WCAP-16943-NP-A
Revision 0

September 2012

Enhanced GRCA Rodlet Design

Westinghouse

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September 2012

Authors:

M. E. Conner
M. J. Hone
J. Long
R. Pomirleanu
G. Wikmark

Prepared by:
N. T. Brichacek*
Fuel Engineering Licensing

Approved by:
K. W. Cummings, Manager*
Fuel Engineering Licensing

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Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066

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Section A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 12, 2012

Mr. James A. Gresham, Manager
Regulatory Compliance
Westinghouse Electric Company, LLC
Suite 428, 1000 Westinghouse Drive
Cranberry Township, PA 16066

SUBJECT: WESTINGHOUSE ELECTRIC COMPANY'S FINAL TOPICAL REPORT SAFETY EVALUATION FOR WCAP-16943, "ENHANCED GRAY ROD CLUSTER ASSEMBLY RODLET DESIGN"

Dear Mr. Gresham:

The U.S. Nuclear Regulatory Commission staff prepared a final Topical Report Safety Evaluation (TRSE) for WCAP-16943, "Enhanced Gray Rod Cluster Assembly Rodlet Design," in support of the AP1000 post – licensing activities submitted by Westinghouse Electric Company (WEC).

The staff requests that WEC publish the applicable version(s) of the TRSE listed above within 1- month of receipt of this letter. The accepted version of the topical report shall incorporate this letter and the enclosed TRSE and add an "-A" (designated accepted) following the report identification number.

The staff has found that WCAP-16943 is acceptable for referencing in licensing applications for AP1000 designed pressurized water reactors to the extent specified and under the limitations delineated in the topical report and in the enclosed final TRSE. The final TRSE defines the basis for our acceptance of the topical report.

If the U.S. Nuclear Regulatory Commission's (NRC) criteria or regulations change, so that its conclusion that the TRSE is acceptable is invalidated, WEC and/or the applicant referencing the TRSE will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the TRSE without revision of the respective documentation.

Prior to placing the public version of this document in the public document room the staff requests that WEC perform a final review of the TRSE for proprietary or security-related information not previously identified. If you believe that any additional information meets the criteria, please identify such information line by line and define the basis pursuant to the criteria established in Title 10 of the *Code of Federal Regulations* Part 2, Section 390.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosure, this document is "DECONTROLLED."

If after a 10–day period, you do not request that all or portions of the TRSE be withheld from public disclosure, the TRSE will be made available for public inspection through the NRC Public Document Room and the Publicly Available Records component of NRC's Agencywide Documents Access and Management System and placed on the NRC's public web page for this application.

If you have any questions or comments concerning this matter, please contact me at 301-415-6715 or via e-mail address at Bruce.Bavol@nrc.gov.

Sincerely,

Bruce M. Bavol, Project Manager
Licensing Branch 4
Division of New Reactor Licensing
Office of New Reactors

Docket No.: 52-006

Enclosure:
As stated

cc w/o encl.: See next page

DC Westinghouse - AP1000 Mailing List (Revised 07/03/2012)
cc:

Ms. Sara Barczak
Southern Alliance for Clean Energy
P.O. Box 8282
Savannah, GA 31401

Paul M. Bessette
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. Barton Z. Cowan, Esquire
Eckert Seamans Cherin & Mellott, LLC
600 Grant Street, 44th Floor
Pittsburgh, PA 15219

Mr. Paul Gaukler
Pillsbury, Winthrop, Shaw, Pittman
2300 N Street, NW
Washington, DC 20037

Ms. Sophie Gutner
P.O. Box 4646
Glen Allen, VA 23058

Ms. Shannon Bowyer Hudson
Office of Regulatory Staff
State of South Carolina
1401 Main Street
Suite 900
Columbia, SC 29201

Rita Kilpatrick
250 Arizona Ave.
Atlanta, GA 30307

Mr. Tony Robinson
AREVA NP, Inc.
3315 Old Forest Road
Lynchburg, VA 24501

Mr. Paul A. Russ
Director, AP1000 Licensing
Westinghouse
1000 Westinghouse Drive
Cranberry Township, PA 16066

Mr. Gary Wright, Director
Division of Nuclear Facility Safety
Illinois Emergency Management Agency
1035 Outer Park Drive
Springfield, IL 62704

DC Westinghouse - AP1000 Mailing List

Email
agaughtm@southernco.com (Amy Aughtman)
alsterdis@tva.gov (Andrea Sterdis)
amonroe@scana.com (Amy Monroe)
APAGLIA@Scana.com (Al Paglia)
APH@NEI.org (Adrian Heymer)
awc@nei.org (Anne W. Cottingham)
bgattoni@roe.com (William (Bill) Gattoni))
Bill.Jacobs@gdsassociates.com (Bill Jacobs)
BrinkmCB@westinghouse.com (Charles Brinkman)
Carellmd@westinghouse.com (Mario D. Carelli)
cberger@energetics.com (Carl Berger)
crpierce@southernco.com (C.R. Pierce)
CumminWE@Westinghouse.com (Edward W. Cummins)
cwaltman@roe.com (C. Waltman)
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
doug.ellis@shawgrp.com (Doug Ellis)
eddie.grant@excelservices.com (Eddie Grant)
erg-xl@cox.net (Eddie R. Grant)
ewallace@nuscalepower.com (Ed Wallace)
fbelser@regstaff.sc.gov
gcesare@enercon.com (Guy Cesare)
George.Madden@fpl.com (George Madden)
gzinke@entergy.com (George Alan Zinke)
james1.beard@ge.com (James Beard)
jcsaldar@bechtel.com (James Saldarini)
jerald.head@ge.com (Jerald G. Head)
jflitter@regstaff.sc.gov
jim@ncwarn.org (Jim Warren)
john.elnitsky@pgnmail.com (John Elnitsky)
Joseph_Hegner@dom.com (Joseph Hegner)
jrappe@nuscalepower.com (Jodi Rappe)
kinneyrw@dhec.sc.gov (Ronald Kinney)
KSutton@morganlewis.com (Kathryn M. Sutton)
kwaugh@impact-net.org (Kenneth O. Waugh)
lchandler@morganlewis.com (Lawrence J. Chandler)
lindg1da@westinghouse.com (Don Lindgren)
maria.webb@pillsburylaw.com (Maria Webb)
marilyn.kray@exeloncorp.com
mark.beaumont@wsms.com (Mark Beaumont)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
maurerbf@westinghouse.com (Brad Maurer)
media@nei.org (Scott Peterson)
melto1ma@westinghouse.com (Michael Melton)

DC Westinghouse - AP1000 Mailing List

Mitch.Ross@fpl.com (Mitch Ross)
MSF@nei.org (Marvin Fertel)
nirsnet@nirs.org (Michael Mariotte)
nscjiangguang@sina.com (Jiang Guang)
Nuclaw@mindspring.com (Robert Temple)
patriciaL.campbell@ge.com (Patricia L. Campbell)
paul.gaukler@pillsburylaw.com (Paul Gaukler)
Paul.Jacobs@fpl.com (Paul Jacobs)
Paul@beyondnuclear.org (Paul Gunter)
pbessette@morganlewis.com (Paul Bessette)
pshastings@generationmpower.com (Peter Hastings)
Raymond.Burski@fpl.com (Raymond Burski)
rclary@scana.com (Ronald Clary)
rgrumbir@gmail.com (Richard Grumbir)
Richard.Orthen@fpl.com (Richard Orthen)
ritterse@westinghouse.com (Stanley E. Ritterbusch)
RJB@NEI.org (Russell Bell)
robert.kitchen@pgnmail.com (Robert H. Kitchen)
rong-pan@263.net (Pan Rong)
sabinski@suddenlink.net (Steve A. Bennett)
saporito3@gmail.com (Thomas Saporito)
sara@cleanenergy.org
sfrantz@morganlewis.com (Stephen P. Frantz)
shudson@regstaff.sc.gov (Shannon Hudson)
sisk1rb@wetinghouse.com (Rob Sisk)
smsloan@babcock.com (Sandra Sloan)
stephan.moen@ge.com (Stephan Moen)
Steve.Franzone@fpl.com (Steve Franzone)
strambgb@westinghouse.com (George Stramback)
Tansel.Selekler@nuclear.energy.gov (Tansel Selekler)
Timothy.Beville@nuclear.energy.gov (Tim Beville)
tom.miller@hq.doe.gov (Tom Miller)
tomccall@southernco.com (Tom McCallum)
TomClements329@cs.com (Tom Clements)
trsmith@winston.com (Tyson Smith)
Vanessa.quinn@dhs.gov (Vanessa Quinn)
vijukrp@westinghouse.com (Ronald P. Vijuk)
Wanda.K.Marshall@dom.com (Wanda K. Marshall)
wayne.marquino@ge.com (Wayne Marquino)
whorin@winston.com (W. Horin)

1.0 Introduction

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 Regulatory Criteria

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 Summary Of Technical Information

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/cm³	19.3 g/cm³ †
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 3380°C 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.

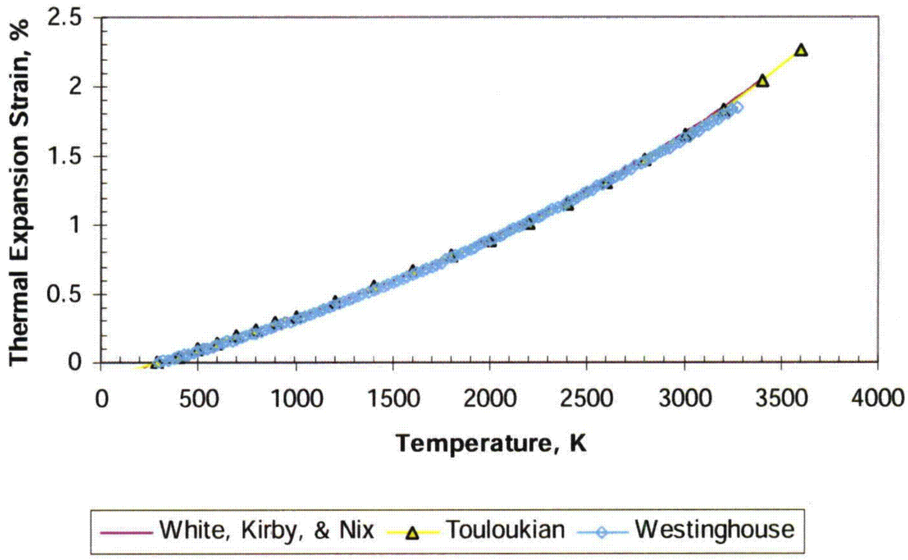

Figure 4.2-1 Thermal expansion of tungsten

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.

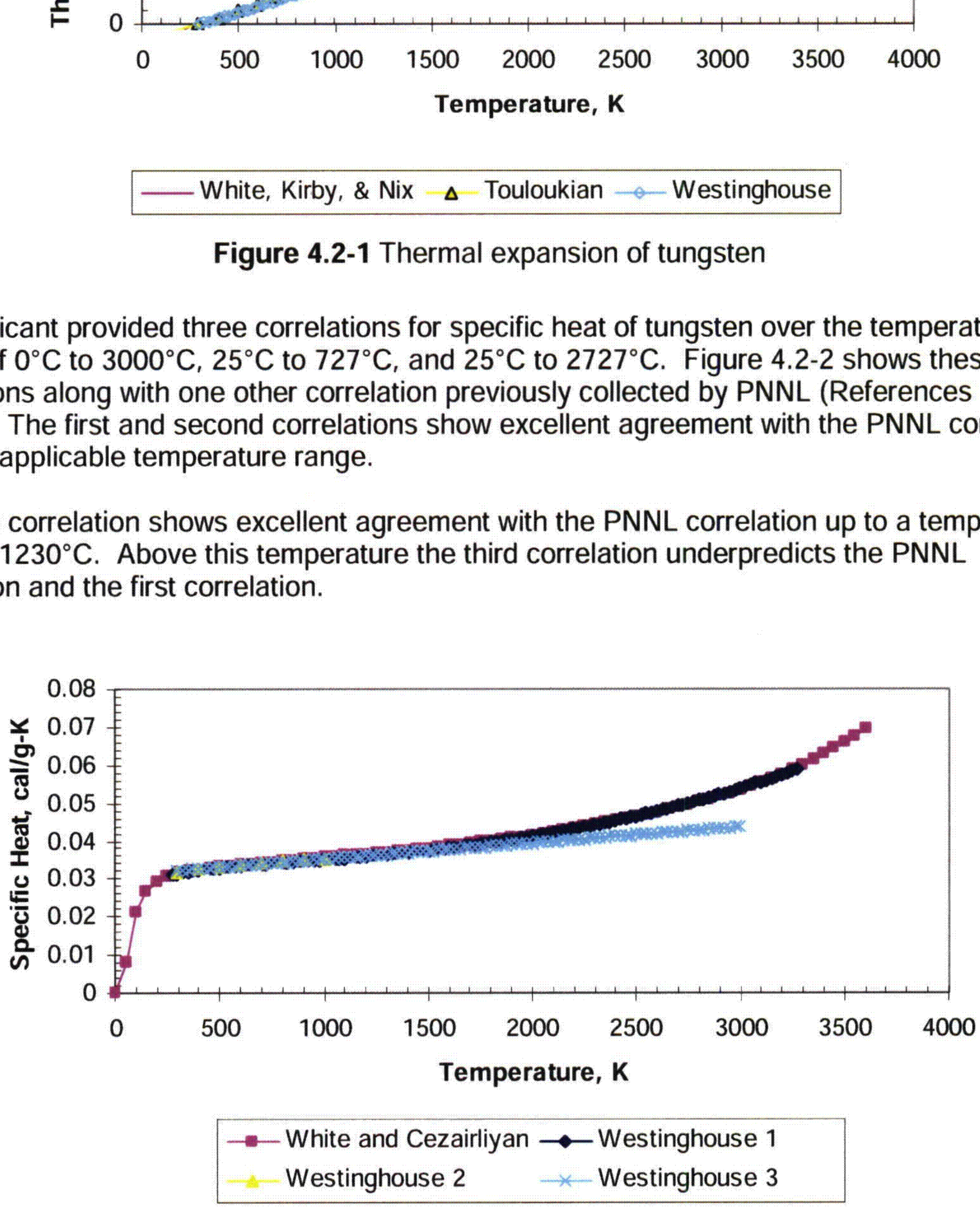

Figure 4.2-2 Specific heat of tungsten

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.

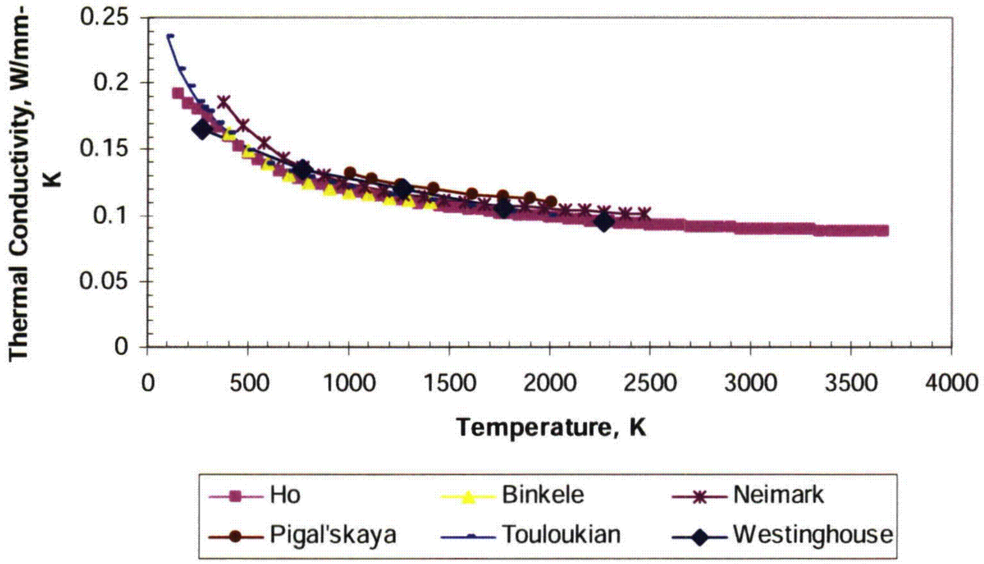

Figure 4.2-3 Thermal conductivity of tungsten

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.

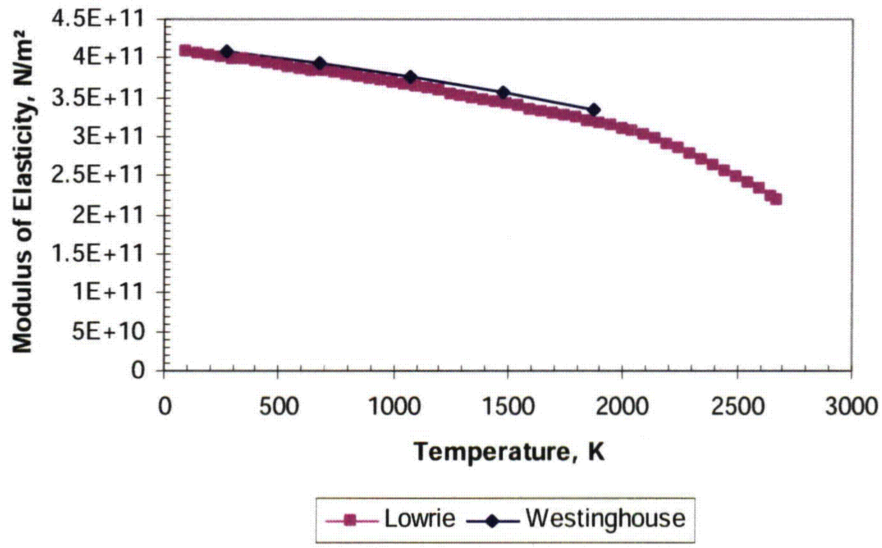

Figure 4.2-4 Young's modulus of tungsten

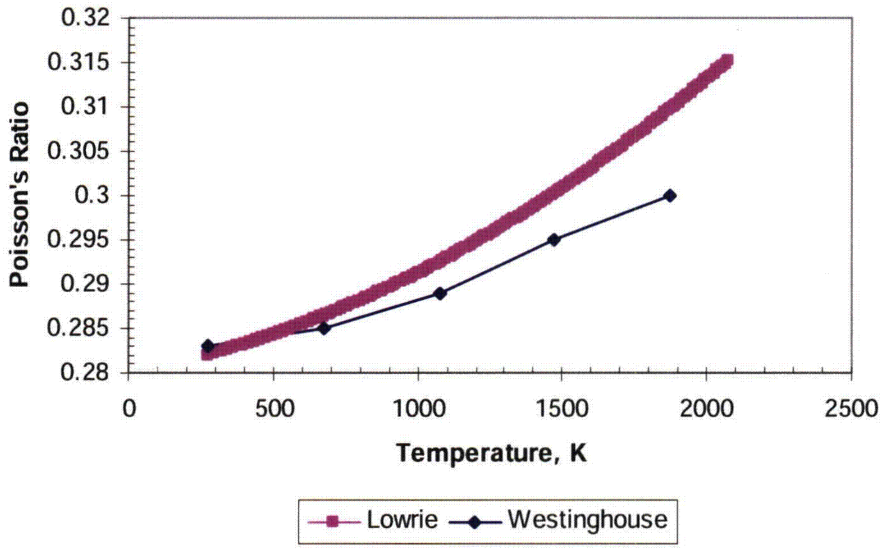

Figure 4.2-5 Poisson's ratio for tungsten (Note: expanded scale magnifies difference)

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.

