Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

May 1, 2006

TVA-SQN-TS-05-10 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - TECHNICAL SPECIFICATIONS (TS) CHANGE 05-10 "EXTENDED BURNUP OF FRAMATOME FUEL ASSEMBLIES"

References:
1. TVA letter to NRC dated February 11, 2000, "Sequoyah Nuclear Plant (SQN) - Request for Exemption in Accordance with 10 CFR 50.12, 'Specific Exceptions' for the use of M5 Advanced Alloy in Fuel Rod Cladding"
2. NRC letter to TVA dated July 29, 2000, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Exemption from the Requirements of 10 CFR 50.44, 50.46, and 10 CFR Part 50 Appendix K, to allow the use of the M5 Alloy for Fuel Cladding and Structural Material (TAC Nos. MA8223 and MA8224)"

D030

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for a TS change (TS-05-10) to Licenses DPR-77 and DPR-79 for SQN Units 1 and 2. The proposed TS change will revise Section 6.9.1.14.a to adopt a recently approved topical report (TR) that extends the burnup limit of the Mark-BW fuel design with M5 alloy. This TR will be utilized as part of the analytical methods to determine the SQN units core operating limits. This change will provide increased core design margin resulting in possible longer fuel residence times and reduced reload feed batch sizes. SQN also proposes an adoption of Industry/Technical Specification Task Force Traveler, TSTF-363, Revision 0, "Revised Topical Report References in Improved Technical Specification (ITS) 5.6.5, Core Operating Limits Report (COLR)." This proposed change makes administrative changes to the format of referenced TRs in TS Section 6.9.1.14.a.

TVA requested an exemption to the requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR 50 Appendix K Paragraph I.A.5 under Reference 1. NRC granted approval of this exemption by Reference 2. This exemption, in conjunction with a TS amendment, allowed SQN to use the Mark-BW fuel design with different cladding material, M5 alloy, as that prescribed by the regulations. TVA has discussed with the NRC staff regarding the necessity to update the current exemption in reference to the recently approved TRs. It was determined that an exemption submittal is not necessary; and that within the safety evaluation for these TS changes, the SQN exemption could be described as revised as a result of adopting the recently approved TR.

TVA has determined that there are no significant hazards considerations associated with the proposed changes and that the TS changes qualify for categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

TVA intends to use the proposed TS changes to support core design for the SQN Unit 1 Cycle 15 Fall refueling outage. Therefore, TVA requests approval of these TS changes by August 2007 and that the implementation of the revised TSs be within 45 days of NRC approval. Upon NRC approval of this TS change, the implementation process will drive the updates to the COLR and subsequent submittal to NRC in accordance with TS 6.9.1.14.c.

There are no commitments contained in this submittal.

If you have any questions about this change, please contact me at 843-7170 or Jim Smith at 843-6672.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 1st day of May, 2006.

Sincerely,

for

P. L. Pace
Manager, Site Licensing and
Industry Affairs

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specifications Changes (mark-up)

cc: See page 4

Enclosures
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY (TVA)
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses DPR-77 and DPR-79 for SQN Units 1 and 2 to extend the burnup limit of the Mark-BW fuel design with advanced alloy material referred to as M5 alloy. This proposed change affects Section 6.9.1.14.a of the SQN technical specifications (TSs). The impact to Section 6.9.1.14.a includes adding an NRC approved topical report (TR) associated with M5 alloy fuel assemblies. This TR will be utilized, among others, in the determination of core operating limits for each fuel cycle. In addition, it is proposed to adopt Industry/Technical Specification Task Force Traveler, TSTF-363, Revision 0, "Revised Topical Report References in Improved Technical Specification (ITS) 5.6.5, Core Operating Limits Report (COLR)," (Reference 10) which removes any references to dates, revision numbers, and supplements in the TS listing of TRs.

2.0 PROPOSED CHANGE

Specifically, the change will revise the SQN TS Administrative Controls, Section 6.9.1.14.a leading sentence by adding, "specifically those described in the following documents:" such that the sentence will read as follows:

> "The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC, specifically those described in the following documents:"

Also, the following sentence is added prior to the listed TRs in Section 6.9.1.14.a:

> "The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements)."

