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A Parametric Study of the Thermal-Hydraulic Response of Supercritical Light Water Reactors During Loss-of-Feedwater and Turbine-Trip Events

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The Idaho National Engineering and Environmental Laboratory is investigating the feasibility of supercritical light water reactors for low-cost electric power production through a Nuclear Energy Research Initiative Project sponsored by the United States Department of Energy. The project is evaluating a variety of technical issues related to the fuel and reactor design, material corrosion, and safety characteristics. This paper presents the results of parametric calculations using the RELAP5 computer code to characterize the thermal-hydraulic response of supercritical reactors to transients initiated by loss-of-feedwater and turbine-trip events. The purpose of the calculations was to aid in the design of the safety systems by determining the time available for the safety systems to respond and their required capacities.

KEYWORDS: Supercritical water reactor, RELAP5, loss of feedwater, turbine trip

I. Introduction

The Idaho National Engineering and Environmental Laboratory (INEEL), Westinghouse Electric Company, Massachusetts Institute of Technology, and the University of Michigan are studying the feasibility of a thermal-spectrum reactor cooled by supercritical light water for electric power production. Supercritical reactors have the potential for improved economics compared to current light water reactor designs due to significant plant simplification and high thermal efficiency.

The design of the supercritical water reactor (SCWR) analyzed here uses a once-through direct cycle with a conventional reactor vessel. The thermal neutron spectrum is obtained using solid moderator boxes containing zirconium hydride.

The purpose of the present work was to perform simple parametric calculations of a preliminary design to characterize the response of the SCWR to various initiating transients so that the required response time and capacities of various safety systems could be determined. The time available was determined by comparing the calculated maximum fuel rod cladding temperature during the transient with a preliminary temperature limit of 840°C. The transients considered were initiated by a loss of feedwater and a turbine trip.

Calculations were performed using the RELAP5[1)] computer code. The RELAP5 computer code was originally developed for the thermal-hydraulic analysis of light water reactors and related experimental systems during loss-of-coolant accidents and operational transients. The code is being improved to support the analysis of potential Generation IV reactors, such as fast reactors cooled by gas or heavy metals.[2)] The code has been recently improved to support the analysis of reactors cooled by supercritical light water.[3)] Improvements have been made to the calculation of water properties and the solution scheme in the supercritical region. Correlations have also been added to the code for the analysis of supercritical water reactors. The correlations of Bishop[4)] and Koshizuka-Oka[5)] for heat transfer to supercritical water and the correlation of Petrov and Popov[6)] for determining the effect of wall temperature on the friction factor at supercritical conditions have been added to the code.

Sections II and III of this paper describe the SCWR and the RELAP5 model of the SCWR. Results are presented in Section IV. Conclusions and references are presented in Sections V and VI, respectively.

II. SCWR Design

The design under consideration[7)] uses a once-through direct cycle with a conventional reactor vessel that is illustrated in **Fig. 1**. The primary flow path is from the cold legs, through an annular downcomer to the lower plenum, up through the core and the upper plenum, and out through the hot legs.

The core contains 157 square canned fuel assemblies. As shown in **Fig. 2**, each fuel assembly contains 217 fuel rods and 36 square moderator boxes containing solid zirconium hydride ($ZrH_{1.6}$). The fuel is uranium dioxide enriched with 4 at% U-235 that is 95% of the theoretical density. The fuel rod cladding and moderator box wall are made of the Ni-based Alloy 718. The geometry of the reactor is summarized in **Table 1**.

III. RELAP5 Model

The RELAP5 model of a SCWR with solid moderator boxes is illustrated in **Fig. 3**. The model represents the reactor vessel including the downcomer (Component 300), lower plenum (Components 310 and 315), core (Components 325, 330, and 335), upper plenum (Component 360), and upper head (Component 370). The core is modeled with three parallel channels, one (Component 325) representing a high-powered fuel bundle, one (Component 330) representing 155 average-powered fuel bundles, and one (Component 335) representing a low-powered fuel bundle. The heated length is divided equally

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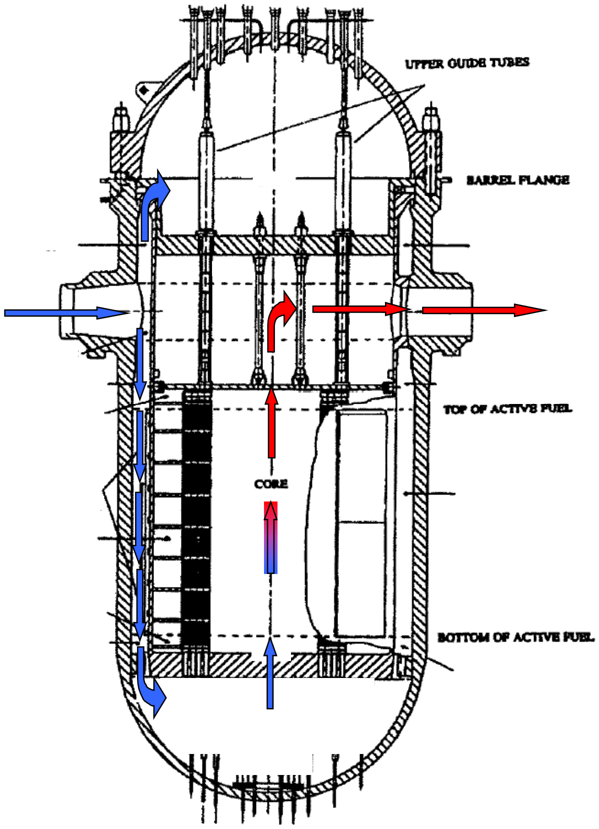

Fig. 1 Reactor vessel layout for the SCWR.