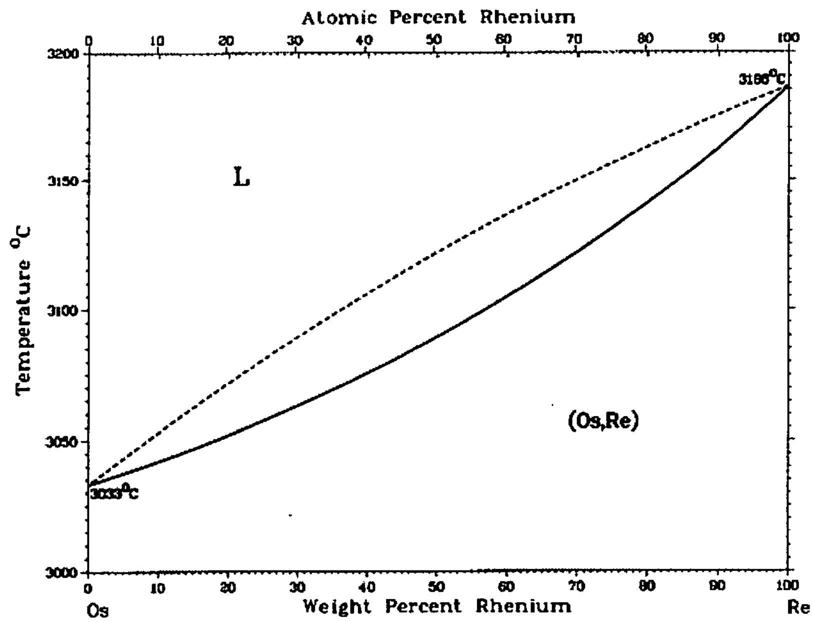

Figure 4.2-6 Re/Os phase diagram

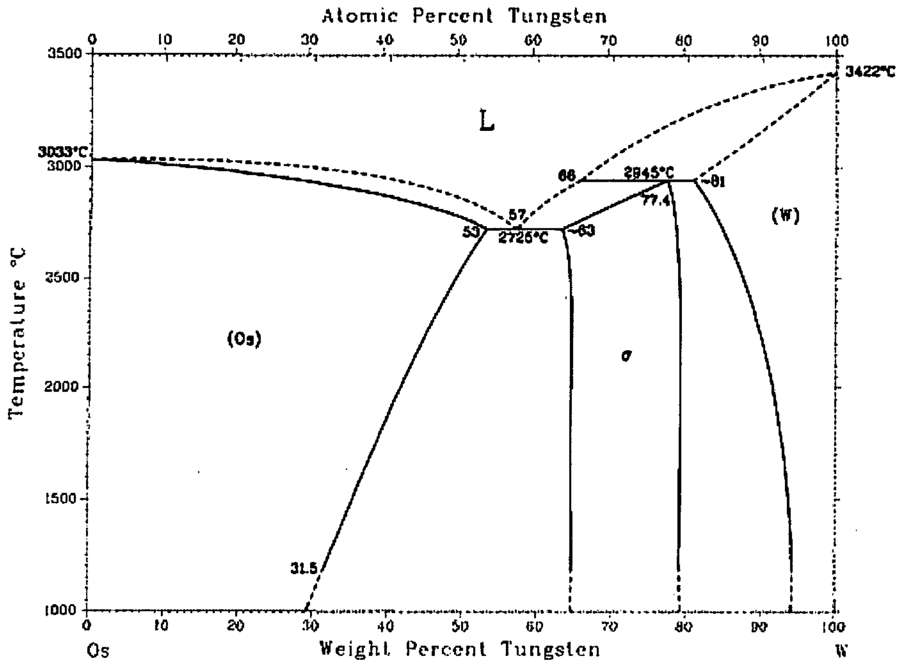

Figure 4.2-7 W/Os phase diagram

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.

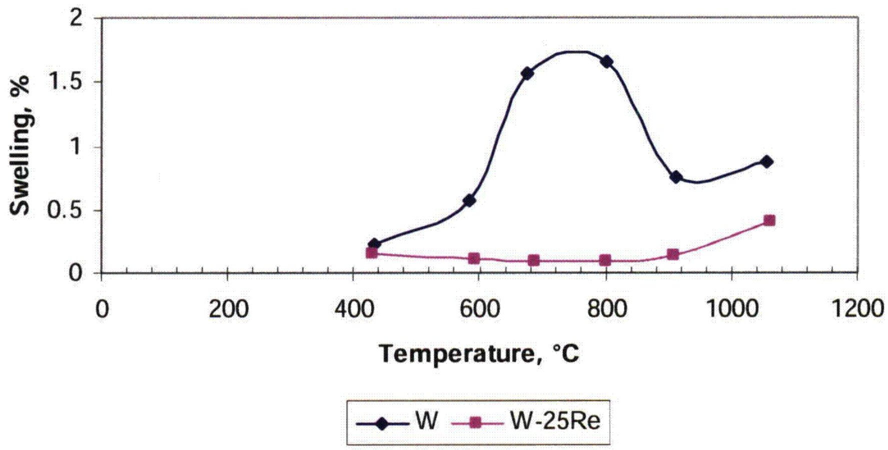

Figure 4.2-8 Radiation swelling in tungsten and W-25Re

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. Mechanical Design Bases And Evaluation

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.

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Section B

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: LTR-NRC-08-28

June 9, 2008

Subject: Submittal of WCAP-16943-P / WCAP-16943-NP, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" Topical Report (Proprietary/Non-proprietary)

Dear Ms. Eileen McKenna:

Enclosed are 5 Proprietary and 3 Non-Proprietary copies of WCAP-16943-P / WCAP-16943-NP, Westinghouse's "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" Topical Report submitted to the NRC for review and approval.

This licensing topical report describes an Enhanced AP1000 Gray Rod Cluster Assembly (GRCA) reactivity control rodlet intended to replace the Ag-Id-Cd (silver-indium-cadmium) GRCA absorber rodlet currently approved for use in the AP1000 Standard Plant. This topical introduces a new gray rod absorber material as an alternative to the use of Ag-Id-Cd. The GRCA function (i.e., for reactor control and operation), outer rodlet dimension, and overall reactivity worth are not changed from the previously approved design. The GRCAs are not required to assure safe shutdown.

As discussed with Mr. Rob Sisk, Manager, AP1000 Licensing and Customer Interface, this "Enhanced GRCA Rodlet Design" Topical Report is not part of the DCD Revision 16 package that is currently under review by the NRC. Instead, this topical is submitted separately in support of the AP1000 Core Reference Report anticipated to be submitted in 2010. As such, an approval need date of September 2009 is requested for NRC review of this topical, in order to support submittal of the Core Reference Report in 2010. It is also requested that the NRC provide an estimate of the man-power resources required for the review and a tentative date for the acceptance meeting.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-08-2433 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-08-2433 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Eileen McKenna, NRO
Joseph Donoghue, NRO
Anthony Mendiola, NRR
Phyllis Clark, NRO
Chris Van Wert, NRO
Jon Thompson, NRR

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Direct tel: 412/374-4643
Direct fax: 412/374-4011
e-mail: greshaja@westinghouse.com

Our ref: AW-08-2433

June 9, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Submittal of WCAP-16943-P / WCAP-16943-NP, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design," Topical Report (Proprietary)

Reference: Letter from J. A. Gresham to NRC, LTR-NRC-08-28, dated June 9, 2008

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-08-2433 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-08-2433 and should be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

AW-08-2433

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse) and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 9th day
of June, 2008.

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-NRC-08-28-Attachment, Submittal of WCAP-16943-P / WCAP-16943-NP, "Westinghouse Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design," Topical Report (Proprietary/Non-proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-08-28) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is for NRC review and approval.

This information is part of that which will enable Westinghouse to:

(a) Obtain generic NRC license approval for the Westinghouse Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design using a new gray rod absorber material as an alternative to that currently used in the AP1000 Standard Plant.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) The information requested to be withheld reveals the distinguishing aspects of a method and/or methodology which was developed by Westinghouse.

(b) Assist customers to obtain license changes.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar fuel design and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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Enhanced GRCA Rodlet Design

1.0 Purpose and Summary

1.1 Purpose

This topical report presents generic design and evaluation information for the introduction of an enhanced Gray Rod Cluster Assembly (GRCA) absorber rodlet by substituting tungsten (W) as the neutron absorber material in lieu of silver-indium-cadmium (Ag-In-Cd) for use in the current NRC approved GRCA design for the AP1000 reactor plant.

This report describes the basic absorber rodlet design using tungsten in a reduced worth control rod, i.e. GRCA. The focus of this topical and the accompanying evaluations is directed at the replacement tungsten absorber material that is internal and double encapsulated in the absorber rodlet, since the basic reduced worth control rod cluster design (i.e., outer rodlet diameter and other overall interface dimensions) remains the same as the previously NRC reviewed and approved GRCA design described in the AP1000 Design Control Document (DCD) (References 1 and 2). This topical report provides the proposed licensing basis for evaluating the enhanced GRCA design and, when approved, will serve as the basis for application of the enhanced GRCA rodlet design.

GRCAs using reduced worth control rods have been approved for use in the AP1000 and other nuclear reactor plants around the world. The introduction of tungsten as a low worth control rodlet absorber will enhance the mechanical and nuclear performance of a low worth control rod (i.e., increased lifetime and constant reactivity) relative to the standard Ag-In-Cd absorber. Tungsten control rodlets can be designed to match the existing NRC approved reduced worth control rods and as a result would not affect the operational characteristics of a reactor core using GRCAs.

1.2 SRP 4.2 Acceptance Criteria Road Map

NUREG 0800 Standard Review Plan (SRP) Section 4.2, "Fuel System Design," (Reference 8) provides the principal guidance that has been used for evaluating the acceptability of the enhanced GRCA reactivity control rod/rodlet design for use in the AP1000 reactor core. In the AP1000 DCD, the overall GRCA design was shown to meet the required design bases and limits in accordance with the Standard Review Plan. As stated in Section 1.1, the purpose of this topical report is to discuss changes in the GRCA rodlet design to incorporate the use of tungsten as the absorber material for the enhanced GRCA design instead of Ag-In-Cd. As such the focus of this report is directed at those aspects of the GRCA design, specifically the rodlet itself, that are impacted by this change. Other aspects of the GRCA design that are not impacted by the absorber material change and were already considered in the DCD will not be addressed in this report.

Section II.1.A.viii of the SRP discusses considerations for changes in control rod design. Specifically, the following items are indicated for review:

1. Changes in control rod configuration
2. Introduction of new materials
3. Changes in neutronics and mechanical lifetime
4. Changes in mechanical design
5. The ability to exclude water/coolant if water-soluble or leachable materials are used

All of these items are specifically addressed in this report. In addition, the specific criteria that are related to the rodlet design changes are addressed in the evaluations of the appropriate design bases.

The report is organized to cover the key SRP areas of design bases, design description, design evaluation, and testing inspection, and surveillance plans as follows:

Section 2.0 - Overall Design Description

Section 3.0 - Nuclear Design Description (which also includes an evaluation of analytical prediction capability)

Section 4.0 - New Material Property Description (including significant irradiation experience with tungsten)

Section 5.0 - Design Bases and Evaluation (Nuclear, Mechanical, and Thermal/Hydraulic)

Section 6.0 - Post-LOCA Survivability

Section 7.0 - Operational Monitoring Program

Based on a thorough review of the SRP 4.2 guidance, the following Table 1.0-1 presents an overview of the SRP acceptance criteria addressed for the introduction of the enhanced GRCA rodlet design, and the corresponding section(s) of the topical report in which the SRP criteria is addressed.

TABLE 1.0-1
STANDARD REVIEW PLAN SECTION 4.2
SUBSECTION II. - ACCEPTANCE CRITERIA

		SRP Subsection	Topical Report Section
Design Bases	**Fuel System Damage**	II.1.A.i. - Stress, Strain or Loading Limits on grids, GT, thimbles, fuel rods, control rods & other fuel system structural members	5.2, 5.3
		II.1.A.viii. (1) - Changes in control rod configuration	2.2
		II.1.A.viii. (2) - Introduction of new materials	4.1, 4.2, 4.3, 4.4, 4.5
		II.1.A.viii. (3) - Changes in neutronics and mechanical lifetime	5.1, 5.2
		II.1.A.viii. (4) - Changes in mechanical design	2.1, 2.2, 5.2, 5.3
		II.1.A.viii. (5) - The ability to exclude water/coolant if water-soluble or leachable materials are used.	4.3
Description & Design		II.2. Description & Design	2.0, 3.0, 4.0
Design Evaluation		II.3.C. - Analytical Prediction	3.1, 3.2, 3.3, 3.4, 3.6
Testing, Inspection and Surveillance Plans		II.3.D. - Test, Inspections, Surveillance	7.0

1.3 Summary

a. The results of the Mechanical Design evaluation performed for the enhanced GRCA rodlet design confirm that the design will perform its intended function and that the design bases are satisified.

b. The results of the Nuclear Design evaluation performed for the enhanced Gray Rod Cluster Assembly design confirm that:

- Standard nuclear design analytical models and methods accurately describe the neutronic behavior of the enhanced GRCA design.

- The enhanced GRCA nuclear design bases are satisfied.

c. The results of the Thermal and Hydraulic Design evaluation for the enhanced GRCA rodlet design confirm that the design bases are satisfied.

d. The evaluation of materials data available for the tungsten absorber material confirm that:

- Material properties of tungsten are understood and well documented such that appropriate design evaluations can be made.

- Significant irradiation experience exists such that the impacts of irradiation on key design characteristics can be assessed for AP1000 operating conditions.

2.0 Design Description of Enhanced GRCAs

2.1 Overall GRCA Design Considerations

Low worth Gray Rod Cluster Assemblies (GRCAs) are used in advanced nuclear reactors to provide a mechanical means of making relatively small adjustments in core reactivity. The use of GRCAs to mechanically adjust (or "shim") the core reactivity reduces the need to make small adjustments to the soluble boron concentration in the reactor coolant using the Chemical and Volume Control System (CVS). This reduces waste water processing required on a daily basis, and allows for simplification of the CVS design and operation. GRCAs are automatically repositioned as necessary to maintain a programmed average coolant temperature during baseload and load follow operation. The movement of GRCAs effectively compensates for reactivity effects due to fuel and burnable absorber depletion, power and temperature changes, and changes in the core xenon condition.

The reactivity control function of low worth GRCAs should be clearly distinguished from that of high worth, or black Rod Control Cluster Assemblies (RCCAs), which are primarily intended to provide rapid shutdown capability for the reactor core. GRCAs are dropped into the core during a reactor trip, but the negative reactivity associated with their insertion is small in comparison to the total negative reactivity available from the RCCAs, and is not needed to safely shut down the reactor. Therefore, GRCAs as considered in this report perform an operational reactivity control function, but are not required for shutdown as the result of a reactor trip.

The basic overall design of the enhanced GRCAs and current GRCAs, as well as RCCAs, is the same except for the absorber design in the inside of the absorber rodlets. The overall outline dimensions as well as the interfaces with the control rod drive mechanisms and the fuel assembly and guide thimbles are identical. In the case of the enhanced GRCAs and RCCAs, there are twenty four rodlets attached to a spider assembly that is attached to the control rod drive mechanism. The outer diameter of the rodlets of the enhanced GRCA and current RCCA/GRCA designs is identical. Table 2.1-1 show a comparison of key interface parameters of the GRCA and RCCA designs.

Table 2.1-1
COMPARISON OF OVERALL INTERFACE CHARACTERISTICS OF GRCA DESIGN WITH RCCA DESIGN

	RCCA (DCD R.16)	GRCA	
		Enhanced GRCA	DCD R.16
Rodlet O.D. (in.)	0.381	0.381	0.381
Overall Assembly Length	Same	Same	Same
Spider Design	Same	Same	Same
No. of Rodlets	24	24	12 SS / 12 Ag-In-Cd
Absorber Material	Ag-In-Cd	Tungsten (W)	Ag-In-Cd

2.2 Tungsten GRCA Rodlet Design

Figure 2.2-1 illustrates the Enhanced GRCA rodlet design. GRCA rodlets have an outside stainless steel cladding with top and bottom end plugs nearly identical to those of current RCCA rodlets.

Inside the cladding, the tungsten material of circular cross-section is further enclosed in a sleeve, which is sealed at the top and bottom. Thus, the tungsten absorber is double encapsulated. [

]^{a,c} The space between the cladding and sleeve is backfilled with an inert gas. The sleeve is held in place by a holddown device.

It should be noted that with the tungsten absorber, due to the optimum worth of this absorber material for GRCA application in the AP1000, all 24 rodlets attached to the spider are identical absorber rodlets. This compares with the GRCA design described in DCD Rev. 16, where only twelve of the 24 rodlets are absorber rodlets with a reduced diameter Ag-In-Cd absorber and the other 12 rodlets are inert stainless steel rodlets.

a,c

FIGURE 2.2-1 CROSS SECTION VIEWS OF ENHANCED GRCA RODLET

3.0 Nuclear Design Considerations

3.1 Methodology

Nuclear design calculations for GRCAs using tungsten as the absorber material are based on unit assembly results and homogenized group cross sections generated by the PARAGON lattice code (Reference 3). The PARAGON code is capable of accurately modeling the tungsten GRCAs. PARAGON is a two-dimensional neutron transport theory code that is typically used to model single or multi-assembly lattice calculations, including the presence of control rod assemblies when appropriate. The PARAGON 70-group neutron cross section library was updated to include appropriate tungsten and rhenium nuclide cross section data from ENDF/B-VI.8, using the same methodology described in Reference 3. Benchmarking of PARAGON against the industry standard MCNP code has been performed. An independent research effort by Rensselaer Polytechnic Institute (RPI) has also confirmed the accuracy of the ENDF/B-VI neutron capture and total cross sections for tungsten. More details regarding the ability of PARAGON to accurately model the tungsten GRCA are included in Section 3.6 of this report.

