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Transient Analysis of the CANDU-9 480/SEU Reactor

J.C. Shin, J.H. Park, K.N. Han, and H.C. Suk
Korea Atomic Energy Research Institute
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CANDU-9 480 / SEU 원자로의 과도변화해석

신정철 · 박주환 · 한기남 · 석호천
한국원자력연구소
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Abstract

The thermal-hydraulic transient analysis of the proposed CANDU-9 plant was performed. Several major transients were analyzed if they meet the heat transport system design requirements. The proposed heat transport system configuration and the preliminary sizes of system equipment are justified by analysis in terms of the fuel integrity and the high system pressure limit during transients. The compliance with AECB R-77 requirements for CANDU-9 reactor was estimated. The analysis results showed that for each postulated accident the peak pressure values in the reactor headers are within the acceptance criteria given in ASME code requirements and the fuel overheating is prevented. One pump start-up during the reactor start-up operation was analyzed to investigate the flow reversal through the fuel channel, which is specific in the proposed CANDU-9 plant.

요 약

제안된 CANDU-9 원자로의 열수력 과도변화상태가 해석되었으며 주요한 몇개의 과도변화가 열수송계통의 설계요건을 만족시키는지에 대해 평가되었다. 열수송계통의 과도변화시 핵연료의 건전성과 계통압력상승의 제한 측면에서 분석된 본 해석결과에 따라서 제안된 열수송계통형상과 열수송계통기기의 예비크기가 확정 및 검증되었다. AECB R-77 요구조건에 대한 CANDU-9 원자로의 만족여부를 평가하였다. 해석결과, 각 과도변화시 원자로 모관의 고압첨두치가 ASME코드의 요구조건에 따른 허용범주내에 있었으며 핵연료의 건전성이 확인되었다. 원자로 가동운전시 제안된 CANDU-9 원자로의 고유적인 핵연료채널을 통한 역류현상을 규명하기 위하여 한개의 펌프가 시동될때의 과도변화현상을 해석하였다.

1. Introduction

The CANDU-9 480/SEU was proposed as one of the advanced heavy water reactor(HWR) by AECL (Atomic Energy of Canada Ltd.) and KAERI(Korea Atomic Energy Research Institute)[1]. This reactor contains 480 horizontal fuel channels using Slightly Enriched Uranium(SEU) with a net electrical output of 1038 MW. KAERI participated this advanced HWR project and several design improvements and analyses for nuclear steam supply system have been made[2]. Although the proven design concept based on the Bruce design is mostly adopted, the overall performance characteristics of the advanced CANDU-9 Heat Transport(HT) system must be confirmed analytically. The thermal-hydraulic performance of HT system was analyzed in this study. The purposes of the present analysis are to confirm if the HT system meets the R-77 Overpressure Protection Requirements[4] during the overpressurization transients and the reactor fuel is cooled at all times so that the fuel integrity is maintained during transients. Also the preliminary sizes of HT system major equipment given in CANDU-9 Technical Description[1] have been justified through the present study. The HT system equipment and piping are classified as nuclear class 1 components. For the process system design point of view, the thermal-hydraulic transient behavior of the HT system equipment must be studied to find the design conditions of that equipment to be supplied based on ASME code requirements as well as to understand the postulated accident scenario. Stress intensity limits for nuclear class 1 components shall be in accordance with the ASME Code Section III Division 1 Subsection NB-3200. The stress limits vary depending on the service limits A(normal), B(upset), C(emergency) and D(faulted). The following major transients were considered in this report :

Level A condition
- One pump start-up

Level B condition
- Reactor trip
- Rapid cooldown
- Loss of class IV power

Level C condition
- Pump seizure

The proposed HT system configuration is a single loop design as shown in Figure 1[1]. With a single pump running, part of the flow from the running pump will bypass the steam generator via the ROH and proceed in the reverse direction through the fuel channels associated with the parallel pump at the same end of the reactor. The purpose of the one pump start-up analysis is to find the flow reversal through the reactor core when only one pump starts during the start-up operation. The rapid cooldown and the reactor trip are the representative transients for the reactor depressurization. The loss of class IV power and the pump seizure are the representative transients for the reactor overpressurization.

2. CANDU-9 Heat Transport System Configuration

The proposed CANDU-9 HT system has two identical loops as shown in Figure 1 in which each loop is connected to one reactor outlet header(ROH) at the end of the reactor so that the loop separation concept like CANDU-6 is not applicable. This kind of HT system configuration is similar to the Bruce HT system design with one ROH and two reactor inlet headers(RIH) at each end of the reactor. The HT system contains two Steam Generators(SG) and two HT pumps at each end of the reactor. Each RIH is fed by a separate HT pump. This is different from the Bruce reactor in which two RIHs are fed by both HT pumps at each end of the reactor. The CANDU-9 reactor has a fully interlaced feeder arrangement. At each end of the reactor every row of fuel channels consists of alternative feeder connections from inlet and outlet headers with alternate inlet feeders connected to the same inlet header. One 12″ interconnect line between two ROHs exists for the flow stability consideration caused by two phase flow phenomena.

