Technical Support to an Operating PWR vis-à-vis Safety Analysis

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Abstract. Currently a PWR of 300 MWe capacity CHASNUPP-I is in operation since the year 2000. Technical support being provided includes in-core fuel management and corresponding safety analysis for the reshuffled core for the next cycle. Currently calculation and analysis were performed for Cycle 6 to achieve the safe and economical loading pattern. The technique used is designated as out in mode (modified). In this technique, most of the fresh fuel assemblies are not directly located at the periphery of the core, but near the boundary. This technique has the advantage that without using burnable absorber we can design a low leakage core with extended cycle and maximum batch averaged burnup.

1. INTRODUCTION

The fuel loading pattern selection calculations for Cycle 6 have been performed in order to determine the arrangement of the new and depleted fuel assemblies in the core. The loading pattern selection, which indicates the beginning of the reload design for a new cycle, is very important for in-core fuel management. The selected loading pattern is directly related to the safety and economy during the operation of a nuclear power plant. To ensure a safe operation of C-1, all criteria defined in the relevant Technical Specifications, Design Criteria and Safety Guide must be obeyed for the reload design for every Cycle keeping in view the balance in utility's demands from economical point of view and engineering reality as well.

1.1. Refueling Design Objectives and Bases

The following design criteria and objectives should be obeyed in the fifth core reload design:

At core full power 998.6 MWth (not including pump power), hot channel factors of 2.70 and 1.60 for F_Q^T and $F_{\Delta H}^N$ respectively, will not be exceeded.
During normal operation, the total peaking factor at elevation Z (QT(Z)) envelope must observe the limit which varies with the core elevation (Z) imposed by thermal hydraulic requirements to limit the consequences of hypothetical Loss of Coolant Accident (LOCA).
At any power level including hot zero power (HZP), the moderator temperature coefficient should not be positive.

With the most reactive control rod stuck out of the core, the remaining control rods shall be capable of shutting the reactor down with a margin not less than 2000 pcm.

The specified fuel loading shall possess sufficient reactivity, to produce the required amount of energy over the specified cycle length at nominal power. The main objective in loading pattern optimization is the achievement of maximum average depletion in the discharged fuel assemblies.
No burnable absorbers are used in Cycle 6.
Quarter core symmetry is to be essentially maintained.

1.2. Refueling Mode

At the end of life (EOL) of Cycle 5, 40 assemblies with initial enrichment of 3.40 w/o of batch 5 and 5 assemblies of batch 1 with 2.4 w/o initial enrichment will be discharged. While 36 twice burnt assemblies of 3.40 w/o of batch 6 and 40 once burnt assemblies of batch 7 will be further used in cycle 6. 45 more assemblies are required for cycle 6. For the refueling mode two options have been used. For option 1 we have used 40 fresh assembles of batch 8 and 5 once burnt assemblies from batch 1 discharged at EOL of cycle 1. For option 2 we have used 44 fresh assemblies and one assembly from batch 1 at core center.

As in cycle 2, 3, 4 & 5 the burnable absorbers had not be used in Cycle 6. The traditional "out-in" mode may be applied for the reloading, that is, the depleted fuel assemblies are located in the inner region and the fresh fuel assemblies in the peripheral region of the core. The main advantage of this mode is to flatten the radial power distribution and to increase the safety margin of the core. In order to decrease the fast neutron irradiation on the pressure vessel, to prolong core life time and improve economic gain, a few fresh assemblies may be moved towards the inner of the core, with some depleted assemblies loaded in the peripheral region. This type of loading will decrease the radial neutron leakage, increase the batch-averaged discharged burn-up and increase the cycle length without using burnable absorbers. Such a loading pattern may be referred to as "modified out-in" mode. In this mode, most of the fresh fuel assemblies are not directly located at the periphery of the core, but still near the boundary, some fresh asseblies can be loaded near the core center.

1.3. Loading pattern Selection

The available fuel assemblies for cycle 6 are: batch 1 (5 assemblies of 2.4 w/o), 36 fuel assemblies of 3.4 w/o batch 6, 40 assemblies of 3.4 w/o batch 7 and 44 fresh assemblies of 3.4 w/o.

Finite difference based two dimensional (x-y) computer code has been used for reload design. With given objective function (for example, the maximal batch-averaged discharge burnup) and given restrictions (for example, the power peaking factor in the whole cycle not larger than a certain value), for given schemes of the reload core, the semi-automatic calculation are performed with this computer code.

