No-core-melt assessment for Canadian-SCWR under LOCA/LOECC Postprint

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Abstract

The safety analysis code SCTRAN for SCWR (Super Critical Water Reactor) is modified to own the capability to assess the radiation heat transfer with developing a two-dimensional heat conduction solution scheme and incorporating a radiation heat transfer model. The verification of the developed radiation heat transfer model is conducted through code-to-code comparison with CATHENA. The results show that the modified SCTRAN code is successful for that the maximum absolute error and relative error of the surface temperature between results of SCTRAN and CATHENA are 6.1 ℃ and 0.9%, which are acceptable in temperature prediction. Then, with the modified SCTRAN code, the loss of coolant accident with a total loss of emergency core cooling system (LOCA/LOECC) of Canadian-SCWR is carried out to evaluate its "no-core-melt" concept. The following conclusions are achieved: 1) in the process of LOCA, the decay heat can be totally removed by the radiation heat transfer and the natural convection of the high-temperature coolant, even without an intervention of ECCS (Emergency Core Cooling System); 2) The peak cladding temperature of the fuel pins in the inner and outer rings of the high power group are 1236 ℃ and 1177 ℃ respectively, which are much lower than the melting point of the fuel sheath. It indicates that the Canadian-SCWR can achieve "no-core-melt" concept under LOCA/LOECC.

Full Text

Preamble

No-Core-Melt Assessment for Canadian-SCWR Under LOCA/LOECC

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The safety analysis code SCTRAN for SCWR (Supercritical Water Reactor) has been modified to assess radiation heat transfer by developing a two-dimensional heat conduction solution scheme and incorporating a radiation heat transfer model. Verification of the developed radiation heat transfer model was conducted through code-to-code comparison with CATHENA. The results demonstrate that the modified SCTRAN code is successful, with maximum absolute and relative errors in surface temperature between SCTRAN and CATHENA results of 6.1 °C and 0.9%, respectively, which are acceptable for temperature prediction. Using the modified SCTRAN code, a loss of coolant accident with total loss of emergency core cooling system (LOCA/LOECC) for the Canadian-SCWR was analyzed to evaluate its "no-core-melt" concept. The following conclusions were achieved: (1) During LOCA, decay heat can be completely removed by radiation heat transfer and natural convection of high-temperature coolant, even without ECCS intervention; (2) The peak cladding temperatures of fuel pins in the inner and outer rings of the high-power group are 1236 °C and 1177 °C, respectively, which are much lower than the melting point of the fuel sheath. This indicates that the Canadian-SCWR can achieve the "no-core-melt" concept under LOCA/LOECC.

Keywords: Canadian-SCWR, LOCA/LOECC, No-Core-Melt, SCTRAN, Radiation heat transfer

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Introduction

The Canadian-SCWR is a pressure-tube type supercritical water reactor that promises to satisfy all major Generation IV International Forum (GIF) goals regarding enhanced safety, sustainability, economics, and proliferation resistance [**?**]. It employs supercritical light water as coolant and subcritical heavy water as moderator. The high-efficiency channel (HEC) design enhances the reactor' s inherent safety, enabling it to achieve "no-core-melt" under postulated accident scenarios involving loss of emergency core cooling, such as LOCA/LOECC (loss of coolant accident with coincident loss of emergency core cooling), because radiation heat transfer inside the HEC and passive heat rejection through the insulator into the low-temperature moderator can remove decay heat. Safety analysis is required to demonstrate the feasibility of the continuously updated Canadian-SCWR concept. Since LOCA/LOECC leads to total loss of core coolant and represents the most severe accident scenario, its safety analysis serves as an important reference for evaluating the inherent safety of the Canadian-SCWR.

