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MAR 3 1976

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendments No. 20 and 17 to Facility Licenses No. DPR-33 and DPR-52 for the Browns Ferry Nuclear Plant, Units 1 and 2. These amendments are in response to your request of November 5, 1975, as supplemented November 28, 1975 and February 5, 1976.

These amendments authorize modification to Browns Ferry Nuclear Plant, Units 1 and 2 by approving the drilling of the fuel assembly lower tie plates of Types 2 and 3 fuel assemblies to provide bypass flow. This bypass flow was originally provided for by holes in the lower core support plate. By Amendments 17 and 14 to Licenses DPR-33 and DPR-52 for Units 1 and 2, respectively, authorization was issued to plug the holes in the lower core support plate to eliminate significant in-core instrument tube vibrations. These amendments do not authorize operation of Units 1 and 2 with the plugged core support plate and drilled fuel assemblies. Operation with these modifications will not be authorized until a later safety evaluation is completed that addresses the effects on operation.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 5, 1975, as supplemented November 28, 1975 and February 5, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by adding paragraph 2.C(6) to read as follows:

2.C(6) The facility may be modified by drilling bypass flow holes in Type 2 and Type 3 fuel assemblies as described in NEDO-21091, "Browns Ferry Nuclear Plant, Units 1 & 2 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations: and NEDE-21156, "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations", dated January 1976.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Date of Issuance: March 3, 1976

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated November 5, 1975, as supplemented November 28, 1975 and February 5, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by adding paragraph 2.C(6) to read as follows:

2.C(6) The facility may be modified by drilling bypass flow holes in Type 2 and Type 3 fuel assemblies as described in NEDO-21091, "Browns Ferry Nuclear Plant, Units 1 & 2 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations: and NEDE-21156, "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibrations", dated January 1976.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Date of Issuance: March 3, 1976

SAFETY EVALUATION REPORT
ON THE REACTOR MODIFICATION TO
ELIMINATE SIGNIFICANT IN-CORE VIBRATION
IN BROWNS FERRY UNIT 1 AND UNIT 2
DOCKET NOS. 50-259 AND 50-260

By

Office of Nuclear Reactor Regulation

U. S. Nuclear Regulatory Commission

Table of Contents

1.0 Introduction

By letter dated November 5, 1975, Tennessee Valley Authority (TVA) applied for amendments to Operating Licenses DPR-33 and DPR-52 for Browns Ferry Nuclear Plant Unit 1 and Unit 2 to authorize plugging of the bypass flow holes in the lower core support plate and drilling new bypass flow holes in the fuel assembly lower tie plate. In support of the application, TVA provided the General Electric report NEDC-21091, "Browns Ferry Nuclear Plant Units 1 and 2 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations". On November 14, 1975 Amendment 17 to DPR-33 and Amendment 14 to DPR-52 authorized the plugging of bypass flow holes in the lower core support plate. By letter dated November 28, 1975, TVA submitted a non-proprietary version of the above GE report, NEDO-21091.

This safety evaluation addresses the acceptability of drilling the fuel assemblies, but does not address the acceptability of reactor operation with the drill assemblies. The consideration of operation with the drilled fuel assemblies and plugged lower core support plate, along with any associated operating limits, will be the subject of a later safety evaluation report that must be completed prior to issuing amendments that will authorize such operation.

2.0 Background

In late 1974, a foreign BWR observed a change in the characteristics of the readings from certain of the in-core instruments. Subsequent examination of the fuel bundle channel boxes in the foreign reactor revealed significant wear on the corners of channel boxes adjacent to instrument and source tubes. This wear had led to cracking and holes in the channel boxes adjacent to the instrument that had displayed the anomalous readings. The General Electric Company notified the NRC immediately of a possibly similar problem in domestic BWR/4 plants. Subsequently, the NRC ordered all the utilities with a similar reactor to inspect for this characteristic noise* and to notify the NRC if the noise level exceeded the predetermined acceptable level. The channel degradation was caused by vibration of instrument and source tubes excited by high velocity coolant flow from the 1-inch diameter bypass holes in the core support plate.