3.2 GRCA Reactivity Control Function

The targeted rod worth for GRCAs in a given plant design is dependent on the intended operating strategy for the plant, the kinetic characteristics of the reactor core, and the control rod pattern. The specific example discussed in this report is a GRCA design for the Westinghouse AP1000 reactor (Reference 1), which uses 16 GRCAs in the control rod pattern. The AP1000 plant is designed to be capable of performing many types of load follow maneuvers without requiring a change in the core soluble boron concentration during the maneuver. For AP1000, this targeted operational capability is the most limiting factor defining the minimum rod worth needed from the GRCAs. The maximum rod worth of GRCAs is not strictly limited, but GRCA designs that provide significantly more rod worth than required can affect the core power distribution more than necessary, as discussed in Section 3.3. Ideally, the GRCA rod worth should be maintained within a relatively narrow band about the optimum worth, in order to provide the intended control capability with minimal effect on power distribution. The tungsten based GRCA design described in Section 2.0 has been confirmed to meet the intended GRCA rod worth target for the AP1000 reactor, as described below.

The rod worth of each individual GRCA or bank of GRCAs will vary, depending on the core location, time in life, and fuel loading pattern of the core. Therefore, in order to allow for valid comparisons, GRCA rod worth has been studied on a relative basis, using single unit assembly geometry calculations performed with the PARAGON transport theory lattice code. Unit assembly rod worth calculations performed for the enhanced GRCA rodlet design were compared to the results of identical calculations for a reference GRCA design that is known to meet the required rod worth for AP1000. These calculations included modeling depletion of the GRCA absorber material for a neutron fluence equivalent to approximately 20 years of service in the AP1000 reactor.

The AP1000 reactor was initially designed to utilize GRCAs with four standard Silver-Indium-Cadmium (Ag-In-Cd) black absorber rodlets in each GRCA component, and was approved on that basis (Reference 1). The four rodlet Ag-In-Cd GRCA design was originally confirmed to provide the correct total GRCA rod worth needed to achieve the intended load follow capability for AP1000, without changing the soluble boron concentration. This GRCA design is thus considered the reference design, and was modeled in unit assembly geometry to establish the reference target rod worth for subsequent comparisons. Sensitivity studies were also performed on the reference design to determine how much GRCA worth could be allowed to decrease before boron concentration changes would become necessary to perform some of the required load follow sequences. While loss of the intended load follow capability would not cause any safety concerns or prevent continued baseload operation of the reactor, it could result in more complexity of operations and more waste water generation if a large load change maneuver became necessary when there is insufficient GRCA worth. The minimum acceptable GRCA rod worth established from the sensitivity studies was determined to be [ ]^{a,c} of the initial undepleted reference worth. This was used to establish the lower limit on GRCA worth, corresponding to the end of useful life time.

In Reference 2, an improved Ag-In-Cd based GRCA design was defined for the AP1000. The revised design reduced the Ag-In-Cd absorber diameter within each rodlet and spread the absorber material out among 12 of the 24 available rodlets in the control assembly. This change significantly improved the local pin power distribution characteristics for rodded operation and met the required GRCA rod worth. However, because of the high fluence accumulation rates associated with gray rod operation and the natural depletion characteristics of the absorber material, neither Ag-In-Cd GRCA design is capable of maintaining the minimum [ ]^{a,c} targeted reference rod worth for the desired service life time.

The enhanced GRCA rodlet design described in this report replaces the Ag-In-Cd primary absorber material with commercially pure tungsten. Tungsten has a significantly lower total neutron absorption cross section than Ag-In-Cd, which results in a naturally slower absorber depletion rate. In addition, the depletion of tungsten leads to the formation of ^{187}Re, which is a slightly stronger neutron absorber than natural tungsten. Over the life time of a tungsten GRCA, the buildup of ^{187}Re more than compensates for the depletion of the tungsten absorber nuclides. The result is that the rod worth of a tungsten GRCA will have a slightly increasing, but nearly flat trend with long term neutron exposure due to ^{187}Re buildup. This is illustrated in Figure 3.2-1, which plots the unit assembly GRCA rod worth for tungsten and both Ag-In-Cd GRCA designs discussed above at various stages of depletion during their life time. Figure 3.2-1 also shows the optimum targeted GRCA rod worth specific to AP1000, and the [ ]^{a,c} rod worth corresponding to the end of useful life time, when the ability to load follow without changing the soluble boron concentration can no longer be demonstrated for all intended scenarios. As shown in Figure 3.2-1, the overall change in rod worth for a tungsten GRCA is significantly smaller than the decrease in rod worth predicted for a similar period of depletion for both Ag-In-Cd GRCA designs. This relatively flat depletion worth characteristic of tungsten makes possible GRCA designs that utilize a minimum amount of absorber material in all 24 rodlets of the control assembly, without having any adverse effects on the GRCA useful life time. A further expected benefit of the relatively flat depletion worth characteristic for tungsten is that it minimizes the potential for long term formation of undesirable axial gradients in the GRCA absorber strength, which could also limit useful life time.

Figure 3.2-1 demonstrates that the AP1000 GRCA targeted rod worth requirement can be met with the enhanced tungsten based GRCA design, and that absorber depletion effects will not limit the useful service life time of the component.

a,c

FIGURE 3.2-1 COMPARISON OF ENHANCED GRCA ROD WORTH VARIATION DUE TO ABSORBER DEPLETION

3.3 GRCA Effect on Power Distribution

Power distribution effects from GRCA operation are minimized by minimizing the rod worth of the GRCAs, and by evenly distributing the absorber material in the GRCA to the maximum extent possible.

Figure 3.2-1 demonstrates that the tungsten based GRCA design matches the optimum targeted GRCA rod worth for AP1000 very closely. The relatively flat depletion worth curve for the tungsten design allows for a long service life time without requiring that additional rod worth be built into the design to account for absorber depletion effects. The ability to use the minimum required GRCA rod worth minimizes the potential increase in peaking factors from flux redistribution in other unrodded core locations when a GRCA bank is fully inserted. This allows core designs to be better optimized, while still maintaining the full power operating capability. AP1000 cores are designed to meet peaking factor limits with allowed GRCA insertion, and will be monitored on-line to ensure that the limits are met at all times.

Evenly distributing the GRCA absorber material among all 24 absorber rodlets minimizes the effect of GRCA operation on the local pin power distribution in the assembly containing the GRCA. This is illustrated in Figure 3.3-1, which shows representative rodded pin power distributions for the various Ag-In-Cd and tungsten GRCA designs in single assembly calculations, and shows the predicted change in local pin power when changing from a rodded to unrodded configuration. Figure 3.3-1 presents the assembly rodded pin power distributions for only 1/8 of the assembly for simplicity, since the assembly used in the model is fully symmetric in this geometry. Figure 3.3-1 also illustrates the relative position of the GRCA absorber rodlets in the guide thimbles for each calculation and the relative blackness to neutrons of each GRCA rodlet design. The results clearly demonstrate that the use of 24 gray absorber rodlets minimizes the effect of GRCA operation on the local pin power distribution of the rodded assembly.

a,c

FIGURE 3.3-1 COMPARISON OF REPRESENTATIVE GRCA RODDED PIN POWER DISTRIBUTIONS FOR 1/8 ASSEMBLY GEOMETRY, AND CHANGE IN PIN POWER DURING GRCA REMOVAL

3.4 Predicted Neutron Fluences and Transmutation in the Enhanced Tungsten GRCA

The tungsten GRCA design has been simulated and depleted under a neutron fluence equivalent to [ ]^{a,c}, using the PARAGON code. PARAGON performs a 70-group neutron transport theory calculation, and through the use of representative fuel and thermal conditions, correctly captures the average intensity and energy spectrum of the neutron flux expected in the AP1000 reactor. As described in Section 3.6, PARAGON has been modified to be capable of depleting a tungsten neutron absorber rodlet. The PARAGON depletion calculations are roughly equivalent to [ ]^{a,c}. This assumption is consistent with the intended operating strategy for GRCAs in AP1000. The resulting maximum predicted absorber fluences for the simulated operation are shown in Table 3.4-1. The maximum predicted fluences are described in terms of the applicable neutron energy range, in order to allow for comparison with published irradiation data on tungsten. Results are presented for only the outer surface of the tungsten absorber, which accumulates the highest fluence. Thermal and resonance energy self shielding effects reduce the fluence in the interior of the tungsten absorber below that of the surface.

Figure 3.4-1 illustrates graphically how the tungsten absorber nuclide concentrations are predicted to change over the simulated operating fluence. This figure represents the average nuclide concentration and total fluence over the entire absorber zone. Figure 3.4-2 presents similar results for the outer surface of the tungsten absorber. Transmutation effects are slightly larger on the absorber surface due to self shielding effects.

TABLE 3.4-1 MAXIMUM PREDICTED ABSORBER FLUENCE ACCUMULATIONS (NEUTRONS/CM^2) FOR TUNGSTEN GRCA ON THE ABSORBER SURFACE

[ ]a,c

a,c

FIGURE 3.4-1 PREDICTED AVERAGE TRANSMUTATION OF TUNGSTEN ABSORBER

a,c

FIGURE 3.4-2 PREDICTED OUTER SURFACE TRANSMUTATION OF TUNGSTEN ABSORBER

3.5 Treatment of Tungsten GRCAs in Safety-Related Nuclear Design Analysis for AP1000

Westinghouse nuclear design models for AP1000 cores containing tungsten GRCAs will include the GRCAs in the model, and predictions of the GRCA rod worth and core operating characteristics will be made in advance of core operation. However, Westinghouse intends to exclude crediting negative reactivity insertion from tungsten GRCAs during analytical shutdown margin calculations made prior to the startup of the core. Therefore, the pre-operation analysis will confirm that there is sufficient margin to shut down the reactor without crediting any negative reactivity insertion from GRCAs falling into the core. It is expected, however, that when the reactor is in a shut down or hot standby condition with GRCA banks confirmed to be inserted by position indication, that utility and Westinghouse personnel may credit the presence of the inserted GRCAs for any shutdown margin or shutdown boron calculations made subsequent to startup physics testing for the cycle, providing that the total measured GRCA rod worth has been confirmed to meet the physics testing acceptance criteria.

Other analytical calculations that involve modeling the insertion of GRCAs (e.g., steam line break) will be performed in the most conservative manner, assuming either that the gray rods insert normally or do not insert at all. If the results from an analytical calculation are sensitive to the ability to predict GRCA worth, then a conservative allowance will be included in the predicted GRCA worth within the analysis, consistent with the review criteria established for GRCA bank worth during startup physics testing. Peaking factor uncertainties will not be changed. Typical GRCA banks have reactivity worths that are [ ]^{a,c}. The effect on core peaking factors associated with GRCA movement is therefore naturally smaller than for allowed RCCA movement, and core peaking factors are continuously monitored in AP1000 with the BEACON™ direct margin monitoring system.

3.6 Description and Benchmarking of Nuclear Methods for Tungsten GRCAs

GRCAs containing tungsten as the primary absorber material can be accurately modeled with the Westinghouse PARAGON lattice code. PARAGON is a two-dimensional neutron transport theory code that is typically used at Westinghouse to model single or multi-assembly lattice calculations, including the presence of control rod assemblies when appropriate. PARAGON includes a resonance self shielding model that correctly captures the effect of self shielding on cross sections for materials which have large resonance absorption peaks (e.g., both tungsten and silver are in this category). PARAGON is also capable of depleting fuel and control rod absorber nuclides within the calculated flux environment. Results available from PARAGON include eigenvalues, pin power distributions, fluxes in up to 70 energy groups, reaction rates, depleted number densities, and collapsed group constants that are used in courser mesh nodal models such as the Westinghouse ANC code (Reference 4). ANC is typically used to model the core in 3-dimensions and 2 energy groups, and includes a control rod insertion model that implements cross section modifiers, which are based on comparisons of rodded and unrodded results from the PARAGON lattice code.

In preparation for modeling tungsten based GRCAs in PARAGON, preliminary studies of the depletion characteristics of tungsten were made using the industry standard generalized depletion code ORIGEN (Reference 5). ORIGEN was used to simulate depletion of a gram of pure tungsten in a representative neutron flux for the [ ]^{a,c}. ORIGEN is also capable of predicting gas generation in tungsten from n-p or n-α reactions. The results from ORIGEN indicated that total gas generation in tungsten is very low, and is expected to be less than 0.1 appm for a fluence [ ]^{a,c}. Based on the isotopic content predicted by ORIGEN after this time period, and also a review of the basic neutron cross sections of the various isotopes, the following dominant nuclear reactions were identified for tungsten depletion:

$^{182}W + n \rightarrow \quad ^{183}W + n \rightarrow \quad ^{184}W$

$^{186}W + n \rightarrow \quad ^{187}W \quad \rightarrow \quad ^{187}Re$ (after beta decay)

$^{187}Re + n \rightarrow \quad ^{188}Re \rightarrow \quad ^{188}Os$ (after beta decay)

The PARAGON 70-group neutron cross section library was updated to include appropriate tungsten and rhenium nuclide cross section data from ENDF/B-VI.8, using the same methodology described in Reference 3. In addition, depletion chains for tungsten were added to PARAGON in order to enable detailed absorber depletion studies. The PARAGON tungsten depletion chains enable tracking specific nuclide concentrations for ^{182}W, ^{183}W, ^{184}W, ^{186}W, and ^{187}Re, as these are the nuclides that are predicted by ORIGEN to be present in abundance in depleted tungsten. ^{187}W, ^{188}Re, and ^{188}Os are not explicitly tracked in PARAGON, either because their abundance was predicted to be very low by ORIGEN, or because they are not significant contributors to the total absorption rate. Figure 3.4-1 shows the PARAGON average predicted transmutation occurring in a tungsten GRCA over a fluence equivalent to [ ]^{a,c}. It should be noted that the total fluence reached on this figure includes all neutron energies from 0-10 MeV, and is representative of the maximum fluence that will be accumulated in a GRCA absorber in service [ ]^{a,c}, after accounting for the intended operating strategy and GRCA shuffling scheme between cycles.

PARAGON modeling methods for tungsten GRCAs were benchmarked against the industry standard Monte-Carlo code MCNP (Reference 6). This is similar to the benchmark calculations performed initially for PARAGON in Reference 3. Specifically, identical single assembly benchmark calculations were performed using both codes at hot and cold temperature conditions, for configurations with and without inserted tungsten GRCAs. Furthermore, the configurations with inserted GRCAs were simulated assuming both fresh and depleted tungsten absorber isotopics. The depleted tungsten calculations included the effects of ^{187}Re buildup representative of [ ]^{a,c}. The results of the benchmark calculations are summarized in Table 3.6-1, and are presented both in terms of an absolute eigenvalue comparison and a GRCA rod worth comparison. The results presented in Table 3.6-1 provide independent confirmation that the PARAGON code has implemented the tungsten cross sections correctly and can predict tungsten GRCA worth with an accuracy comparable to that of black neutron absorbers like Ag-In-Cd, regardless of temperature or absorber depletion state. Therefore, it is expected that tungsten GRCA rod worths predicted by nodal core models using

PARAGON to generate the homogenized few group constants, will be accurate to within the same level of uncertainty as current black control rod designs.

Each MCNP benchmark comparison case was also compared for pin-by-pin power distribution results. The results indicate that PARAGON is capable of matching the MCNP rodded pin powers to within +/- 1%, which is similar to results that have been obtained in benchmark comparisons performed for other types of control rods.

In addition to benchmarking PARAGON against the industry standard MCNP code as discussed above, Westinghouse has identified a published research effort (Reference 7) conducted in 1997 by Rensselaer Polytechnic Institute (RPI) that confirmed acceptable accuracy of the neutron capture and total cross sections for natural tungsten that are currently in the ENDF/B-VI database. In particular, the independent measurements of tungsten performed in 1997 were made using three separate measurement techniques (transmission, capture, and self indication) over the thermal and epithermal neutron energy range that is of most interest to this application of tungsten (0.01 eV to 200 eV). The thermal capture cross section for natural tungsten at 0.0253 eV was reported as 17.09+/- 0.14 barns by RPI, compared to the ENDF/B-VI value of 18.19 barns. The resonance integral was reported as 384 barns by RPI, compared to the ENDF/B-VI value of 360 barns. The differences in the two reported numeric values are both approximately 6-7% and in opposing directions. The RPI study also included plots comparing the ENDF/B-VI capture cross section data to the independently measured capture cross sections, as a continuous function of neutron energy. The plots indicated only very minor differences between the RPI measurements and the ENDF/B-VI database for natural tungsten as a function of energy. In particular, the plotted capture cross sections demonstrated excellent agreement at the energy of the peak thermal neutron flux in the AP1000 neutron spectrum (which ocurrs at approximately 0.12 eV) and within the largest resonance absorption peaks for tungsten occurring at approximately 20 eV. The plots confirm that the basic neutron capture cross sections for tungsten in the ENDF/B-VI database contain no obvious gross errors that could significantly affect accuracy in the nuclear models for this application of tungsten. In addition, PARAGON and MCNP benchmark cases were performed using both natural tungsten and individual tungsten nuclides. The results indicated that the predicted worth of a tungsten GRCA is not significantly different depending on whether or not the natural tungsten cross sections or the individual nuclide cross sections are accessed. Furthermore, an investigation of the ENDF/B-VII cross section library confirmed that the capture and total cross sections for individual tungsten nuclides were not modified over the thermal and epithermal energy range of interest for this application. The combination of diverse MCNP benchmarks and basic cross section confirmation minimizes the risk that there will be any significant misprediction of tungsten GRCA worth using PARAGON based core models.

Finally, in the unlikely event there is a misprediction of tungsten gray rod worth outside of the normal physics testing acceptance criteria, the consequences of such a misprediction will be very minor for plants such as the AP1000. Since the gray rods will not have been credited for shutdown margin made during the core analysis before the startup (see Section 3.5), there will be no impact on the core safety evaluation. Operation of the plant will still be possible both in baseload and load follow modes of operation. In baseload operation, the GRCAs are intended to operate in automatic mode to maintain the core average temperature program. If the GRCA rod worths are mispredicted then the rods may move at slightly different rates while in auto to achieve the reactivity balance, but the temperature signals that feed the control system will be unaffected. If the rate of GRCA motion is different from predictions, this may ultimately affect the frequency of soluble boron adjustments slightly, but the effects of the misprediction on base load operation will otherwise be unnoticeable. On-line core monitoring will continue to confirm that peaking factor limits are met when the GRCAs are inserted. In load follow operation, if the actual GRCA worth is significantly less than the predicted worth, it is possible that some extreme load follow maneuvers may require a change in the soluble boron concentration. However, this situation, though inconvenient, is well within the capabilities of the plant. Startup physics testing measurements of GRCA worth will be performed as with other RCCA banks, and will identify any problems in the predicted data, allowing for correction in future cycles.

