RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
Date of RAI Issued:	**06/18/2015**

Question No. 11

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(v) provides guidance stating that the fuel failure criteria should address excessive fuel enthalpy.

This fuel failure criterion is addressed on Page 3-9 of APR1400-F-M-TR-13001-P for the PLUS7 fuel design. Section 3.4.4 of the topical report states that the code used to analyze this fuel failure mechanism (FATES3B) over predicts fuel thermal conductivity at high burnup. Therefore, it will also under predict fuel enthalpy. The staff notes that Section 3.4.4 provides a qualitative argument to state that the effects of burnup dependence of TCD are bounded by the reduced power capabilities at higher burnups. This raises concerns from the staff on the ability of the excessive fuel enthalpy analysis to demonstrate compliance with the excessive fuel enthalpy SAFDL given all core loading options available.

Please include a discussion, supported by analysis, within Section 3.2.10 and/or 3.4.4 of APR1400-F-M-TR-13001-P regarding the impacts of the fuel enthalpy under prediction and how excessive fuel enthalpy is precluded.

Response

A new section 3.4.5 has been added to discuss the impacts of fuel enthalpy.

“The impacts of TCD result in increasing the fuel enthalpy and the fuel centerline temperature. The impact on the fuel enthalpy due to TCD would be negligible for fresh fuel even though the maximum power peaking factors exist in the low burnup fuel in the APR1400 reload core designs. For the high burnup fuel which is affected by TCD, the fuel enthalpy and the fuel centerline temperature increase due to TCD are not significant because of the peaking factor

burndown effect at higher burnups. The detailed evaluation was performed for the impacts of TCD and submitted in TCD Technical Report in Reference 3-14."

TCD Technical Report APR1400-F-A-NR-14002-P has been added as a reference and a pointer to it has been placed in the introduction of Section 3.4.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

PLUS7 fuel design topical report (APR1400-F-M-TR-13001-NP) will be revised as indicated on the attached markups.

As shown in Figure 3-2, the predicted values are much higher than those of measured values for both H614 and H605 PLUS7 fuel assemblies. In addition, the means values and standard deviations of M/P are summarized in Table 3-9 for H615 and H605 assemblies.

- Comparison of operating conditions of OPR1000, Westinghouse Type Plants and APR1400

As explained in previous section, oxide thickness data to use the development of corrosion model multiplier and verification were measured at Ulchin unit 3, Yonggwang unit 2, Yonggwang unit 4 and Yonggwang unit 5. However, there are no available APR1400 plant specific corrosion data because the APR1400 plant was not started its first commercial operation yet.

It is, therefore, necessary to compare the operating conditions of Ulchin unit 3, Yonggwang unit 4 Yonggwang unit 4 with those of APR1400 because the corrosion buildup on cladding material is mainly dependent on operating conditions in terms of coolant temperatures, mass flow rate, lithium concentration and core average power.

Table 3-10 shows the operating conditions of APR1400 and OPR1000 (Yonggwang unit 4 and Ulchin unit 3) as well as Westinghouse type plant of Yonggwang unit 2. As shown in Table 3-10, the coolant inlet temperature and outlet temperature of APR1400 plant are less than those of OPR1000 and Yonggwang unit 2 as well as the core average coolant mass flow rate of APR1400 plant is well within the range of OPR1000 and Yonggwang unit 2. In addition, the allowable maximum lithium concentration of APR1400 plant is the same as those of OPR1000 and Yonggwang unit 2. On the other hand, the core average linear heat rate of APR1400 plant is about four percent higher than that of OPR1000. However, it is expected that four percent increase of core average power does not give a significant effect on oxide buildup of cladding tube. Therefore, the applicability of PAD code with increased corrosion multiplier to PLUS7 fuel in APR1400 for corrosion evaluation was confirmed.

3.4 Impact of TCD on Fuel Rod Design Criteria

FATES3B does not explicitly model fuel thermal conductivity degradation (TCD) with burnup. FATES3B uses the burnup independent thermal conductivity of Lyons correlation. Compared with the thermal conductivity model with TCD effect, the Lyons model produces a relatively less conservative temperature distribution within fuel pellet.

Many of cladding-related criteria, such as cladding corrosion and hydrogen pickup, cladding collapse and fuel rod growth are not affected by TCD. Cladding temperature is not affected since the heat flux is not changed by TCD, so cladding corrosion and hydrogen criteria are unaffected. Fuel densification is not also affected by TCD, so cladding collapse criterion is not impacted by TCD. Fast neutron fluence does not change due to TCD, so fuel rod growth criterion is also not impacted by TCD.

The evaluations for the fuel rod design criteria that are affected by TCD are described as follows. The detailed evaluation results ~~will be provided in TCD Technical Report which is planned to submit to NRC.~~ were described in TCD Technical Report in Reference 3-14.

3.4.1 Cladding Stress

As described in Section 3.2.1, KNF cladding stress criterion is established to prevent fuel damage from the excessive primary stress which results from the pressure difference between rod internal pressure and system pressure.

Considering the amount of potential increase in rod internal pressure by TCD impact and design margin to the limit, it is judged that the cladding stress criteria are still met with consideration of TCD.

3.4.2 Cladding Strain and Fatigue

The cladding strain is affected by TCD due to the increased fuel thermal expansion. However, the increased thermal expansion can be offset with available design margin to the cladding strain and fatigue limits and conservatism in the input variables such as power history and assumed rod internal pressure. Therefore, cladding strain and fatigue criteria are still satisfied with consideration of TCD.

3.4.3 Fuel Rod Internal Pressure

The effect of increased fuel temperature due to TCD on fission gas release is inherently accounted for in the current performance code, FATES3B (References 3-1 through 3-4) because the model was calibrated to measured data for a full range of fuel rod burnup and operating conditions. Additionally, conservatism is considered in the original FATES3B calibration process and in fuel rod design analysis. However, the rod internal pressure may still increase with TCD due to the increased fuel thermal expansion, which reduces the total fuel rod void volume.

Evaluations show that the reduction of void volume due to increased thermal expansion is not significant in PLUS7 fuel rod design. In addition, the rod internal pressure limit calculation is inherently conservative in that actual gap reopening is predicted to occur at higher pressures.

In conclusion, the increased rod internal pressure can be offset with available design margin to the rod internal pressure limits. Therefore, rod internal pressure criteria are still satisfied considering the effects of TCD.

3.4.4 Overheating of Fuel Pellets

The power to melt limit depends on fuel burnup but the reduced power capability with burnup, which is caused by the depletion of the fissile material in the fuel and the buildup of fission products, offsets the TCD impact. Therefore, it is judged that there will be no safety concerns due to TCD. However, the power to melt values with burnup considering the impact of TCD are calculated and will be provided in the TCD Technical Report which will be submitted to NRC.

In summary, KNF fuel rod design criteria have been reviewed with respect to the potential impacts of TCD, and it is concluded that TCD can be accommodated such that approved fuel rod design criteria will remain satisfied for current fuel rod designs.

3.4.5 Impacts of Fuel Enthalpy

The impacts of TCD result in increasing the fuel enthalpy and the fuel centerline temperature. The impact on the fuel enthalpy due to TCD would be negligible for fresh fuel even though the maximum power peaking factors exist in the low burnup fuel in the APR1400 reload core designs. For the high burnup fuel which is affected by TCD, the fuel enthalpy and the fuel centerline temperature increase due to TCD are not significant because of the peaking factor burndown effect at higher burnups. The detailed evaluation was performed for the impacts of TCD and submitted in TCD Technical Report in Reference 3-14.

3.5 Conclusion

The PLUS7 fuel rod design is verified to maintain the rod integrity up to rod average burnup of 60,000 MWD/MTU based on the thermal performance and mechanical integrity evaluation results

6. REFERENCES

2-1 CENPD-178-P, Revision 1-P, “Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading,” Combustion Engineering Inc., August 1981.

