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SuperCritical Light Water Reactor (SCLWR) with Intermediate Heat Exchanger (IHX)

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Abstract – *All recently proposed SCLWR concepts are once-through designs, based on the BWR concept, where the supercritical water from the reactor directly feeds the turbine. This concept clearly excels in system simplification and cycle efficiency, but the drawback is that the turbine and the Balance Of Plant inevitably become activated. In a BWR, soluble and suspended radioactive products remain in the reactor by the separation of water and steam, but in the SCLWR this process does not occur. In case of an accident, the radioactivity in the supercritical water can easily be transported out of the containment building.*
In this paper, a SCLWR concept with an IHX is proposed. Like the SG in a PWR, the IHX separates the primary loop from the secondary loop. The primary loop can then be completely enclosed within the reactor building. Such a concept will inevitably lead to a higher investment cost. But the advantage is that all primary activity remains within the primary loop and within the reactor building and no contamination of the turbine and BOP occurs. Moreover, this concept allows a separate chemistry for the primary and secondary loops. It also allows the use of a soluble neutron poison for reactivity control.
A conceptual design of the reactor vessel and the IHX is proposed; a RELAP model of the primary and secondary systems has been built and some design base accidents have been analyzed using the RELAP5/mod3.3 code. These analyses were performed to investigate the general behavior of a SCLWR with IHX, to have an idea of the grace time available before fuel damage occurs and to obtain some indication such as which type of safety systems would be needed for this concept. The purpose of the exercise is to determine whether the advantages of this concept with IHX sufficiently outweigh its drawbacks and consequently whether it is worth pursuing the development of this concept with IHX.

I. INTRODUCTION

The very first developments of a supercritical light water reactor (SCLWR) date back to the late 1950's, early 1960's, see historical overview[1,2]. In the late 1960's, there were also some attempts to design a steam cooled fast reactor. But the rapid and spectacular development of the LWR's (both PWR and BWR) on the one hand and the important R&D that was required to develop such a SCLWR on the other hand, stopped their further development and the idea of a SCLWR was abandoned for quite some time. But in fossil fired power plants the development continued and supercritical boilers have been in operation for over 30 years now.

In the late 1980's, the Kurchatov institute took up the idea again and proposed a concept of a small, integral type PWR with the primary loop operating at supercritical conditions[3]. But the concept of a SCLWR was really revived by Prof. Y. Oka at the University of Tokyo in the 1990's[4-9]. The important novelty in the concept of the University of Tokyo was the use of the BWR as starting point for their development. This led to important simplifications and cost savings with respect to current PWR's, while at the same time overall cycle efficiency was increased.

Following the work at the University of Tokyo, several R&D projects were launched:

1. In Canada, the CANDU-X project by AECL to study supercritical versions of the CANDU reactor;
2. In the US, some smaller projects were financed within NERI by the US DOE;
3. In Europe, the HPLWR project[10] was funded by the EC 5th framework program, funding continues in the 6th framework program.

In 2002, the Generation IV project selected the SCLWR as one of the six most promising concepts for future nuclear reactors. Since then, there is a worldwide renewed interest in SCLWR's and a large, international R&D program is launched within GenIV.

The work at the University of Tokyo clearly demonstrated the main advantages of the SCLWR concept with respect to current LWR's:

1. Increased thermal efficiency, leading to reduced fuel cost and waste disposal cost per kWh;
2. Important plant simplifications and consequently a reduced investment cost per installed kWe;
3. Possibility of both a thermal and a rapid neutron spectrum core.

The main viability issues of the SCLWR concept are the selection of suitable materials and the core design. The most difficult problem is probably the fuel rod cladding material, which will require extensive testing in material test reactors. Besides the material problem, core design is also a major viability issue. Due to the large enthalpy rise over the core, the cladding temperature is very sensitive to the hot channel factors, making core design a very difficult task.

All recently proposed SCLWR concepts are once through or direct cycle designs, based on the BWR concept, where the supercritical water from the reactor directly feeds the turbine. This concept excels in system simplification and cycle efficiency, but the drawback is that the turbine and the BOP inevitably become activated. In a BWR, soluble and suspended radioactive products remain in the reactor by the process of separation between water and steam. In a SCLWR however, all activity in the supercritical fluid is transported to the turbine. A leaking fuel rod will in this situation inevitably lead to an immediate reactor shutdown. Current PWR's on the other hand can continue to operate with several leaking fuel rods till the next planned outage, provided the activity remains below the Plant Technical Specification allowed limit.

In this paper, a SCLWR concept with an Intermediate Heat Exchanger (IHX) is proposed. Like the SG's in current PWR's, the IHX separates the primary loop from the secondary loop or power conversion system. The primary loop can than be completely enclosed within the reactor building. Such a concept will lead to a higher investment cost and a higher reactor temperature for the same cycle efficiency. But the advantage is that the whole primary activity remains within the primary loop and within the reactor building and no contamination of the turbine and BOP occurs. This concept also allows a separate chemistry for the primary and secondary loops and allows the use of a soluble neutron poison for reactivity control. Moreover, it is hoped that this concept will behave much like a current PWR during design base accidents, so that the vast experience in this field with today's PWR's can to a large extend be recuperated.

