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A Preliminary Design Study of a Pool-type FBR "ARES" Eliminating Intermediate Heat Transport Systems

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ABSTRACT

An innovative reactor concept "ARES" (Advances Reactor Eliminating Secondary system) is proposed to aim at reducing the construction cost of a liquid metal cooled fast breeder reactor (LMFBR). This concept is developed to show the ultimate cost down potential of LMFBR's at their commercial stage. The electrical output is 1500 MW, while the thermal output is 3900 MW. Main components of the primary cooling system are four electromagnetic pumps (EMP) and eight double-wall-tube steam generators (SG). Both of them are installed in a reactor vessel like pool type LMFBR's. An intermediate heat transport system which a previous LMFBR has is eliminated, main components of which are intermediate heat exchangers (IHX), secondary pumps and secondary piping. Further, a high reliable SG could decrease the occurrence of water leak accidents and reduce the related mitigation systems. In this study, structure concept, approach to embody a high reliable SG and accidents analyses are carried out.

Flow path configuration is mainly discussed in investigation of the structure concept. In case of a water leak accident in a SG, the fault SG must be isolated to prevent a reaction production from flowing into the core. The measure to cut both inlet and outlet coolant flow paths by siphon-break mechanism is adopted to be consistent with the decay heat removal operation.

The safety design approach of the double-wall-tube SG is investigated to limit the accident occurrence below 10^{-7} (1/ry). A tube-to-tube weld is excluded from the reference design, because the welding process is too difficult and complicated to prevent adhesion of the double-wall-tube effectively. The reliability of the tube-to-

tube-sheet was evaluated as 10^{-10} (1/hr) for an inner tube and 10^{-9} (1/hr) for an outer tube with reference to the failure experience of previous SG's. The failure must be detected within 60 to 120 minutes. Finally, a seamless U tube type of double-wall-tube SG is adopted.

Transient events due to unique structure and configuration of ARES are evaluated with plant dynamic analysis codes. Representative events are identified to reveal the thermal hydraulic characteristics of ARES, main events of which are a miss-operation of the siphon-break system, a rapid change of feed water flow rate, a station blackout and a water leak accident. The results predict that the core is sufficiently cooled without sodium boiling nor cladding failure, and that the fault SG is properly isolated to restrain the amount of solid reaction product from forming blockage in the fuel assembly.

INTRODUCTION

Fast breeder reactors (FBR's) are expected to provide a major energy source in the middle of 21st century. However, construction costs of previous FBR concepts, which account for more than 60% of electricity costs, found to be high as electric power generation plants. The major reasons why the construction of a FBR is higher than that of a light water reactor (LWR) are the use of sodium for the reactor coolant and the need of an intermediate heat transport system (IHTS), which prevents a water leak incident in a SG from directly affecting on the integrity of the reactor core. IHTS consists of a long and large-diameter piping system and a number of supplemental devices such as sodium leak detectors, pre-heaters and steel lined cells. These are the major cost-increasing factor of FRB's.

CRIEPI (Central Research Institute of Electric Power Industry) has been struggling to develop the cost reduction concept. Three measures devised to deal with the IHTS system are found. The first measure is the pool configuration of IHTS as the pool type FBR integrates the primary heat transfer system in a reactor vessel. The double-pool FBR (Kinoshita, 1991) is an innovative concept which aims to accommodate the whole IHTS by installing SG's and secondary pumps into a sodium-filled annular plenum formed between the primary and the secondary concentric vessels. The combination of the double pool concept and modularization showed that the construction cost of FBR's would be reduced to levels competitive with those of large LWR's. The second measure is the shortening the secondary piping system by integrating the secondary components. A united component of a SG and the secondary pump has been developed. The third measure is the eliminating the IHTS by introducing a high reliable SG against the sodium-water reaction accidents. Normally, a double-wall-tube SG is adapted to the previous reactor concepts (Kubota, 1991) and the reactors have the primary piping. Aiming at further reduction and simplification of the sodium system, an advanced reactor concept of the eliminating IHTS was proposed (Yoshida, 1997), where the preliminary cost has evaluated that the n-th kind of construction cost could be compatible as those of the light water reactors. A design modification and measures against a sodium-water reaction accident are described in this paper. In short word, the ARES is the pool type LMFBR of eliminating IHTS. All of the major components are installed in a single sodium vessel. The reactor concept and key features are described.