3-1 CENPD-139-P-A, “C-E Fuel Evaluation Model Topical Report,” Combustion Engineering Inc., July 1974.

3-2 CENDP-139, Supplement 1, Revision 01, ““C-E Fuel Evaluation Model Topical Report,” Combustion Engineering Inc., July 1974.

3-3 CEN-161(B)-P-A, “Improvements to Fuel Evaluation Model,” August 1989.

3-4 CEN-161(B)-P Supplement 1-P-A, “Improvements to Fuel Evaluation Model,” January 1992.

3-5 CENPD-404-P-A, “Implementation of ZIRLOTM Cladding Material in CE Nuclear Power Fuel Assembly Designs,” November 2001.

3-6 CEN-372-P-A, “Fuel Rod Maximum Allowable Gas Pressure,” May 1990.

3-7 CENPD-275-P, Revision 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers,” May 1998.

3-8 CENPD-275-P, Revision 1-P, Supplement, 1-P-A, “C-E Methodology for Core Designs Containing Gadolinia-Urania Absorbers,” April 1999.

3-9 CEN-193(B), Supplement 2-P, Partial Response to NRC Questions on CEN-161(B)-P, “Improvements to Fuel Evaluation Model,” March 21, 1982.

3-10 WCAP-16500-P-A, Rev.0 “CE 16x16 Next Generation Fuel Core Reference Report,” August 2007.

3-11 WCAP-15063-P-A, Rev.1, with Errata, “Westinghouse Improved Performance Analysis and Design Model (PAD4.0),” July 2000.

3-12 CENPD-187-P-A, CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Combustion Engineering, Inc., April 1976 ; Supplement 1-P-A, June 1977.

3-13 EPRI NP-3966-CCM, “CEPAN Method of Analyzing Creep Collapse of Oval Cladding-Volume 5: Evaluation of Interpellet Gap Formation and Cladding Collapse in Modern PWR Fuel Rods,” Combustion Engineering, Inc., April 1985.

3-14 APR1400-F-A-NR-14002-P, "The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analysis," KEPCO NF, September 2014.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

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Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
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Question No. 12

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(iv) provides guidance in regards to GDC 10 by stating that overheating of fuel pellets should be avoided by preventing centerline melting. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and should account for the effects of burnup and composition on the melting point.

Sections 3.4.4 of the PLUS7 fuel design topical report (APR1400-F-M-TR-13001-P) and 2.2.2 of the thermal conductivity degradation (TCD) technical report (APR1400-F-A-NR-13002-P) discuss overheating of the fuel pellets. The PLUS7 fuel design includes UO_2-Gd_2O_3 pellets. The impact of the UO_2-Gd_2O_3 pellet composition on the overheating of fuel pellets analysis is not discussed in either Section 3.4.4 of APR1400-F-M-TR-13001-P or Section 2.2.2 of APR1400-F-A-NR-13002-P. UO_2-Gd_2O_3 has a lower melting temperature and lower thermal conductivity than UO_2 which has caused the staff to question the ability of the fuel centerline melting analysis provided to demonstrate compliance with GDC 10.

Update the topical report, as applicable, to address fuel pellet overheating considering UO_2-Gd_2O_3 to ensure that no fuel centerline melting occurs for all fuel compositions and normal operation/AOO conditions.

Response

Figure 12-1 shows the normalized radial power fall-off curves for UO_2 and Gd_2O_3-UO_2 rods. As shown in Figure 12-1, the highest radial powers for Gd_2O_3-UO_2 rods are lower than those for UO_2 rods over all burnup ranges. Also, the radial powers decrease as burnup increases due to the reduced power capability caused by the depletion of the fissile material in the fuel and the buildup of fission products.

Figure 12-2 shows the calculated PTM (Power-to-Melt) values using the FRAPCON code for both UO_2 and Gd_2O_3-UO_2 rods, which are represented by black and red solid lines, respectively. Figure 12-2 also shows the highest attainable powers for UO_2 and Gd_2O_3-UO_2 rods in black and red dotted lines, respectively.

In Figure 12-2, each maximum attainable power of UO_2 and Gd_2O_3-UO_2 rods can be obtained from radial fall-off values, namely from the normalized radial power in Figure 12-1. Since the maximum attainable power for both UO_2 and Gd_2O_3-UO_2 rods is limited up to [ ]TS which is a SAFDL value for APR1400, it can be defined that the maximum attainable power of[ ]TS is only available with a normalized UO_2 rod radial power of [ ]TS. In addition, attainable power after about [ ]TS in Figure 12-2 can be interpreted from the proportional decrease of normalized radial power fall-off curve in Figure 12-1.

Based on the same reasoning, the normalized radial power ratio of UO_2 and Gd_2O_3-UO_2 rods over all burnup ranges in Figure 12-1 was considered and the attainable power of Gd_2O_3-UO_2 rods in Figure 12-2 was derived.

As shown in Figure 12-2, the PTM values are well above the attainable powers for Gd_2O_3-UO_2 rods. Therefore, it can be concluded that there will be no melting of Gd_2O_3-UO_2 rods even when the rod power for UO_2 reaches the SAFDL value of [ ]TS.

In conclusion, the lower power of Gd_2O_3-UO_2 burnable absorber fuel rods compared to UO_2 fuel rods allows for the UO_2 fuel rods to be the limiting case for centerline melt.

The text in Section 3.4.4 of the PLUS7 fuel design topical report (APR1400-F-M-TR-13001-P) will be revised to provide a discussion on Gd_2O_3-UO_2 melting.

TS

Figure 12-1 Rod Power Histories Used for PLUS7 Fuel

TS

Figure 12-2 Power to Melt and Maximum Attainable Powers for UO_2 and Gd_2O_3-UO_2 Rod as a Function of Burnup

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

PLUS7 fuel design topical report (APR1400-F-M-TR-13001-NP) will be revised as indicated on the attached markups in response to Question 13.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
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Question No. 13

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(iv) provides guidance in regards to GDC 10 by that overheating of fuel pellets should be avoided by preventing centerline melting. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and should account for the effects of burnup and composition on the melting point.

Section 2.2.2 of APR1400-F-A-NR-13002-P provides a scoping analysis using FRAPCON3.4 to investigate the impacts of burnup dependent TCD on the fuel centerline temperature analysis. The staff has concerns regarding the methodology and results presented in that the methodology is different than what is presented in APR1400-F-M-13001-P and the results do not support the stated conclusions. This in turn has caused the staff to question the ability of the fuel centerline melting analysis provided to demonstrate compliance with GDC 10.

Address the following concerns, as appropriate, to demonstrate compliance with GDC 10:

a) Provide a basis for the assumed uncertainty of 9.7% on best estimate fuel centerline temperature used to calculate the conservative power to melt results.

b) Section 2.2.2 of APR1400-F-A-NR-13002-P provides a SAFDL of 20 kW/ft. Figure 2-10 shows that the fuel would melt above 30 GWd/MTU assuming a conservative analysis. Update the topical report, as applicable, to ensure that the linear heat rate SAFDL is conservatively met.

c) Provide a description and update the topical report, as necessary, to explain how the melt analysis will be performed on a cycle specific basis since FRAPCON-3.4 was used to perform the scoping analysis, or update the topical report methodology to not require FRAPCON-3.4 to confirm compliance with GDC 10.

Response

a) The assumed uncertainty of 9.7% in Section 2.2.2 of APR1400-F-A-NR-14002-P is a typo and the assumed uncertainty should be [ ]^{TS}. The value comes from measured and predicted centerline temperature comparison for the UO_2 rod described in the FRAPCON manual [Reference 13-1]. The measured data is within [ ]^{TS} level of lower and upper predicted limit in Figure 13-1. Therefore, the uncertainty of [ ]^{TS} was conservatively determined by assuming a standard deviation of [ ]^{TS} and multiplying by [ ]^{TS}.