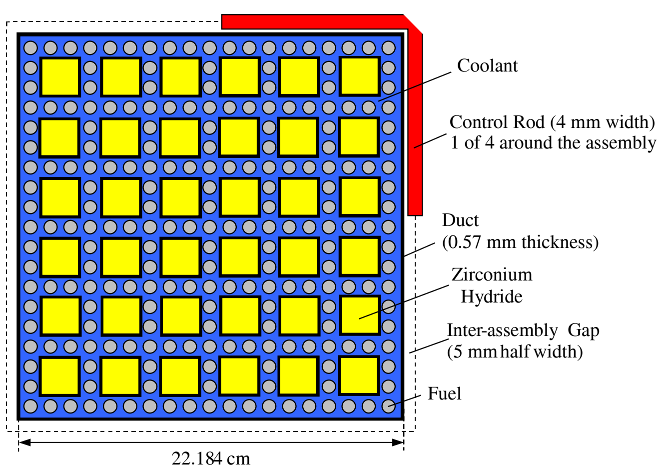

Fig. 2 Fuel assembly design

Table 1. Geometry of the SCWR

Parameter	Value
Fuel:	
Fuel outer diameter, mm	8.19
Cladding inner diameter, mm	8.36
Cladding outer diameter, mm	9.50
Fuel rod pitch, mm	11.50
Heated length, mm	4.27
Vessel:	
Inner diameter, m	4.41
Outer diameter, m	5.13
Height, m	12.37
Core barrel:	
Inner diameter, m	3.80
Outer diameter, m	3.90

into ten axial control volumes. Heat structures are used to represent fuel rods and moderator boxes in each core channel, as well as the reactor vessel wall and core barrel. A separate heat structure is used to model four hot rods in the high-powered fuel bundle.

Orifices are simulated at the bottom of each core channel to achieve a uniform power-to-flow ratio across the core. Three core bypass paths are simulated. Component 345 simulates the gaps between assemblies and between the outer assemblies and the core barrel. Junction 365 represents the cooling flow from the downcomer to the upper head. Component 366 represents leakage from the downcomer to the upper plenum around the hot leg nozzle. Form loss coefficients in the bypass flow paths were adjusted to obtain the desired flow rates at normal operating conditions. Boundary conditions are used to represent the feedwater (Junction 105) and main steam (Component 405) systems.

The local peaking factor of the hot rods is 1.108 based on the calculations of Ref. 7. The axial power profile is computed using a two-zone enrichment scheme and has a peak value of 1.41. The radial peaking factors of the high-powered and low-powered fuel bundles are assumed to be 1.30 and 0.70, respectively. The moderator boxes receive 2.6% of the total core power.[8)]

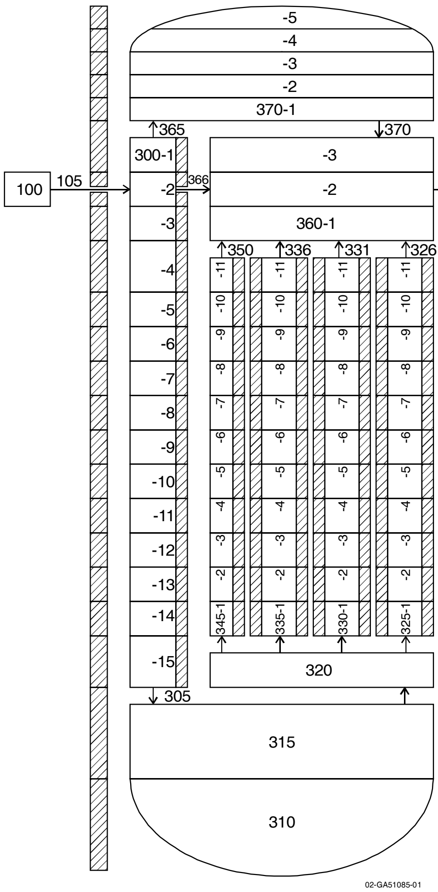

Fig. 3 RELAP5 model of the SCWR.

Transient reactor power was calculated using a best-estimate point kinetics model. The model used representative pressurized water reactor values for the kinetics and decay heat parameters. The reactivity feedback model was used to simulate the effects of changes in the fuel temperature and coolant density. The Doppler and density feedback coefficients were -3.23×10^{-3} \$/K and $+3.23\times10^{-3}$\$ /(kg/m^3), respectively, based on the calculations described in Ref. 7 and 8. Power-squared averaging was used to determine the weighting factors in the feedback model. For cases with reactor scram, the control rods began moving 0.8 s after the scram signal was generated, and were fully inserted 2.5 s later. The total control rod worth was about 11\$.