A conceptual design of the reactor vessel and the IHX is proposed, a RELAP model of the primary loop has been built and some design base accidents have been analyzed using the RELAP5/mod3.3 code. These analyses are performed to investigate the general behavior of a SCLWR with IHX, to have an idea of the grace time available before fuel damage occurs and to obtain some indication such as the type and capacity of the safety systems that would be needed for this concept.

The purpose of the whole exercise is to determine whether the advantages of this concept with IHX sufficiently outweigh its drawbacks and consequently whether it is worthwhile pursuing the development of this concept with IHX.

II. CONCEPTUAL DESIGN

II.A. Operating Conditions

The operating conditions are given in Table I and are those of the conceptual design for a direct cycle SCLWR developed by INEEL[11]. The reference design of the power conversion cycle[11] considers turbine inlet conditions of 25 MPa, 500 °C and generates 1600 MWe with a net thermal efficiency of 44,8 %. The corresponding core thermal power is 3575 MWt.

TABLE I

Operating Conditions

Primary System	
Core Power	3575 MWth
Reactor Inlet Temperature	300 °C
Reactor Outlet temperature	520 °C
Reactor Operating Pressure	28 MPa
Reactor Flow Rate	1916 kg/s
Power Conversion System	
Net Electric Power	1600 MWe
Turbine Inlet Temperature	500 °C
Turbine Inlet Pressure	25 MPa
Feedwater temperature	280 °C
Feedwater Flow	1847 kg/s

In order to preserve the performance of the power conversion cycle, the same operating conditions were kept on the shell side of the IHX. The reactor operating temperatures were consequently increased with 20 °C. The reactor operating pressure was also increased to 28 MPa. This has a favorable impact on the size of the IHX, see section II.D.

II.B. Core and Fuel Assembly Design

Recent papers on SCLWR core design seem to converge to a square lattice Fuel Assembly (FA) design with water rods[8,9,12,13,14] for a thermal spectrum reactor. The core and FA design proposed by INEEL[11] is adopted.

The core design parameters are given in Table II. The FA and fuel pin relevant dimensions are given in Table III and Table IV. The reference core is shown in Fig. 1, the FA cross section in Fig. 2.

TABLE II

Reference Core Design[11]

Number of Fuel Assemblies	145
Fuel Assembly Type	square 25x25array
Fuel Assembly Pitch	0,288 m
Core Barrel inner/outer Diameter	4,3/4,4 m
Axial/Radial/Local/Total Peaking Factor	1,4/1,4/1,2/2,35
Average/Peak Linear Power	192,6/453,0 W/cm

Part of the coolant flows downwards through the water rods to provide sufficient moderation in the core. In the bottom nozzle of the FA, this flow is mixed with the remaining fraction and the total amount of coolant flows upwards to cool the fuel pins. In the literature widely different values are proposed for the downflow fraction. In this paper, the value of 90 % proposed by INEEL[11] is adopted. Because the heating of the downward flowing water in the water rods has an important impact on its density, the value of the downflow fraction can only be optimized using coupled neutronic and thermal hydraulic calculations.

TABLE III

Reference Fuel Assembly Design[11]

Fuel Assembly Side	286 mm
Fuel Assembly Duct thickness/material	3 mm/MA956
Number of Fuel Pins/Water Rods	300/36
Fuel Pin Pitch	11,2 mm
Water Rod Side	33,6 mm
Water Rod thickness/material	1 mm/MA956
Water Rod Insulation	2 mm/Zirconia

TABLE IV

Reference Fuel Pin Design[11]

Fuel Pin Outside Diameter	10,2 mm
Cladding thickness/material	0,63 mm/MA956
Fuel Pellet Outside Diameter	8,78 mm
Fuel Column Length	14 ft/4,2672 mm
Gas Plenum Length	0,6 m
Fill Gas Pressure at ambient conditions	6 MPa

For the cladding material, INEEL[11] suggests the use of the oxide dispersion strengthened ferritic alloy Incoloy MA956. This is a Fe-Cr-Al steel, mechanically alloyed with Yttrium oxide particles. This alloy has excellent oxidation resistance and high creep strength up to temperatures as high as 1300 °C. Compared to Ni based alloys such as Inconel MA754, which shows similar strength at elevated temperatures, MA956 has an advantage in neutron economy because it is Ni free. No data are available on the behavior of MA956 under irradiation. MA956 has been assumed as cladding material and as structural material for the fuel assemblies in this concept.