The CANDU-9 HT system arrangement can accommodate a range of reactor sizes from 480 channels up to 640 channels. Sufficient margins must be provided to accommodate current minimum allowable performance specification(MAPS), increased void reactivity uncertainty, the effects of larger core sizes, and reasonable future requirements. For a LOCA with critical sized inlet header breaks, the reactivity introduced by voiding of fuel channels causes the reactivity to increase until the shutdown systems eventually overtake the void-induced reactivity transient. The above HT system configuration has design margin to avoid the prompt critical condition[3].

Due to a specific reactor lattice design, the HT system arrangement sets the rate of void reactivity insertion on a LOCA, the magnitude of short term void reactivity and alternate HT system arrangements were examined in an effort to increase shutdown margin on LOCA relative to that provided by two loop arrangements[3]. Assessments of other important safety requirements, including pressure tube integrity and core refill after a LOCA, were also made[3]. Each HT system arrangement was also evaluated for pipe whip, accessibility and maintainability, simplicity, and economics. All relevant design requirements of the HT system were considered in the selection of the configuration.

3. Design Requirements

The HT system design shall comply with Atomic Energy Control Board(AECB) Regulatory Document R-77 Requirements for Overpressure Protection[4]. The stresses in this system must not exceed those allowables for ASME service levels specified by R-77 document. This selection depends on the expected frequency of the event and whether the first shutdown system is credited or assumed to be available. The "first shutdown system" is that which would normally trip first for the event. It may be either of two shutdown systems, SDS1 or SDS2. According to R-77 document when the first shutdown system (SDS1) fails for high pressure transients, then one higher service level of service limit must be satisfied for that event with SDS1 unavailable. Also this document requires that only the second trip parameter is credited unless the first trip parameter is the HT high pressure trip. In the analyses to demonstrate that the requirements are met, the process system protective action including regulating system action is not credited.

For level B service condition, the maximum pressures must be within 110% of the design pressure limit. For level C service condition, the maximum pressure must be within 120% of the design pressure, or the calculated stress intensity and other design limitations for service limit C specified in NB-3000 are not exceeded for each of the components in the protected system. For level D service condition, the maximum pressure must be limited such that the limitations specified in the ASME code Section III Subsection NB-3200. These pressure limits are based on the assumption that the internal pressure is by far the most important contributor to the primary stress among all loadings. However, these limits are not a final acceptance criteria. The ASME Code stress intensity limits must be met by stress analysis.

If the flow reversal exists during the one pump start-up operation, the flow rate should be limited not to damage the fuel bundle or the fuel channel components and feeders. From the operating experience the maximum flow rate to move the fuel bundle is considered to be 10kg/s per channel. When the parallel pump is started later, the flow in the pass will push the fuel forward again. The fuel impact during start-up may cause additional stress to the fuel bundle. The flow network analysis should confirm the acceptability of this one pump start-up operation.

During the overpressurization transients the steam space in the pressurizer shall limit the pressure increase within the acceptance value required by the ASME Code. The pressurizer connecting line should be sized such that the HT system pressure shall remain above NPSH required by the HT pumps after a reactor trip or a rapid cooldown. The HT pump inertia shall be selected to ensure that there is no fuel overheating due to a flow reduction in cases involving pump trips. A loss of class IV power transient will be produced to aid in selection of the pump inertia. The fuel integrity must be maintained during all kinds of transients. The high pressure peaks acting on the pressure tube during the overpressurization transients should not exceed the allowable limit required by the R-77 document.

4. Code Modelling

SOPHT is a thermal-hydraulic code to generate the transient curves for the HT system equipment of CANDU type reactors during normal as well as abnormal operation. The mathematical theory and modelling of thermal-hydraulic system employed in the SOPHT code are described in [5] through [8] in detail.

The formulation of physical network system was initially suggested by Porsching et al. [6] and modified to include the two-phase flow pressure drop multipliers, the pump, and the Nahavandi form[7] of momentum flux term. The method of solution based on Porsching et al. was also used to solve the above transient equations, which are coupled by the equation of state. The steady state program solves the steady state forms of the foregoing conservation equations simultaneously, choosing the pressure, enthalpy and flow as dependent variables to conveniently satisfy the known boundary conditions within the network. The program performs an initial steady state solution using the Newton-Raphson iteration technique. In the transient cases, the thermal-hydraulic equations are linearized and integrated by the fully implicit difference method of Porsching et al. [6].

The nuclear fuel and heat exchanger pipe wall temperature profiles and heat transfer at the boundaries are dependent on the local fluid boundary conditions and are evaluated by solving the one dimensional thermal diffusion equation in cylindrical coordinates. A point kinetic model was used to simulate the reactor kinetics. The fission product decay power is considered in the reactor power calculation.

There are components or equipment in the thermal-hydraulic network whose operating conditions set the boundary conditions or flow restrictions for the network hydrodynamic solution. These components are identified by their type code in the input data and are treated separately according to their design characteristics and operating conditions.

The digital computer control programs including the logic decisions, the control equations and the periodical program executions are modelled by SOPHT code within the thermal-hydraulic networks as a part of the time step solution.

