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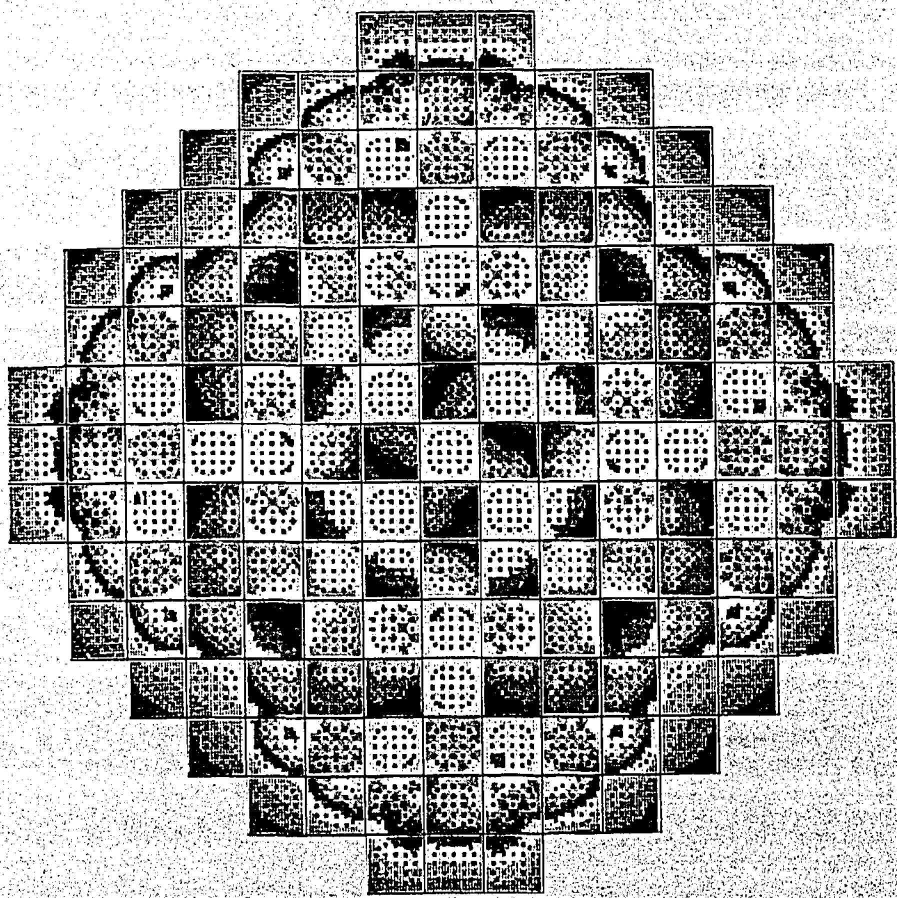
Reload Safety Evaluation Report for Yonggwang Nuclear Power Plant Unit 1 Cycle 7

June 1992



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RELOAD SAFETY EVALUATION REPORT

FOR

YONGGWANG NUCLEAR POWER PLANT

UNIT 1, CYCLE 7

June 1992

Released by :

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LWR Fuel Division

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UNIT 1, CYCLE 7

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1 INTRODUCTION AND SUMMARY

1.1 Introduction

The Yonggwang Nuclear Power Plant, Unit 1 (YGN-1) is anticipated to be refuelled with 17x17 Korean Fuel Assemblies (KOFA).

This report presents the reload safety evaluation for YGN-1, Cycle 7 and demonstrates that the reactor core being entirely composed of KOFA as described below will not adversely affect the safety of the public and the plant.

The evaluation of YGN-1, Cycle 7 was accomplished utilizing the methodology described in "Reload Transition Safety Report for YGN-1&2" (Ref./1-1/). The reload core for YGN-1, Cycle 7 is entirely comprised of 17x17 KOFA. In the YGN-1 licensing documentation to KEPCO the reference safety evaluation was provided for the operation of a reactor core fully loaded with KOFA as well as associated proposed changes to the YGN-1 Technical Specifications.

The reload for YGN-1, Cycle 7 also introduces UO_2/Gd_2O_3 containing fuel rods. The use of fuel rods with Gd_2O_3 poisoning of the fuel has been approved as a part of the above-mentioned licensing documentation.

All of the accidents comprising the licensing bases which could potentially be affected by the reload fuel assemblies have been reviewed for the Cycle 7 core design described herein.

1.2 General Description

The reactor core for YGN-1, Cycle 7 is comprised of 157 fuel assemblies whose core loading pattern configuration, burnable absorber pattern and control and shutdown rod locations are given in Figure 1-1, Figure 1-2 and Figure 1-3, respectively. The 8 RCCA's (B_4C+Hf Tip) in control bank D (4 assemblies) and shutdown bank SC (4 assemblies) will be replaced by chrome plated full length AgInCd RCCA's during Cycle 6/7 refuelling outage. During the refuelling outage for Cycle 7, 57 Optimized Fuel Assemblies (OFA) and 7 KOFA's will be discharged and then 64 fresh KOFA's will be reloaded. Among 64 fresh KOFA's 36 fresh KOFA's will contain UO_2/Gd_2O_3 fuel rods.

