1.0 <u>Introduction</u>

By letter dated June 9, 2008, Westinghouse submitted Licensee Topical Report (LTR) WCAP-16943-P, “Enhanced Gray Rod Cluster Assembly (GRCA) Rodlet Design” (Reference 1) for review and approval. This Safety Evaluation Report (SER) is based on the submitted letter, and responses to requests for additional information (RAIs). Although the GRCA system is not a safety system, the staff reviewed the potential impact on safety systems following the appropriate portions of guidance provided in Chapters 4.2, “Fuel System Design”; 4.3, “Nuclear Design”; 4.4, “Thermal and Hydraulic Design”; and 15, “Accident Analysis”, of the Standard Review Plan (SRP), NUREG-0800 (Reference 2) as applicable. The GRCA design as documented in WCAP-16943-P would be referenced as part of a future post-COL licensing amendment. WCAP-16943-P described an enhanced AP1000 GRCA intended to replace the Ag-In-Cd (AIC) GRCA absorber rodlet currently approved for use in the AP1000 Standard Plant. This SER does not cover any plant designs beyond the AP1000 Standard Plant.

This SER is divided into sections: Section 2 presents a summary of applicable regulatory criteria and guidance, Section 3 contains a summary of the information presented in the topical report, and Section 4 contains the technical evaluation of the major components of the enhanced tungsten GRCA design. These components include: (1) nuclear design, (2) material properties, (3) mechanical design, and (4) thermal hydraulic design. Section 5 presents the conclusions of this review, and Section 6 contains the restrictions and limitations on the use of the Enhanced GRCA Rodlet design.

2.0 <u>Regulatory Criteria</u>

The AP1000 GRCA system is a non-safety system used to allow for small adjustments in core reactivity. However, as the GRCA system interacts with the fuel assemblies by being inserted into the guide tubes while in use, the impacts of the GRCA system are evaluated against potentially impacted regulatory criteria that govern the fuel system. The following sections present the relevant requirements and guidance that were used to guide the staff’s review.

2.1.1 Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10, “Reactor Design,” requires, in part, that control and protection systems be designed with appropriate margin to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during normal operation, including the effects of anticipated operational occurrences. There are no specific SAFDLs associated with the GRCA system; however, the potential impacts of the GRCAs upon the fuel system SAFDLs are considered as part of this evaluation.

10 CFR Part 50, Appendix A, GDC 27, “Combined Reactivity Control Systems Capability,” requires:

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Enclosure 1

10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," requires:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

2.1.2 Relevant Guidance

NUREG-0800, "Standard Review Plan," (SRP) provides detailed review guidance that is acceptable to the staff in meeting the applicable regulatory requirements. In particular, NUREG-0800 sections that contain guidance relevant to this review are:

1. Section 4.2, "Fuel System Design,"
2. Section 4.3, "Nuclear Design,"
3. Section 4.4, "Thermal and Hydraulic Design,"
4. Section 15, "Accident Analysis."

3.0 <u>Summary Of Technical Information</u>

The tungsten GRCAs are designed for use in advanced nuclear reactors to allow for small adjustments in core reactivity. The use of these low worth rods reduces the need for soluble boron concentration adjustments via the chemical and volume control system (CVCS).
The use of these tungsten GRCAs is non-safety in nature as compared to the high worth black Rod Cluster Control Assemblies (RCCAs), which are intended to provide shutdown capability for the core. In the event of a reactor trip, the GRCAs are dropped into the core along with the RCCAs, but their reactivity effect is comparatively small and is not credited in the safety analyses.

The general design of each GRCA consists of 24 double-walled rodlets attached to a spider assembly. The overall outline dimensions, as well as interfaces with the control rod drive mechanisms and fuel assembly, are identical to the RCCAs. The GRCA rodlet's outer cladding is made from stainless steel and the inner sleeve is made from [ ], thereby encapsulating the tungsten absorber in a double-walled design to ensure no contact occurs between the tungsten and coolant. The gap between the inner sleeve and outer cladding is backfilled with an inert gas. The bottom of the rodlet contains a spacer, which allows for a gradual transition in worth of the rodlet as the GRCA is inserted.

The methodology described in Topical Report WCAP-16943-P can be divided into four main categories:

1) Nuclear Design
2) Material Properties
3) Mechanical Design
4) Thermal Hydraulic Design.

The evaluation presented in WCAP-16943-P includes normal operation and transient conditions.

3.1. Nuclear Design

The applicant presents discussions and analyses for various nuclear design considerations for the tungsten GRCA design. The first topic of discussion is the reactivity control function design of the tungsten GRCAs. They are designed to provide a comparatively flat depression to the assembly power as compared with the previously approved 4- and 12-rod AIC GRCA designs. The nuclear lifetime is defined as the GRCA depletion point at which the plant can no longer perform load-follow maneuvers without requiring a change in the core soluble boron concentration during the maneuver. The presented analysis of the tungsten GRCA design demonstrates a marked increase in lifetime over the reference Ag-In-Cd GRCA design. The application then provides a discussion of the GRCA effects on the power distribution. Analyses are presented that show the power distribution effects of the tungsten GRCA design on neighboring assemblies as compared with the previously approved Ag-In-Cd GRCA designs. The change in assembly power for the neighboring assemblies in the tungsten GRCA design is shown to be less, due to the lower initial assembly worth as compared with the 12 reduced diameter Ag-In-Cd rodlets GRCA design. Predicted neutron fluences and transmutation in the enhanced GRCA rodlet is presented based on the PARAGON transport code. These calculations are based on the intended operating parameters of the GRCAs during their lifetime within an AP1000 plant.

Per Section 3.5 of the application (Reference 1), safety-related nuclear design analyses will conservatively account for the use of tungsten GRCAs. GRCA's will not be credited in shutdown margin calculations made prior to the startup of the core but will be accounted for in analyses in which their use leads to more limiting results.

The application provides a detailed description of the codes used in the analyses of rod worth and nuclear lifetime. ORIGEN was used for preliminary depletion calculations, identification of dominant nuclear reactions for tungsten, and prediction of gas buildup as a product of tungsten depletion. The detailed 2D transport calculations are performed by PARAGON with an updated 70-group neutron cross section library to include appropriate tungsten and rhenium nuclide cross section data from ENDF/B-V1.8.

The nodal code ANC (which includes a control rod insertion model) is used for 3D core modeling. PARAGON was benchmarked against MCNP (Reference 1), and experimental data were used to validate the neutron capture cross section data for tungsten.

3.2. Material Properties

The applicant provided tungsten material properties including various physical properties, thermal expansion, specific heat, and thermal conductivity. In addition, mechanical properties including elastic, tensile, and phase-diagrams were provided by the applicant based on available literature.

Material property information is provided for both unirradiated and irradiated tungsten. Irradiated properties are based on various fast reactor and space application experience, which is based on a harder neutron spectrum than would be expected in a light water reactor core. Additional data is provided on tungsten-rhenium alloys to account for transmutations that would occur during the lifetime of the GRCAs.

The expected neutron fluence for the tungsten GRCAs over approximately [ ] has been calculated and is the basis for the analyses presented in WCAP-16943-P.

3.3. Mechanical Design

The mechanical design of the tungsten GRCA includes the mechanical performance under normal and accident conditions, based on the material properties of the GRCA design. The mechanical analysis included compatibility with reactor coolant, and corrosion by water and steam.

As tungsten and coolant would undergo a corrosion reaction in the event that they interact, the design of the tungsten GRCAs includes a double-walled encapsulation of the tungsten. The applicant provides information regarding corrosion rates of various tungsten alloys in water environments in the range of 180-320°C. Although this information is provided, the design is such that both the inner and outer cladding would need to fail in order for the tungsten to interact with the coolant.