The previous verification of the original version of SOPHT code is given in [9] through [13]. The validation of the workstation version of SOPHT used in the present study was made by the comparison of SOPHT results with the site data from Point Lepreau Generating Station(PLGS) as shown in Wolsong-2, 3, 4 PSAR. The detailed information about the verification of the present version can be found in [14]. As a result the overall performance of the simulation was reasonable although the differences can be attributed to primarily modelling.

The SOPHT model of Heat Transport and Auxiliary systems were developed as shown in Figure 2 based on the HT system configuration in Figure 1. This model is called a "four quadrant model : which means that four similar loops of the HT circuit are modeled. One core path in Figure 2 represents 120 fuel channels since CANDU 9 has 480 fuel channels. In addition to the HT system, the Pressure and Inventory Control(PIC) system, the Feedwater system and the Main Steam system are also included in the computer modeling. The part of Shutdown Cooling (SDC) lines attached to two RIHs at each side of the reactor was modelled to analyze the one pump start-up transient. The HT Purification system was not included since it does not affect the transient results considered in this report.

The steady state input data was prepared using the SOPHT User's Manual[15]. The SG size was estimated according to the data provided by the supplier[16]. In general the SG height is about the same as that of Wolsong-2 but the diameter and the num-

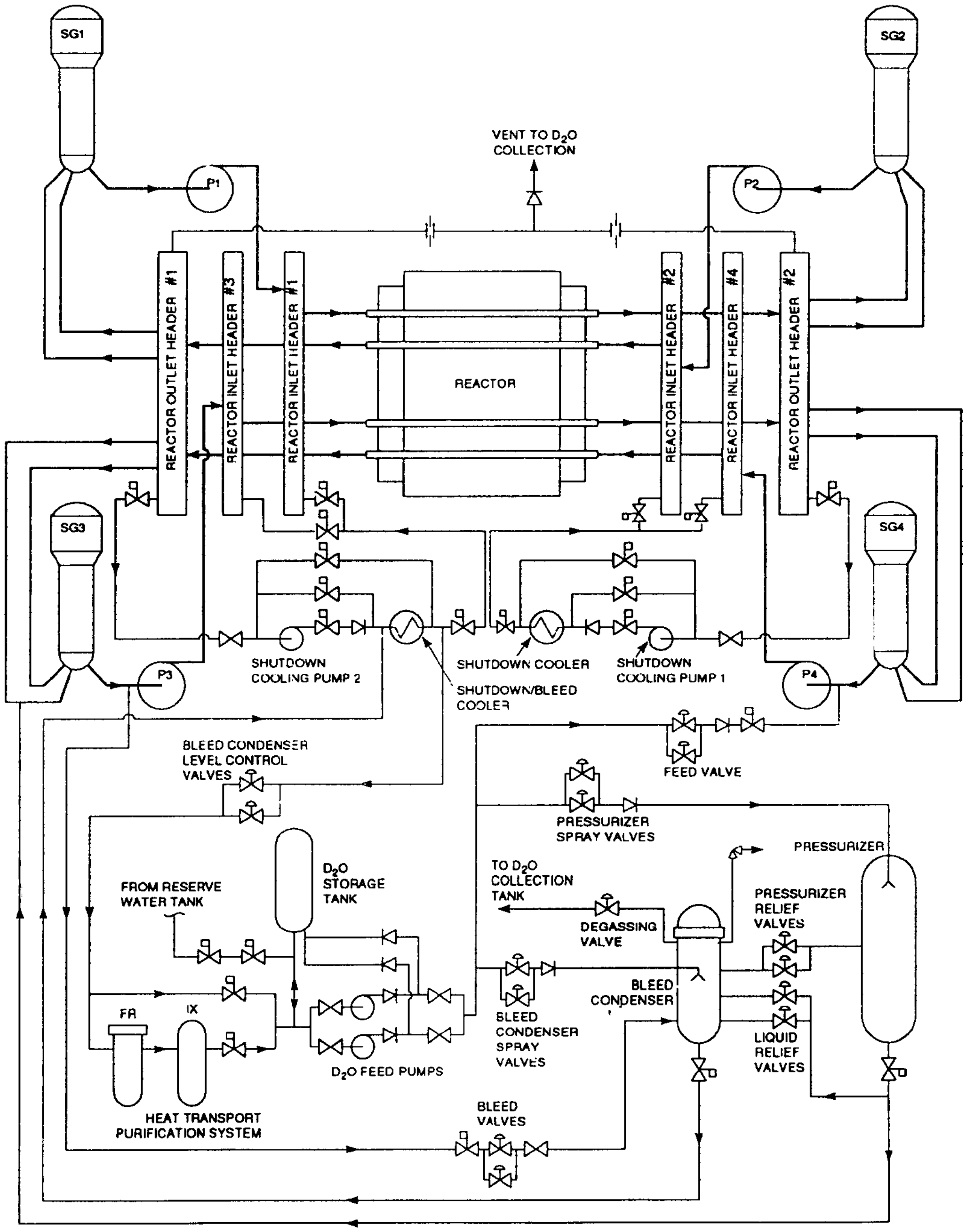

Fig. 1. Simplified Flow Diagram of CANDU-9 Heat Transport and Auxiliary Systems.