The presence of cracks or holes in a channel box is of concern since it would allow part of the cooling water that normally flows through the fuel bundle to flow out of the cracks or holes and bypass the fuel rods. Such a change in flow pattern would decrease the safety margins for the thermal performance of the fuel. These reduced margins could lead to overheating and damage to the fuel in the event of some anticipated operating transients or some postulated accidents. Significant wear and cracking of the channel boxes would also affect their mechanical strength for transients and accidents.

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*Noise is defined as the ratio of fluctuations in the signal in the frequency range of interest (generally 1-4 Hz), divided by the mean value of the signal.

If large cracks occur in channel boxes, there could be a potential for direct impacting of the tubes on fuel rods or interference with control rod movement.

The NRC ordered those plants with a high Traversing In-Core Probe (TIP) noise level to lower coolant flow and power to minimize the damage to the channel. On July 18, 1975, the staff issued a safety evaluation report[(1)*] stating that no further damage to the channel boxes is expected when the flow rate is reduced. Also, the staff concluded that when the reactors are operated at the reduced power level described in the GE submittal[(2)] the reactors will not present an undue risk to the health and safety of the public, even with degraded channel boxes. Some utilities, e.g., operators of the Duane Arnold and the Vermont Yankee BWR/4's, decided to shut down the reactors and plug the bypass holes in the lower core plate. The NRC approved such an action[(3)] and stated that plugging only could result in an allowable power penalty for some reactors.

Concurrent with this action, GE has developed a permanent reactor modification to eliminate significant in-core vibration. The permanent modification consists of both drilling two holes in each fuel bundle lower tie plate to provide an alternate bypass flow path and at the same time plugging the 1-inch bypass holes. The GE development of this permanent modification for the channel box wear problem has been completed and reported to the staff[(6)] . The staff has completed

*References are numbered and listed in Section 8.

its generic review of the permanent modification only for reactors employing fuel bundles with the holes drilled in all lower tie plates in conjunction with plugging of all the 1-inch bypass holes (e.g., Browns Ferry 3). The review is summarized in this safety evaluation report.

Concurrently the staff has reviewed the effects of drilling holes in the lower tie plates for some but not all of the fuel bundles within the core (e.g., Browns Ferry 1 and 2). Since the number of bundles with holes drilled in the lower tie plate directly affects the bypass-region-to-bundle flow rates and the reflood rate for ECCS, the safety analysis for those reactors not having holes drilled in all fuel bundles must be reviewed on an individual basis. Thus, the complete evaluation for operating limits on any reactor having drilled holes in only some of the fuel bundles is excluded from the scope of this summary. However, the mechanical and hydraulic considerations of operating with only some of the fuel bundles having drilled holes were considered.

3.0 Fuel Channel and Reactor Internal Inspection

3.1 Inspections and Wear Criteria

As a routine part of planned reactor shutdowns, the channel boxes and instrument and source tubes are visually inspected for corner wear. Cracks or holes in the channel boxes are readily apparent in the spent fuel pool without optical aids. The results on each channel are compared with predetermined acceptance criteria for reuse. The bases for establishing acceptable wear limits as well as the inspection plan are discussed in the GE report NEDC-20994[4].

The radial depth of the wear on the channel box corners was estimated from an inspection procedure used at several BWR/4 reactor sites. The inspection station was located at the fuel storage pool using a fuel preparation machine, a borescope and a visual standard. The channel wear was observable visually by the contrast between the Zircaloy-4 metal and the zirconium oxide adhering to the unworn portion of the channel box. Cracks and penetrations were observable by their lack of light reflection. The widths of the wear marks were measured by direct comparison with the known dimensions on the visual standard. The depth of wear was inferred from a simple Pythagorean derivation for the radial overlap of two eccentric circles (Figure 2-2, reference 4). This inference assumes no horizontal wiping of the tube on the channel. The depth from uniquely wiping wear is only 42% of that inferred by this technique. Thus, the technique used to estimate corner wear was conservative.