The text in Section 2.2.2 of APR1400-F-A-NR-14002-P will be revised to reflect the correction of uncertainty from 9.7% to [ ]^{TS}.

TS

Figure 13-1 Measured and Predicted Centerline Temperature for the UO_2 Assessment Cases throughout Life

b) Divided by a SAFDL of 20 kW/ft, the conservative Power-To-Melt (PTM) values are normalized and plotted in Figure 13-2. As shown in Figure 13-2, the normalized Power-To-Melt (PTM) values indicate a gradual decline below a [ ]^{TS} kW/ft from [ ]^{TS}. The fuel will not melt because the descent ratio of actual fuel rod power (radial power fall-off) after 30 GWd/MTU will be much higher than that of the normalized PTM due to the reduced power capability caused by the depletion of the fissile material in the fuel and the buildup of fission products.

Furthermore, Figure 13-2 shows that the margin between Radial Power Fall-off and PTM tends to increase after 30 GWd/MTU. This analysis will be verified each cycle using the cycle specific radial power fall-off curve.

[ ]^{TS}

Figure 13-2 Normalized Power to Melt and Radial Power Fall-off Curve vs. Burnup

c) A cycle specific radial fall-off curve based on the core loading pattern will be generated and bounded by conservative radial fall-off curve limits used in the FRAPCON analysis. This will ensure that the linear heat rate [ ]TS is valid.

The text of Section 3.4.4 of APR1400-F-M-TR-13001-P will be revised to explain how the melt analysis will be performed on a cycle specific basis.

References

[13-1] [ ]TS

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

Topical Report APR1400-F-M-TR-13001-NP will be revised as indicated in Attachment 1.
Technical Report APR1400-F-A-NR-14002-NP will be revised as indicated in Attachment 2.

Considering the amount of potential increase in rod internal pressure by TCD impact and design margin to the limit, it is judged that the cladding stress criteria are still met with consideration of TCD.

3.4.2 Cladding Strain and Fatigue

The cladding strain is affected by TCD due to the increased fuel thermal expansion. However, the increased thermal expansion can be offset with available design margin to the cladding strain and fatigue limits and conservatism in the input variables such as power history and assumed rod internal pressure. Therefore, cladding strain and fatigue criteria are still satisfied with consideration of TCD.

3.4.3 Fuel Rod Internal Pressure

The effect of increased fuel temperature due to TCD on fission gas release is inherently accounted for in the current performance code, FATES3B (References 3-1 through 3-4) because the model was calibrated to measured data for a full range of fuel rod burnup and operating conditions. Additionally, conservatism is considered in the original FATES3B calibration process and in fuel rod design analysis. However, the rod internal pressure may still increase with TCD due to the increased fuel thermal expansion, which reduces the total fuel rod void volume.

Evaluations show that the reduction of void volume due to increased thermal expansion is not significant in PLUS7 fuel rod design. In addition, the rod internal pressure limit calculation is inherently conservative in that actual gap reopening is predicted to occur at higher pressures.

In conclusion, the increased rod internal pressure can be offset with available design margin to the rod internal pressure limits. Therefore, rod internal pressure criteria are still satisfied considering the effects of TCD.

3.4.4 Overheating of Fuel Pellets

~~The power to melt limit depends on fuel burnup but the reduced power capability with burnup, which is caused by the depletion of the fissile material in the fuel and the buildup of fission products, offsets the TCD impact. Therefore, it is judged that there will be no safety concerns due to TCD. However, the power to melt values with burnup considering the impact of TCD are calculated and will be provided in the TCD Technical Report which will be submitted to NRC.~~

In summary, KNF fuel rod design criteria have been reviewed with respect to the potential impacts of TCD, and it is concluded that TCD can be accommodated such that approved fuel rod design criteria will remain satisfied for current fuel rod designs.

3.5 Conclusion

The PLUS7 fuel rod design is verified to maintain the rod integrity up to rod average burnup of 60,000 MWD/MTU based on the thermal performance and mechanical integrity evaluation results using by NRC approved design codes and methodologies.

The power to melt values with burnup considering the impact of TCD were calculated and provided in the TCD Technical Report (Reference 3-14). For UO_2 rod, the power to melt is decreased below a SAFDL of 20 kW/ft above 30 GWd/MTU but the reduced power capability with burnup, which is caused by the depletion of the fissile material in the fuel and the buildup of fission products, offsets the TCD impact. For Gd_2O_3-UO_2 rod, the attainable powers are below PTM of Gd_2O_3-UO_2 rod over all burnup range even when the power for UO_2 rod is reached at a SAFDL of 20 kW/ft. Therefore, it is judged that there will be no fuel centerline-melting due to TCD for UO_2 and Gd_2O_3-UO_2 rods.

This melt analysis will be verified each cycle by comparing the conservative radial fall-off curve limits used in the analysis of Reference 3-14 with a cycle specific radial fall-off curve to ensure that the linear heat rate SAFDL of 20 kW/ft is still valid.

2.2.2 Fuel centerline temperature

The power to fuel melting decreases as fuel rod burnup increases because the fuel melting temperature decreases. The local linear powers that preclude fuel centerline melting are calculated as a function of burnup using the FRAPCON3.4 fuel performance code [6] considering the thermal conductivity degradation effect for the transient accident analysis. The results are presented in Table 2-2 and Figure 2-10. The conservative power to fuel melting is calculated while considering an uncertainty of ~~9.7~~ 9.4 % for the best estimated fuel centerline temperature. Based on the results of power to fuel melting with TCD, the linear heat rate specified acceptable fuel design limit (SAFDL) of 20.0 kW/ft was determined for the APR1400. It should be noted that an SAFDL of 20.0 kW/ft has been applied to the current APR1400 design control documents.

2.2.3 Cladding stress

The cladding stress criteria for a PLUS7 fuel rod loaded in the APR1400 are as follows [7].

- The primary tensile stress must not exceed [ ]^{TS} of the material at the applicable temperature.
- The primary compressive stress must not exceed [ ]^{TS} of the material at the applicable temperature.

It should be noted that the criteria of cladding stress are for primary stress not secondary stress. It is well known that the primary stress does not depend on the contact of the fuel pellet and cladding but on the pressure difference between the rod internal pressure and system pressure. The rod internal pressure is determined by the combination of fission gas release and total void volume. As explained in section 2.2.1, it was determined that the rod internal pressure predicted by FATES3B includes the effect of TCD.

Therefore, considering the conservatism of FATES3B with regard to the rod internal pressure and the design margin for cladding stresses, it is confirmed that the result of the cladding stress design criteria without TCD are still applicable to the results with TCD.

2.2.4 Cladding strain

The cladding strain was calculated using the convenient code FATES3B. Since the fission gas release data are provided by FATES3B, it is possible to evaluate the impact of TCD on cladding strain using codes in which the fuel thermal conductivity model, the Lyons correlation, is replaced with the modified NFI thermal conductivity model.

In order to assess the impact of TCD on the cladding strain, strain analysis was conducted using the cycle specific rod power history as well as the transient peak power as a function of rod burnup. The evaluation results show that the cladding plastic stain during normal operation and single AOO, and total strain (elastic plus plastic) during a single AOO, are [ ]^{TS}, respectively, when TCD is accounted for; however, the design criteria are still satisfied with the consideration of TCD. A summary of the evaluation results is shown in Table 2-3.

2.2.5 Cladding fatigue

The cladding cumulative fatigue damage factor is determined as the sum of the fatigue damage factors resulting from the daily load following operation, reactor trips and startup/shutdown operation. The fatigue damage factor for the daily load following operation is calculated using the same convenient code, FATES3B, as was used for the cladding strain evaluation. On the other hand, the fatigue damage factors for reactor trips and startup/ shutdown are determined by hand calculation using a simple formula.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Question No. 14

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(iv) provides guidance in regards to GDC 10 by stating that overheating of fuel pellets should be avoided by preventing centerline melting. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and should account for the effects of burnup and composition on the melting point.