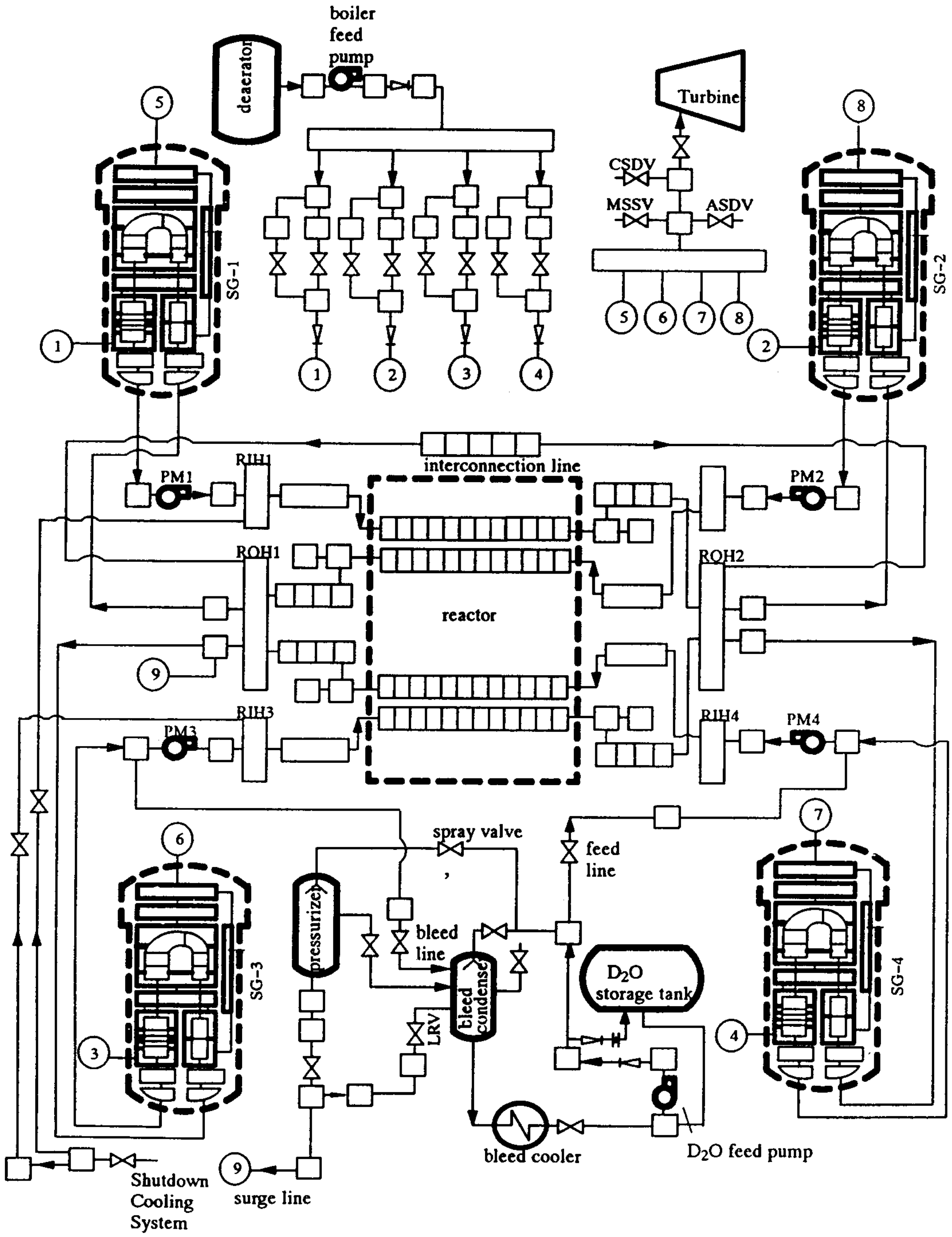

Fig. 2. CANDU-9 4 Quadrant Model.

ber of U-tubes becomes larger. The pressurizer diameter was also increased from Wolsong-2 data to accommodate the larger inventory of CANDU-9.

The HT pump head and flow data were based on the CANDU-9 Technical Specification[1]. The pump characteristic curve was tuned against the pump curve used in NUCIRC analysis[17] for CANDU-9. NUCIRC is a steady state thermal-hydraulic code designed to analyze the HT system for a variety operating conditions. It determines the critical channel power ratios for dryout during overpower for each channel or any required number of channels and determines the feeder pipe sizes that meet the established thermal-hydraulic criteria. The valve opening and stroking time of most control valves in CANDU 9 are assumed to be the same as Wolsong-2. However the different valve size and mass flow rate are input according to the transient analysis basis document[18]. Although the feeder sizes in reality are different each other, one average feeder size(G5 feeder) and the following one fuel channel was modelled. At the present time the detailed layout, equipment and line sizes for most auxiliary systems of CANDU-9 are not available. However CANDU-9 PIC system, Main Steam system and Feedwater system are similar to CANDU 6 design. Therefore CANDU 9 SOPHT model for these systems was scaled up from the corresponding CANDU 6 model[19] and details are described in [20].