General Electric has performed visual inspections specifically for channel box wear at 18 reactors (9 with bypass flow holes in the lower core plate and 9 without bypass holes). The results of all the reported inspections have been reviewed in detail by the staff. More than 1600 channel boxes were examined during these inspections at those plants with bypass flow holes. Only some in-core tubes are adjacent to bypass holes. No unusual wear was observed at any channel box corner not adjacent to in-core instrument or source tube. The reject rate for channels adjacent to bypass holes is about two times higher than the reject rate for channels adjacent positions with no bypass holes. Thus, the staff has concluded that the joint presence of both in-core instrument and source tubes and bypass flow holes was necessary to cause significant channel box corner wear.

The results of the more detailed inspections at nine other reactors having no bypass holes in the core plate have also been reviewed. The inspections were focused upon more than 100 channels adjacent to in-core instrument and source tubes. The results show that reactors without bypass holes in the lower core support plate have exhibited no significant channel box corner wear.

General Electric recommends two types of channel inspections: diagnostic and general. The procedure is to diagnose the extent of wear by sampling selected channels and by performing a general inspection for all the channels adjacent to an in-core instrument tube only when the diagnosis yields evidence of significant wear.

When the channel wear problem was first identified GE, re-investigated their channel box design margins. They found that when a channel box corner was worn less than .01 to .02 inches (the nominal wall thickness is 0.08 inches) the original design limits were not violated. This conclusion was based upon a stress analysis of the channel boxes considering all modes of loading conditions such as steady state, fatigue, steam line break and seismic . GE identified fatigue as the limiting design loading. The fatigue loadings result from pressure variations from normal operations (e.g., startups and shutdowns, daily and weekly load reductions, and rod worth tests) as well as the various abnormal transients (e.g., pump trip, turbine trip, generator load rejection, etc.). The information supplied[4] was not sufficiently comprehensive to perform an exhaustive review of the channel integrity. However, the staff performed several bounding calculations for maximum allowable wear and found that GE wear limits are acceptable.

There are four types of instrument and source tubes in a BWR. They are Local Power Range Monitor (LPRM), Source, Intermediate Range Monitor (IRM), and Source Range Monitor (SRM). When there is excessive vibration, these stainless steel tubes impact or rub against the Zircaloy channel box corners. GE has inspected over half of the total number of in-core instrument tubes at two different BWR/4 reactors. Two LPRM tubes were replaced because they exceeded GE's wear limits. It should be noted though that those two tubes were located where channels experienced through-wall wear and some pieces of the channel were torn off.

The GE criterion for allowable wear on the instrument tube is approximately 20% of the nominal thickness and could mean that the tube resistance to collapse was reduced to half its original resistance. The staff's calculation indicated that .01 inches of wear does not constitute a significant reduction from the original safety margin. We therefore conclude that the allowable wear for the SRM and IRM tube should not exceed 0.01 inches and the criterion be applied in all future plant inspections. Furthermore, we require that all the in-core tubes be inspected prior to restart when the diagnostic inspection indicates that there is significant wear on the channels in a BWR/4.

3.2 In-Core Instrument Noise

When the core flow exceeds about 40 percent of rated flow for BWR/4's with bypass flow holes, the signal from the fission detectors of the LPRM subsystem and the TIP subsystem exhibit a characteristic noise associated with vibrating LPRM instrument tubes. This characteristic noise in the TIP traces and LPRM time traces has a frequency range of about 1 to 4 Hz. However, other low frequency noise is also observed in these signals and is similar to that observed in BWR/3's.

