TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379

February 18, 2000

TVA-SQN-TS-00-16 10 CFR 50.90

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

Gentlemen:

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

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"

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment to SQN's Licenses DPR-77 and 79 to change the TSs for Units 1 and 2. The proposed change will revise Section 5.3.1 of the SQN TSs to include the M5 alloy as an option for the cladding material of the fuel rods. Additionally, the topical report for the use of the M5 alloy is being added to the analytical methods used for determining the core operating limits described in Section 6.9.1.14.a of the TSs.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of SQN Units 1 and 2, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

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Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Units 1 and 2 marked up to show the proposed change. Enclosure 3 forwards the revised TS pages for Units 1 and 2 which incorporate the proposed change.

TVA plans to include fuel assemblies in the Unit 2 Cycle 11 core loading that are composed of the M5 fuel rod cladding material. The current schedule for the start of this refueling outage is late October 2000. Therefore, TVA requests NRC review and approval of the proposed TS revision by October 2000, and that the revised TS be made effective within 45 days of NRC approval. In addition to this TS revision request, TVA has submitted an exemption request to the requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR 50 Appendix K for the use of the M5 fuel rod cladding material. This exemption request, in conjunction with the proposed TS revision, must be approved prior to the use of the M5 alloy. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 18th day of February

Peggy M. Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures
cc: See page 3

cc (Enclosures):

Mr. R. W. Hernan, Project Manager
Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Mr. Michael H. Mobley, Director (w/o Enclosures)
Division of Radiological Health
Third Floor
L&C Annex
401 Church Street
Nashville, Tennessee 37243-1532

NRC Resident
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37384-2000

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303-3415

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2
DOCKET NOS. 327 AND 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 00-16
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 TSs to allow the use of Framatome Cogema Fuels (FCF) advanced fuel rod cladding alloy designated M5. This revision affects the requirements in Sections 5.3.1 and 6.9.1.14.a of the SQN TSs. Currently, TS Section 5.3.1 only permits the use of zircaloy for fuel rod cladding. The proposed revision will add the option for the cladding to be comprised of zircaloy or M5. The impact to TS Section 6.9.1.14.a is the addition of the NRC approved topical associated with the analysis for M5 that will be utilized in the determination of core operating limits for each fuel cycle.

II. REASON FOR THE PROPOSED CHANGE

TVA has been using FCF fuel in the SQN Units 1 and 2 reactors for the Cycle 9 and 10 cores. FCF has developed a new and improved fuel rod cladding and fuel assembly structural material designated as M5. Use of the M5 alloy, with its enhanced corrosion resistance and reduced growth rate, will permit longer fuel residence times, higher fuel burnups, and reduced reload feed batch sizes, with corresponding improvements in fuel cycle economics. Reduced feed batch sizes will also help to reduce the spent fuel storage burden at SQN. TVA plans to use fuel assemblies with M5 fuel rod cladding starting with the Unit 2 Cycle 11 core.

III. SAFETY ANALYSIS

The M5 advanced alloy was developed by FCF's parent company in France as an improved fuel rod cladding and fuel assembly structural material. Detailed properties of the M5 alloy are presented in referenced FCF Topical Report BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

The M5 alloy is an FCF proprietary material composed of primarily zirconium (approximately 99 percent) and niobium (approximately 1 percent). This composition provides for improvements in fuel rod cladding corrosion and hydrogen pickup, fuel assembly and fuel rod growth, and fuel rod cladding creep relative to the zircaloy-4 rod cladding and structural components currently in use at SQN. The M5 alloy has been tested in both reactor and nonreactor environments to ascertain its mechanical and structural properties, as described in the reference.

Results of test irradiations of M5 fuel rod cladding in commercial power reactors in both the United States and Europe have demonstrated that both the maximum fuel rod cladding corrosion rate and the hydrogen pickup rate are only about 50 percent of that observed for low-tin zircaloy-4 rod cladding of the type currently used at SQN. Therefore, use of M5 rod cladding will provide significantly greater margin to the 100 micron corrosion limit than can be afforded by low-tin zircaloy-4. These improvements in corrosion and hydrogen uptake are also applicable to fuel assembly structural components, such as guide tubes and spacer grids.