A summary of Cycle 7 fuel inventory is given in Table 1-1.

Nominal core design parameters utilized for Cycle 7 are as follows :

Core Power (MWt)	2775.0
System Pressure (bar)	155.1
Core Inlet Temperature(HFP)(°C)	291.2
Thermal Design Flow (m^3/sec)	17.907
Average Linear Power Density	
without Gamma-Heating(kw/m)	17.83

1.3 Conclusion

From the evaluation presented in this report, it is concluded that the core design for YGN-1, Cycle 7 does not result in the violation of any safety limits for any accident.

This conclusion is based on the following :

1) End of Cycle 6 burnup is in the range from 11,730 to 12,530 MWD/MTU.

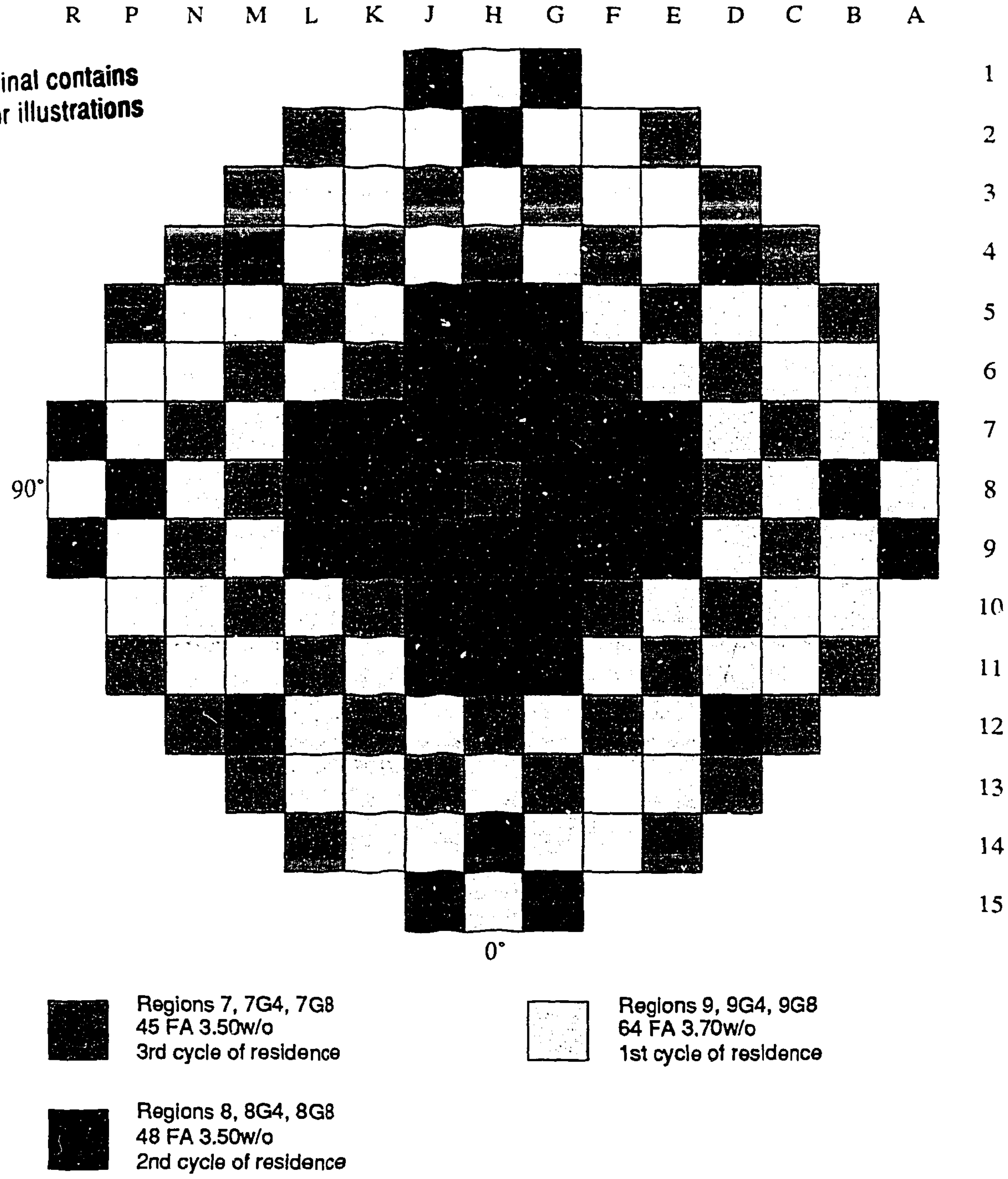
2) The burnup of Cycle 7 will not exceed the end-of full power capability. *)

3) There is adherence to the plant operating limitations for Cycle 7 as given in the YGN-1 Technical Specifications, and the proposed YGN-1, Cycle 7 changes summarized in Section 6.

*) Definition - with control rods fully withdrawn and approximately 0 to 10 ppm of residual boron at the Cycle 7 rated power condition of 2775.0 MWt.