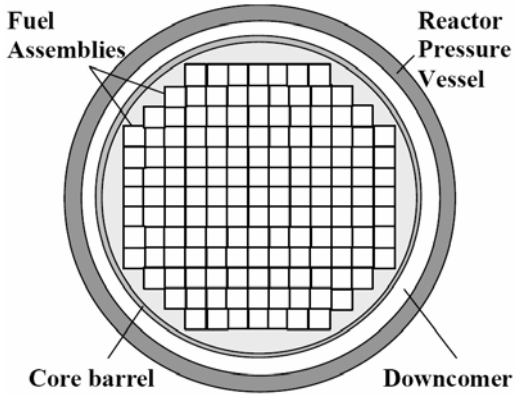

Fig. 1. Core Layout[11].

To avoid unacceptable cladding temperatures during normal operation, the coolant flow through each FA must be proportional to the FA power. This can be achieved by placing orifices at the outlet of each FA to obtain a flow distribution that matches the power distribution. This also requires that the fuel assemblies are ducted. No cross flow between FA can be allowed. This is a major difference with a traditional PWR core.

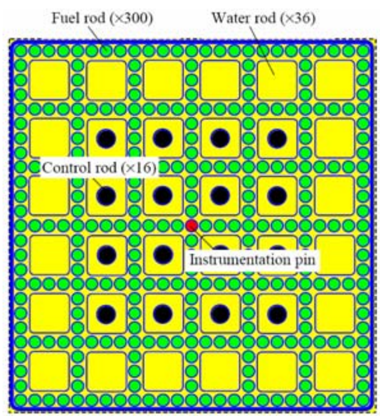

Fig. 2. Fuel Assembly Design[11].

The original FA design[11] proposed a water rod wall thickness of 0,4 mm, all metal. But the need for insulated walls was already suggested in this report. Test calculations with parallel fuel assemblies of different powers indicated that it was not possible to obtain a stable flow distribution proportional to the FA power with the thin, all metal wall for the water rods. The heat transfer through this thin wall is very high and in the higher powered fuel assemblies, the water inside the water rods can be heated to temperatures above the pseudo critical temperature while flowing downwards. This leads to large density changes inside these water rods resulting in an unstable flow behavior and flow reversal. To obtain a stable flow distribution proportional to the FA power, it is necessary to keep the temperature inside the water rods well below the pseudo critical temperature. This requires insulation of the water rod walls.

A suitable insulating material might be stabilized Zirconium oxide or Zirconia. This ceramic material combines a high strength at elevated temperatures with low heat conductivity. The use of Zirconium also has the advantage of low neutron absorption. Therefore, an insulating insert of 2 mm Zirconia is assumed inside the water rods. Also the prolongation of the water rods through the upper plenum of the reactor vessel is assumed insulated with 2 mm of Zirconia. A temperature difference of more than 200 °C develops over this thin Zirconia wall during normal operation. The resulting thermal stresses might be a problem for this thin wall. Some information on the behavior of Zirconia under irradiation is available from tests with ceramic nuclear fuels. Little is known on the long term behavior of Zirconia in an aqueous environment.

II.C. Reactor Vessel Design

Compared to a typical large size PWR vessel, the SCLWR vessel must accommodate the additional requirement that a significant part of the cold leg flow must be directed to the volume under the reactor vessel head and then must flow downwards to feed the water rods. In a typical PWR vessel layout, this would require a very complex construction at the level of the FA top nozzle and upper core plate, where the colder fluid coming from the dome and feeding the water rods must be separated from the hot fluid coming out of the FA and flowing counter currently towards the upper plenum and the hot legs.

To avoid this difficulty, an alternative reactor vessel layout is proposed in Fig. 3. The hot and cold legs are connected to the reactor vessel at different elevations, the cold leg having the higher elevation. The upper core plate has been eliminated altogether. Instead, the FA duct is prolonged with 1,8 m, bringing the total FA length to about 7,1 m. The FA is positioned between the lower and upper core support plates. The side walls of the duct prolongation are perforated so that the hot fluid from the FA can escape sideways towards the hot legs. Each water rod in Fig. 2 is prolonged with a circular tube (with 2 mm Zirconia insert) running inside the duct prolongation up to the upper core support plate.

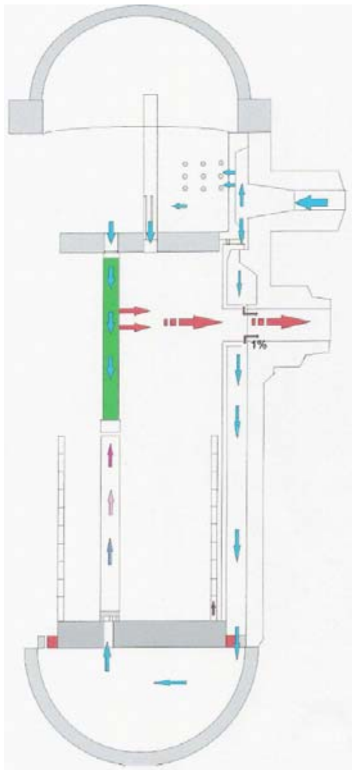

Fig. 3. Reactor Vessel.