The current TRs' date and revision number will be removed as detailed in the following table:

Current Listing of TRs	Proposed Listing of TRs
1. BAW-10180P-A, Rev. 1, "NEMO - Nodal Expansion Method Optimized," March 1993 (FCF Proprietary) (Methodology for Specification 3.1.1.3- Moderator Temperature Coefficient)	1. BAW-10180P-A, "NEMO - Nodal Expansion Method Optimized"
2. BAW-10169P-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology For Recirculating Steam Generator Plants," October 1989 (FCF Proprietary) (Methodology for Specification 3.1.1.3- Moderator Temperature Coefficient)	2. BAW-10169P-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology For Recirculating Steam Generator Plants"
3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989 (FCF Proprietary) (Methodology for Specification 2.2.1, - Limiting Safety System Settings [f1(ΔI), f2(ΔI) limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)	3. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs"
4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants (FCF Proprietary) (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)	4. BAW-10168P-A, "RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants"

Current Listing of TRs	Proposed Listing of TRs
5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants (FCF Proprietary) (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)	Same as 4. (above)
6. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (W Proprietary) (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)	5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"
7. WCAP-10266-P-A, Rev. 2, "The 1981 Revision Of Westinghouse Evaluation Model Using Bash Code," March 1987 (W Proprietary) (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).	6. WCAP-10266-P-A, "The 1981 Revision Of Westinghouse Evaluation Model Using Bash Code"
8. BAW-10227P-A, "Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000 (FCF Proprietary) (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)	7. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"
	8. BAW-10186-A, "Extended Burnup Evaluation"

The parenthetical information following each TR is also proposed for removal as shown in the above table.

Lastly, the change will add the published NRC approved TR justifying the extended burnup as Item 8 to Section 6.9.1.14.a:

> BAW-10186P-A, "Extended Burnup Evaluation," (Framatome ANP Proprietary).

In summary, TVA has proposed to add a recently approved TR to support core operating limits determination. Also proposed is the removal of TR revision levels, revision dates, as well

as parenthetical information while maintaining the full TR citation in the SQN COLR.

3.0 BACKGROUND

By letter dated February 18, 2000, TVA submitted a license amendment request, Reference 1, for review by the staff. The license amendment proposed a revision to SQN TSs to allow the use of a newer fuel design that uses an advanced zirconium alloy as the fuel cladding (i.e., Mark-BW fuel design with M5 alloy). The licensing amendment was technically supported by an NRC approved Framatome Cogema Fuels TR (Reference 2). NRC approved the TVA license amendments on July 31, 2000 (Reference 3). In parallel with license amendments request, TVA applied for an exemption from 10 CFR 50.44, "Standards for Combustible Gas Control in Light-Water-Cooled Power Reactors;" 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors;" and 10 CFR 50 Appendix K, "ECCS Evaluation Models," Paragraph I.A.5 (Reference 4). NRC approved the TVA exemption application on July 29, 2000 (Reference 5). SQN began using the Mark-BW fuel assemblies with M5 alloy in Unit 1 Cycle 12 (fall 2001) and Unit 2 Cycle 11 (fall 2000).

Because of limited irradiation data, a burnup restriction of 60,000 MWd/MTU was applied to the Mark-BW fuel design. On November 19, 2001, Framatome ANP submitted Supplement 1 to TR BAW-10186P-A Revision 1, "Extended Burnup Evaluation" for staff review and approval. Supplement 1 provided justification (i.e., additional irradiation data) for extension of the burnup limits for the Mark-BW fuel design with M5 alloy to 62,000 MWd/MTU. NRC approval of the supplement removed the burnup restriction of the Mark-BW fuel design (Reference 6). In addition, the NRC safety evaluation (SE) superseded the original SEs' burnup restriction for both References 7 and 8.

Extension of the burnup limit will provide SQN additional flexibility in core designs. This is achieved by relaxing restrictions on core location for select fuel assemblies and allowing assemblies to operate to higher burnup limits. The extension of burnup limits also provides a potential increase in fuel residence time and a possible reduction in feed batch sizes. Reduced feed batch sizes would help reduce the spent fuel storage burden at SQN.

The addition of the recently approved TR provided an opportunity to adopt TSTF-363 Revision 0.

4.0 TECHNICAL ANALYSIS

As presented in the above section, NRC reviewed and approved the use of Mark-BW assemblies with M5 alloy in SQN Units 1 and 2 by References 3 and 5. For this reason, the technical information presented below will focus on the considerations necessary for the increased burnup limit.