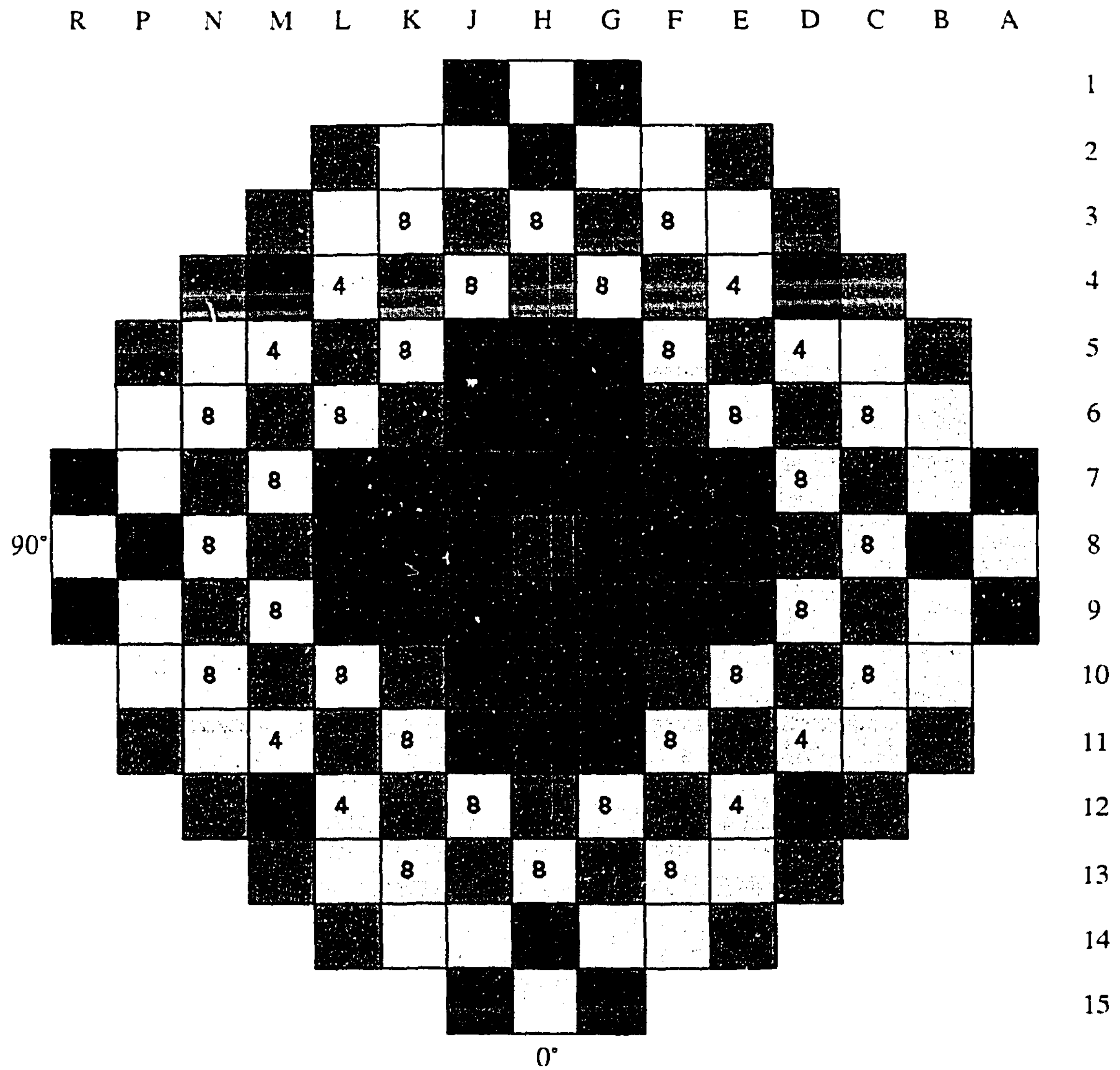
Table 1-1 Fuel Assembly Design Parameters
[Yonggwang-1, Cycle 7]

Region	7	7G4	7G8	8	8G4	8G8	9	9G4	9G8
Enrichment (w/o U235)	3.50	3.48	3.46	3.50	3.48	3.45	3.70	3.67	3.65
Number of Assemblies	32	5	8	28	12	8	28	8	28
Approximate Burnup at BOC (GWD/MTU)	27.1	29.5	29.1	13.2	15.4	15.3	0.0	0.0	0.0



original contains color illustrations

Figure 1-1 Core Loading Pattern (Regions)
[Yonggwang-1, Cycle 7]

