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Study on Flexible Fuel Management Strategy of Haiyang Nuclear Power Plant

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Abstract. Flexible fuel management in a nuclear power plant relates to the operational efficiency of the whole plant, which not only satisfies flexible fuel demands, but also has a positive effect on environmental pollution control. The proposed management strategy aims at improving energy utilization rate of nuclear fuel in nuclear power plants, meeting the needs of peak electricity consumption in summer and nuclear energy heating in winter, enhancing the economic efficiency of nuclear power plants, and reducing carbon emissions, thus contributing to energy carbon neutrality. In this paper, the Core and System Integrated Engine for design and analysis (COSINE) nuclear design software package was adopted to study the fuel management strategy for alternate refueling in 68 groups and 60 groups of assemblies, based on the current 64 groups of refuel assemblies at Haiyang Nuclear Power Plant (NPP). The results show that the Haiyang NPP has the capability to refuel flexibly between 68 and 60 groups of assemblies, and the reactor core design meets the relevant design guidelines and requirements.

Keywords. COSINE, fuel management, nuclear power, EFPD

1. Introduction

Units 1 and 2 of the Haiyang Nuclear Power Plant (HNPP) are currently using a low-leakage and high-fuel-consumption fuel management strategy of 18-month refueling [1]. According to the current fuel management strategy, there are 157 fuel assemblies in the core, and the number of assemblies per cycle is 64. The cycle lengths of the first, second and third cycles (balance cycle) are approximately 450 EFPD, 516 EFPD and 512 EFPD respectively, which are relatively fixed.

With the commencement of official commercial operation, Haiyang NPP was faced with the requirement to adjust the cycle length appropriately, for example, to avoid scramming the core for overhaul during the summer months when electricity consumption is at its peak, to avoid overhaul both units at the same time to protect people's livelihood from heating in winter, to prevent overlapping overhauls and to increase the overhaul interval of both units, etc. Based on above, the study on flexible fuel management strategies for adding and subtracting four groups of new fuel assemblies at Haiyang NPP was carried out to meet the requirements on flexible refueling, reduced carbon emissions, and to contribute to the carbon neutrality in China.

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2. Research Method

Based on the current fuel management strategy with 64 assemblies refueling at the Haiyang Nuclear Power Plant, the fuel management solution design and safety analysis [2] were carried out using the perturbation model of adding or deleting four assemblies successively and alternatively, which is the most perturbing to the refueling solution [3]. Since addition and subtraction of four assemblies successively and alternatively will bring perturbation to the current 64-assembly refueling pattern to the largest extent, thus the safety and feasibility of adding and subtracting four refueling assemblies.

In order to demonstrate the flexible refueling management strategy, the existing final safety analysis report for the Haiyang NPP was referred, the layout of flexible cycle refueling core, the core generic safety analysis parameters, validation of relevant design guidelines, power capacity demonstration, validation of critical accident envelope limits [4] and other aspects that may be affected, were taken into consideration. This study is based on the core fuel management report for Units 1 and 2 in Haiyang NPP. Four groups of new fuel assemblies were added and deducted successively and alternatively from the third cycle onwards. In the design of reactor core in this study, 68 groups and 60 groups of fuel assemblies were refueled alternatively, and then transited to a balanced cycle.

3. Design Requirements and Criterions

The following design requirements and criterions have been followed in the proposed strategy in this paper.

(1) 68 groups and 60 groups of fuel assemblies were refueled successively and alternatively.

(2) Following safety requirements were satisfied [5].

- When under full power in heat condition, the nuclear enthalpy rise hot channel factor $F_{\triangle H}^{N}$≤1.72.
- When under full power in heat condition, power peak factor F_Q≤2.60, thus to meet LOCA limits on accident consequences.
- When the core is working under any power level (including Hot Zero Power (HZP)), the moderator temperature coefficient (MTC) should not be positive, MTC ≤ 0 pcm/°C.
- When the core is working under any power level (including HZP), the shutdown margin (SDM) 1,600 pcm.
- Maximum average fuel rod burnup in the core should not exceed 62000 MWd/tU to meet the requirements of the fuel rod performance analysis.