The neutronic signals generated by the fixed LPRM detectors and the moveable (or parked) TIP detectors and as recorded by plant or special recording instrumentation can be correlated with the impacting of channel box corners and instrument tubes in a number of ways. A direct approach consists of estimating the 1 to 4 Hz noise content in a TIP trace or an LPRM time trace. Another approach consists of using noise analysis techniques and computing either the power spectral density (PSD) as a function of frequency for a detector or the cross power spectral density (CPSD) as a function of frequency for any two detectors. The acoustic* noise caused by impacting instrument or source tubes on channel boxes can also be measured with accelerometers attached to instrument/source tube components that are external to the reactor pressure vessel. Other approaches which use piezoelectric affects (TIP detector as a sensor) may also be used as an indicator of vibration.

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*The signals recorded with the accelerometers are termed "acoustic noise" in this report for the sake of brevity and convenience.

All of the various methods of relating observations on this impacting and vibration of instrument/source tubes indicate the same trends. BWRs with plugged bypass flow holes in the lower core support plate indicate little neutronic or acoustic noise characteristic of the vibrating or impacting of instrument tubes on channel box corners. BWR/4s with bypass flow holes open but with core flows restricted to 40 percent or less of rated flow also indicate similar results. But BWR/4s with bypass flow holes open and operating in the range of 40 to 100 percent of rated flow exhibit neutronic/acoustic noise varying from slight to considerable for the affected instrument/source tubes.

The measured channel box corner wear for several BWR/4's has been shown to correlate with neutronic noise, either directly estimated or computed PSDs or CPSDs. However, the correlations are not strong. All that can be said is that the greater the neutronic noise with a frequency content of 1 to 4 Hz at a given location the greater the expectation of channel box corner wear. Establishing a reliable correlation is difficult due to the complexity of the phenomena (e.g., number and placement of bypass flow holes around an instrument source tube, the motion of the affected tube and fuel channels, the control rod position and previous history, the in-channel void content, the bypass region void content, core wide flux gradients, microphonic noise of the detectors, variations in core flow, and the quality of the plant measuring systems). Quantitative aspects of the effect of position and voids on the detector signal have been studied by our consultants at the Brookhaven National Laboratory [5]. The calculations performed by our consultants generally support the previously stated observations concerning neutronic noise caused by vibrating instrument tubes.

Although the effect of instrument tube movement and channel box corner wear on neutronic noise is generally understood, it is currently not possible to predict the occurrence of holes, splits, or cracks in channel boxes. We believe that the general complexity of the associated phenomena, the range of reactor operating states and the lack of sophistication of plant instrumentation precludes exact predictions of the occurrence of holes, splits, or cracks in channel boxes. However, we conclude that trends in measurements over a period of time, with reactor operation at substantial core flow rates permits an assessment of the potential for substantial channel box damage.

Therefore, based on our own analysis of the channel box corner wear data and neutronic noise, the study performed by our consultants, and a review of the information from domestic BWR/4s concerning channel wear and noise, we conclude that:

(1) BWRs with plugged or no bypass flow holes in the lower core support plate do not have any significant

neutronic or acoustic noise of the type associated with the channel wear problem,

(2) BWR/4s with bypass flow holes do not have any significant neutronic or acoustic noise, of the type associated with the channel wear problem, if the core flow is restricted to about 40 percent of rated flow or less,

(3) the measured neutronic and acoustic noise, for BWR4s with bypass flow holes open, increase as a function of increased core flow,

(4) neither neutronic or acoustic methods are presently capable of indicating the occurrence of holes, splits, or cracks in a channel box, and

(5) noise measurements need to be evaluated over a period of time to monitor any changes or abnormalities as an indication of potential for channel box wear.

