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Safety Analysis in AP-1000 Nuclear Power Plant Against Primary Loop Coolant Pump and Fuel Failures

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Abstract
This study focused on the best-estimated thermal-hydraulic calculations performed regarding the AP-1000 nuclear power plant (NPP) under primary coolant loop pump and fuel failures. Primary coolant pump and fuel failure transients are important design-basis accidents in NPP safety assessments. This work examined the primary reactor’s safety margins of fuel and cladding peak temperature, primary and secondary loop pressure, hydrogen formation, and primary and secondary flow rates. The Adaptive System Thermal-Hydraulics program was used in the simulation and event progression analysis. The results showed that the peak fuel surface temperature was kept below the licensing design limit. No observation of corium was seen during the transient calculation. The structural integrity of the reactor was not jeopardized by changes in the flow rates and pressure of both primary and secondary coolant loops. This result validates the defense-in-depth concepts and safety principles incorporated into the design of the AP-1000 NPP. Finally, based on observations of the beneficial effects in controlling accidents aided by the scram of control rods and reactor thermal-hydraulic feedbacks, the AP-1000 NPP can be concluded certainly to be safe against the combination of primary coolant loop pump and fuel failures. These findings provide useful information for assessing reactor safety capacities to withstand the effects of coolant flow loss and fuel failure at power.

Keywords: AP-1000, fuel, pump, safety, coolant, Adaptive System Thermal-Hydraulics
1. Introduction
The AP-1000 pressurized water nuclear reactor designed by the Westinghouse Electric Company has enhanced passive safety systems as well as design elements aimed at lowering capital costs and enhancing economics [1]. The AP-1000 plant—a tow-loop pressurized water reactor (PWR)—takes a simplified, cutting-edge, and successful approach to safety [2]. Advanced passive safety measures and significant plant simplifications are part of the AP-1000 design, which improves the plant’s construction, operation, and maintenance [3]. The plant’s architecture relies on tried-and-true technology and builds on approximately 40 years of PWR operation. Around the world, PWRs account for 74% of worldwide light-water reactors. Most of these reactors are based on Westinghouse PWR technology [4].

The configuration and passive safety features of the AP-1000 reactor are the same as that of the AP-600. To accommodate the higher power rating, the capabilities of the main reactor components have been enhanced. The process of design for the containment cooling and core cooling passive safety features involves assessing each feature to see whether any adjustments are required to then provide enough safety margins at the higher power rating [5]. According to preliminary safety evaluations, the AP-1000 passive safety systems exhibit satisfactory effectiveness in limiting design-based accidents (DBAs) [6]. However, other hypothesized improbable scenarios should be examined to prevent uncontrollable mishaps and improve this reactor’s emergency readiness.

Some examples of DBAs in AP-1000 include fuel failure and loss of reactor power followed by safety control rod axe man (scram) accidents. They happen as a result of a rupture, fuel creep, or loss of all external alternating current supply [7]. Studies have been done on the effect of power loss incidents on a nuclear power plant’s (NPP’s) transient responses. They show that the responses change depending on the type of reactor and position at which the rupture occurs [8]. The potential reactions of the reactor in the event of a fuel failure, followed by loss of integrity of primary pumps, is not well understood. These scenarios can be quite terrifying.

In light of these accidents, this research concentrated on the investigation of the changes to the AP-1000 reactor. The postulated events were fuel failure in the reactor at full power, followed by reactor coolant main pump failure. The pump in the primary loop, which is connected to the pressurizer, was assumed to fail. Calculations for steady-state and transient scenarios were made using the Adaptive System Thermal-Hydraulics (ASYST) program. Investigations were conducted into how the reactor’s characteristics responded along the transient runtime, such as core heat removal, core damage, and cladding failure. The ASYST program has been shown through previous research to be a dependable simulation tool because produces results that agree with those of other safety models and AP1000 design safety document [9]. The International Atomic Energy Agency has also suggested ASYST as a technique for safety evaluation in both operational and under-design NPPs.
Decisions on potential occupational, public, and environmental radiation exposure could benefit from these calculations. The calculations could also be used as a guide for modifying core and reactor safety mechanisms in the future. This study may be used as needed for generation IV safety system modifications and generation V and small modular reactor safety system preparation.