Most of the water coming from the cold legs flows immediately to the volume under the dome through the perforated skirt of the upper internals. The upper core support plate has an opening corresponding with each FA. The water flows through these openings into the water rods and flows downwards to the FA bottom nozzle. The remaining part of the coolant flows downwards between the barrel and the vessel to the lower plenum. The lower core support plate also has an opening for each FA. Inside the FA bottom nozzle, the two fractions are mixed. The coolant then flows upwards along the fuel pins and escapes sideways out of the duct prolongations towards the hot legs.

Just as in a typical PWR vessel, a control rod guide tube is placed on top of a selected number of FA openings in the upper core support plate. A penetration in the vessel head is aligned with each control guide tube for the control rod drive shaft. The control rod clusters are largely the same as in a typical PWR vessel, only the rods are now considerably longer because of the duct prolongation. Note that the control rods are guided inside the water rods and are therefore not in contact with the high temperature water.

The flow path through the reactor is such that the pressure retaining boundary (vessel + head) remains at the cold leg temperature of 300 °C. This allows the use of current state of the art LWR vessel materials with a stainless steel cladding on the inside. The thickness will however be significantly larger because of the much higher design pressure compared to a typical PWR vessel. For those internals exposed only to the cold leg temperature, the same austenitic stainless steels used for the internals of current PWR vessels can be used. For the internals exposed to higher temperatures, the ferritic 9Cr steels like the P91 or P92 could be used. These materials are extensively used in supercritical fossil fired power plants for the high temperature components. But no data are available on the evolution of the mechanical properties of these materials under irradiation. In this model SA 508 Grade 3 Class 1 steel has been assumed as vessel material and the P91 9Cr-1Mo-V steel has been assumed for the internals.

II.D. Intermediate Heat Exchanger Design

The IHX design is inspired by the once-through Steam Generator design[15]. The primary fluid is on the tube side, the secondary fluid on the shell side.

Sizing calculations have been performed for three types of IHX: straight tubes pure counter current; baffled design with cross flow on the shell side; helically wound tube bundle. The helically wound tube bundle clearly gives the best performance for this application. According to these calculations, it seems possible to transfer the heat using only two helically wound IHX's with dimensions of the same order of magnitude as a typical once through SG. Incoloy 800HT was assumed as tube material. The main dimensions of the IHX as used in this concept are given in Table V.

TABLE V

Intermediate Heat Exchanger Main Dimensions

Tubes OD/wall thickness	12,7/1,7 mm
Number of Tubes	15475
Tube Bundle Length	24,78 m
Inner/outer Radius of Tube Bundle	0,2/2,142 m
Helical Angle	20 °
Total Heat Transfer Area	44730 m²
Tube side flow	958 kg/s
Tube side inlet/outlet temperature	520/300 °C
Shell side flow	923,5 kg/s
Shell side inlet/outlet temperature	280/500 °C

The sizing calculations were performed using the Dittus Boelter heat transfer correlation[2,16] on the tube side and the Zukauskas correlation[17] on the shell side. It is acknowledged that the Dittus Boelter correlation is not the most appropriate correlation and overestimates the heat transfer coefficient around the pseudo critical point[2,16]. As for the Zukauskas correlation, no information is available on its applicability for supercritical fluids. It uses however a form similar to the Dittus Boelter correlation and it can therefore be assumed that it suffers from the same shortcomings. To the authors' knowledge however, no other correlation is available for the heat transfer on a helical tube bundle in supercritical fluids. The two correlations mentioned were selected for the sizing calculations because they require knowledge of the fluid properties only at the bulk fluid temperature and allow therefore a direct calculation without iteration on the wall temperature. This greatly simplifies the sizing calculation. Notwithstanding the known shortcomings, it is believed that a reasonable first guess of the size of the IHX is obtained, which is judged to be sufficient for the purpose of this paper.

Because a helical tube bundle is used, the differential thermal expansion between tubes and shell poses no problem. The feedwater inlet is moved to the top of the shell and the feedwater flow is directed downwards between bundle wrapper and shell. In this way the outer shell is only exposed to the feedwater temperature, which allows a smaller shell thickness. Nevertheless, wall thickness will be much larger than current SG's due to the much higher design pressure.

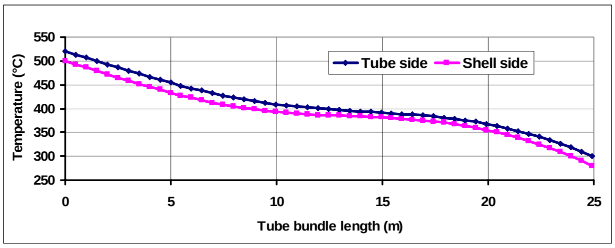

Fig. 4. Calculated Temperature Evolution in IHX].