TABLE 3.6-1 PARAGON VS. MCNP BENCHMARK RESULTS FOR EIGENVALUE AND GRCA WORTH

[ ] a,c

3.7 Summary of Nuclear Design Considerations

1. The PARAGON lattice code is capable of accurately modelling the tungsten GRCA design.

2. The enhanced GRCA meets the GRCA worth requirements for the AP1000 reactor specified in the approved DCD.

3. The flat depletion characteristic of the enhanced GRCA will not limit the GRCA service lifetime.

4. Power distribution effects from GRCA operation are minimized with the enhanced GRCA design by minimizing rod worth and allowing dispersion of the absorber over all 24 rodlet locations.

5. Transmutation effects in the enhanced GRCA are well understood.

4.0 Material Properties of Tungsten

This section addresses both the unirradiated material properties of tungsten as well as the impact of irradiation on key characteristics that could influence the GRCA design performance. Section 4.1 addresses the physical properties, Section 4.2 addresses the mechanical properties, and Section 4.3 addresses the compatibility of tungsten with the reactor coolant. These properties are utilized in the design evaluations discussed in Section 5.0 and can be found in the materials handbooks as referenced.

Section 4.4 addresses the effects of irradiation on tungsten. These effects have been examined in a significant number of studies which include high irradiation fluences. [

]^{a,c}

4.1 Physical Properties

The following representative set of unirradiated physical properties for high purity tungsten is well documented and is provided in the accompanying References.

4.1.1 Physical Properties of Unirradiated Tungsten

Density: 19.254 Mg/m^3 (Ref. 9)
Atomic weight: 183.85 g/mole (Ref. 9)
Melting point: 3410 ± 20°C (Ref. 9); 3380°C (Refs. 11, 15)
Boiling point: 5700 ± 200°C (Ref. 9); 5500°C (Refs. 11, 15)
Crystal structure: Body centered cubic, A2, at all temperatures below the melting point (Refs. 9,11,15)

Based on the above melting point and uncertainty, a minimum absorber melting point of 3380°C is used as a design value.

4.1.2 Thermal Expansion

From 25°C to 2500°C (Reference 9), powder metallurgy rod:

$$\frac{L - L_{25°C}}{L_{25°C}} * 100 = -8.69 * 10^{-3} + 3.83 * 10^{-4} * T_C + 7.92 * 10^{-8} * T_C^{\ 2}$$

4.1.3 Specific Heat

Specific heat (Reference 9), from 0 to 3000°C:

$$c_p = 135.76(1 - 4805T^{-2}) + (9.1159 * 10^{-3})T + (2.3134 * 10^{-9})T^3$$

where c_p is in J kg^{-1} K^{-1} and T is in K

Specific heat (Reference 11):

Table 4.1-1

T, K	C_p, cal K^{-1} g-atom^{-1}	C_p, J K^{-1} kg^{-1}
298.15	5.80	132
400	6.00	137
500	6.12	139
600	6.22	142
700	6.30	143
800	6.37	145
900	6.44	147
1000	6.52	148

Specific heat (Reference 15), 298 – 3000K:

$c_p = 5.69 + 7.8 * 10^{-4}$ T

where c_p is in cal K^{-1} mole^{-1} and T is in K

4.1.4 Thermal Conductivity

Thermal conductivity (Reference 9):

Table 4.1-2

Temperature, °C	P, W $m^{-1}K^{-1}$
0	165
500	135
1000	120
1500	105
2000	95

4.2 Mechanical Properties

4.2.1 Elastic Properties

Elastic Properties, Polycrystalline Tungsten (Reference 9):

Table 4.2-1

Temperature, °C	E, GPa	ν
0	407	0.283
400	393	0.285
800	376	0.289
1200	357	0.295
1600	335	0.300

4.2.2 Tensile Properties

In general, unirradiated tungsten gains ductility and loses strength with increasing temperature.

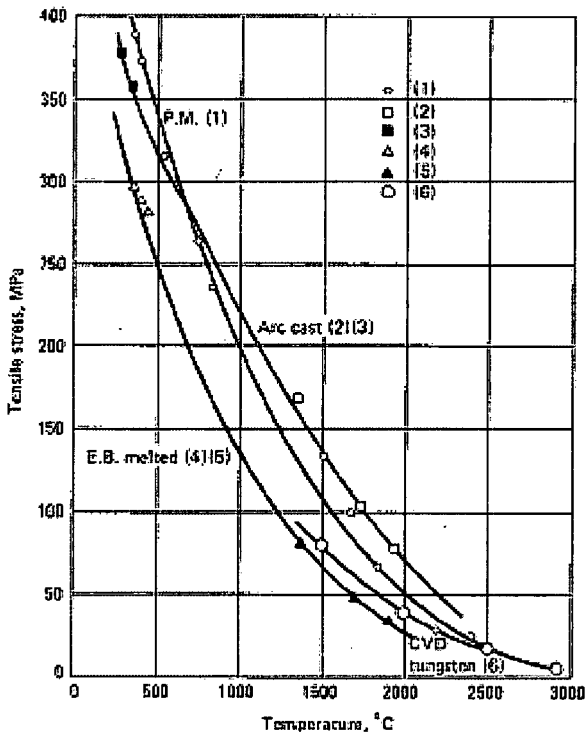

FIGURE 4.2-1 TENSILE PROPERTIES (REFERENCE 9)

4.2.3 Hardness

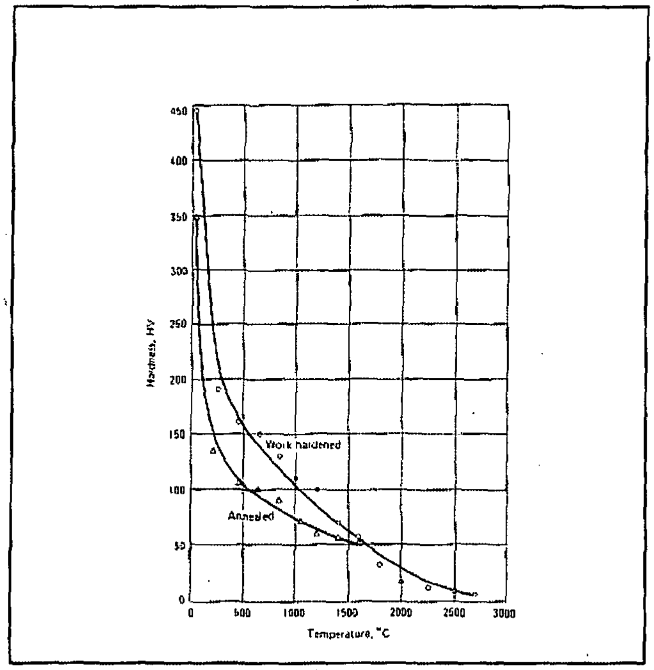

FIGURE 4.2-2 HARDNESS (REFERENCE 9)

4.2.4 Phase diagram, Re-W

Rhenium is formed as tungsten is transmuted.

Phase diagram, Re-W (Reference 12). Tungsten-rhenium alloys remain body-centered-cubic, isotropic with W, at all compositions predicted to form from transmutation by thermal neutron transmutation in GRCA service.

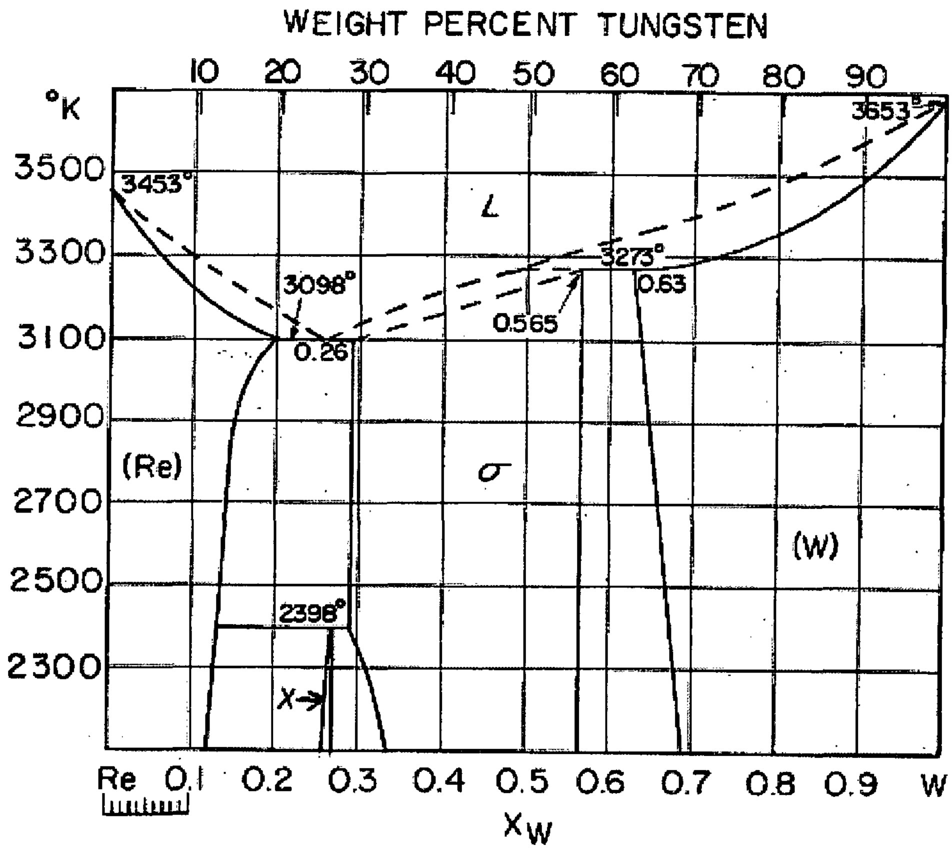

FIGURE 4.2-3 PHASE DIAGRAM, Re-W (REFERENCE 12)

4.3 Compatibility with Reactor Coolant

In the application of tungsten as an absorber in an LWR, the absorber material could possibly come into contact with the reactor coolant in the unlikely event of cladding failure and cause corrosion of the tungsten. For this reason the enhanced GRCA design employs an inner sleeve in addition to the outer cladding to double encapsulate the tungsten, thereby doubly protecting it from the coolant. The coolant conditions involve water of temperatures up to 330°C, with up to 2000 ppm or more of boron, up to 3.5 ppm or more of lithium ions, with up to 4 ppm or more of hydrogen, and corrosion products and impurities at a much lower level, i.e. ppb levels. The pH is maintained between 6.9 and 7.4.

4.3.1 Corrosion by Water and Steam

Tungsten corrodes in contact with air (Ref. 9) and starts forming a thin film of tungsten trioxide at room temperature (Ref. 47). Temperatures above 200°C and humidity will accelerate the corrosion process. The corrosion rate is dependent on the temperature and the oxygen pressure (Ref. 47). The oxide will be protective only to a limited extent. At temperatures above 700°C, the oxide will start sublimating (Ref. 47) with some acceleration in the presence of steam (Refs. 46, 45).

Tungsten reacts slowly in contact with water even at temperatures below 100°C (Ref. 47). The corrosion rate has been reported to be 160 $mg/dm^2/h$ at elevated temperature (150 to 360°C) and higher pressures (70 to 80 atm) (Ref. 47). In a more extensive study, it was found that the corrosion rate in water at elevated temperature is dependent on alloying elements and the metallurgical processing of the tungsten (Ref. 16). Pure tungsten has a lower corrosion rate than tungsten-rhenium alloys, as illustrated in Figure 4.3-1. The negative mass gain values illustrate corrosion product solubility in the water. Only the different forms of tungsten processing, such as polycrystalline (W-P, HPW), single crystal (W-S), CVD formed tungsten, and the various tungsten-rhenium alloys, up to 26% Re (W-26Re) are discussed in this report.

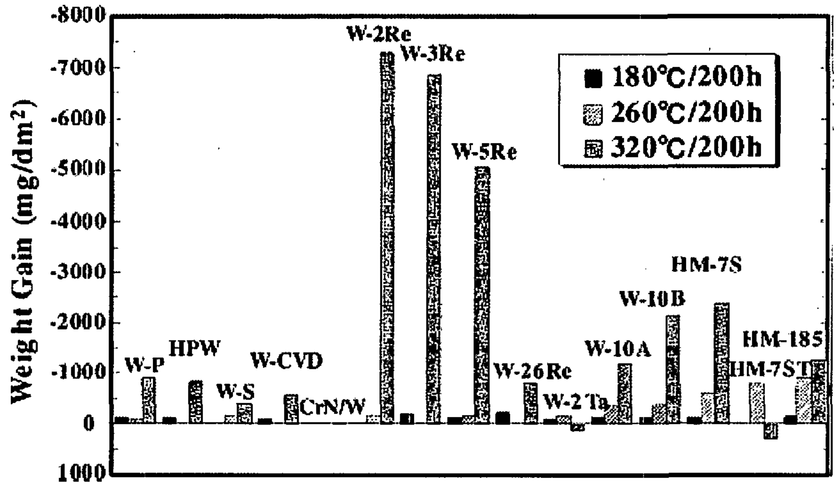

FIGURE 4.3-1 CORROSION OF VARIOUS TUNGSTEN ALLOYS IN HIGH TEMPERATURE WATER (REFERENCE 16).

An alkaline environment will increase the corrosion rate due to the increased solubility of the tungsten trioxide in alkaline environments (Ref. 44).

The density of the formed oxide, WO_3, is considerably lower than for tungsten (Refs. 16, 44) and an oxidation of tungsten by oxygen, air, steam or water would increase the volume of a oxidized body of tungsten.

As described earlier, the enhanced GRCA rodlet design utilizes double encapsulation consisting of an outer clad and inner sleeve to prevent the reactor coolant from coming into contact with the tungsten absorber. Even in the unlikely event of breach of the cladding, the inner sleeve serves to protect the absorber from the coolant.

4.4 Material Irradiation Experience

4.4.1 Tungsten Irradiation Experience

Irradiation effects on tungsten have been studied since 1968 (Refs. 27, 28, 37, 38, 41). The interest in tungsten for nuclear application has evolved since then due to the use or planned use of tungsten in three major fields in the nuclear industry. These fields are space power generation (Refs. 19, 20, 43), spallation neutron source targets (Ref. 22, 29) and more recently and most extensively, for the proposed use of tungsten as armor in the development of fusion power reactors, such as Tokamaks (Refs. 18, 21, 23, 25, 26, 30, 31, 33, 34, 36, 39, 40).

In these applications, the space power generation has generally been limited to rather low neutron fluences, less than $2x10^{21}$ $n \cdot cm^{-2}$ although up to ten times higher fluences are considered in advanced applications (Ref. 19). The studies for the fusion field and the limited studies with regard to spallation sources have been performed at higher fluences, up to $144x10^{21}$ $n \cdot cm^{-2}$ (E>0.1 MeV) (Ref. 30), [

]^{a,c}

4.4.2 General Discussion of Irradiation Effects on Materials

Irradiation effects on solid materials are either microstructural changes due to particle impact or transmutation. Microstructural changes are generally produced by fast neutrons of sufficient impact energy to cause a knock-on effect disturbing the crystal lattice of the material. In reality this means that the fast neutrons must have an energy above 0.1 or 1 MeV to cause such significant impact by forming defect clusters in the material. The effect of the defect cluster formation by each neutron impact in the material is the formation of vacancies and interstitials in the material's crystal structure. Most of the vacancies and interstitials in the cluster recombine resulting in no net effect, but some vacancies and interstitials will remain after the cluster has disappeared. The vacancies and interstitials may then diffuse to sinks, i.e. lower energy locations in the crystal structure, and thereby form dislocations and dislocation loops. The dislocations may in turn grow three-dimensionally to form voids. Such voids will contribute to the volumetric swelling of the material due to the irradiation.

High energy neutrons can also generate other cavities in a material by gas formed in the material by nuclear reactions, i.e. n,p reactions, producing hydrogen, or n,alpha reactions, producing helium. The gas atoms can also diffuse to low energy locations and grow hydrogen and helium gas bubbles in the material. Such gas bubbles, when grown to sufficient size, will also contribute to the irradiation swelling and embrittlement of a material.

The formation of interstitials, vacancies, dislocations and voids can also lead to formation of secondary phases, which would otherwise not be formed without the microstructural change of the material.

The resulting microstructural changes discussed previously will depend on the material, some important impurities in the material, the fast neutron fluence, the temperature during irradiation and after the irradiation has ceased, and the time at temperature.

The microstructural changes can also result in changes in the mechanical properties of the material. Most materials are, for instance, hardened by irradiation. This means that the yield stress of the material will increase while the ductility, i.e. ability to deform plastically, will decrease. Void formation and precipitation of second phases can also lead to material embrittlement.

Transmutation is generally caused by thermal neutrons or absorption of epithermal neutrons due to resonance peaks. Transmutation will cause a chemical change in the material since a new chemical element is formed. Occasionally the transmutation reaction can also cause microstructural effects if the energy released during the neutron capture or a subsequent nuclear reaction has high energy. The new element can sometimes take the place of the mother element in the crystal lattice or form new phases if the solubility of the formed element is low or if the energy of formation has been lowered due to other microstructural changes by the fast neutron irradiation.

The thermal and epithermal neutrons do not directly contribute to the microstructural changes as described for fast neutrons. The new phases formed could, however, cause disturbance to the crystalline structure of a material, resulting in embrittlement and swelling, just as for fast neutrons, but obviously due to quite different causes.

4.4.3 Irradiation Effects in Tungsten

As discussed in section 4.4.2, irradiation effects from fast neutrons for tungsten will be as described for the materials in general. Details about the magnitude of the various effects are discussed in the following.

Pronounced absorption peaks are found in the epithermal absorption spectrum around 20 eV for tungsten, see section 3.6. The transmutation will lead to production of rhenium, Re, and some other products at lower concentrations, among them osmium, Os (Refs. 18, 24, 30, 32, 39).

A significant number of irradiation studies on tungsten have been reported, as shown by Table 4.4-1.

TABLE 4.4-1 IRRADIATIONS OF TUNGSTEN AND TUNGSTEN ALLOYS

a,c

Most of the studies, and especially those at higher fluences, have been performed in reactors with a pronounced fast flux, such as the FFTF, EBR-II and BOR-60. Irradiations in a predominantly fast flux will produce the microstructural changes discussed in Section 4.4.2, but the simultaneous transmutation, to rhenium and eventually osmium, will not have occurred to the same extent as would be experienced in a PWR core with similar fast flux fluence. However, a number of the studies with regard to the irradiation impact of neutrons on tungsten in Table 4.4-1 have been performed on W-Re alloys up to 26% Re, the solubility limit of rhenium in tungsten (see section 4.2.5). [

]a,c

[ ]a,c Specific characteristics will be discussed in Section 4.4.5.

