VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 4, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 960
NO/WRM:jmj
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Dear Mr. Denton:

ADDITIONAL INFORMATION
SURRY POWER STATION UNITS NO. 1 AND NO. 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE

In our letter dated May 15, 1980 (Serial No. 400), we proposed changes to the Technical Specifications for Surry Units No. 1 and No. 2 to permit an increase in the enrichment limit for new fuel.

In our letter dated September 23, 1980 (Serial No. 773A), we agreed, in response to a verbal request from a member of your staff, to provide a general evaluation of reactor operation to extended fuel burnups possible with higher enrichment fuel. This evaluation is provided in Attachment 1.

If you require any additional information, please contact this office.

Very truly yours,

B R Sylvia

B. R. Sylvia
Manager - Nuclear
Operations and Maintenance

Attachment

cc: Mr. James P. O'Reilly
Office of Inspection and Enforcement
Region II

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1.0 Introduction and Conclusions

In the past batch average discharge burnups at the Surry Power Station have averaged 28,000 to 33,000 megawatt days per metric ton (MWD/MTU) of uranium. In order to improve uranium utilization, reduce nuclear fuel cycle costs and reduce the number of discharge fuel assemblies, batch average discharge burnup extensions to approximately 45,000 MWD/MTU are being proposed. Vepco is participating in a program sponsored by the Department of Energy to demonstrate extended burnup technology. The Department of Energy (Reference 1) has determined that this improved fuel utilization program will have no significant impact on the environment. From a general safety viewpoint, Reference 1 states that, "no changes to existing facilities or to any aspects of fuel design or fuel use will be required. Radionuclide release from extended burnups of the current design fuel will be within normal facility design considerations and no change in the safety and accident considerations of light water reactors are expected." The purpose of this safety evaluation is to document that the general safety assessment provided in Reference 1 is also applicable specifically to the Surry Power Station.

The potential safety impacts of higher burnup fuel include the two basic areas of fuel performance and safety analysis. Sections 2.0 and 3.0 document the evaluations of the potential fuel performance and safety analysis impacts, respectively.

From these evaluations, the following conclusions can be drawn:

1. Westinghouse fuel performance at the Surry Power Station will

retain its current high level of reliability at increased fuel burnup levels.

2. The safety analyses for the Surry Power Station are not significantly impacted at increased fuel burnup levels.

3. Westinghouse reload safety evaluation methodology (Reference 2) assures an effective cycle by cycle check to insure continued plant safety at increased fuel burnup levels. Should any fuel performance changes be identified, they will be appropriately incorporated in the safety analysis as required.

2.0 Fuel Performance

High burnup fuel will not have a significant impact on either fuel design or operation. Westinghouse fuel is designed to meet the fuel rod design bases/criteria listed below:

1. Fuel Centerline Temperature
2. Fuel Rod Internal Pressure
3. Cladding Stress
4. Cladding Strain
5. Cladding Fatigue
6. Cladding Collapse

Further description of these design bases/criteria is given in Reference 3. At the present time, Westinghouse fuel designed for the Surry Power Station must meet the indicated design criteria. The higher burnup fuel designed for the Surry Power Station will also be required to meet the design criteria or the designs will be precluded from further consideration. Furthermore, the high burnup fuel performance will be assessed against the current fuel related Technical Specifications. Specifically, high burnup fuel will be required to meet the current peaking factor and reactor coolant activity limits or the impact on the current safety analyses will be assessed as discussed in Section 3.0.

3.0 Safety Analysis

As indicated in Reference 1, no fundamental change in the safety and accident considerations of the LWR are anticipated as a result of greater discharge exposure fuel. Higher burnup fuel could potentially impact those safety analyses for which fuel failures were originally postulated. As indicated in the Surry Power Station Final Safety Analysis Report (Reference 4), these accidents include Steam Generator Tube Rupture, Steam Pipe Rupture, Fuel Handling Accident, Volume Control Tank Rupture, Waste Gas Decay Tank Rupture and the Loss of Coolant Accident. An evaluation of the impact of higher burnup fuel on each of the accidents is discussed in Sections 3.1 through 3.5. Section 3.6 discusses the Westinghouse methodology which verifies the applicable licensing analyses on a cycle by cycle basis.

3.1 Steam Generator Tube Rupture

It is assumed that the accident takes place at power and while the reactor coolant is contaminated with fission products corresponding to continuous operation with one percent of the fuel rods defective. The accident leads to contamination of the secondary systems due to leakage of radioactive coolant from the reactor coolant system and in the event of a coincident loss of offsite power, there will be a discharge of activity to the atmosphere through the steam generator safety and/or power operated relief valve. Since the release will occur because of leakage of the primary coolant, maintaining the same technical specification limits on coolant activity would insure that the quantitiy of radionuclides available for release from the coolant would be unaltered. In addition, the offsite radiological consequences of this accident are based on the airborne releases of volatile fission products (noble gases and radioiodine). All of the important radioactive iodine and noble gas nuclides are of short half-life compared to the fuel cycle time, with the exception of Kr-85. For these nuclides, equilibrium inventories in the fuel are attained relatively quickly. Thus, the quantity of these volatile fission products released in the event of a fuel failure is independent of burnup. In the case of Kr-85, both the small inventory present and the small contribution of Kr-85 to the total dose result in insignificant changes in the radiological consequences of this accident.