Section 3.2.9 of the PLUS7 fuel design topical report, APR1400-F-M-TR-13001-P, provides the overheating of fuel pellets analysis for the APR1400 design. On Page 3-9, it is stated that the linear heat rate corresponding to the centerline melt of Gd_2O_3-UO_2 burnable absorber fuel rods is always less than that of the UO_2 fuel rods. The lower thermal conductivity of Gd_2O_3-UO_2 burnable absorber fuel rods causes the staff to question the claimed bounding nature.

Provide linear heat generation rate limits for UO_2 and Gd_2O_3-UO_2 to support the position presented in APR1400-F-M-13001-P, or revise the topical report as necessary.

Response

Figure 14-1 shows the Power-to-Melt (PTM) for both UO_2 and Gd_2O_3-UO_2 rods calculated by FATES3B code. It also shows the maximum attainable power for both UO_2 and Gd_2O_3-UO_2 rods. As mentioned in the response to Question No.12, the attainable power of Gd_2O_3-UO_2 rods in Figure 14-1 can be derived by considering the normalized radial power ratio of UO_2 and Gd_2O_3-UO_2 rods shown in Figure 12-1. As shown in Figure 14-1, the attainable powers for Gd_2O_3-UO_2 rods are well below the PTM values over all burnup ranges.

Therefore, it can be concluded that there will be no melting of Gd_2O_3-UO_2 rods even when the rod power for UO_2 rods reaches a SAFDL value of [ ]TS.

TS

Figure 14-1 Power to Melt and Maximum Attainable Power of Gd_2O_3-UO_2 Rod vs. Burnup

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Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environmental Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Question No. 15

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(vi) provides guidance in regards to GDC 10 by stating that rod internal gas pressures should be limited in order to (1) prevent cladding liftoff during normal operation, (2) prevent radial reorientation of hydrides in the cladding and (3) account for additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and should account for the effects of burnup and composition on the melting point.

Section 3.2.5 of APR1400-F-M-TR-13001-P provides the APR1400 fuel rod internal pressure analysis. Page 3-17 of APR1400-F-M-TR-13001-P qualitatively discusses the impact of TCD on fuel rod internal pressure, stating that the impact of TCD on fuel rod internal pressure is negligible. While the stated overall calculated rod internal pressure is less than system pressure, the actual limit proposed by KHNP is not clear. This has caused the staff to question the specific rod internal pressure limit proposed by KHNP.

In order to assist the staff to perform confirmatory calculations to investigate the statements that TCD has a negligible effect on the rod internal pressure safety analyses, provide the rod internal pressure limit used for the PLUS7 fuel rod internal pressure safety analysis. If the limit is greater than system pressure, provide a basis and update the topical report, as applicable.

Response

As described in Section 3.2.5 of APR1400-F-M-TR-13001-P, the criterion for internal gas pressure is to prevent clad lift-off (No Clad Lift-Off, NCLO). When the internal gas pressure exceeds the system pressure, the rod internal pressure should be less than the critical pressure limit, which could cause clad lift-off. The critical pressure limit for NCLO is the internal hot gas pressure, where the outward tensile creep rate of the cladding due to the differential pressure load would equal the fuel pellet swelling rate. Therefore, if rod internal pressure is less than system pressure, the NCLO criterion can be satisfied without critical pressure limit consideration.

The critical pressure limit depends on the power history. And it is determined by the FATES3B code calculation regarding creep and swelling rates. The FATES3B code automatically generates the critical pressure limit only when the internal gas pressure exceeds the system pressure. Generally, the calculated critical pressure limit would be [ ]TS.

Since the calculated maximum internal gas pressure of PLUS7 fuel rod did not exceed the system pressure as described in Section 3.2.5 of APR1400-F-M-TR-13001-P, clad lift-off did not occur and thus, satisfied NCLO criterion without a need to consider specific critical pressure limit.

Section 3.2.5 will be revised to provide a discussion on how the NCLO criterion is satisfied without a need to consider a critical pressure limit.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

Topical Report APR1400-F-M-TR-13001-P will be revised as indicated on the attached markups.

3.2.4 Cladding Oxidation and Hydriding

(1) Basis

Fuel system will not be damaged due to excessive oxidation under normal operation including AOOs.

(2) Criteria

The best estimate cladding oxide thickness shall be less than [ ]^{TS}.

The clad hydrogen pickup is limited to [ ]^{TS} at EOL to preclude loss of ductility due to hydrogen embrittlement by formation of zirconium hydride platelets.

(3) Evaluation

The cladding oxide thickness and hydrogen content of the PLUS7 fuel rod are evaluated by the same methodology and model as are used for Westinghouse PWR fuel rod designs. The best estimate oxide thickness and hydrogen content at the end of irradiation are calculated with NRC approved PAD code [3-11] during normal operation. Since cladding oxide thickness and hydrogen content do not increase during AOOs due to short time, it is not necessary to consider the AOOs for oxide thickness and hydrogen content.

The following parameters were used for evaluation.

a) Nominal design inlet temperature, system pressure, and core mass velocity based on Thermal Design Flow Rate
b) A crud thickness of [ ]^{TS}
c) A ZrO_2 thermal conductivity of [ ]^{TS}

The maximum cladding thickness and hydrogen content up to rod average burnup of 60,000 MWD/MTU for APR1400 are 85.0 μm and 596.0 ppm, respectively. The criteria of cladding oxide thickness and hydrogen content are satisfied.

3.2.5 Fuel Rod Internal Pressure

(1) Basis

exceed

Fuel system will not be damaged due to excessive rod internal pressure under normal operation.

(2) Criteria

The fuel rod internal hot gas pressure shall not the critical maximum pressure determined to cause an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length of the fuel rod.

Reorientation of the hydrides in the radial direction in the cladding shall not occur

The radiological dose consequences of DNB failures shall remain within the specified limits.

(3) Evaluation

FATES3B was used to calculate the rod internal pressure and corresponding critical limit according to the NRC approved methodology described in Reference 3-6. Where appropriate, the approved gadolinia methodology of References 3-7 and 3-8 has been applied. The critical limit is the internal hot gas pressure at which the outward tensile creep rate of the cladding exceeds the fuel pellet radial growth rate due to fuel swelling, thus creating any potentially damaging effects on the fuel rod due to detrimental thermal feedback effects within the fuel rod during normal operation.

Maximum rod internal pressure is calculated using conservative biasing of nominal fuel rod data including cladding outer diameter, cladding inner diameter, pellet outer diameter, active fuel length, fill gas pressure and will usually include additional conservatisms in the power levels representing each successive cycle of projected residency in the reactor core. The input power history to the code is important for rod internal pressure calculation. The methodology and conservatism for determining the rod power history are described in Reference 3-9. The main parameters and rod power histories considered in representative fuel rod internal pressure calculation are summarized in Table 3-4 and Figure 3-1, respectively.

The evaluation shows that the maximum rod internal pressures are [ ]TS and [ ]TS for the UO_2 fuel rod and Gd_2O_3-UO_2 burnable absorber fuel rod, respectively. ~~Therefore, no clad lift-off criterion is satisfied since the calculated gas pressures are less than critical pressure limit.~~

Therefore, no clad lift-off (NCLO) criterion is satisfied since the calculated gas pressures are less than the system pressure. FATES3B code calculates the critical pressure limit to prevent clad lift-off only when the rod internal pressure exceeds the system pressure.

DNB propagation evaluations for transients and DNB accidents are performed using the NRC approved methodology with INTEG code, which is a standalone computer code to predict fuel rod deformation and heat balance under conditions of DNB (Reference 3-6). [illegible] analysis methodology for any given plant. These inputs include time, heat flux, quality, mass flow, system pressure, rod internal pressure, and fuel rod initial geometry. To evaluate the potential for DNB propagation, the limiting DNB transients and internal pressure of [ ]TS are applied.

