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 Issue:	**06/18/2015**

Question No. TR PLUS7 Fuel Design for the APR1400-16

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)(A)(ii) provides guidance in regards to GDC 10 by stating that the cumulative number of strain fatigue cycles on the structural members should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor.
Pages 3-17 and 2-5 of the TCD report (APR1400-F-A-NR-13002-P) discuss cladding strain and fatigue. Section 3.4.2 states that the increased thermal expansion can be offset with available design margin in the cladding strain and fatigue limits. Sample calculations provided in the TCD report show that the fatigue damage factor will increase from 0.28 to 0.77 when TCD is considered. This has caused the staff to question the claim that increased thermal expansion can be offset with available design margin in the cladding strain and fatigue limits.

a) Discuss how the fatigue analysis will be performed on a cycle specific basis given this demonstration that FATES3B alone is inadequate to assess the fatigue damage fraction. Update the topical report, if necessary, to include the clarification.

b) The fatigue damage factors (FDF) presented in topical report APR1400-F-M-TR-13001-P and technical report APR1400-F-A-NR-13002-P (for the operating condition "without TCD") do not appear to match. Update the report(s) as necessary to reflect the correct FDF.

Response

a) According to Reference 16-1, a license condition that will impose more restrictive operation/design radial power fall-off (RFO) curve limits for St. Lucie 2 was accepted by NRC. This approach is applied to the licensing application of APR1400 Topical Reports.

Even though the FATES3B fuel performance code does not explicitly model Thermal Conductivity Degradation (TCD), FATES3B centerline temperature predictions are well matched to Halden measured data up to intermediate levels of rod average burnup [ ]TS. However, as the burnup goes beyond [ ]TS, the extent of underprediction gap linearly increases and finally reached [ ]TS at the rod average burnup of [ ]TS and then remains constant for higher burnup.

Based on these comparisons, a more restrictive (management) radial power fall-off (RFO) curve limit is determined by reducing the rod power from the analysis RFO power by an amount sufficient to reduce the fuel centerline temperature corresponds to [ ]TS over the burnup range from [ ]TS and [ ]TS for higher burnup. This temperature penalty will set aside margin to account for the burnup dependent effects of TCD. Figure 16-1 shows the derived management RFO curve limit accounting for the TCD penalty and the analysis RFO curve limit which is used for fuel performance analyses and temperature generation for safety analysis.

The management RFO curve limit will be validated on a cycle-by-cycle basis to confirm the applicability for a given subsequent cycle. This assures that the plant will be operated under the management RFO curve limit that accounts for the TCD penalty.

Fatigue analysis in the topical report is performed using higher rod power history (radial peaking factor) than the analysis RFO curve limit. Therefore, the fatigue analysis results presented in the topical report can be continually used for fatigue criterion confirmation if the management RFO curve limit is confirmed to be valid on a cycle specific basis. Section 3 in the PLUS7 topical report will be updated with mark-ups to provide a discussion on the derivation of both analysis and management RFO curve limits.

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Figure 16-1 Radial Fall-Off Curve Limits

b) The fatigue damage factors presented in topical report APR1400-F-M-TR-13001-P and technical report APR1400-F-A-NR-13002-P(for the operating condition without TCD) are [ ]^{TS} and [ ]^{TS}, respectively. The higher fatigue damage factor in topical report APR1400-F-M-TR-13001-P is mainly attributed to the excessive conservatism of power history assumed in the calculation. Rod power history (radial peaking factor) used in the fatigue analysis of the topical report is much higher than that of technical report as shown in Figure 16-2. The other conservatism in the fatigue analysis for the topical report is to assume higher End Of Life (EOL) rod average burnup of [ ]^{TS}. In contrast, the fatigue analysis result (fatigue damage factor=[ ]^{TS}) presented in the technical report was obtained by assuming the EOL burnup of [ ]^{TS} (the target licensing burnup) and reducing the excessive conservatism in the rod power history (radial peaking factor) as shown Figure 16-2.

