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OPERATING TEMPERATURES FOR AN LMFBR

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1 INTRODUCTION

The case for the Fast Breeder Reactor (FBR) stems from its ability to extract sixty times more energy from a given mass of uranium than in thermal reactor, thereby transforming uranium from a minor to a major world fuel resource. This has led to the FBR being regarded as the ultimate goal of fission reactor development. Further deployment of FBR is essential to establish a closed fuel cycle. However, despite its technical maturity and undisputable strategic advantages, the FBR will only be commercially used when its electricity generating cost becomes competitive with the established generating systems like fossil power stations and PWRs. This requires the set of design features which involve considerable reduction in size and weight of components, compared to the previous FBR designs. In addition, it is also important to introduce some high technology, like high thermal efficiency, high fuel burn up. Considering the high boiling point of sodium (about 1200 K), recent improvements in the material properties and maturity achieved in the high temperature design & analysis methodology, a FBR system has the potential for improvement towards cost effectiveness, by generating steam at high temperature and pressure. By this, it is possible to adopt the standard turbines which are commonly used in the modern fossil power plants, operating with the steam condition at about 810 K and 17 MPa pressure. In addition to direct cost benifits due to high thermal efficiency, the effects on the environment because of low radioactive waste generation and less thermal pollution are also significant.

In this paper the possibilities of higher operating temperatures are investigated for the sodium cooled pool type FBR. A complete study of this kind needs very thorough investigations on the high temperature materials, clad hot spot factors, design code aspects, structural analysis and operating experiences. But the scope of the present paper is limited to structural mechanics aspects that are associated with this technology. However, for the purpose of comprehensive presentation, all the other related issues are also highlighted. For this study, a Prototype Fast Breeder Reactor (PFBR) with 500 MWe capacity is taken as the reference design. Accordingly, some critical high temperature components of PFBR are analysed in detail for elastic, inelastic and viscoplastic behaviour towards life prediction as per the requirement of design codes (RCC-MR 87) which form basis for justifying the possibility of higher operating temperatures for LMFBRs.

2 OVERALL DESIGN FEATURES OF PFBR

The overall flow diagram of PFBR is shown schematically in Fig.1. Heat generated in the core is transported by primary coolant circuit to an Intermediate Heat Exchanger(IHX) in which the heat is transferred to a secondary Na circuit which contains the steam generators. Steam produced in the SG is supplied to the turbine through steam-water system which is of conventional design. As shown in Fig.1, the whole of the primary circuit which comprises core, circulating pumps and IHXs is all contained in a single vessel, called main vessel (MV). Secondary sodium leaving the IHXs is carried by high temperature piping to the SGs and is returned through low temperature piping which contains the secondary sodium circulating pumps and a surge tank.

The MV, which is about 14 m dia. contains internal structures for supporting core as well as other reactor components. In order to avoid a complete loss of coolant, even in the unlikely event of Na leak from the MV, it is surrounded by a safety vessel. Under normal operating conditions, sodium at the temperature of 653 K is drawn from the cold pool by the primary pumps and is discharged through pipes into grid plate which supports the core subassemblies as well as distributes flow through them. The high temperature sodium (815 K), leaving the core, impinges on an above core structure, called 'control plug (CP)' which destroys the sodium velocity head and directs the flow into the high temperature pool. CP also contains shroud tubes to protect the CRDMs from flow buffeting.

Both hot and cold pools have a free sodium surface blanketed by argon and the flow of sodium through 8 IHXs is driven by a level difference (1 m of Na head) between the hot and cold pool free surfaces. The hot and cold temperature pools are separated from each other by a self standing thin shell structure, called Inner Vessel (IV) which is supported on core support structure (CSS).

It is worth to mention here about the primary sodium outlet/inlet temperatures for the various FBRs in the world: 833 K/658 K for PHENIX, 833 K/673 K for PFR, 818 K/668 K for SPX1, 823 K/653 K for BN-600, 802 K/670 K for MONJU and 818 K/668 K for EFR.

