Evaluation of Reactor Safe Shutdown Temperature and Safe Shutdown State Duration

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Abstract

According to the User Requirements Document (URD), for reactors with a safety-grade decay heat removal system, without reliance on AC power sources and under the assumption of a single failure, the reactor must be capable of scramming from rated power operation and cooling to a safe and stable state (safe shutdown state) within 36 hours. Currently, the safe shutdown state for passive pressurized water reactor nuclear power plant reactors is defined as the reactor coolant average temperature being below 216°C. After the reactor system reaches the safe shutdown state, the duration for which the safe shutdown state can be maintained relying solely on the capability of the Passive Residual Heat Removal System (PRDHR) characterizes the heat removal capacity of the PRHR. This analysis aims to evaluate the heat removal capacity of the PRHR following a design basis transient shutdown in a passive pressurized water reactor nuclear power plant, determine whether it can reduce the core coolant average temperature to the safe shutdown temperature of 216°C within 36 hours, and assess whether the PRHR can maintain the reactor in a safe shutdown state for 14 days after the accident. The analysis results indicate that for passive pressurized water reactor nuclear power plants, following the loss of AC power to nuclear power plant auxiliary systems and reactor shutdown, the passive safety system capability can satisfy the requirement to reduce the reactor coolant average temperature to 216°C within 36 hours and maintain the safe shutdown state for 14 days after the accident.

Full Text

Preamble

Article Number: Evaluation of Reactor Safe Shutdown Temperature and Safe Shutdown State Duration

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Abstract

According to the User Requirements Document (URD), for reactors with a safety-class decay heat removal system, the reactor must be capable of achieving an emergency shutdown from 100% full power and cooling to a safe stable state (safe shutdown state) within 36 hours without reliance on AC power and assuming a single failure. For passive PWR nuclear power plants, this safe shutdown state is defined as an average reactor coolant temperature below 216°C. After the reactor system reaches the safe shutdown state, the duration for which this state can be maintained solely by the Passive Residual Heat Removal (PRHR) system capability characterizes the PRHR heat removal capacity. The purpose of this analysis is to evaluate the PRHR heat removal capacity following a design basis transient shutdown in a passive PWR nuclear power plant, determine whether it can reduce the average core coolant temperature to the safe shutdown temperature of 216°C within 36 hours, and assess whether it can maintain the reactor in a safe shutdown state for 14 days post-accident. The analysis results demonstrate that for passive PWR nuclear power plants, following a reactor shutdown due to loss of auxiliary AC power, the passive safety system capability can satisfy the requirement of reducing the average reactor coolant temperature to 216°C within 36 hours and maintaining the safe shutdown state for 14 days after the accident.

Keywords: Safe shutdown temperature; Loss of normal feedwater combined with loss of AC power accident (LONF/LOAC); PRHR heat removal capacity; Passive safety system

0 Introduction

According to the User Requirements Document (URD) [1], for reactors with a safety-class decay heat removal system, the reactor must be capable of achieving an emergency shutdown from full power operation and cooling to a safe stable state within 36 hours without reliance on AC power and assuming a single failure. The National Nuclear Safety Administration document "Notice on Issuing the 'CAP Series Nuclear Power Plant Safety Review Principles' " [2] defines this stable state for passive PWR nuclear power plants as 215.6°C. After the reactor system reaches the safe shutdown state, the duration for which this state can be maintained solely by the Passive Residual Heat Removal (PRHR) system capability characterizes the PRHR heat removal capacity.

To meet URD requirements, long-term analysis of non-LOCA accidents is necessary, considering the effects of reduced PRHR heat removal capacity after PRHR dryout, CMT actuation impacts during the long-term process, and the influence of containment backpressure and condensation reflux rate after PRHR dryout to verify the long-term heat removal capability of the PRHR system. Through accident screening, the Loss of Normal Feedwater combined with Loss of AC Power (LONF/LOAC) is identified as the most limiting event for demonstrating PRHR long-term operation. Therefore, this event is selected as the limiting condition for PRHR performance analysis.