The applicant also provides an analysis of the mechanical impact of tungsten due to irradiation. Specifically, discussions regarding the irradiation effects on the ductile to brittle transition temperature (DBTT) and yield stress are presented. Additionally, the applicant includes discussions regarding swelling and concludes that it will not adversely affect the performance of the cladding.

3.4. Thermal Hydraulic Design

In Section 5.3 of the report (Reference 1), the applicant provides the thermal hydraulic design bases and evaluation for the tungsten GRCAs. The design bases include:

1. Maximum absorber temperature
2. No surface boiling within the dashpot region
3. No bulk boiling in the guide thimble tubes
4. Bypass flow in the guide thimble tubes.

4.0 Technical Evaluation

4.1.1. Nuclear Design

The topical report outlines the use of tungsten as a GRCA absorber material instead of the currently used AIC-based GRCA design. The use of GRCAs is to allow the reactor to follow load demands, fuel burnup, temperature changes, and xenon changes without adjusting the boron concentration in the coolant. This reduces waste water processing, and allows for a simpler CVCS. The overall basic mechanical design is the same as that for all the control rod assemblies.

There are twenty four rodlets attached to a spider assembly, which is attached to the control rod drive mechanism. In the case of the tungsten GRCA design, all 24 rodlets contain absorbing material. This is not the case for the AIC GRCA designs (some have 4, others have 12 absorbing rodlets – the remaining positions are stainless steel blanks). The use of tungsten has the following operational advantages:

1) A tungsten-based GRCA system is able to provide the desired control margin up to the maximum analyzed neutron fluence values as listed in Table 3.4-1 of WCAP-16943-P (Reference 1). This is due to the fact that depletion products resulting from tungsten burnup are as good as, or slightly better than, tungsten at absorbing neutrons.

2) Fast neutron irradiation experiments indicate that tungsten would perform as expected when subjected to the fast fluence expected to impinge on the tungsten during its lifetime in an AP1000.

3) Global and local power peaking is minimally influenced by the tungsten-based GRCA compared to the currently designed AIC systems.

The GRCA design methodology is based on suitably modified accepted Westinghouse design codes PARAGON-ANC using ENDF/B-VI nuclear data. The data for tungsten were further validated by comparison to recently measured cross sections (Reference 25). In order to determine the extent of the depletion chain that should be retained in the PARAGON code, a preliminary calculation was performed using the ORIGEN code and then an appropriate depletion chain was created to represent the transmutation reactions. Lifetime worth estimates that were carried out confirmed the viability of the concept, as well as the superior performance with regard to power peaking with and without the rods present.

In RAI-01, the staff requested more information regarding the radial dimensions and types of stainless steel used in the design. The approximate radial dimensions are as follows (Reference 2):

- Absorber diameter [ ]
- Sleeve inner diameter [ ]
- Sleeve outer diameter [ ]
- Cladding inner diameter [ ]
- Cladding outer diameter [ ]

These dimensions indicate that the overall rodlet dimension is equal to the current RCCA rodlet dimension and that there is a small gap between the sleeve and cladding, which would help to contain any swelling. Given the consistency with the outer diameter of the current RCCA dimensions and the gap to account for possible expansion, no follow-up information was necessary.

The cladding material consists of cold worked stainless steel [ ]. The use of stainless steel [ ] is discussed in Section 4.2.5 of this SER and is considered acceptable for this design.

In RAI-07, the staff inquired as to the spatial distribution of the transmutation products (rhenium and osmium) from the tungsten. In response, the applicant stated that the spatial distribution of the transmutation products peak on the surface of the rodlet, and are minimal at the center of the rodlet. This distribution is more pronounced in the resonance region, and less pronounced in the thermal region. The shape of this profile demonstrates to the staff that the model contained the geometry detail necessary to accurately reflect the depletion of the GRCA rodlet. This profile supports the isotopic distribution presented in figures 3.4-1 and 3.4-2 of the topical report and agrees with the staff's expectations regarding the flux profile throughout the GRCA rodlet based on experience with the burnup of RCCAs. No further questions were necessary on this topic.

In RAI-08, the staff requested additional information regarding the post-irradiation examination of the rodlets. In response, the applicant outlined the post-irradiation examination of the rodlets. Two types of examinations are proposed. The first are non-destructive in nature and include eddy current testing, ultra-sound testing, and cladding profilometry and wear. Destructive testing of the rodlets, inside a hot cell, would be carried out after the rodlets had experienced a significant amount of exposure. This might require approximately [ ], which would imply a fluence with a magnitude of approximately [ ]. This examination would determine the physical condition of the tungsten rod, the relative quantities of the transmutation products (both the average and surface elemental content of the depleted tungsten absorber), and mechanical integrity of the rodlet. The results of this examination, particularly the relative quantities of the transmutation products, would be the ultimate validation of the PARAGON code system for modeling tungsten GRCAs, including the nuclear data and the length of the depletion chain. Until these results are available, however, the validity of the depletion step of the PARAGON code is supported by the physical reasonableness of its predictions for tungsten GRCA depletion along with the previous PARAGON approval for similar reactor predictions (Reference 24).

The staff concludes that the examination and testing protocol in this response adequately demonstrates that any unanticipated degradation of the GRCA rodlets would be identified well before large scale failure and therefore satisfies the request. No further questions were necessary.

In RAI-09, the staff requested a copy of the ORIGEN input deck, which the applicant supplied in its response. This deck was executed to determine if the truncation of the depletion chains used in the submittal and implemented in the PARAGON code were valid. The most important rhenium isotopes were retained together with the associated osmium isotopes resulting from decay in rhenium isotopes. The most important isotopes are those with the highest thermal capture cross sections, or the highest resonance integrals. In this way the buildup and destruction of tungsten, rhenium, and osmium can be determined and hence the time dependent isotopic content of the rodlet can be tracked. The staff therefore concluded that the applicant had correctly truncated the depletion chains and that the libraries used by PARAGON were correct.

In RAI-11, the staff requested additional supporting evidence to validate the nuclear data based on thermal neutron spectrum conditions instead of the fast reactor conditions presented in the report. In response, the applicant stated that after the submission of WCAP-16943-P, a set of critical experiments was carried out with tungsten absorbers in simulated thermal reactor conditions at the [

]. Critical configurations were constructed with and without tungsten absorber rods in the test assemblies in order to allow worth predictions to be assessed. [

]

Criticality was achieved in each case by [

]. MCNP models were constructed for each experimental configuration and the multiplication factors were predicted. The staff finds that the good agreement between the MCNP results and the measured data and, by extension, the good agreement between MCNP and the PARAGON code gives confidence that the PARAGON code gives an accurate representation of the rodlet behavior at Beginning of Life (BOL) conditions.

The critical experiments described above do not include any simulations that include the phenomena associated with depletion of tungsten, and the buildup and destruction of rhenium and osmium. The depletion calculations carried out using PARAGON indicate that there will be a slow increase in rodlet worth with time.

The staff finds this slow increase to be reasonable, since rhenium has a larger capture cross-section than tungsten. Eventually this increase drops off as the rhenium is depleted or decays and is replaced by osmium, which generally has a smaller capture cross-section than rhenium. This last statement depends on the isotopic mix resulting from the depletion process. The values of thermal neutron capture cross section and corresponding resonance integral for selected isotopes that could be involved in a more extensive depletion chain are shown in Table 4.1-2. The staff concludes that this behavior, although not confirmed experimentally, is physically reasonable based on publically available data.