Simulations evaluating and optimizing thermal performance of the Canadian-

SCWR following LOCA/LOECC were performed by AECL (Atomic Energy of Canada Limited) using CANFLEX bundle [?]. Transient simulations were carried out with assumed decay power variation, analyzing effects of insulator properties and moderator temperature on fuel cladding temperature. Results showed that the Canadian-SCWR has potential to significantly reduce core damage frequency. Shan et al. [?] performed sub-channel analyses with ATHAS code and radiation heat transfer analyses with CATHENA code for a 54-element Canadian-SCWR bundle. Sub-channel analysis results showed maximum fuel cladding temperatures of 761 °C at BOC (Beginning of Cycle) and 808 °C at EOC (End of Cycle). Radiation heat transfer calculations at different decay power levels indicated that the pressure tube with 54-element fuel bundle can remove about 2% of rated power to the moderator through radiation heat transfer. Licht and Xu [?] provided simulation results and analyses for a 78-element Canadian-SCWR bundle during LOCA/LOECC, showing that fuel sheath temperature with a non-porous insulator remains below melting temperature for less than 3% of rated power. While other authors have proposed suggestions, few published studies have addressed "no-core-melt" assessment of the Canadian-SCWR.

Previous analyses of the Canadian-SCWR focused primarily on radiation heat transfer rather than natural convection effects during LOCA/LOECC. Additionally, the transition process from supercritical to subcritical pressure was not simulated. This paper incorporates a two-dimensional heat conduction model and a radiation heat transfer model into SCTRAN [?]. The SCTRAN thermal-hydraulic idealization for the Canadian-SCWR is based on the latest conceptual design utilizing a 64-element fuel bundle. LOCA/LOECC is analyzed considering both radiation and convection effects in the HEC.

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II. Modification of SCTRAN

SCTRAN, a safety analysis code developed at Xi' an Jiaotong University, can simulate most accidents for SCWR including LOCA [?], thus applicable to accidents at both subcritical and supercritical pressures. A homogeneous model and a four-equation dynamic slip model are implemented as optional modes for subcritical pressure conditions. Its capability for SCWR safety analysis has been verified [?], and it has been applied to analyze accident consequences for Chinese pressure vessel type concepts such as CSR1000 [?] and CGNPC SCWR [?].

To simulate radiation heat transfer inside the HEC of the Canadian-SCWR, a radiation heat transfer model was incorporated into SCTRAN. Since fuel pins absorb different amounts of radiation heat from different directions, resulting in prominent circumferential heat conduction, the original one-dimensional heat conduction model in SCTRAN was updated to a two-dimensional version.

A. Development of Two-Dimensional Heat Conduction Model

The differential equation of heat conduction with an internal heat source is given as

$$\rho c_p \frac{\partial T}{\partial t} = \nabla \cdot [\lambda \nabla T] + S,$$

where ρ is density, c_p is specific heat capacity, T is temperature, t is time, λ is conductivity, and S is heat source. The first term on the right side represents spatial energy variation while the second term represents the heat source in the heat structure.

In two-dimensional polar coordinates, the conduction equation can be rewritten as

$$\rho c_p \frac{\partial T}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(\lambda r \frac{\partial T}{\partial r}\right) + \frac{1}{r^2}\frac{\partial}{\partial \phi}\left(\lambda \frac{\partial T}{\partial \phi}\right) + S,$$

where r is the radial distance and ϕ is the circumferential angle. To reduce calculation iteration while guaranteeing accuracy, a fuel rod is divided into four sectors in the circumferential direction. The mesh layout for the 2D heat conduction is shown in Fig. 1 [Figure 1: see original paper]. Points P, N, S, W, and E represent nodes requiring temperature calculation. Material properties and temperatures in a half mesh interval are assumed constant over a time interval. For the control volume of point P (shadowed in Fig. 1), the internal energy increase equals the internal heat source plus heat conducted from surrounding control volumes. Thus, the discrete form of Eq. (2) can be given as

$$(\rho c_p)_p \frac{(r_n + r_s)}{2} \Delta r \Delta\theta \frac{(T_P - T_P^0)}{\Delta t} = \left[\frac{r_n \lambda_n (T_N - T_P)}{(\delta r)_n} + \frac{r_s \lambda_s (T_P - T_S)}{(\delta r)_s}\right] \Delta\theta + \left[\frac{\lambda_e (T_E - T_P)}{(\delta\theta)_e r_e} + \frac{\lambda_w (T_P - T_W)}{(\delta\theta)_w r_w}\right]$$

