Core analysis of the first cycle of Chashma nuclear power plant

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Abstract. The upcoming 300 MWe CHASHMA NPP will provide the opportunity to study the burn-up behavior of the fuel. Our experience is limited to the incore fuel management studies when fuel burn-up remains within the design limits. The initial core is loaded in three regions with fuel of three different enrichments 2.4 w/o, 2.67 w/o & 3.0w/o. It is intended to study the enhanced fuel burn-up vis-à-vis the expected cost benefit in due course of time. The core of the Chashma nuclear power plant is that of a typical PWR NPP of 300 MWe capacity. It has 121 fuel assemblies and all of them have identical external dimensions and hydraulic characteristics. The core height is 290 cm and equivalent diameter is 248.6 cm. The core is cooled and moderated by H2O and surrounded by a Stainless Steel baffle. Each fuel assembly consists of 15 x 15 rod array and the assembly pitch is 20.03 cm. The average discharge burn-up is 30,000 MWd/MTU. Core analysis was carried out for the first cycle at hot full power (HFP). Two dimensional calculations were performed for burn-up analysis including core multiplication, flux distribution, burn-up length, isotopic inventory, peaking factor and critical boron concentration to achieve the economical fuel management within the constraints imposed by safe reactor operation. Calculations indicate that expected burn-up of the first cycle is 13479 MWd/MTU equivalent to 485 EFPD, with 25 ppm of boron is still in the system, which is very near to the design value. Similarly assembly power distribution, pin by pin power distribution and reactivity coefficients, calculated are within the acceptable limits. Efforts are on to improve further these calculations.

1. INTRODUCTION

Chashma Nuclear Power Plant is a 300MWe two loop PWR presently under construction on the bank of Indus river, at the south of Mianwali, in the Punjab province, about 300 Km from Islamabad. It will be operational some time in the second half of 1999. Incore fuel management programme for Chashma NPP has to be developed in order to have an economical and safe power generation within safety [1,2] limits from the plant. Therefore efforts are underway to have an extensive and reliable incore fuel management programme for the plant.

2. REACTOR CORE

2.1. Description

The core of the CHASHMA nuclear power plant is that of a typical PWR NPP of 300 MWe rated capacity. It has 121 fuel assemblies and all of them have identical external dimensions and hydraulic characteristics. The core height is 290 cm and the equivalent diameter is 248.6 cm. The core is cooled and moderated by H2O and surrounded by a Stainless Steel baffle. Each fuel assembly consist of 15x15 rod array and assembly pitch is 20.03 cm. The cladding material is Zircaloy-4, burnable poison material is Borosilicate Glass, and the control rod material is Ag-In-Cd alloy. The spacer grids and top & bottom fittings hold the fuel rod within the assembly. The fuel assembly is provided with guide thimbles for cluster of control rods. All other details and features are according to the standard PWR assembly [3].
2.2. Fuel Enrichment & Loading

Fuel assemblies of three different enrichments are used in the initial core, in order to obtain a favourable radial power distribution and burn-up.

Region I  2.40 w/o U235
Region II  2.67 w/o U235
Region III 3.00 w/o U235

The first two regions, which consist of the lower enrichment are arranged in modified checkerboard pattern towards the center of the core. The third region is arranged around the periphery of the core. Configuration of the core including the burnable poison (BP) and rod cluster control assembly (RCCA) in the standard PWR core.

3. METHODOLOGY

3.1. Cross-section generation

Core analysis was carried out with a two dimensional computer code using the following sets of cross section:

1)  2.4 w/o U235 without BP, with 16 BP and 20 control rods,
2)  2.67 w/o U235 without BP, with 8 & 16 BP and 20 control rods,
3)  3.0 w/o U235 without BP, with 8 & 16 BP,
4)  Core baffle,
5)  Reflector,

Four Group Microscopic Cross section were calculated from Cross Section Generation codes.

3.2. Two dimensional analysis

The mesh point scheme in xy plane is shown in figure 1. One fourth core was analyzed for the following cases.

- Criticality analysis as a function of burn-up with no soluble Boron,
- Critical Boron concentration with burn-up,
- Peaking factor as a function of burn-up,
- Burnable poison remaining fraction with burn-up,
- Expected cycle length of first cycle,
- Assembly power distribution as a function of burn-up.
3.3. Reactivity coefficients

During reactor operation, the plant condition variations, operator interventions and abnormal or accidental transients will lead to the corresponding response of the core. The core response is determined by the core kinetics characteristics, which are associated with reactivity coefficients. The reactivity coefficients reflect the change in the neutron multiplication due to varying plant conditions such as power, moderator or fuel temperature or less significantly due to change in pressure or void conditions. A three-dimensional code based on nodal method employing perturbation techniques, calculates reactivity coefficients. Various reactivity coefficients have been calculated at the beginning of life (BOL). Two energy group average cross section have been used in the analysis.
3.3.1. Moderator temperature and density coefficients

The moderator temperature coefficient curves as a function of moderator temperature at BOL, ARO (all rods out) and at boron concentration 0, 500, 1000, 1500 and 2000 ppm were computed. The moderator coefficient is calculated for the various plant conditions by performing two group perturbation calculations, varying the moderator temperature by about ± 1°C each of the mean temperatures. The temperature range covered is from cold 20°C to about 320°C. Moderator temperature coefficient has been calculated as a function of boron concentration at Hot zero power (HZP) and hot full power (HFP).