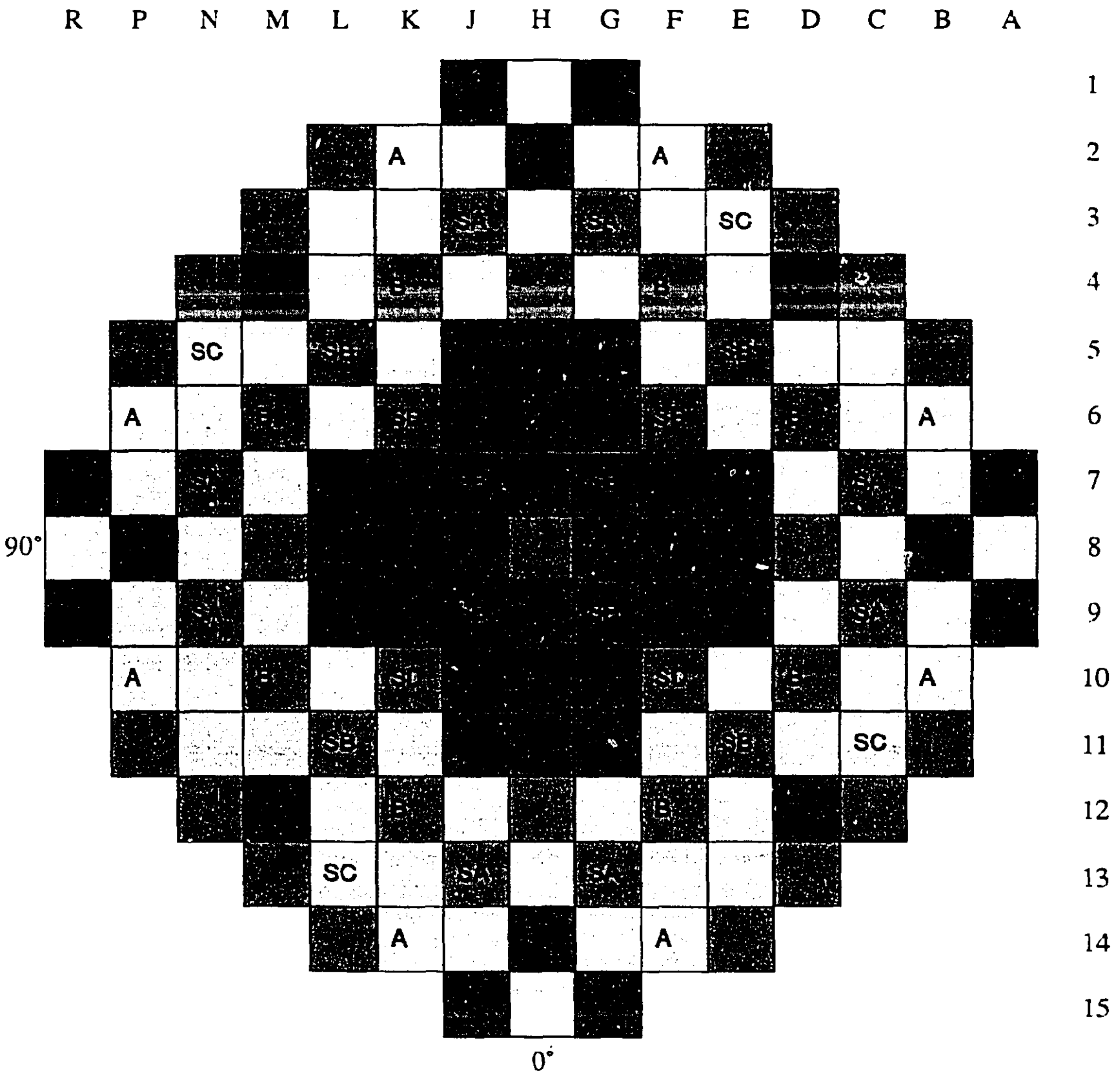


BURNABLE ABSORBER PATTERN

NUMBERS INDICATE NUMBER OF GD-POISONED FUEL RODS IN FEED ASSEMBLY

TOTAL : 256 Gd-RODS (9.0w/o Gd_2O_3 - 1.8 w/o U235)

Figure 1-2 Burnable Absorber Pattern (Regions 9G4 and 9G8)
[Yonggwang-1, Cycle 7]



CONTROL BANK	NUMBER OF RODS	MATERIAL	SHUTDOWN BANK	NUMBER OF RODS	MATERIAL
D	4	AgInCd	SD	4	AgInCd
C	8	AgInCd	SC	4	AgInCd
B	8	AgInCd	SB	8	AgInCd
A	8	AgInCd	SA	8	AgInCd
TOTAL	28		TOTAL	24	

Figure 1-3 Control and Shutdown Rod Locations
[Yonggwang-1, Cycle 7]

2 MECHANICAL DESIGN

2.1 Introduction

The task of fuel mechanical design consists basically of the following items :

- an evaluation of the geometrical and mechanical compatibility of the existing fuel assemblies and the new fuel assemblies,
- checking of the fuel rod and fuel assembly integrity under the authorized operating conditions,
- generation of the fuel rod mechanical design (FMD)-related safety parameters for the LOCA/Non-LOCA analyses.

The 17x17 fuel assemblies loaded in YGN-1, Cycle 7 consist of 157 KOFA's. The KOFA's loaded in Cycle 7 contain the following design features :

- 6 intermediate Zircaloy spacer grids, with Inconel springs and without mixing vanes, in the active region,
- Inconel spacer grids in the lowest and uppermost region,
- removable bottom and top end pieces in order to make the fuel assembly repairable.

In order to confirm the fuel rod integrity under the authorized operating conditions, the fuel rod design criteria are to be checked for every reload. The results of evaluation and analysis for the fuel rod integrity are described in Section 2.2.

To confirm the fuel assembly integrity under the authorized operating conditions, all mechanical design data of the fuel assemblies are to be checked and evaluated. The results of design data check and evaluation are described in Section 2.3.