4. Description of the Core Fuel Management Strategy

This study was carried based on the second cycle of the core fuel management scheme for Unite 1 and 2 at Haiyang NPP. 68 groups and 60 groups of fuel assemblies were refueled successively and alternatively. A low-leakage fuel distribution method [6] and only integral fuel burnable absorber (IFBA) were adopted. 68 refueled assemblies were divided into two batches according to enrichment. The average enrichments of refueled assemblies (excluding the axial low enrichment zone) were 4.45 w/o and 4.95 w/o, and the numbers of assemblies were 36 and 32 respectively. 60 refueled assemblies were divided into two batches the same way. The average enrichments of refueled assemblies (excluding the axial low enrichment zone) were 4.45 w/o and 4.95 w/o, and the numbers of assemblies were 36 and 24 respectively. In order to reduce neutron leakage at the ends of the active zone of the core, axial low enrichment zones with an enrichment of 3.20 w/o were used at the top and bottom ends of all the refuel assemblies. Fuel pellets of 101.6 mm were distributed at both ends of the non-low enrichment zone in the active zone without IFBA, to balance the axial power distribution of the core.

5. Introduction of Software COSINE

The core fuel management solution was calculated using the reactor core design software cosNU in the COSINE (Core and System Integrated Engine for design and analysis) software package for nuclear power plant safety analysis and engineering design. The COSINE software package is a major national project approved by the National Energy Administration and undertaken by the State Power Investment Institute of Science and Technology Company. The project is dedicated to the development of a complete set of software for the design and safety analysis of second and third generation advanced pressurized water reactors with fully independent intellectual property rights in China. The cosNU software includes the component parameter calculation software cosLATC [7] and the core physical analysis software cosCORE [8].

6. Calculation Result

Using cosNU software, the key physical parameters in the core fuel management strategy were calculated and the relevant design guidelines were validated.

6.1. Calculation Result of Key Physical Parameters

The main core physical parameters for each cycle are given in Table 1. And it can be seen that the critical safety parameters of the core [9] are within the design guideline requirements under the flexible cycle fuel management strategy.

6.2. Variation of Core Operating Parameters with Fuel Consumption in Each Cycle

Figures 1-4 present the full power critical soluble boron concentration, variation of core AO [10], critical heat flux hot channel factor, and nuclear enthalpy rise hot channel factor [11], for each burnout of the core during the simulated operation. It can be seen that the power factor under each cycle does not exceed the design criterion; the critical boron concentration shows two patterns under 68 groups and 60 groups, which is as expected; the AO tends to stabilize after deepening of the burn-up, which is in line with the core operation regularity.

Table 1. Calculation result of key physical parameters in each cycle.

Key physical parameters	3rd cycle	4th cycle	5th cycle	6th cycle	7th cycle	8th cycle
Number of refueling assemblies (group)	68	60	68	60	68	60
Number of burnable poison rod (IFBA)	8128	7104	8128	7104	8128	7104
Initial (BOL, NOXE, hot full power (HFP)) critical boron concentration (ppm)	1950.9	1796.0	1979.7	1767.9	1968.7	1760.0
Effective full power days (EFPD)	532.2	478.7	534.9	476.4	535.5	476.2
Maximum $F_{\triangle H}^{C}$ (HFP)	1.494	1.516	1.497	1.523	1.494	1.516
Maximum $F_{\triangle H}^{N}$ (HFP)	1.614	1.637	1.617	1.645	1.614	1.637
Maximum F_{Q}^{C} (HFP)	1.775	1.775	1.844	1.813	1.804	1.818
Maximum F_{Q}^{T} (HFP)	1.949	1.945	1.994	1.961	1.951	1.966
Average of discharge burnup (MWd/tU)	45548	49521	49831	49458	49824	49438
Maximum assembly burnup (MWd/tU)	53233	53294	54461	53666	54661	53771
Maximum rod burnup (MWd/tU)	54501	56043	58736	58213	58779	58306