5. Results and Discussion

5.1. Steady-State Condition

In order to verify the assumption of using one average fuel channel in the SOPHT model, the steady state condition was compared to the NUCIRC analysis[17] and results are shown in Table 1. The major

Table 1. Comparison of the Steady State Condition Between NUCIRC and SOPHT.

1. 100% Full Power

Parameter	NUCIRC	SOPHT
Total Core Flow	12090kg/s	12088kg/s
Header to Header Pressure Drop	1.464 MPa	1.455 MPa
Outlet Header Pressure	10 MPa(a)	10 MPa(a)
Outlet Header Enthalpy	1383 kJ/kg	1385 kJ/kg
Outlet Header Quality	2.73%	2.71%
Outlet Header Temperature	310 °C	310°C
Inlet Header Pressure	11.46 MPa(a)	11.45 MPa(a)
Inlet Header Temperature	264.7 °C	264.1°C

2. 85% Full Power

Parameter	NUCIRC	SOPHT
Total Core Flow	12380 kg/s	12354 kg/s
Header to Header Pressure Drop	1.427 Mpa	1.441 MPa
Outlet Header Pressure	10 MPa(a)	10 MPa(a)
Outlet Header Enthalpy	1340 kJ/kg	1341 kJ/kg
Outlet Header Quality	0%	0%
Outlet Header Temperature	308.0 °C	307.9°C
Inlet Header Pressure	11.43 MPa(a)	11.43 MPa(a)
Inlet Header Temperature	265.2°C	264.4°C

thermal-hydraulic parameters are in good agreement with those in the NUCIRC analysis. For 85% FP case in which the ROH quality is zero, the parameters also agree well each other. During the normal operating condition, all process, control and safety systems retain their functional capability as designed. The reactor power is maintained at the design full power level by the reactor regulating system. The reactor core is continuously being refuelled in accordance with a prescribed refuelling scheme. During the normal operation, the ROH conditions are maintained at 10 MPa(a), 310°C and 3020 kg/s.

5.2. One Pump Start-up

In a single loop design when a single pump is started, the initial flow in that pass may push the fuel bundles in the channels associated with the parallel pump backwards towards the inlet shield plug. However, with an interconnection via the Shutdown Cooling(SDC) inlet header isolation valves, the flow reversal could be alleviated. SOPHT input data was prepared to simulate this one pump start-up operation with SDC line modelled.

To reduce the HT system pressure from the normal power operation, the SG pressure should be lowered. The secondary side energy was adjusted from the high pressure heater and the flow rate through the feedwater control valves. Consequently, very low flow through the SG was achieved to balance the decay heat in the primary side. All four pumps are running initially and then trip all pumps at t=0 to establish the initial condition for the simulation of one pump start-up operation.

Two cases, SDC valve closed and open, were studied to investigate the flow reversal through the core as shown in Figures 3 and 4. From 90 to 150 seconds, a low condition flow exists due to thermosyphoning. At t=150 only one pump(PM#1) starts and then the flow reversal in core pass #3, which is connected to RIH#3, occurs. The flow from RIH#1 passes through the core and then 100% of HT

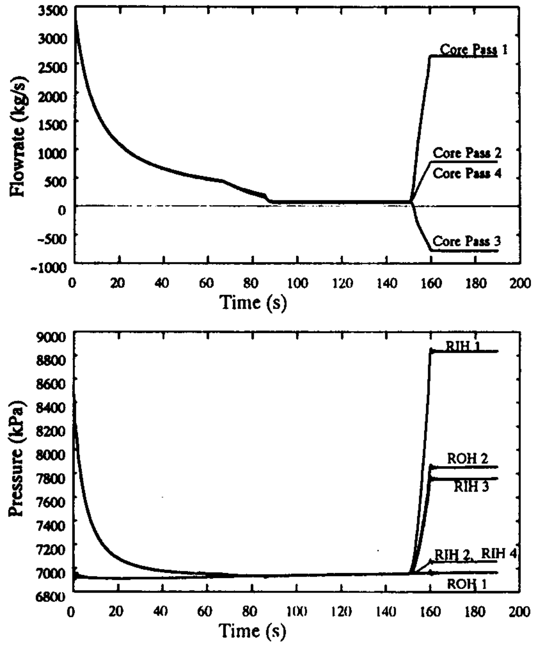

Fig. 3. One Pump Start-up(SDC Valves Closed).