REACTOR DESCRIPTION

Figure 1 shows a schematic diagram of the ARES reactor structure. Major plant specifications are listed in Table 1. The main features of the concept are as follows.

- Reactor core and major components of the sodium cooling system are installed in a single sodium vessel.
- Siphon-break mechanism and a vertical partition walls are arranged in the main sodium flow path to isolate the reactor core in case of a sodium-water reaction accident.
- An outer plenum is divided into four sectors circumferentially to limit the propagation of reaction production. An each sector contains one EMP and two SG's. And the outer plenum is divided into hot and cold part.
- DRACS (direct reactor auxiliary cooling system) are installed in outer hot plenum.

Mixed oxide (MOX)-fueled core is used to elevate the core outlet temperature and improve the thermal efficiency. Core inlet and outlet temperatures are 395 and 550 [^{o}C] respectively. This temperature condition is determined referring to the Japanese DFBR design (Inagaki, 1998). The diameter of the core barrel is 6.25 [m].

The reactor arrangement is similar to that of a conventional pool type FBR. Intermediate heat exchangers in the pool type FBR are replaced by the double-wall-tube SG's. A vertical partition wall is installed in the reactor vessel. The reactor core and the upper core structure are installed inside the wall, while four EMP's, eight SG's and four DRACS's are installed outside. DRACS is put in above space of EMP. The size of the reactor vessel is 19 [m] in diameter and 16 [m] in height. The circumferential arrangement of the SG's and EMP's is the critical factor to determine the diameter. The vertical size is dominated by the height of the SG.

Figure 2 shows the coolant flow path under normal operation. Hot sodium coolant from the core rises up inner plenum and overflows into the outer hot plenum. Then it flows into SG inlet with siphon structure through hole on the vertical wall. After heat exchange with water/steam, it flows out to the cold plenum, which is the lower part of outer plenum. The EMP pumps it from the cold plenum, and discharges it into the core inlet plenum through siphon piping in the EMP shell.

Table 1 Plant main specifications

Items	Specifications
Thermal output	3636 MW
Electrical output	1500 MW
Core and fuel	Mixed oxide (MOX)
Core inlet/outlet temperature	395/550 °C
Main sodium pump	Immersed electromagnetic pump, 4 units
Steam generator	Double-wall-tube, reversed U shape, 8 units
Decay heat removal system	Direct reactor auxiliary cooling system, 4 units
Fuel storage	In-vessel storage
Refueling	Flexible-arm refueling machine And cask-car transporting machine
Seismic protection	3D seismic isolation