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Figure 16-2 Radial Peaking Factors used for Fatigue Analyses

Note) Figure 16-1 indicates that both RFO curves are conservatively regenerated to secure the applicability for a subsequent reload cycle. Thus, the radial peaking factor (rod power history) for fatigue analysis is also conservatively regenerated to bound the newly developed analysis RFO curve limit as shown in Figure 16-3. The fatigue result using the regenerated radial peaking factor in Figure 16-3 is [ ]^{TS}. Both regenerated RFO curve limits in Figure 16-1 and the recalculated fatigue result will be reflected in topical report (APR1400-F-M-TR-13001-P). Affected contents due to the TCD penalty approach will be provided as mark-up and also reflected in the topical report.

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Figure 16-3 Previous and Regenerated Radial Peaking Factors used for Fatigue Analyses

References

[16-1] Letter, ML#12198A202, "FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE FLORIDA POWER AND LIGHT ST. LUCIE, UNIT 2, LICENSE AMENDMENT REQUEST FOR AN EXTENDED POWER UPRATE," July 23, 2012.

Impact on DCD

The effect on the Design Control Document (DCD) Tier 2 will be reviewed when the TCD penalty methodology results are applied. DCD Tier 2 will then be updated if necessary.

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

The PLUS7 fuel design topical report (APR1400-F-M-TR-13001-NP) will be updated as indicated on the attached markups to include the TCD penalty approach. Technical Report APR1400-F-A-NR-14002-NP will be withdrawn.

Fuel coolability applies to postulated accidents. To meet the requirements of GDC 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability should be provided for all severe damage mechanisms. Fuel coolability criteria involves cladding embrittlement, violent expulsion of fuel, generalized cladding melting, fuel rod ballooning and structural deformation.

In this section, evaluations have been done to verify that the fuel rod design bases and criteria can be met for the PLUS7 fuel design. The fuel rod design criteria for cladding stress, cladding strain , cladding fatigue, cladding corrosion/hydriding, rod internal pressure, cladding collapse, overheating of fuel pellets, pellet-to-cladding interaction have been evaluated using the NRC approved Westinghouse fuel rod performance codes and methodologies (References 3-1 through 3-9, 3-12, 3-13). The codes applied for evaluating each fuel rod design criterion are summarized in Table 3-2.

The design bases and limits for PLUS7 fuel rod are the same as NRC approved design bases and limits for Westinghouse CE PWR fuel designs (Reference 3-10).

The stress and loading limit for other than cladding, fretting wear, dimensional change, assembly lift-off, rod mechanical fracturing and structural deformation are described in Chapter 2. The criteria and evaluations related to safety analyses will be described in DCD Chapter 15 in detail.

FATES3B does not explicitly model fuel thermal conductivity degradation (TCD) with burnup. In order to preserve margin and conservatively offset the effects of TCD, a burnup dependent temperature penalty is suggested and approved (Reference 3-14). In detail, FATES3B centerline temperature predictions were well matched to Halden measured data up to intermediate levels of rod average burnup (35,000 MWD/MTU). However, temperature underprediction starts at 35,000 MWD/MTU burnup and increases linearly up to 200°F at 50,000 MWD/MTU rod average burnup, then remains constant for higher burnup. Based on these comparisons, a more restrictive (management) radial power fall-off (RFO) curve limit is determined by reducing the rod power from the analysis RFO power by an amount sufficient to reduce the fuel centerline temperature to correspond to 0 - 200°F over the burnup range from 35 to 50 GWD/MTU and 200°F for higher burnup. Figure 3-1 shows the derived management RFO curve limit accounting for TCD effect and analysis RFO curve limit which is used for fuel performance analyses and temperature generation for safety analyses. The management RFO curve limit will be validated on a cycle-by-cycle basis to confirm the applicability for a given reload cycle. This assures that the plant will be operated under the management RFO curve limit. Therefore, the calculation results from the analysis RFO curve limit can be maintained as applied to fuel rod design criteria confirmation and downstream safety analyses as long as the management RFO curve limit is valid on a cycle specific basis. The detailed evaluation results using the analysis RFO curve limit are described in this section.