The calculated temperature evolution on the shell and tube side is given in Fig. 4. This figure highlights a specific problem of heat exchangers with supercritical fluid where the fluid crosses the pseudo critical temperature on both sides of the tubes. It is observed that around the pseudo critical temperature, a zone of nearly constant temperature develops. This is due to the very high specific heat of the fluid around this temperature. In this region, nearly all of the transferred heat is used to overcome the high thermal capacity and the resulting temperature change is very small. This behavior is somewhat similar to a boiling system. Therefore the pseudo critical temperature must be sufficiently higher on the primary side than on the secondary side. If not, the temperature difference becomes very small in the pseudo critical region, resulting in very low heat transfer and therefore requiring a much larger heat transfer area. For the same reason, it is not possible to operate the primary system at a lower pressure than the secondary system.

The impact of the primary pressure on the required heat transfer area is shown in Fig. 5. The impact of the primary to secondary temperature difference on the heat transfer area is likewise shown in Fig. 6. The chosen primary system operating conditions are a reasonable compromise, but there is certainly room for optimizing the IHX dimensions.

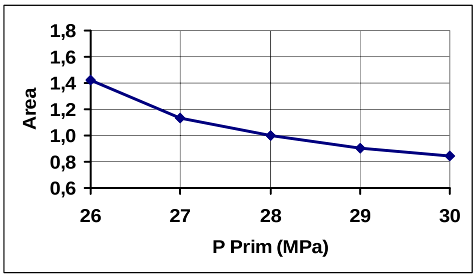

Fig. 5. Relative IHX area vs. Primary Pressure.

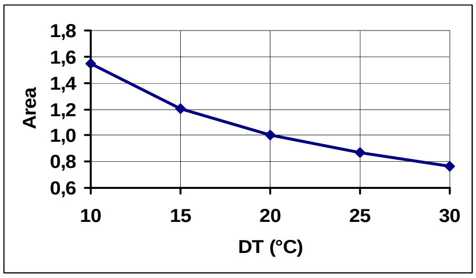

Fig. 6. Relative IHX area vs. ΔT.

Only a rough thermal-hydraulic design of the IHX has been performed, just to obtain an order of magnitude of the main dimensions. This is sufficient for the purpose of this paper. The mechanical design of the IHX has not been looked at, although it is quite clear that this will be very challenging. It is not at all clear if such an IHX can be build and perhaps a solution with three or four IHX will be necessary.

II.E. Primary Loops and Coolant Pumps

The layout of the primary loops is inspired by the AP1000[18] and the B&W plants with OTSG's[19]. Both designs use two SG's., connected to the reactor vessel by two hot legs, four circulation pumps and four cold legs.

The same layout has been adopted in this concept. There are 4 cold legs and 2 hot legs. The 9Cr steel P91 has been assumed for the pipes.

High inertia canned motor pumps are assumed, 2 pumps per IHX. The pumps are roughly half the size of the AP1000 pumps. Design temperature is about the same as in the AP1000 design, but the design pressure would be much higher.

III. RELAP MODEL

A RELAP model has been built of the reactor vessel, both primary loops, both IHX's and the feedwater and steam lines up to the isolation valves. Accumulators, an auxiliary feedwater system and a limited number of reactor shutdown signals were also added to the model. The code version used is RELAP5/mod3.3gl.

The adequacy of the RELAP5/mod3 code to calculate transients in the supercritical regime has been investigated[20]. The authors[20] conclude that, although the heat transfer correlations are not the most appropriate for supercritical conditions, the overall prediction capability of RELAP5/mod3 is sufficient to investigate the general behavior of a system with supercritical water as long as the pressure remains supercritical. But the code version used[20] failed whenever the pressure had to cross the critical pressure during depressurization transients. The code version used in this paper still suffers from the same problem. For this reason, no Loss Of Coolant Accident (LOCA) or Steam Line Break (SLB) scenarios could be calculated so far.

During the course of this project, the authors were in close contact with the code developers at ISL Inc., reporting any code problem that was encountered. This resulted in a number of modifications to the 3.3gl version to improve the code performance in the supercritical regime. However the problem of crossing the critical pressure is not yet resolved and the code developers are currently working on it.

Many quantities vary strongly along the core height e.g. coolant temperature and density, fuel and cladding temperatures. These variations are also highly non linear. To capture these variations correctly, a fine mesh in the axial direction of the core is needed. In this model, 21 nodes along the heated length of the fuel were used. The hottest FA including the hottest fuel rod has also been modeled explicitly in parallel with the averaged core. A bottom peaked power distribution with the power peaking factors as in table II is used. This power distribution is typical for a core with a higher moderation in the bottom part than in the top part.