The staff notes that physics tests would be carried out at the start of every cycle to confirm the worth of the rodlets, at which time any unexpected deviation would be noted, and compensated for. Additionally, it is noted that the tungsten GRCAs are used for reactivity control only and, if a problem should arise with this proposed system, the in-place boron system could be used as an alternative option to achieve the same goals. The staff has reviewed the response to this RAI and has found the added information on thermal reactor experiments clarifies the issues raised in the RAI. This closes the questions raised in the RAI.

Table 4.1-2 Nuclear Properties of Selected Isotopes in the Tungsten Depletion Chain.
(Reference 23)

Isotope	Thermal Capture (barns)	Resonance Integral (barns)
Ta-181 (s)*	20.5	660.0
W-180 (s – 0.12)**	30.0	214.0
W-182 (s – 26.31)	20.0	604.0
W-183 (s – 14.28)	10.0	337.0
W-184 (s – 30.64)	1,7	14.7
W-185 ((T½ - 75.1 d., isomeric state 1.67 min.)	No Data No Data for isomeric	No Data No Data for isomeric
W-186 (s – 28.64)	37.9	485.0
W-187 (T½ - 23.72 hrs)***	64.0	2760.0
Re-184 (T½ - 38.0 d., isomeric state 169 d.)	No Data No Data for isomeric	~ 8900 No Data for isomeric
Re-185 (s)	112.0	1717.0
Re-186 (T½ - 90.64 hrs)	No Data	No Data
Re-187 (s)	76.4	300.0
Re-188 (T½ - 17.02 hrs., isomeric state 18.59 min.)	~ 2.0 No Data for isomeric	No Data No Data for isomeric
Os-186 (s)	80.0	280.0
Os-187 (s)	320.0	500.0
Os-188 (s)	4.7	152.0
Os-189 (s)	25.0	674.0
Os-190 (s)	13.1	22.0
Os-191 (T½ - 15.4 d., isomeric state 13.1 hrs.)	~ 383.0 No Data for isomeric	No Data No Data for isomeric
Ir-191 (s)	954.0	3500.0
Ir-192 (T½ - 73.83 d.)	1420.0	3242.0
Ir-193 (s)	111.0	1350.0
Pt-192 (s)	10.0	115.0

* (s) = stable isotope
** (s – abundance)
*** (T½ - Half-life in hours or days)

4.2. Material Properties

Section 4.0 of WCAP-16943-P (Reference 1) describes the material properties of tungsten. Sections 4.2.1 through 4.2.4 of this report show comparisons of the material properties provided in Reference 1 to material properties for tungsten that have previously been compiled by PNNL. Section 4.2.5 evaluates the material properties for stainless steel used in the GRCA rodlet cladding. Section 4.2.6 evaluates the material properties for [ ] used in the GRCA rodlet sleeve. The staff compared the mechanical properties provided in Reference 1 to a database of material properties that the staff's contactor (Pacific Northwest National Laboratory (PNNL)) has based on reliable test data.

4.2.1. Tungsten Physical Properties

Some basic physical properties of tungsten were provided in Reference 1. The properties are compared in Table 4.2-1 to those previously compiled by PNNL. The staff agrees that these properties show reasonable agreement.

Table 4.2-1. Physical Properties of Tungsten

Property	Westinghouse value	PNNL value
Density	19.254 g/cm^3	19.3 g/cm^3 [†]
Atomic weight	183.85 g/mol	183.84 g/mol [†]
Melting point	3410±20°C; 3380°C	3422°C [†]
Boiling point	5700±200°C; 5500°C	5555°C [†]
Crystal structure	BCC	BCC [‡]

[†] Reference 3

[‡] Reference 4

It was noted that the applicant gave two different values for melting point and boiling point. The staff asked in RAI-02 (Reference 2) which of these values is used in the safety analyses. The applicant responded (Reference 2) that a melting point of 3380ºC is used in the thermal analysis, and that the boiling point is not used in any analyses. This response is acceptable as the selected melting point is the minimum value of the values shown in Table 4.2-1.

The applicant provided a correlation for thermal expansion of tungsten over the temperature range of 25°C to 2500°C. Figure 4.2-1 shows this correlation along with two other correlations previously collected by PNNL (References 5, 6, 7, and 8). The staff agrees that these correlations all show excellent agreement.

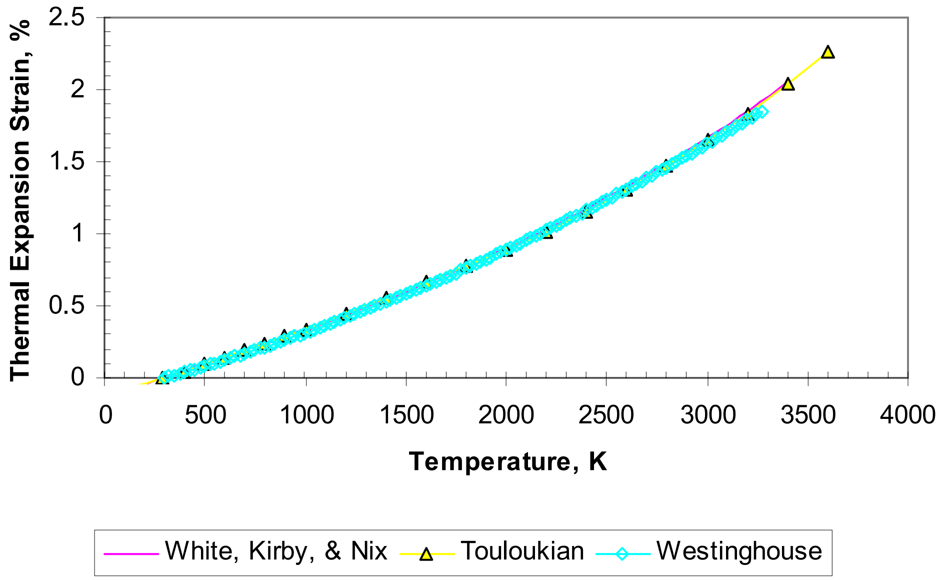

Figure 4.2-1 Thermal expansion of tungsten

The applicant provided three correlations for specific heat of tungsten over the temperature ranges of 0°C to 3000°C, 25°C to 727°C, and 25°C to 2727°C. Figure 4.2-2 shows these correlations along with one other correlation previously collected by PNNL (References 9 and 10). The first and second correlations show excellent agreement with the PNNL correlation over the applicable temperature range.

The third correlation shows excellent agreement with the PNNL correlation up to a temperature of about 1230°C. Above this temperature the third correlation underpredicts the PNNL correlation and the first correlation.

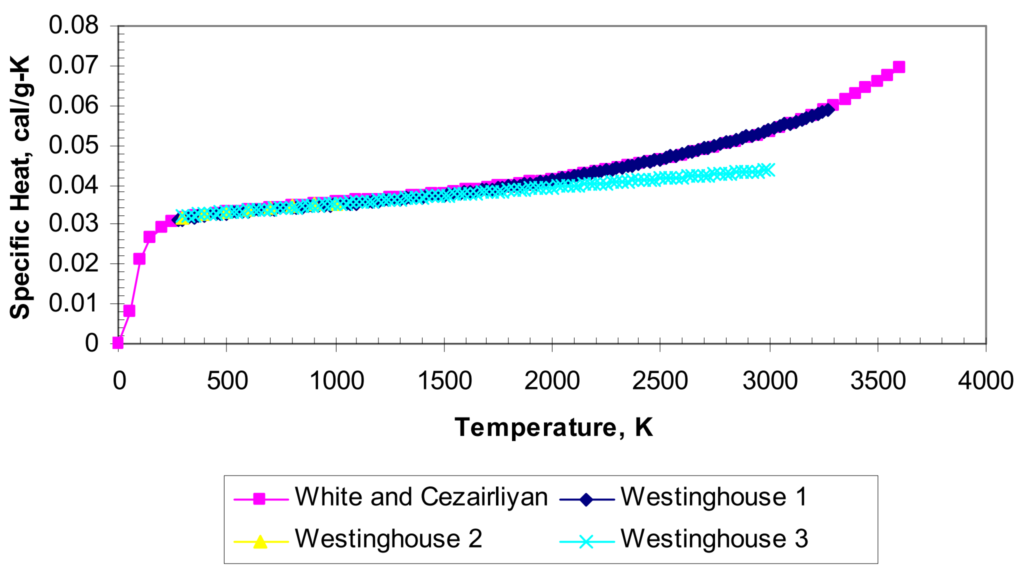

Figure 4.2-2 Specific heat of tungsten

Since the applicant gave three different correlations for specific heat of tungsten, the staff asked in RAI-02 (Reference 2) which of these correlations is used in the safety analyses. The applicant responded that there are no analyses that use the correlations for specific heat. This response is acceptable since it has no impact on any applicable safety analysis, but does provide additional general description of the material properties for tungsten found in open literature.