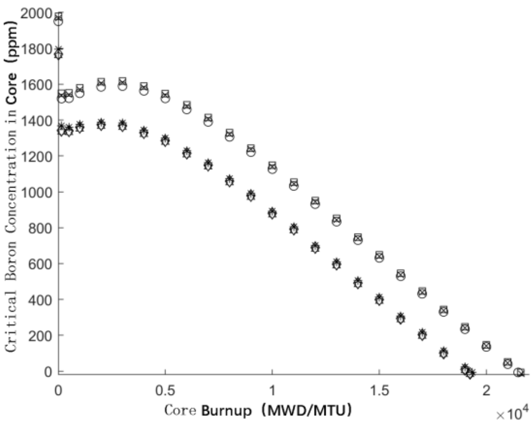

Figure 1. Variation of critical boron concentration with core burnup in each cycle.

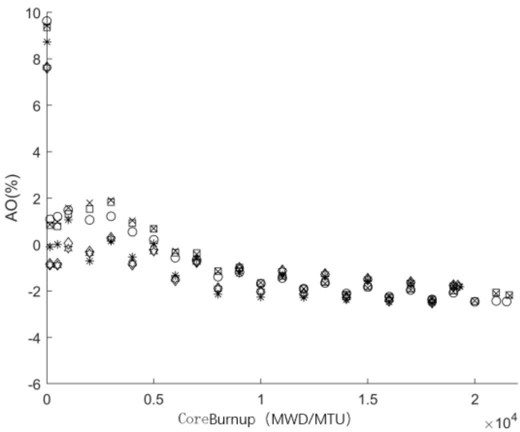

Figure 2. Variation of AO with core burnup in each cycle.

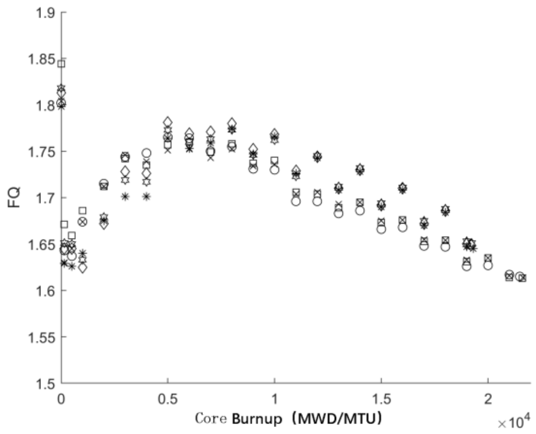

Figure 3. Variation of heat flux hot channel factor with core burnup in each cycle.

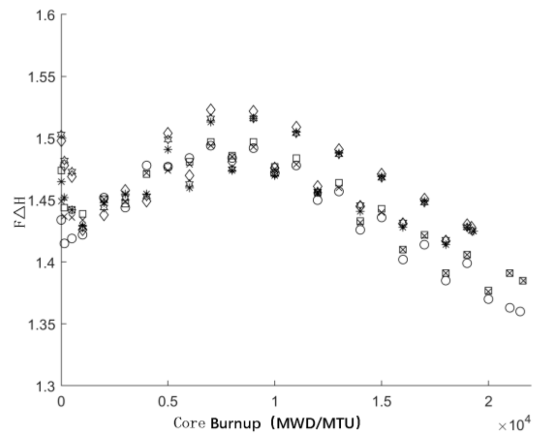

Figure 4. Variation of nuclear enthalpy rise hot channel factor with core burnup in each cycle.

6.3. Validation of the Moderator Temperature Coefficient

Table 2 gives the moderator temperature coefficients (MTC) for the conditions corresponding to the transition and equilibrium cycles. The moderator temperature calculation starts with an analysis of the most conservative HZP all rod out (ARO) no xenon (NOXE) condition, which does not really exist in plant operation. For this condition where the moderator temperature coefficient exceeds the limit, this report validates the moderator temperature coefficient by simulating the actual start-up process of the nuclear power plant. Although the most positive MTC for the HZP ARO NOXE lifetime of the 68 groups refuel assemblies cycle is slightly greater than 0, the MTC is less than 0 during the whole start-up process, so the MTC value can meet relevant requirements.