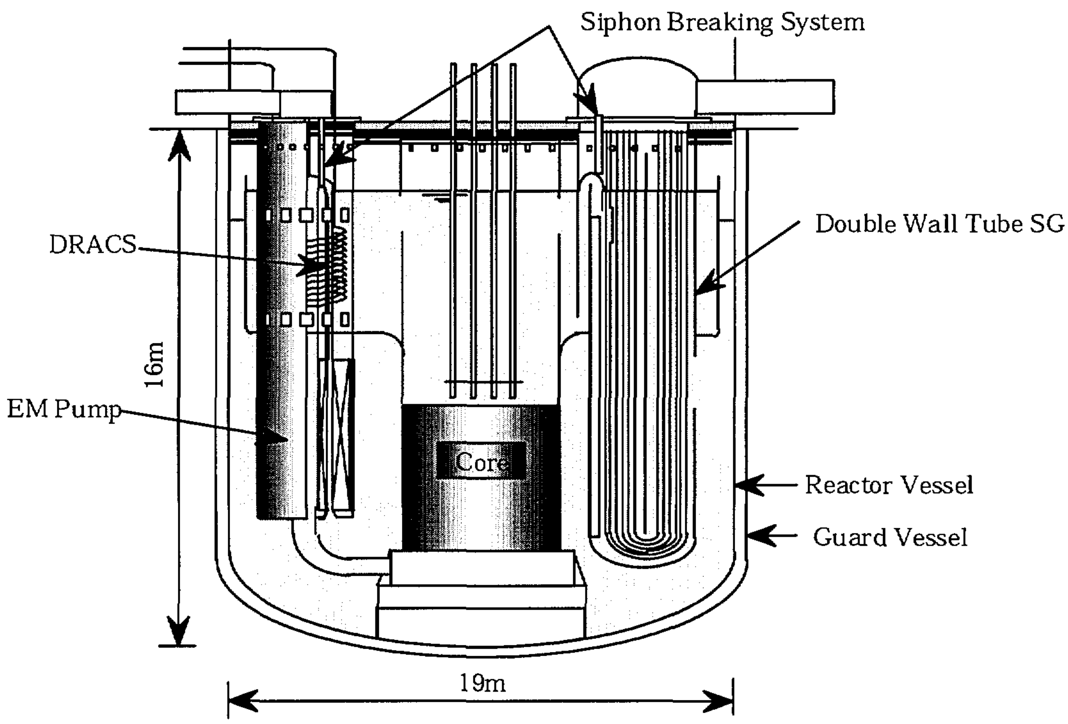

Figure 1 Schematic Diagram of ARES

Under decay heat removal operation, cooled sodium at DRACS coil flows down the bottom of outer hot plenum. SG sucks it due to the circulation head of the EMP under low flow operation or the natural circulation head at the inner plenum as the chimney effect. The coolant flow path in sodium-water reaction accident will be described in the further section.

An introduction of a flexible-arm refueling machine enables to eliminate a rotating plug as costly equipment and to simplify the roof-deck structure.

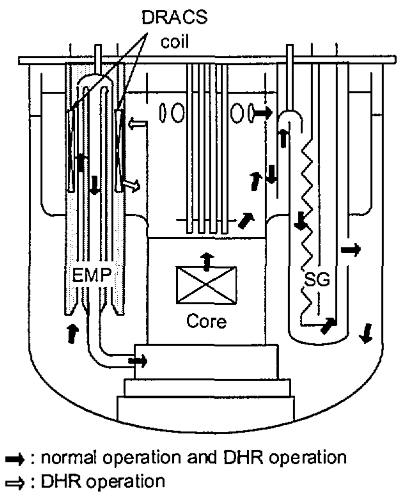

Figure 2 Coolant flow path

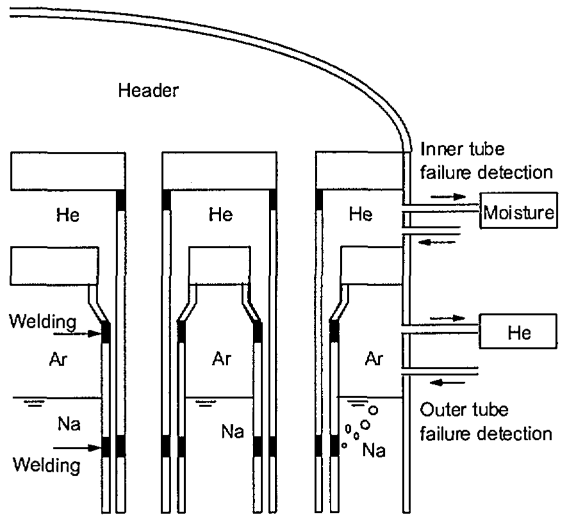

Figure 3 Schema of double-wall-tube sturucture

RELIABILITY OF DOUBLE-WALL-TUBE SG

A double-wall-tube has been proposed to exclude sodium-water reaction accident from design basis events because the sodium coolant and the water/steam are separated each other by the double boundaries, which are rarely violated by a single failure. Design target is set to below 10^{-7} (1/ry) to exclude sodium-water reaction accident from design basis events. As the present design of ARES has eight SG's, the reliability for single SG is set to 10^{-8} (1/unit-year).