3.3.2. Doppler temperature coefficient

Doppler temperature coefficients as a function of fuel effective temperature were calculated at BOL and ARO for moderator temperature 280°C and 302°C with 1°C change in fuel temperature.

3.3.3. Doppler Power coefficient

Doppler Power coefficient as a function of rated power (0 to 100%) at ARO and BOL were calculated with power change of 9.989E06 Watt. Doppler induced negative reactivity was also calculated at ARO and BOL as a function of Power when boron concentration was 1000 ppm.

3.3.4. Total Power coefficient

Power coefficient at BOL, ARO were calculated as a function of rated power (0 to 100%) when Boron concentrations were 750, 1000 and 1250 ppm. Power defects due to the power increase were also calculated at BOL and ARO with Boron concentration 750, 1000, and 1250 ppm. Differential worth of soluble boron verses power level at BOL ARO were estimated with soluble boron concentration ranging from 750 to 1250 ppm.

3.4. Power Distribution

The axial power distribution and pin-by-pin power distribution in the radial direction has been calculated. A one-dimensional and two-dimensional diffusion theory based computer codes were used for this purpose. Analysis has been performed for various conditions for hot rod assembly, core central assembly and core periphery.

The one dimensional axial computer code calculates the mesh wise axial power distribution for the core. The one-dimensional group constants are converted from two dimensional group constants by space and flux averaging. The effective multiplication factor calculated for two-dimensional core has been adjusted for the one-dimensional configuration by modifying the \( \nu_{GR} \) and radial buckling.

The two dimensional diffusion theory based code we use calculates power distribution at desired points in an assembly by using the four group microscopic cross section homogenized for a quarter of the assembly. Power distribution output within an assembly does not include the heterogeneous effect due to different types of cells viz.: fuel, burnable poison, control rod, thimble etc. in an assembly. The shortcoming of the code is overcome by using the pin-by-pin flux distribution obtained from the thermal spectrum calculation code.
The relative (output of two dimension code, as discussed above) power distribution in each pin is multiplied with relative (output of spectrum code, as discussed above) flux in the corresponding pin to get the pin-by-pin power distribution map for the complete core. The input of two dimensional codes was prepared in a manner to print the power at every fuel pin center in a quarter of core.

4. RESULTS & DISCUSSIONS

Burn-up analysis was performed and results obtained were analyzed to check their conformity with the well established reactor physics principles.

Criticality analysis as a function of burn-up without Boron is shown in figure 2. Initially decrease in $K_{eff}$ is due to Xenon-equilibrium. The reactor remains critical at 13000 MWD/MTU. Critical Boron concentration versus burn-up is given in figure 3. The initial decrease in boron concentration up to 135 MWD/MTU is due to the formation of Xenon and after 135 MWD/MTU, the burn-up is consistent with the boron concentration.

Radial Peaking Factor as a function of burn-up is described in figure 4. The graph shows that peaking factor decreases in the beginning and after reaching Xenon-equilibrium peaking factor increases with burn-up up to 6000 MWD/MTU. This increase is due to uneven distribution of burnable poison. At the End of Life (EOL), the power distribution is more flat due to burnable poison depletion with U-235.

Burnable poison remaining fraction with burn-up is depicted in figure 5. Its behavior conforms to the criteria, i.e. 25 ppm of soluble Boron concentration should be present at EOL. Expected burn-up of first cycle is 13145 MWD/MTU, with 25 ppm of Boron still in system. The calculated burn-up is greater than the design value, which is more conservative.

The analysis for moderator temperature coefficient behavior has been studied for the BOL/ARO conditions. Moderator temperature coefficient as a function of moderator temperature is plotted in figure 6 for 0 ppm to 2000 ppm for full range of operation. Here it is observed that increase in temperature of moderator or decreases in boron concentration gives a more negative moderator temperature coefficient by hardening the neutron spectrum. The neutron spectrum is hardened both by increase in moderator temperature as the moderator density decreases reducing moderation and by decreasing boron concentration as the boron density decreases giving a reduced effect of neutron poison. These curves are of importance from stand point of boron dilution/addition.

The Doppler temperature coefficient as a function fuel temperature is shown in figure 7 for the full range of operation for two moderator temperatures 280°C and 302°C. This is because the Doppler coefficient is less negative at higher fuel temperature owing to the saturation of resonance broadening effect. It becomes more negative at higher moderator temperature due to hardening of neutron spectrum.

The axial power distribution results give relative power for various core heights for different core conditions viz.: beginning of life (BOL), middle of life (MOL) and end of Life (EOL), including zero and Equilibrium Xenon at all rod out (ARO) positions. Figure 8 shows Relative axial power distribution at ARO for the no Xe at BOL and Equilibrium Xe at BOL, MOL, EOL conditions whereas Figure 9 shows relative axial power distribution for the BOL core at RCCA inserted at 120 steps. It has been observed that all these vital parameters are well behaved and in good agreement (within ±5%) with quoted design values[4]. Efforts are on to improve further these calculations.
FIG. 2.

FIG. 3.
FIG. 4.

FIG. 5.
FIG. 6.

Moderator temperature (°C)

FIG. 7.

Fuel effective temperature (°C)
FIG. 8.

FIG. 9.
REFERENCES