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Supplement No. 2

Safety Evaluation Report
related to the operation of Sequoyah Nuclear Plant, Units 1 and 2

Docket Nos. 50-327 and 50-328

Tennessee Valley Authority
Supplement No. 2

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In March 1979, the Nuclear Regulatory Commission (NRC) issued its Safety Evaluation Report regarding the application by the Tennessee Valley Authority (TVA or licensee) for licenses to operate the Sequoyah Nuclear Plant Units 1 and 2. The Safety Evaluation Report was supplemented by Supplement No. 1 which documented the resolution of several outstanding issues in further support of the licensing activities. Further review of the operating license application resulted from the number of reviews conducted on the accident at the Three Mile Island Unit 2 reactor plant. This resulted in a short-term licensing pause during which additional requirements were established to improve the overall safety of reactor plants. These actions have been applied to the Sequoyah Units 1 and 2 plants.

On February 29, 1980, a license was issued to allow Unit 1 operation at power levels not to exceed 5% of rated power. The license permitted fuel loading and zero power testing. The license was subsequently amended: Amendment No. 4, dated July 10, 1980, permitted the licensee to perform the low-power test program identified in Section 8.6 of the license. As of July 10, 1980, the licensee has been restricted to operation not to exceed 5% percent of rated core thermal power.

The purpose of this supplement is to further update our Safety Evaluation Report by providing (1) our evaluation of additional information submitted by the licensee since the issuance of Supplement No. 1 to the Safety Evaluation Report, (2) our evaluation and status of the Non-TMI-2 outstanding issues identified in Part I of SER Supplement No. 1, (3) our evaluation of TMI-2 requirements which must be completed prior to the issuance of a full-power operating license, (4) our evaluation of dated requirements which the licensee must implement by the dates identified in NUREG-0694, "TMI-Related Requirements for New Operating Licenses," and (5) our evaluation of additional information for those sections of the Safety Evaluation Report where further discussion or changes are in order.

Our review of TMI-2 requirements is based on Commission guidance provided in S. Chilk memorandum of June 5, 1980 (SECY-80-23) for current operating license applications; the requirements are derived from NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating Licenses." The Sequoyah Nuclear Plant Units 1 and 2 were measured against the NRC regulations as augmented by these requirements.

The ACRS has considered the Sequoyah Units 1 and 2 reactor plants and reports its finding in a letter to the Chairman, dated July 15, 1980. This is discussed further in Section 18.0 of this report.

Each of the following sections of this supplement is numbered the same as the corresponding section of the Safety Evaluation Report and Supplement No. 1, except Section 22.0 which addresses TMI-2 requirements and Section 23.0 which presents our conclusions.

In this supplement where the staff concludes that a licensee action is "acceptable," we mean that the action complies with the Commission regulations as stated in Title 10, Chapter 1 of the Code of Federal Regulations and with criteria provided therein and, in several instances, with additional Commission guidance as provided in Regulatory Guidelines and NUREG documents.

Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and Supplement No. 1 thereto, except where specifically noted. Appendix A is a continuation of the chronology of any principal actions related to the processing of the application. Appendix D contains the July 15, 1980, ACRS letter to the Commission on Sequoyah, and Appendix E contains the Emergency Preparedness Evaluation Report.

Except for the issue of hydrogen control, as discussed in Section 22.2 Item B.7, we conclude that the Sequoyah facility may be operated safely at full power in accordance with the facility Technical Specifications without undue risk to the health and safety of the general public. An additional supplement to the SER will be issued dealing with the hydrogen control matter.

2.0 SITE CHARACTERISTICS

2.2 Nearby Industrial, Transportation, and Military Facilities

In our SER of March 1979, we stated that the new essential raw cooling water (ERCW) intake will be protected against barge collisions by a dike which will be constructed on the upstream side of the intake structure and by the skimmer wall on the downstream side. Since the applicant indicated that the ERCW intake was protected by barriers up to a river level of 705 feet above mean sea level (MSL) (normal river elevation is 683 feet above MSL), the applicant analyzed the probability of a flood causing a river level greater than 705 feet above MSL coincident with a drifting barge striking the intake structure. The applicant concluded, with our concurrence, that the probability of this event was sufficiently low (about 4×10^{-8} per year) that it need not be considered as part of the design for the plant.

Recently, a question was raised by the ACRS in their review of the Sequoyah operating license (letter dated July 17, 1980 from R. F. Fraley to W. J. Dircks) concerning the vulnerability of the ERCW intake structure to collision of a barge at full speed from any credible direction, including a tow proceeding in the upstream direction, and the probability of such an event. In addition, information was requested on the ability of the ERCW intake to withstand the effects of barges carrying flammable cargoes including liquid natural gas (LNG).