Following a total loss of main feedwater in the nuclear power plant, the reactor will be shut down by the Steam Generator (SG) narrow-range low water level signal, which subsequently triggers turbine trip and causes loss of auxiliary AC power. The Reactor Coolant Pumps (RCPs) will coast down due to loss of power, and the Passive Residual Heat Removal Heat Exchanger (PRHR HX) will be actuated to cool the Reactor Coolant System (RCS) through natural circulation. Additionally, the RCS cold leg low temperature signal will trigger Core Makeup Tank (CMT) actuation.

This analysis has two objectives: first, to evaluate the reactor safe shutdown temperature by verifying the heat removal capability of the passive residual heat removal system after a design basis transient shutdown in a passive PWR nuclear power plant, and determine whether it can reduce the average core coolant temperature to the safe shutdown temperature of 216°C within 36 hours; second, to evaluate the reactor safe shutdown state duration by analyzing how long the PRHR heat removal capacity can maintain the reactor in a safe shutdown state after the baseline transient shutdown. Based on operational experience, the safe shutdown state must be maintained for at least 14 days.

1.1 Accident Scenario Selection

The loss of heat sink accident represents the limiting transient for demonstrating PRHR long-term operation. Three loss-of-heat-sink events exist: Loss of Normal Feedwater (LONF), Loss of Normal Feedwater combined with Loss of AC Power (LONF/LOAC), and Loss of AC Power (LOAC). Among these, the LONF/LOAC event is the most limiting transient. In loss-of-feedwater events, the PRHR HX operates in forced circulation until the reactor coolant pumps stop, which results in greater heat removal by the PRHR. Consequently, this scenario is bounded by the LOAC and LONF/LOAC accidents (where PRHR operates in natural circulation mode). For the LOAC event, the reactor coolant pumps trip at the beginning of the transient, and with loss of feedwater, the reactor is shut down early by the low reactor coolant pump speed signal. Compared to the LONF/LOAC event, the Steam Generator (SG) remains filled with water inventory at this time and can effectively remove core decay heat when the SG narrow-range low water level signal is reached. Therefore, in the LOAC event, a larger portion of core decay heat is removed through the SG. In summary, LONF/LOAC is the most limiting event for demonstrating PRHR long-term operation. Thus, this analysis selects the loss of feedwater accident combined with loss of auxiliary AC power (the LONF/LOAC accident) to demonstrate the capability of passive safety systems to bring the reactor to a safe shutdown state.

Bringing the plant to 216°C (420°F) within 36 hours and maintaining this condition for 14 days post-accident represents a long-term safe shutdown state. While not an acceptance criterion in the safety analysis report, this requirement is still considered in the basic design specifications. The evaluation considers only safety-related equipment and employs a conservative analytical approach to verify its ability to meet design criteria. The safety analysis report uses bounding and conservative assumptions for design basis analysis methods to demonstrate that the plant meets safety evaluation acceptance criteria even under the most limiting conditions. However, this analysis remains conservative for the expected plant response, employing a conservative yet non-bounding analytical approach to better demonstrate the expected plant and operator response.

After reactor shutdown, the Passive Residual Heat Removal (PRHR) heat exchanger is available to remove core decay heat. Within hours after the accident, the water in the In-Containment Refueling Water Storage Tank (IRWST) will reach saturation due to PRHR heat transfer. Steam generated from IRWST water boiling will cause containment pressure and temperature to increase. Following the containment high-2 pressure signal, Passive Containment Cooling System (PCS) valves will open, and gravity-driven chilled water will flow from the Passive Containment Cooling Water Storage Tank (PCCWST) to the outer surface of the containment steel shell, mitigating the accident through evaporative heat transfer. The Passive Core Cooling System (PXS) condensate return tank collects condensate from the inner surface of the containment steel shell and returns most of this condensate to the IRWST, ensuring long-term PRHR submergence and enhancing heat removal to the IRWST.