Design approach is considered with two ways as follows.

- Quantitative approach: probabilistic analysis
- Qualitative approach: measure against common cause failures

Quantitative design approach

In the quantitative approach, a water leakage occurrence is estimated based on the tube failure rate and the failure detection time. Basically, a water leakage probability P_{leak} [-] is estimated as below except common cause failures.

$$\frac{P_{leak}}{T} = C \cdot p_{in} \cdot p_{out} \cdot \Delta t_d \, , \tag{1}$$

where C is the number of combination of failure tubes defined by leakage paths, p_{in} and p_{out} are the failure rate of inner tubes and outer tubes respectively and Δt_d is the detection time including the duration time to mitigating action. T is the normalization time; $T = 1[\mathrm{yr}] = 8760[\mathrm{hr}]$ to evaluate annual occurrence of the left side of eq. (1).

Figure 3 shows the schema of double-wall-tube structure and the tube failure detection systems. There are two tube-sheets. One is a water/steam tube-sheet to which the inner tubes are welded; the other is a sodium tube-sheet to which the outer tubes are welded. Helium gas flows in an intermediate region between the tubes. In this design, an inner tube failure is detected by moisture detection in helium gas. An outer tube failure is detected by gas analysis of leakage helium in the cover gas of argon.

In case that there are N_{in} inner tubes and N_{out} outer tubes, and each inner or outer tube has n_{in} or n_{out} welding, simply, C is expressed

$$C = N_{in} \cdot n_{in} \cdot N_{out} \cdot n_{out} \,. \tag{2}$$

However, the leakage path composed by failure combination of different tube-to-tube welding, which includes intermediate space between tube-sheets, would be neglected, because flow resistance in a gap between tubes is large enough to detect inner tube failure faster than water/steam leakage through the outer tube defect.

Table 2 Summary of causes and parts of tube failures

		cause of failure						total
		SCC	initial defect	poor welding	fatigue	material defect	vibration	
part of failure	tube-to-tube sheet welding	33	13(13)	11		1		58(13)
	tube-to-tube welding				6(6)			6(6)
	basic tube material	1				1	1	3
	tube support welding	2		1			1	4
	other welding		17					17
total		36	30(13)	12	6(6)	2	2	88(19)

() the number of failures not to be excluded after modification

The operating experience data have been surveyed to evaluate the occurrence of tube failures based on open papers. Eighty-eight failures are found among seven reactors and five SG test facilities. The reactors are KNK-1, Phenix, PFR, BN-350, BN-600, EBR-II and Enrico Fermi. The five facilities are Hengero 50MW SG, PNC 1MW SG, ALCO/BLH SG, GE 2MW SG and GVE 45MW SG. The cause of failures and part of failures are summarized in table 2. Cause of SCC (stress corrosion cracking) is mainly due to the austenitic steel. After the replacement to the ferritic steel, no failure has occurred. Eleven failures due to poor welding at tube-to-tube-sheet are caused by insufficient treatment in re-welding. Seventeen failures due to initial defect at other welding are occurred in bayonet type SG, which is so different type SG not to be taken in account in this study.

Nineteen failures are left through a screening of failures in this way and are expressed in Table 2 with parentheses. Six failures in western reactors due to thermal fatigue at tube-to-tube welding are caused by the DNB instability, which can be excluded in tube-to-tube-sheet welding. The failures are divided into two groups of nineteen and thirteen. Furthermore, thirteen failures consist of three western facilities and ten of BN-600. Operating time of all reactors and facilities is 3.7×10^{9} $[\text{tube}\cdot\text{hr}]$ and that of western reactors and facilities is 3.9×10^{8} $[\text{tube}\cdot\text{hr}]$. Table 3 shows the failure rates categorized in consideration of thermal fatigue and of including BN-350 and BN-600 or not. Failure rate to assess the reliability of double-wall-tube SG in this paper can be estimated to be up to 1×10^{-9}[1/hr], because most of the failures have occurred at the early stage of operation and there has been little failure after modifications. As the latest SG using modified material and welded by high reliable method has experienced no failure, one order higher reliability would be expected.