The results indicate that the clad strains induced by high temperature creep for limiting transients are less than [ ]TS This amount of strain does not induce DNB propagation to adjacent fuel rods.

Finally, hydride reorientation does not occur at internal pressure of [ ]TS, which is well above the predicted rod internal pressure of [ ]TS on PLUS7 fuel (Reference 3-6).

3.2.6 Internal Hydriding

(1) Basis

Fuel system will not be damaged due to excessive hydriding.

(2) Criteria

Primary hydriding is prevented by maintaining the level of moisture very low during the pellet manufacturing. The moisture content shall remain below the limit of 2.0 ppm (hydrogen from all sources for fuel pellets).

(3) Evaluation

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
Date of RAI Issued:	**06/18/2015**

Question No. 17

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(3)(A)(i) provides review guidance related to the assessment of fuel system damage related to dimensional changes (including hydrogen uptake induced swelling) that should be presented and reviewed.

It is stated on Page 4-7 of APR1400-F-M-13001-P that a 13% hydrogen pickup fraction will be used for ZIRLO. This pickup fraction is lower than the staff expected based on previous experience and has caused the staff to question the basis for this hydrogen pickup fraction.

Please justify the use of a 13% hydrogen pickup fraction. Provide a figure based on Figure 4-44 with an overlay of calculated hydrogen pickup assuming 13%, 15%, and 17.5% pickup fractions. Also include the data shown in Figure 10-1 of the response to RAI 4-7542 on this figure.

Response

The differences and ratio between measured and predicted hydrogen contents using [ ]TS hydrogen pickup fraction are shown in Figure 17-1 and Figure 17-2, respectively. The average difference is [ ]TS in Figure 17-1 and the measured to predicted ratio is [ ]TS in Figure 17-2. The predicted values represent the best estimate tendency of measured hydrogen content. But under prediction tendency the measured hydrogen content, to some degree, ascends as oxide thickness increases. Therefore, a hydrogen pickup fraction of [ ]TS is newly suggested as shown in Figure 17-3 and Figure 17-4.

The hydrogen contents calculated by assuming [ ]TS were provided in Figure 17-5, Figure 17-6, and Figure 17-7. These Figures include the data shown in Figure 10-1 of the response to RAI 4-7542 and Figure 4-44 in Topical Report.

As suggested with the hydrogen pickup fraction of [ ]^{TS}, the relevant oxide thickness and hydrogen content in Topical Report APR1400-F-M-13001-P were recalculated as follows.

The oxide thickness in Topical Report APR1400-F-M-13001-P was calculated using the imaginary power histories, which could conservatively cover all the power histories to be postulated. Because the imaginary power histories include unrealistic power histories, the oxide thickness and hydrogen pickup were very conservatively calculated.

Therefore, to remove the excessive conservatism, the oxide thickness and hydrogen content were re-evaluated with actual power histories based on nuclear physics data. It is noted that the [ ]^{TS} hydrogen pickup fraction was used in the re-evaluation. As a result, the maximum oxide thickness of [ ]^{TS} microns and a maximum hydrogen content of [ ]^{TS} ppm were calculated. The results will be reflected into APR1400-F-M-13001-P as shown in the attachment.

[ ]^{TS}

Figure 17-1 Measured Hydrogen Contents Minus Hydrogen Contents Predicted by [ ]^{TS} Hydrogen Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-2 Measured to Predicted Hydrogen Contents Using [ ]TS Hydrogen Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-3 Measured Hydrogen Contents Minus Hydrogen Contents Predicted by [ ]TS Hydrogen Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-4 Measured to Predicted Hydrogen Contents Using [ ]TS Hydrogen Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-5 Comparison of Measured and Predicted Hydrogen Contents with [ ]TS Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-6 Comparison of Measured and Predicted Hydrogen Contents with [ ]TS Pickup Fraction vs. Predicted Oxide Thickness

TS

Figure 17-7 Comparison of Measured and Predicted Hydrogen Contents with [ ]TS Pickup Fraction vs. Predicted Oxide Thickness

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

PLUS7 fuel design topical report (APR1400-F-M-TR-13001-NP) will be revised as indicated on the attached markups.

3.2.4 Cladding Oxidation and Hydriding

(1) Basis

Fuel system will not be damaged due to excessive oxidation under normal operation including AOOs.

(2) Criteria

The best estimate cladding oxide thickness shall be less than [ ].TS

The clad hydrogen pickup is limited to [ ]TS at EOL to preclude loss of ductility due to hydrogen embrittlement by formation of zirconium hydride platelets.

(3) Evaluation

The cladding oxide thickness and hydrogen content of the PLUS7 fuel rod are evaluated by the same methodology and model as are used for Westinghouse PWR fuel rod designs. The best estimate oxide thickness and hydrogen content at the end of irradiation are calculated with NRC approved PAD code [3-11] during normal operation. Since cladding oxide thickness and hydrogen content do not increase during AOOs due to short time, it is not necessary to consider the AOOs for oxide thickness and hydrogen content.

The following parameters were used for evaluation.

a) Nominal design inlet temperature, system pressure, and core mass velocity based on Thermal Design Flow Rate
b) A crud thickness of [ ]TS
c) A ZrO_2 thermal conductivity of [ ]TS

The maximum cladding thickness and hydrogen content up to rod average burnup of 60,000 MWD/MTU for APR1400 are ~~85.0~~ 69.4 µm and ~~596.0~~ 554.3 ppm, respectively. The criteria of cladding oxide thickness and hydrogen content are satisfied.

3.2.5 Fuel Rod Internal Pressure

(1) Basis

Fuel system will not be damaged due to excessive rod internal pressure under normal operation.

(2) Criteria

The fuel rod internal hot gas pressure shall not the critical maximum pressure determined to cause an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length of the fuel rod.

Reorientation of the hydrides in the radial direction in the cladding shall not occur

The radiological dose consequences of DNB failures shall remain within the specified limits.

TS

Figure 4-44 Hydrogen Content of ZIRLO Cladding

TS

TA

Figure 4-44 shows the measured hydrogen contents in cladding as a function of oxide thickness. Figure 4-44 includes CO3 fuel rod data and ZIRLO data irradiated in commercial reactors of Yonggwang unit 4, V.C. Summer, [ ]TS. The hydrogen contents of LTA fuel are within the ZIRLO cladding database.

The hydrogen content absorbed in cladding depends on the hydrogen pickup fraction and oxide layer thickness. The hydrogen pickup fraction is one of the material properties, which is [ ]TS for ZIRLO based on best estimate. This value is not dependent on the fuel types as well as the plant operating conditions. Therefore, the hydrogen content absorbed in ZIRLO cladding is based on oxide layer thickness generated by oxidation reaction.

Using the oxide thicknesses in Figure 4-9, the hydrogen contents averaged over the entire wall thickness is calculated using a hydrogen evolution model. The results are shown in Figure 4-45. For discharge burnup up to 60,000 MWD/MTU, the circumferential average hydrogen concentrations are less than [ ]TS for PLUS7 fuel in APR1400.
.
A photomicrograph of the etched C03 fuel rod cladding is shown in Figure 4-46. Figure 4-46 shows that the concentration of hydrides increases with increasing distance from the cladding inner surface due to the radial thermal gradient in the cladding. The cladding hydrogen contents exist mostly in the form of circumferential hydrides distributed in the outer region of the clad wall as shown in Figure 4-46.

4.3.6 Fission Gas Release

(1) Measurement Method

During the hot cell examination, A14 and C03 fuel rods of PLUS7 LTA irradiated in UCN-3 cycle 5 through cycle 7 were punctured and gases from these rods were collected and analyzed to determine the fractional fission gas release.