IV. Results

1. Steady State

A RELAP5 calculation was performed to determine the steady-state thermal-hydraulic conditions at normal operating power. These normal operating conditions are summarized in **Table 2**. As mentioned previously, orifices were simulated at the inlet of each core channel to achieve the same power-to-flow ratio, which results in a uniform fluid temperature at the outlet of each channel. The orifice size was set to achieve a core pressure drop of 0.150 MPa, which is similar to that in most operating light water reactors. The simulated orifice at the bottom of the high-powered channel had a diameter near 4.8 cm. The pressure loss due to the orifice exceeded the losses due to the grid spacers and wall friction. The pressure drop across the core was only about 0.080 MPa in preliminary calculations in which the high-powered channel did not contain an inlet orifice. However, preliminary calculations of loss-of-feedwater events without the orifice resulted in flow oscillations in the high-powered channel. These oscillations disappeared when the pressure drop across the core was increased to 0.150 MPa.

Table 2. Calculated initial conditions at rated power

Parameter	Value
Core Power, MWt	2700
Pressure, MPa	25.0
Feedwater temperature, °C	280
Vessel outlet temperature, °C	450
Core outlet temperature, °C	474
Peak cladding temperature, °C	572
Feedwater flow rate, kg/s	1568
Core flow rate, kg/s	1474
Core differential pressure, MPa	0.150
Bypass flow rates:	
Core bypass, %	4.0
Outlet nozzle leakage, %	1.0
Upper head cooling, %	1.0

The RELAP5 model was used to perform a series of steady-state calculations in which the feedwater flow and core power were varied so that their ratio remained constant. **Figure 4** presents the results of these calculations in the form of maximum core cladding temperature as a function of normalized power. A normalized power of 1.0 corresponds to the normal operating conditions presented in Table 2. Lower normalized values correspond to operation at reduced power and flow. Because the power-to-flow ratio was held constant, the fluid temperature distribution in the core was roughly the same for each calculation, differing only because the fraction of core bypass varied with flow.

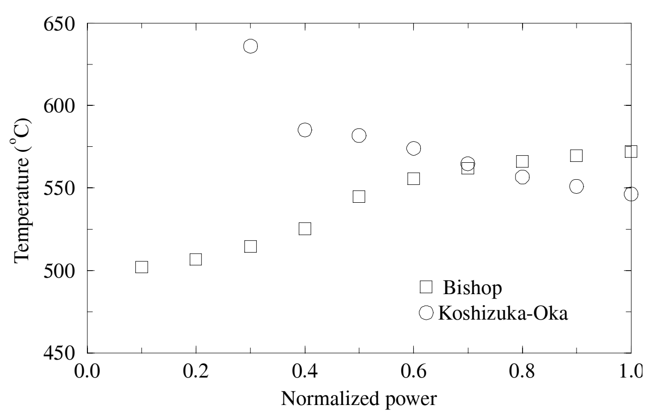

Fig. 4 Maximum cladding temperature as a function of normalized power

Figure 4 shows that the maximum cladding temperatures obtained with the Bishop and Koshizuka-Oka correlations were similar at a normalized power of 1.0. However, the two correlations predicted completely different trends with respect to power. The Bishop correlation predicted that the maximum cladding temperature increased with power, while the Koshizuka-Oka correlation predicted that it decreased with power. The location of the maximum cladding temperature also varied between correlations. The maximum occurred in the eighth (out of ten) heated control volumes with the Bishop correlation, but varied between the fourth and sixth control volumes with the Koshizuka-Oka correlation. The calculated conditions in the SCWR core represented by Fig. 4 required both correlations to be extrapolated from their respective databases. For example, the calculated heat and mass fluxes dropped below the lower limits of the Bishop correlation near normalized values of 0.4 and 0.7, respectively. The Koshizuka-Oka correlation required a larger extrapolation as the mass flux during normal operation was 10% below the lower limit of the correlation. The results shown in Fig. 4 demonstrate the dangers of extrapolating correlations beyond their database. The results also demonstrate the importance of obtaining heat transfer data that cover the range of interest for the SCWR.

Heat transfer data should also be taken that cover the transitions between the forced convection, natural convection, and laminar heat transfer regimes as these transitions will be encountered during loss-of-flow transients. RELAP5 normally calculates the heat transfer coefficient as the maximum of forced convection, natural convection, and laminar correlations. However, the natural convection contribution was suppressed for these calculations to demonstrate the difference between the Bishop and Koshizuka-Oka forced convection correlations. The heat transfer coefficient from the code's natural convection correlation exceeded that from the Koshizuka-Oka correlation, particularly at low flow rates.

2. Loss of Feedwater

Calculations were performed to investigate the effects of various parameters on the peak cladding temperature during a loss-of-feedwater transient. The parameters investigated include the main feedwater (MFW) coastdown time, occurrence of scram, auxiliary feedwater (AFW) flow rate, steam relief, step changes in MFW flow rate, and reactivity feedback coefficient.

The event initiated by loss of feedwater is of particular importance to the SCWR because it is a once-through flow system without recirculation. Loss of feedwater in the SCWR corresponds to a simultaneous loss-of-flow and loss-of-feedwater event in current light water reactors and has the potential for rapid overheating. Complete loss-of-flow events in pressurized water reactors are generally classified as accidents, while complete loss-of-feedwater events are generally classified as transients. Transients have a higher probability of occurrence than accidents and are required to meet more restrictive thermal limits. For example, the thermal limits in a Japanese SCWR[9)] are 1260°C for accidents and 840°C for transients. Loss-of-feedwater events in the Japanese SCWR have been classified as accidents, which is consistent with the categorization of complete loss-of-flow events in pressurized water reactors. However, it is not obvious that the U. S. Nuclear Regulatory Commission will support the probability classification of the loss-of-feedwater event in the SCWR as an accident. Furthermore, the U. S. has a history of meeting transient thermal limits during relatively infrequent accidents. Therefore, the transient temperature limit of 840°C identified in Ref. 9 was conservatively applied to the loss-of-feedwater event in this study.

