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Reactor power setback: A procedure to reduce thermal shock on FBR components

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ABSTRACT

Reactor power setback is a procedure to avoid reactor SCRAM for events originating from balance of plant, which do not affect the operation of boiler feed pumps that supply coolant to steam generators. This procedure is envisaged to avoid reactor components from being subjected to thermal shock due to SCRAM for some of the events which do not affect nuclear safety. In this procedure, all control rods of the reactor are driven down simultaneously to achieve a pre-determined lower power level. Appropriate plant parameters have been identified for the automatic triggering of power setback procedure on the occurrence of those events which are envisaged to be managed through this procedure. Knowledge of the transient thermal hydraulic behavior of the whole plant during various events is essential to formulate the operating procedure. Plant dynamics code DYANA-P developed for PFBR has been utilized for this purpose. This paper discusses (i) events for which this procedure can be adopted, (ii) detailed implantation scheme of power setback, (iii) transient thermal hydraulic behavior of the whole plant during this procedure and (iv) benefits of this procedure.

1. Introduction

PFBR is a 500 MWe (1250 MWt) liquid sodium cooled fast breeder reactor and it is in an advanced stage of construction in India. The primary sodium circuit, which removes heat from the reactor core transfers heat to two secondary sodium circuits through intermediate heat exchangers (IHXs). The secondary sodium circuits transfer heat to a steam water system also known as balance of plant (BoP) to produce steam that runs the turbine-generator to produce power. Schematic flow sheet of PFBR is shown in Fig. 1. Design of a fast breeder reactor (FBR) is governed largely by thermal hydraulics and structural mechanics. Lower operating pressure of sodium systems leads to a component design often controlled by thermal loads. High operating temperature of the reactor and the consequent thermal deformation of components (resulting in possible misalignment among various components), demand critical attention in the minimization of thermal fatigue loading due to various plant transients. Several design and operating features are adopted in FBR to accomplish this. For example, primary and secondary sodium pumps are provided with dedicated flywheel so as to avoid rapid flow reduction thereby to prevent sharp rise in coolant temperature during pump trip events. Following an emergency shutdown of fission power generation in reactor core (SCRAM), sodium pumps are operated at a reduced speed to attenuate the magnitude of cold shock (Natesan et al., 2000) on structures immersed in sodium pools. Manual controlled shutdown procedure is implemented for the management of events which do not cause immediate concern on nuclear safety and consequences of which evolve rather slowly.

PFBR is designed to take care of ~1040 design basis events in the plant life of 40 y, out of which ~750 events require SCRAM. In the remaining 290 events, ~110 events can be managed by controlled shutdown of the plant. The remaining 180 events, originates from the BoP, excluding the boiler feed pumps (BFPs). These events are:

1. Turbine-Generator (TG) trip.
2. Plant-Grid disconnection.
3. One condenser extraction pump (CEP) trip and standby not starting.
4. One condenser cooling water pump (CCWP) trip.

The above events can be managed by reducing reactor power automatically to a lower value and then continuing plant operation at this level till the initiating fault is corrected. This procedure is termed as "Reactor power setback". The selection of the lower power level for this procedure depends on the design provisions available in the plant for the management of above events. This procedure avoids components from being subjected to severe thermal shock due to SCRAM following the above 180 events. Power setback procedure being automatic, appropriate plant parameters have to be identified for the initiation of the procedure. Knowledge of the transient core neutronics and thermal hydraulic behavior of

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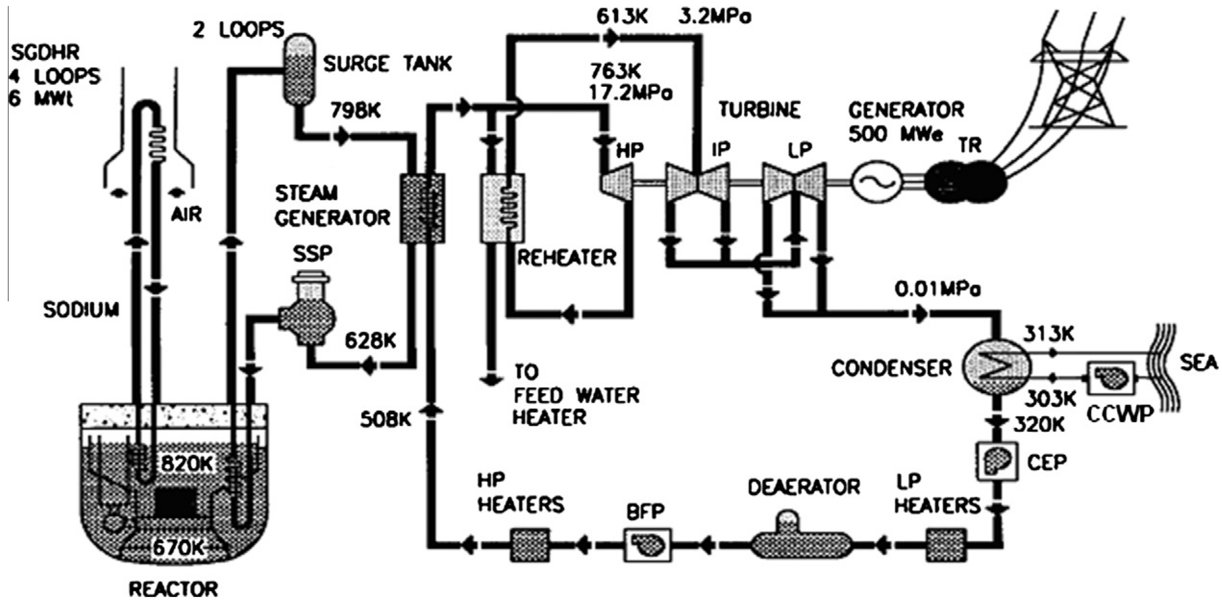