where θ is the angle in the circumferential direction. The left side represents the internal energy increase of the control volume at point P. The first term on the right side is heat conducted from points N and S. The second term is heat conducted from points E and W, while the last term denotes the internal heat source. Eq. (3) can be simplified as

$$a_P T_P = a_E T_E + a_W T_W + a_N T_N + a_S T_S + b,$$

where T represents mesh node temperature and a and b are coefficients. Every node in the mesh layout has the same heat conduction equation as Eq. (4), with five unknown node temperatures in each equation. Simultaneously solving temperature equations for all nodes is time-consuming. To save computer memory and calculation time, conduction equations are solved sector by sector. Node temperatures in one radial sector are solved together, assuming that node properties of the west and east sectors apply values from the last iteration:

$$a_P T_P^{(n)} = a_N T_N^{(n)} + a_S T_S^{(n)} + a_E T_E^{(n-1)} + a_W T_W^{(n-1)} + b.$$

Eq. (5) has only three unknown parameters and can be solved by the Tridiagonal Matrix Algorithm (TDMA) with corresponding boundary conditions [?]. When solution of one radial sector is complete, physical properties of the nodes are updated. Radial heat conduction calculation proceeds to the next sector clockwise until all node temperatures satisfy the convergence criterion (for each node, temperature difference < 1 °C between two iterations). In this way, the 2D heat conduction solution is transformed into multiple 1D solutions, greatly promoting calculation efficiency.

B. Development of Radiation Heat Transfer Model

As a boundary condition for heat conduction solution, radiation heat transfer inside the HEC is highly related to fuel configuration. The cross-section of an HEC channel is illustrated in Fig. 2 [Figure 2: see original paper]. The radiation enclosure consists of the outer surface of the central channel, surfaces of fuel rods in inner and outer rings, and the tube liner. The method for solving radiation heat transfer adopted from RELAP5 is referenced [?].

The following assumptions are made:

1. Radiation heat transfer between different elevations is ignored.
2. The coolant in the pressure tube neither emits nor absorbs radiant thermal energy.
3. Radiation exchange between water steam and fuel surface does not produce a significant effect on LOCA/LOECC results.
4. Reflectance from a surface is independent of reflected direction and radiation frequency.

1. Net Radiation Heat Flux The radiosity of a surface is the radiant energy flux leaving a surface (i.e., emitted energy flux plus reflected energy flux). Energy balance for the ith surface is

$$R_i = \varepsilon_i \sigma T_i^4 + \eta_i \sum_{j=1}^{n} \frac{A_j}{A_i} R_j F_{ji},$$

where R is radiation heat flux, σ is Boltzmann coefficient, η is reflectivity, A_i is area of surface i, A_j is area of surface j, and F_{ji} is the view factor from surface j to surface i. Considering energy conservation, we have

$$A_iF_{ij} = A_jF_{ji},$$

where F_{ij} is the view factor from surface i to surface j. Substituting Eq. (7) into Eq. (6) yields

$$R_i = \varepsilon_i \sigma T_i^4 + \eta_i \sum_{j=1}^{n} R_j F_{ij}.$$

The net heat flux at surface i, Q_i, is the difference between the radiosity of surface i and radiosity incident on surface i from all surfaces, given by

$$Q_i = R_i - \sum_{j=1}^{n} R_j F_{ij}.$$

Combining Eqs. (8) and (9), the net heat flux of surface i is expressed by

$$Q_i = \frac{\varepsilon_i}{1-\varepsilon_i}(\sigma T_i^4 - R_i).$$

In Eq. (10), reflectivity and emissivity are basic physical properties of surface i. By energy conservation, the sum of reflectivity and emissivity for a sector equals 1. In calculations, surface temperature from the previous time step is applied, and therefore radiosity is calculated explicitly.