Calculations were performed to investigate the effect of MFW coastdown, with the coastdown time ranging from 0 to 10 s, during a total loss of flow. The normalized feedwater flow rates are shown in **Fig. 5**. In each case, a linear flow coastdown was assumed beginning at 0 s. The point kinetics model was used to calculate reactivity feedback, but no scram was assumed. The reactor pressure was assumed to remain constant due to the operation of turbine bypass valves.

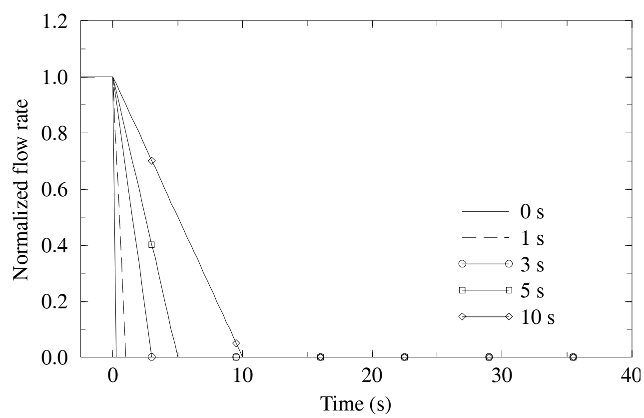

Fig. 5 Normalized feedwater flow rates

The effect of the flow coastdown on the maximum cladding temperature is shown in **Fig. 6**. The effect of additional MFW flow was to slow the increase in cladding temperature. Each additional second of full feedwater flow (for example, the 5-s coastdown represents 2.5 full flow seconds) caused the peak cladding temperature to reach the 840°C transient limit about one second later. The scram system does not have to respond as quickly if the coastdown time can be extended.

A 5-s flow coastdown was assumed for the analysis of the Japanese SCWR as described in Ref. 9 A 5-s value was also assumed in the Safety Analysis Report[10)] for the Grand Gulf boiling water reactor (BWR). Therefore, subsequent calculations were performed with a 5-s coastdown. The maximum cladding temperature reached 840°C at 4.8 s with a 5-s coastdown.

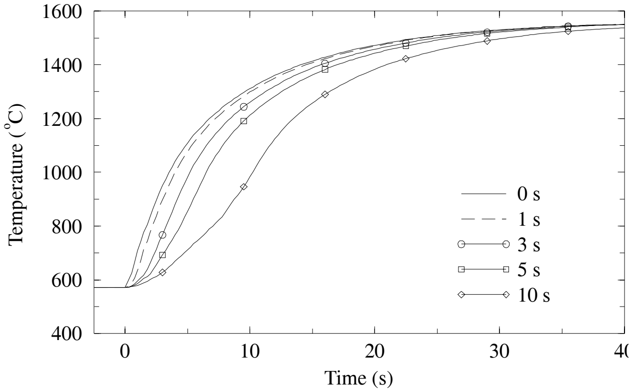

Fig. 6 The effect of MFW coastdown on maximum cladding temperature

The effect of scram on the peak cladding temperature is illustrated in **Fig. 7**. The scram signal was assumed to be generated at 0.5 s, corresponding to a 10% reduction in flow. The maximum cladding temperature exceeded the transient limit 6.5 s into the event with scram, compared to 4.8 s without scram. The peak cladding temperature with scram was 964°C and occurred at 26 s. The cladding temperature then decreased slowly compared to current light water reactors. The slower temperature decrease occurs because the thermal conductivity of the supercritical steam near the hot spot is less than that of subcooled liquid, there is no nucleate boiling in supercritical water, and the natural circulation flow rate is relatively small in the SCWR. Natural circulation occurs between the downcomer and the core because of flow through the bypass paths. The SCWR does not contain external loops that would enhance natural circulation in the absence of forced circulation.

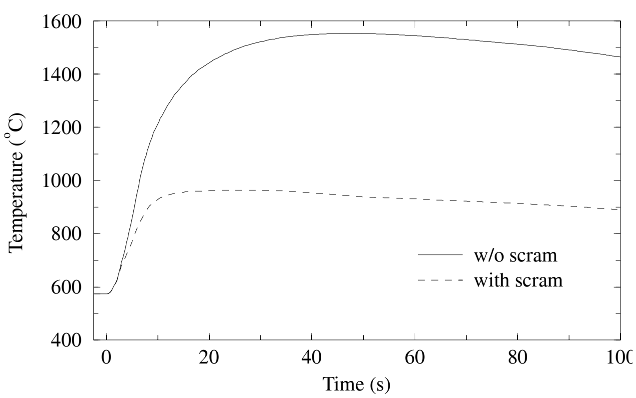

Fig. 7 The effect of scram on maximum cladding temperature

Calculations were performed to determine the effect of AFW flow combined with a 5-s MFW flow coastdown, rector scram, and with the reactor pressure held constant by the turbine bypass valves. AFW flow rates corresponding to 10%, 20%, and 30% of the initial feedwater flow were assumed as shown in **Fig. 8**. The effect of the AFW flow rate on the maximum cladding temperature is shown in **Fig. 9**. An interpolation of the calculated results indicates that the peak cladding temperatures will remain below the transient limit of 840°C if the AFW flow is at least 15% of the initial MFW value. Considering that additional conservatisms may be required in the model, a higher AFW flow is desirable. The 15% AFW flow rate would have to be generated within 4.25 s of the start of the event to be consistent with the assumed flow rates shown in Fig. 8.