In response to Framatome ANP's request to extend the burnup limit of the Mark-BW fuel design with M5 alloy, NRC used the acceptance criteria of Section 4.2, "Fuel Design System" of the Standard Review Plan (SRP). The NRC review was focused mainly on those areas where the staff determined that the data was insufficient in the previous review (Reference 2), and could have impact in the fuel performance of Mark-BW fuel design. These areas are discussed below:

High Burnup Data

Framatome ANP has a comprehensive post irradiation examination campaign program with commitments for continued data compilation. As part of this program, Framatome ANP has collected and presented updated irradiation data from various lead test assembly (LTA) plants. The data was comprised of poolside and hotcell examinations of both Mark-B and Mark-BW fuel designs with either Zircaloy-4 or M5 cladding. Based on the additional data and commitments, adequate information is available to support peak average burnup limits of 62,000 MWd/MTU for the Mark-BW fuel design with M5 alloy.

Fuel Rod and Assembly Growth

Fuel rods and fuel assemblies grow at differing rates under irradiated and thermal conditions. If not accounted for, fuel rods could contact the fuel assembly's top and bottom nozzles resulting in undesirable conditions, such as fuel rod bowing. Framatome ANP fuel design basis accounts for irradiated growth by establishing a clearance referred to as shoulder gap.

The fuel rod bowing phenomenon is evaluated against three core design and operational criteria, in the predecessor to Reference 8 that includes: thermal-hydraulic design [departure from nucleate boiling ratio (DNBR)], local power changes, and fuel cladding mechanical design. Correlations were developed for these criteria with respect to shoulder gap clearance as a function of fuel assembly burnup. The effect of rod bowing on the DNBR is incorporated in the statistical core design methodology as a component of the engineering hot channel factor on hot pin average power. This penalty is applied to SQN's core reload analyses (Reference 9). Local power peaking changes due to local neutron moderation variations resulting from rod bowing have been analyzed, and are considered in the core operating

limits development as peaking uncertainty. The fuel cladding mechanical design is concerned with fretting of the cladding surfaces at 100 percent closure of the shoulder gap clearance. Although shoulder gap clearance is accounted for, Framatome ANP has concluded, in the unlikely event of rod-to-rod contact, an insignificant wear depth would result from the small relative motion and low contact force.

As part of the extended burnup limit increase (Reference 8), Framatome ANP updated their shoulder gap model and supporting physical data. The update largely included data for Zircaloy-4 assemblies; however, data for lead test assemblies with M5 alloy growth was also presented. Results showed that the fuel design basis for the Zircaloy assemblies is conservative with adequate clearance for 62,000 MWd/MTU. With regards to fuel design with M5 alloy, Reference 8 shows that M5 alloy fuel rod growth rates are consistently below that of Zircaloy-4 rods. In addition, the correlations of shoulder gap closure developed with respect to the three core design and operation criteria remain conservative.

Oxidation and Crud

The presence of cladding oxidation (i.e., corrosion) and crud presents potential fuel damage mechanisms. The damage mechanism from cladding oxidation in part includes a thinning of the base metal and increase of clad stress, higher base metal temperature due to lower thermal conductivity of the oxide layer, spalled oxide material contributing to crud formation, and lowered ductility due to hydrogen loading. Crud formation on the oxidation layer of the fuel rod also assists in lowering the thermal conductivity of the assembly and increasing the oxidation rate. Crud deposit can also increase core and local pressure drops.

NRC has not established limits on cladding oxidation and crud, but specifies that their effects be accounted for in the thermal and mechanical analysis. Furthermore, recent post irradiation examination of high burnup Zircaloy cladding indicates that decreased ductility exists when oxide thicknesses begin to exceed 100 microns. NRC has strongly recommended that fuel vendors and licensee take measures to limit corrosion in order to maintain adequate fuel cladding ductility at high burnups for normal operation and anticipated operational occurrences. As a result Framatome ANP has adopted a corrosion thickness limit of 100 microns.

Framatome ANP's recent data shows that M5 cladding corrosion rate tends to level off and does not increase significantly as Zircaloy-4 does in the high burnup regime. The behavior indicates that M5 cladding has a quality of lower sensitivity to oxidation kinetics for various irradiation conditions. Consequently, the M5 cladding has a greater margin to the

established corrosion limit than Zircaloy-4 cladding. In addition, crud buildup as a result of oxidation will be lessened. Based on the new corrosion data, the Mark-BW fuel design with M5 alloy cladding is acceptable to support peak rod average burnup limits of 62,000 MWd/MTU.

Control Rod Drop Time

Industry events occurred in the mid-1990's that raised concerns about control rod operability in high burnup fuel assemblies. Specifically, control rods were identified as not fully inserting within the allowed drop time or binding was experienced during component shuffles. NRC issued Bulletin 96-01, "Control Rod Insertion Problems," to address the issues. The industry took action to redesign the affected fuel assemblies and improve core management to eliminate the problem; however, NRC requires new fuel designs or burnup extension to address this concern.