(2) Measurement Results and Evaluation

Table 4-8 shows the predicted values and measured data obtained in hot cell examination. As shown in Table 4-8, the fission gas released are [ ]TS for A14 and C03 fuel rods, respectively. In addition, the rod internal pressures were measured for A14 and C03 fuel rods. The measured rod internal pressures of [ ]TS are well within the database of PWR fuel rod internal pressure at the condition of room temperature.

4.4 Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel and post-irradiation surveillance are described in DCD tier 2, Section 4.2.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
Date of RAI Issued:	**06/18/2015**

Question No. 18

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

To perform accurate confirmatory calculations to evaluate the application's conformance with GDC 10, NRC must use the correct input information for the APR1400 design, including rod geometry, reactor conditions, power history, and axial power profile.

Please provide the following sample calculations using FATES3B. For each case, include all appropriate input information including rod geometry, reactor conditions, power history, and axial power profile:

a. Provide sample calculations of cladding strain under AOOs for a typical AOO overpower event. Provide calculations at rod average burnup of 0 GWd/MTU, 20 GWd/MTU, 40 GWd/MTU, and 60 GWd/MTU.

b. Provide a sample calculation of rod internal pressure for a bounding power history up to a rod average burnup of 62 GWd/MTU. Provide pressure calculations as a function of time.

c. Provide a sample calculation of power to melt at the following rod average burnup levels; 0 GWd/MTU, 10 GWd/MTU, 20 GWd/MTU, 30 GWd/MTU, 40 GWd/MTU, 50 GWd/MTU, 60 GWd/MTU.

d. Provide a sample calculation of fuel stored energy for a bounding power history up to a rod average burnup of 62 GWd/MTU. Provide stored energy calculations as a function of time.

Response

Design input information including rod geometry, power history, axial power profile and reactor conditions are described in Tables 18-1 - 18-5. Sample calculations of transient strain, rod internal pressure, power to melt, and stored energy are below

Table 18-1 Fuel Rod Geometry

TS

Table 18-2 Generic Power Histories and Axial Power Profile for Transient Strain

TS

Table 18-3 Bounding Power Histories for Rod Internal Pressure, Power to Melt and Stored Energy

TS

Table 18-4 Axial Power Profile for the Calculation of Rod Internal Pressure, Power to Melt and Stored Energy

TS

Table 18-5 Reactor Conditions

TS

Results of sample calculations for transient strain, rod internal pressure, power to melt and stored energy are as follows;

a) Transient strain under AOO at rod average burnup of 0 GWd/MTU, 20 GWd/MTU, 40 GWd/MTU, and 60 GWd/MTU.

TS

Re-examined total (elastic plus plastic) strain will be reflected in topical report as shown in the attached markup.

b) Rod Internal gas pressure up to fuel rod average burnup of 60 GWd/MTU TS

Rod internal pressure was calculated up to 60 GWd/MTU which is a license target burnup.

c) Power to melt at 0 GWd/MTU, 10 GWd/MTU, 20 GWd/MTU, 30 GWd/MTU, 40 GWd/MTU, 50 GWd/MTU, 60GWd/MTU.

TS

d) Fuel stored energy for one fuel rod as a function of time

TS

Fuel stored energy was calculated up to 60 GWd/MTU which is a license target burnup.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

Topical Report APR1400-F-M-TR-13001-NP will be revised as indicated on the attached markups.

normal operation involving fuel handling and storage, reactor servicing, power operation and reactor trip, heatup and cooldown, and minor fuel handling accident using the standard formulas.

The stress analyses for AOOs such as “Inadvertent Opening of an Atmospheric Dump Value”, Uncontrolled CEA Withdrawal at Power”, “Total Loss of Reactor Coolant Flow and “Loss of Condenser Vacuum” events were performed using the FATES3B code [3-1 through 3-4].

The results of the evaluation indicate that the primary tensile and compressive stresses in the cladding and end cap welds are within the allowable limits as shown in Table 3-3.

3.2.2 Cladding Strain

(1) Basis

Fuel system will not be damaged due to excessive strain under normal operation including AOOs.

(2) Criteria

At any time during the fuel rod lifetime, the net unrecoverable circumferential tensile cladding strain shall not exceed [ ]TS, based on the beginning-of-life (BOL) cladding dimensions. This criterion is applicable to normal operating conditions and following a single AOO.

The total (elastic plus plastic) circumferential cladding strain increment produced as a result of a single AOO shall not exceed [ ]TS relative to the pre-transient condition.

Ductility is a function of irradiation and hydride formation in the cladding wall. Because the waterside corrosion for ZIRLO is significantly lower and should result in less hydrogen uptake and less hydride formation, total stain capability of ZIRLO is projected to be in excess of [ ]TS at burnups of 60,000 MWD/MTU. Thus, a [ ]TS strain limit will continue to be applied as a strain criterion.

(3) Evaluation

The evaluation methodology for the fuel rod cladding strain is discussed in Reference 3-5, which was reviewed and approved by NRC for Westinghouse CE PWR fuel designs.

The first part of the strain limit concerns that the total plastic strain incurred as a result of cladding creep and cladding yielding during long term normal operation and the following short transient conditions. Cladding creep strain and plastic strain due to cladding yielding are driven by the stress in the cladding that results from differential pressure and interface with the fuel pellets. The method used to evaluate the strain accounts for power dependent and time dependent changes (e.g., fuel rod void volume, fission gas release and gas temperature, differential cladding, cladding creep and thermal expansion) that can produce strain in the fuel cladding. In addition, the strain analysis accounts for both long term, normal operating condition, and short term, transient condition.

For the present application, the predicted plastic strain using FATES3B code is [ ]TS, and the total (elastic plus plastic) circumferential cladding strain increment produced as a result of a single AOO is [ ]TS. As the calculated strains are less than [ ]TS, the design criteria are satisfied.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
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Question No. 19

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(3)(C)(i) provides guidance in regards to GDC 10 and the phenomenological models important to fuel temperature (stored energy) calculations. This fuel temperature calculation is important to the pellet swelling and clad creep models.
Table 4-8 on Page 4-12 of APR1400-F-M-TR-13001-P shows predicted and measured results from the puncture analysis. The staff notes that FGR is overpredicted but rod internal pressure is underpredicted. This has caused to the staff to question the validity of the void volume calculations and the potential subsequent impacts on the pellet swelling and clad creep models.
Provide a discussion to explain why FGR is overpredicted but rod internal pressure is underpredicted, and update the topical report if applicable. If the pellet swelling or clad creep models are non-conservative, also address all other impacted analyses and update the topical report, if applicable.

Response

Generally, rod internal pressure is dependent on fission gas release as well as void volume. With regards to fission gas release, it can be easily inferred that the more fission gas release, the more rod internal pressure. However, Table 4-8 on Page 4-12 of APR-1400-F-M-TR-13001-P shows two comparison results in which fission gas release is overpredicted but rod internal pressure is underpredicted. The main reason for this unexpected behavior is attributed to the fact that FATES3B overpredicts the void volumes as shown in Table 4-8.

The measured values for void volume are compared with the predicted ones, which are shown in Figure 2-7 and Figure 2-8 of TCD technical report (APR1400-F-M-TR-13002-P). As can be seen in those figures, the measured void volumes for some rods are greater than the volumes of the FATES3B predictions but for others the volumes are smaller than the volumes of the FATES3B predictions. Two data in Table 4-8 didn’t result from the non-conservatism of the pellet swelling or ZIRLO clad creep models. They happened to fall into a case where the predicted void volume is greater than the measured one. Those models had already been verified and approved and their descriptions can be found in References 19-1 and 19-2. As explained in Section 2.1.3 of APR1400-F-M-TR-13002-P, in overall, over conservatism to predict void volume with an average value of [ ]TS is applied to the FATES3B code.