The irradiation temperature is important, mainly for the fast neutron impact, since a diffusion rate enhanced by elevated temperature during the irradiation will allow annealing of radiation effects such as vacancies and dislocations. An elevated temperature will also allow faster growth of voids and gas bubbles, i.e. increase swelling. As is seen in Table 4.4-1, a wide temperature range has been studied, along with the wide neutron fluence range, in the available neutron irradiation studies on tungsten.

4.4.4 Expected Limiting Fluences

[ ]a,c

[ ]a,c

4.4.5 Impact of Irradiation

In the application of tungsten in the enhanced GRCA design, the two potentially most significantly impacted properties by irradiation are ductility and swelling. These are discussed in the following sections. It should be noted that the double encapsulation feature of the enhanced GRCA design plays a key role in addressing the impact of these characteristics.

4.4.5.1 Mechanical Impact on Tungsten due to Irradiation

Neutron irradiation of tungsten will increase the ductile to brittle transition temperature (DBTT) of tungsten. This temperature, which is around room temperature for the unirradiated material depending on various metallurgical parameters, will increase significantly even after a low attained fluence of fast neutrons (Refs. 25, 47). The DBTT of tungsten was reported to change from 75°C to 880°C from unirradiated state to a fluence of 5.6×10^{21} n cm^{-2} (E>0.1 MeV) obtained at 300°C (Ref. 22). The increase of the DBTT occurs at lower fluences for W-Re alloys (Refs. 22, 25) although the DBTT is lowered for the unirradiated material by adding rhenium to the tungsten (Ref. 25).

The yield stress at 300°C was reported to decrease from 780 MPa to about 100 MPa by irradiation to $1x10^{21}$ n cm^{-2} neutron fluence (energy range unspecified) with no elongation and brittle fracture in a tensile test (Ref. 31). Further irradiation to $15x10^{21}$ n cm^{-2} lowered the yield stress to approximately 70 MPa in this study. In another study (Ref. 28), the yield stress was actually increased by irradiation to $9x10^{21}$ n cm^{-2} (E>0.1 MeV) in tensile tests at 427°C and an irradiation temperature of 382°C. Some ductility, well above 1%, remained in the material at these fluences.

The brittle cracking of the material seems to be both intergranular and transgranular at both low (Ref. 43) and high (Ref. 30) fast neutron fluences.

The variation in the results in the studies on the mechanical behavior implies that processing or maybe impurities of the tungsten will affect the embrittlement of the tungsten material. In some cases, rather good ductility can be obtained even at higher fluences while in at least one reported case, the embrittlement is rather pronounced.

As a consequence, it is considered that tungsten will become brittle even after a very short exposure (months) in a PWR neutron spectrum in the designed application as GRCA absorber material. Even so, the irradiation experience has shown that the irradiated pure tungsten will still be intact, i.e. obviously not powdered, allowing measurements of dimensions and performing immersion density determination of samples irradiated to $55x10^{21}$ n cm^{-2} (E>0.1 MeV) (Ref. 37) and even $144x10^{21}$ n cm^{-2} (E>0.1 MeV) (Ref. 30). In the latter case, the thin TEM samples occasionally broke during handling with tweezers. In another instance, TEM samples were manufactured from W-26%Re irradiated to $95x10^{21}$ n cm^{-2} (E>0.1 MeV) (Ref. 21), again showing that the material integrity is not jeopardized by even high fluences of fast neutrons. Although tungsten becomes brittle, the inner sleeve that provides double encapsulation serves to provide support for the tungsten such that the brittleness is directly considered in the design.

4.4.5.2 Swelling of Tungsten due to Irradiation

Swelling of tungsten has been studied at rather high fluences, up to $5.5x10^{22}$ n cm^{-2} neutron fluence (E>0.1 MeV). As expected, a notable effect of temperature was found and the maximum swelling was found at around 800°C, see Figure 4.4-1. [

]^{a,c} Note that this amount of swelling is much lower than that of other control rod absorber materials.

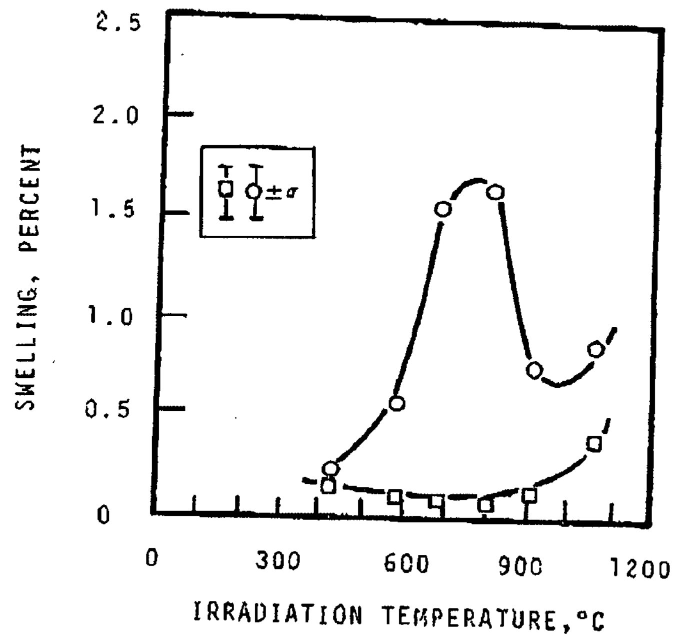

FIGURE 4.4-1 Swelling of tungsten (circles) and W-25%Re (squares) by immersion density measurement at a fluence of 5.5×10^{22} n cm^{-2} neutron fluence (E>0.1 MeV) (1×10^{22} n cm^{-2} neutron fluence (E>1 MeV)) in the EBR-II reactor (Ref. 37)

Irradiation of dynamically compacted tungsten, with a resulting density of 95.3%, to 14.4×10^{22} n cm^{-2} neutron fluence (E>0.1 MeV) at 600°C resulted in a densification of 2 to 3% rather than swelling (Ref. 35). This implies that the volumetric swelling for the dense tungsten material must be less than 3% even if we assume the swelling could perfectly fill all the voids present in the material originally. This is observed despite the very high fluence, which is almost three times higher than for the material in Figure 4.4-1. The calculated transmutation in this material, despite being exposed in a hard spectrum (Ref. 32), was found resulting in 1% rhenium and about 1% osmium and obviously these irradiation-created alloying elements do not significantly change the materials behavior. The rhenium and osmium were not found in the grain boundaries but seemed to be evenly distributed in the material (Ref. 35) as expected from the rather high solubility of Re and Os in body-centered-cubic tungsten and the substitutional nature of the alloy.

It is further shown in Figure 4.4-1 that the presence of rhenium, the main transmutation product of tungsten in a PWR spectrum, will decrease the swelling behavior in relation to pure tungsten. This is consistent with the observation that tungsten alloys with 25 to 26% rhenium do not exhibit any voids even at high fluences, i.e. $19x10^{21}$ n cm^{-2} (E>0.1 MeV) (Ref. 42) or more (Refs. 21, 38), although some voids were detected in one study (Ref. 49). The swelling of a W-25%Re material with a fluence of $37x10^{21}$ n cm^{-2} (E>0.1 MeV) irradiated at a temperature of 700°C to 900°C was reported to be less than 0.3% (Ref. 38), consistent with the data in Figure 4.4-1. The limited swelling actually recorded in the W-Re material has been reported to be due to precipitation of second phase particles due to the fast fluence (Ref. 38), which also explains a limited fluence and temperature impact on the W-Re alloy swelling below approximately 1000°C. In a tungsten alloy with 3% rhenium irradiated at 800°C, some voids were observed although less than in pure tungsten. No voids were observed in a W-26%Re alloy in the same study. The resulting swelling, evaluated from the microstructural measurements, was about 10 times lower in the W-3%Re alloy relative to the pure tungsten (Ref. 40). Voids were not found in a W-10%Re alloy irradiated to a fluence of $40x10^{21}$ n cm^{-2} (E>0.1 MeV) at 575°C to 675°C (Refs. 48, 49). The mitigating effect of rhenium on void formation in tungsten hence occurs in the range of 3 to 10% rhenium alloyed in the tungsten.

The results shown in Figure 4.4-1 are quite consistent with the general rule that swelling is not seen for materials at temperatures below $T < 0.3\ T_m$, i.e. temperatures below 30% of the melting temperature (on the Kelvin scale), since voids will not be formed below that temperature (Refs. 27, 29). Voids have, however, been observed in tungsten above 1000°C (1273 K) (Ref. 27).

Helium and hydrogen formation in tungsten from fast neutron impact is very limited compared with many other elements (Refs. 27, 33). Only about 0.2 appm of helium has been calculated to form in tungsten even with the 14 MeV high energy neutrons in the ITER design at a fluence of $2x10^{21}$ n cm^{-2} (Ref. 25). This very low helium, and hydrogen, formation is supported by calculations performed by Westinghouse, see section 3.6. In addition, a rather high temperature is needed to allow He diffusion and gas bubble formation in such a refractory material as tungsten (Ref. 29). Gas bubble formation will hence have negligible impact on the swelling for tungsten under irradiation in a PWR neutron spectrum relative to the void formation.

4.4.5.3 Tungsten Swelling from Transmutation

The formation of rhenium by transmutation of tungsten will have little effect on the structure of the tungsten since rhenium has a high solubility (more than 25%) in tungsten. The W-Re alloys within the solubility range also have a slightly higher density (Refs. 36, 38) and will not cause a swelling effect, contrary to what has been noted by phases formed by transmutation for Ag-In-Cd alloys at high neutron fluences. In addition, the presence of rhenium has been shown to impede the already limited fast neutron induced swelling in tungsten if alloyed (Ref. 37) as illustrated by the lower curve in Figure 4.4-1 and discussed above.

4.4.6 Summary of Material Property Impacts

1. Significant irradiation experience exists such that the impacts of irradiation on key characteristics of the GRCA design can be assessed under AP1000 operating conditions. Among the various studies conducted there is a strong consistency in the trends of the data.

2. In order to protect the tungsten against embrittlement and oxidation, the enhanced GRCA design will double encapsulate the tungsten absorber. The double encapsulation insures that no coolant will come into contact with the tungsten absorber even in the unlikely event of an outer cladding breach. The inner sleeve also serves to provide support for the tungsten under the effects of irradiation embrittlement.

3. Some limited swelling is expected for tungsten in a GRCA application. Swelling less than 0.7% diametral or axial elongation is inferred for fluence up to $55x10^{21}$ n cm^{-2} (E>0.1 MeV) from the published data. An absorber operating temperature below 800°C will lower the swelling rate further. The encapsulating internal sleeve is not required to have significant ductility or creep rate since only a very limited dimensional change will be experienced by the tungsten even at high fluences.

5.0 Design Bases and Evaluations

5.1 Nuclear Design

5.1.1 GRCA Reactivity Worth

Design Basis:

GRCAs shall have sufficient control rod worth over their life time such that they are capable of compensating for the intended reactivity effects as assumed in the plant's load follow operating strategy.

Evaluation:

Simulations of AP1000 load follow maneuvers down to 50% rated thermal power have established that GRCA bank worths of [ ][a,c] on average are ideal for meeting the operational requirement to perform such maneuvers without requiring a change in the core soluble boron concentration. The original approved four-rodlet Ag-In-Cd GRCA design was confirmed to provide this average targeted worth in actual core designs, and to meet the load follow operational requirement. The four-rodlet Ag-In-Cd design was therefore established as the basis for comparison with all other enhanced GRCA design concepts that were considered. The tungsten based enhanced GRCA design described in this report has been confirmed to meet the targeted GRCA control rod worth for AP1000, through comparison with the four-rodlet Ag-In-Cd GRCA design using unit assembly control rod worth calculations. This evaluation is described in Section 3.2.

5.1.2 GRCA Effect on Power Distribution

Design Basis:

Subject to meeting the minimum worth requirements in Section 5.1.1, GRCAs shall be designed to minimize the effect on core power distribution associated with their operation. This criterion helps to ensure that peaking factor limits can be met while maintaining an economical fuel management strategy, and also minimizes local pin power distribution effects in rodded assemblies.

Evaluation:

The tungsten based GRCA design described in this report has been confirmed to minimize both global and local effects on power distribution. This evaluation is described in Section 3.3.

5.2 Mechanical Design Bases and Evaluations

Design Basis:

For Conditions I and II, the stress and strain categories, and the strength theory presented in the ASME Code, Section III, are used as a general guide in the design of the GRCA structural parts. For conditions III and IV, code stresses are not limiting.

The design conditions considered under the ASME Code, Section III, are as follows:

- External pressure equal to the reactor coolant system operating pressure with appropriate allowance for overpressure transients
- Wear allowance equivalent to 1,000 reactor trips
- Bending of the rod due to a misalignment in the guide thimble
- Forces imposed on the rods during rod drop
- Loads imposed by the accelerations of the control rod drive mechanism
- Radiation exposure during maximum core life.
- Temperature effects at operating conditions.

In addition to meeting the above ASME conditions, specific enhanced GRCA evaluations are provided in the following Sections 5.2.1, 5.2.2, 5.2.3, and 5.2.4.

5.2.1 Internal Pressure and Cladding Stresses during Normal, Transient, and Accident Conditions

The design of GRCA rods provides a sufficient cold void volume to accommodate the minor internal pressure increase during operation. Vey little gas is -released by the tungsten absorber. Sufficient diametral and end clearances have been provided to accommodate the relative thermal expansion between the sleeve and the surrounding clad and end plugs. Thermal expansion of the sleeve and cladding is greater than that of the tungsten absorber.

GRCA rodlet and fuel assembly guide thimble and dashpot analyses indicate that the flow is sufficient to prevent coolant boiling within the guide thimble. Therefore, clad temperatures at which the clad material has adequate strength to resist coolant operating pressures are maintained.

5.2.2 Irradiation Stability of the Absorber Material, Taking into Consideration Gas Release and Swelling

The irradiation stability of the tungsten absorber material is discussed in Section 4.4.5.1. Irradiation produces no deleterious effects in the absorber material. Some minor cracking of the tungsten material may occur, but this cracking does not affect the absorber column geometric stability, due to the very small diametral clearance between absorber and sleeve in operation.

As previously mentioned, gas release is not a concern for the gray control rod material because very little gas is produced by the absorber material.

5.2.3 Absorber Material Swelling

The irradiation swelling of tungsten material is discussed in Sections 4.4.5.2 and 4.4.5.3. Sufficient diametral and end clearances have been provided to accommodate the absorber volumetric swelling and the relative thermal expansion between the sleeve and the surrounding cladding and end plugs, such that the cladding will not be mechanically loaded due to absorber swelling. The small amount of tungsten swelling noted in Section 4.4.6 is also accommodated by the inner sleeve [ ]^{a,c}

5.2.4 Potential for Interaction Between the Absorber and the Coolant

As discussed in Section 4.3, tungsten is subject to oxidation if it comes into contact with the reactor coolant. However, the double encapsulation feature of the enhanced GRCA rodlet design provides additional protection. Even in the highly unlikely event of a breach of the outer cladding, the inner sleeve will provide protection for the tungsten absorber.

5.3 Thermal and Hydraulic Design Bases and Evaluation

The GRCA is operated at various insertion elevations into the core, from fully inserted to fully withdrawn. The limiting condition for thermal calculations is when the GRCA is fully inserted and the core is at 100% power. Long-term power shapes are of interest for determining the operating temperature of the GRCA components as well as defining the heating rate for boiling criteria evaluations. In addition, all normal operation (Condition I) or upset (Condition II) conditions need to be evaluated to determine the maximum GRCA temperatures.

The rodlet linear heating rate of the GRCA is a key parameter in the thermal evaluations. This includes the heating of the tungsten absorber, the sleeve, and the cladding. Compared to a typical RCCA (silver-indium-cadmium), the enhanced GRCA has a lower rodlet heating rate by approximately 40%.

The following sections provide a summary of the thermal and hydraulic design bases and the associated evaluations performed using the standard Westinghouse design and analysis methodology for the these types of in-core components. Appropriate heating rates are use to reflect the incorporation of the tungsten absorber in the GRCA design.

5.3.1 Maximum Absorber Temperature

Design Basis:

The maximum temperature of the absorber material shall not exceed its melting temperature during normal operation (Condition I) or an upset (Condition II) conditions. The melting temperature of Tungsten is approximately 3380 °C (> 6000 °F) as stated in Section 4.1.1.

Evaluation:

The maximum absorber temperature was calculated using maximum $F_{\Delta h}$, maximum fuel rod F_Q, and overpower conditions (Condition II). This provides the maximum local heating in the absorber, thus resulting in the maximum temperature. Thermal evaluation shows that the maximum absorber temperature is much less than the tungsten melting temperature. Likewise, the maximum sleeve and cladding temperatures are much lower than the melting temperature of the respective materials.

In addition to maintaining the temperatures below the material melting temperature for maximum heating during Condition II conditions, it is desirable to maintain relatively low GRCA temperatures during normal, long-term operation (Condition I). Therefore, the GRCA rodlet design has features that are used to keep the operating temperature as low as possible. The gaps between the cladding, sleeve, and absorber are minimized as much as practical to reduce the temperature increase across them. In addition, helium gas is used instead of air in the largest gap, which is the cladding-to-sleeve gap. Helium has a higher thermal conductivity than air, thus improving the heat transfer and reducing the temperature rise across this gap.

To calculate realistic operating temperatures, the local power suppression in the GRCA rodlet due to presence of the absorber is considered. Therefore, the calculation of operating temperatures utilizes lower local values than maximum $F_{\Delta h}$ and maximum fuel rod F_Q.

Thermal-hydraulic analysis has shown that the criterion of maximum temperature of the absorber material not exceeding its melting temperature is met for this GRCA design.

5.3.2 No Surface Boiling within the Dashpot Region

Design Basis:

During normal operation (Condition I), there will be no surface boiling on the absorber cladding in the dashpot region.

Evaluation:

The flow in the dashpot is much lower than the flow in the guide thimble tube above the dashpot, where side holes in the guide thimble tube inject more coolant into this tube. Therefore, the temperature conditions in the dashpot need to be evaluated.

The GRCA utilizes [ ]^{a,c} This is the only portion of the GRCA that is in the dashpot, and this only occurs when the GRCA is inserted deeply into the fuel assembly. The limiting condition for surface boiling on the absorber cladding is when the GRCA is fully inserted in the fuel assembly, and this condition was evaluated.

Thermal-hydraulic analysis has shown that this criterion of no surface boiling on the absorber cladding is met for this GRCA design.

5.3.3 No Bulk Boiling in the Guide Thimble Tubes

Design Basis:

During normal operation (Condition I), there will be no bulk boiling in the guide thimble tubes.