The Mark-BW fuel design has a dashpot integral to the guide thimbles. This is unique to the Mark-BW fuel assemblies relative to the Mark-B fuel design. Since the original TR approval, Reference 2, Framatome ANP has accumulated control rod drop data, Reference 8. This data indicates that no Mark-BW fuel design with M5 alloy has experienced any incomplete rod insertion and for differing burnup regimes there is no appreciable drop time changes. Based on this recent control rod drop data, and the fact that M5 alloy guide thimbles are less likely than Zircaloy guide thimbles to involve this problem due to in part slower growth rates and reduced corrosion, the Mark-BW fuel design with M5 alloy cladding is acceptable to support peak average burnup limits of 62,000 MWd/MTU.

Post-Irradiation Examination

Operating experiences have identified a series of fuel issues such as accelerated growth of rods and assemblies, higher cladding oxidation, and incomplete control rod insertion events that are inherent to higher burnup limits. Because of these issues the NRC has established a few basic burnup extension guidelines. These guidelines stress the need for lead test assembly (LTA) post irradiation examinations (PIEs).

This license amendment request basis is established by recently reviewed PIE data compiled by Framatome ANP, as discussed previously. Framatome ANP has committed in References 7 and 8 to performing additional PIEs as part of their LTA program. A list of the specific commitments in regards to data collection is found in Reference 8.

Based on Framatome ANP's LTA commitment, which is consistent with the NRC guidelines, the program is acceptable for the

Mark-BW fuel design with M5 alloy cladding peak average burnup limit of 62,000 MWd/MTU.

Application of TRs

TVA has utilized the Reference 7 TR, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," on a cycle-to-cycle basis for core reload designs since fall 2000. A review of the recently approved TR (Reference 8) and associated safety evaluation did not identify the need for any additional plant specific requirements. The standing requirement includes the regulatory exemption as discussed above in the "Background" section.

TVA has reviewed the change of a peak rod average burnup increase against SQN's analysis of record and concludes this change may be utilized for future core reload designs and is bounded by our analysis of record. Considering the TRs (References 7 and 8) predict and have shown similar mechanical design performance for the Mark-BW fuel design with M5 alloy as compared to Zircaloy fuel design into higher burnup regimes and in some cases (i.e., oxidation) provide better performance, SQN remains within its thermal and hydraulic design parameters for normal operating and accident conditions. Furthermore, the increase of the burnup limits has not changed the conclusion that the Mark-BW fuel design with M5 alloy will continue to meet 10 CFR 50.46 loss-of-coolant accident (LOCA) acceptance criteria. In the event of a LOCA, testing shows the M5 peak cladding temperature (PCT) remains below 2200 degrees Fahrenheit. Oxidation testing of the M5 cladding at elevated temperatures proves that total oxidation of the fuel cladding will remain less that 17 percent. Contributions of hydrogen to the containment during a LOCA is not limited to that produced by zirconium-water reaction, additionally, Framatome ANP has shown that core wide oxidation remains well within less than 1 percent for fuel designed with M5 alloy. Framatome ANP methodology for determining flow blockage was determined by NRC to be conservative. The increase in the burnup limit has not challenged this methodology. As such, the consequences of both thermal and mechanical deformation of the fuel assemblies in the core have been assessed and the resultant deformations have been shown to maintain coolable core configurations. The emergency core cooling system (ECCS) is evaluated against the thermal power immediately after shutdown. The thermal power is largely a function of short-lived fission products, which tend to saturate at relatively low burnup limits and are not appreciably affected by extended-burnup. SQN is not proposing a change to any systems and as such long-term cooling is maintained.

Extending the burnup of fuel assemblies does result in different mixtures of fission product nuclides. However, the

source terms applied at SQN are correlated to the effective full power day (EFPD) burnup of the average core or per assembly, dependent upon the accident evaluation. No change is proposed to the established safety analysis fuel assembly inputs, specifically fuel assemblies are still limited to a maximum 1500 EFPD burnup and the reactor core average maximum burnup will remain at 1000 EFPD burnup ensuring the present accident analyses remain bounding.

Application of TSTF-363

TVA has reviewed TSTF-363, Revision 0 and finds it is appropriate for adoption to SQN's TSs without variances. To meet the specific requirement of TSTF-363, a sentence is added to Sections 6.9.1.14.a, which states, "The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements)." This ensures that full citations are included in the COLR used to determine the core limits for the particular fuel cycle.