To confirm that the current limit values assumed in the reference safety analysis are still bounding, the FMD-related key parameters are to be checked for every reload. The results of the FMD-related key parameters are described in Section 2.4.

2.2 Fuel Rod Mechanical Design Analysis

In the fuel rod mechanical design, it should be demonstrated that the fuel rod integrity can be maintained over the whole residence time under the authorized operating conditions. The key parameters for the fuel rod integrity check are fuel rod and reactor thermal hydraulic data, and rod power history. Compared with the previous cycle, fuel rod and reactor thermal hydraulic data are same and only the rod power history is changed.

Table 2-1 summarizes the results of the fuel rod design calculations. From this table, it can be seen that the design criteria considered are met under the authorized operating conditions. The other design criteria such as elastic buckling, plastic deformation, cladding stresses and alternating bending stress, which depend only on the fuel rod and reactor thermal hydraulic data are safely met since the fuel rod and reactor thermal hydraulic data of Cycle 7 are the same as those of the previous cycle and there were much safety margins for those criteria, as explained in the previous cycle.

2.3 Fuel Assembly Mechanical Integrity

In the fuel assembly mechanical design, it should be demonstrated that the fuel assembly integrity can be maintained over the whole residence time under the authorized operating conditions. In order to confirm the integrity of KOFA loaded in YGN-1, Cycle 7, all mechanical design data check and evaluation should be performed to find out whether further design analyses for the specific cycle are necessary or not.

The key fuel assembly-related design data for YGN-1, Cycle 7 were checked and evaluated. As the result of the data check and evaluation, the KOFA design data of YGN-1, Cycle 7 KOFA are not so deviated from the previous standard design data that the re-analysis of the fuel assembly integrity is not necessary. The results of the design data check and evaluation indicate that the KOFA integrity loaded in YGN-1, Cycle 7 is maintained.

2.4 FMD-related Safety Parameters Generation

The FMD-related safety parameters which characterize the conditions of a fuel rod at the onset of/during the accidents such as LOCA and Non-LOCA are required for the evaluation of the fuel rod behavior during the accidents considered. The FMD-related key safety parameters for the LOCA analysis are fuel stored energy and internal gas pressure while those for the Non-LOCA analysis are maximum and minimum gap conductances.

Table 2-2 summarizes the results of the FMD-related key safety parameters for YGN-1, Cycle 7. From this table, it can be seen that the current limit values of fuel stored energy, internal gas pressure and maximum gap conductance are greater than the reload values while the current limit value of minimum gap conductance is smaller than the reload value. It is, therefore, proved that the current limit values are still bounding for YGN-1, Cycle 7.

2.5 Conclusion

The fuel rod and fuel assembly integrity of the YGN-1, Cycle 7 were checked. The results show that all the design criteria related to the fuel rod and fuel assembly integrity are satisfied under the authorized operating conditions. In addition, the FMD-related key safety parameters for YGN-1, Cycle 7 were generated. Comparison of the calculated reload values with the current limit ones shows that the current limit values are still bounding for YGN-1, Cycle 7.

Table 2-1 Summary of the Fuel Rod Design Analyses
[Yonggwang-1, Cycle 7]

ANALYSIS	PARAMETER		DESIGN CRITERIA	RESULT IN WORST CASE
HOT CHANNEL	Fuel Centerline Temperature	UO_2	No Melting (< 2792 °C)	2580 °C
		UO_2/Gd_2O_3	No Melting (< 2453 °C)	2270 °C
	Total Tangential Strain		≤ 1 %	0.78 %
LONG TERM	Rod Internal Pressure		Cladding Non-Lift-Off	126.0 bar ($<P_{coolant}$)
	Equivalent Plastic Strain		≤ 2.5 %	1.87 %
	Oxide Layer Thickness		≤ 100 μm	81.7 μm
	Hydrogen Content in the Cladding		≤ 500 ppm	< 481 ppm(*)

(*) The hydrogen content of 481 ppm was calculated at the oxide layer thickness design limit of 100 μm. Note that hydrogen content of the cladding for 17x17 KOFA is still below 500 ppm when the oxide layer thickness of the cladding reaches the design limit (100 μm).

Table 2-2 Summary of the FMD-related Key Safety Parameters
[Yonggwang-1, Cycle 7]

a) Key Parameters for the LOCA Analysis

LHGR [W/cm]	FUEL STORED ENERGY [kJ/kg]		INTERNAL GAS PRESSURE [bar]	
	Current Limit	Reload (Cycle 7)	Current Limit	Reload (Cycle 7)
100	140	130	NA (*)	NA
200	209	190	NA	NA
300	279	249	NA	NA
400	350	310	NA	NA
434.5	372	330	75.0	74.0

(*) : NA means "Not Applicable"

b) Key Parameters for the Non-LOCA Analysis

LHGR [W/cm]	MIN. GAP CONDUCTANCE [W/K/m^2]		MAX. GAP CONDUCTANCE [W/K/m^2]	
	Current Limit	Reload (Cycle 7)	Current Limit	Reload (Cycle 7)
550	9100	9703	47000	41420

3 NUCLEAR DESIGN

3.1 Introduction

This section presents the results of the nuclear design analysis performed for Cycle 7 of YGN-1 to demonstrate that the core reload with KOFA will satisfy the nuclear design related safety criteria. In addition, it is checked whether the reload values of mostly relevant nuclear key safety parameters for Cycle 7 are within the bounds used in the reference accident analyses presented in the RTSR. The evaluation was accomplished utilizing the methodology described in the RTSR.