The applicant provided a data table for thermal conductivity of tungsten over the temperature range of 0°C to 2000°C. Figure 4.2-3 shows these data along with five other correlations previously collected by PNNL from the open literature (References 11, 12, 13, 14, and 15). The data provided by the applicant show excellent agreement with the open literature correlations within the uncertainty of the data.

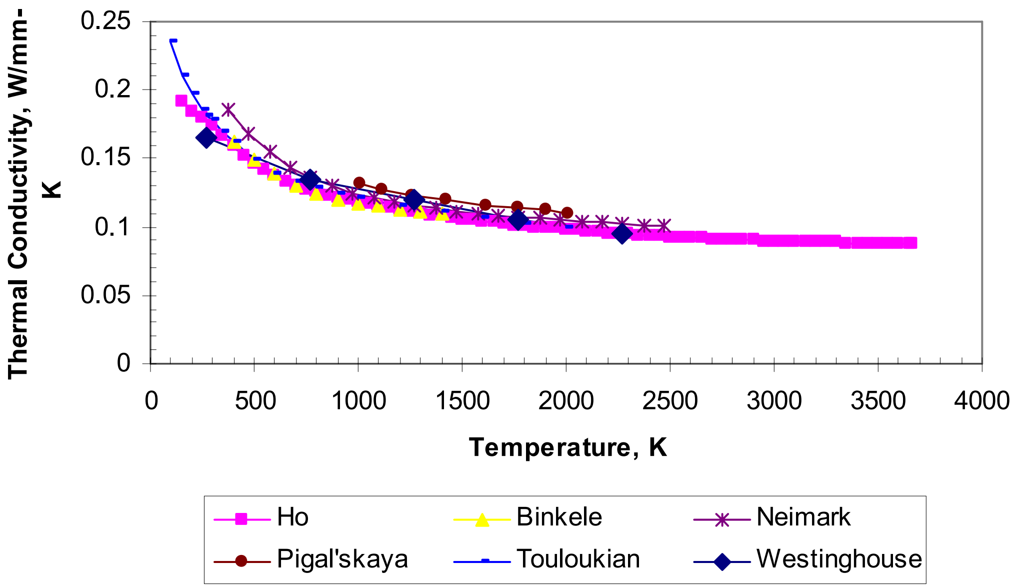

Figure 4.2-3 Thermal conductivity of tungsten

4.2.2. Tungsten Mechanical Properties

The applicant provided a data table for elastic modulus and Poisson's ratio of tungsten over the temperature range of 0°C to 1600°C. Figures 4.2-4 and 4.2-5 show these data along with a correlation previously collected from the open literature by PNNL for elastic modulus and Poisson's ratio, respectively (Reference 16). The data provided by the applicant show excellent agreement with the open literature correlations within the uncertainty of the data. The scale is expanded on Poisson's ratio in Figure 4.2-5, which suggests the difference is large, but the maximum difference is only 3%, which is considered to be within the uncertainty of Poisson's measurements; the difference is even smaller within the temperature range of this application.

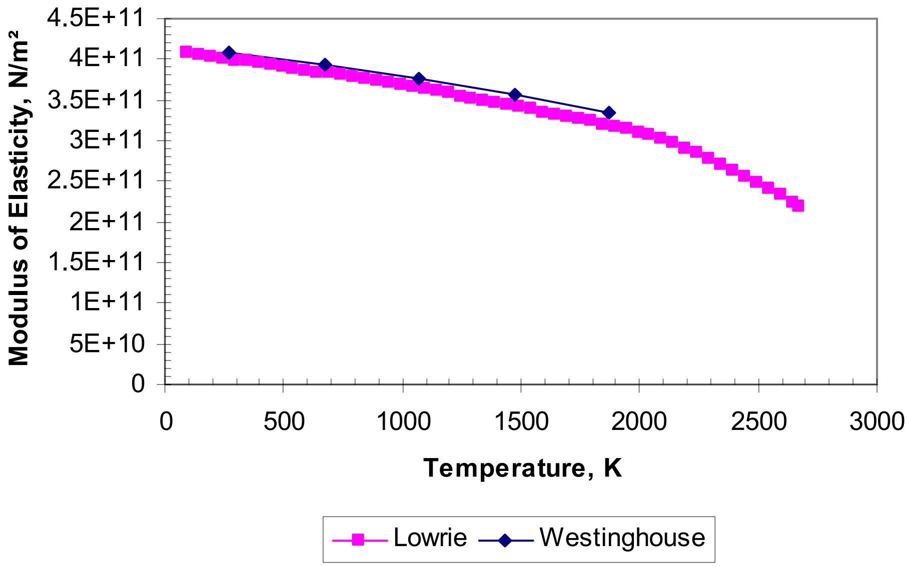

Figure 4.2-4 Young's modulus of tungsten

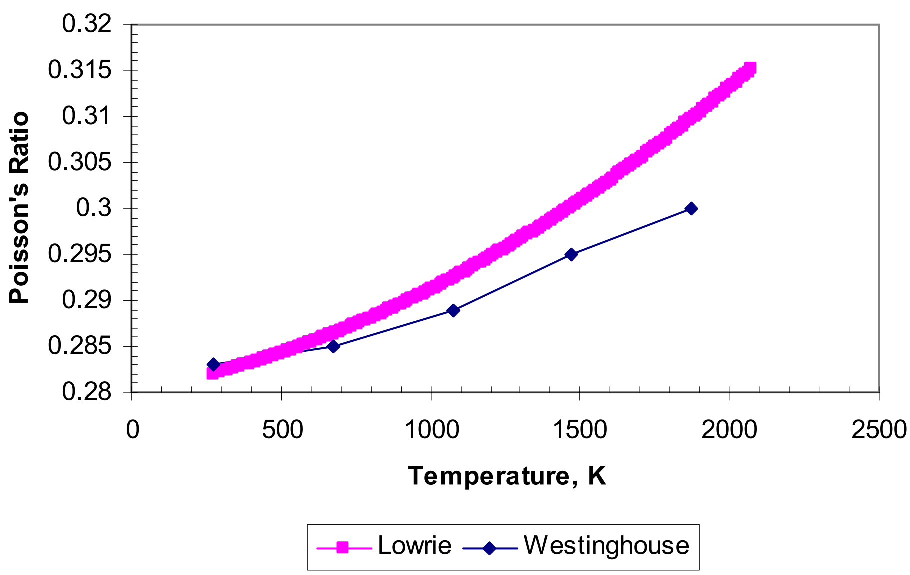

Figure 4.2-5 Poisson's ratio for tungsten (Note: expanded scale magnifies difference)

The applicant provided data that show that tensile stress and hardness of tungsten decrease with increasing temperature. The applicant also makes the claim that ductility increases with increasing temperature, but no data were provided to substantiate this claim. Due to the qualitative nature of these claims, the staff asked in RAI-02 (Reference 2) if the strength or hardness of tungsten is used in a quantitative way in any of the safety analyses. The applicant responded (Reference 2) that the tungsten tensile stress and hardness are not used in a quantitative way in any analysis. Since the applicant confirmed that the tensile strength properties were not used in the analysis, the staff finds the response acceptable.