Currently, SQN utilizes only vendor TRs ensuring that the published TR consist of both the TR and NRC safety evaluation (SE). Plant specific TRs may not incorporate the NRC SE, as the practice is to with vendor TR, and therefore may not capture any conditions of use in the NRC SE. Because SQN does not utilize plant-specific TRs, this is not a concern for the proposed change.

TVA is also proposing to remove the parenthetical information, shown in the table of the above Section 3.0, "Proposed Change," detailing which limits are associated with each referenced TR. Removal of this information is not addressed by TSTF-363. This type of information has been maintained in SQN TSs beginning in the early 1990s with approval of TS changes implementing the guidance of Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" (Reference 11). NRC GL 88-16 allows the removal of cycle-dependent variables from the TSs, provided that the values of these variables are included in a COLR and are determined with NRC reviewed and approved methodology, which is referenced in the TSs. This information has been included in the TSs only as an additional level of detail for each referenced TR, and is planned to be retained in the COLR.

Although not disclosed in the above Section "Application of TRs" and for your information, the TR (Reference 8) proposed for addition will support determination of LCO 3/4.2.2, "Heat Flux Hot Channel Factor - F_Q (X,Y,Z)."

Review of the parenthetical information in regards to 10 CFR 50.36, TVA finds this information does not meet the definition of an administrative control as defined in 10 CFR 50.36(c)(5) and does not satisfy any of the four criteria provided in 10 CFR 50.36(c)(2)(ii)(A) through (D), for inclusion in the TS as a LCO.

Any changes to the COLR will be made in accordance with the requirements of 10 CFR 50.59, with a copy of the revised COLR sent to NRC as required in Section 6.9.1.14 of the TSs.

This change is considered administrative and is consist with the NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants." This change will result in resource savings for both TVA and NRC by eliminating any future license amendment requests, when referenced TRs are revised and approved for use by the NRC.

Conclusion

NRC has reviewed and approved Framatome's request to extend the burnup limit of the Mark-BW fuel design with M5 alloy to a peak rod average burnup limit of 62,000 MWd/MTU. A limiting burnup restriction of 60,000 MWd/MTU had been place on the Mark-BW fuel design because of insufficient data to support the peak rod average burnup limit of 65,000 MWd/MTU. TVA has evaluated the change of the burnup limit and has determined that it is acceptable for application and that all records of analysis remain bounding. To paraphrase SQN TS Section, "Core Operating Limits Report," each core operating limit shall be established and documented in the COLR, the limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, shutdown margin, et cetera) of the safety analysis are met, and the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC. As such, approval of the proposed TS change will allow SQN more flexibility in reload core designs, yet, continue to ensure core operating limits are maintained within the safety limits values.

TVA has reviewed NRC approved license application changes and associated correspondences (References 12-19) in regards to the proposed adoption of TSTF-363. This review identified several NRC concerns with adoption of TSTF-363. The above information has attempted to address these concerns, relative to SQN's proposed change, in support of NRC review.

5.0 REGULATORY SAFETY ANALYSIS

This letter is a request to amend Operating Licenses DPR-77 and DPR-79 for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The proposed changes would adopt a recently approved Framatome ANP Topical Report (TR) BAW-10186P-A that extends the peak rod average burnup limit of the Mark-BW fuel design with M5 alloy. This change will add a NRC approved analytical method, for extended fuel burnup evaluations, to SQN Technical Specification (TS) "Administrative Controls" section. Lastly, this request proposes to adopt the format of Industry/TSTF Standard Technical Specification Change Traveler TSTF-363, "Revise Topical Report References in ITS [Improved TS] 5.6.5, COLR."

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   In general, fuel assemblies and more specifically fuel rod cladding, of any burnup level, is not a precursor to accidents previously evaluated. An evaluation has been performed of the Mark-BW design fuel assembly for all loss-of-coolant accidents (LOCA) and non-LOCA transient events. This evaluation confirmed and justified the use of Mark-BW fuel for operation in SQN Units 1 and 2.

   The ability of the M5 fuel rod cladding material to provide a barrier against the release of radioactive fuel material has not been reduced with respect to the Zircaloy-4 material. The approved TR evaluated postulated accidents that involved adverse core conditions and the release of radionuclides, and found that higher burnup limits have very little impact on the overall radiological consequences. Radiological consequences, as well as other safety limits, are evaluated on a cycle-to-cycle basis to confirm that the analyses of record remain bounding. If a proposed extended burnup core design exceeds bounding safety analysis values, then either the

core design would be changed or the safety values would be changed.