These same tests have also shown that the M5 alloy exhibits significantly less irradiation-induced growth in fuel rods and fuel assembly guide tubes when compared to zircaloy-4. This will provide additional margin to the fuel assembly and fuel rod growth limits for fuel assemblies with high burnups. Reduced fuel assembly growth will also help reduce irradiation-induced fuel assembly bow and distortion, which can be detrimental to fuel handling.

In evaluating the properties of the M5 alloy, FCF determined in the referenced topical report that the use of the M5 alloy would have either no significant impact or would produce a benefit for the following parameters and analyses:

- Fuel assembly handling and shipping loads
- Fuel rod internal pressure
- Fuel rod cladding transient strain
- Fuel centerline melting temperature
- Fuel rod cladding fatigue

* Fuel rod cladding creep collapse

* Fuel rod axial growth

* Fuel rod bow

Thus, FCF has determined that the M5 advanced alloy will perform acceptably at all normal operating conditions.

FCF evaluated, in the reference topical report, the performance of the M5 alloy for non-loss-of-coolant accident (LOCA) and LOCA accident scenarios representative of the events defined in the SQN Updated Final Safety Analysis Report. The non-LOCA events in which the rod cladding material could affect the overall accident outcome are those events that either approach or exceed departure from nucleate boiling (DNB). These events include those in which acceptable margin to DNB is an acceptance criterion (Condition I, II, and III events) as well as those events (Condition IV events) for which limited rod cladding failure is predicted.

For the Condition I, II, and III accidents, a change from zircaloy-4 to M5 fuel rod cladding and fuel assembly spacer grids produces no adverse consequences in DNB performance. This is to be expected, since both M5 and zircaloy-4 have very similar heat transfer properties. In some cases, due to the reduced rod cladding creep rate of M5, a DNB benefit can be produced since the reduced rod cladding creep rate results in greater heat transfer surface area; therefore, lower heat flux.

FCF determined that the five LOCA acceptance criteria from 10 CFR 50.46 are applicable to fuel rods clad with the M5 alloy. Generic LOCA analyses for Westinghouse-type plants and fuel have been performed using M5-specific material properties which accounted for the following:

* lower alpha-beta phase transition temperature for M5 relative to zircaloy-4,

* slower rod cladding creep for M5 relative to zircaloy-4,

* slightly lower beginning of life yield strength for M5 relative to zircaloy-4 (by 3 GWd/mtU (approximately 3 months of irradiation) the yield strength of M5 is equivalent to that of unirradiated zircaloy-4), and

- M5-specific rod clad swelling and rupture models determined through experimental measurements.

These generic LOCA analyses using M5-specific material properties have demonstrated that: (1) all five of the LOCA acceptance criteria mandated by 10 CFR 50.46 can be readily met in cores using M5 rod cladding, (2) the use of M5 rod cladding will not require any significant change in LOCA linear heat rate limits, and (3) there are no adverse LOCA-related issues that would prevent the acceptable use of M5 rod cladding.

Also, FCF determined in the reference topical, that for those accidents which result in radionuclide release (e.g., LOCA, rod ejection accident, fuel handling accident), the use of M5 rod cladding and structural components will have no adverse impact on radiological doses. Again, this is due to the similar material properties and similar DNB performance of both M5 and zircaloy-4 during these accident scenarios.

TVA has reviewed referenced Topical Report BAW-10227P-A and has determined that it is applicable to the proposed revision to the SQN TSs and is an acceptable basis for the use of the new advanced M5 alloy for fuel rod cladding at SQN. Based on the above, it can be concluded that the use of the M5 alloy for fuel rod cladding will have no adverse effect on safety.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of SQN Units 1 and 2, in accordance with the proposed change to the technical specifications (TSs), does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS revision will allow the use of a new advance alloy material for the fuel rod cladding. The new M5 alloy properties are not significantly different than the characteristics of the currently used zircaloy-4 as demonstrated in the NRC approved Topical Report BAW-10227P-A for the use of the M5 alloy for fuel rod cladding. In this topical, the M5 alloy was shown to perform very similar to the zircaloy-4 with improved performance in several areas including fuel cladding corrosion, hydrogen pickup, fuel rod and fuel assembly growth, and fuel rod cladding creep. The proposed revision will not alter the operating characteristics of the plant or plant components. The fuel rod cladding function will not be changed even though some of the rod cladding properties could be enhanced.

