Virginia Electric and Power Company
Richmond, Virginia 23261

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

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Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS NO. 1 AND 2
NOTIFICATION OF INTENTION TO USE
LEAD FUEL ASSEMBLIES WITH ADVANCED CLADDING MATERIALS

This letter provides notification of Virginia Electric and Power Company plans to load four lead test fuel assemblies supplied by Framatome Cogema Fuels (FCF), formerly the B&W Fuel Company, into North Anna Unit 1. It is our intention to begin irradiation of these assemblies in North Anna Unit 1 Cycle 13, which is currently scheduled to begin operation in June, 1997. This program will be limited to the irradiation of the four lead test assemblies, with the objective of demonstrating the performance of the FCF fuel assembly design features in the North Anna units under operating conditions typical of our normal fuel management.

The North Anna lead test assemblies will be very similar to Mark-BW17 assemblies previously irradiated in other Westinghouse-designed reactors. However, the North Anna assemblies will incorporate several new features, including use of an advanced zirconium-based alloy, designated as M5, for the fuel assembly structural tubing. The fuel rod cladding in these assemblies will be fabricated from two advanced zirconium based alloys, M4 and M5, which have previously been approved for use as cladding materials in demonstration assemblies in the McGuire Unit 1 and Three Mile Island Unit 1 reactors. Additional information on the design features of the lead test assemblies is provided in Attachment 1, along with a summary of the evaluations that will be performed to support their use at North Anna.

The Design Section of the North Anna Technical Specifications currently defines the fuel rod cladding material as either Zircaloy-4 or ZIRLO. The references listed in Administrative Section 6.9 for the Core Operating Limits Report (COLR) address evaluations with Zircaloy-4 or ZIRLO rod cladding materials. Therefore, use of these lead test assemblies will require an amendment to the operating license, in the form of a

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license condition and changes to the appropriate Technical Specifications. Further, use of a material other than Zircaloy-4 or ZIRLO for the cladding material will require an exemption to the requirements of 10 CFR 50.46, which requires the use of an approved ECCS evaluation model for reactors with Zircaloy clad fuel.

In addition, the NRC's July 29, 1986 safety evaluation report which approved our normal reload nuclear design methodology (VEP-FRD-42 Rev. 1-A) specified that, in its present form, our methodology could be applied only to Westinghouse-supplied fuel in Westinghouse-supplied reactors. Therefore, we will also be seeking NRC concurrence that our standard reload design methodology may be applied to the North Anna cores which contain the four FCF lead test assemblies. The remainder of the fuel in the cores will continue to be supplied by Westinghouse.

We plan to submit our proposed license amendments for the lead assemblies by August 1, 1996, for your review and approval. To provide sufficient time for including the lead assembly features into the North Anna Unit 1 Cycle 13 reload core design, we will be requesting that NRC approve our submittal by February 1, 1997.

While it is our intent to insert these assemblies into Unit 1 for three consecutive cycles of irradiation, we will also request that the proposed license conditions be applicable to both North Anna Units 1 and 2 to allow additional flexibility in the irradiation schedule for the lead test assemblies. To support this request, the evaluations in our submittal will be performed to support operation of the FCF assemblies in either North Anna Unit 1 and 2. Any effect of using the lead test assemblies will be incorporated into each appropriate cycle specific reload analysis.

If you have any questions about the proposed lead test assembly program, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachment

cc: U. S. Nuclear Regulatory Commission
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R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