RAI-05 asked (Reference 2) what the effects of osmium build-in with irradiation were on tungsten performance. The applicant responded (Reference 2) that, if the tungsten had a complete loss of ductility, the [ ] sleeve would confine the tungsten and maintain the functionality of the GRCA. The staff agrees that the properties of the sleeve would serve to confine the tungsten even in the event of a complete loss of ductility, and therefore agrees that tungsten tensile stress and hardness does not represent a concern for GRCA functionality.

4.2.3. Tungsten Compatability With Reactor Coolant

The applicant provided data to demonstrate that tungsten is reactive with both water and air. However, the applicant notes that the stainless steel cladding and [ ] sleeve provide double encapsulation of the tungsten absorber to prevent interaction between the absorber and the coolant.

The staff agrees that the double encapsulation of the tungsten absorber is an acceptable double barrier to prevent reaction with the reactor coolant. The staff review of a potential single failure point of both the sleeve and cladding due to mechanical design issues is documented in Section 4.3.

4.2.4. Tungsten Material Irradiation Experience

There are typically two areas of concern with irradiation of solid materials.

The first is transmutation of the elements in the material to other elements that can cause the formation of different crystal phases, thus significantly altering the material properties from those of the unirradiated material properties.

The applicant provided a phase diagram of tungsten and rhenium (Re) to demonstrate the impact of the transmutation of tungsten to rhenium throughout the service life of the GRCA rodlet. The applicant provided predicted transmutation throughout life for the GRCA rodlet for the absorber surface and the radial average. The staff asked in RAI-07 for the applicant to provide a calculation of expected flux profile across the radius of the GRCA rodlet. The applicant responded (Reference 2) by showing a plot of thermal flux as a function of radius and a plot of flux for an energy group containing the largest resonance peaks in tungsten. In the response, the applicant predicts that the maximum rhenium content at end of life will be [ ]. The W/Re phase diagram demonstrates that at this level the rhenium will be soluble in the body-centered cubic (BCC) tungsten matrix. Therefore the properties of [ ] are expected to be similar to those of tungsten. Based on the calculated low Re content at end of life conditions, the staff agrees that the irradiated absorber material properties will be similar to pure tungsten even after irradiation.

The staff noted that osmium (Os) is also formed throughout the service life of the GRCA rodlet. It is predicted that the maximum osmium content at end of life will be [ ]. Figure 4.2-6 shows the Re/Os phase diagram, which demonstrates that rhenium and osmium are mutually soluble in each other at all compositions (Reference 17). Figure 4.2-7 shows the W/Os phase diagram, which demonstrates that osmium is only soluble in tungsten up to 6 wt% (Reference 17). [ ] The staff asked in RAI-05 for the applicant to discuss the impact of buildup of osmium on the material properties of irradiated tungsten. The applicant responded (Reference 2) by noting that the osmium content [ ]. The applicant also referenced a paper that stated that exceeding the osmium solubility limit is expected to cause a slight decrease in lattice parameter which would cause slight shrinkage of the material rather than swelling. This response demonstrates that the structural integrity of the sleeve is not compromised by the build-up of Os and is therefore found to be acceptable.

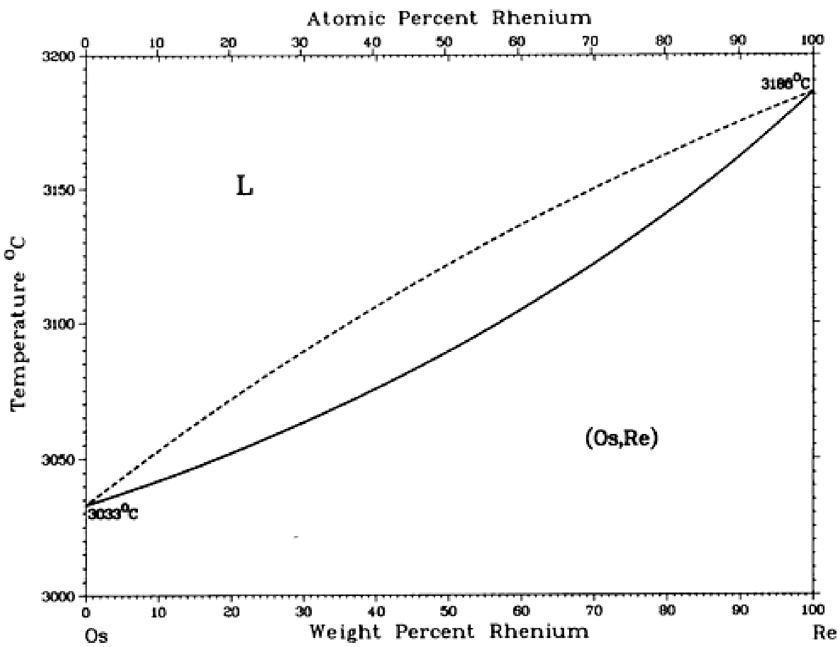

Figure 4.2-6 Re/Os phase diagram

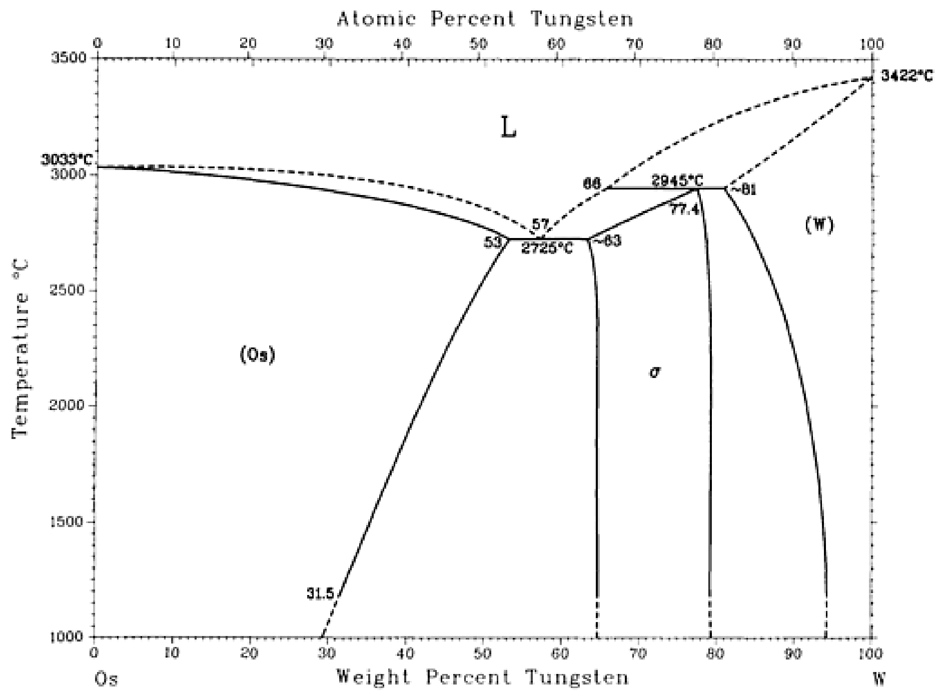

Figure 4.2-7 W/Os phase diagram

The second area of concern for irradiated materials is volumetric swelling caused by formation of defect clusters in the material, which causes vacancies and interstitials, and nuclear reactions that produce hydrogen and helium gas.