2. A Brief Description in AP-1000 Safety Design and ASYST Program
The fundamental system architecture of the AP-1000 is a traditional two-loop, two-steam generator setup that has been upgraded in a number of ways, as shown in Figure 1. The core power of the AP-1000 is 3400 MW(t) with a nominal output of 1117 MW dependent on on-site conditions. The core, similar to Doel 4 and Tihange 3, includes 157 fuel assemblies (Figure 2). The containment cooling and passive emergency core cooling systems are available on the AP-1000 [10]. These systems imply that in AP-1000, simpler, passive systems depending on gravity, compressed gases, or natural circulation to operate them instead of pumps have taken the role of active systems required only to alleviate DBA situations. Additionally, AP-1000 does not require safety-grade alternating current power supplies. When an improbable circumstance compels the activation of the passive emergency system, Class 1E batteries supply the necessary electrical needs [11].
Plant operability and maintainability are important aspects of the AP-1000 design philosophy. The AP-1000 design incorporates features such as a simplified system design to improve operability with fewer components and associated maintenance requirements. The AP-1000 safety’s systems in particular reduce surveillance requirements by enabling significantly simplified technical specifications [2].

To guarantee a high level of reliability with little maintenance needed, emphasis has been placed on the selection of tried-and-true components [12]. Component standardization eliminates the need for spare parts, cuts down on maintenance and training needs, and permits shorter maintenance intervals. For important components, built-in testing capability is included [13].

The design of the plant enables easy access for maintenance and inspection. Equipment staging, removal routes, and room for mobile units and remotely operated service equipment are all provided by laydown space. At strategic areas, access platforms and lifting equipment are available, along with utility services such as demineralized water, electrical power, breathing and service air, lighting, and ventilation [14].

Principles for reducing radiation exposure are also incorporated into the AP-1000 design to keep the occupation-to-dose ratio as low as reasonably achievable (ALARA). The design considers the following key criteria: source reduction, shielding, distance, and exposure length. To satisfy the ever-tightening standards for NPP safety, the AP-1000 safety concept considers the most recent global trends in this area. The need to enhance the NPP’s economic effectiveness was also considered. Through the elimination of components, the reduction of bulk quantities, and the reduction of building
sizes, various characteristics are included in the design to shorten costs and construction time [15].

b. ASYST in NPPs
The pressurized water reactor systems’ performance is forecasted by the ASYST computer program in both steady-state and transient situations. Reactor Excursion and Leak Analysis Program (RELAP) and Severe Core Damage Analysis Package (SCDAP) are complementary sections of it. RELAP5 calculates the thermal-hydraulic responses on all coolant systems as well as the kinetic conditions of system elements such as pumps and valves. SCDAP calculates changes in the core and vessels. Additionally, it computes the production of debris and molten pools, interactions between debris and vessels, creep rupture, and structural failure during serious incidents [16]. As a result, ASYST is a reliable thermal-hydraulics program that may be used to explain fuel and pump failure in the AP-1000.

3. Study Simulation

a. Validation of ASYST Simulator Program
Validation of safety analysis codes is required in NPPs. Validation was required to ensure that the simulator used in this study was complete and correct. Firstly, a functional fidelity representing the AP-1000 reactor’s design was assembled. Secondly, the systematic representation was validated by the ASYST. ASYST had to reproduce the reactor system’s nominally measured steady-state condition. Finally, ASYST had to demonstrate an acceptable condition for time-dependent scenarios.

b. Description of Conditions and the Modeling Procedure in Pump Failure Accident
Hydrodynamics, control systems, heat structures, and neutronics were the divisions of ASYST input. The solid components of the reactor model were represented by thermal and hydrodynamic structures. The number of tripped reactor coolant pump (RCP) sets determined the sequence of events and the design operation of the systems. The current analysis examined two cases of RCP sets tripping while the reactor was running at full power—two-trip of two RCP sets.

Despite the scram command for control rods being sent instantly during a pump failure accident (in both loops), the control rods began falling after 1.4 s because of the inevitable delay time in protection system performance. However, in the event of a pump failure accident in two loops, the delay time for control rods to fall was 6.5 s.

4. Results and Discussion

a. Steady-State Results
Before simulating the accident’s condition, understanding the start and boundary conditions, which should be as close to a real AP-1000 plant as possible, was required. This understanding aided in certifying the simulation program, resulting in reliable
simulated results. A steady-state calculation was done to achieve real plant conditions. Following that calculation, ASYST ran for a while until all the plant’s main variable values attained stability around their nominal values. This steady-state calculation ran for 20 s. The results of the steady-state calculation are shown in Table 1. The results were also compared with the AP-1000’s design specifications. The comparison among plant variables revealed that the modeled results were within acceptable bounds. As a result, this work was convincing toward carrying out a transient study on the built input for AP-1000 on the ASYST program.

Table 1. Design and steady-state values.