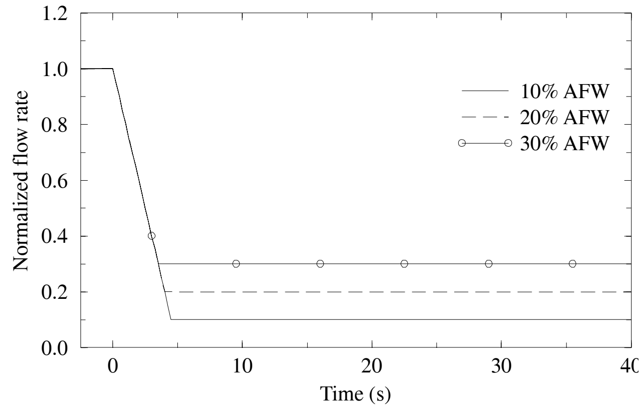

Fig. 8 Total feedwater flow rates with AFW

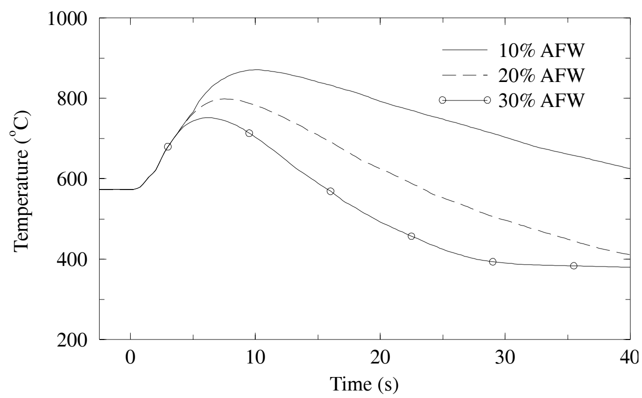

Fig. 9 The effect of AFW on peak cladding temperature

Calculations were performed to determine the effects of step reductions in MFW flow. The step reductions varied from 25% to 100% of the initial flow rate, with the latter value corresponding to a complete loss of feedwater flow. The calculations were performed without reactor scram. Turbine bypass valves were assumed to hold the reactor pressure constant.

Figure 10 shows the effects of step reductions in MFW flow rate on the maximum cladding temperature. The flow reductions caused the cladding temperature to increase, with larger increases calculated for the more severe reductions in flow. Reactivity feedback caused the reactor power to decrease until a new steady state was obtained corresponding to the reduced flow and power.

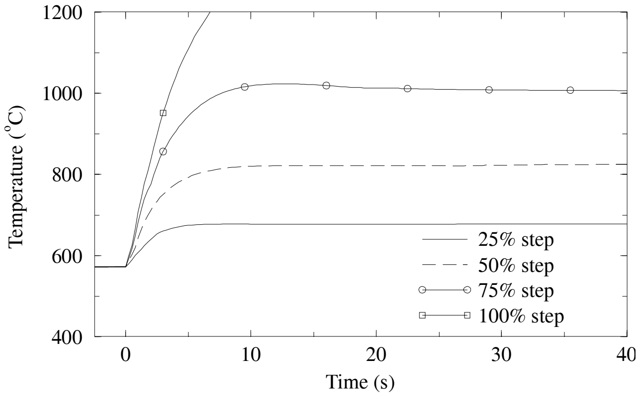

Fig. 10 The effect of a step reduction in MFW flow on maximum cladding temperature

The calculated results indicate that the peak cladding temperature will remain below the transient limit of 840°C for step reductions in flow less than 52% of the initial value. In particular, it appears that a scram is not required to meet the transient temperature limit for an instantaneous loss of half of the feedwater pumps.

Calculations were performed to determine the effectiveness of steam relief to enhance flow through the core. In the original calculation, the turbine bypass valves were assumed to modulate to hold the reactor pressure constant. A second calculation was performed in which a relief valve was opened at 2 s. The relief valve was sized to discharge 20% of the rated steam flow at normal operating conditions. A check valve was placed in the main steam line to prevent steam from flowing from time-dependent Volume 405 (see Fig. 3) through the relief valve. A third calculation was performed in which 100% steam relief was simulated by assuming that the turbine throttle valves remained fully open. These calculations assumed a 5-s MFW coastdown, no AFW, and a reactor scram at 0.5 s.

Figure 11 shows the total steam flow for the three calculations. The steam flow rate closely followed the MFW flow rate when the turbine bypass valves modulated to hold the pressure constant. Slightly more steam flow was obtained when the 20%-capacity relief valve opened at 2 s and significantly more flow was obtained in the case with 100% steam relief.