3.2.1 Cladding Stress

(1) Basis

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

The rod internal pressures used to perform the stress analyses of the fuel rod designs accounts for power dependent and time dependent changes (e.g., fuel rod void volume, fission gas release and gas temperature, differential cladding pressure, cladding creep and thermal expansion) that can affect stresses in the fuel rod cladding. The rod external pressures are consistent with the event being analyzed and are biased in the conservative direction (maximum for compressive stresses, minimum for tensile stresses). The maximum tensile and compressive stresses were calculated for

3.2.3 Cladding Fatigue

(1) Basis

Fuel system will not be damaged due to excessive fatigue under normal operation.

(2) Criteria

For the number and type of transients which occur during normal operation, end-of-life (EOL) cumulative fatigue damage in the cladding must be less than [ ]TS.

A fatigue analysis for the ZIRLO cladding is performed using the Langer-O'Donnell fatigue design curve. A safety factor of 2 on the stress or a safety factor of 20 on the number of cycles is imposed on this curve conservatively. The fatigue damage is defined as the ratio of the number of calculated cycles in a given strain range to the permitted number of cycles determined by fatigue curve at strain range.

(3) Evaluation

The method used for fatigue analysis of fuel rod accounts for power dependent and time dependent changes (e.g., rod void volume, fission gas release and gas temperature, cladding creep and thermal expansion, and pellet swelling and thermal expansion) that can produce cyclic straining of the fuel cladding. With respect to determining changes in cladding and pellet diameters, the methodology used to evaluate the fatigue damage from power cycling is the same for the strain evaluation during normal operation, except that some of the parameters are biased in the opposite direction to provide results that are conservative for fatigue analysis.

The number of load cycles assumed in the fatigue evaluation is defined below for startup/shuddown, power variations and reactor trip during normal operation. The stress amplitudes for the cyclic loads are calculated using the FATES3B code.

a) Startup/shutdown

A startup/shutdown cycle is defined as the change from the 0% cold stand-by-state to the 0% hot stand-by state. [ ]TS startups/shutdowns between 0% power and room temperature conditions are considered for fatigue evaluation.

b) Power variation during normal operation

Load follow operation is the most limiting condition of normal operation for fatigue evaluation. The power variation for load following is conservatively assumed to vary between 10% and 100% on a daily basis, and for each day.

c) Reactor Trip

The limiting condition for power change due to the AOOs is the reactor trip from 100 power to hot condition, conservatively. [ ]TS reactor trips are considered for fatigue evaluation.

The total cumulative fatigue damage factor from power cycling, reactor trips and startups/shutdowns is [ ]TS, which is below the limit of [ ]TS. It was demonstrated that the fuel rod fatigue criterion was satisfied.

(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.

3-2

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 burst behavior under conditions of DNB (Reference 3-6).

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

Fuel rod failure will not occur due to the overheating of cladding under normal operation including AOOs.

(2) Criteria

There should be a 95% probability at the 95% confidence level that a hot fuel rod in the reactor core will not experience a DNB during normal operation or AOOs. For postulated accidents, the rods that experience DNB are assumed to fail for radiological dose calculation purposes.

(3) Evaluation

The evaluation for overheating cladding is addressed in plant specific transient and accident analysis (Chapter 15 of DCD).

3.2.9 Overheating of Fuel Pellets

1) Basis

Fuel rod failure will not occur due to the overheating of fuel pellets under normal operation including AOOs. For postulated accidents, the total number of rods that experience centerline melting should be considered for radiological dose calculation.

(2) Criteria

During normal operating and AOOs, the fuel centerline temperature shall not exceed the melting temperature accounting for degradation due to burnup and addition of burnable absorbers.

The fuel rod is considered to be failed [ ]TS during postulated accidents. In the event of fuel failure, the radiological consequences of fuel failure must be accounted in the dose calculations.

The melting temperature of UO_2 is taken to be 5,080 °F (unirradiated) and to decrease [ ]TS per 10,000 MWD/MTU fuel burnup. For Gd_2O_3-UO_2 burnable absorber fuel rod, the melting temperature decreases additionally [ ]TS per weight percent of Gd_2O_3.

(3) Evaluation

The fuel centerline temperatures as a function of burnup are calculated using the NRC approved FATES3B code and methodology described in References 3-1 through 3-5 for UO_2 rod and References 3-7 and 3-8 for Gd_2O_3-UO_2 burnable absorber fuel rod.