<table>
<thead>
<tr>
<th>Unit</th>
<th>Design value</th>
<th>Steady-state value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor power</td>
<td>MWth</td>
<td>1,090</td>
</tr>
<tr>
<td>Steam pressure</td>
<td>Psia</td>
<td>1200</td>
</tr>
<tr>
<td>Clad average temperature</td>
<td>°F</td>
<td>566</td>
</tr>
<tr>
<td>Feedwater temperature</td>
<td>°F</td>
<td>440</td>
</tr>
<tr>
<td>Main steam flow rate</td>
<td>kg/s</td>
<td>1889</td>
</tr>
<tr>
<td>Steam pressure</td>
<td>MPa</td>
<td>5.76</td>
</tr>
<tr>
<td>Core coolant outlet temp</td>
<td>°C</td>
<td>324.7</td>
</tr>
</tbody>
</table>

b. Safety Analysis of ASYST Results for Primary Loop Coolant Pump and Fuel Failures

In the simulation, the reactor scrambled at 13.4 s. After the scram, the core thermal power fell to below 250 MWth in 90 s (Figure 3) and then continuously slowly decreased until the end of calculation. This decrease was a result of the reactor being in shut mode. The decrease also indicated that the reactor was under subcritical once the scram happened. This result confirms the prompt role of the safety systems. When the pump’s function was lost, the primary pressure in the pressure vessel instantly went up to a maximum of 15.7 f from 15.5 MPa in 20 s. This increase might be a result of continuous heat generated by reactor fuel. Then, the pressure dropped below 15.25 MPa in approximately 30 s. Safety valves on the pressure vessel might have partially opened to depressurize the reactor. This opening is a passive safety response. The primary pressure gradually increased once the safety valves closed until it normalized at approximately 15.8 MPa in 1400 s, and pressure remained constant up to the end of simulation (Figure 4). This pressure is within the design limit of the AP-1000. A long time simulation would show more changes in the primary pressure; thus, this study proposes that work in the future. The pressures in both steam generators (SGs) followed the same trend as the primary pressure; SG pressures increased and peaked at 7.3 MPa at 100 s (Figure 5). These pressures then fell below 10 psi in both SGs. This result indicates that the tubes in the SGs were not compromised.
Figure 3. Core thermal power.

Figure 4. Pressure in pressure vessel.
According to Figure 6, peak temperatures fell to 350 K from 1800 K after 75 s. Pressure then slowly, continuously dropped until the end of the ASYST run. This result affirms that fission in the fuel was stopped by the reactor safety control. The peak temperature of the cladding increased once the pumps lost their function. This temperature increase happened because the primary coolant could not leave the reactor vessel; thus, it continued absorbing the heat. After 110 s, the temperature then started to drop as a result of the reactor’s depressurization or passive coolant injection in the pressure vessel (Figure 7). Similarly, the water level in the pressure vessel increased once the coolant was trapped in the core, which resulted in water heating up more and increasing in volume. The water level then also dropped sharply once reactor depressurization happened at approximately 150 s. Thereafter, a slow increase was observed, which was due to decay heat originating from the reactor core (Figure 8). The presence of primary coolant in the core was critical in keeping the reactor under control.
Figure 6. Peak fuel temperature.

Figure 7. Peak cladding temperature.
The boiling departure ratio suddenly increased from 1.5 to 50 in 10 s of the transient run and then slowly increased up to the end of the calculation (Figure 9). During the reactor transient, no H₂ was produced by Zr–H₂O (Figure 10). Finally, during the simulation period, neither substantial radiation traces nor corium were observed.
5. Conclusions
This study examined the thermal-hydraulic parameters of the AP-1000 in the event of a primary coolant pump and fuel failure. After a successive steady-state run for 20 s, accident conditions were introduced. This study found that the fuel rod cladding temperature was below 500 K, which is a significantly lower value than prescribed DBAs (1200 K). This result is the major indicator that the reactor is in safe mode. Furthermore, primary and secondary coolant pressures were also lower than the operation values. This result confirms the rest of the system’s structural integrity. The ASYST simulation showed that the probability of fuel cladding failure is 0% in the AP-1000, indicating that no clad oxidation occurs. No changes occurred in reactor core geometry, and no corium was formed. Therefore, this study concluded that the AP-1000 reactor has enough provisions to mitigate the effects of a primary coolant pump and fuel failure without the operator’s intervention. These findings can also be used to raise public awareness, particularly among those living near potential sites, and to confirm the core and safety systems of high engineering standards.

6. Acknowledgments
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