ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION REVISION

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2, AND 3

(TVA BFN TS 253)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.E. Control Room Emergency Ventilation	4.7.E Control Room Emergency Ventilation
* 1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.	1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to to be less than 6 inches of water at system design flow rate (± 10%).
2. a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.	2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.
	d. Each circuit shall be operated at least 10 hours every month.

* LCO not applicable until just prior to withdrawing the first control rod for the purpose of making the reactor critical from the unit 2 cycle 5 outage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.E. Control Room Emergency Ventilation	4.7.E. Control Room Emergency Ventilation
* 3. From and after the date that one of the control room emergency pressurization systems is made or found to be INOPERABLE for any reason, reactor operation or refueling operations is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.	3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.
* 4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in Cold Shutdown within 24 hours for reactor operations and refueling operations shall be terminated within 2 hours.	4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the following dampers operate as indicated: Close: FCO-150 B, D, E, and F Open: FCO-151 FCO-152

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b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.
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b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.
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3.7.E. Control Room Emergency Ventilation	4.7.E. Control Room Emergency Ventilation
* 3. From and after the date that one of the control room emergency pressurization systems is made or found to be INOPERABLE for any reason, reactor operation or refueling operations is permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.	3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.
* 4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in Cold Shutdown within 24 hours for reactor operations and refueling operations shall be terminated within 2 hours.	4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the following dampers operate as indicated: Close: FCO-150 B, D, E, and F Open: FCO-151 FCO-152

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ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

Description of Change

The Browns Ferry Nuclear (BFN) Plant Technical Specifications require specific safety-related systems to be operable during the handling of spent fuel, operations over the spent fuel pool, and loading fuel in the reactor vessel. BFN is proposing the attached temporary changes to units 1, 2, and 3 technical specifications for the Control Room Emergency Ventilation System (CREVS). This request involves denoting limiting conditions for operations (LCO) 3.7.E.1, 3.7.E.3, and 3.7.E.4 by an asterisk and defining them as not being applicable until just before the withdrawal of the first control rod for the purpose of making the reactor critical from the unit 2, cycle 5 outage.

Reason for Change

The proposed temporary changes to the technical specifications, as shown in attachment 1, are to provide a relaxation of the system operability requirements for the CREVS to support the BFN fuel load activities and the additional activities needed to support unit 2 restart, just before the withdrawal of the first control rod for the purpose of making the reactor critical from the unit 2, cycle 5 outage.

Technical specification, LCO 3.7.E.1, presently requires both CREVS to be operable at all times when any reactor vessel contains irradiated fuel, except as specified in LCO 3.7.E.3. When one of the CREVS is found or made to be inoperable, refueling operations are permissible only for the succeeding seven days (LCO 3.7.E.3). If both CREVSs are inoperable, refueling operations must terminate within two hours (LCO 3.7.E.4).

BFN submitted and NRC approved (safety evaluation report dated July 20, 1988) technical specification amendments (151, 147, 122) which allow both CREVSs to be inoperable while no fuel was in any reactor vessel. This was based on the fact that the BFN fuel has decayed for at least three years, therefore, the radiological consequences due to potential fuel handling accident are far below that evaluated by the current BFN Final Safety Analysis Report. Additional activities are required for unit 2 restart which involve loading fuel in the unit 2 reactor vessel (RV) and performing tests (e.g., RV hydrostatic test) with the RV intact. These activities are no different than a typical refueling operation. With CREVS inoperable, current technical specifications, LCOs 3.7.E.1, 3.7.E.3, and 3.7.E.4, would prevent loading fuel into the reactor vessel. The ability to temporarily relax the CREVS operability requirements during these activities would greatly facilitate currently planned unit 2 restart work while not compromising nuclear safety.

Reason for Change (Cont'd)

The proposed technical specification changes are written to allow these reload and testing operations to take place even though work on the CREVS may still be in progress. Since the CREVS is a common system and since its operability is required for the operation of any unit, the technical specification change is written to ensure that LCOs 3.7.E.1, 3.7.E.3, and 3.7.E.4 become applicable just before the withdrawal of the first control rod for the purpose of making unit 2 critical from the current outage.