2. Radiosity Solution Based on Eq. (8), the radiosity of all surfaces can be calculated by:

$$\begin{cases} i=1: & (1-\eta_1F_{11})R_1 + (0-\eta_1F_{12})R_2 + (0-\eta_1F_{13})R_3 \cdots + (0-\eta_1F_{1n})R_n = \sigma\cdot\varepsilon_1T_1^4 \\ i=2: & (0-\eta_2F_{21})R_1 + (0-\eta_2F_{22})R_2 + (0-\eta_2F_{23})R_3 \cdots + (0-\eta_2F_{2n})R_n = \sigma\cdot\varepsilon_2T_2^4 \\ i=i: & (1-\eta_iF_{i1})R_1 + (0-\eta_iF_{i2})R_2 \cdots (0-\eta_iF_{ii})R_i \cdots\cdots (0-\eta_iF_{in})R_n = \sigma\cdot\varepsilon_iT_i^4 \\ i=n: & (0-\eta_nF_{n1})R_1 + (0-\eta_nF_{n2})R_2 + (0-\eta_nF_{n3})R_3 \cdots + (0-\eta_nF_{nn})R_n = \sigma\cdot\varepsilon_nT_n^4 \end{cases}$$

Surface emissivity is regarded as constant. From [?], surface emissivity of the central channel is 0.34 and emissivity of fuel rods and liner is 0.8. For a grey body, absorptivity equals emissivity, hence $\eta = 1-\varepsilon$, which simplifies the formula. Due to central symmetry of the fuel assembly (Fig. 2), the pressure tube and center channel are circumferentially divided into 32 sectors and each fuel rod into 4 sectors. With view factors for each surface calculated by GEOFAC code [?], the radiosity matrix can be solved and net radiation heat flux obtained using Eq. (11).

C. Heat Conduction Solution Considering Radiation and Convection Effects

The net heat flux created by radiation and convective heat transfer serves as the boundary condition for the 2D heat conduction:

$$\lambda \frac{\partial T}{\partial r}\bigg|_i = h_i(T_i - T_{sk}) + Q_i,$$

where h_i is the convective heat transfer coefficient and T_{sk} is the coolant temperature. The radiation heat flux Q_i in Eq. (12) is also a function of surface temperature. Bulk-fluid temperature for all sectors in an enclosure is assumed identical. Natural convection heat transfer coefficient inside the HEC is calculated by Churchill-Chu correlation [?]. Forced convection heat transfer in the HEC is primarily calculated by Dittus-Boelter correlation [?] at subcritical pressure and Jackson correlation [?] at supercritical pressure during LOCA/LOECC. To simplify solution, surface temperatures from the previous time step are adopted to compute radiation heat flux explicitly, avoiding iterative solution of radiation heat flux.

III. Verification of SCTRAN Radiation Heat Transfer Model

Due to lack of experimental data, verification of the radiation heat transfer model was performed through code-to-code comparison with the Canadian system code CATHENA. As a one-dimensional, two-fluid thermal-hydraulic code [?], CATHENA includes 1D and 2D heat conduction models (GENHTP). Its capability to evaluate radiation heat transfer has been validated by Lei and Goodman [?].

The 64-element Canadian-SCWR bundle was analyzed for verification. The HEC cross-section is shown in Fig. 2. Bundle power was 9.34 MW, with fuel pins in inner and outer rings occupying 44% and 56% of power, respectively. Moderator temperature outside the pressure tube was 80 °C and heat transfer coefficient between moderator and pressure tube was 1000 W/($m^2 \cdot K$). Three steady-state cases with different decay heat levels (2%, 3%, and 4% of rated power) were simulated by SCTRAN and CATHENA.

In simulations, decay heat was transferred from fuel rod to inner surface of pressure tube only by radiation heat transfer. Each sector in the HEC was marked by a number (Fig. 3 [Figure 3: see original paper]). Table 1 lists surface temperatures of the 64-element fuel bundle at 2%, 3%, and 4% power levels calculated by SCTRAN and CATHENA. SCTRAN results agree well with CATHENA, with maximum absolute error of 6.1 °C and maximum relative error of 0.9%. This demonstrates that the 2D heat conduction and radiation heat transfer models developed for SCTRAN provide guaranteed calculation accuracy.