The applicant provided irradiation data for tungsten and W-25Re alloy to high fast fluences [ ] in the submittal. These data can be seen in Figure 4.2-8. These data demonstrate that the radiation swelling at [ ] will be a maximum of [ ], corresponding to a length change of [ ]. The staff noted that the lowest temperature at which the swelling data was taken was 430°C. The staff is aware of some materials such as SiC that have more severe swelling at lower temperatures than at high temperatures since vacancies and interstitials are not annealed out as rapidly at low temperatures. The staff asked in RAI-04 that the applicant provide an estimate of the tungsten absorber at normal operating conditions and, if this temperature is less than 430°C, that data be provided to justify the assumption of [ ] over the expected range of operating temperature. The applicant responded (Reference 2) that in evaluating the absorber swelling, [ ] would be used based on that measured in the tungsten at [ ]. This is conservative since the maximum tungsten temperature is expected to be [ ] swelling in pure tungsten of [ ], and swelling appears to decrease with rhenium additions due to transmutation. The applicant addressed the concern over low temperature swelling with an argument that swelling in refractory metals such as tungsten only occurs where the temperature is greater than 0.3 times the melting temperature. Based on the [ ] assumptions used by the applicant and inherent reduction in swelling with rhenium additions, the staff concluded that the applicant's response is acceptable.

Furthermore, the staff agrees that the swelling in tungsten is not expected to have a large contribution due to hydrogen and helium formation. Calculations have been performed by

Westinghouse that demonstrate very little helium is formed in irradiated tungsten and that very high temperature is necessary to allow helium diffusion and gas bubble formation in tungsten.

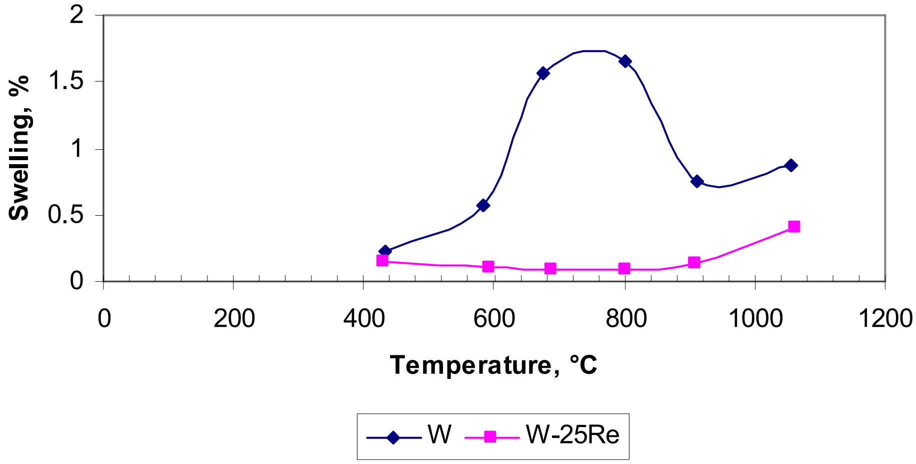

Figure 4.2-8 Radiation swelling in tungsten and W-25Re

4.2.5. Stainless Steel Properties

No material properties for stainless steel were provided in WCAP-16943-P (Reference 1). The staff asked in RAI-01 what type of stainless steel is used in the cladding of the GRCA rodlet. The applicant responded (Reference 2) that cold-worked stainless steel [ ] is used.

The staff also asked in RAI-03 that the applicant provide data that show the stainless steel cladding will have sufficient ductility up to the maximum target fluence level because some stainless steels [ ] are susceptible to stress corrosion cracking (SCC).

The staff also asked that the applicant provide any stainless steel properties that are used in the mechanical design evaluations. The applicant provided in response to RAI-03 the mechanical properties of stainless steel cladding used in the mechanical design evaluations. Since the gap between the sleeve and the cladding is not expected to close, and therefore cladding stresses are expected to be low with no plastic strain expected, there are no ductility requirements for the stainless steel cladding. Sample calculations that demonstrate no gap closure and thus support this assertion are discussed in Section 3.0 of this report.

The staff concluded that since [ ] is known to be compatible for various reactor applications and the stainless steel cladding is not expected to experience swelling-induced strains from the absorber material, the use of stainless steel for the cladding material is acceptable.

4.2.6. [ ] Properties

[ ] is used to sleeve the tungsten. The applicant states in Section 5.2.3 of WCAP-16943-P that [ ] will have [ ] ductility at the maximum expected fluence. No material properties for [ ] were provided in WCAP-16943-P. The staff asked in RAI-03 that the applicant provide data that show the [ ] sleeve will have

sufficient ductility up to the maximum target fluence level because the sleeve is calculated to result in a small amount of plastic deformation [ ]. The staff also asked that the applicant provide any [ ] properties that are used in the mechanical design evaluations.

The applicant responded by providing mechanical property data for [ ] that demonstrated that total elongation was greater than [ ]. However, these total elongation data do not preclude potential failure at strains less than [ ].

PNNL has examined failure data for Zircaloy-4 cladding that demonstrated that total elongation is a poor predictor of failure strains and that cladding has failed with less than [ ] even though measured total elongation was [ ]. Examination of these data has suggested that uniform elongation data better predicts failure strains. A similar observation has been made in the automotive industry where uniform elongation is used to predict failure probability in the deformation of body panels from welded steel sheet (Reference 18). Therefore, the staff cannot confirm that [ ] can [ ] without failure at the fluences expected for the GRCA. However, the staff concludes that even if the [ ] sleeve were to fail, the tungsten would remain within the sleeve as long as [ ] stainless steel cladding does not fail. The cladding is not expected to fail because it is not calculated to experience any plastic deformation and only small elastic strains due to the low pressure differential between the internal GRCA and external system pressures. Accordingly, the staff concludes the [ ] performance is acceptable due to the double encapsulation design for the GRCA.

4.3. <u>Mechanical Design Bases And Evaluation</u>

In addition to the design conditions considered under the ASME Code Section III, specific GRCA evaluations have been provided in WCAP-16943-P. These design considerations and the evaluations done to ensure that they are met are discussed in the following sections.

4.3.1. Internal Pressure and Cladding Stresses During Normal, Transient, And Accident Conditions

The applicant states that the design of the GRCA rods provides sufficient cold void volume to accommodate internal pressure increase during operation and that very little additional gas will be released by the tungsten absorber. The applicant further stated that no closure of the sleeve-to-clad gap would result at either cold or hot operating conditions at end-of-life. The staff was not able to confirm these assertions based on the information provided in Reference 1. The staff asked in RAI-06 that a sample calculation be provided of the expected component temperatures, including the assumed gamma and neutron heating values, such that an audit calculation could be performed to verify no sleeve-to-clad gap closure and the rod internal pressures as calculated by Westinghouse.

The applicant provided rodlet component dimensions and tolerances and linear heat rates for each component. The applicant stated that the [

]. The applicant also provided expected temperature values for each component. The applicant assumed the cladding surface temperature was equal to the saturation temperature of water at the system pressure [ ]. Since Westinghouse demonstrated that there will be no boiling on the GRCAs, as discussed in Sections 4.2 and 4.3 of Reference 1, this is conservative. The staff did a calculation to verify these temperatures. The average temperatures calculated by PNNL for each component are shown with the Westinghouse temperatures in Table 4.3-1. This table shows that the Westinghouse temperature calculations are slightly conservative (higher) relative to the staff calculations.