Selection of loading pattern of C-1, Cycle 6, has been performed in three stages. In stage first computer code executes all possible loading patterns for a selected refueling mode at beginning of life (BOL). In second stage selectected loading patterns from stage 1 are analysed up to end of life by using quarter core symmetry. After the second stage, some loading patterns are obtained for detailed analysis. Core calculations are performed with different computer codes in order to determine whether the power distribution, reactivity coefficient, control bank worth, cycle length, shut down margin and so on satisfy the safety guide and economical requirement. The patterns which satisfy the saftey requirment enter in the final recommended list.

By using the experience upto cycle 5 for cycle 6 only three modified out in type loading pattern modes have been analyzed. These are with 4 once burnt assemblies of batch 1 and 12 twice burnt assemblies of batch 6 (case 1), 4 once burnt assemblies of batch 7 and 12 twice burnt assemblies of batch 6 (case 2 and 3) at core periphery. For these schemes some fresh assemblies are shifted slightly inside but near to the core periphery and four fresh assemblies have been shifted near the core center.
The key parameters of core characteristic are listed in Tables 1. It is noticed that the calculation conditions for initial boron concentration is BOL, hot full power (HFP), ARO, No Xe and critical. The maximum hot channel factor $F^{C}_{\Delta H}$ is calculated under HFP, ARO and critical condition for the whole cycle, including No Xe and EQ Xe.

2. SAFETY ANALYSIS

The safety criterion for the selected loading patterns is that these patterns should satisfy the requirements set forth in the Technical Specifications, Design Criteria and Safety Guide. In selecting the loading pattern of Cycle 6, the following aspects are considered:

Power distribution, moderator temperature coefficient, depletion of the discharged fuel assemblies and hot shutdown margin.

2.1. Power distribution

For selected fuel loading patterns, it is specified that the nuclear enthalpy rise hot channel factor $F_{\Delta H}^{N}$ and heat flux hot channel factor F_{Q}^{T} at HFP should not exceed the design limitations, that is, F_{Q}^{T} should not be larger than 2.70 and $F_{\Delta H}^{N}$ not larger than 1.60. Both the radial and axial power distributions determine the power peaking factors F_{Q}^{T} and $F_{\Delta H}^{N}$. The radial power distribution is relative fixed and may easily be bounded with upper limits. While the axial power distribution is more complex, it is affected by power level, control bank motion, axial Xenon transient, and so on. Therefore, the operation regulation for axial distribution is specified in the Technical Specifications. The axial power distribution may be bounded by setting control bank insertion limitation and controlling of the neutron flux axial offset. In the reload design, by changing the relative position of fuel assemblies with different initial enrichment or different depletion, the radial power distribution and the peaking factors F_{Q}^{T} and FNΔH may be adjusted. In practice, with consideration of the calculation uncertainties and correction factors, the calculated $F_{\Delta H}^{C}$ values are compared to their upper limits. On the basis of experience gained from last five cycles, if the $F_{\Delta H}^{C}$ is within limit then F_{Q}^{C} will also be within limit, so at loading pattern selection stage F_{Q}^{C} has not been calculated.

2.2. Moderator temperature coefficient

For the reload design the criteria for the moderator temperature coefficient is that it should not be positive when the core operates at any power level including HZP. However, for economy reasons, reactor cores with higher initial excess reactivity are usually used. This would lead to a higher initial critical boron concentration and a positive moderator temperature coefficient, especially under the condition of BOL, HZP and NOXE. When the positive moderator temperature coefficient at BOL, HZP, NOXE is not big enough, the control bank withdrawal limitation would be set in order to keep a negative moderator temperature coefficient under operation condition. For some loading patterns, this value might be too large, so the control bank must be very deeply inserted, which would be in conflict with the restriction of control bank insertion limitation, i.e. T4=2 steps and T3=192 steps for C-1. It is shown in table 1 at BOL, HZP, No Xe, ARO, critical, the moderator temperature coefficient is positive, for the three selected loading patterns. This phenomenon is inevitable for reactor with larger excess reactivity and without burnable absorber. In order to meet the reload design criteria, the control bank withdrawal limit should be set which are given in table 2.

2.3. Depletion

For the safety analysis, two parameters are important which are related to depletion. These are the batch-averaged burnup for discharged fuel assemblies and the maximal assembly burnup. For the sixth Cycle, maximum burnup as given in Table 1 are less than the limit values.