IV. Safety Analysis of LOCA/LOECC

The ability to maintain core components below melting temperatures in postulated accidents is referred to as the "no-core-melt" concept, an important safety goal for the Canadian-SCWR. The LOCA with loss of ECCS was simulated with modified SCTRAN to evaluate whether the Canadian-SCWR can achieve this "no-core-melt" goal.

A. Introduction to Canadian-SCWR

The conceptual Canadian-SCWR design features a modular configuration that separates coolant from moderator, similar to CANDU reactors. The reactor core consists of 336 fuel channels, each housing a 5-m long fuel assembly, designed to generate 2540 MW thermal power or approximately 1200 MW electric power. In the conceptual design (Fig. 4 [Figure 4: see original paper]), light water coolant enters the inlet plenum and flows downward through the central flow tube of the channel. Near the channel bottom, coolant exits the central flow tube, flows upward through fuel elements (fuel assembly), and arrives at the outlet plenum. From the outlet plenum, high-temperature and high-pressure coolant is fed to the high-pressure turbine. The cylindrical vessel houses relatively low-pressure and low-temperature heavy water moderator. Main parameters of the Canadian-SCWR are listed in Table 2 .

HEC is adopted for the Canadian-SCWR conceptual design, consisting of a pressure tube, outer liner tube, insulator, and inner liner tube. The pressure tube is surrounded by heavy water moderator. A key advantage of HEC is that in the unlikely event of LOCA/LOECC, heat in the fuel would be transferred by thermal radiation to the liner tube and then conducted to the moderator through the insulator. The 64-element fuel bundle with two concentric rings is equipped inside the pressure tube.

B. SCTRAN Model

The SCTRAN model idealization of the Canadian-SCWR was developed based on the conceptual design. To more accurately simulate radial power distribution effects, the 336 channels in the reactor core are divided into four groups (84 channels each): "average power channels (AP)" , "high power channels (HP)" , "medium power channels (MP)" , and "low power channels (LP)" [?], as shown in Fig. 5 [Figure 5: see original paper] in the radial direction. In the axial direction, fuel bundles and coolant channels are divided into 10 parts. Axial power distribution for fuel bundles refers to [?]. Fuel pins in inner and outer rings of each group are simulated by independent heat structures, which exchange heat separately with coolant.

Heat exchange between central channel and coolant channel is also considered. A time-dependent junction at 350 °C and 1254 kg/s, and a time-dependent volume at 625 °C and 25 MPa, are set as boundary conditions for the main coolant line and main steam line, respectively.

The moderator cooling system contains active and passive moderator cooling systems. Under accident conditions, only the passive moderator cooling system (PMCS) is used [?]. The PMCS is simulated by volume 800 (Fig. 5), serving as an ultimate heat sink that activates automatically during transients. Moderator temperature of 80 °C is assumed, with constant heat transfer coefficient of 1000 W/($m^2 \cdot$K) between moderator and pressure tube [?], indicating that moderator removes core heat through natural convection. No active system prepares for the Canadian-SCWR system when LOCA/LOECC occurs. The moderator system is assumed to remain intact throughout the entire process. Turbine stop valves installed on main steam lines are tripped by the "power scram" signal.

Other initial conditions and assumptions are as follows: - Reactivity feedback in the core is neglected in both steady-state and accident simulations. - Decay heat is calculated according to previous work [?]. - Emissivity of fuel sheath and liner tube is 0.8, while emissivity of central channel is 0.34. - Natural convection heat transfer coefficient in HEC is calculated by Churchill-Chu correlation [?]. - Referring to the ASM specialty handbook [?], melting temperature of SS310 is 1400–1450 °C; thus, fuel cladding melting temperature is set as 1400 °C. - Due to radiation heat transfer effects, each fuel sheath has four sector temperatures; the maximum sector temperature is applied to describe surface temperature variation.