Evaluation:

Like the dashpot region evaluation, the limiting condition is when the GRCA is fully inserted into the fuel assembly. The flow in the guide thimble tubes is much higher than the flow in the dashpot region. However, the temperature of this tube flow increases with increasing elevation in the core so the temperature conditions in the guide thimble tube need to be evaluated.

Thermal-hydraulic analysis has shown that the no bulk boiling in the guide thimble tubes criterion is met for this GRCA design.

5.3.4 Bypass Flow in Guide Thimble Tubes

Design Basis:

The core bypass flow through the guide thimble tubes is limited to assure that sufficient coolant flow is provided to the fuel rod channels to meet fuel and thermal hydraulic design criteria. The maximum guide thimble tube bypass flow, when combined with the other bypass flow paths in the reactor (see Section 4.4.4.2.1 of the DCD, Rev. 16), must be maintained below 5.9% of the overall core flow.

Evaluation:

Section 4.4.4.2.1 of the DCD, Rev.16, has shown that the core bypass is no greater than 5.9% for calculations using worst case drawing tolerances and uncertainties in pressure losses. This GRCA design has no impact on these calculations. Therefore, this bypass flow criterion is met.

5.4 Summary

The principal difference between the Enhanced GRCA and current GRCA design described in DCD Chapter 4 is the tungsten absorber material. The Enhanced GRCA rodlet design is not significantly changed other than the substitution of the tungsten absorber material and the associated double encapsulation, as noted in Section 1.0. Evaluations have been performed to verify that the current approved/licensed DCD GRCA design bases and design criteria continue to be met for the enhanced GRCA design. The enhanced GRCA design has been evaluated using the NRC-approved Westinghouse methods consistent with the guidance in SRP 4.2.

6.0 Post-LOCA Survivability

An assessment of the enhanced GRCA was made regarding post-LOCA survivability for control rods to assure that this new design is no more limiting than the Rod Cluster Control Assembly (RCCA) or the previous GRCA. The primary issue is the potential formation of a eutectic between the stainless steel clad and the zirc-based thimble tube that would lower the melting point of the cladding and cause melting of the control rod design. The same materials are utilized and [

][a,c]. Thus, the assessment performed for the RCCA in PA-ASC-0313, "Assessment of Post LOCA Control Rod Survivability," bounds the GRCA design and the GRCA design is acceptable.

An assessment was also made of the potential formation of low-temperature melting eutectics between the tungsten and the structural material. Phase diagrams for combinations of tungsten with iron and nickel, the main components in stainless steel and nickel-base materials, respectively, show that there are no such concerns for tungsten. The phase diagrams for tungsten and iron as well as for tungsten and nickel are not exhibiting any such low-temperature melting phases, as illustrated by Figure 6.0-1 and Figure 6.0-2.

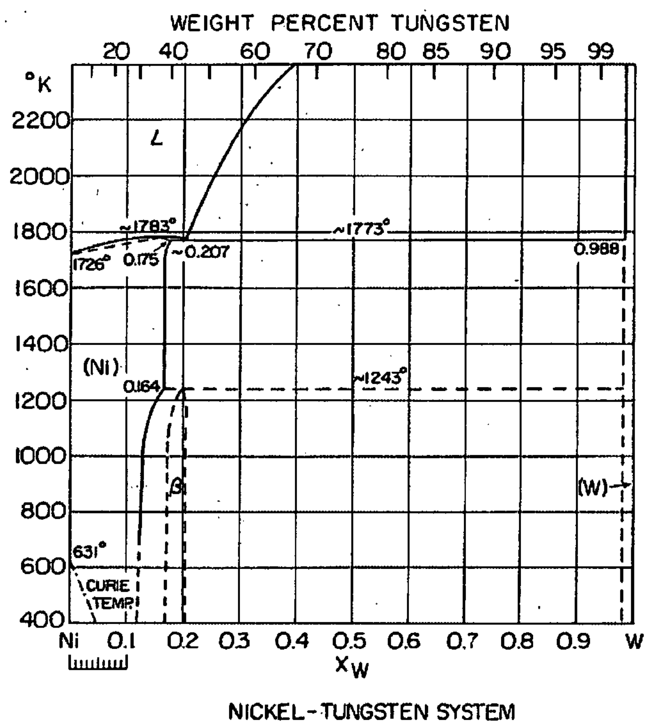

FIGURE 6.0-1 TUNGSTEN-NICKEL PHASE DIAGRAM (REFERENCE 12)

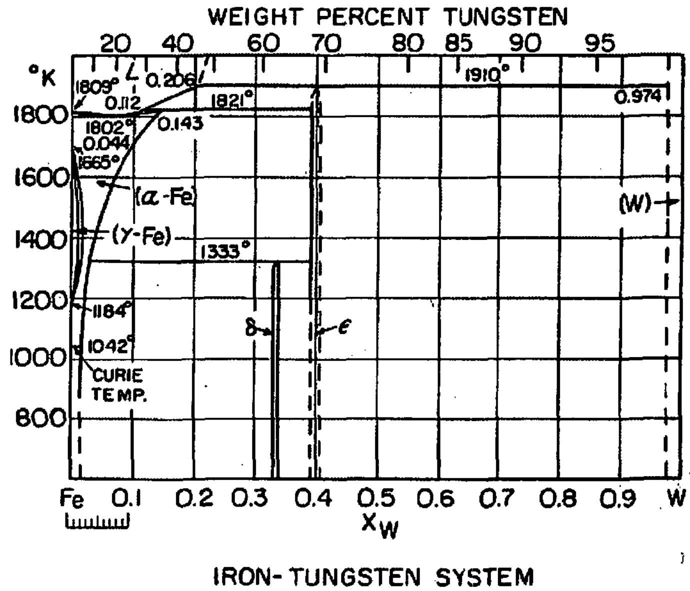

FIGURE 6.0-2 TUNGSTEN-IRON PHASE DIAGRAM (REFERENCE 12)

7.0 Operational Monitoring Program

In order to validate the expected performance of the enhanced GRCA design in the reactor core, Westinghouse will monitor specific aspects of the design over its lifetime. This approach will provide assurance that continuing successful performance is achieved and also identify any unforeseen issues in a timely fashion.

The two basic aspects of this approach include startup physics testing, as well as post-irradiation examination of the hardware itself.

Although there are no specific drop time requirements for the GRCAs (i.e. their insertion is not required to assure safe shut down of the reactor), rod drop testing will be routinely conducted, similar to that conducted for RCCAs, to track and identify any trends that might indicate adverse impacts on the GRCA design.

With regard to startup physics testing, GRCA bank worths will be measured each cycle during the beginning of cycle startup in the same manner as other control rod banks. The measurement of GRCA bank worth will confirm the adequacy of the nuclear design calculation methods and provide data to support improvements in future cycles, if necessary.

With regard to post-irradiation examination, pool-side and hot cell exams will be planned at specific points in the lifetime of the GRCA from a lead plant. In order to establish a baseline for PIE exams, selected GRCAs will be pre-characterized during the manufacturing process prior to shipment. A visual exam will be performed at an appropriate point after the first or second cycle. The purpose of the visual exams will be to identify any characteristics that may indicate adverse performance. Key areas of interest include cladding wear and any changes in cladding diameter, e.g. due to swelling. It should be noted that these mechanical and material attributes generally take place slowly over time. Based on the results of the early life visual exams and drop testing trends, subsequent visual exams, more detailed non-destructive exams and a hot cell exam will be planned at appropriate points in the lifetime of the GRCA. The non-destructive exams will provide more detailed information regarding wear and any other changes to the cladding. The hot cell exam will also provide information regarding the conditions inside the rod, including the performance of the tungsten absorber material and the sleeve. A factor in determining the timing of the hot cell exam will be the desire to obtain these detailed measurements at a time that will provide the best indication of integrated lifetime performance, while validating that ongoing experience meets the expected performance.

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Section C

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-3846
e-mail: greshaja@westinghouse.com

LTR-NRC-10-25
April 7, 2010

Subject: Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" (Proprietary/Non-Proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary versions of the responses to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design."

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-10-2795 (Non-Proprietary) with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-10-2795 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Phyllis Clark, NRO
Eileen McKenna, NRO
J. Donoghue, NRO
E. Lenning, NRR

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Direct tel: 412/374-4643
Direct fax: 412/374-3846
e-mail: greshaja@westinghouse.com

AW-10-2795

April 7, 2010

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-10-25 P-Enclosure, "Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, 'Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design'" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-10-25, dated April 7, 2010

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-10-2795 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying Affidavit should reference AW-10-2795 and should be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

cc: Phyllis Clark, NRO
Eileen McKenna, NRO
J. Donoghue, NRO
E. Lenning, NRR

AW-10-2795

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse) and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 7th day of April 2010.

Renee' Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-NRC-10-25 P-Enclosure, "Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, 'Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design'" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-10-25) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company includes responses to NRC requests for additional information.

This information is part of that which will enable Westinghouse to:

(a) Obtain NRC approval for the Westinghouse Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design using new gray rod absorber material.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) The information requested to be withheld reveals the distinguishing aspects of a method and/or methodology which was developed by Westinghouse.

(b) Assist customers to obtain license changes.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar fuel design and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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Response to NRC Requests for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" (Non-Proprietary)

Westinghouse Electric Company
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Response to NRC Requests for Additional Information (RAIs) Re: WCAP-16943-P

RAI 1. Provide more details of the Gray Rod Cluster Assembly (GRCA) design including the following:

a. Inner and outer diameters of cladding, sleeve, and tungsten
b. Type of stainless steel used in the cladding
c. Geometry of upper plenum regions of sleeve and cladding
d. Helium fill pressure of cladding and sleeve

Westinghouse Response to RAI 1:

The details of the GRCA design used to generate the results reported in the topical report are provided below. The primary purpose for providing this information is to address the NRC's need to understand the specifics of the design application and be able to perform audit calculations of material presented in the topical report. [ ]^{a,c}

a. The dimensions used to generate the calculation results reported in the topical are as follows:
   - Absorber diameter is [ ]^{a,c}
   - Sleeve inner diameter is [ ]^{a,c}
   - Sleeve outer diameter is [ ]^{a,c}
   - Cladding inner diameter is [ ]^{a,c}
   - Cladding outer diameter is 0.381 inches.
   - Minimum radial gap between sleeve and cladding, including manufacturing tolerances is larger than [ ]^{a,c}

b. The cladding material is cold-worked stainless steel [ ]^{a,c}. [ ]^{a,c}

c. The cladding upper plenum diameter is the same as the cladding inner diameter. The length of the cladding upper plenum is [ ]^{a,c}

   [ ]^{a,c}

d. The sleeve backfill is [ ]^{a,c} The cladding backfill is [ ]^{a,c}

RAI 2. Section 4.0 documents the material properties of Tungsten

a. Are these properties used for all safety analyses documented in Section 5?
b. Section 4.1.1 gives 2 values for melting point and boiling point. Which of these values is used in the safety analyses?
c. Section 4.1.3 gives 3 correlations for specific heat. Which of these correlations is used in the safety analyses?
d. Section 4.2.2, and 4.2.3 show tensile stress and hardness data. Are these data used in a quantitative way in any safety analysis?
e. Describe how the tungsten absorber material is manufactured including specifications on impurities and any alloying additions.

Westinghouse Response to RAI 2:

As discussed in the topical report section 2.1, the GRCA function is to perform an operational reactivity control function and is not required for "safety" functions. Therefore, this question will be answered from the standpoint of the design analyses performed and reported in the topical to demonstrate that all of the design criteria listed in WCAP 16943-P will be met.

a. When the analyses in Section 5 require the use of various properties, the properties identified in the topical are those that are employed. More specifics will be described in subsequent answers.

b. The melting point of 3380 deg C is used in the thermal analysis mentioned in Section 5.3.1. The boiling point is not used in any analyses.

c. There are no analyses that use the correlations for specific heat.

d. Tungsten tensile stress and hardness are not used in a quantitative way in any analysis.

e. The manufacture of tungsten is conducted by the following process:

TUNGSTEN ROD PRODUCTION

[ ] a,c

[ ][a,c] The product is defined by the following chemistry requirements:

[ ] a,c

The product chemistry is over checked by an independent laboratory to assure conformance to the requirements. The above requirements yield a product that is [ ][a,c] tungsten.

RAI 3. Provide data that shows the stainless steel cladding and sleeve will maintain sufficient ductility up to the target fast neutron fluence level. Section 5.2.3 states that the sleeve will have sufficient ductility at the irradiation damage saturation. Provide data to support this claim and that ductility saturates. Also provide any material properties for stainless steel cladding and sleeve that are used in mechanical design evaluations.

Westinghouse Response to RAI 3:

Background

[

]a,c Because of these favourable properties and the experience base, it was selected as the sleeve material to contain the tungsten in the Gray Rod Cluster Assembly (GRCA design). [

]a,c

Supplemental data at high fluence and elevated temperature is available from test reactor irradiation.

a,c

[

]a,c

Review of Irradiation Data [ ]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[ ]^{a,c}

The mechanical property, swelling, and microstructural data [ ]^{a,c} can be summarized as follows.

- [ ]^{a,c}
- The strength of the material is not strongly affected by irradiation. [ ]^{a,c}
- Ductility is decreased by irradiation. [ ]^{a,c}
- [ ]^{a,c}
- The peak swelling [ ]^{a,c} was low, [ ]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

[ ]^{a,c}

[ ]a,c

[ ]a,c

[

]a,c

[

]a,c

[

]a,c

a,c

[

]a,c

a,c

[

]a,c

The overall trend of the data is that ductility greater than [ ]a,c is retained throughout life which is sufficient to accommodate the maximum projected absorber swelling.

For the Gray Rod design, the sleeve could experience a maximum diameter strain of less than [ ]a,c which is well within the sleeve material strain capability [
]a,c

In the Gray Rod design, the stainless steel cladding is in compression throughout life.
[
]a,c

Table 2 lists the material properties used in the mechanical evaluation of the cladding and sleeve.

[ ] a,c

[ ] a,c

RAI 4. Figure 4.4-1 shows swelling data for W and W-25% Re. What is the expected operating temperature of the W? It has been observed for some materials that swelling increases with decreasing temperature due to a lack of annealing at lower temperature. If the expected operating temperature is below 430°C, what data can be used to justify the assumption that linear swelling will be less than that calculated?

Westinghouse Response to RAI 4:

The expected range of operating temperature for the tungsten absorber in the GRCA is from [ ]^{a,c} depending on host assembly power and GRCA position. In evaluating the anticipated absorber swelling [,]^{a,c} the temperature [ ]^{a,c} in Figure 4.4-1 of WCAP-16943-P was used. [ ]^{a,c} The data from Figure 4.4-1 is in the form of volumetric swelling. Diametral swelling is 1/3 of the value for the isotropic volumetric swelling expected for tungsten.

Thus, the swelling value used for mechanical evaluations bounds tungsten irradiation swelling over the temperature range from [ ]^{a,c} . Below 430°C (lower bound of the data in Figure 4.4-1 of WCAP-16943-P), the temperature is too low to produce recordable swelling, as stated below.

There is a general observation, as mentioned also in the WCAP-16943-P page 34, that irradiation swelling is only observed for materials at temperatures, T, approximately above $T>0.3T_m$, where T_m is the melting temperature of the material. This has been shown to be valid for stainless steels (Ref. 1, Ref. 2) materials used in electronic components, such as silica, (Ref. 3) and refractory materials, among them tungsten (Ref. 4). The explanation for this general behavior in so many different materials is that the diffusion rate in the materials at temperatures, T, below $0.3T_m$, is too low to allow formation of voids causing the swelling in reasonable times (Ref. 3). Typically, a maximum in swelling is seen at temperatures around $0.5T_m$, where thermal vacancy formation is overruling irradiation vacancy and void formation, hence impeding irradiation vacancy agglomeration.

Ref. 1 Maziasz, P.J., "Overview of microstructural evolution in neutron-irradiated austenitic stainless steels," J. Nucl. Mater. 205 (1993) 118-145.

Ref. 2 Chung, H. M., "Assessment of Void Swelling in Austenitic Stainless Steel Core Internals," NUREG/CR-6897 (ANL-04/28), 2004.

Ref. 3 Ghose, D and Karmohapatro, S.B., "Topography of Solid Surfaces Modified by Fast Ion Bombardment," ADVANCES IN ELECTRONICS ELECTRON PHYSICS V79, Volume 79 (Advances in Imaging and Electron Physics), Academic Press 1990, ISBN 0-12-014679-7. p. 73-146.

Ref. 4 Zinkle, Steven J. and Wiffen, F.W., "Radiation Effects in Refractory Alloys," Space Techn. Appl. Int'l Forum, STAIF 2004, ISBN 0-7354-0171-3, CP6999, p. 753-740

RAI 5. Section 4.4 discusses the impact of the build-in of rhenium on the material properties. No discussion was made of the expected impact of the build-in of osmium. However, it was shown in Section 3.4 that the end-of life osmium concentrations are close to that of rhenium. Discuss the impact of the build-in of osmium on the material properties. It should be noted that the solubility of Os in W is only 6 wt%.

Westinghouse Response to RAI 5:

The formation of osmium by transmutation of tungsten after [ ]^{a,c} It is further expected that any increase in brittleness will be at least partly balanced by the more significant increase of rhenium contents from transmutation. Still, the confinement of the absorber in the sleeve ensures that even a complete loss of ductility does not compromise the absorber function in operation.

The maximum osmium concentration from transmutation in the tungsten absorber has been calculated to be approximately [ ]^{a,c}

The solubility of osmium in tungsten at temperatures of 1000°C and below has been reported to be approximately 6% (Ref. 1). [ ]^{a,c}

It has also been shown (Ref. 1) that the lattice parameter of the alfa-tungsten phase will decrease as the osmium concentration is increasing in the alloy, as expected from the smaller osmium atom size. This decrease is rather small, approximately only 0.17% smaller lattice parameter with 6% osmium compared to pure tungsten. Similar shrinkage was reported also for higher osmium concentrations (Ref. 1), up to 10% osmium, even with the formation of a sigma phase above 6% osmium in the alloy.

The tungsten-osmium alloy is further hardened relative to tungsten due to the fast fluence and at least at higher osmium concentrations, i.e. when the sigma phase is formed, the alloy is very brittle.

Ref. 1 Taylor, A, Kagle, B.J. and Doyle, N.J., "The Constitution Diagram of the Tungsten-Osmium Binary System," J. Less Common Metals, 3 (1961) 333-347.

RAI 6. The evaluations for the mechanical design bases did not provide sufficient detail to ensure that the safety limits are met. The following areas need more information:

a. 5.2.1. Provide a sample calculation demonstrating that at the expected component temperature, the thermal expansion will not cause gap closure for the cladding sleeve gap or the sleeve tungsten gap.
b. Also document what gamma and neutron heating values are assumed for each component.