The M5 alloy will maintain fuel rod cladding integrity such that the potential for rod cladding failures is not increased. The fuel rod cladding is not assumed to arbitrarily fail as an accident initiator even though it does function to ensure that initial core conditions are within the analysis assumptions and to provide a barrier to the release of radiation. Therefore, the proposed revision will not increase the possibility of an accident based on the new M5 alloy having similar properties as the zircaloy-4 material.

The ability of the new 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 and the generation of hydrogen has been reduced. The approved topical report evaluated postulated accidents that involved adverse core conditions and the release of radionuclides and found the M5 alloy to perform similar to the current fuel rod cladding material. Rod cladding failures are assumed to occur in the fuel handling accident; however, the consequences of this event is independent of the properties of the fuel rod cladding. This is based on the fuel handling event assuming the rupture of fuel rods regardless of the rod cladding material. Therefore, based on the topical report results, the proposed revision to allow the use 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.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed M5 rod cladding material has been demonstrated to have properties that are not significantly different than the current zircaloy-4 in maintaining the integrity of the fuel rods. The new material will not alter the functions of the rod cladding which is to provide a barrier against the release of radioactive material. Initial plant conditions, which is 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 new M5 rod cladding is not postulated to become an accident initiator based on the similarity with zircaloy-4, the possibility of a new or different kind of accident is not created.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

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 loss-of-coolant accident (LOCA) acceptance criteria are unchanged:

- peak cladding temperature of ≤ 2200 degrees Fahrenheit,
- maximum cladding oxidation of ≤ 17 percent of the total cladding thickness before oxidation,
- maximum hydrogen generation of ≤ 1 percent of the hypothetical amount if all of the cladding metal were to react,
- coolable geometry such that the core remains amenable to cooling, and
- long-term cooling to maintain core temperature at an acceptably low value and removal of decay heat for an extended period.

These requirements continue to be met with the new M5 fuel rod cladding material. The acceptance criteria for Departure from Nucleate Boiling (DNB) events has not changed and is still the 95 percent probability and 95 percent confidence interval that DNB is not occurring during the transient. The changes to material properties have been evaluated in BAW-10227P-A and all applicable acceptance criteria are met. In addition, the proposed revision to allow the use of M5 fuel rod cladding 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.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

VI. REFERENCE

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

ENCLOSURE 2

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

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 00-16
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1

5-4
6-13a

Unit 2

5-4
6-14

II. MARKED PAGES

See attached.

5.3 REACTOR CORE

or M5

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. R184

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 $\pm$ 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

CORE OPERATING LIMITS REPORT (continued)

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) R227

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).

6.9.1.14.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. R159

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.

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4. R76

6.9.2.2 This specification has been deleted.

R245

8. BAW-10227P-A, "Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000. (FCF Proprietary)
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or M5

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rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited
substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved R172
applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have
been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply
with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative
testing may be placed in nonlimiting core regions. Sequoyah is authorized to place a limited number of lead test R234
assemblies into the reactor, as described in the Framatome Cogema Fuels Report BAW-2328, beginning with the
Unit 2 Operating Cycle 10 core.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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VOLUME

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CORE OPERATING LIMITS REPORT (continued)

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) R214

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).

6.9.1.14.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. R146

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.

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6.9.2.2 This specification has been deleted. R231

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)

ENCLOSURE 3

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

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 00-16
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1

5-4
6-13a

Unit 2

5-4
6-14

II. REVISED PAGES

See attached.

CLEAN PAGES PROVIDED TO NRC AND EDMS ONLY)

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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. R159

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