3.2 Steam Pipe Rupture

The limiting steam pipe rupture accident for the Surry Power Station will not result in departure from nucleate boiling. Consequently, no radioactivity is released to the environment because of a steam line break unless there is or has been primary to secondary system leakage in a steam generator. Current Technical Specifications limiting primary to secondary leakage and reactor coolant system activity will not change as a result of higher burnup fuel. These limits are well below values assumed in the licensing analysis. In addition, the offsite radiological consequence results will not be significantly impacted as discussed in Section 3.1.

3.3 Fuel Handling Accident

For the purposes of evaluating the radiological consequences of this accident, all rods in a fuel assembly are assumed to rupture with a sudden release of the gaseous fission products held in the voids between the pellets and cladding of the fuel rod. The low temperatures of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Although larger quantities of long lived fission products will be present in the fuel,[5] review of the source terms applicable to this accident indicates that the major contributors to the dose result from short-lived isotopes of such radioactive gases and volatiles as xenon, krypton, and iodine.[6] Since these short-lived gases and volatiles reach equilibrium concentrations after about one year of irradiation, there is no significant increase in site boundary dose due to the release of radioactive gases and volatiles during a fuel handling accident from high burnup fuel as compared to standard burnup fuel.

3.4 Radioactive Gas Release

The concentration of radioactive waste gases in the primary and auxiliary systems is a function of the rate of fission gas release to the coolant from defective fuel and the rate of removal via the auxiliary systems. The components which retain significant concentrations of radioactive gases are the volume control tank and the waste gas decay tanks. The radioactive gas release analysis considers the rupture of the volume control tank and a waste gas decay tank with the instantaneous release of the radioactive gas inventories of each to the environment. Since the Technical Specifications limit on reactor coolant activity is not changing, the volatile fission products available for the airborne releases are not significantly impacted as discussed in Section 3.1.

3.5 LOCA

The loss-of-coolant accident (LOCA) would not become more adverse if the use of higher burnup fuel is implemented. In the loss-of-coolant accident, peak linear heat rates and the design of the emergency core coolant system must be such as to limit clad temperatures and oxidation to values prescribed by regulations (10 CFR 50, Appendix K). The primary fuel characteristics which can influence clad temperatures and resulting oxidation are the amount of stored energy and the decay heat source immediately after shutdown. The quantity of stored energy present in the fuel is dependent upon the peak linear heat rate during operation, but is independent of burnup. Since the fuel is designed and the plant is operated to limit the peak linear heat rate to values which are established to be acceptable for LOCA, stored energy will not increase as a result of the proposed changes.

The decay heat source term immediately after shutdown is due almost exclusively to the decay of short-lived fission products which reach a saturated concentration during the first year of irradiation.[1] Thus, the proposed changes, and particularly the increase in discharge exposure, will not significantly alter the decay heat generation during the loss-of-coolant event. Furthermore, the Nuclear Regulatory Commission requires the use of the ANS decay heat standard (see 10 CFR 50, Appendix K) based upon infinite operation plus 20% uncertainty, which is thus conservative and already encompasses extended burnup.[7] Since neither the stored energy nor decay heat generation rate increases, the proposed changes are not expected to result in an increase in the severity (i.e., in peak clad temperatures or oxidation) of the loss-of-coolant accident or to increase the chances of meltdown, and hence improvements or changes in emergency core cooling systems are not required.

The proposed changes (in particular the increase in discharge exposure) also are not expected to significantly alter site boundary doses as a result of the loss-of-coolant accident. Site boundary doses are due almost exclusively to the short-lived fission products which reach saturated concentration during the first year of irradiation,[6] so that the inventory (curies) of isotopes which are significant in the calculation of site boundary dose during the loss-of-coolant accident is virtually independent of burnup.[5] Although increased fuel failures during the LOCA event are not anticipated, the Nuclear Regulatory Commission requires that site boundary doses be evaluated under the conservative assumption of 100% fuel failure.[8] Thus, any tendency for increased fuel failures during LOCA of high burnup fuel will not alter the perceived consequences of the loss-of-coolant accident.

3.6 Reload Safety Evaluation Methodology

The Westinghouse reload safety evaluation methodology will continue to serve as the mechanism employed to confirm the validity of the existing safety analyses for each fuel cycle. The methodology is used to identify technical specification changes and potential unreviewed safety questions. This methodology will readily identify any potential impact higher burnup fuels may have on plant safety and will verify the existing safety analyses.

Should the reload safety evaluation process identify a change in the fuel performance (e.g., an increase in the number of rods predicted to have a minimum DNBR less than or equal to design limits), these changes will be reviewed by the appropriate radiation specialists to determine the impact of these changes on the radiological consequences of the particular event in question. Such changes will be documented in accordance with the requirements of 10 CFR 50.59.

4.0 References

1. DOE/EA-0118, "Environmental Assessment, DOE Program to Improve Uranium Utilization in Light Water Reactors," August 1980.

2. WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology," March 1978.

3. Docket No. STN 50-572, "RESAR-414, The Reference Safety Analysis Report Documenting the Westinghouse 3820 MWth Nuclear Steam Supply System," October 8,1976.

4. Final Safety Analysis Report - Surry Power Station Units 1 and 2, Virginia Electric and Power Company, December 1969.

5. "NASAP Preliminary Safety and Environmental Information Document (Volume I, Pressurized Water Reactors) - Responses to NRC Comments", August, 1979.

6. "Reactor Safety Study",WASH-1400 (NUREG-75/014), October 1975.

7. Title 10, Part 50, Code of Federal Regulations, Appendix K.

8. Regulatory Guide 1.4, "Assumptions Used For Evaluating The Potential Radiological Consequences Of A Loss Of Coolant Accident For Pressurized Water Reactors", Revision 2, June 1974.