Westinghouse Response to RAI 6:

a. The following sample calculation for sleeve-to-cladding and absorber-to-sleeve gap closures uses the dimensions provided in the response to RIA #1 above:
   - Absorber diameter is [ ]^{a,c}
   - Sleeve inner diameter is [ ]^{a,c}
   - Sleeve outer diameter is [ ]^{a,c}
   - Cladding inner diameter is [ ]^{a,c}
   - Cladding outer diameter is 0.381 inches.
   - Minimum radial gap between sleeve and cladding, including manufacturing tolerances is larger than [ ]^{a,c}

   The following expected temperatures are bounding for this geometry: [ ]^{a,c} Therefore, even if the cladding thermal expansion is not accounted for, there is ample margin to sleeve-to-cladding gap closure.

   Tungsten thermal expansion for the dimensions shown above using Section 4.1.2 of the topical report is [ ]^{a,c} The volumetric swelling of tungsten is less than [ ]^{a,c} (per Figure 4.4-1 in the topical report) leading to a diametral increase of [ ]^{a,c} Combining these effects (irradiation swelling + thermal expansion), the tungsten total expansion [ ]^{a,c} Therefore, the tungsten-to-sleeve gap remains open at both hot and cold conditions.

   [ ]^{a,c}

[

]a,c Even with these conservative assumptions, the sleeve strains are smaller than [ ]a,c at both hot and cold conditions.

b. The following conservative linear heating rates (including both neutron and gamma heating) were used to determine the temperature of each component
   - Tungsten absorber: [ ]a,c
   - Sleeve: [ ]a,c
   - Cladding: [ ]a,c

RAI 7. Figures 3.4-1 and 3.4-2 show the difference in average transmutation and surface transmutation. Please provide a plot that shows the expected flux profile across the radius of the GRCA rodlet.

Westinghouse Response to RAI 7:

The transmutation of tungsten occurs more at that surface of the absorber than in the interior largely because of resonance self shielding effects. The following plots are provided which show the average thermal flux profile and the flux profile for an energy group containing the largest resonance peaks in tungsten. The flux profiles are shown in relation to the tungsten absorber, sleeve, and cladding, and are representative of undepleted tungsten material in an assembly operating at the core average power level. This information was determined using 70 energy group transport theory lattice calculations in the PARAGON code. The transmutation data provided in Figures 3.4-1 and 3.4-2 of the report was obtained from depletion calculations made using the same PARAGON model.

a,c

a,c

RAI 8. The operational monitoring program is presented in Section 7 and discusses startup physics testing and post-irradiation examination of the hardware. Provide a more detailed schedule of planned non-destructive and destructive examinations including the fluence levels at which inspections will be performed on the GRCAs in this program.

Westinghouse Response to RAI 8:

Post-irradiation evaluation of the GRCAs will consist of both non-destructive and destructive examinations. Non-destructive exams will begin after the first cycle of operation in a lead plant and will continue after each subsequent cycle. These exams will include eddy current testing (ECT) and ultrasonic testing (UT) on selected lead GRCAs to determine cladding profilometry and wear. Four GRCAs will be examined after the first cycle. The number of GRCAs to be examined after subsequent cycles may vary depending on the initial observations. Visual exams will be included in this effort. The GRCAs chosen will have the highest fluence. Operation in the highest flux core locations in AP1000 will expose the tungsten absorber to maximum total fluences on the order of [

]a,c

A hot cell exam will be performed to evaluate conditions inside the GRCA rodlet, including the tungsten absorber and sleeve. The exact timing of this exam will be determined by the desire to obtain these measurements at a time that will provide the best indication of integrated lifetime performance, while validating that ongoing experience meets expected performance. One consideration is the results of the non-destructive exams. If these exams indicate performance that is different from expectations, the hot cell may be warranted earlier. From a lifetime standpoint the hot cell exam would be preferred later, but after a certain lifetime or fluence under normal circumstances, sufficient data can be obtained to validate expected performance. [

]a,c

RAI 9. Provide the ORIGEN calculation inputs that were used to provide input to the depletion calculations.

Westinghouse Response to RAI 9:

An ORIGEN depletion was performed to determine the dominant nuclear absorption reactions in tungsten. The input is provided below. [

]a,c

a,c

a,c

a,c

RAI 10. Section 4.4.3 provides a discussion regarding irradiation effects in tungsten and includes a study of available databases. Provide a comparison of the material composition with the Westinghouse GRCA design.

Westinghouse Response to RAI 10:

The manufacturing of the tungsten absorber is similar to the manufacture of the test material that was cited to verify performance. The tests listed in the Table below are from Table 4.4-1 in WCAP-16943-P. The tests are from relevant investigations concerning higher neutron fluences. Manufacturing and chemical composition when available are reported. The information for the AP1000 GRCA is shown first in the table below for comparison.

[

]a,c

a,c

References in the Table refer to the reference identification given in WCAP-16943-P.

Properties of materials during irradiation may be correlated through the displacement-per-atom index, which for the tungsten absorber in the intended application was calculated using the damage energy cross-section and the effective displacement energy provided in Ref.1.

Ref. 1 Doran, D.J. and Graves, N.J., "Neutron Displacement Damage Cross Sections for Structural Metals," Irradiation Effects on the Microstructure and Properties of Metals, ASTM STP 611, American Society for Testing and Materials, 1976, p. 463-482.

RAI 11. WCAP-16943 provides transmutation chains and supporting evidence from industry data. However, the relevance of fast reactor data to the thermal reactor conditions representative of the AP1000 reactor design is debatable. What additional data can be provided to improve the support for the transmutation chains and worth estimates as listed in WCAP-16943 (e.g. Idaho National Laboratory's advanced test reactor program or a lead test assembly program)? If there is no additional data available at the current date, how will the inspection program be used to obtain this supporting information before the end of life?

Westinghouse Response to RAI 11:

The PARAGON transmutation chains were initially developed based on the results of an ORIGEN depletion (see RAI #9) and a study of the neutron absorption cross sections and radioactive decay characteristics of the nuclides determined to be present after long term depletion. ORIGEN was chosen because it analyzes the full isotopic transition matrix.

Specifically, the tungsten, rhenium, osmium, and tantalum nuclide number densities predicted by ORIGEN to be present in depleted tungsten were tabulated along with the absorption cross sections, and the relative contribution of each nuclide to the total neutron absorption was estimated. Once the most important neutron absorbers were identified, the PARAGON depletion chains were developed through determination of the nuclear reactions that produce and destroy those nuclides. Some engineering judgment was used to simplify the chains for reactions that were determined to contribute little to the production or destruction of the most important absorbers, or for short lived radioactive isotopes. [

]a,c

Through comparison with the ORIGEN results, it can be confirmed that the PARAGON depletion chains capture the transmutation reactions that are most important to the neutron absorption characteristics of the gray control rods.

PARAGON tungsten rod worth calculations were benchmarked against MCNP Monte Carlo calculations with continuous energy cross section models. This confirmed that the cross sections and self shielding model implemented in PARAGON can accurately model a tungsten based absorber approximately as well as a Monte Carlo code.

Since the topical report was submitted, a set of critical experiments has been performed with tungsten absorber rods in thermal reactor conditions (Reference 1). [

]a,c

[

]a,c

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

]a,c

[ ]a,c MCNP models were constructed of each critical experiment. Figures 11-1 and 11-2 show an example of a tungsten critical experiment [ ]a,c, as modeled in MCNP. [

]a,c The results from the MCNP eigenvalue calculations for all the experiments are tabulated below in Table 11-1 for each experiment. Note that the case names in Table 11-1 reflect the simulated moderator temperature and boron conditions, the enrichment in the test assembly zone, and the type of sample rods that were employed in each experiment. [

]a,c

The results shown in Table 11-1 confirm directly that the Monte Carlo benchmark basis for PARAGON is correctly modeling tungsten in a thermal reactor environment, and by extension that PARAGON is correctly modeling tungsten as well. Based on the results of the critical experiments, Westinghouse expects to be able to correctly predict tungsten rod worths with the same or better uncertainty levels than are experienced with our current Ag-In-Cd based control rod technology.

While the critical experiments described above did not include any simulations of depleted tungsten, they do confirm that the total neutron absorption rate predicted in undepleted tungsten control rods is correct. The long term depletion calculations performed in PARAGON indicate that tungsten depletion will result in a very slow increase in rod worth with time. Physics testing performed at the start of each operating cycle will allow us to confirm acceptable accuracy in the predictions, and identify any adverse or unexpected trends in the tungsten rod worth predictions before they become problematic.

Finally, it is noted that the intended use of the tungsten gray rods is related to core reactivity control only. However, since the standard control rods and the boron system are also available to supplement the gray rods for this function, any mis-prediction of the gray rod worth will have no impact on the ability to provide effective core reactivity control. Furthermore, since gray rods are not credited in any safety analysis, even a significant mis-prediction of the gray rod worth would have no impact on plant safety.

Ref. 1 [

]a,c

Table 11-1: Summary of Critical Experiments and MCNP K-effective Results

[ ]a,c

Figure 11-1: Example Critical Experiment Core Model ([ ]^{a,c}, top view)

a,c

Figure 11-2: Example Critical Experiment Core Model ([ ]^{a,c}, axial view)

a,c

RAI 12. Due to the long in-reactor residence time of the GRCAs and the stainless steel cladding design, there exists a potential for shadow corrosion between the stainless steel gray rod and the ZIRLO™ (ZIRLO™ High Performance Fuel Cladding Material) guide tubes. Is there an increased corrosion rate (and hydrogen pickup) that would result? If so, what are the effects on the mechanical properties of the guide tubes (e.g. higher assembly growth)?

Westinghouse Response to RAI 12:

Based on the current experience and knowledge, it is inferred that a steel clad tungsten absorber GRCA will not induce shadow corrosion in an AP1000 plant operating according to current good and well-controlled PWR primary water chemistry standards. The understanding and experience regarding shadow corrosion unambiguously shows that shadow corrosion is not more pronounced after long time exposure but rather at short exposures, and it is hence not implied that a long time exposure should develop shadow corrosion never observed for shorter periods in similar conditions.

Westinghouse has extensive experience of shadow corrosion observations in BWRs and has investigated spacer location shadow corrosion of fuel rods in thousands of spacer positions. Shadow corrosion has not been reported from any Westinghouse PWR fuel, at least not in modern times. This relates to both zirconium alloy and Inconel spacer locations of PWR fuel rods and in guide tubes for stainless steel clad Ag-In-Cd RCCAs.

It has been shown that shadow corrosion is not a matter of material per se, and hence is similar for stainless steel, Inconel and platinum in close vicinity to a zirconium material. It has also been shown that shadow corrosion is not a matter of internal material induced radiation but rather of electrochemistry, i.e. potential differences between materials in a gamma or neutron field in more oxidizing conditions. Furthermore, shadow corrosion development is not a matter of extended exposure time in core but rather showing high corrosion rate early but not late in life relative to common uniform corrosion (Ref. 1, Ref. 2). Hence, typically, in a BWR, the shadow oxide is in the order of 10 to 50 times thicker on a fuel cladding in a spacer location after one year in full power operation relative to the oxide thickness between spacer locations. After 5 years of full power operation the relative thickness is only 2 to 5 times higher. Literature and Westinghouse experience show that the hydrogen pick-up fraction is generally lower for shadow corrosion than typical uniform corrosion.

It is hence concluded that the reducing conditions in PWR and AP1000 mitigates formation of shadow corrosion and that the absorber material inside a PWR control rod when clad with stainless steel will not induce shadow corrosion. In addition, Westinghouse has experienced hundreds of pressurized water reactor years with no shadow corrosion recorded for fuel cladding and guide tube materials, such as Zircloy-4, ZIRLO™, Optimized ZIRLO™ and more experimental alloys.

Ref. 1 Fukuya, Koji; Echigoya, Hironori; Hattori, Yasuhiro; Kobayashi, Kunihide; Kobayashi, Kazuhiro; Sasaki, Taichi; "BWR Fuel Channel Performance and Localized Corrosion at High Burn-ups," Proc. ANS Int'l Topical Meeting LWR Fuel Perform., West Palm Beach, FL, USA, April 17-21, 1994, 580-586, 1994.

Ref. 2 Zwicky, Hans-Urs; Loner, Helena; Andersson, Björn; Wiktor, Clas-Göran, Harbottle, John; "Enhanced Spacer Shadow Corrosion on SVEA Fuel Assemblies in the Leibstadt Nuclear Power Plant," Pres. ANS Int'l Topical Meeting LWR Fuel Perform., Park City, UT, USA, April 10 -14, 2000.

Section D

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-3846
e-mail: greshaja@westinghouse.com

LTR-NRC-10-25, Rev.1

August 26, 2010

Subject: Revised Responses to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, Revision 0, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" (Proprietary/Non-Proprietary)

Reference: Westinghouse letter from J. A Gresham to the USNRC, "Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, 'Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design' (Proprietary/Non-Proprietary)," LTR-NRC-10-25, dated April 7, 2010.

As discussed with the staff reviewers during a conference call held on August 16, 2010, enclosed are revised proprietary and non-proprietary responses to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design." The enclosed revisions to the responses initially submitted in Westinghouse letter LTR-NRC-10-25 dated April 7, 2010, have been marked with change bars located in the margins to identify all changes and additions.

Enclosed are copies of the proprietary and non-proprietary versions of "Revised Responses to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design"

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-10-2924 (Non-Proprietary) with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding Proprietary Information from Public Disclosure and an Affidavit. The Affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the proprietary aspects of the application for withholding or Westinghouse affidavit should reference AW-10-2924 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Phyllis Clark, NRO
Eileen McKenna, NRO
J. Donoghue, NRO
E. Lenning, NRR

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Direct tel: (412) 374-4643
Direct fax: (412) 374-3846
e-mail: greshaja@westinghouse.com

AW-10-2924
August 26, 2010

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-10-25, Rev. 1, P-Enclosure, "Revised Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, 'Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design'" (Proprietary)

Reference: Letter from J. A. Gresham to Document Control Desk, LTR-NRC-10-25, Rev. 1, dated August 26, 2010

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-10-2924 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of this application for withholding or the accompanying Affidavit should reference AW-10-2924 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

JA Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

cc: Phyllis Clark, NRO
Eileen McKenna, NRO
J. Donoghue, NRO
E. Lenning, NRR

AW-10-2924

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse) and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 26th day of August 2010.

Renee' Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-NRC-10-25, Rev. 1, P-Enclosure, "Revised Response to the NRC's Request for Additional Information RE: Westinghouse Electric Company Topical Report WCAP-16943-P, 'Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design'" (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-10-25, Rev. 1) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company includes responses to NRC requests for additional information.

This information is part of that which will enable Westinghouse to:

(a) Obtain NRC approval for the Westinghouse Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design using new gray rod absorber material.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) The information requested to be withheld reveals the distinguishing aspects of a method and/or methodology which was developed by Westinghouse.

(b) Assist customers to obtain license changes.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar fuel design and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Revised Response to NRC Requests for Additional Information Re: Westinghouse Electric Company Topical Report WCAP-16943-P, "Enhanced GRCA (Gray Rod Cluster Assembly) Rodlet Design" (Non-Proprietary)

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

Response to NRC Requests for Additional Information (RAIs) Re: WCAP-16943-P

RAI 1. Provide more details of the Gray Rod Cluster Assembly (GRCA) design including the following:

a. Inner and outer diameters of cladding, sleeve, and tungsten
b. Type of stainless steel used in the cladding
c. Geometry of upper plenum regions of sleeve and cladding
d. Helium fill pressure of cladding and sleeve

Westinghouse Response to RAI 1:

The details of the GRCA design used to generate the results reported in the topical report are provided below. The primary purpose for providing this information is to address the NRC's need to understand the specifics of the design application and be able to perform audit calculations of material presented in the topical report. [

]a,c

a. The dimensions used to generate the calculation results reported in the topical are as follows:
   - Absorber diameter is [ ]a,c
   - Sleeve inner diameter is [ ]a,c
   - Sleeve outer diameter is [ ]a,c
   - Cladding inner diameter is [ ]a,c
   - Cladding outer diameter is 0.381 inches.
   - Minimum radial gap between sleeve and cladding, including manufacturing tolerances is larger than [ ]a,c

b. The cladding material is cold-worked stainless steel [ ]a,c. [

   ]a,c

c. The cladding upper plenum diameter is the same as the cladding inner diameter. The length of the cladding upper plenum is [ ]a,c

   [
   ]a,c

d. The sleeve backfill is [ ]a,c The cladding backfill is [ ]a,c

RAI 2. Section 4.0 documents the material properties of Tungsten

a. Are these properties used for all safety analyses documented in Section 5?
b. Section 4.1.1 gives 2 values for melting point and boiling point. Which of these values is used in the safety analyses?
c. Section 4.1.3 gives 3 correlations for specific heat. Which of these correlations is used in the safety analyses?
d. Section 4.2.2, and 4.2.3 show tensile stress and hardness data. Are these data used in a quantitative way in any safety analysis?
e. Describe how the tungsten absorber material is manufactured including specifications on impurities and any alloying additions.

Westinghouse Response to RAI 2:

As discussed in the topical report section 2.1, the GRCA function is to perform an operational reactivity control function and is not required for "safety" functions. Therefore, this question will be answered from the standpoint of the design analyses performed and reported in the topical to demonstrate that all of the design criteria listed in WCAP 16943-P will be met.

a. When the analyses in Section 5 require the use of various properties, the properties identified in the topical are those that are employed. More specifics will be described in subsequent answers.

b. The melting point of 3380 deg C is used in the thermal analysis mentioned in Section 5.3.1. The boiling point is not used in any analyses.

c. There are no analyses that use the correlations for specific heat.

d. Tungsten tensile stress and hardness are not used in a quantitative way in any analysis.

e. The manufacture of tungsten is conducted by the following process:

TUNGSTEN ROD PRODUCTION

[ ]^{a,c}

[ ]^{a,c} The product is defined by the following chemistry requirements:

[ ]^{a,c}

The product chemistry is over checked by an independent laboratory to assure conformance to the requirements. The above requirements yield a product that is [ ]^{a,c} tungsten.

RAI 3. Provide data that shows the stainless steel cladding and sleeve will maintain sufficient ductility up to the target fast neutron fluence level. Section 5.2.3 states that the sleeve will have sufficient ductility at the irradiation damage saturation. Provide data to support this claim and that ductility saturates. Also provide any material properties for stainless steel cladding and sleeve that are used in mechanical design evaluations.