Rod cladding failures are assumed to occur in the fuel handling accident; however, the consequences of this event are independent of the properties of the fuel rod cladding. This is based on the fuel handling event assuming the rupture of all fuel rods regardless of the rod cladding material.

No change is proposed to the established safety analysis fuel assembly inputs, specifically fuel assemblies are still limited to a maximum 1500 effective full power day (EFPD) burnup and the reactor core average maximum burnup will remain at 1000 EFPD burnup ensuring the present accident analyses remain bounding. Based on above discussion, the proposed revision to extend the burnup limit of M5 fuel rod cladding material will not significantly increase the consequences of an accident and the potential for the release of radioactive material to the environment.

Removing revision numbers, dates, and parenthetical information from the listed TRs has no impact on the actual analytical methods used to determine the core operating limits, nor does the change have impact on the calculations performed for the current or future reloads. This change is administrative in nature. This change has no impact on plant equipment operation nor does is affect the likelihood or consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Mark-BW fuel design with M5 alloy has been demonstrated to have similar characteristics to that of the Mark-B fuel design. Extended burnup of the M5 material has not been shown to alter the functions of the rod cladding, which is to provide a barrier against the release of radioactive material. Initial plant conditions, which are considered in the accident analysis, will also be

maintained such that no new plant conditions will exist that could affect the analysis results. Since plant functions and conditions are not impacted by the proposed revision and the higher burnup limit of the Mark-BW fuel design with M5 alloy material is not postulated to become an accident initiator based on the similarity with Mark-B fuel design and Zircaloy-4 material, the possibility of a new or different kind of accident is not created.

The proposed changes will not alter the plant configuration or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The margin of safety is established by the acceptance criteria used by NRC. Meeting the acceptance criteria assures that the consequences of accidents are within known and acceptable limits. The emergency core cooling system (ECCS) acceptance criteria are not exceeded. Testing has been performed on M5 alloy with respect to criteria for peak cladding temperature (PCT) and maximum cladding oxidation. These tests demonstrate that M5 alloy rod cladding remains within PCT of 2200 degrees Fahrenheit and conservatively bounded by the 17 percent limit for maximum cladding oxidation. M5 alloy oxidation rates are lower than that of Zircaloy at temperatures less than 2200 degrees Fahrenheit and have similar rates for temperatures up to about 2300 degrees Fahrenheit. High-temperature oxidation rates of M5 alloy remain equivalent to Zircaloy and, as such, respond as hydrogen generators to the same extent. Core geometry for amenable cooling is not directly related to rod cladding material; however, it applies equally well to all materials. The consequences of both thermal and mechanical deformation of fuel

assemblies have been assessed, and the resultant deformations have been shown to maintain coolable core configurations. The ECCS is evaluated against the thermal power immediately after shutdown. The thermal power is largely a function of short-lived fission products which tend to saturate at relatively low burnup limits and are not appreciably affected by extended-burnup. Therefore, with no system changes being proposed; long-term cooling is maintained. Additionally, the fuel storage cooling system is capable of supporting the long-term storage of the extended burnup fuel assemblies' decay heat.

The changes to burnup limit have been evaluated against Departure from Nucleate Boiling (DNB) events and all applicable acceptance criteria are met. In addition, the proposed revision to allow an increase in the burnup limit of the Mark-BW fuel design with M5 alloy will not impact plant setpoints that maintain the margin of safety. Based on these results, it is concluded that the margin of safety is not significantly reduced. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Removing revision numbers, dates, and parenthetical information from the listed TRs will not reduce a margin of safety because this information has no effect on any safety analysis assumption nor does it revise any setpoints assumed in the analysis of record. The proposed change is consistent with NUREG-1431, issued by the NRC staff, revising the TSs to reflect the approved level of detail, which indicates that there is no significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of

Federal Regulations (10 CFR), Section 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. The requirements to specify the lowest functional performance levels acceptable for continued safe operation are maintained by specifying the calculation methodologies and acceptance criteria in the TSs as concurred by Generic Letter (GL) 88-16. Full disclosure of the approved methodologies used to determine core-specific parameters is required and maintained in the COLR. The identified parenthetical information is not necessary to maintain safe plant operations and its inclusion was unnecessary during the implementation of GL 88-16.

As stated in 10 CFR 50.59(c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90 if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that NRC approve the TS changes before the changes are implemented. TVA's submittal meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

NUREG-800, Standard Review Plan, Section 4.2, "Fuel Design System," presents acceptance criteria to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. This NUREG section embodies the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 10, "Reactor Design," GDC 27, "Combined Reactivity Control Systems Capability," GDC 35, "Emergency Core Cooling," 10 CFR Part 100, "Reactor Site Criteria," and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

The extension of burnup limits for the Mark-BW fuel design with M5 alloy has been reviewed against these criteria in References 7 and 8. These reviews determined the fuel design to meet these acceptance criteria.