Table 3 Failure rate of welding

failures	operating time	failure rate [1/hr]	failure rate [1/yr]
13	3.7×10^{9} $[\text{tube}\cdot\text{hr}]$	3.5×10^{-9}	3.1×10^{-5}
13+6		5.1×10^{-9}	4.5×10^{-5}
3	3.9×10^{8} $[\text{tube}\cdot\text{hr}]$	7.7×10^{-9}	6.7×10^{-5}
3+6		2.3×10^{-8}	2.0×10^{-4}

<u>Quantitative design approach</u>

In the qualitative approach, dependent failures and common mode failures are investigated. A dependent failure could be occurred as one tube failure causes the other tube failure; for instance, an inner tube failure brings the pressure rise that acts the adjacent outer tube and might result in the both tube failures if the outer tube has not enough strength. A common mode failure is such a failure that both inner and outer tube become failure at the same time by a single cause. If inner and outer tubes metallurgically adhere each other, a crack can develop to penetrate the double-wall. High cycle fatigue brought about by flow vibration might break the double-wall-tube. In many cases, careful numerical analysis and mockup test could exclude abnormal flow vibration.

As the inner and the outer tubes have to be fabricated tightly to get good heat transfer performance, it is difficult to produce long tube separately by welding. Both side welding of butt joint performs tube-to-tube welding of elemental double-wall-tube. There is serious problem that reinforcement of weld at the inner side of outer tube is apt to merge into the adjacent inner tube surface.

Figure 4 shows the ways to weld the double-wall-tube. A type (a) of butt joint by TIG welding has the above problem and requires a sleeve part (a faint meshed part in figure). Though EBW (electron beam welding) cannot need sleeve, the adhesion problem still remains. A type (b) of fillet joint also requires a sleeve part and it makes the tubes fabrication complex that the tube diameter becomes wider. A type (c) is the way of transient liquid phase bonding with an amorphous material, of which a standard has not prepared. For reasons of above consideration, tube-to-tube welding structure has been rejected in SG design of ARES.

There are a butt joint and a fillet joint for the tube-to-tube-sheet welding (Figure 5). Through the survey of failure data, all failure cases at tube-to-tube-sheet were found to be in the type of fillet weld, whereas existing SG's with butt joint of welding have performed well. Butt joint structure is adopted in SG design in ARES in consideration of the reliability, as the manufacturing cost is rather high.

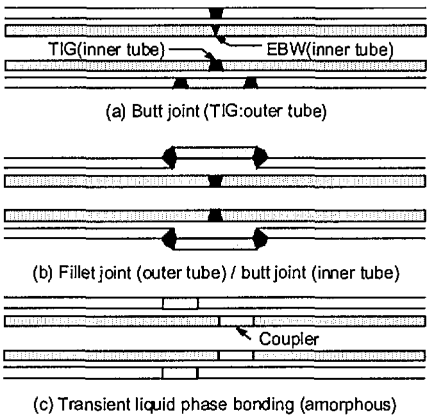

Figure 4 Tube-to-tube welding

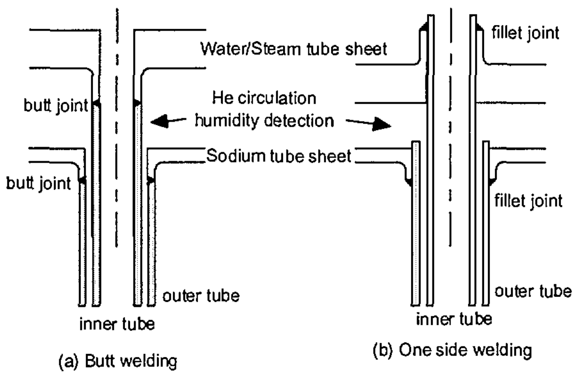

Figure 5 Tube-to-tube-sheet welding

Occurrence of water leak accidents

As the maximum length of a double-wall-tube unit fabricated at the present factory sop is 30 [m], the effective heat transfer length in U-shape tube is about 17 [m]. A single SG requires 2280 tubes to gain the heat exchange capacity of 455 [MWth]. Each tube has two tube-to-tube-sheet welding at the both ends. The failures combination of one side inner tube and the other side outer tube could be neglected. So, the number of combination C in eq. (2) is expressed as follows,