2.4. Hot shut down margin

The reload design criteria states that with the insertion of all control rods without the most reactive one (which is assumed to be stuck out of the core), there is sufficient reactivity present to shut down the reactor. This reactivity should be at least 2000 pcm. For a reactor without burnable absorber, from BOL to EOL, the critical boron concentration at HFP tends to reduce with increasing burnup, the moderator temperature coefficient become more negative. So the reactivity induced during the reactor shutdown at EOL is much larger than at BOL. While the control bank worth at EOL is almost the same as at BOL. Therefore, the hot shut down margin at EOL is more limiting than at BOL. The shutdown margin for the three selected loading patterns at EOL of Cycle 6 is shown in table 3. The shutdown margin of all loading patterns satisfies the reload design criteria.

Table1. Main physics parameters for cycle 6

Parameters		Pattern 6-1	Pattern 6-2	Pattern 6-3
Fuel Loading Mode		Modified "Out-In"	Modified "Out-In"	Modified "Out-In"
Once Burnt Depleted FA Number at Periphery of Core (and Position)		4 (A07) (2.4%)	4 (C11) (3.4%)	4 (A07) (3.4%)
Twice Burnt Depleted FA Number at Periphery of Core (and Position)		12 (A08,C11)	12 (A07,A08)	12 (A08,C11)
Initial BC at BOL, HFP (ppm)		1363	1434	1431
Cycle Length	(EFPD)	378	401.5	401.5
	MWD/MTU	10510	11164	11164
Average power of Periphery FA's		0.400	0.390	0.456
MTC at BOL, HZP, No Xe, ARO (pcm/C^{o})		3.46	4.987	4.8
Maximum Fcxy at ARO		1.381	1.401	1.413
Maximum $F^{C}_{\Delta H}$ at ARO		1.437	1.443	1.469
Maximum assembly burnup (MWD/MTU)		36130	36510	36475
Burnup (MWD/MTU) Increased during Cycle 6 for Batch 6 36 FA, 3.4 w/o)		8563	8891	8993
Burnup at EOL of Cycle 5 (MWD/MTU)				
Batch 1 (5 FA, 2.4 w/o)		21768		
Batch 1 (1 FA, 2.4 w/o)			25010	25315
Batch 6 (36 FA, 3.4 w/o)		33153	33481	33582
Batch 7 (40 FA, 3.4 w/o)		23883	24011	23851
Batch 8 (40 FA, 3.4 w/o)		11027		
Batch 8 (44 FA, 3.4 w/o)		-	11851	11906

Table 2. MTC and withdrawal limits of control rods for three proposed fuel loading schemes to make MTC=0 AT BOL, HZP, NO XE, CYCLE 6

Requirement	Pattern 1	Pattern 2	Pattern 3
MTC (pcm/oC)	3.464	4.986	4.8
Withdrawal limits of T4 and T3 for αm=0 (steps)	T4=80 T3=270	T4=35 T3=225	T4= 45 T3=235

Table 3. Verification of shutdown margin for the proposed fuel loading schemes at EOL, HZP, FOR CYCLE 6

Requirement	Pattern 1	Pattern 2	Pattern 3
Reactivity Insertion (HFP→HZP)	PCM	PCM	PCM
Doppler	820	817	817
Rod Insertion Allowance	500	500	500
Variable TMOD + Void Content	1088	1095	1096
Redistribution	850	850	850
(1) Total Reactivity Requirement	3258	3262	3263
Control Rod Worth (HZP)			
ARI (37 RCCA)	-9068	-8284	-8781
ARI Less Most Reactive Stuck Rod (36 RCCA)	-6965	-6577	-6744
(2) Less 10% for uncertainty	-6269	-5919	-6070
Shutdown Margin			
Calculated Margin [(2) – (1)]	-3011	-2657	-2807
Required Shutdown Margin	2000	2000	2000

3. COMPARISION OF PREDICTED DATA WITH MEASURED DATA

The comparison between various calculated and measured physics parameters are presented in Table 4. Measured critical boron concentration at HZP, BOL is compared with the calculated value, the calculated value is very much close to the measured value and is within the acceptable limits. The difference between measured and predicted boron end point concentration with bank T1 inserted is only 13 ppm. It is also observed that the calculated reactivity worth of bank T1 and RCCA banks in overlap mode agree well with the measured values as shown in Table-3. Important parameters shown in the Table is measured MTC which is 1.97 pcm/oC at ARO, critical boron concentration (CBC) condition whereas the calculated value is 2.82 pcm/oC at predicted CBC. The difference between measured and predicted axial offset at HZP condition is less than 2%. The radial power distribution at Zero power No Xe compared with measured values for 1/8 core is shown in fig. 1. and it is found that all values meet the acceptance criteria.