Fig. 1. Schematic flow sheet of PFBR.

the plant during this procedure is essential for the successful implementation of this procedure. Plant dynamics code DYANA-P (Natesan et al., 2012) developed for PFBR has been utilized to acquire this knowledge.

Several plant dynamics models have been developed world wide. Most of these codes are system specific such as IANUS (Addition et al., 1973) and DEMO (WARD report, 1975) which are developed to simulate the overall response of LMFBR plants in US viz., fast flux test facility (FFTF) and Clinch River Breeder Reactor Plant (CRBRP) respectively. IANUS code models primary and secondary circuits of the plant with a dump heat exchanger whereas DEMO code models steam generating modules also. NALAP (Martin et al., 1975) is another code developed for the transient simulation of LMFBR which is developed basically from the famous system dynamics code, RELAP 3B (Relap-3B Manual, 1974) for water cooled reactors. This code has been established by incorporating sodium properties in place of water in the RELAP 3D code and is capable of predicting rudimentary decay of core flow following a pipe rupture accident. There are also several generalized system codes such as SSC developed by Brookhaven National Laboratory (Guppy, 1983) for loop and pool type LMFBRs. The French code, CATHARE adopted for the safety analysis of PWR has been modified for sodium reactors recently (Geffraye et al., 2009). The Japanese code Super-COPD (Yamada et al., 2009) developed for sodium cooled fast reactors has been validated against startup tests carried out in MONJU reactor. Similarly, NETFLOW++ (Mochizuki, 2009) is a code developed for prediction natural convective flow behavior in sodium systems. Special feature of TRACE code (Chenu et al., 2009) is its capability to simulate one dimensional and two dimensional sodium boiling situations. Most of the codes described above are developed with a view to demonstrate safety of the plant under various design basis events and hence detailed models for BoP systems were not included initially. Events originating from the BoP are simulated by considering conservative boundary conditions for the water inlet side of steam generator (SG).

A BoP modeling capability has been added to SASSYS (Warinner and Dunn, 1985), extending its transient simulation capability to the feedwater/steam circuit. Briggs (1989) and Ku (1989) have reported a number of test problems simulating the feedwater circuit, viz., simultaneous trip of feedwater pumps and closure of turbine admission valve. Another code, NATDEMO (Planchon et al., 1986) was developed by Argonne National Laboratory in a general form to analyze either a pool or loop type reactor system. This code has models for primary, secondary, condenser and feedwater systems. Later versions of DEMO IV, SSC-L Rev-2 and NETFLOW++ codes have models for steam water system included. Dynamic Simulator for Nuclear Power Plants (DSNPs) (Larson et al., 1984) allows a whole plant thermal-hydraulic simulation using specific component and system models.

DYANA-P code consists of neutronic model for core, mass and momentum balance models for primary and secondary sodium circuits and heat balance model for the entire plant upto the SG. Steam–water system of the plant is not modeled in the code. Reactor power setback is envisaged for events originating from balance of plant other than those affecting operation of boiler feed pumps. Therefore, SG cooling is not affected due to the initiating event. However, SG cooling is adjusted in response to the reactor power reduction adopted during this procedure. Process dynamics of steam–water system does not affect the performance of SG during the initiating events for which the power setback is envisaged till this procedure is initiated. Dynamics of steam–water system decides only the time of initiation of power setback. Once the reactor power setback is initiated, then, SG cooling is adjusted in response to the power evolution in the core by the SG sodium outlet temperature controller which manipulates the feedwater flow. SG is the process link between nuclear steam supply system and the steam water system. Hence, the transient thermal hydraulics of power setback procedure in the nuclear steam supply system can be studied without a coupled modeling of steam–water system and nuclear steam supply system. Modeling of steam water system is required only for estimating the time of triggering of reactor power setback procedure which can be estimated through simplified models. This paper discusses detailed implementation of power setback scheme and transient thermal hydraulic behavior of the plant during this procedure.

2. Reactor power level for power setback

Steam–water system of PBFR is provided with large inventory of feedwater in the deaerator tank and condenser well. Therefore, the events originating from the steam–water system affects the reactor side slowly. Automatic trip of the plant is generally not essential for ensuring nuclear safety. From the point of view of reducing