Figure 12 shows calculated reactor pressures for the three cases. Opening the 20%-capacity valve resulted in a significant reduction in pressure, but did not result in flashing of the cold fluid in the downcomer and lower plenum during the time period of interest. Without flashing, there was little enhancement of the core flow and a relatively small effect on cladding temperature as shown in **Fig. 13**. In the case with 100% steam relief, flashing in the downcomer and lower plenum occurred near 12 s. This flashing caused a significant increase in core flow and a rapid reduction in cladding temperature. In the constant-pressure case, the peak cladding temperature was 964°C and occurred at 26 s. The 20%-capacity relief valve was relatively ineffective in reducing the peak cladding temperature. With 100% steam relief, the peak cladding temperature was 874°C and occurred at 11 s. The calculations show that a rapid opening of 100%-capacity turbine bypass valves could help reduce the cladding temperature, but by itself would not be sufficient to prevent the peak cladding temperature from exceeding the transient limit of 840°C.

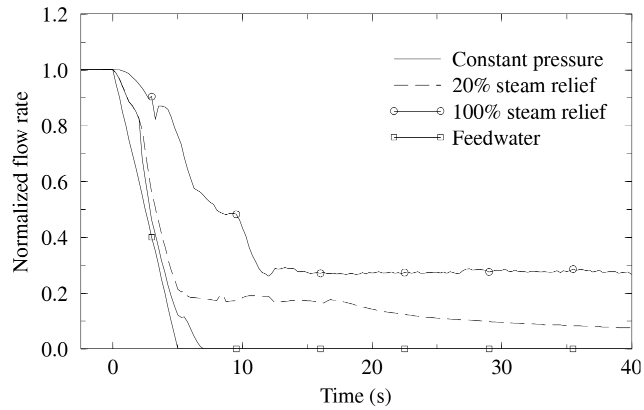

Fig. 11 Steam flow rates following loss of feedwater

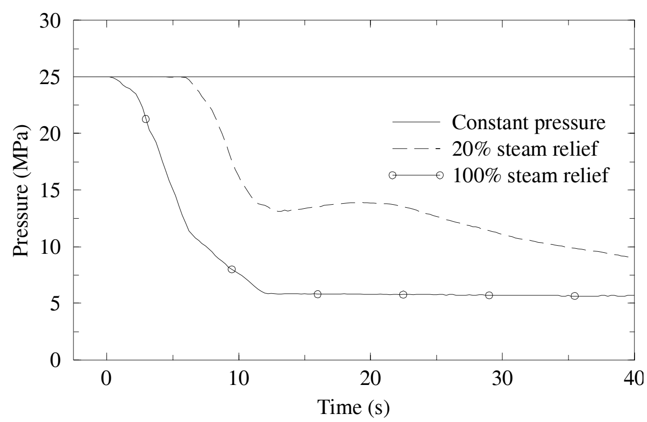

Fig. 12 The effect of steam relief on reactor pressure

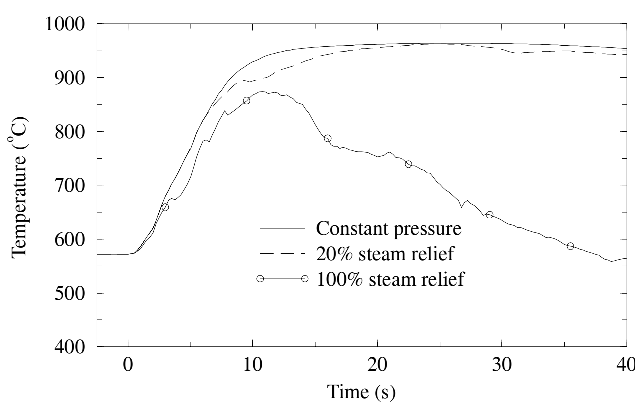

Fig. 13 The effect of steam relief on maximum cladding temperature

The results shown previously were obtained with reactivity feedback coefficients that were calculated for a design with solid moderator boxes. The Doppler feedback coefficient for a reactor design using water rods (Ref. 9) is similar to that used here, but the coolant density feedback coefficient is about 19 times greater than that used here. Because burnup considerations favor a design with water rods, a sensitivity calculation was made in which the density feedback coefficient was increased by a factor of 19 to 6.15×10^{-2}\$/(kg/m^3). These sensitivity calculations are expected to show trends but will not exactly represent a design using water rods because the thermal response time of the water rods differs from that of the coolant.

A calculation was performed to determine the effect of the coolant density feedback coefficient on a transient initiated by a 50% step change in feedwater flow without reactor scram. **Figure 14** shows that the reactor power decreased much faster, but tended to overshoot the equilibrium power, with the higher feedback coefficient. The equilibrium power was 67% of the initial value with the lower feedback coefficient, which corresponds to the solid moderator. With the higher coefficient, the equilibrium power was 54%, just slightly exceeding the normalized feedwater flow rate. The more rapid reduction in power with the higher feedback coefficient resulted in a major reduction in peak cladding temperature as shown in **Fig. 15**.