Westinghouse Response to RAI 3:

Background

[

]a,c Because of these favourable properties and the experience base, it was selected as the sleeve material to contain the tungsten in the Gray Rod Cluster Assembly (GRCA design). [

]a,c

Supplemental data at high fluence and elevated temperature is available from test reactor irradiation.

a,c

[

]a,c

[

]a,c

Review of Irradiation Data [ ]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

The mechanical property, swelling, and microstructural data [
]a,c can be summarized as follows.

- [ ]a,c
- The strength of the material is not strongly affected by irradiation. [
]a,c
- Ductility is decreased by irradiation. [
]a,c
- [ ]a,c
- The peak swelling [ ]a,c was low, [
]a,c

[ ]a,c

[

]a,c

[

]a,c

a,c

a,c

[

]a,c

[

]a,c

[

]a,c

a,c

[

]a,c

[ ] a,c

[

]a,c

The overall trend of the data is that ductility greater than [ ]a,c is retained throughout life which is sufficient to accommodate the maximum projected absorber swelling.

For the Gray Rod design, the sleeve could experience a maximum diameter strain of less than [ ]a,c which is well within the sleeve material strain capability [ ]a,c

In the Gray Rod design, the stainless steel cladding is in compression throughout life. [ ]a,c

Table 2 lists the material properties used in the mechanical evaluation of the cladding and sleeve.

a,c

a,c

RAI 4. Figure 4.4-1 shows swelling data for W and W-25% Re. What is the expected operating temperature of the W? It has been observed for some materials that swelling increases with decreasing temperature due to a lack of annealing at lower temperature. If the expected operating temperature is below 430°C, what data can be used to justify the assumption that linear swelling will be less than that calculated?

Westinghouse Response to RAI 4:

The expected range of operating temperature for the tungsten absorber in the GRCA is from [ ]^{a,c} depending on host assembly power and GRCA position. In evaluating the anticipated absorber swelling [,]^{a,c} the temperature [ ]^{a,c} in Figure 4.4-1 of WCAP-16943-P was used. [ ]^{a,c} The data from Figure 4.4-1 is in the form of volumetric swelling. Diametral swelling is 1/3 of the value for the isotropic volumetric swelling expected for tungsten.

Thus, the swelling value used for mechanical evaluations bounds tungsten irradiation swelling over the temperature range from [ ]^{a,c} . Below 430°C (lower bound of the data in Figure 4.4-1 of WCAP-16943-P), the temperature is too low to produce recordable swelling, as stated below.

There is a general observation, as mentioned also in the WCAP-16943-P page 34, that irradiation swelling is only observed for materials at temperatures, T, approximately above $T>0.3T_m$, where T_m is the melting temperature of the material. This has been shown to be valid for stainless steels (Ref. 1, Ref. 2) materials used in electronic components, such as silica, (Ref. 3) and refractory materials, among them tungsten (Ref. 4). The explanation for this general behavior in so many different materials is that the diffusion rate in the materials at temperatures, T, below $0.3T_m$, is too low to allow formation of voids causing the swelling in reasonable times (Ref. 3). Typically, a maximum in swelling is seen at temperatures around $0.5T_m$, where thermal vacancy formation is overruling irradiation vacancy and void formation, hence impeding irradiation vacancy agglomeration.

Ref. 1 Maziasz, P.J., "Overview of microstructural evolution in neutron-irradiated austenitic stainless steels," J. Nucl. Mater. 205 (1993) 118-145.

Ref. 2 Chung, H. M., "Assessment of Void Swelling in Austenitic Stainless Steel Core Internals," NUREG/CR-6897 (ANL-04/28), 2004.

Ref. 3 Ghose, D and Karmohapatro, S.B., "Topography of Solid Surfaces Modified by Fast Ion Bombardment," ADVANCES IN ELECTRONICS ELECTRON PHYSICS V79, Volume 79 (Advances in Imaging and Electron Physics), Academic Press 1990, ISBN 0-12-014679-7. p. 73-146.

Ref. 4 Zinkle, Steven J. and Wiffen, F.W., "Radiation Effects in Refractory Alloys," Space Techn. Appl. Int'l Forum, STAIF 2004, ISBN 0-7354-0171-3, CP6999, p. 753-740.

RAI 5. Section 4.4 discusses the impact of the build-in of rhenium on the material properties. No discussion was made of the expected impact of the build-in of osmium. However, it was shown in Section 3.4 that the end-of life osmium concentrations are close to that of rhenium. Discuss the impact of the build-in of osmium on the material properties. It should be noted that the solubility of Os in W is only 6 wt%.

Westinghouse Response to RAI 5:

The formation of osmium by transmutation of tungsten after [

]^{a,c} It is further expected that any increase in brittleness will be at least partly balanced by the more significant increase of rhenium contents from transmutation. Still, the confinement of the absorber in the sleeve ensures that even a complete loss of ductility does not compromise the absorber function in operation.

The maximum osmium concentration from transmutation in the tungsten absorber has been calculated to be approximately [

]^{a,c}

The solubility of osmium in tungsten at temperatures of 1000°C and below has been reported to be approximately 6% (Ref. 1). [

]^{a,c}

It has also been shown (Ref. 1) that the lattice parameter of the alfa-tungsten phase will decrease as the osmium concentration is increasing in the alloy, as expected from the smaller osmium atom size. This decrease is rather small, approximately only 0.17% smaller lattice parameter with 6% osmium compared to pure tungsten. Similar shrinkage was reported also for higher osmium concentrations (Ref. 1), up to 10% osmium, even with the formation of a sigma phase above 6% osmium in the alloy.

The tungsten-osmium alloy is further hardened relative to tungsten due to the fast fluence and at least at higher osmium concentrations, i.e. when the sigma phase is formed, the alloy is very brittle.

Ref. 1 Taylor, A, Kagle, B.J. and Doyle, N.J., "The Constitution Diagram of the Tungsten-Osmium Binary System," J. Less Common Metals, 3 (1961) 333-347.

RAI 6. The evaluations for the mechanical design bases did not provide sufficient detail to ensure that the safety limits are met. The following areas need more information:

a. 5.2.1. Provide a sample calculation demonstrating that at the expected component temperature, the thermal expansion will not cause gap closure for the cladding sleeve gap or the sleeve tungsten gap.
b. Also document what gamma and neutron heating values are assumed for each component.

Westinghouse Response to RAI 6:

a. The following sample calculation for sleeve-to-cladding and absorber-to-sleeve gap closures uses the dimensions provided in the response to RIA #1 above:
   - Absorber diameter is [ ]^{a,c}
   - Sleeve inner diameter is [ ]^{a,c}
   - Sleeve outer diameter is [ ]^{a,c}
   - Cladding inner diameter is [ ]^{a,c}
   - Cladding outer diameter is 0.381 inches.
   - Minimum radial gap between sleeve and cladding, including manufacturing tolerances is larger than [ ]^{a,c}

   The following expected temperatures are bounding for this geometry: [

   ]^{a,c} Therefore, even if the cladding thermal expansion is not accounted for, there is ample margin to sleeve-to-cladding gap closure.

   Tungsten thermal expansion for the dimensions shown above using Section 4.1.2 of the topical report is [ ]^{a,c} The volumetric swelling of tungsten is less than [ ]^{a,c} (per Figure 4.4-1 in the topical report) leading to a diametral increase of [ ]^{a,c} Combining these effects (irradiation swelling + thermal expansion), the tungsten total expansion [ ]^{a,c} Therefore, the tungsten-to-sleeve gap remains open at both hot and cold conditions.

   [

   ]^{a,c}

[

]a,c Even with these conservative assumptions, the sleeve strains are smaller than [ ]a,c at both hot and cold conditions.

b. The following conservative linear heating rates (including both neutron and gamma heating) were used to determine the temperature of each component
   - Tungsten absorber: [ ]a,c
   - Sleeve: [ ]a,c
   - Cladding: [ ]a,c

[ ]a,c

RAI 7. Figures 3.4-1 and 3.4-2 show the difference in average transmutation and surface transmutation. Please provide a plot that shows the expected flux profile across the radius of the GRCA rodlet.

Westinghouse Response to RAI 7:

The transmutation of tungsten occurs more at that surface of the absorber than in the interior largely because of resonance self shielding effects. The following plots are provided which show the average thermal flux profile and the flux profile for an energy group containing the largest resonance peaks in tungsten. The flux profiles are shown in relation to the tungsten absorber, sleeve, and cladding, and are representative of undepleted tungsten material in an assembly operating at the core average power level. This information was determined using 70 energy group transport theory lattice calculations in the PARAGON code. The transmutation data provided in Figures 3.4-1 and 3.4-2 of the report was obtained from depletion calculations made using the same PARAGON model.

a,c

a,c

RAI 8. The operational monitoring program is presented in Section 7 and discusses startup physics testing and post-irradiation examination of the hardware. Provide a more detailed schedule of planned non-destructive and destructive examinations including the fluence levels at which inspections will be performed on the GRCAs in this program.

Westinghouse Response to RAI 8:

Post-irradiation evaluation of the GRCAs will consist of both non-destructive and destructive examinations. Non-destructive exams will begin after the first cycle of operation in a lead plant and will continue after each subsequent cycle. These exams will include eddy current testing (ECT) and ultrasonic testing (UT) on selected lead GRCAs to determine cladding profilometry and wear. Four GRCAs will be examined after the first cycle. The number of GRCAs to be examined after subsequent cycles may vary depending on the initial observations. Visual exams will be included in this effort. The GRCAs chosen will have the highest fluence. Operation in the highest flux core locations in AP1000 will expose the tungsten absorber to maximum total fluences on the order of [

]a,c

A hot cell exam will be performed to evaluate conditions inside the GRCA rodlet, including the tungsten absorber and sleeve. The exact timing of this exam will be determined by the desire to obtain these measurements at a time that will provide the best indication of integrated lifetime performance, while validating that ongoing experience meets expected performance. One consideration is the results of the non-destructive exams. If these exams indicate performance that is different from expectations, the hot cell may be warranted earlier. Specifically, if there is any indication of outward deformation of the rodlet cladding beyond normal measurement uncertainty, then consideration would be given to near term destructive examination to understand the cause. From a lifetime standpoint the hot cell exam would be preferred later, but after a certain lifetime or fluence under normal circumstances, sufficient data can be obtained to validate expected performance. [

]a,c

RAI 9. Provide the ORIGEN calculation inputs that were used to provide input to the depletion calculations.

Westinghouse Response to RAI 9:

An ORIGEN depletion was performed to determine the dominant nuclear absorption reactions in tungsten. The input is provided below. [

]a,c

a,c

a,c

a,c

RAI 10. Section 4.4.3 provides a discussion regarding irradiation effects in tungsten and includes a study of available databases. Provide a comparison of the material composition with the Westinghouse GRCA design.

Westinghouse Response to RAI 10:

The manufacturing of the tungsten absorber is similar to the manufacture of the test material that was cited to verify performance. The tests listed in the Table below are from Table 4.4-1 in WCAP-16943-P. The tests are from relevant investigations concerning higher neutron fluences. Manufacturing and chemical composition when available are reported. The information for the AP1000 GRCA is shown first in the table below for comparison.

[

]a,c

a,c

References in the Table refer to the reference identification given in WCAP-16943-P.

Properties of materials during irradiation may be correlated through the displacement-per-atom index, which for the tungsten absorber in the intended application was calculated using the damage energy cross-section and the effective displacement energy provided in Ref.1.

Ref. 1 Doran, D.J. and Graves, N.J., "Neutron Displacement Damage Cross Sections for Structural Metals," Irradiation Effects on the Microstructure and Properties of Metals, ASTM STP 611, American Society for Testing and Materials, 1976, p. 463-482.

RAI 11. WCAP-16943 provides transmutation chains and supporting evidence from industry data. However, the relevance of fast reactor data to the thermal reactor conditions representative of the AP1000 reactor design is debatable. What additional data can be provided to improve the support for the transmutation chains and worth estimates as listed in WCAP-16943 (e.g. Idaho National Laboratory's advanced test reactor program or a lead test assembly program)? If there is no additional data available at the current date, how will the inspection program be used to obtain this supporting information before the end of life?

Westinghouse Response to RAI 11:

The PARAGON transmutation chains were initially developed based on the results of an ORIGEN depletion (see RAI #9) and a study of the neutron absorption cross sections and radioactive decay characteristics of the nuclides determined to be present after long term depletion. ORIGEN was chosen because it analyzes the full isotopic transition matrix.

Specifically, the tungsten, rhenium, osmium, and tantalum nuclide number densities predicted by ORIGEN to be present in depleted tungsten were tabulated along with the absorption cross sections, and the relative contribution of each nuclide to the total neutron absorption was estimated. Once the most important neutron absorbers were identified, the PARAGON depletion chains were developed through determination of the nuclear reactions that produce and destroy those nuclides. Some engineering judgment was used to simplify the chains for reactions that were determined to contribute little to the production or destruction of the most important absorbers, or for short lived radioactive isotopes. [

]a,c

Through comparison with the ORIGEN results, it can be confirmed that the PARAGON depletion chains capture the transmutation reactions that are most important to the neutron absorption characteristics of the gray control rods.

PARAGON tungsten rod worth calculations were benchmarked against MCNP Monte Carlo calculations with continuous energy cross section models. This confirmed that the cross sections and self shielding model implemented in PARAGON can accurately model a tungsten based absorber approximately as well as a Monte Carlo code.

Since the topical report was submitted, a set of critical experiments has been performed with tungsten absorber rods in thermal reactor conditions (Reference 1). [

]a,c

[

]^{a,c}

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

]^{a,c}

[ ]^{a,c} MCNP models were constructed of each critical experiment. Figures 11-1 and 11-2 show an example of a tungsten critical experiment [ ]^{a,c}, as modeled in MCNP.
[

]^{a,c} The results from the MCNP eigenvalue calculations for all the experiments are tabulated below in Table 11-1 for each experiment. Note that the case names in Table 11-1 reflect the simulated moderator temperature and boron conditions, the enrichment in the test assembly zone, and the type of sample rods that were employed in each experiment.
[

]^{a,c}

The results shown in Table 11-1 confirm directly that the Monte Carlo benchmark basis for PARAGON is correctly modeling tungsten in a thermal reactor environment, and by extension that PARAGON is correctly modeling tungsten as well. Based on the results of the critical experiments, Westinghouse expects to be able to correctly predict tungsten rod worths with the same or better uncertainty levels than are experienced with our current Ag-In-Cd based control rod technology.

While the critical experiments described above did not include any simulations of depleted tungsten, they do confirm that the total neutron absorption rate predicted in undepleted tungsten control rods is correct. The long term depletion calculations performed in PARAGON indicate that tungsten depletion will result in a very slow increase in rod worth with time. Physics testing performed at the start of each operating cycle will allow us to confirm acceptable accuracy in the predictions, and identify any adverse or unexpected trends in the tungsten rod worth predictions before they become problematic.

Finally, it is noted that the intended use of the tungsten gray rods is related to core reactivity control only. However, since the standard control rods and the boron system are also available to supplement the gray rods for this function, any mis-prediction of the gray rod worth will have no impact on the ability to provide effective core reactivity control. Furthermore, since gray rods are not credited in any safety analysis, even a significant mis-prediction of the gray rod worth would have no impact on plant safety.

Ref. 1 [

]a,c

Table 11-1: Summary of Critical Experiments and MCNP K-effective Results

a,c

Figure 11-1: Example Critical Experiment Core Model ([ ]a,c, top view)

a,c

Figure 11-2: Example Critical Experiment Core Model ([ ]^{a,c}, axial view)

a,c

RAI 12. Due to the long in-reactor residence time of the GRCAs and the stainless steel cladding design, there exists a potential for shadow corrosion between the stainless steel gray rod and the ZIRLO™ (ZIRLO™ High Performance Fuel Cladding Material) guide tubes. Is there an increased corrosion rate (and hydrogen pickup) that would result? If so, what are the effects on the mechanical properties of the guide tubes (e.g. higher assembly growth)?

Westinghouse Response to RAI 12:

Based on the current experience and knowledge, it is inferred that a steel clad tungsten absorber GRCA will not induce shadow corrosion in an AP1000 plant operating according to current good and well-controlled PWR primary water chemistry standards. The understanding and experience regarding shadow corrosion unambiguously shows that shadow corrosion is not more pronounced after long time exposure but rather at short exposures, and it is hence not implied that a long time exposure should develop shadow corrosion never observed for shorter periods in similar conditions.

Westinghouse has extensive experience of shadow corrosion observations in BWRs and has investigated spacer location shadow corrosion of fuel rods in thousands of spacer positions. Shadow corrosion has not been reported from any Westinghouse PWR fuel, at least not in modern times. This relates to both zirconium alloy and Inconel spacer locations of PWR fuel rods and in guide tubes for stainless steel clad Ag-In-Cd RCCAs.

It has been shown that shadow corrosion is not a matter of material per se, and hence is similar for stainless steel, Inconel and platinum in close vicinity to a zirconium material. It has also been shown that shadow corrosion is not a matter of internal material induced radiation but rather of electrochemistry, i.e. potential differences between materials in a gamma or neutron field in more oxidizing conditions. Furthermore, shadow corrosion development is not a matter of extended exposure time in core but rather showing high corrosion rate early but not late in life relative to common uniform corrosion (Ref. 1, Ref. 2). Hence, typically, in a BWR, the shadow oxide is in the order of 10 to 50 times thicker on a fuel cladding in a spacer location after one year in full power operation relative to the oxide thickness between spacer locations. After 5 years of full power operation the relative thickness is only 2 to 5 times higher. Literature and Westinghouse experience show that the hydrogen pick-up fraction is generally lower for shadow corrosion than typical uniform corrosion.

It is hence concluded that the reducing conditions in PWR and AP1000 mitigates formation of shadow corrosion and that the absorber material inside a PWR control rod when clad with stainless steel will not induce shadow corrosion. In addition, Westinghouse has experienced hundreds of pressurized water reactor years with no shadow corrosion recorded for fuel cladding and guide tube materials, such as Zircloy-4, ZIRLO™, Optimized ZIRLO™ and more experimental alloys.

Ref. 1 Fukuya, Koji; Echigoya, Hironori; Hattori, Yasuhiro; Kobayashi, Kunihide; Kobayashi, Kazuhiro; Sasaki, Taichi; "BWR Fuel Channel Performance and Localized Corrosion at High Burn-ups," Proc. ANS Int'l Topical Meeting LWR Fuel Perform., West Palm Beach, FL, USA, April 17-21, 1994, 580-586, 1994.

Ref. 2 Zwicky, Hans-Urs; Loner, Helena; Andersson, Björn; Wiktor, Clas-Göran, Harbottle, John; "Enhanced Spacer Shadow Corrosion on SVEA Fuel Assemblies in the Leibstadt Nuclear Power Plant," Pres. ANS Int'l Topical Meeting LWR Fuel Perform., Park City, UT, USA, April 10 -14, 2000.