A special problem occurred when simulating heat exchangers with small temperature differences as in Fig. 4. Like all nodal codes, the RELAP code basically uses the energy balance over the node to calculate the temperature in the node. But this means that the averaged node temperature is in reality the node outlet temperature. In a counter current heat exchanger, this results in an even smaller temperature difference between tube and shell side than in reality. The only possibility to obtain a reasonable simulation of the heat exchanger is to use a very fine meshing along the tube bundle. In this model, 60 nodes on either side of the tubes were needed to represent the tube bundle temperature profile.

The RELAP model of the IHX was first calibrated to reproduce the calculated design conditions. The vessel model was also calibrated to obtain the correct flow distribution between averaged and hot FA, the necessity of which was discussed in section II.B. Next, the model was assembled and the pumps calibrated to give the correct primary loop flow. A satisfactory steady state solution at nominal power was obtained for the complete model as starting point for the accident analysis. This steady state solution reproduces the operating conditions of Table I. Resulting hot rod maximum cladding temperature is 625 °C and occurs at the top of the fuel rod. Maximum fuel centerline temperature is 1950 °C at about 40 % elevation. Required pump head is 0,75 MPa, pump power is 630 kW.

IV. DESIGN BASE ACCIDENTS

IV.A. Loss of one Reactor Coolant Pump

The first accident analyzed is the loss of one out of four reactor coolant pumps. The results are shown in Fig. 7.

Pump trip is postulated at t = 5 sec in the transient. Reactor shutdown occurs on low pump speed, set at 90 % of nominal speed. The core power reduction is simulated using a power curve in function of time following reactor shutdown, taken from a typical PWR with comparable core power. The reactor shutdown also causes turbine trip and loss of feedwater. The primary system relief valves setpoint is 30,0 MPa, and the steam line relief valves open at 26,5 MPa. The primary system relief valves are assumed connected to the top elevation of both hot legs.

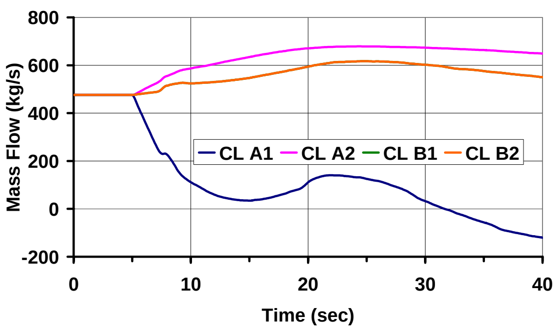

Fig. 7a. Cold Leg Flow.

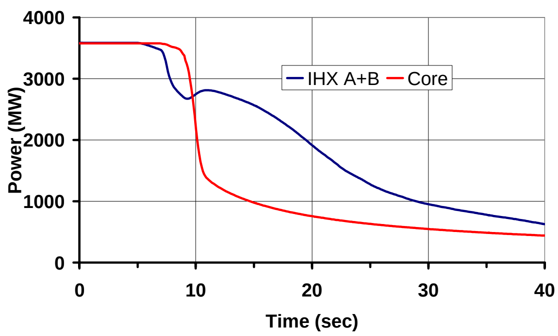

Fig. 7b. Core and IHX Power.

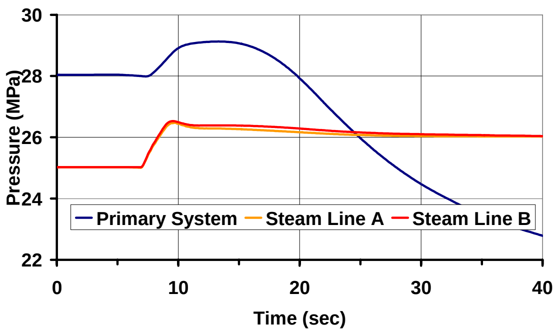

Fig. 7c. Primary and Secondary Pressure.

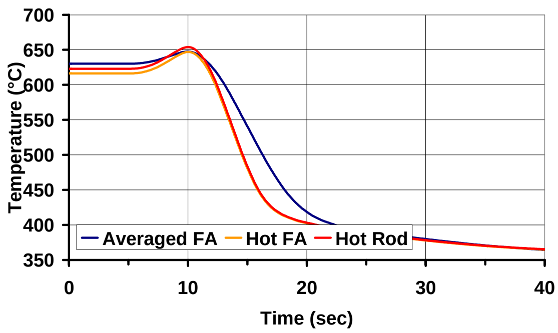

Fig. 7d. Maximum Cladding Temperature.

Following the pump trip, the mass flow in the corresponding cold leg rapidly decreases and becomes negative about 28 sec after pump trip. The mass flow in the other cold leg of the same loop increases with about 40 %. The mass flow in the other loop increases with about 30 %. After reactor shutdown, the heat evacuated by the IHX's rapidly becomes larger than the core power, assuring the cooling of the primary loop. The pressure in the primary system rises but the setpoint pressure of the pressure relief valves is not reached. The secondary pressures are controlled by the steam relief valves. The increase of the cladding temperature is rather limited. The maximum cladding temperature of the hot rod increases with only 25 °C and reaches a maximum of 650 °C.