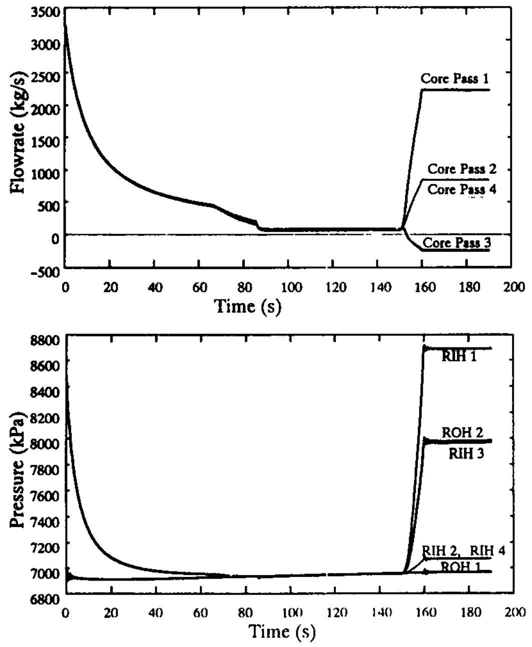

Fig. 4. One Pump Start-up(SDC Valves Open).

pump flow arrives at ROH#2. From ROH#2, the flow splits into four directions ; 30% to SG#2, 30% to SG#4, 10% to the stability line, and 30% to the core pass #3. The flow reversal occurs through the core pass about 30% of nominal flow. Also one quadrant of the HT system(PM#3 side) has a flow reversal. The flow distribution corresponds to the pressure distribution in the HT system is shown also in Figure 3. The ROH#2 pressure is higher than RIH#3, which causes the reversed flow through the core pass #3.

If the SDC valve is open, the pressure difference between ROH#2 and RIH#3 is smaller, as shown in Figure 4, since part of the flow from HT pump #1 bypasses the reactor core through the SDC interconnection line. As a result, only 10% of nominal flow reverses through core pass #3. The amount of flow is 2.4 kg/s per channel with the channel pressure drop of 5 kPa. Even in the case of SDC valve being closed, the amount of flow is 6.5 kg/s with the channel pressure drop of 50 kPa. Therefore it is considered that the one pump operation is acceptable without damaging the fuel bundle if the SDC valve is open.

5.3. Reactor Trip

Initially the reactor is operating at 102% FP allowing 2% for instrumentation errors and all four HT pumps are running. The HT system pressure and the SG pressure are controlled as designed. Fouled SG and equilibrium fuel conditions are assumed. The reactor is tripped by SDS1 with all shutdown rods available. The boiler pressure control system(BPCS) is functioning as designed. Feed and bleed valves are closed and pressurizer heaters are not available. Other pressure control and safety systems are functioning as designed. After the reactor trips four HT pumps are running continuously.

Due to the reactor trip the ROH pressure decreases rapidly from 10.0 MPa to 7.5 MPa in 10 seconds and then increases to equilibrate with the pressurizer to HT system as shown in Figure 5. The minimum HT pressure is above the NPSH required by the HT pump. After around 15 seconds the HT system pressure increases due to the disappearance of void in the HT system at this time and equilibrates with the pressurizer condition. The reactor trips at t = 0 and the reactor power decreases from 102% FP to 10% FP in about 2 seconds and then approaches zero power gradually. The fuel integrity is maintained by keeping the normal flow rate through the core during this transient. The SG flow rate is continuously decreasing due to the reactor trip. The ROH temperature and pressure drop quickly with reactor trip, there is a flow from the pressurizer to the HT system for about 20 seconds, which results in the decrease of the pressurizer level. Since the pressurizer heaters are not available, the pressurizer pressure does not increase.

5.4. Rapid Cooldown

The rapid cooldown is the bounding transient for the reactor depressurization. At t = 0 all the main

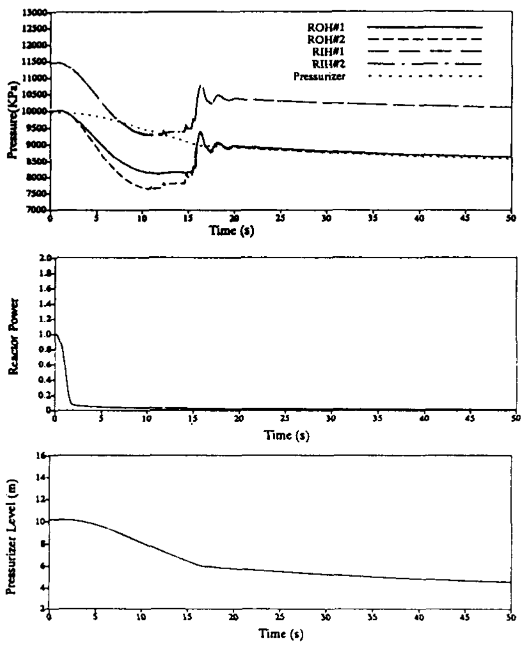

Fig. 5. Reactor Trip.

steam safety valves(MSSV) are fully open when the reactor trips. The transient behavior is similar to the reactor trip except more severe depressurization occurs the MSSV's opening as well as the reactor trip. The mismatch between the heat addition and removal causes a depressurization of the secondary side of the SG. As a result, the primary side temperature and pressure are decreased. The SG flow rate is large at the beginning and then reduced. The fuel is cooled at all times during this transient with high CPR values. The pressurizer level is set at the 100% FP level to see if there is enough D_2O during the rapid cooldown transient. The pressurizer outsurge during the rapid cooldown is the largest from the transients simulated in Table 2. The 16" pressurizer surge line is acceptable in terms of the flow velocity through this line and the HT system low pressure limit.