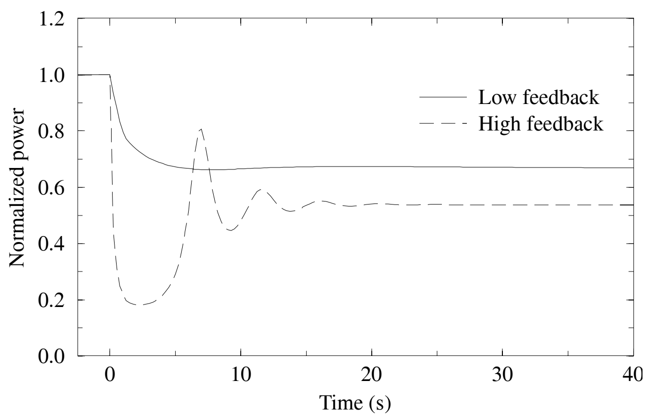

Fig. 14 The effect of reactivity feedback on reactor power following a 50% step reduction in feedwater flow

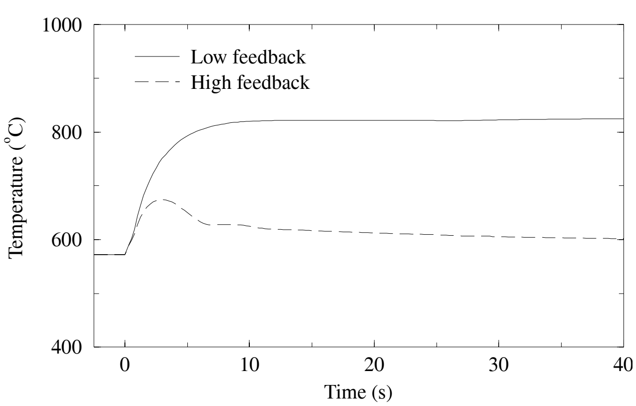

Fig. 15 The effect of reactivity feedback on maximum cladding temperature following a 50% reduction in MFW

The effect of the reactivity coefficient on a transient initiated by a complete loss of feedwater was also determined. These calculations assumed a 5-s MFW coastdown, no AFW, and a reactor scram beginning at 0.5 s. The higher feedback coefficient caused the reactor power to decrease significantly before the control rods were released as shown in **Fig. 16**. The more rapid reduction in core power lowered the peak cladding temperature as shown in **Fig. 17**. The peak cladding temperature was 860°C with the higher feedback coefficient, slightly exceeding the transient limit of 840°C, but a significant improvement from the 964°C calculated with the lower feedback coefficient.

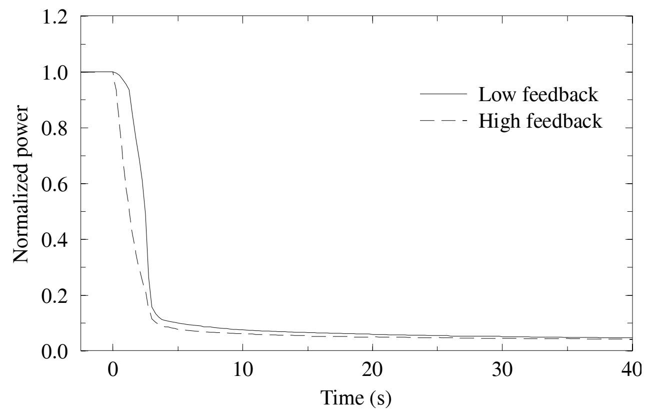

Fig. 16 The effect of reactivity feedback on reactor power following total loss of feedwater

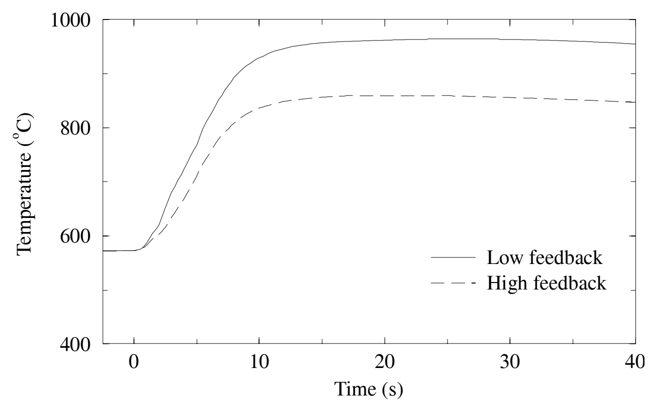

Fig. 17 The effect of reactivity feedback on maximum cladding temperature following total loss of feedwater

3. Turbine Trip

Three calculations of a turbine trip were performed using the RELAP5 model shown in Fig. 3. The three calculations assumed an instantaneous closure of the turbine control valves and continued MFW flow at its rated value to obtain conservative predictions of the reactor vessel pressurization. The three calculations differed relative to their assumptions concerning scram and the capacity of the safety relief valves (SRVs). The first calculation assumed that no scram occurred and that no SRVs were available, thus bounding the pressure response of the reactor. The second calculation assumed that no scram occurred and that the SRV capacity was 90% of the steam flow at normal operating conditions.

The third calculation assumed a relief capacity of 80% of the normal steam flow and that a scram signal was generated 0.1 s into the event. The SRVs were assumed to open at a pressure of 27.0 MPa and close at a pressure of 26.25 MPa.

The transient and accident pressure limits were the same as those developed in Ref. 9. Specifically, the transient pressure limit was taken to be 28.87 MPa, which corresponds to 1.05 times the design pressure limit, which was assumed to be 1.10 times the operating pressure of 25.0 MPa. The accident pressure limit was assumed to be 30.25 MPa, corresponding to 1.10 times the design pressure of 27.5 MPa.

Figure 18 shows the pressure response from the three calculations. Without scram or SRVs, the pressure exceeds the accident limit at 0.45 s. Without scram and an SRV capacity of 90%, the pressure remains below the transient limit. The reactor approaches a new steady state with continuous flow through the SRVs. With scram, the relief capacity can be reduced to 80% of the steam flow at normal operating conditions and still remain below the transient limit. Intermittent operation of the SRVs occurs after 5 s.