Clearly the accidental loss of one reactor coolant pump poses no problem for the cladding temperature, provided that reactor trip occurs on low pump speed or low primary flow.

IV.B. Blackout

The second accident analyzed is the blackout or loss of non emergency electrical power. This results in the simultaneous loss of all four reactor coolant pumps, turbine trip and loss of feedwater. The results are shown in Fig. 8 for the short term and Fig. 9 for the long term.

In the short term, it is verified that the reactor shutdown signals are capable of protecting the core against too high cladding temperatures. This accident could also lead to severe overpressure in the primary system. In the long term, it must be checked that the core can be adequately cooled by natural circulation.

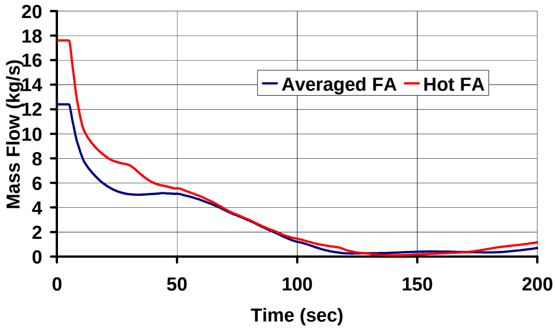

Fig. 8a. Fuel Assembly Flow

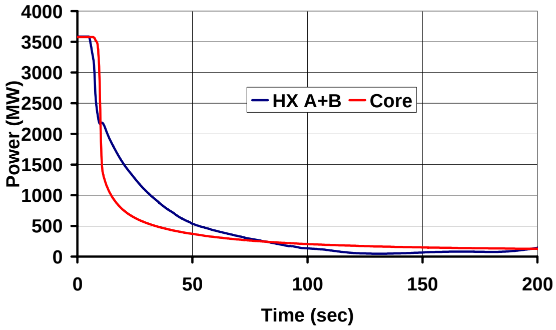

Fig. 8b. Core and IHX Power

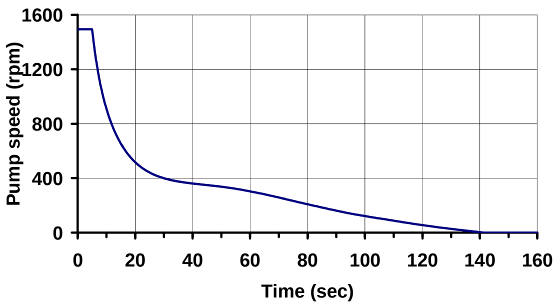

Fig. 8c. Primary Pump Coast Down

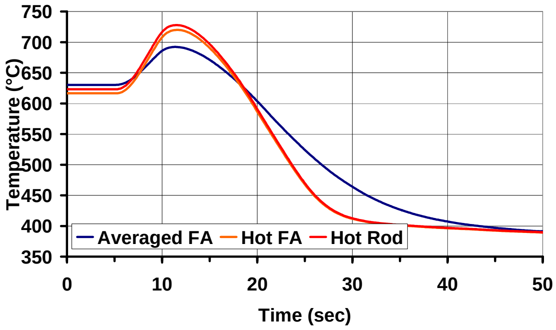

Fig. 8d. Maximum Cladding Temperature

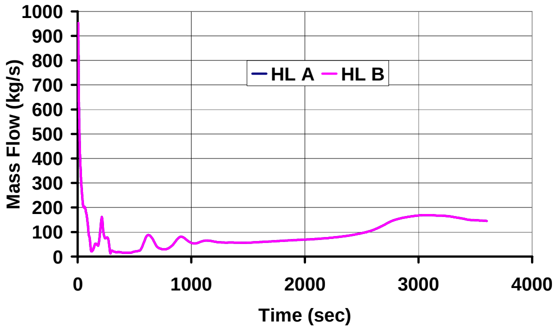

Fig. 9a. Hot Leg Flow

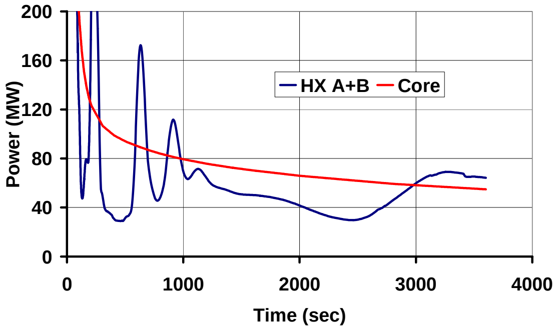

Fig. 9b. Core and IHX Power

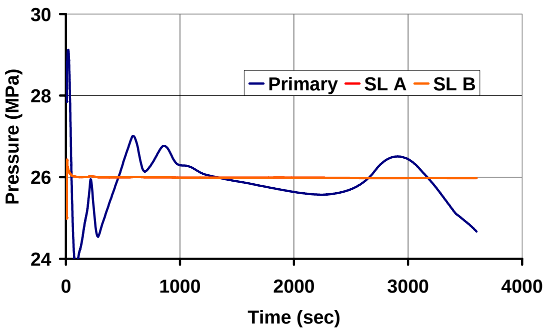

Fig. 9c. Primary and Secondary Pressure

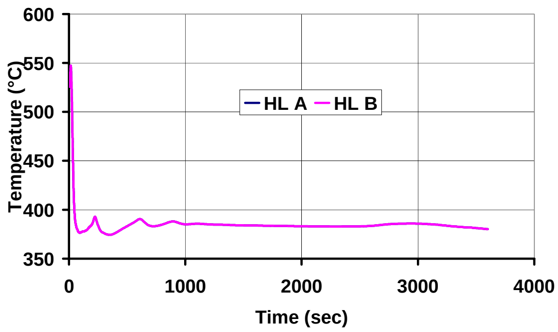

Fig. 9d. Hot Leg Fluid Temperature

The blackout is postulated at t = 5 sec in the transient and results in turbine trip, loss of feedwater and coast down of all four reactor coolant pumps. The reactor shutdown is delayed until the signal on low pump speed, set at 90 % of nominal speed, occurs. In this way, the analysis also covers the loss of all reactor coolant pumps accident. The core power reduction is simulated using a power curve in function of time following reactor shutdown, taken from a typical PWR with comparable core power. The primary system relief valves open at 30,0 MPa, and the steam line relief valves open at 26,5 MPa.

Following the pump trip, the mass flow through the FA's rapidly decreases, leading to a heat up of the system, a pressure increase and an increase of the cladding temperature. The core power is rapidly interrupted by the reactor shutdown signal on low pump speed or low primary flow. Maximum cladding temperature reached is 725 °C. At these temperatures, the ultimate tensile strength of the cladding material is still above 100 MPa (in unirradiated condition), which should be sufficient to withstand the short term loads under these conditions.

The calculated cladding temperature is however at best a first approximation of the real value. A more accurate calculation using an appropriate subchannel code an applying the required uncertainties will most probably lead to even higher values. The result shows that cladding temperature will be a very sensitive licensing parameter.

The pressure rise in the primary system remains limited. The maximum pressure reached in the primary system is 29,1 MPa and remains below the opening setpoint of the primary pressure relief valves. The large density changes following reactor shutdown on the contrary are responsible for a sharp drop of the primary pressure down to 23,5 MPa.