Justification for Change

The CREVS is designed to protect the control room operators by pressurizing the main control room (MCR) with filtered air during a fuel handling accident condition. The CREVS uses charcoal adsorbers to assure the removal of radioactive iodine from the air and high efficiency particulate absolute (HEPA) filters for removing particulate matter. These filters and adsorbers will keep the resulting doses, in the event of a design basis fuel handling accident, less than the allowable levels stated in criterion 19 of the General Design Criteria for Nuclear Power Plants Appendix A to 10 CFR 50.

TVA is proposing to relax the operability requirements (3.7.E.1, 3.7.E.3, and 3.7.E.4) of the CREVS until just before the withdrawal of the first control rod for the purpose of making the reactor critical. At this point, the existing technical specifications LCOs 3.7.E.1, 3.7.E.3, and 3.7.E.4 will become applicable. This temporary change will enable work to be performed on the CREVS and the associated control room HVAC ducting, as necessary. This consists of a one time change to the technical specifications.

The filtration function that the CREVS provides is not presently needed in the event of a fuel handling accident. 10 CFR 50 Appendix A (GDC 19) requires that in the event of an accident, the radiation dosage to the occupants in the MCR not exceed 5 REM whole body or its equivalent to any part of the body for the duration of the accident. This same radiation dose limit is endorsed in section 6.2.4 of NUREG 0800. TVA has evaluated the potential consequences to the control room operators in the event of a fuel handling accident. Currently, all three units are defueled with the irradiated fuel stored in the spent fuel pool. The irradiated fuel has decayed for approximately three years and the only remaining volatile fission product of any significance is Kr-85. Kr-85 is an inert gas that is not filtered by the CREVS. Essentially, no iodine is present in the decayed fuel. Because of the "scrubbing" effect of the fuel pool water and since Kr-85 is the only radioisotope of any significance, virtually no particulates would enter the CREVS intake ductwork. Since essentially no iodine is present in the fuel, the inhalation dose is negligible, and therefore, assuming the failure of two assemblies (i.e., 124 fuel pins), the MCR doses would be .002 REM whole body gamma, 0.200 REM beta, and 0.0 REM inhalation. These calculated doses are far below the dose level acceptable in the event of an accident. In order to reach the dose limit of 10 CFR 50 Appendix A, approximately 300 of the assemblies currently stored in the BFN fuel pool would have to fail.

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Justification for Change (Cont'd)

Other events that might occur during fuel load were reviewed. The only other event that has a potential to cause fuel damage, other than the fuel handling accident, is a pipe break inside the primary containment after the fuel has been loaded in the vessel. This would result in a loss of reactor water inventory. The BFN Technical Specifications require the Core and Containment Cooling System (CCCS) to be operable when there is irradiated fuel in the reactor vessel (3.5.A). Therefore, if a pipe break occurred, the CCCS would provide an adequate supply of water to mitigate any fuel cladding damage which would result in a release of fission products. Again, because of the current fission product inventory of the fuel, the only significant isotope is Kr-85. Since CREVSs function is to filter any iodine, it would not be needed to perform any mitigation function.

The operation of the CREVS is not needed to mitigate any of the applicable design basis events which could occur during the time between loading fuel in the unit 2 reactor vessel and just before the withdrawal of the first control rod for the purpose of making the reactor critical from the current outage. For this reason, TVA is requesting the temporary relaxation of the CREVS technical specifications as specified in attachment 1. This relaxation will allow unit restart work to proceed and not compromise the health and safety of the public.

ENCLOSURE 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

Description of Proposed Technical Specification Amendment

The proposed amendment to the Browns Ferry Nuclear Plant Units 1, 2, and 3 Technical Specifications requests temporary changes to the operability requirements for the Control Room Emergency Ventilation System (CREVS). This will allow system modifications and maintenance needed for restart to proceed in parallel with those activities just before the withdrawal of the first control rod making the reactor critical from the unit 2, cycle 5 outage.