6.4. Validation of SDM

Table 3 shows the results of the SDM calculations for the transition cycle and the equilibrium cycle, both of which meet the requirement that SDM should be less than 1600 pcm.

Table 2. MTC of transition cycle and equilibrium cycle.

	Transition cycle (68 groups)	**Transition cycle (60 groups)**	**Equilibrium cycle (68 groups)**	**Equilibrium cycle (60 groups)**
HZP, ARO, NOXE condition most positive MTC (pcm/°C)/ corresponding burnup point (MWd/tU)	0.58/0	-2.34/0	0.99/0	-2.81/0
HFP, ARO, equilibrium xenon (EQXE) condition most positive MTC (pcm/°C)/ corresponding burnup point (MWd/tU)	-17.71/ 2000	-23.18/ 1000	-17.14/ 2000	-23.70/ 1000

Table 3. SDM of transition cycle and equilibrium cycle.

Parameters	**Transition cycle (68 groups)**		**Transition cycle (60 groups)**		**Equilibrium cycle (68 groups)**		**Equilibrium cycle (60 groups)**	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
Control requirement								
Total power loss (%Δρ)	1.74	3.15	1.86	3.20	1.74	3.18	1.87	3.19
Redistribution effect (only unfavorable Xenon) (%Δρ)	0.29	0.33	0.36	0.31	0.30	0.32	0.36	0.31
Remaining control rod (%Δρ)	2.50	2.50	2.50	2.50	2.50	2.50	2.50	2.50
Void acctivity (%Δρ)	0.05	0.05	0.05	0.05	0.05	0.05	0.05	0.05
Overall control requirement (%Δρ)	4.58	6.03	4.77	6.06	4.59	6.05	4.78	6.05
Value of control rod (69 bunches)								
a. all control rods inserted (%Δρ)	9.70	10.12	9.36	10.55	9.95	10.26	9.66	10.61
b. 68 bunches of control rods inserted (to reach the maximum value) (%Δρ)	8.24	8.80	7.98	8.96	8.33	8.88	8.22	9.04
Value of control rod without 7% uncertainty (%Δρ)	7.66	8.18	7.42	8.33	7.74	8.26	7.64	8.40
SDM (%Δρ)	3.08	2.15	2.65	2.27	3.15	2.21	2.86	2.35

7. Conclusion

Using the COSINE program, a study of alternate refueling in 68 groups and 60 groups of assemblies was carried out on the basis of the existing fuel management strategy for the Haiyang NPP, and a fuel management scheme for the Group 68 and Group 60 alternate cores was designed. The calculations of the main physical parameters and verification of the relevant design guidelines were carried out for each cycle. The results and conclusions of the calculation and analysis are as follows.

(1) The life span of the 68 groups refuel assemblies in equilibrium cycle is

approximately 535.5 EFPD, the average unloading fuel consumption is approximately 49,824 MWd/tU, and the average fuel consumption of the largest rod is 58,779 MWd/tU. The life span of the 60 groups refuel assemblies in equilibrium cycle is approximately 476.2 EFPD, the average unloading fuel consumption is approximately 49,438 MWd/tU, and the average fuel consumption of the largest rod is 58,306 MWd/tU which meets the design requirements.

(2) The relevant design parameters of the transition cycle and the equilibrium cycle meet the requirements of the relevant design guidelines, i.e. the nuclear enthalpy rise hot channel factor, heat flux hot channel factor, moderator temperature coefficient and maximum fuel rod burn-up are all lower than ultimately required, and the SDM is greater than the minimum requirement.

In summary, this study proved that Haiyang has the capability to refuel flexibly between 68 and 60 groups of assemblies and the designed core scheme meets the relevant design guidelines and requirements. It is possible to actively adjust the strategy according to the energy demand and carry out large scale nuclear energy heating and water-heat cogeneration projects to reduce carbon emissions and enhance energy carbon neutrality.

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