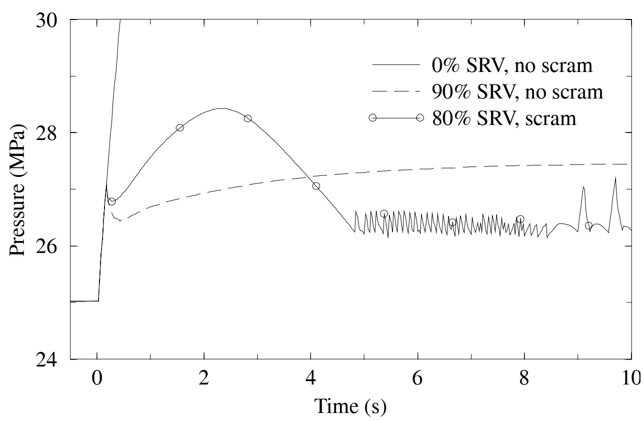

Fig. 18 Reactor pressure following a turbine trip

Calculated reactor power is shown in **Fig. 19**. The maximum power remained relatively low even in the cases without scram. The maximum power was only 2.5% higher than the initial value with a relief capacity of 90% and without a reactor scram. For reference, in the Grand Gulf BWR6 (Ref. 10), the power increase for a turbine trip was about 6%. However, this relatively low power increase was obtained taking credit for a simultaneous trip of the recirculation pumps. In the Browns Ferry BWR4[11)], a turbine trip coupled with a failure of the turbine bypass system resulted in a power increase of 170%. The BWR4 results are more comparable with the SCWR because they were obtained without the recirculation pump trip, which is consistent with the constant feedwater flow assumed here.

The relatively low power increase and the continued feedwater flow caused a relatively small increase in maximum cladding temperature as shown in **Fig. 20**.

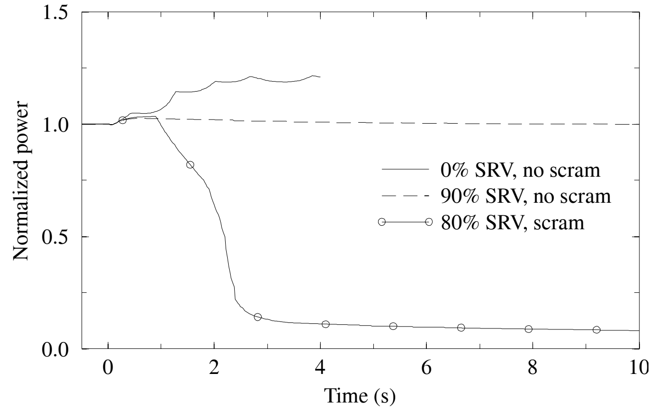

Fig. 19 Total reactor power following a turbine trip

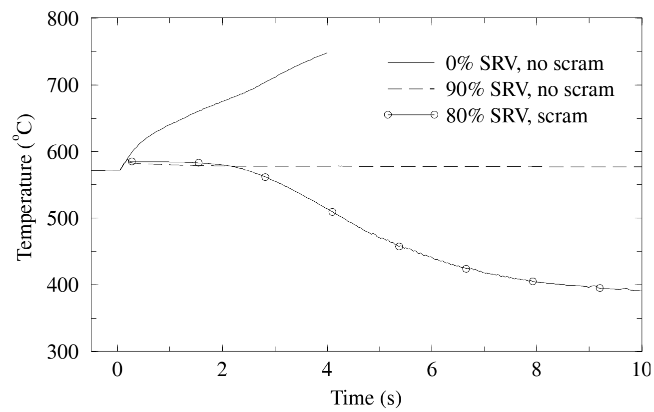

Fig. 20 Maximum cladding temperature following a turbine trip

IV. Conclusions

The parametric calculations showed that the SCWR with solid moderator boxes could tolerate a 50% instantaneous reduction in feedwater flow without a reactor scram and still meet a transient temperature limit of 840°C. Transients involving total loss of feedwater pose a more serious challenge to the design. Calculations indicated that acceptable temperature results could be obtained with a 5-s MFW flow coastdown, a reactor scram, and an AFW flow rate that is 15% or more of the initial feedwater flow. Calculations also showed that a fast-opening, 100%-capacity turbine bypass system could significantly reduce the peak cladding temperature. An increase in the coolant density feedback coefficient also significantly lowered the peak cladding temperature. Since the density feedback coefficient is about 19 times larger when moderation is achieved with water rods rather than with solid moderator, water-rod designs show potential for increased safety margins.

The parametric calculations showed that the SCWR could meet reactor vessel pressure limits following a turbine trip provided that the SRV capacity at normal operating conditions is 90% or more of the rated steam flow. The power increase following a turbine trip was much smaller than in comparable BWRs.

Finally, the calculated results for the SCWR are sensitive to the choice of heat transfer correlation. Large variations in calculated results were obtained due to the choice of correlation. Furthermore, the databases of the existing correlations do not cover a sufficiently wide range of thermal-hydraulic conditions to fully support analysis of the SCWR at off-normal conditions and during transients. Heat transfer experiments that are prototypical with respect to thermal-hydraulic conditions and geometry should be performed to support analysis of the reactor. The experiments should also cover the transitions between the forced convection, natural convection, and laminar heat transfer regimes as these transitions will be encountered during loss-of-flow transients.

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