10 CFR Part 50 GDC 19, "Control Room" requires that a control room shall be provided from which actions can be taken to operate the nuclear power plant unit under

normal and accident conditions, including loss-of-coolant accident. The increase in burnup limit of the Mark-BW fuel design with M5 alloy does not challenge the analysis of record for radiation dose to the occupants of the control room. This is maintained by not exceeding the EFPD burnup of a fuel assembly or the reactor core average maximum burnup, 1500 EFPD and 1000 EFPD, respectively.

10 CFR Part 50 GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires that the fuel handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. Configuration of plant equipment subject to this GDC is maintained by the proposed TS changes. Extended fuel burnup does not propose a challenge of these systems such that adequate safety under normal and postulated accident conditions is compromised.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or SR. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 50.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. TVA letter to NRC dated February 18, 2000, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 Technical Specification (TS) Change No. 00-16, Revision of Fuel Rod Cladding Requirements to allow the use of the M5 Advanced Zirconium Alloy Cladding"

2. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," Framatome Cogema Fuels, Lynchburg, VA., February 2000

3. NRC letter to TVA dated July 31, 2000, "Sequoyah Nuclear Plant (SQN), Units 1 and 2 Issuance of Amendments Regarding use of M5 Alloy in the Construction of Fuel Assemblies (TAC Nos. MA8490 and MA8491)"

4. TVA letter to NRC dated February 11, 2000, "Sequoyah Nuclear Plant (SQN) - Request for Exemption in Accordance with 10 CFR 50.12, 'Specific Exceptions' for the use of M5 Advanced Alloy in Fuel Rod Cladding"

5. NRC letter to TVA dated July 29, 2000, "Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Exemption from the Requirements of 10 CFR 50.44, 50.46, and 10 CFR Part 50 Appendix K, to allow the use of the M5 Alloy for Fuel Cladding and Structural Material (TAC Nos. MA8223 and MA8224)"

6. NRC letter to Mr. James F. Mallay dated June 18, 2003, "Safety Evaluation of Framatome ANP Topical Report BAW-10186P-A, Revision 1, Supplement 1, 'Extended Burnup Evaluation' (TAC Nos. MB3650 and MB7548)"

7. BAW-10227P-A, Revision 1 "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," Framatome Cogema Fuels, Lynchburg, VA., June 2003

8. BAW-10186P-A, Revision 2, "Extended Burnup Evaluation," Framatome ANP, Lynchburg, VA., June 2003

9. BAW-10220P, Revision 0, " Mark-BW Fuel Assembly Application for Sequoyah Nuclear Plants Units 1 and 2," Framatome Cogema Fuels, Lynchburg, VA., 24506, March 1996

10. Industry/TSTF Standard Technical Specification Change Traveler TSTF-363, "Revise Topical Report References in ITS [Improved TS] 5.6.5, COLR," April 2000.

11. NRC letter to Mr. Dan A. Nauman dated October 23, 1991, "Issuance of Amendments (TAC No. 80498) (TS 91-08 and 91-11)" and NRC letter to the Senior Vice President, Nuclear Power Tennessee Valley Authority, dated March 30, 1992, "Issuance of Amendments (TAC No. 80499) (TS 91-08 and 91-11)"

12. NRC letter to Mr. David A. Christian dated August 27, 2003, "Surry Units 1 and 2 - Issuance of Amendments Re: Revisions of The Core Operating Limits Report References (TAC Nos. MB9502 and MB9503)"

13. NRC letter to Mr. J. V. Parrish dated May 12, 2003, "Columbia Generating Station - Issuance of Amendment Re: The Addition of Depleted Uranium to The Fuel Assembly Composition Described in Technical Specifications 4.2.1 and 5.6.5.b (TAC No. MB6319)"

14. NRC letter to Mr. J. A. Price dated December 19, 2001, "Millstone Nuclear Power Station, Units No. 2 - Issuance of Amendment Re: Number Revision to List of Documents in Technical Specification 6.9.1.8b (TAC No. MA1780)"

15. NRC letter to Mr. Otto L. Maynard dated March 28, 2002, "Wolf Creek Generating Station - Issuance of Amendments Re: Relocation of Cycle Specific Parameters to The Core Operating Limits Report (TAC No. MB1638)"