References

[19-1] [ ]TS
[19-2] [ ]TS

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environmental Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
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Question No. 20

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(iv) provides guidance in regards to GDC 10 by stating that overheating of fuel pellets should be avoided by preventing centerline melting. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and should account for the effects of burnup and composition on the melting point.

Table 3-10 on Page 3-24 of APR1400-F-M-TR-13001-P lists various core inlet and outlet temperatures for plants. The staff notes that APR1400 is listed as having a higher core average linear heat rate while having essentially the same ΔT and core average mass flow as the other examples. This has caused the staff to question the accuracy of the core average linear heat rate calculations presented in Table 3-10.

Please describe the core average linear heat rate calculations in sufficient detail to explain how APR1400 calculates a higher linear heat rate and make the supporting calculations available for staff audit or submit them on the docket.

Response

To obtain the core average linear heat rate, the core thermal power [ ]^{TS} of the APR1400 should be divided by number of fuel rods and flow path length of the active core. The number of fuel rods in a core is [ ]^{TS}. And the flow path length of the active core is [ ]^{TS}. Therefore, the core average linear heat rate can be acquired by following the calculation shown as equation 20-1. The parameters are also presented in the Table 4.1-1, 4.1-2 and 4.4-8 in Chapter 4 Reactor in DCD.

[ ]TS

The core thermal power of the OPR1000 reactor type such as Yonggwang unit 4, Ulchin unit 3, and Yonggwang unit 5 is [ ]TS. And the core thermal power of Westinghouse type of Yonggwang unit 2 is [ ]TS. Therefore, the difference in thermal power is the main reason for the different linear heat rate presented in Table 3-10.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environmental Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
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Question No. 21

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 provides review guidance related to the development of acceptance criteria based on test data, in part or in whole, for various phenomena (e.g. fretting wear, oxidation/hydriding/crud buildup, dimensional changes, PCI, cladding embrittlement, etc.).

Appendix A of APR1400-F-M-TR-13001-P provides a summary of PLUS7 fuel assembly tests. During the review, the staff noted that the test conditions were listed but not compared with the APR1400 operational ranges for all of the tests. This has made it difficult for the staff to complete the review of the adequacy of the testing.

Provide a table that compares the test condition ranges to the APR1400 condition ranges for the liftoff tests, assembly vibration tests, and buckling strength discussed in Appendix A so it can be determined that the test conditions bound the expected reactor conditions. Update the topical report as necessary.

Response

As described in Appendix A of APR1400-F-M-TR13001-P Rev.0, the liftoff test and assembly vibration test were conducted in the Fuel Assembly Compatibility Test System (FACTS), and the buckling strength test was conducted by using a dynamic grid crush test apparatus. These facilities are located at Westinghouse in Columbia, SC. The test conditions were compared with the APR1400 operational ranges as below:

1. The liftoff tests were performed by increasing the flow rate from [ ]TS and from [ ]TS by increments of [

]TS. When the flow rate reached [ ]TS, the flow increased by increments of [ ]TS for a better estimate of the flow rate that could liftoff the fuel assembly. The PLUS7 fuel assembly was lifted off at the flow rates of [ ]TS, respectively.

Compared with the in-reactor mechanical design flow of [ ]TS for the PLUS7 fuel assembly in the APR1400, the corresponding mechanical design flow in FACTS liftoff test was [ ]TS, and it was due to the difference in the fuel assembly channel size between the APR1400 and the FACTS loop. The conditions are compared in Table 21-1.

2. During fuel assembly vibration tests, the loop flow rate was increased from [ ]TS to the maximum achievable loop flow rate of [ ]TS. The corresponding mechanical design flow for the assembly vibration test in FACTS is [ ]TS. As shown in Table 21-2, the flow range in FACTS covers the mechanical design flow in APR1400.

3. The grid buckling strength test was conducted by using a dynamic crush grid test apparatus at [ ]TS which is conservative considering the average temperature of APR1400 reactor is [ ]TS. In addition, the pendulum weight of [ ]TS used in the grid buckling strength test was selected for the PLUS7 fuel assembly span weight of [ ]TS. The conditions are compared in Table 21-3.

Note:

TS

Table 21-1 Comparison of the liftoff test condition ranges to the APR1400 operating condition ranges

TS

Table 21-2 Comparison of the assembly vibration test condition ranges to the APR1400 operating condition ranges

TS

Table 21-3 Comparison of the buckling strength test condition ranges to the APR1400 operating condition ranges

TS

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

PLUS7 fuel design topical report (APR1400-F-M-TR-13001-NP)will be revised as indicated on the attached markups.

A.2.0 SUMMARY OF TESTS

A.2.1 Fuel Assembly Vibration Test (FACTS)

1.0 Introduction and Objectives

The objective of this test is to confirm that the PLUS7 design is not susceptible to high resonance flow-induced assembly vibration over a range of plant operating flow rates. This is assessed by reviewing the vibration spectra from displacement transducers (see Figure A.2.1-1) and the instrumented fuel rod accelerometers for the presence of abnormal, flow-dependent, resonant vibration peaks. The absence of such peaks will be sufficient to determine the acceptability of the fuel assembly design.

2.0 Test Conditions

The test flow conditions were systematically varied in an effort to excite the vibration modes of the fuel assembly. Such a variation consisted of setting the loop at the required temperature of [ ]TS [ ]TS and sweeping the loop flow rate from [ ]TS to the maximum achievable flow rate. The flow rate was then returned to [ ]TS at a same rate. The maximum achievable flow rate for this test was [ ].TS The PLUS7 fuel assembly will be placed at [ ].TS The equivalent mechanical design flow in the FACTS test is [ ].TS Although the FACTS loop could not reach the elevated test flow rate [ ]TS the maximum achievable flow rate does bound all possible operating flow rates, representing over 117% of the best estimate flow [ ],TS which is the flow rate at which PLUS7 fuel assembly will be placed. The maximum achievable flow rate is more than adequate to confirm that the PLUS7 design does not exhibit flow-induced resonant fuel assembly vibration.

The test conditions and APR 1400 operating conditions are compared in Table A.2.1-1.

3.0 Test Results

Insert Table A.2.1-1 in Appendix 21-1

The flow sweep test was performed at [ ].TS In this test, the loop flow rate was increased in a constant manner from [ ]TS to the maximum achievable loop flow rate of [ ]TS [ ]TS in six minutes. Because the instrumented rod accelerometers were mounted at mid-grid elevations with as-built cell conditions, the accelerometer outputs represent the fuel assembly vibration.

Figure A.2.1-2 shows that the assembly did not experience the high resonance flow-induced fuel assembly vibration. There was no indication of abnormal flow-induced vibration response throughout the test flow range.

4.0 Summary and Conclusion

A FACTS loop test was conducted to verify that the PLUS7 design did not exhibit high resonance flow-induced fuel assembly vibration. Displacement transducers and instrumented fuel rods at grid locations measured the fuel assembly vibration. There was no indication of abnormal flow-induced vibration response throughout the test flow range.

A.2.3 Mid Grid Crush Test

1.0 Introduction and Objectives

The dynamic crush strength of the mid grid is required to obtain the structural characteristics to show that both seismic and LOCA loads are met.
The specific test objectives for the dynamic crush test of the PLUS7 ZIRLO mid grid design at elevated temperatures were as follows:

- To obtain the impact force as a function of impact velocity
- To determine the grid ultimate load capability
- To characterize the grid failure mode
- To obtain the data to determine the grid dynamic stiffness

2.0 Test Conditions

weight

The dynamic crush test was performed at operating temperature and the pendulum ~~inertial mass~~ was calculated as the weight of one span of PLUS7 fuel assembly.

The pendulum angle was increased by 1° from 7° until grid crushed.

Delete

- ~~Elevated temperature: 600°F±20°F~~
- ~~Pendulum inertial mass: [ ]TS~~
- ~~Pendulum initial angle: 7°~~

The test conditions and APR 1400 operating conditions are compared in Table A.2.3-1.