3 HIGH TEMPERATURE DESIGN PROBLEMS OF CRITICAL COMPONENTS

The components which operate at high temperature are fuel and blanket subassemblies, control plug, CRDMs, IV, IHXs, Sec. Na hot leg piping, SG, steam pipes and turbines. Out of which, steam pipes and turbine are designed for high temperature and high pressure steam conditions (810 K & 17 MPa) in the modern fossile power plants and hence, are not limitations. Design of fuel and blanket subassemblies, particularly, cladding, under high temperature for longer life in the core is very important for the development of high burn up fuel which in turn reduces the fuel cycle cost. This is governed by selection of advanced materials, rather than the design optimisation and analysis methodology, etc. Further core subassemblies are replaceable in the course of reactor operation. Then, the overall layout of piping is designed to minimise thermal expansion stresses in the secondary circuit but these are nevertheless significant. Thermal transients lead to the need for relatively thin walled piping to minimise the thermal transient stresses but which is adequate to withstand the internal pressure (<1 MPa). However, the thermal expansion stresses concentrate at bends and lead to potential problems of creep buckling. The CRDMs are not critical in the temperature range of the present discussion (750-900 K), due to the presence of low level primary and secondary stresses and further these components are replaceable also.

Finally, the principal high temperature components in LMFBR in general, and PFBR in particular, which dictate the choice of operating temperature are IV, CP, IHX and SG. Predominant failure modes in these components are due to creep, fatigue and buckling. Though at high temperatures, the strength reduction may pose risks of buckling problems, in the temperature range of the present study, it is not of very much critical issue in limiting the operating temperature. Thus creep and fatigue are considered to be controlling parameter for estimating the operating temperatures. It is to be noted that the stresses developed in the above said component structures are mainly due to self weight, pressure, seismic, steady state and transient thermal loadings. Though seismic events may alter the residual stress distributions and enhance inelastic strain accumulation at structural discontinuities, the associated creep-fatigue damages are negligible. Hence for each of these critical components, the thermo-mechanical loadings which are important from the creep-fatigue damage point of view are detailed in the following paragraphs.

3.1 Inner Vessel (IV)

Inner Vessel contributes to the overall hydraulics of the reactor assembly, bears the IHX pressure drop, and is subjected to the temperature difference between the hot and cold plenum. A single wall concept (Fig. 2) consisting of a lower cylindrical shell of 8.03 m diameter, an upper cylindrical shell of 13.3 m diameter and a conical shell (also called redan) as a connecting structure, has been considered for the design. It is also provided with twelve stand pipes for the passages of eight IHXs and four primary pumps. The whole structure is made up of stainless steel type 316LN.

The inner vessel is subjected to both mechanical and thermal loads during normal and transient conditions of the reactor. The mechanical loads include the self weight of structures (83 t) and IHX pressure drop (1 m of sodium head). The hot and cold pool temperatures are 813 K and 653 K respectively during nominal steady state conditions. Under this condition, the inner surface of the IV sees a maximum temperature of 798 K with associated outer surface temperature at 708 K, resulting in 90 K temperature gradient through the thickness. During cold transient following a reactor scram, the temperature gradient through thickness is reversed, yielding a value of -60 K with corresponding inner and outer surface temperatures of 640 K & 700 K respectively, at a critical instant during the transient. According to design specifications of the inner vessel, the component encounters 1000 cold transients during its life time of 30 years with a total hold time of $2x10^5$ h (for 75% load factor). These also envelope all the other possible less severe thermal transients following load variations, decay heat removal conditions etc.

3.2 Control Plug (CP)

The control plug is illustrated diagrammatically in Fig. 3. The bottom portion of CP which is immersed in hot sodium pool, mainly consists of a circular cylindrical shell of about 2.25 m dia with a bottom plate called core cover plate (CCP). This shell is divided into 3 regions by means of 2 intermediate supporting plates viz. lower and upper stay plates (LSP & USP resp.). Thermal shields are provided at the bottom of core cover plate to protect it against thermal shocks. The shell is perforated in the upper and intermediate regions, in an optimum way to achieve minimum possible temperature gradients in the plate-shell junctions during cold shock following a reactor scram. CP is manufactured from SS 316LN.

Because of its proximity to the core, CP is subjected to the steady state and transient temperature distributions of sodium discharged from the core and is therefore in the most severe thermal environment of the reactor. Applied loads from pressure and dead weights are very small so that virtually all stresses are thermally induced. During steady state conditions, temperature gradients in the parts of CP are found to be negligible [Anil Lal, S. 1991]. But reactor scram causes severe transient, through thickness thermal stresses which produces yielding at the plate-shell junctions. During a reactor start up after shut down, these thermal stresses also produce reverse plastic strain, inducing residual stresses at these junctions, which inturn may relax during period of subsequent normal operation.

