailable. The MSIV was tripped when the downcomer water level reached "Level 1" and the Automatic Depressurization System (ADS) was activated with a 120 second delay. The transient was calculated up to 450 seconds. The results showed generally good agreement with the test data. However, in the calculation, the level 1 was reached about 10 seconds later and the depressurization was slightly slower after ADS activation than in the test. This resulted in about 30 seconds delay in Low Pressure Core Spray (LPCS) and Low Pressure Core Injection (LPCI) initiations. This also caused a delay of the rod heatup.

Test 6DBA1B was a large break test with a 200% recirculation line break for a BWR/6. Additionally two LPCI pumps were assumed to fail. In general, the core predicted the test results very well. The magnitude of the bundle heatup was also adequately predicted by the TRAC-BD1 calculation. However, the core showed a consideraly shorter heatup period than in the test.

Test 6MSB1 was a main steamline break test, simulating a BWR/6 response with a double-ended break at the upstream of the flow limiter in one of the four main steamlines. This test was initially simulated using the correct break area as given in the test report. However, the calculation resulted in much larger beak flow than in the test. Therefore, the calculation was repeated with reduced break area to match the break flow. Even with the break area half of that of the test, the calculated break flow was still substantially higher than the break flow in the test; yet the pressure did not decrease as fast in the calculation as in the test. This indicates that if the break flow was further reduced to match the test data, the pressure would be even higher than in the test. Since the pressure and mass inventory in the system are among the most important parameters determining other behavior in the reactor, the calculation was terminated at this point without further trials reducing the area. It appears that this inconsistent reactor behavior was caused by the faster increase of downcomer water level in the calculation than in the test due to the level swelling phenomenon and more liquid entrained through the break.

It appeared that the TRAC-BD1/MOD1 code adequately predicted the large and small break tests, and the MSIV closure ATWS test. However, it overpredicted the break flow in the main steamline break test. Furthermore, the code did not appear to be completely robust numerically as manifested by occasional failures and the need for restarting with small time steps. The code also needed some manipulation for geometric data such as cell length, area and/or hydraulic diameter around some of the "VALVE" components used to simulate breaks and SRVs, to avoid taking excessively small time steps due to the material Courant limit. This difficulty was caused by the semi-implicit numerical scheme used in the code and is expected to be eliminated in the new code version (TRAC-BF1) with a SETS (Stability Enhancing Two Step) numerical method.

REFERENCES

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