Table 4.3-1 Westinghouse and PNNL expected GRCA component temperature

	Westinghouse temperature	PNNL temperature
Tungsten	[ ]	[ ]
Sleeve	[ ]	[ ]
Cladding	[ ]	[ ]

The staff calculated the gas temperature inside the sleeve and cladding plenums. These temperatures were calculated to be [ ] for the cladding plenum and [ ] for the sleeve plenum. Both the cladding and the sleeve are pressurized to [ ]. The calculated temperature increase at operating conditions leads to in-reactor pressures of [ ] for the cladding and sleeve plenums respectively. These pressures are well below the system pressure of 2250 psi and, therefore, the staff confirmed the applicant's claim that the GRCA rods provide sufficient cold void volume to accommodate internal pressure increase during normal operation.

The applicant also stated that there will not be significant hoop stress in any of the components due to sleeve-to-cladding gap closure during reactor operation because the thermal expansion of the cladding is greater than that of the sleeve, both of which are greater than that of the absorber.

The thermal expansion of each component is plotted in Figure 4.3-1. It can be seen that the thermal expansion of the cladding (Reference 19) is greater than that of the sleeve (Reference 20), which is greater than that of the absorber.

However, this evaluation does not ensure that there will be no stress between the components because the inner component temperatures will be greater than the outer component temperatures. Therefore, the staff considered it necessary to calculate each component temperature and the associated thermal expansion in order to determine if there will be any interference.
[

]

Figure 4.3-1 Thermal expansion for cladding, sleeve and absorber

In the response to RAI-06 the applicant performed a sample calculation [ ], which demonstrated no gap closure for the cladding/sleeve gap [ ]. The staff has reviewed these calculations and finds them to be acceptable. Based on the methodology and results of calculated temperatures and thermal expansion results presented in response to RAI-06, the staff concludes that the sleeve-to-clad gap will remain open [ ]. As noted previously in Section 4.2.6, the staff was not able to confirm that the [ ] limit will not result in failure of the [ ] sleeve. However, the double encapsulation of the tungsten and the calculation in the response to RAI-06 demonstrating no gap closure provides reasonable assurance that the sleeve will not fail under operating conditions.

Finally, the applicant has performed an analysis to determine that there will be no boiling along the length of the rodlet. These analyses are discussed in Sections 4.2 and 4.3 of Reference 1. The staff reviewed these calculations and agrees with the applicant's basis for concluding that they demonstrate the cladding will operate at sufficiently low temperatures such that it has adequate strength to resist cladding collapse.

For the reasons above, the staff finds the methodologies given in Reference 1 (as supplemented with the response to RAI-06) are acceptable for evaluating internal pressure and cladding collapse.

4.3.2. Irradiation Stability Of The Absorber Material, Taking into Consideration Gas Release And Swelling

The applicant references industry experience of swelling that demonstrates that there will be little radiation swelling in the tungsten absorber. The staff confirmed the maximum swelling assumed by Westinghouse of [ ] appears to be conservative based on available data. The applicant notes that there could be minor cracking in the absorber material due to small clearances between the absorber and the sleeve, although there is not expected to be significant absorber relocation. The applicant also notes that very little gas will be produced in the absorber.

The staff reviewed the transmutations caused by neutron capture and radioactive decay of the tungsten GRCAs and agrees with the applicant's assertion that very limited gaseous product is produced. The staff therefore concludes that this limited gas release combined with limited radiation swelling of the absorber prevents large scale absorber material relocation.

4.3.3. Absorber Material Swelling

The applicant references industry experience of irradiation swelling of tungsten and claims that there will be sufficient diametrical and end clearances to accommodate absorber volumetric swelling and relative thermal expansion between the sleeve and cladding and end plugs such that the cladding will not be mechanically loaded. The analyses discussed in Section 4.3.1 of this SER demonstrate that the cladding will not be loaded diametrically (no plastic strain) and that the strain in the sleeve will not exceed [ ]. The predicted swelling of tungsten is expected to cause the tungsten to expand axially [ ]. The sleeve upper plenum length of [ ] is sufficient not to cause any axial loading on the sleeve end caps. The cladding upper plenum length of [ ] is more than enough to accommodate any differential expansion between the cladding and the sleeve.

Based on the available gap between the sleeve and surrounding components in comparison to the predicted volumetric changes, the staff concludes that radiation swelling will not result in failure of the sleeve or cladding.

4.4. <u>Thermal-Hydraulic Design Bases And Evaluation</u>

Specific GRCA thermal-hydraulic design bases and evaluations have been provided in WCAP-16943-P. The rodlet linear heating rate of the GRCA is a key parameter in the thermal evaluations. Compared to the typical RCCA (silver-indium-cadmium), the heating rate in the tungsten GRCA is about 40% lower. Based on this, it is expected that any analyses for typical RCCA would be more limiting than for the tungsten GRCA. Nevertheless, these evaluations have been performed by the applicant for the tungsten GRCA.

These design bases and the evaluations done to ensure that they are met are discussed in the following sections.

4.4.1. Maximum Absorber Temperature

This design basis is that the maximum temperature of the absorber material shall not exceed the melting temperature during normal operation (Condition I) and accident (Condition II) conditions. Thermal hydraulic analysis is used to calculate the maximum absorber temperature using the maximum $F_{\Delta h}$, maximum fuel rod F_Q, and overpower conditions (Condition II). This analysis demonstrates that the absorber, sleeve, and cladding will not exceed their melting temperatures.

In order to calculate realistic normal operation (Condition I) component temperatures, the thermal hydraulic analysis uses lower values of $F_{\Delta h}$ and F_Q to account for the local power suppression due to the presence of the absorber.

The staff concludes that this evaluation demonstrates that the tungsten GRCAs will not exceed their melting temperatures for normal and AOO conditions and is therefore acceptable.

4.4.2. No Surface Boiling Within the Dashpot Region

This design basis is that there is no surface boiling on the absorber cladding in the dashpot region during normal operation (Condition I).

The applicant notes that the GRCA has a stainless steel type [ ] tip spacer, which is the only portion of the GRCA that is in the dashpot region. This spacer has a [ ] heating rate than the sleeve/absorber material. The applicant performed a thermal hydraulic evaluation of the most limiting condition when the GRCA is fully inserted in the fuel assembly. This analysis showed that there was no surface boiling on the absorber cladding.

The staff therefore concludes that the use of a stainless steel type [ ] tip spacer is an acceptable way to prevent overheating in the dashpot region based on the applicant's thermal-hydraulic analysis.

4.4.3. No Bulk Boiling In the Guide Thimble Tubes

This design basis is that there is no bulk boiling in the guide thimble tubes.

The applicant performed a thermal hydraulic evaluation of the most limiting condition, when the GRCA is fully inserted in the fuel assembly. This analysis showed that there was no bulk boiling in the guide thimble tubes.

The staff concludes that, based on the lower heating rate for tungsten GRCA versus the AIC RCCA designs, the RCCA analysis bounds the GRCA design.

4.4.4. Bypass Flow In Guide Thimble Tubes

This design basis is that the maximum guide thimble tube bypass flow combined with other bypass flow paths must be below 5.9%.

The applicant references the DCD for AP-1000, which shows the total core bypass is less than 5.9% using worst case drawing tolerances and uncertainties in pressure losses. The applicant states that the GRCA design has no impact on these calculations, and therefore the previous calculations are still applicable.

The staff confirmed that the rodlet outer diameter dimension for the enhanced GRCA design and DCD RCCA design are identical. Therefore the bypass flow in the guide thimble tubes is not changed.

4.4.5. Post-LOCA Survivability

The applicant provided an assessment of the GRCA to show that, with regard to post-LOCA survivability, the tungsten GRCA is no more limiting than the RCCA or the previous GRCA.

The primary concern is the potential for the formation of a eutectic between the stainless steel cladding and the zirconium alloy thimble tubes. Because the same materials are used and the cladding thickness of the GRCA [ ], the assessment for the RCCA will bound the GRCA.