5.5. Loss of Class Ⅳ Power

Initial reactor power was assumed to be 102% FP. The HT pumps are running and the pressurizer was connected to the HT system. All four pumps are tripped at t=0 to initiate the loss of class Ⅳ power event. The sixteen MSSVs have a role to depressurize the SG secondary side. For conservative analysis, only fourteen out of sixteen MSSVs are credited. The pressurizer heaters are not available. The feedwater pumps are also tripped. The service cooling water is not available for all coolers or heat exchangers. The turbine trip is modelled by a closure of the governor valve in 0.1 second.

The conservative condition for overpressurization transients should cover the input of new SG condition(zero fouling) as well as fouled SG at the end of life condition. Various deposits coated on the SG tubing after a long operation deteriorate heat exchanging performance. During the normal operating condition the HT system using the new SG loses more energy to the secondary system due to the increased heat transport through the new SG. This might cause very low steam quality in the HT system. Accordingly during the transient state the damping

Table 2. Transient Analysis Results

Transient Description	Service Limit	RIH Peak* Press. MPa(a)	RIH Peak* % of Design Press.	ROH Peak* Press. MPa(a)	ROH Peak* % of Design Press.	Min. HT Sys. Press. MPa(a)	Min. Core Flow kg/s	PRZ Max. Flow Rate kg/s	PRZ Level Range m
Loss of Class Ⅳ Power	B	12,250	94.2	11,888	110	9.4	400	−1800	9−15
Loss of Class Ⅳ Power with SDS1 Failure	C	12,000	92.3	11,500	106	9.4	400	−1800	9−14
Pump Shaft Seizure (Case 3)	C	12,500	96.2	11,490	106	7.5	−550	1700	8−13
Pump Shaft Seizure with SDS1 Failure	D	12,280	94.5	11,260	104	7.5	−500	1900	8−14
Rapid Cooldown	B	−	−	−	−	7.5	2700	2100	4−12
Reactor Trip	B	−	−	−	−	7.5	2700	2040	4−11

* Maximum value between two ROHs in four quadrant SOPHT model of CANDU-9 plant.

effect from steam voids in the system is less, which results in higher ROH peak pressure. Therefore the new SG condition was used for all overpressurization transients.

The new fuel produces higher neutron flux which would lead to higher pressure and transients. For each transient that leads to overpressurization, the new fuel condition was assumed.

The trip parameters for the loss of class Ⅳ power transients are the ROH high pressure. The high ROH pressure trip set point was set by 0.5% higher than the nominal set value to take into account instrumental errors.

The HT system pressure increases rapidly following the initiation of event due to the reduction of HT system coolant circulation as shown in Figure 6. The peak pressure value is below the allowable limit required by the ASME Code, 110% of design value for level B service limit as shown in Table 2 and accordingly satisfies the R-77 requirements. After the HT pump stops, there is an insurge into the pressurizer from the HT system due to the HT system pressure increase. The pressurizer size is acceptable during the insurge due to the HT system overpressurization. The Liquid Relief Valve(LRV) is open due to high ROH pressure and allows outflow to the degasser condenser. The fuel is cooled at all times during this transient and the fuel integrity is maintained.

The reactor trip caused by the high HT system pressure results in the following decrease of the HT system pressure. The pressurizer level decreases with an outsurge to the HT system after the reactor trip. This outsurge continues as the HT system pressures equilibrate. The neutron power increases up to about 120% FP after the initiation of the event. After reactor trip, the reactor power decreases rapidly to decay power level. The HT system pressure and temperature reach their peak values after the reactor trip and then decrease.

For all overpressurization transients, the HT system pressure increases until the reactor trip signal is initiated. In general the later the reactor trip signal is initiated, the higher the HT system pressure and temperature increase for high pressure transients. The HT system behavior for the loss of class Ⅳ power with SDS1 failure is similar to the one with SDS1 available except that the reactor trip occurs lately due to the late SDS2 trip with higher trip pressure set point. As seen in Table 2, the ROH peak value is lower than the allowable limit required by the ASME Code Section III Division 1 Subsection NB-3200.

The pressure and temperature of the secondary coolant in the steam generator also increases due to the increased pressure and temperature in the primary side. The MSSVs start to open due to high steam pressure at about the same time as the reactor trip and allows steam outflow to the atmosphere.

5.6. Pump Seizure

This event starts when one of the HT pumps experiences a shaft seizure. One HT pump is assumed to be run down at $t=0$ and comes to a complete

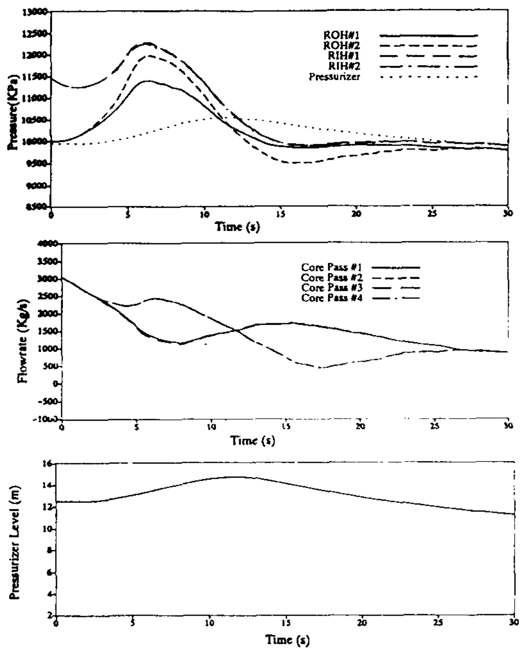

Fig. 6. Loss of Class IV Power.