Attachment

Planned Evaluations to Support Use of FCF Lead Test Assemblies in North Anna

General Description

The North Anna lead test assemblies will be very similar to Mark-BW17 assemblies previously irradiated in the McGuire Units 1 and 2, Catawba Units 1 and 2, and Trojan reactors. However, the North Anna assemblies will be an advanced Mark-BW17 design, incorporating several new features, including: mid-span mixing grids, an advanced (fine mesh) debris filter bottom nozzle, a quick disconnect top nozzle, a floating top end grid (only the middle grids on the Mark-BW17 design 'float'), and use of an advanced zirconium-based alloy, designated as M5, for the fuel assembly structural tubing. The fuel rod cladding in these assemblies will be fabricated from two advanced zirconium based alloys, M4 and M5. The majority of the rods will use M5 for the fuel rod cladding, but two of the assemblies will also contain a limited number of fuel rods with cladding fabricated from the M4 alloy. These two alloys have previously been used as cladding materials for limited numbers of fuel rods in demonstration assemblies in the McGuire Unit 1 and Three Mile Island Unit 1 reactors. The North Anna lead test assemblies will differ from these demonstration assemblies in using advanced alloys as the cladding material for all fuel rods in the assemblies, as well as using alloy M5 for the guide thimbles.

Evaluations to be Performed

Evaluation of the lead test assemblies will be performed jointly by FCF and Virginia Electric and Power Company. These evaluations will include both testing and analyses, and will address all aspects of safety, including mechanical, thermal hydraulic, neutronic, transient, and accident analyses.

- Material testing for the new cladding alloys includes strength measurements, corrosion behavior, and swelling and rupture characteristics. Because the M5 alloy will comprise the vast majority of the cladding and because M4 closely parallels the Zircaloy-4 alloy specification, the brittle fracture testing (cold water plunge tests) and the high temperature oxidation rate testing were conducted only for the M5 alloy. The results of these tests indicate that the 17 percent local oxidation limit remains applicable and that the Baker-Just metal-water reaction rate correlation remains conservative. Swelling and rupture data, for use in benchmarking the rupture characteristics of the LOCA evaluation model, have been obtained for both alloys. The testing covers the expected range of parameter variation, providing sufficient basis for an adjustment to the assumed materials characteristics if required.

- Mechanical and hydraulic tests will also be conducted on components and a prototype fuel assembly. Component testing will address holddown spring and nozzle compression, hydraulic pressure drop, grid dynamic crush testing, and general functional tests. The prototype mechanical tests, comprising measurement of assembly static and dynamic responses to various steady state and impact loading situations, have been completed. Additional tests of assembly pressure drop, control rod drop time, life and wear, and flow induced vibration susceptibility will also be performed.

- Using approved methods and established design limits, FCF will verify that the static and dynamic structural characteristics of the lead test assemblies are compatible with the resident fuel in the core. The lead test assemblies will be designed to maintain their mechanical integrity through the planned operating life of the fuel. The evaluations will address normal operation, faulted conditions (seismic and LOCA), and shipping and handling loads. The lead test assemblies will also be designed for compatibility with all core internals, instrumentation, and control rod assemblies.

- FCF evaluations of the fuel rod mechanical and thermal performance for the lead test assemblies will be completed using their approved codes and methods (BAW-10162P-A and BAW-10084P-A). The fuel rod strength analysis will be performed in accordance with BAW-2133P, "Mark-BW Advanced Cladding Fuel Rod Evaluation," which was submitted to the NRC, and referenced in subsequent analyses, to support Technical Specifications changes for the use of the advanced cladding materials at other reactors. The mechanical analyses will address shipping and handling, stress, creep collapse, strain and fatigue. The thermal performance analyses will demonstrate that criteria for fuel rod temperature and internal pressure are met. The transient portions of these evaluations will initially be performed using generic FCF input for bounding plant transient conditions. During the design phase for each actual operating cycle, plant and cycle specific information will be provided to FCF by the Virginia Electric and Power Company to allow verification that the generic data remain applicable, or - if necessary - to allow reevaluation for the actual operating conditions. These cycle specific data will be generated with Virginia Electric and Power Company codes using methods which are consistent with FCF's methodology approved by the NRC.

- The thermal hydraulic design evaluation of the lead test assemblies will be performed by FCF using their approved methods and information on the characteristics of the core internals and the resident fuel supplied by Virginia Electric and Power Company. Mixed and full core evaluations will be performed to account for the lead test assembly impacts on minimum DNBR, pressure drop, fuel assembly lift and lateral flow velocities. It is expected that the DNBR performance for the lead test assemblies will be bounded by the resident fuel. The thermal hydraulic (DNB) evaluation of the

applicable North Anna reload cores would then be performed by Virginia Electric and Power Company assuming a full core of Westinghouse fuel.