4.0 Evaluation of Reactor Changes

4.1 Mechanical Effects

General Electric has proposed to reduce the vibration of instrument and source tubes by eliminating adverse crossflow because of the 1-inch bypass holes in the lower core support plate adjacent to these tubes. The design change proposed to eliminate adverse coolant crossflow at in-core tube elevations is to both drill two holes in each fuel bundle lower tie plate and to plug the bypass holes in the lower core support plate. The two drilled holes are always located at the narrow-narrow interchannel gap and not at the wide-wide gap where the flow might impinge on the control blades. With all the bundles drilled there are approximately ten times as many holes as there were in the core support plate, and the total flow area is slightly less. The holes in the fuel bundle lower tie plate are slanted to direct coolant flow down toward the core support plate prior to mixing into the total bypass flow which is upward. This results in a more uniform flow throughout the core at elevations adjacent to the in-core tubes. The uniformity of flow was demonstrated at the GE cold flow test facility by measuring axial velocity distributions.

Drilling only some of the fuel bundles is expected to provide a partial benefit of reduced adverse crossflow at elevations adjacent to in-core tubes. Thus, no adverse effect on channel box wear is expected when operating with only some of the bundles having holes drilled in their lower tie plates.

The lower tie plate serves to support the weight of the fuel bundle and rests on a fuel support casting (see Figure 5-3, reference 4). Both components are stainless steel. The thickness of the tie plate wall is approximately 1/2 inch at the holes. A stress analysis (including the stress concentration factor for the holes) indicated that the stress levels are an order of magnitude below the allowable stress when all the expected loads are considered for normal, abnormal and postulated accident conditions.

GE also investigated implications of a misoriented bundle where the flow would be directed toward the control blade. Simulated tests in the cold flow facility at San Jose showed no abnormal control rod vibration. GE further examined the effect of this design change on other internal components (e.g., core support plate, guide tubes, shroud support) and found no significant effect.

Plugging the bypass holes is also a part of the reactor modification. The staff's safety evaluation of such plugs was performed prior to issuance of the license amendments on November 14, 1975 that authorized plugging of the bypass flow holes in the lower core support plates. The conclusions of that evaluation are supported by the service experience of plugs at the Vermont Yankee and the Pilgrim 1 reactors where plugs were installed to eliminate control curtain vibration. Post-service examination of an extracted plug exhibited neither degradation nor wear of the plug after one fuel cycle. The possibility of plug vibration from the flow through the two drilled tie plate holes was investigated by GE at the same cold test facility with full size plugs and tie plates. No unacceptable plug vibrational response was found as measured by accelerometers.

Long-term fatigue, creep and relaxation of parts of the plug however, should be monitored by reasonable sampling inspection at each outage of the lead plants including some non-destructive and destructive tests. GE proposed an extensive plug surveillance[4] program which the staff considers mandatory (see section 6).

While developing and demonstrating the plant modification to eliminate wear caused by in-core tube vibration, GE has also developed a method of machining the lower tie plates. The implementation will be performed in two steps: drilling and deburring of the fuel bundle lower tie plate. These operations on exposed fuel will be performed in the fuel storage pool under about 25 feet of water.

The implementation procedure employs pneumatic drills and clamping devices. Care has been taken in the design of the equipment to preclude misorientation of the fuel bundle. The verification that all debris can be removed was demonstrated in a full-scale underwater test facility. We observed the underwater machining procedure. The rigors of the underwater machining procedure will necessitate close adherence by the personnel doing the machining to the specific Quality Assurance requirements.

General Electric has established several levels of contingency plans for possible difficulties during implementation. The plans begin with simple procedures and progress to the replacement of the entire fuel bundle. All contingency plans will be demonstrated before their implementation.

4.2 Nuclear Performance and Thermal Hydraulic Effects

Since only some of the fuel bundles are being drilled, we will require prior to issuing amendments authorizing operation with the drilled fuel assemblies that either

(1) a plant specific evaluation be submitted for a partially modified reactor, or

(2) the plant nuclear and thermal hydraulic parameters, characteristics, and performance for normal, transient and accident conditions be based on the more conservative plugged-only core configuration (e.g., reference 3).