Insert Table A.2.3-1 in Appendix 21-1

3.0 Test Results

Twelve grid crush tests were sequentially performed. The PLUS7 ZIRLO mid grid crush test results are summarized as follows: (Table.A.2.3-2)

A.2.3-2

Table ~~A.2.3-1~~ PLUS7 ZIRLO Mid grid Crush Test Results

[ ]TS

4.0 Summary and Conclusion

The crush strength values for the PLUS7 ZIRLO mid grid tested with rod-in-cell are shown in Figure A.2.3-3.

A summary of the dynamic crush test results is as follows:

- Dynamic Crush Strength:

A.2.9 FACTS Lift-off Test for the PLUS7 Fuel Assembly

1.0 Introduction and Objectives

The objective of the lift-off tests of the PLUS7 fuel assembly is to determine the flow rates at which the fuel assembly lifts off under specified temperatures and holddown spring compressions. The test was performed with the Fuel Assembly Compatibility Test System (FACTS) as shown in Figure A.2.9-1 for the PLUS7 fuel assembly design with 40/20 ([ ]TS [ ])TS Debris Filter Bottom Nozzle (DFBN).

2.0 Test Conditions

The test conditions and APR 1400 operating conditions are compared in Table A.2.9-1.

To ensure that the fuel assembly would lift-off during the tests, the holddown springs were compressed by [ ].TS The lift-off tests were performed by increasing the flow rate from [ ]TS with increments of [ ].TS When the flow rate reached [ ],TS the flow increased with an increment of [ ]TS for a better estimate of the flow rate that could lift-off the fuel assembly.

Two uni-axial accelerometers were mounted on the base of the pressure vessel to monitor the output impact signal. The output from the accelerometers was amplified and monitored through an oscilloscope and a visicorder.

Insert Table A.2.9-1 in Appendix 21-1

Table A.2.9-2

Table ~~A.2.9-1~~ Fuel Assembly and Specifications

Component Descriptions	
Top Grid Material	Inconel 718
Protective Grid Material	Inconel 718
Bottom Grid Material	Inconel 718
Protective Grid Inner Strap Heights	[ ]TS
Bottom Grid Inner Strap Heights	[ ]TS
Top Grid Inner Strap Heights	[ ]TS
Mid Grid Material	ZIRLO
Number of Mid grids	9
Mid Grid Inner Strap Heights	[ ]TS
Number of Rods	236
Rod OD	[ ]TS
Test Rod Length	[ ]TS
Number of Guide Thimble	4
Diameter of Guide Thimble Tubes	[ ]TS
Number of Instrument Tube	1
Diameter of Instrument Tube	[ ]TS
Dashpot Elevation (Top of Thimble Tube Smaller OD Elevation)	[ ]TS
Rod-to-Bottom Nozzle Gap	[ ]TS
Thimble Tube Plugging Device	None. However the flow is blocked by the standoff tubes
Test Condition	
[ ]	

Appendix 21-1

Table A.2.1-1 Comparison of the assembly vibration test condition range to the APR1400 operating condition ranges

TS

Table A.2.3-1 Comparison of the buckling strength test condition range to the APR1400 operating condition ranges

TS

Table A.2.9-1 Comparison of the liftoff test condition range to the APR1400 operating condition ranges

TS

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
Date of RAI Issued:	**06/18/2015**

Question No. 22

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(3)(A)(i) provides review guidance related to the assessment of fuel system damage by stating that stress limits must be presented and reviewed.

Section 2.2.2 of APR1400-F-M-TR-13001-P provides the structural integrity design basis, criteria, and evaluation. Within this section, it is stated that the evaluation of fuel assembly for seismic and LOCA loads will be addressed in the APR1400 DCD Section 4.2. The staff notes that APR1400 DCD Section 4.2 in turn points back to APR1400-F-M-TR-13001-P. Therefore, the staff is unable to ascertain if the proposed stress limits are violated by the analysis.

Provide the stress analysis results for the PLUS7 fuel design, and update the topical report as necessary.

Response

As described in Section 2.2.2 of APR1400-F-M-TR-13001-P, the evaluation of the fuel assembly for seismic and LOCA loads is addressed in Section 4.2 of APR1400 DCD tier 2. In addition, Section 4.2.3.5.3 of the DCD refers to technical report APR1400-Z-M-NR-14010-P (proprietary) and NP (non-proprietary) “Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading for the APR1400” in order to show the analysis results in detail. This technical report contains information on the test, model, analysis methodology, criteria, and stress analysis results for the fuel assembly during postulated accidents. Table 6-1 of Reference #35 (APR1400-Z-M-NR-14010) in DCD Section 4.2 provides the stress analysis results of the fuel assembly for seismic and LOCA loads.

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environmental Report.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. PROJ 0782

RAI No.:	**5-7954**
SRP Section:	**4.2 Fuel System Design**
Application Section:	**PLUS7 Fuel Design for the APR1400 (APR1400-F-M-TR-13001-P)**
Date of RAI Issued:	**06/18/2015**

Question No. 23

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). SRP Section 4.2 (II)(1)(B)(viii) and Appendix A provides review guidance related to mechanical fracturing based on seismic and LOCA applied loads. It is also stated specifically that control rod insertability must be maintained.

This topic is addressed in Section 2.2.2 of APR1400-F-M-TR-13001-P and also in the response to Question 2 of RAI 4-7542 (ML14177A220). The staff notes that for postulated accidents, the limits proposed in the topical report are based on ASME Section III Service Level D requirements. This service level could result in "faulted" conditions for the guide tubes. A faulted guide tube could affect the ability to insert RCCAs, and therefore challenge GDC 27.

Provide a discussion that covers the proposed stress-strain limits and what level of damage could occur to the components based on those limits. If damage could occur to the guide tubes, include a description of the tests and results that demonstrate control rod insertability. Update the topical report, as necessary, to capture these points.

Response

For control rod insertability, standard review plan (SRP) Section 4.2 Appendix A (IV-1) describes as follows: "Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit), as defined above, then significant deformation of the fuel assembly would not occur and lateral displacement of the guide tubes would not interfere with control rod insertion. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that the deformation is not severe enough to prevent control rod insertion."

Based on the guideline above, control rod insertability for PLUS7 fuel is satisfied because buckling in the PLUS7 fuel assembly grid does not occur as a result of seismic and LOCA loads [Reference 23-1].

For the evaluation of guide tube stresses induced by the lateral displacements and the axial loads on the fuel assembly during seismic and LOCA events, Appendix F of ASME Section III is used as the general stress criteria: 1) the general primary membrane stress intensity P_m shall not exceed the lesser of $2.4S_m$ and $0.7S_u$, 2) the primary membrane plus primary bending stress intensity P_m+P_b shall not exceed 150% of the limit for P_m. [

]TS.

In addition, the guide tube behavior related to the control rod insertability was evaluated through the PLUS7 fuel assembly lateral stiffness test [Appendix A.3 in Reference 23-1 and Section 3 in Reference 23-2]. [

]TS. The maximum lateral displacement of guide tube as a result of seismic and LOCA analyses is [ ]TS based on the seismic and LOCA analysis described above. The location and numbering of strain gages attached to the guide tubes are presented in Figure 23-1.

In summary, the control rod insertability during seismic and LOCA events is maintained because the maximum guide tube stress based on the PLUS7 seismic/LOCA analysis and PLUS7 lateral stiffness test is below the yield strength ([ ]TS ksi at temperature) as described above. The guide tube design stress limits during seismic and LOCA events could be adjusted if necessary.

References

[23-1] [ ]TS

[23-2] [ ]TS.

[1)] : the project name during PLUS7 fuel development

TS

Figure 23-1 PLUS7 Mechanical Test Fuel Assembly Strain Gage Location and Numbering
[Figure 3.2.3 in Reference 23-2]

Impact on DCD

There is no impact on the DCD.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environmental Report.