$$C = 2 \cdot N_{in} \cdot N_{out} . \quad (3)$$

Assuming $p_{in} = p_{out} = 1\times10^{-9}$ [1/hr], the detection time Δt_d is required below about 13 [min.] from eq. (1). The inner tube can be welded with a welding torch inside a tube. As this welding method is mature one, a higher reliability for inner tube welding would be expected. The required detection time is 44[min.] for $3x10^{-10}$ [1/hr] and 132[min.] for$1x10^{-10}$ [1/hr]

SODIUM-WATER REACTION

In this paper, the sodium-water reaction accident is assumed to be a beyond design basis event with the double-wall-tube SG's, after the FBR has been commercialized. The reason is that reliable fabrication and manufacturing have carried out, and successful operation experience has been accumulated by that time. This accident, however, has to be evaluated in consideration of plant risk management.

The basic policy in this study is to minimize the countermeasures against sodium-water reaction. Safety related issues in this event are the structural integrity challenged by the pressure rise due to the reaction and the flow blockage in the core region and the chemical corrosion by reaction products. Leakage detection systems, which consist of tube failure detection systems and reaction detection systems, are required.

With regard to the pressure rise, some mitigation systems, which are a quick steam/water blow system and a pressure release system, could be excluded because the ARES configuration has the large cover gas volume.

With regard to reaction products, a fault SG' should be isolated to prevent the reaction production flowing into the core. In case of the detection of tube failure (either the inner tube or the outer tube), the reactor is immediately shutdown and the two siphon-break lines installed in an EMP outlet and a SG inlet are actuated. Feed water pumps are also tripped and steam/water blow down lines are opened. The actuated siphon-break lines are in an only one section out of four divided sections to which the fault SG belongs. The amount of inflow of reaction products to the reactor core is analyzed and evaluated.

A flow network code named SPROUT is used to analyze the sodium-water reaction products behavior, in which the liquid phase sodium and the gas phase hydrogen of reaction product are modeled by the drift flux and the solid phase of reaction products flow with the liquid phase assuming a correlation. The solid reaction products perform dissolution and separation in the coolant, adhesion and deposit on the structures. After main pumps trip, the gas injection to make the siphon-break does not work until pump head becomes low enough. This duration time is the dead time for the siphon-break action.

The water/steam leakage condition is assumed that a single tube breaks near SG outlet of which a fluid is the single phase of water and a breaking structure is a double-ended-guillotine cut. Two water leak rates are assumed; 9 [kg/s] is for the nominal design and 3 [kg/s] is for the optional design with a orifice at an inlet tube-sheet against flow instability. A dead time for siphon-break initiation is strongly related a flow halving time of main pump and a gas injection pressure. The reference design adopts flow halving time of 5.5 [sec.]. Three values of 5, 10 and 20 [sec.] are assumed.

Table 4 Evaluated amount of reaction products

Water leak rate	9 kg/s			3 kg/s		
Dead time	5 sec.	10 sec.	20 sec.	5 sec.	10 sec.	20 sec.
Isolated area	395	500	660	140	175	225
The other region	26	35	65	0.6	4.0	18.5

Maximum solubility in other region: 49 kg

The results of parametric survey analyses are summarized in Table 4. Coolant region is divided into two regions. One is a quarter sector including the fault SG, the other is the rest three quarter sectors and the inside plenum including the core region. The maximum solubility of the other region for the reaction products is 49 [kg]. In the worst case, insoluble reaction product amounts to 16 [kg]. If all the insoluble reaction products flow into the core region, total inlet blockage would hardly occurred due to the number of 523 sub-assemblies. As corrosion rate of NaOH to austenitic steel is varied from 0.5 to 1.3 [mm/yr.] at 400 [^{o}C] and the saturation concentration, the structural integrity is secured.