5.0 Demonstration Tests

GE performed a cold hydraulic test at its San Jose facility to first determine the cause of in-core instrument tube vibration and channel box damage and secondly, to see that their proposed modifications will perform satisfactorily as expected. Thirty-two fuel bundles (4x8 array) were installed in a test tank with as-manufactured channel boxes, lower tie plates, control rod plates, fuel support castings and in-core instrument tubes. Plan views are given on pages 5-64 through 5-86 in reference 4.

There are some differences between the test and an in-reactor configuration. The LPRM tubes in the test are cut short to approximately 15′ and attached to a spring whereas these tubes are more than 40′ long in-reactor. All the internals in the LPRM tubes (TIP tube, fission chamber and cables) were removed to facilitate installing an accelerometer. The flow orifices of the fuel support castings were slightly altered to simulate the bypass flow volume. In some tests, fuel rods were removed from the channels and replaced by dummy weights. Also, the top of the fuel bundle is sealed (due to limited pump capacity) to simulate only bypass region flow and not flow through the fuel.

5.1 Mechanical

For the initial BWR/4 simulation, GE was able to produce significant impacting of an LPRM tube and channel box. When the proposed modification for operating reactors was tested, the impacting level was considerably reduced. The staff monitored these tests and observed them on several different occasions.

Additional tests were performed at the Moss Landing facility.[6] The test facility consisted of sixteen fuel bundles (4x4 array), one 0.750 inch OD LPRM tube, four control rod blades, a shroud and a pressure vessel. It simulated in-reactor temperatures and pressures but no two phase flow was introduced.

Two conclusions were drawn from the tests. First, the amount of bypass flow measured was more than expected. Secondly, the impact level between fuel bundle and LPRM tube was higher than the value observed in the previous cold tests at San Jose. GE reduced the lower tie plate hole size from the original to correct for the desired bypass flow.

The reasons for the higher "g" level observed by the accelerometer in the LPRM tube were also investigated. The difference can be attributed to the in-bundle flow. In the cold test, in-bundle flow was sealed off because of a limited pump capacity thus only simulating bypass flow between channels. When the flow was allowed to pass through the fuel bundle in a channel box at Moss Landing it caused a slight excitation of the fuel bundle thus adding to the LPRM tube vibration and impact. GE confirmed bundle vibration at the cold facility by opening the flow seal to four fuel bundles. Further tests were performed at Moss Landing for both the BWR/3 simulated configuration and the fully plugged BWR/4 mockup. GE found that the impact levels are the same as that of the BWR/4 with the complete modification (ranging between 4 to 8 g's). They also confirmed, at the same facility, that the BWR/4 with bypass flow holes in the core support plate produced accelerations about an order of magnitude higher. GE concluded that since the impacts for the BWR/3 and for the modified BWR/4 were equivalent and since no significant wear was observed in the BWR/3 channel inspections after full service life, the proposed BWR/4 modifications should eliminate the significant wear.

The Moss Landing tests employed those core components for use in both the BWR/3's and the BWR/4's (both modified and unmodified). Although the scale of the entire core was not simulated in the tests, the relative effects for the hydraulic and mechanical responses of the components were measured at Moss Landing. The measured impactings for tests from both the BWR/3 components and the modified BWR/4 components were significantly improved relative to those from the unmodified BWR/4 components. Based upon the above observations and the assumption that the outreactor tests are a scaled equivalent of reactor hydraulic and mechanical environments, we conclude that the instrument and source tube impact levels in the modified BWR/4's are expected to be equivalent to the BWR/3's. General Electric reported data to show that no significant wear from impacting has been observed in their BWR/3 surveillance program.