- Virginia Electric and Power Company will perform the neutronic evaluation of the cores containing the FCF lead test assemblies in accordance with our normal reload nuclear design methodology (VEP-FRD-42 Rev.1-A). The nuclear design will ensure that the lead test assemblies are not placed in the highest rod power density locations.

- Virginia Electric and Power Company will assess the impact of the lead test assemblies on the non-LOCA core accident analyses. Any lead test assembly design features which differ from the resident fuel and which might affect input to the safety analyses will be defined, and the impact of these design changes on the analyses of record will be assessed. Because the lead test assemblies will not be placed in the highest rod power density locations, it is expected that they will be bounded by the safety analyses performed for the resident fuel.

- The LOCA performance of the lead test assemblies will be calculated with the Framatome Technologies RSG LOCA Evaluation Model. This evaluation model is described in BAW-10168 and has been approved by the NRC. Inputs which are descriptive of the Mark-BW17 lead test assemblies will be compiled and incorporated into a large break LOCA model representing the North Anna units. The treatment of swelling and rupture in the advanced Mark-BW17 LOCA calculations will be based on ongoing evaluations of the rupture testing data. The rupture characteristics of the M4 alloy are essentially the same as those of Zircaloy-4. Thus the modeling of NUREG-0630 is directly applicable to that alloy. The rupture characteristics of the M5 alloy are similar to M4 and Zircaloy-4, but a judgement as to the applicability of NUREG-0630 has not yet been finalized. If the rupture characteristics of the M5 alloy are determined to be within the range of applicability of the NUREG-0630 data correlations, the material properties as determined within NUREG-0630 for Zircaloy-4 will be applied. However, if the ongoing evaluations determine that a meaningful difference exists between the rupture properties determined in NUREG-0630 and those which would be appropriate for the M5 alloy, the material properties for the NUREG-0630 model will be rederived or adjusted based on the rupture test results. The LOCA calculations would then be conducted with the revised material properties appropriate for the advanced alloy.

The calculational base to be developed will include a burnup sensitivity study, a 3-break mini-spectrum, and a 2-elevation K_z validation study. These studies will be conducted for the M5 cladding because its properties differ from Zircaloy-4 more than the M4 material properties. Upon completion of the studies, the predicted peak cladding temperatures will be compared to those for the resident Westinghouse fuel, and a LOCA differential result, DPCT, will be established. If the DPCT appears to be strongly dependent on material

properties which differ for the M4 and M5 alloys, selected calculations will be redone using the M4 properties to establish a DPCT specific to the M4 alloy. The DPCT(s) will be applied to the licensing calculational results for the resident fuel design to provide the licensing basis for the lead test assemblies. It is expected that the DPCT(s) will be substantially negative, so that the LOCA analyses of record (based on a full core of Westinghouse fuel) will remain bounding for the cores which incorporate the FCF lead test assemblies. At this time, there are no plans to impose peaking requirements on the lead test assemblies for the LOCA analyses which differ from those applied to the remainder of the core.

Where any differences exist between the North Anna Unit 1 and Unit 2 designs, a bounding approach will be taken for the aforementioned evaluations to support operation of the FCF assemblies in either North Anna core. This will ensure that the lead test assemblies are technically capable of operating in either unit. This degree of flexibility may be desirable if reinsertion is delayed to allow time (off critical path) to perform more extensive characterization of the assemblies after one or two cycles of operation: upon completion of the examinations, rather than waiting a year or more for the next refueling outage to reinsert the LTAs, the assemblies could instead be incorporated into the next cycle of the other unit. Similarly, if NRC review and approval of the program can not support the proposed use in North Anna 1 Cycle 13, it would be possible to irradiate the lead test assemblies in the North Anna Unit 2 reactor rather than wait a full cycle to conduct the program in North Anna Unit 1. Any effect of using the lead test assemblies will be incorporated into the cycle specific reload analyses for each applicable cycle.