The applicant also assessed the possibility of the formation of a low-melting temperature eutectic between tungsten and the structural materials in the GRCA. Table 4.5-1 lists the composition of the stainless steel cladding and the [ ] sleeve (Reference 21).

Table 4.5-1 Elemental composition of cladding and sleeve

Component	**Material**	**Composition (wt%)**
Cladding	[ ]	[ ]
Sleeve	[ ]	[ ]

The applicant provided phase diagrams for W/Ni and W/Fe since nickel (Ni) and iron (Fe) are the main components in these materials. Neither of these phase diagrams showed the presence of any low temperature eutectics. To further ensure the lack of any low temperature eutectics, the staff looked at the binary phase diagrams for tungsten and any element from Table 4.5-1 with more than 1 wt%. The phase diagrams for W/Cr, [ ] all showed no low temperature eutectics (Reference 17). No phase diagram for W/Mn was found, but a note was found that tungsten is insoluble in liquid Manganese (Mn). This indicates that there will be no low temperature eutectics for the W/Mn system.

Based on these analyses, the staff concludes that the assessment of the GRCA for post-LOCA survivability is acceptable.

4.4.6. Operational Monitoring Program

In order to validate the expected performance of the tungsten GRCA design, the applicant will monitor two basic aspects of the design over its lifetime. These aspects are startup physics testing and post-irradiation examination of the hardware.

For startup physics testing, the applicant will monitor rod drop time to identify any trends. The applicant notes that there are no requirements on rod drop time for GRCA since they are not required for safe shutdown of the reactor. The applicant will also measure GRCA bank worth during each cycle startup to confirm the adequacy of the nuclear design calculations.

For post-irradiation examination, Westinghouse has various pool-side and hot cell exams planned. The staff requested in RAI-08 that the applicant describe in detail a schedule of the planned non-destructive and destructive examinations that will be performed on the GRCA. The applicant responded (Reference 2) that non-destructive examination on GRCAs would be performed for every cycle following the first cycle to measure, at a minimum, wear and cladding outer diameter. GRCAs would be examined from different core locations. GRCAs with the highest fluence would be examined and if any clad diameter plastic deformation beyond normal uncertainties in diameter measurements was observed, a destructive examination would be considered to evaluate the tungsten and sleeve swelling or other cause for unexpected clad deformation.

If no significant clad deformation was observed after approximately [ ], then destructive examination would be performed to confirm performance of the GRCA components.

5.0 Staff Conclusions

The staff has completed its review of the Enhanced GRCA Rodlet Design as described by WCAP-16943-P and supplemented by RAI responses and concludes that the applicant has demonstrated that the GRCAs will not impede the functions of the systems and structures surrounding them. The staff notes that, while the GRCAs do not have a safety function and therefore do not have the same regulatory requirement burden that RCCAs do, the applicant has demonstrated compliance with GDCs 27 and 35 by demonstrating that the GRCAs will not fail in such a way as to impact the surrounding fuel assemblies and RCCAs. The staff therefore approves the use of the Enhanced GRCA Rodlets as described by WCAP-16943-P and supplemented by RAI responses in the AP1000 design.

6.0 Conditions and Limitations

1. Post-irradiation examinations as described in response to RAI-08 (ML102460149) are to be conducted. As stated, destructive examination may be conducted if non-destructive examinations indicate performance is different than expectations. As a condition of the staff's acceptance and approval of this topical report, if plastic deformation beyond measurement uncertainties is observed in the GRCA stainless steel cladding from non-destructive examinations, the licensee must submit to the NRC the results of any examinations along with justification for continuing to operate with unexpected deformation of GRCA cladding.

2. To be consistent with the approximate analyzed lifetime of [ ] the [ ] must be tracked and confirmed to remain below the value in [ ], and thereby within the analyzed lifetime.

7.0 REFERENCES

1. M.E. Conner, M.J. Hone, J. Long, R. Pomirleanu, and G. Wikmark. 2008. *Enhanced GRCA Rodlet Design*, WCAP-16943-P, Westinghouse Electric Company, Monroeville, PA (ML081680553).

2. J.A. Gresham, 2010. *Response to NRC Requests for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design (Proprietary)*, LTR-NRC-10-25 P-Enclosure, Westinghouse Electric Company, Monroeville, PA (ML102460149).

3. Knolls Atomic Power Laboratory. 2002 *16th chart of the nuclides*.

4. Sargent-Welch Scientific Company. 1979. *Periodic Table of the Elements*.

5. G.K. White, R.B. Roberts, High Temperatures-High Pressures, vol. 15, p. 321, 1983.

6. R.R. Kirby, High Temperatures-High Pressures, vol. 4, p. 459, 1972.

7. F.C. Nix, D. MacNair, Physical Review, vol. 61, p. 74, 1942.

8. Y.S. Touloukian et al., “Thermophysical Properties of Matter. Vol. 12: Thermal Expansion – Metallic Elements and Alloys,” IFI/Plenum, New York-Washington, 1975.

9. G.K. White and S.J. Collocott, Journal of Physical Chemistry Reference Data, vol. 13, no. 4, p. 1251,1984.

10. A. Cezairliyan and J.L. McClure, Journal of Research National Bureau Standards, vol. 75A, p. 283, 1971.

11. C.Y. Ho, R.W. Powell and P.E. Liley, Journal of Physical Chemistry Reference Data, vol. 1, p. 279, 1972.

12. L. Binkele, “Significance of discrete Lorenz function levels at high temperatures resulting from new metallic conductivity measurements,” High Temperature-High Pressures, vol. 18, pp. 599-607, 1986.

13. B.E. Neimark and L.K. Voronin “Thermal conductivity, electrical resistivity and emissivity of refractory metals at high temperatures,” Teplofiz.Vys.Temp. vol. 6 pp. 1044-1056, 1968.
14. L.A. Pigal'skaya et al., “Thermal diffusivity of tungsten at high temperatures,” High.Temperature, vol. 4, pp. 290-292, 1966.

15. Y.S. Touloukian et al., “Thermophysical Properties of Matter. Vol. 1: Thermal Conductivity – Metallic Elements and Alloys,” IFI/Plenum, New York-Washington, 1970.

16. Lowrie R. and Gonas A.M., Journal of Applied Physics, vol.38, p. 4505, 1967.

17. The Materials Information Society. *ASM Handbook: Volume 3, Alloy Phase Diagrams*,

18. Davies, R. W., G.J. Grant, H. Eddie Oliver, M.A. Khaleel, and M.T. Smith, “Forming-Limit Diagrams of Aluminum Tailor-Welded Blank Weld Material,” Metallurgical and Materials Transactions A, Vol. 32A, February 2001, pp 275-283.

19. A.P. Miller, A. Cezairliyan, High Temperatures-High Pressures, v23, p205 (1991).

20. U.S. Department of Defense. 1998. *Military Handbook: Metallic Materials and Elements for Aerospace Vehicle Structures*, MIL-HDBK-5H, p.6-51.

21. AMS International. 2001. *Advanced Materials & Processes*, December 2001, Volume 159, No. 12.

22. NUREG-0800, Section 4.2, “Fuel System Design,” March 2007 (ML 070740003).

23. S.F. Mughabghab, "Neutron Cross Sections – Neutron Resonance Parameters and Thermal Cross Sections", Vol. 1, Part B (Z= 61 – 100). Academic Press Inc. (Harcourt Brace Jovanovich, Publishers) New York (1984).

24. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," Ouisoumen M., et al, August 2004 (ML042250345).

25. KAPL-P-000069, "Neutron Total and Capture Cross Section Measurements of Tungsten," Werner C. J., et al, June 1998.