To provide verification of the expectations on actual operating reactors, we believe that a comprehensive surveillance program is needed which is further discussed in section 6. Final confirmation of the modification can only occur after the alternative flow path configuration has experienced a full fuel cycle of service. The plants employing this modified configuration need to schedule a post-irradiation surveillance on the channels at each outage for that purpose (see section 6).

5.2 Thermal and Hydraulic

Alternate flow paths and finger spring flow tests were performed by General Electric in the ATLAS facility which simulated the inlet geometry and bypass region for one fuel bundle under typical BWR operating conditions. GE has stated that all components used in these tests were typical of those in production and currently operating in BWR's which incorporate finger springs in the fuel design. The test results provided the applicant with flow loss coefficients for different hole sizes and leakage flow rates around the finger springs. General Electric used these test results to determine the hole size to be drilled in the fuel bundle lower tie plates.

6.0 Post Reactor Modification Surveillance

In the previous sections we have discussed the necessity of having a surveillance program during reactor operation to guard against the possible recurrence of channel box degradation. We believe that two different types of sensors can be used to monitor vibrations during power operations:

(1) in-core neutron detectors (TIPs), and

(2) accelerometers attached on the tube beneath the reactor which detects the mechanical energy of impact.

6.1 TIPs

Excessive instrument tube-channel box interaction previously has been determined from the neutronic noise level in unfiltered TIP traces. The plant modifications, including the plugging of the bypass flow holes, are expected to affect the noise content of the TIP traces. In particular, the noise in the 1 to 4 Hz frequency range caused by vibration of instrument tubes should be reduced relative to power dependent noise.

Based on our previous surveillance requirements, unfiltered TIP traces were taken prior to any plant modifications at the highest flow and power permitted. For some plants, TIP traces were also taken at a number of power and flow conditions. These data provide part of the basis for evaluating the efficacy of the reactor modifications. After the reactor modification, comparison of similar measurements with the pre-modification data will be made to confirm that the mechanical vibration of the instrument tubes has been substantially reduced. The unfiltered TIP traces taken during return to power operation will also provide baseline data which can be used to monitor any changes in the 1 to 4 HZ noise level not attributable to such causes as power level, core flow and control rod pattern.

Therefore, we conclude that

(1) surveillance using unfiltered TIP traces to monitor the efficacy of the plant modifications, and

(2) the frequency of taking TIP traces in accordance with GE Standard Technical Specifications (about 4 to 6 weeks of full power operation),*

are an acceptable means for monitoring neutronic noise of the type associated with instrument tube vibrations.

6.2 Accelerometer

Since April 1975, when we first learned of in-core tube vibration, considerable experience has been accumulated both at various reactors and the San Jose facility regarding the capability of accelerometers to detect significant impact. The Cooper, Duane Arnold and Peach Bottom reactors all demonstrated with acceleometers at different flow rates that there is a definitive transition in the flow rate below which no significant

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*GE STS Table 4.3.1-1 Item 2e and footnote f (December 1, 1975 revision).

impact of the in-core tube can be detected. This was the basis for allowing plants to operate at lower flow even though we suspected that some reduced wear rate may continue.

GE performed an experiment with a full-length LPRM tube mounted upright in the air. They then impacted the tubes with a hammer and monitored the stress wave with an accelerometer at various locations along the tube. NRC consultants and personnel from Philadelphia Electric Company, TVA and GE jointly experimented with a piezo-electric accelerometer at the Brown's Ferry plant during the current shutdown. All came to the conclusion that the accelerometer is a viable sensor that detects any significant impact of the in-core tube.

The first two reactors to employ the modified configuration should install accelerometers on the in-core instrument tubes. We regard this action necessary to provide further evidence of the efficacy of the modified reactor. The applicants involved should establish a one month surveillance interval and report to us any anomalous behavior observed in the accelerometer.

GE has already accumulated some accelerometer experience in a BWR/3 plant. This together with the experience obtained during power ascension flow tests at the Duane Arnold reactor[3] and other reactors with plugs only provides a reference for comparison.