stop in a few seconds. During overpressurization, the HT system equipment at the other side of the pressurizer may experience a higher pressurization than that at the same side of the pressurizer. Four different cases were analyzed to find the bounding condition in terms of the overpressurization of HT system and fuel dryout. For all 4 cases, new SG, fresh fuel and SDS1 trip were assumed. For cases 1 and 2, HT pump#1 which is located at the pressurizer side was seized. For cases 3 and 4, the HT pump#2 which is located at the other side of the pressurizer was seized. In addition the pump seizing period may affect the transient behaviour of the HT system. For cases 1 and 3, 2 seconds seizing time was assumed while 6 seconds seizing time was assumed for cases 2 and 4. As a result it was found that case 3 is the bounding condition in terms of both the overpressurization and the fuel dryout points of view.

Figure 7 shows that RIH#2 pressure decreases abruptly due to the seizure of HT Pump#2. Due to the pump seizure, the flow rate through the pump#2 decreases in 2 seconds and then increase. The flow through the core pass #2 reverses for a short period of time and then recovers toward its normal direction. As a result the fuel dryout occurs for a short period of time. However, preliminary safety analysis indicated that the fuel sheath temperature is acceptable. If the seizing time increases to 6 seconds(case 4) the flow reversal disappears but the fuel dryout still occurs with the value of critical power ratio(CPR) less than 1.0.

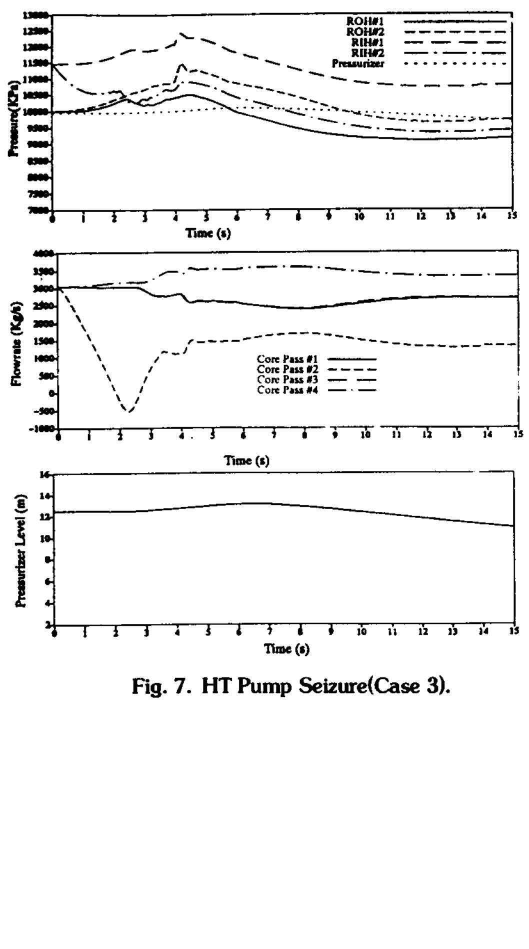

Fig. 7. HT Pump Seizure(Case 3).

There is an insurging flow to the pressurizer from the HT system for a short period of time due to the increases of the temperature and pressure of the HT system and then the outsurge flow occurs as the temperature and pressure of the HT system decrease. Table 2 shows that the high pressure peaks are much less than the ASME code requirements. With the governor valve closed after the reactor trip, the temperature and pressure in the steam generator are increased until the MSSVs open. The pressure of ROH#2 is higher than that of ROH#1, which makes the flow through the ROH interconnecting line.

6. Conclusions and Recommendations

The simulation of primary heat transport system of the proposed CANDU-9 480/SEU plant was investigated and the related major transients were analyzed. The HT system configuration and the preliminary sizes of system equipment were justified by the HT loop analysis in terms of the ASME code requirements and the reactor fuel integrity. For the overpressurization transients the high pressure peaks were below the service limit required by the ASME code and therefore the R-77 overpressure protection requirements were satisfied. The design condition of CANDU-9 PHT pump was adequate to prevent the fuel overheating during depressurization transients.

Differently from the current CANDU-6 plant, the proposed CANDU-9 plant had a flow reversal through the fuel channels associated with the parallel pump at the same end of the reactor during the one pump start-up operation. The analysis results showed that the flow reversal through the fuel channel occurred but didn't result in any damage on the fuel bundle.

Although the flow reversal and fuel dryout occurs at the seized pump core pass during the pump seizure event, the preliminary analysis results indicated that the fuel sheath temperature was acceptable. Further analysis is recommended to evaluate the effect of various trip parameters on the transient behavior in detail design stage.

The preliminary size of pressurizer was adequate to accommodate the swell and shrinkage during the HT system overpressurization transients. The pressurizer connecting line was designed such that HT system pressure remained above NPSH of HT Pump during depressurization transients.

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