Following turbine trip, the secondary side pressure rapidly increases until the steam line relief valves open. For the remainder of the transient, steam pressure is controlled by the steam relief valves and remains nearly constant.

After this initial phase, the temperature in the primary system rapidly decreases, but then remains nearly constant around the pseudo critical temperature. A natural circulation flow develops in the loops. On average, the heat evacuated by the IHX's matches the core power, but large variations in heat transfer can be observed. These are again due to the large density variations around the loops following reactor shutdown. Similar to a boiling system, the primary pseudo critical temperature more or less stabilizes slightly above the secondary pseudo critical temperature. The latter is controlled by the secondary relief valves. If sufficient AFW flow is supplied to compensate for the flow through the relief valves, the long term cooling of the core is assured.

Further cool down of the primary loop will require an adapted guideline, allowing the operators to cool down the plant while avoiding boiling in either primary or secondary system.

An AFW flow of 25 kg/s to each IHX is assumed in this model. This AFW flow is injected in the feedwater line upstream of the IHX as is the usual practice in PWR's. But in this case about 1 h after the start of the transient, this cold AFW water has not yet reached the tube bundle. An optimization of the AFW flow and of the injection point will be necessary in order to be able to perform a correct cooling of the primary system.

V. CONCLUSIONS

A concept of a SCLWR with IHX's has been presented. A reactor vessel concept has been proposed that allows a large fraction of the cold leg flow to feed the water rods in downflow without too much complication. Sizing calculations have been performed for the IHX, which indicate that it could be possible to transfer the core power with only two IHX. But further optimization of the dimensions of the IHX is necessary and perhaps a solution with three or four smaller heat exchangers might be more feasible and/or economical. A RELAP model of the entire system has been built and calculations have been performed with the RELAP5/mod3.3gl version of the code. Satisfactory simulations of the steady state power operating conditions have been obtained. Unfortunately, the code version used fails when during depressurization of the system at some point the fluid conditions are passing through the critical region. Therefore, no LOCA or SLB transients could be studied so far and only the blackout scenario was investigated. Cladding temperatures remain acceptable in case of a loss of all circulation pumps. A natural circulation flow does develop in the primary loop in case of loss of all circulation pumps. The long term cooling of the system with the IHX and the AFW is not straightforward and will require some careful optimization of the AFW system.

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