1.2 Analysis Program Description

This analysis employs thermal-hydraulic analysis programs and containment analysis programs for calculation and evaluation. The thermal-hydraulic analysis program can simulate up to four-loop Reactor Coolant Systems (RCS), including the reactor pressure vessel, hot and cold legs, steam generators (tube side and shell side), and pressurizer. The program can also simulate pressurizer electric heaters, spray systems, and safety valves; it includes a point neutron kinetics model capable of simulating reactivity effects from moderator, fuel, boron, and fuel rods. The steam generator secondary side uses a homogeneous saturated fluid mixture model for thermal transient analysis, with additional water level relationships for indication and control. The turbine, condenser, and feedwater heaters are not modeled directly but are represented by steam demand and feedwater flow rate and enthalpy. The protection and safety monitoring system can simulate reactor trips triggered by high neutron flux, overtemperature ΔT, pressurizer high and low pressure, low RCS flow, and high pressurizer water level; it can also simulate control systems including control rod control, main steam bypass, main feedwater control, and pressurizer pressure and level control; additionally, it can simulate the passive core cooling system (including accumulators). The program can also perform core DNBR calculations during transient accidents based on core limit lines.

The program is applicable for non-LOCA accident analysis, including reactor coolant system overcooling, loss of secondary side heat sink, loss of forced reactor coolant flow, reactivity insertion, and RCS depressurization accidents.

The containment analysis program can simulate multiphase flow by solving mass, energy, and momentum conservation equations for multicomponent flow in integral form. The momentum conservation equations are formulated separately for each phase of the flow field (droplets, liquid region, atmospheric region), including terms for storage, convection, surface tension, body forces, boundary source terms, interfacial source terms between phases, and equipment source terms.

For containment wall regions, the containment analysis program employs turbulent natural convection heat transfer with the correlation:
[Equation reference preserved]

Turbulent natural convection mass transfer is employed with the correlation:
[Equation reference preserved]

Wall liquid film conduction heat transfer is employed with the correlation:
22.0Re
[Incomplete equation preserved as in original]

The containment backpressure and condensation reflux rate used in the analysis are obtained through iterative calculations between the thermal-hydraulic analysis program and the containment analysis program to ensure conservatism.

2.1 Assumptions and Initial Conditions

Based on the passive cooling system design of the nuclear power plant, the Passive Containment Cooling Water Storage Tank (PCCWST) can provide cooling water to the Passive Containment Cooling System (PCS) for 72 hours. After 72 hours, operators can employ multiple water sources and methods to replenish the PCCWST for long-term containment cooling. These sources and methods include: (1) using PCS recirculation pumps (if available) to provide an additional 4 days of cooling from PCCAWST to the containment; (2) using mobile diesel-driven emergency makeup pumps (if PCS recirculation pumps are unavailable) through safety Class C flanged connections in the plant area to provide cooling water from PCCAWST and other available plant water sources including the Condensate Storage Tank (CST), fire water tank, demineralized water storage tank (DWST), conventional island water production workshop tank, water plant, etc.; (3) if the above plant water sources are depleted or unavailable, operators can use external water sources (such as fire trucks) to provide long-term cooling water to the containment.

Additional assumptions and initial conditions are presented in Table 1 :

1) The containment steel shell wall heat and mass transfer correlation factor is set to 1.0;
2) The IRWST water inventory is set to the rated water volume;

3) Increasing heat sink surface area reduces the condensate return flow to the IRWST, which is conservative for condensation reflux analysis; the metal heat sink volume is increased by 10% while the concrete heat sink volume remains unchanged;
4) The Uchida heat transfer correlation conservatively underpredicts the condensation efficiency of internal heat sinks, expressed as:
[Equation structure preserved]
where vs is steam density and vg is non-condensable gas density;
5) The steel shell inner wall condensate loss rate is defined as the ratio of condensate loss (that cannot return to IRWST) to total condensate;
6) The transient development of the liquid film is not simulated; heat transfer from the liquid film is not considered within 400 seconds after PCS actuation, with this delay time based on full-scale water distribution tests and scaled liquid film flow calculations;
7) The PCS cooling water flow rate is set to the rated flow, with containment response calculated through evaporative limit flow iteration to reduce liquid film sensible heat absorption. PCS flow rates after 72 hours use the 72-hour PCS flow rate.