6.3 Internals

GE presented a plan to inspect channel boxes at the earliest refueling outage. The first two reactors which implemented the plant modification will be required to perform detailed visual examinations of a statistically significant number of channel boxes for the first two refueling cycles after the modification. The results of current inspections indicate that outer pheripheral bundles may be more susceptable to a corner wear. The statistical sampling should emphasize channel boxes which appear more susceptable to wear.

GE provided a satisfactory program for the plug surveillance. It includes removal of two plugs each from the core after two, five and ten years of service. The plugs will be examined for wear, spring force relaxation and any deformation.

As discussed in section 2.1, all the in-core instrument and source tubes should be inspected when the channel box inspection indicates that there is significant corner wear in the channels. Furthermore, an in-core IRM or SRM tube must be replaced when its wear exceeds 0.01 inches.

7.0 Environmental Considerations

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

8.0 Conclusions

We have reviewed the proposed reactor modification and found that:

(1) the outreactor flow test sufficiently demonstrated that the modification will reduce significantly in-core tube vibration and hence channel box damage;

(2) the effects of the holes on the mechanical strength of the fuel assembly lower tie plate are insignificant;

(3) the fuel rods and cladding of modified fuel will not be damaged by the drilling operation;

(4) measures to ensure that all drilling scraps and burrs are removed from the modified fuel are acceptable; and

(5) the underwater drilling procedures satisfactorily protect the workers from radiation exposures.

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed modification to the fuel assembly, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

The operation of Units 1 and 2 with the modified fuel assemblies will be the subject of later license amendments.

Date: MAR 3 1976

References

1. "Safety Evaluation Report for Partial Power Operation in BWRs with Channel Wear" from V. Stello to K. Goller, July 18, 1975.

2. Letter from R. Engle, GE, to V. Stello, NRC, July 11, 1975.

3. "Safety Evaluation Report for Duane Arnold Operation with Plugged Bypass Flow Holes", from V. Stello to K. Goller, June 30, 1975.

4. "Peach Bottom Atomic Power Station Units 2 and 3: Safety Analysis Report for Plant Modifications to Eliminate Significant In-core Vibration", NEDC-20994, GE, September 1975. (Proprietary)

5. "A Study of the Effect of Position and Voids on BWR In-Core Detector Readings" by Hsiang-Shou Cheng, BNL-20547, Sept. 1975.

6. "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration", NEDE-21156, January 1976. (Proprietary)

7. Georgia Power Company, Letter to Mr. A. Giambusso, Director, Office of NRR, from I. S. Mitchell III, July 9, 1975.

8. Hatch Unit 1 FSAR, Docket 50-321.

9. Wilson, Grenda and Patterson, "The Velocity of Rising Steam in a Bubbling Two-Phase Mixture, ANS Transactions, 5(1), p. 151-152 (1962).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-33 and Amendment No. 17 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority (the licensee) for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments authorize modification to Browns Ferry Nuclear Plant, Units 1 and 2 by approving the drilling of the fuel assembly lower tie plates of Types 2 and 3 fuel assemblies to provide bypass flow. This bypass flow was originally provided for by holes in the lower core support plate. By Amendments 17 and 14 to Licenses DPR-33 and DPR-52 for Units 1 and 2, respectively, authorization was issued to plug the holes in the lower core support plate to eliminate significant in-core instrument tube vibrations. These amendments do not authorize operation of Units 1 and 2 with the plugged core support plate and drilled fuel assemblies. Operation with these modifications will not be authorized until a later safety evaluation is completed that addresses the effects on operation.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act),

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and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 5, 1975, as supplemented November 28, 1975 and February 5, 1976, (2) Amendment No. 20 to License No. DPR-33 and Amendment No. 17 to License No. DPR-52, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this day of

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Operating Reactors

OFFICE ➤						
SURNAME ➤						
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