Basis for Proposed No Significant Hazards Consideration Determination

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed temporary changes to the technical specifications involve relaxations to system operability requirements for the CREVS during those activities leading to and just before withdrawal of the first control rod for the purpose of making the reactor critical from the unit 2, cycle 5 outage. The fuel that will be moved from the spent fuel pool to the reactor vessel has decayed for approximately three years, thus reducing the need for this system to be operable by the technical specifications for postaccident iodine removal.

   The fuel handling accident evaluated in the Final Safety Analysis Report (FSAR), Section 14.6.4, represents the most severe event in terms of radioactive release and dose consequences that are applicable. The movement of the fuel from the fuel pool to the reactor vessel is a typical refueling operation in which the current FSAR analysis is still valid. The current conditions of the fuel are well within the bounds of the FSAR analysis. The FSAR calculations used freshly irradiated fuel (unloaded from the core 24 hours after reactor shutdown) which contains large amounts of fission products, specifically iodine. The irradiated fuel presently being handled has decayed approximately three years and the only remaining volatile fission product of any significance is Kr-85, which is an inert gas. Because of this decay time, there is essentially no iodine present and therefore no need for the operability of this system with iodine removal capability.

Basis for Proposed No Significant Hazards Consideration Determination (Cont'd)

The proposed temporary changes to the technical specifications do not affect the precursors for any accident analysis and therefore do not involve a significant increase in the probability of an accident previously evaluated. The present required availability of systems in the technical specifications is based on FSAR accident analysis assumptions and limitations. The present condition of the fuel in the spent fuel pool is such that over 300 assemblies would have to fail before the FSAR limiting assumptions for releases and dose consequences could be reached, thus allowing a reduction in the number of systems required to mitigate such a limiting event. The requested relaxation in system operability for the CREVS has been evaluated and a determination reached that the present FSAR assumptions and limitations will be maintained. Therefore, the proposed temporary changes do not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from an accident previously evaluated. The proposed temporary changes will relax present system operability requirements, however, no new modes of plant operations are introduced which could contribute to the possibility of a new or different kind of accident. The fuel handling accident is the most severe event that could occur during fuel load or any other activity being conducted just before withdrawal of the first control rod for the purpose of making the reactor critical from the unit 2, cycle 5 outage.

3. The proposed amendment does not involve a significant reduction in a margin of safety. The proposed temporary technical specification changes will reduce the operability requirements of the CREVS during fuel load and those activities leading to the withdrawal of the first control rod for the purpose of making the reactor critical from the current outage. However, the irradiated fuel has decayed for approximately three years and the only remaining volatile fission product of any significance is Kr-85. Essentially, no iodine is present in the decayed fuel. Because of the "scrubbing" effect of the fuel pool water and since Kr-85 is the only radioisotope of any significance, virtually no radioactive particulates would be present in the CREVS intake ductwork. There is essentially no iodine currently present in the CREVS intake ductwork. Since essentially no iodine is currently present in the fuel, the filtration function that CREVS provides would not be needed until after reactor critically in which the production of iodine would begin. Thus, the relaxation in the system operability requirements for CREVS until just before the withdrawal of the first control rod for the purpose of making the reactor critical from the current outage allows restart work to be completed and does not reduce the margin of safety.

Basis for Proposed No Significant Hazards Consideration Determination (Cont'd)

The proposed temporary changes will ensure that the appropriate safety-related systems needed to mitigate a fuel handling accident are operable and will be able to perform their intended safety function if called upon. Therefore, the proposed changes do not represent a significant reduction in a margin of safety.

Determination of Basis for Proposed No Significant Hazards

Since the application for amendment involves a proposed change that is encompassed by the criteria for which no significant hazards consideration exists, TVA has made a proposed determination that the application involves no significant hazards consideration.