REACTOR DYNAMICS

Basically, the reactor configuration of ARES is very similar to that of the pool type FBR. Different configurations are SG's are directly installed of IHX's and a unique flow path with the siphon-break mechanism. Due to these unique configurations, representative events are identified, which are a miss-operation of the siphon-break system, a rapid change of feed water flow rate, a station blackout and a water leak accident. These transient events are analyzed by FBR plant dynamic analysis code CERES and the flow network code (Nishi, 1999).

A steady state of full power operation and a manual trip event are analyzed to understand the behavior in a trip sequence and to set up the thermal transient condition for the analyses of the structural integrity. There is neither stagnant region nor thermal stratification in the steady state. Varied design parameters are surveyed in the reactor trip and the continuous sequence to define the core flow rate in the scheduled decay heat removal operation. Even a 5% core flow rate can maintain the stable heat removal with DRACS.

The miss-operation of the siphon-break system causes the main pump trip concerned quarter sector, which results in the similar event of the main pump stick in previous FBR. As mentioned in sodium-water accident, the siphon-break begins after decreasing the head of the main pump. For the conservative assumption, the siphon -break cuts the flow path suddenly and makes the core flow rate gradually decrease to 75% of normal operation in 1.2 [sec.] in safety analysis. The result shows that the temperature rise estimated for the hottest fuel pin is around 50 [^{o}C], which satisfies the safety criteria.

The rapid change of feed water event causes the reactor power change due to the negative temperature reactivity coefficient. The feed water increase makes the core inlet temperature decrease which brings the core power increase. Comparing to the previous FBR's, the time constant to change the core inlet temperature of ARES is shorter, because ARES does not have IHTS. The transient sequence is assumed that the feed water pumps run higher up to the limit of the emergency level when the feed water flow rate is 130% of normal operation in 30 [sec.]. The analytical result predict that the time constant of core inlet temperature rise is one-third of the value of the Japanese prototype reactor MONJU, and that the reactor power raised by about 9% which does not reach the scram level.

The station blackout transient, which is defined as the loss of the external AC power and the on-site AC power, is selected to evaluate natural convection capability of ARES. After the reactor shutdown, for a time the core flow rate decreases to below 1% of normal operation, it recovers to 1.8% with progression of the natural circulation head. The nominal hottest coolant temperature which does not include the engineering safety factor rises to 760 [^{o}C], and the temperature of coolant boundary rises to 620 [^{o}C] below the criteria of 650 [^{o}C].

It is confirmed through the reactor dynamics calculation that the core and the structural integrity are properly secured.

CONCLUDING REMARKS

A new innovative FBR concept, the ARES (1500MWe), has been proposed to show the economical advantage of FBR's in their commercial stage. The intermediate heat transfer system is eliminated, and the reactor core and the major components of main sodium coolant system are installed in a single sodium vessel. Consequently, the main sodium piping system is eliminated and the sodium handling area is minimized.

In order to classify water leak accidents into a beyond design basis event, a high reliable double-wall-tube SG is introduced, which has no tube-to-tube welding causing measure tube failures. The reliability of the tube-to-tube-sheet was evaluated as 10^{-10} (1/hr) for an inner tube and 10^{-9} (1/hr) for an outer tube with reference to the failure experience of previous SG's. The failure must be detected within 60 to 120 minutes. Finally, a seamless U tube type of double-wall-tube SG is adopted.

The representative events for safety analyses are identified in consideration of the unique configuration of ARES. The main events are a miss-operation of the siphon-break system, a rapid change of feed water flow rate, a station blackout and a water leak accident. The plant dynamic analyses are conducted to reveal the thermal hydraulic characteristics and to confirm the core and structure integrity.

The effect of a water leak accident is evaluated to validate the siphon-break system. With conservative design parameters, the result shows that the amount of solid reaction products would be low enough to make no total flow blockage in a sub-assembly.

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