Table 4. Comparision of Key Predicted parameters with Measured data

PARAMETERS	MEASURED VALUE	PREDICTED VALUE	DIFFERENCE (Predicted.-Measured)
Critical Boron concentration (ppm)	1453	1457	4
Boron end point concentration Bank T1 In (ppm)	1254	1267	13
Control banks worth with Overlap (pcm)	1654.22	1708.63	3.28%
Control bank T1 worth (pcm)	1801.5	1771	-1.6%
Moderator Temperature Coefficient (pcm/oC)	1.97	2.82	-0.85
Axial Offset (HZP, T4=200)	-2.76%	-1.28%	<2%
$F_{\Delta H}^{N}$ (HZP, BOl, T4=200)	1.55	1.51	-2.45%
F_{Q}^{T} (HZP,BOl, T4=200)	2.11	2.09	-0.94%

Fig. 1. Comparision of radial measured and predicted power distribution (cycle 6, 0.0 EFPD, HZP, T4 = 200 STEPS)

	G	F	E	D	C	B	A
0 7	0.638	0.779	0.991	1.098	1.172	1.295	0.415
	0.63	0.785	0.986	1.089	1.171	1.294	0.411
	-1.20%	0.75%	-0.48%	-0.79%	-0.08%	-0.09%	-0.94%
0 8		1.165	0.955	1.131	1.008	1.227	0.354
		1.164	0.949	1.128	1.001	1.22	0.351
		-0.13%	-0.60%	-0.28%	-0.68%	-0.54%	-0.60%
0 9			0.955	1.092	1.299	1.272	
			0.949	1.094	1.302	1.272	
			-0.60%	0.25%	0.26%	-0.01%	
1 0				1.131	1.299	1.051	
				1.13	1.306	1.061	
				-0.05%	0.58%	0.97%	
1 1	Measured Value (M) ---------------				1.008		
	Calculated Value(C) ---------------				1.006		
	Deviation, [(C-M)/M]*100 -------				-0.18%		

4. ACCIDENT ANALYSIS

Safety evaluation of cycle 6 was carried out for the reshuffled core. All the probable accident scenarios based on initiating events as given in the FSAR were evaluated with respect to input parameters. For a specific event, the comparison of critical safety related core physics parameters between the reference case and current cycle was undertaken. In addition, the brief qualitative analysis is also provided if that of the reference case could cover the above parameters adopted in the refueling design. Otherwise, a re-calculation and corresponding analysis report is needed. “Long Window and Short Window” should be considered for the safety evaluation of the accident analysis. Thus, the conservative parameters are adopted in the accident analysis for “Long Window and Short Window” i.e. all analysis and evaluation are within the range of “Normal Window” ±500 MWd/tU.

4.1. Main Design Parameters

NSSS power is the same as that of the reference analysis (i.e. 1035 MWt).
Average coolant temperature under hot zero power (HZP) condition is the same as that of the reference analysis (i.e. 2800C). According to the actual operation condition, the average coolant temperature under hot full power (HFP) condition is 2980C which is 40C less than that of the reference analysis.
Pressurizer pressure is same as that of the reference analysis (i.e. 15.3 MPa).
Thermal design flow rate is same as that of the reference analysis (i.e. 24000 t/h).

4.1.1. Initial Conditions

In the same way as the reference analysis, the initial conditions used for the accident analysis are obtained by adding the maximum steady state allowances to the rated values. The positive or negative allowances are chosen from conservative point of view, depending on the accident analyzed.

The steady state allowances are the same as that of the reference analysis. The following conservative steady state errors were assumed in the analysis:

NSSS power	+3%
Average coolant temperature	±3°C
Pressurizer pressure	±0.196 Mpa

4.1.2. Power Distributions

The transient response of the reactor system is dependent on the initial power distribution. Power distribution may be characterized by the radial peaking factor (i.e. nuclear enthalpy rise hot channel factor $F_{\Delta H}^{N}$) and the total peaking factor (i.e. heat flux hot channel factor F_{Q}^{T}). The design value of $F_{\Delta H}^{N}$ and F_{Q}^{T} are 1.60 and 2.70 respectively in reference analysis, and the same values are taken as design limits for C-1 Cycle 6.

4.1.3. Reactivity Coefficients Used in Accident Analysis

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. In the analysis of certain events, conservatism requires the use of maximum reactivity coefficient values; whereas in the analysis of other events, conservatism requires the use of minimum reactivity coefficient values. The justifications for the conservative use of reactivity coefficient values are treated on an event-by-event basis.