3.2 Power Capability

For the evaluation performed to address overpower concerns, the maximum linear heat generation rate limit of 591 W/cm corresponding to the fuel centerline temperature limit can be accommodated with margin in the Cycle 7 core using the methodology described in the RTSR.

The current LOCA F_Q limit is 2.30 under normal operating conditions as defined in the Technical Specifications and analyzed in the RTSR. For Cycle 7 the predicted F_Q at any core conditions including the most unfavorable situations permissible within the Technical Specifications is less than the limit at all core elevations. Therefore, frequent axial power distribution monitoring is not required.

3.3 Safety Parameter Generation

A core reload can typically affect accident analysis input parameters in the following areas : core kinetics characteristics, control rod worths and core peaking factors. Cycle 7 parameters in each of these three areas were determined as discussed below to ascertain whether the conclusions of the reference analysis, i.e., the RTSR safety analyses remained valid or new accident evaluations/analyses were required.

3.3.1 Kinetics Parameters

The values of all relevant kinetics parameters were determined for Cycle 7 and compared to the current limits used in the RTSR safety analyses. Table 3-1 summarizes some of the nuclear key safety parameters checked. The result of this overall check is that all relevant kinetics parameters are within the bounds of their current limits.

3.3.2 Control Rod Worths

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths and trip reactivity. Table 3-1 indicates that the Cycle 7 value for the minimum trip reactivity satisfies the current limit.

Table 3-2 provides the control rod worths and shutdown margin at the most limiting conditions during the cycle. The available shutdown margin for Cycle 7 exceeds the minimum required.

The Cycle 7 values of the differential rod worths satisfy the current limits which were used in the reference accident analysis of the bank withdrawal from a subcritical condition. The ejected rod worths for Cycle 7 are below the current limits in the corresponding RTSR accident analysis.

The Cycle 7 values of the D-bank characteristics from HFP rod insertion limit to

ARO which are relevant for the analysis of the rod drop accident satisfy the current limit values.

3.3.3 Core Peaking Factors

The peaking factors for the rod drop events and the statically misaligned RCCA accident have been evaluated for Cycle 7 and are below the limits used in the reference safety analyses. Thus, it is concluded that the DNB design basis is met. For the single RCCA withdrawal accident the fuel rod failure fraction was determined to be below the applicable limit of 5 percent. The evaluation of the maximum linear heat generation rate after rod ejection indicates that the Cycle 7 values satisfy the current limits applied in the reference analysis of the rod ejection accident.

3.4 Conclusion

The evaluation of core power capability indicates that YGN-1 can be safely operated at 100 percent of rated power during Cycle 7. During overpower transients the maximum linear heat generation rate will not exceed the current limit. The evaluation of nuclear key safety parameters for Cycle 7 also shows that the current reload design bases remain satisfied and there is no impact on the reference safety analyses.

Table 3-1 Kinetics Characteristics [Yonggwang-1, Cycle 7]

Parameter	Current Limit	Value in Cycle 7
Moderator Temperature Coefficient (pcm/^{o}C)		
most positive at HZP	+ 9.0	+ 4.4
least negative at HFP	0.0	- 13.2
most negative	- 71.3	- 60.3
Doppler Temperature Coefficient (pcm/^{o}C)		
least negative	- 2.84	- 3.49
most negative	- 5.98	- 4.94
Delayed Neutron Fraction β_{eff}(%)		
maximum	0.68	0.64
minimum	0.49	0.53
Prompt Neutron Lifetime (μsec)		
maximum	28	25
minimum	17	18
Minimum Trip Reactivity (pcm)	5760	6070

Table 3-2 Summary of Control Rod Reactivity Requirements and Shutdown Margin [Yonggwang-1, Cycle 7]

Requirements	BOC ($\% \Delta\rho$)	EOC ($\% \Delta\rho$)
Total Power Defect (*) (Doppler + Variable T_m + Redistribution)	1.60	3.03
Rod Insertion Allowance	0.50	0.50
Total Control Bank Requirement (1)	2.10	3.53
Control Rod Worth (HZP)		
Control Rods Inserted Less Most Reactive Stuck Rod	7.30	7.74
Less 10 percent (2)	6.57	6.97
Shutdown Margin		
Calculated Margin [(2)-(1)]	4.47	3.44
Minimum Required Shutdown Margin	1.77	1.77

(*) includes 10% uncertainty

4 THERMAL AND HYDRAULIC DESIGN

4.1 Introduction

This section describes the thermal hydraulic design considerations for YGN-1, Cycle 7. The description consists of the DNB power capability and of the f(ΔI) function verification.

The cycle specific operating conditions given in Section 1.2 are the same as those used in the design of previous cycle. Therefore, the general thermal hydraulic design data used for previous cycle are still valid for Cycle 7.