C. Results for LOCA/LOECC

The core loses most of its inventory and suffers deteriorated cooling conditions when LOCA occurs. Heat in the fuel can only be removed by passive natural convection of high-temperature steam and thermal radiation exchange between fuel sheath and liner tube, while ECCS is simultaneously lost. In this process, radiation heat transfer plays a dominant role. The double-ended break accident at the cold leg is analyzed as it represents the most severe LOCA scenario. Since the HP group has the largest power fraction and highest cladding temperature, it is selected as the main analysis object.

In the simulation, break at the cold leg occurred at 0.5 s. At 1 s, control rods started to drop and core power decreased, while the turbine stop valve closed. Coolant flowrate from the main coolant pump was maintained for the first 10 s and then decreased to zero within 5 s (determined by pump inertia).

Fig. 6 Figure 6: see original paper shows coolant flowrate through the break. When break occurred, large amounts of coolant drained through the break at critical velocity. Core pressure decreased quickly to subcritical level due to coolant loss. Coolant flowrates in different channels are shown in Fig. 6(b). Break location in the cold leg resulted in reversed flow in each channel. Large amounts of coolant at 625 °C from the outlet plenum were directed to the fuel channel, leading to greatly degraded cooling conditions. Consequently, fuel sheath temperatures increased rapidly during this period.

Fig. 7 Figure 7: see original paper shows heat removed from fuel rings and heat variation in the HP group during the first 50 s. Fig. 7(b) shows maximum cladding surface temperature (MCST) of components. Initially, nuclear power produced in fuel rods could not be completely removed to coolant, and MCST of both rings increased sharply. With power decreasing and reversed flow establishing, MCSTs showed a slower increasing trend but dropped slightly at about 15 s. However, when coolant in the outlet plenum was exhausted, cooling conditions in coolant channels deteriorated. Channels filled with high-temperature steam and radiation heat transfer became dominant in HEC at around 30 s. Heat transferred by radiation and natural convection at this time was insufficient to remove decay heat from inner rings, while decay heat from outer rings could be completely removed. MCST of the inner ring began rising again at around 15 s, while that of the outer ring kept decreasing.

As time progressed, fuel cladding temperature of the inner ring increased continuously while the outer ring experienced MCST decrease. As shown in Fig. 8 [Figure 8: see original paper], MCST of the inner ring of the HP group began dropping slowly after reaching a peak value of 1236 °C. The fuel rod would reach steady state when heat removed by radiation and convection equals decay heat. However, decay heat decreased slowly with time, resulting in continuous MCST decrease. Behaviors of the other three power groups are similar to the HP group, which has the highest MCST due to its highest power fraction.

According to the analysis, decay heat of Canadian-SCWR fuel pins during LOCA/LOECC can be transferred to moderator outside the pressure tube through natural convection and radiation heat transfer. Fuel rods in the two rings experience different sheath temperature variations. Fuel rods in the outer ring are closer to the pressure tube, enhancing radiation heat exchange with the pressure tube, while radiation heat transfer for fuel rods in the inner ring is less effective. This explains why fuel rods in the inner ring experience a longer period of sheath temperature increase. Peak cladding temperatures of the two rings are 1236.3 °C and 1177.3 °C, respectively, much lower than fuel cladding melting temperature.

V. Conclusion

This paper assesses the inherent safety of the 64-element Canadian-SCWR. A two-dimensional heat conduction model and radiation heat transfer model were developed and successfully incorporated into SCTRAN. Validity of the newly developed models was verified by CATHENA, a system code developed by AECL.

A SCTRAN idealization of the Canadian-SCWR conceptual design was developed and used to simulate steady-state thermal-hydraulic conditions. LOCA/LOECC analysis shows that even with total loss of core coolant inventory, radiation heat exchange among sheaths and natural convection of high-temperature steam can remove a certain amount of core decay heat. Based on conservative assumptions, maximum fuel sheath temperatures in inner and outer rings of the HP group are 1236 °C and 1177 °C, respectively, lower than cladding material melting point. Results demonstrate that the Canadian SCWR is capable of achieving the "no-core-melt" design objective under LOCA/LOECC.

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