4.1.4. Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is determined both by the function of the rod cluster control assembly (RCCA) position versus time and the variation in rod worth as a function of rod position. RCCA insertion characteristics following refueling are the same as that of the reference analysis.

4.1.5. Residual Decay Heat

In the reference analysis, residual heat in a sub-critical core is calculated for the loss of coolant accident according to the requirements of Appendix K of 10CFR50.46. These requirements determine fission product decay heat which assumes infinite irradiation time before the core goes sub-critical. For all other accidents, the same models are used except that fission product decay heat is based on core average exposure at the end of the equilibrium cycle. Thus, the assumptions used in the reference analysis are conservative to cover that of C-1 Cycle 6.

4.1.6. Burnup Influence on Fuel Material Characteristics

The material property of the fuel is varied with the fuel burnup. This effect was considered in the Thermal-hydraulic Design of FSAR Chapter 4. For example, the effect of the fuel burnup on the fuel melting temperature was considered and 25900C is conservatively selected as the fuel melting temperature. And a conservative gap conductance across the pellet-clad gap was used to cover various burnup conditions.

The parameters adopted in the reference analysis which related to the burnup effects on fuel material characteristics bound that condition of C-1 Cycle 6.

4.1.7. Others

The assumptions are same as that of the reference analysis such as the reactor trip setpoint and its time delay, systems and components available for mitigation of accident effects, limiting single failure and so on.
Each critical parameter of the current cycle is compared with that of the reference analysis for each event to justify the effectiveness of the reference analysis.

4.2. Feedwater System Malfunction Causing a Reduction in Feedwater Temperature or an Increase in Feedwater Flow

Reduction in feedwater temperature or addition of excessive feedwater flow will cause an increase in the heat transfer from the primary side to the secondary side in the steam generator and a decrease in RCS temperature. And it causes an increase in reactor power by a reactivity insertion due to the effect of the negative moderator temperature reactivity feedback. Meanwhile, the fuel temperature increases with reactor power and it causes a reduction in power due to the Doppler reactivity feedback effect. Thus, the most negative moderator temperature coefficient and the least negative Doppler coefficient are the critical parameters.

With the plant at no-load conditions, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load transient is less severe than the full power case.
The critical parameters for both Cycle 6 and reference analysis are listed as follows.

	Reference Analysis	Cycle 6
Maximum MTC (pcm/0C)	-59.0	-52.4
Minimum DC (pcm/0C)	-2.0	-2.32
Note:	Maximum: Most negative; Minimum: Least negative. MTC: Moderator Temperature Coefficient. DC: Doppler Coefficient.	

As shown above, the critical parameters of Cycle 6 are bounded by the reference analysis. So re-analysis is not needed.

4.3. Partial/Complete Loss of Forced Reactor Coolant Flow

Either partial or complete loss of forced reactor coolant flow will result in a rapid increase in the coolant temperature. Reactor power is reduced by the negative reactivity insertion. The minimum moderator reactivity feedback results in a most conservatively high reactor power preceding the trip. And Doppler reactivity feedback retards the power decrease due to the moderator reactivity feedback. For a conservative analysis, the least negative moderator temperature coefficient and the most negative Doppler coefficient are the critical parameters.

The critical parameters for both Cycle 6 and reference analysis are listed as follows.

		Reference Analysis	Cycle 6
Minimum MTC (pcm/0C)		0.0	-1.9
Maximum DC (pcm/0C)		-3.60	-3.04
Note:	Maximum: Most negative; Minimum: Least negative. MTC: Moderator Temperature Coefficient. DC: Doppler Coefficient.		

As shown above, the critical parameters of Cycle 6 are bounded by the reference analysis. So re-analysis is not needed.

5. CONCLUSION

The calculated values are generally in good agreement with the measured values. All the calculated values are satisfying the stipulated criteria for each physics value. Therefore it once again shows the soundness of the design tools used for calculations of various physics parameters. Since all initiating events evaluated have relevant parameters within the limits, it is concluded that reactor will operate safely during cycle 6.

REFERNCE

[1] IAEA Safety Series No. 50-CD IAEA Safety Standards

[2] IAEA Safety Series No. 50-SG-D14 IAEA Safety Standards

[3] "Chashma Nuclear Power Plant Unit-1, Final Safety Analysis Report", SNERDI, 1998.

[4] Mohammad Khan, Mohammad Kamran Chughtai & Abid Hussain, "Cycle 6 Fuel Loading Pattern Selection for CHASNUPP Unit-1", May 2008.