3.3 Intermediate Heat Exchanger (IHX)

The schematic sketch of IHX is shown in Fig.4. IHX in the PFBR design is a straight tube, cross flow, vertical, sodium to sodium heat exchanger in which the primary sodium flows in shell side and the secondary sodium flows in the tubes. The inlet/outlet temperatures of the primary and secondary sodium are 806 K/653 K and 623 K/778 K respectively, thus the top tube sheet and its connected shell is subjected to temperature where the creep is significant. The material of construction is stainless steel type 316LN. The dominant stresses in the top tube sheet arises from the temperature difference between the hot primary and secondary sodium streams and also from the high pressure of the secondary relative to the primary sodium (about 0.85 MPa). Further under reactor trip conditions, the rate of change of sodium temperature is such that transient thermal stresses are significant to cause severe creep-fatigue damage in the top tube sheet, particularly at the junction with the outer shell.

3.4 Steam Generators (SGs)

For PFBR, modular type concept is used for SGs. In the whole secondary sodium circuit, there are totally 12 modules (3 modules/loop) with each module comprising of one evaporator, one superheater and one reheater. The individual SG units (Fig.5) are once through, vertical, straight tube type geometry with an expansion bend on each tube in sodium path. Materials of construction is 2.25 Cr-1 Mo for evaporator and 9 Cr-1 Mo (normalised & tempered) for superheater and reheater. In addition to above normal loadings, these components are subjected to significant thermal transients due to failure of secondary sodium and feed water pumps, total power failure etc. The top tube sheets are subjected to high temperature wherein creep effects are significant.

4. PRINICIPAL FACTORS LIMITING THE OPERATING TEMPERATURES

It is well known that, from the thermal efficiency point of view, higher primary sodium temperature is preferred (a 20 K increase in steam temperature results in about 0.5 % increase in thermal efficiency). Further higher ΔT across the core is advantageous in terms of lesser coolant flow rate and/or heat transfer area. While the maximum temperature is limited by the strength of materials at high temperatures, ΔT across the core is restricted by the associated thermal stresses & strains, developed in the components. The possibility of higher primary sodium outlet temperature along with the higher ΔT depends strongly on the improvements in the material properties, maturity of the design criteria and the advances in the structural analysis capability.

Use of stainless steel type 316LN for the hot pool components including hot leg sodium piping is very much justified because of operating experience (SPX-1 material), availability of complete data (codified in RCC-MR appendix Z) , acceptability to operate at about 880 K for long duration, etc. As far as ferritic steels for SGs are concerned, 9Cr-1Mo is considered to be an attractive material. A lot of developmental works are on, towards a) applicability of current material models, b) identification of failure criteria and c) validation of inelastic analysis methodology. It is expected that this material will be codified in ASME-N47 soon (Dhalla,A.K, 1991). Operation of this material at about 810 K in SG for a required life of 30 years is now possible. Though new materials like, nickel base super alloys are more suited for very high temperature applications, they are quite expensive. Development of a new excellant material for core components is continuing research towards achieving high burnup. Core material can be changed in the course of reactor operation depending upon the experience and hence it is considered here as not very critical isssue for determining maximum operating temperatures.

Regarding the design codes for LMFBR application, ASME-N47 1992 and RCC-MR 87 are the latest editions available. Out of these, RCC-MR is considered to be the most suited code for LMFBR applications. The recent edition of ASME-N47 1992, major changes are introduced in creep fatigue damage assessment which are now in line with RCC-MR 87. Further the creep damage is dominant failure mechanism for the most of LMFBR structures for which RCC-MR code rules which take into account the cyclic hardening behaviour of SS 316LN are more appropriate. SS 316 material which is codified in ASME-N47 is different from 316LN, codied in RCC-MR.

Other aspects like effects of Na, weldments, are not very critical since, effects of Na on creep rupture propetries of 316LN is not significant and it is possible to avoid welding in the most critical locations of the high temperature components. Finally, it is worth to note that the capability to analyse the FBR components by using sophisticated material models, even with the use of viscoplastic theory, has increased dramatically in the recent years.

With all the above arguements, it can be stated that an LMFMR can operate with the primary sodium outlet temperature of about 870 K. However, considering the lack of experience in operation with this temperature, it is better to raise the operating temperature in stages. In the reference (Shigehiru, AN. 1991), 3 stages are proposed to achieve elevated temperature as: 803 K to 823 K in stage 1 , more than 823 K in stage 2 and 858 K in the final stage so as to produce steam at about 810 K.

5. STRUCTURAL ANALYSIS RESULTS

It is clear that the realistic life assessment of the above said high temperature components requires sophisticated inelastic analysis and a detailed understanding of the material response under all relevant conditions. Inelastic analysis is very costly and time consuming and hence the general approach is to use elastic calculations in the design stage for which criteria exists in the design codes, which will be followed by a more comprehensive inelastic analysis