The powers to fuel melting are calculated as a function of rod burnup. To preclude fuel melting, the peak local power experienced in normal operation and AOOs should be less than the power to fuel melting at all burnups.

The minimum design margin occurs at the end of cycle 1 because of 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. The calculated power to fuel melting at the end of cycle 1 is [ ]TS, which is bounded by the local power density for trip setpoint of [ ]TS.

- ZIRLO Corrosion Data Base

ZIRLO in-reactor corrosion data have been obtained from the BR-3 and North Anna fuel program. The BR-3 test fuel residence times extend to 47,500 hours, and the North Anna fuel has achieved a 10,150 EFPH residence time after one cycle of irradiation. Comparison of the corrosion data from the BR-3 ZIRLO rods with sibling Zr-4 rods show ZIRLO peak corrosion ranging from 66% to 83% of Zr-4 peak corrosion. Peak corrosion data from the one cycle North Anna rods range from 0.3 to 0.6 mil for the ZIRLO rods, and from 0.5 to 0.9 mil for the Zr-4 rods.

Based on these data, a corrosion multiplier of 0.75 relative to the Zr-4 corrosion model has been determined for use in ZIRLO fuel rod performance evaluations.

- Increase of Corrosion Model Multiplier

However, it was found that the model multiplier of 0.75 is less conservative for ZIRLO claddings irradiated in Yonggwang unit 2, Yonggwang unit 4 and Ulchin unit 3 through the Post Irradiation Examination Program.

Using the measured oxide layer thickness data of Yonggwang unit 2, Yonggwang unit 4 and Ulchin unit 3, the new corrosion model multiplier was determined to evaluate the best estimate predictions of oxide thickness. A total of 25 oxide thickness data with high burnup exceeding 50 MWD/kgU was used to determine best estimate corrosion model multiplier for Yonggwang unit 2, Yonggwang unit 4 and Ulchin unit 3. Those oxide thickness data consist of 6 fuel rods of V5H fuel irradiated in Yonggwang unit 2, 6 fuel rods of GUARDIAN fuel irradiated in Yonggwang unit 4 and 13 fuel rods of PLUS7 fuel irradiated in Ulchin unit 3 were used.

Through the process of statistical analyses and evaluations, the corrosion model multiplier of 0.92 was determined for best estimate oxide thickness prediction for ZIRLO cladding. Table 3-8 shows the comparison of measured oxide thickness and predicted oxide thickness using corrosion model multiplier of 0.92. The results of statistical analyses are as follows.

Number of Data :	25
Mean Value of M/P :	0.89
Standard Deviation of M/P :	0.19

- Verification of Corrosion Model Multiplier

In order to verify the accuracy of corrosion model multiplier determined based on measured oxide thickness data from Yonggwang unit 2, Yonggwang unit 4 and Ulchin unit 3, the verification of modified corrosion model multiplier was performed using the measured oxide thickness data of PLUS7 fuel rods irradiated in Yonggwang unit 5. The 72 thrice burnt fuel rods with fuel rod average burnups of up to 58,000 MWD/MTU were selected for verification of corrosion model multiplier.

Figure ~~3-2~~ shows the comparison of measured oxide thickness and predicted oxide thickness. The predicted oxide thicknesses were generated using the corrosion model multiplier of 0.92.

3-3

3-3

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.~~

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~~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.

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Figure 3-2 Comparison of Measured Oxide Layer Thickness and Predicted Oxide Layer Thickness for H615 and H605 Assemblies

Non Proprietary

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Figure 3-~~1~~ Rod Power History Used for PLUS7 Fuel Rod Performance Analysis

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Figure 3-~~2~~ Comparison of Measured Oxide Layer Thickness and Predicted Oxide Layer Thickness for H615 and H605 Assemblies

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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.

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

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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.

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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.~~

3-14 Letter, ML#12198A202, "FINAL SAFETY EVALUATION REPORT ASSOCIATED WITH THE FLORIDA POWER AND LIGHT ST. LUCIE, UNIT 2, LICENSE AMENDMENT REQUEST FOR AN EXTENDED POWER UPRATE," July 23, 2012.