4.2 DNB Power Capability

The design system limit DNBR's for Cycle 7 are 1.47 for thimble channel and 1.49 for typical channel, respectively. And design safety analysis limit DNBR's are 1.58 for thimble channel and 1.60 for typical channel, respectively. These DNB margins cover all DNBR penalties sufficiently for Cycle 7.

The core thermal limits which form the basis for the overpower/overtemperature ΔT trip system setpoints are determined from plant operating conditions, safety analysis limit DNBR and design radial and axial power distribution. Since there are no changes in basic design parameters due to Cycle 7 reload, the core thermal limits and overpower/overtemperature ΔT trip functions as defined in the RTSR (Ref./1-1/) remain valid for Cycle 7 operation.

4.3 f(ΔI) Function Verification

The f(ΔI) function for Cycle 7 was changed as presented in Ref./4-1/ to enhance operational flexibility with respect to the axial power shape control. Justification for these changes are given in Ref./4-2/. The validity of changed f(ΔI) function was verified by using cycle specific axial power shapes generated from the loading pattern of Cycle 7. As shown in Figure 4-1, the result indicates that this f(ΔI) function remains valid for Cycle 7.

4.4 Conclusion

No significant variations in thermal margins will result from Cycle 7 reload. Therefore, the design basis of Ref./1-1/ is maintained for Cycle 7 power operation.

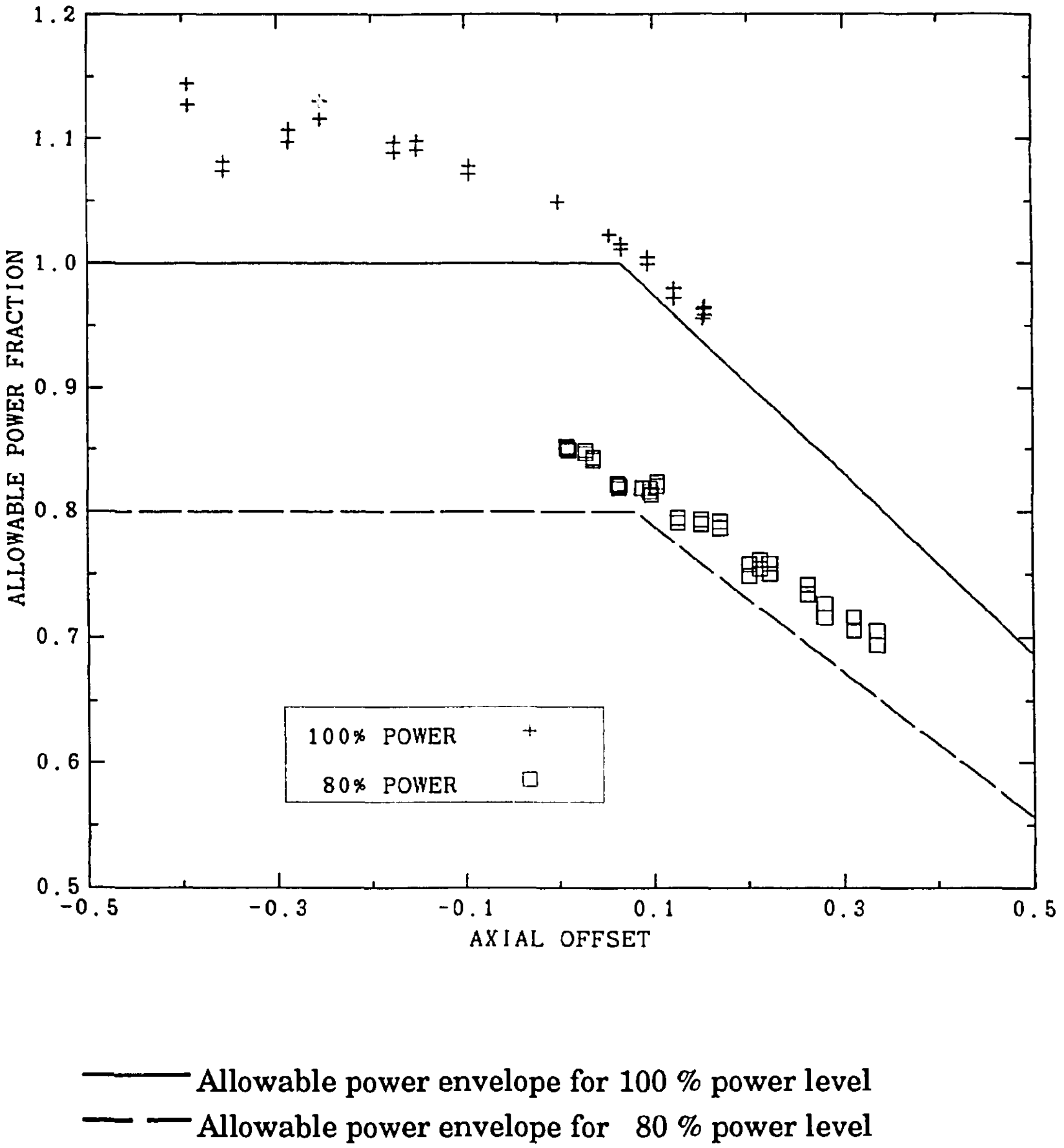


Figure 4-1 Verification of the Allowable Power Envelope of the Existing $f_1(\Delta I)$ Function

5 ACCIDENT EVALUATION

5.1 Introduction

This section justifies the compatibility of Cycle 7 specific conditions with the conclusions of the reference document for reload cores of KOFA, including transition cores, that is the RTSR (Ref./1-1/). In conclusion, it is demonstrated that the reload core for Cycle 7 does not affect the safety of the plant and that the conclusions of the RTSR remain valid.