16. NRC letter to Mr. Jeffrey S. Forbes dated March 23, 2005, "Arkansas Nuclear One, Unit 2 (ANO-2) - Issuance of Amendments Re: License Amendment Request to Support Cycle 18 Core Reload (TAC No. MC3246)"

17. NRC letter to Mr. Bryce L. Shriver dated February 25, 2003, "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Generic Changes to Improved Technical Specifications (TAC Nos. MB3269 AND MB3270)"

18. NRC letter to Mr. J. A. Scalice date December 30, 2003, "Browns Ferry Nuclear Plants, Units 2 and 3 - Issuance of Amendment Regarding Core Operating Limits (TAC Nos. MB8433 and MB8434)"

19. NRC letter to Mr. J. A. Stall dated May 6, 2004, "St. Lucie Unit 1 - Issuance of Amendments Regarding Updated Core Operating Limits Report (COLR) Methodologies (TAC No. MB5178)"

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

Proposed Technical Specification Changes (mark-up)

I. AFFECTED PAGE LIST

Unit 1
6-13
6-13a

Unit 2
6-13
6-14

II. MARKED PAGES

See attached.

Insert 1

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

Insert 2

1. BAW-10180P-A, "NEMO - Nodal Expansion Method Optimized"

2. BAW-10169P-A, "RSG Plant Safety Analysis - B&W Safety Analysis Methodology for Recirculating Steam Generator Plants"

3. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWRs"

4. BAW-10168P-A, "RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants"

Insert 3

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"

6. WCAP-10266-P-A, "The 1981 Revision of Westinghouse Evaluation Model Using BASH CODE"

7. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"

8. BAW-10186-A, "Extended Burnup Evaluation"

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 DELETED.

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.

2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

4. Control Bank Insertion Limits for Specification 3/4.1.3.6,

5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,

6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and

7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

Insert 1

6.9.1.14.a The analytical methods ~~used to determine the core operating limits shall be those previously~~ reviewed and approved by NRC ~~in~~, *specifically those described in the following documents*:

Insert 2

1. BAW-10180P-A, Rev. 1, "NEMO - NODAL EXPANSION METHOD OPTIMIZED", March 1993. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

2. BAW-10169P-A, "RSG PLANT SAFETY ANALYSIS - B&W SAFETY ANALYSIS METHODOLOGY FOR RECIRCULATING STEAM GENERATOR PLANTS", October 1989. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989. (FCF Proprietary)
   (Methodology for Specification 2.2.1, - Limiting Safety System Settings [f1(ΔI), f2(ΔI) limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

CORE OPERATING LIMITS REPORT (continued)

Insert 3

6. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985, (W Proprietary)
(Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

7. WCAP-10266-P-A, Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

8. BAW-10227P-A, "Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000, (FCF Proprietary)
(Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, "RCS Pressure and Temperature (P/T) Limits"

Specification 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System"

6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.16 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.8.4.k, Steam Generator (SG) Program. The report shall include:

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 DELETED

CORE OPERATING LIMITS REPORT

6.9.1.14 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. $f_1(\Delta I)$ limits for Overtemperature Delta T Trip Setpoints and $f_2(\Delta I)$ limits for Overpower Delta T Trip Setpoints for Specification 2.2.1.

2. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

3. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

4. Control Bank Insertion Limits for Specification 3/4.1.3.6,

5. AXIAL FLUX DIFFERENCE Limits for Specification 3/4.2.1,

6. Heat Flux Hot Channel Factor and K(z) for Specification 3/4.2.2, and

Insert 1

7. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3.

6.9.1.14.a The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in, *specifically those described in the following documents*:

Insert 2

1. BAW-10180P-A, Rev. 1, "NEMO - NODAL EXPANSION METHOD OPTIMIZED", March 1993. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

2. BAW-10169P-A, "RSG PLANT SAFETY ANALYSIS - B&W SAFETY ANALYSIS METHODOLOGY FOR RECIRCULATING STEAM GENERATOR PLANTS", October 1989. (FCF Proprietary)
   (Methodology for Specification 3.1.1.3-Moderator Temperature Coefficient)

3. BAW-10163P-A, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989. (FCF Proprietary)
   (Methodology for Specification 2.2.1, - Limiting Safety System Settings [f1(ΔI), f2(ΔI) limits], 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3/4.2.1 - Axial Flux Difference, 3/4.2.2 - Heat Flux Hot Channel Factor, 3/4.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

4. BAW-10168P-A, Rev. 2, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

5. BAW-10168P-A, Rev 3, RSG LOCA - B&W Loss of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)