2.2 Analysis Results

Figures 1 [Figure 1: see original paper] and 2 [Figure 2: see original paper] compare the containment pressure and condensation reflux rate between the containment analysis program and the thermal-hydraulic system analysis program. As the PRHR transfers heat to the IRWST water, the water temperature continuously increases, initially dominated by sensible heat with minimal evaporation, resulting in a slight containment pressure increase. Once the water temperature reaches saturation, evaporation increases significantly, causing a rapid containment pressure rise. When the containment high-2 pressure setpoint is reached, PCS actuation is triggered. After a 400-second delay, a stable liquid film is established, followed by heat removal through liquid film evaporation. Through several iterations between the thermal-hydraulic system program and the containment analysis program, the containment pressure and condensation reflux rate used in the thermal-hydraulic system program can bound the results from the containment analysis program, providing a conservative basis for accident analysis.

3.1 Assumptions and Initial Conditions

The reactor safe shutdown temperature evaluation and safe shutdown state duration evaluation employ the following assumptions: (1) Initial reactor power level is 100% of nominal; (2) Initial reactor coolant average temperature considers the nominal value plus uncertainty; (3) Initial pressurizer pressure considers the nominal value minus uncertainty, which reduces RCS discharge through pressurizer safety valves; (4) Single steam generator initial water inventory corresponds to the value at SG maximum water level; (5) PRHR and CMT employ nominal safety-grade equipment parameters; (6) The core decay heat model uses the best-estimate decay heat plus 10% uncertainty.

3.2 Analysis Results

Assuming loss of normal feedwater at 10 seconds, the resulting SG water level decrease reaches the SG narrow-range low water level shutdown setpoint at 78.4 seconds, with control rod insertion into the core beginning after a 2-second delay (as shown in Figure 3 [Figure 3: see original paper]). This subsequently triggers turbine trip and simultaneous loss of auxiliary AC power, causing the reactor coolant pumps to coast down, RCS flow to decrease, and pressurizer pressure, pressurizer inventory, and RCS average temperature to increase. The SG safety valve opens at 86.5 seconds when the opening pressure is reached.

PRHR actuation is triggered by the SG wide-range low water level setpoint signal at 146.7 seconds, causing pressurizer pressure, pressurizer inventory, and RCS average temperature to begin decreasing. At 2816.5 seconds, the cold leg low temperature "S" signal is reached, with main steam line isolation triggered after a 7-second delay and CMT actuation triggered after a 12-second delay. CMT actuation causes a rapid RCS temperature decrease, with pressurizer pressure and inventory decreasing correspondingly. However, the RCS temperature decrease reduces PRHR heat removal capacity, causing RCS temperature, pressurizer pressure, and pressurizer inventory to increase again. At approximately 35,000 seconds, PRHR heat removal reaches equilibrium with core decay heat. At approximately 26.25 hours, the RCS average temperature decreases to 216°C and can be maintained for 14 days post-accident. The event sequence is detailed in Table 2 .

Figure 3 [Figure 3: see original paper] Reactor Power
Figure 4 [Figure 4: see original paper] RCS Temperature
Figure 5 [Figure 5: see original paper] IRWST Water Level
Figure 6 [Figure 6: see original paper] PRHR Heat Removal

4 Conclusions

(1) Through iteration between the thermal-hydraulic analysis program and containment analysis program, a set of conservative containment pressure and condensation reflux rate values were obtained for reactor safe shutdown evaluation and safe shutdown state duration evaluation. The maximum containment pressure is 0.2 MPa(a) and the minimum pressure is 0.1 MPa(a); the maximum condensation reflux rate is 0.8.

(2) Using the thermal-hydraulic analysis program, this analysis evaluated the reactor shutdown temperature and safe shutdown state duration, verifying that following a loss of normal feedwater combined with loss of AC power accident, the passive safety system can reduce the reactor coolant average temperature to the safe shutdown temperature of 216°C within 36 hours

and maintain the safe shutdown state for 14 days post-accident, satisfying URD requirements.

(3) This analysis considered the effects of reduced PRHR heat removal capacity after PRHR dryout, CMT actuation, and containment backpressure and condensation reflux rate after PRHR dryout, providing further verification of PRHR long-term heat removal capability.

(4) Nuclear power plants should be enhanced to withstand accidents more severe or longer-lasting than design basis accidents to avoid unacceptable consequences and further improve nuclear power plant safety.

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Note: Figure translations are in progress. See original paper for figures.