5.2 Safety Parameters Evaluation

The effects of the reload core on the design basis accidents were evaluated by examining the input parameters of safety analysis such as kinetics parameters, control rod worths, core peaking factors and the other parameters specific to the accidents.

5.2.1 Kinetics Parameters

The core kinetics parameters of Cycle 7 which influence the postulated licensing accidents showed that all values remain within the boundings of the analysis limits.

5.2.2 Control Rod Worths

Changes in control rod worth may affect shutdown margin, ejected rod worth and trip reactivity. All the control rod worths for Cycle 7 remain in the boundings of the RTSR limits.

5.2.3 Core Peaking Factors

Evaluation of the Cycle 7 peaking factors for all accidents sensitive to peaking factors shows that the DNBR is maintained above the appropriate safety analysis limit DNBR.

5.2.4 Parameters for Specific Accident

All the safety parameters related to the specific accidents do not exceed the limit values.

5.2.5 Technical Specification Parameters

The fully withdrawn position of control banks was changed from 228 to 231 step. Even though this change causes the slight increase in D-bank worth from the RIL to ARO, the worth is still within the limit value. Therefore, the relevant plant parameters are maintained within the allowable limits and thus the conclusions of the RTSR are still valid.

5.3 Accident Re-analysis/Evaluation

Since there is no violation of the boundings of parameters described in Section 5.2, the safety analyses given in the RTSR are effective and thus no cycle specific investigations have to be performed.

5.4 Conclusion

Non-LOCA Accidents :

For the Cycle 7, all the safety parameters were evaluated to ensure the safety of the reload core and those were found to be within the boundings applied for the analysis for the basic document (RTSR). The change of Technical Specifications described in the Section 5.2.5 does not affect the safety of the reload core.

LOCA Accidents :

For LOCA a comparison of Cycle 7 relevant key safety parameters referring to LOCA results of previous cycle RSE showed that all bounding values used for the calculations were not violated and they remain therefore valid. With respect to LOCA accidents the conclusions of the basic document RTSR (Ref./1-1/) remain valid.

6 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The Technical Specifications with respect to f(ΔI) function for Cycle 7 were changed as presented in Ref./4-1/ based on the Ref./4-2/. Actual Technical Specification changes for Cycle 7 operation are summarized in Table 6-1.

Table 6-1 Summary of Actual Changes of Technical Specifications for Cycle 7

Page	Section	Description	Changes
16.2.6	Table 16.2.2-1 (4 of 6)	Setpoint Constant	between - 50% and 5%
16.2.6	Table 16.2.2-1 (4 of 6)	Setpoint Constant	exceeds - 50%
16.2.6	Table 16.2.2-1 (5 of 6)	Setpoint Constant	exceeds 5%
16.2.6	Table 16.2.2-1 (5 of 6)	Setpoint Constant	reduced by 1.65%

7 PEAKING FACTOR LIMIT REPORT

This section describes a core operating limit that is applicable only to Cycle 7. The Radial Peaking Factor Limit is provided in accordance with subparagraph 16.6.9.1.9 of the YGN-1 Technical Specifications.

The F_{XY} limits for RATED THERMAL POWER within specific core planes shall be :

1. F_{XY} for all core planes containing bank "D" control rods :

   $F_{XY} \leq 1.83$

2. F_{XY} for all unrodded core planes :

   $F_{XY} \leq 1.70$ up to 50% of core height

   $F_{XY} \leq 1.74$ above 50%

These $F_{xy}(Z)$ limits were used to confirm that the total peaking factor $F_Q(Z)$ will be limited to the following values specified in the Technical Specifications:

$$F_Q(Z) \leq \left[\frac{2.30}{P}\right][K(Z)] \quad \text{for} \quad P > 0.5 \quad \text{and ,}$$

$$F_Q(Z) \leq [\,4.60\,][K(Z)] \quad \text{for} \quad P \leq 0.5$$

where K(Z) represents the normalized LOCA F_Q limit versus axial core height.

8 REFERENCES

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TR-TH-YG9-91001E, 1991.

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Abstract (About 300 Words)

This report presents the reload safety evaluation for YGN-1, Cycle 7 and demonstrates that the reactor core being entirely composed of KOFA as discribed below will not adversely affect the safety of the public and the plant.

All of the accidents comprising the licensing bases which could potentially be affected by the reload fuel assemblies have been reviewed for the Cycle 7 core and results are described in this report.

Subject Keywords (About 10 Words): Yonggwang Unit 1, Cycle 7, reload, safety, accident analysis, key s[illegible] parameters, safety analysis

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