In a letter dated August 5, 1980, TVA has stated that the existing (currently used) ERCW pumping station, which is now relied upon for operation of Unit 1, will act as a backup to the new ERCW pumping station after the new pumping station is put into service. In view of the separation distance between these two intakes (about 2000 feet) and the location of the existing ERCW pumping station, together with the fact that the existing ERCW intake can supply the cooling water needs for Unit 1, we conclude that the full-power operation of Unit 1 can proceed prior to the resolution of this issue. However, since the new ERCW intake is required for operation of both units 1 and 2, we will not permit operation of Unit 2 until this matter is resolved.

TVA stated that it will provide the analyses regarding the above information. We will review these and provide our evaluation regarding the vulnerability of the new intake structure and additional protection required, if any, in a future supplement prior to operation of Unit 2. Our review has been performed consistent with GDC 4 to 10 CFR 50, Appendix A and the Standard Review Plan Section 2.2.3.

2.3 Meteorology

2.3.3 Onsite Meteorological Measurements Programs

As indicated in the Sequoyah SER, the operational meteorological data will be transmitted to a TVA meteorological forecast center in Muscle Shoals, Alabama, as part of the radiological emergency plan. Since that report was written, NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," has been issued. A description of and completion schedule for an upgraded meteorological program in substantial compliance with NUREG-0654, Appendix 2 is required by NUREG-0694, "TMI-Related Requirements for New Operating Licenses," before issuance of a full-power license.

The essential elements of the NUREG-0654, Appendix 2 criteria are:

1. A primary meteorological measurement program with redundant power sources.

2. A backup meteorological measurements program with redundant power sources.

3. A system for making real-time predictions of the atmospheric effluent transport and diffusion, including Class A and Class B models as described in Appendix 2.

4. A capability for remote interrogation on demand of the atmospheric measurements and prediction systems by the licensee, emergency response organizations, and the NRC staff with primary and backup communications systems.

Regulatory requirements for onsite meteorological programs are addressed in Regulatory Guide 1.23. Appendix E to 10 CFR 50 requires plans for coping with radiological emergencies. Such plans make it necessary for a licensee to establish and maintain a meteorological program capable of rapidly assessing critical meteorological parameters. NUREG 0654, Appendix 2 provides additional guidance to licensees in this matter. In addition, such programs are necessary to determine ongoing compliance with 10 CFR 20.105 and Appendix B to 10 CFR 20.

TVA has provided in letters dated August 1 and 5, 1980, a description and completion schedule for these essential elements as required by NUREG-0694. The details of the meteorological program will be reviewed on a schedule consistent with TVA's implementation schedule for its Emergency Response Plan. (See section 22.2 III.A.1.2 and Appendix E of this supplement.) These schedules are in accordance with emergency planning schedules as delineated in the amendments to the regulations approved by the Commission July 23, 1980 (10 CFR 50.54(s)).

2.3.4 Short-Term (Accident) Diffusion Estimates

In order to determine compliance with 10 CFR 20.105, Appendix B to 10 CFR 20 and to have acceptable radiological emergency plans as called for in Appendix E to 10 CFR 50, acceptable means must be available to determine site-specific relative concentrations (X/Q). The present staff position for determining the X/Q values used to describe a postulated accident is, in part, the atmospheric dispersion model specified in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." This model varies slightly from the modified model described in Section 2.3.4 of the Sequoyah SER. Since the model described in Regulatory Guide 1.4 (Rev. 2) (and previously presented in Section 2.3.4) is more conservative than that of Regulatory Guide 1.145, we conclude that acceptable means are available to determine site-specific relative concentration (X/Q) values at Sequoyah.

2.3.6 Conclusions

The operational meteorological program is in compliance with Regulatory Guide 1.23. TVA has made commitments with respect to an updated meteorological program emergency preparedness which are in compliance with NUREG-0694. The technical details of the meteorological program as it relates to emergency planning will be reviewed for acceptability as the program is developed.

2.5 Geology and Seismology

Piping and Components

In the December 11, 1979 ACRS report to the Commission (from ACRS Chairman M. Carbon to NRC Chairman J. Ahearne) on interim low-power operation of Sequoyah Nuclear Power Plant Unit 1, the ACRS recommended that a seismic margin program be continued and expanded to the extent necessary to determine the seismic design margin of all structures and equipment necessary to accomplish safe shutdown. During the June 1980 ACRS meeting, the expanded seismic design margin program for Sequoyah was discussed in some detail. The TVA program, as described by letter dated May 27, 1980, is intended to ensure that all structures, piping, components, and equipment necessary for decay heat removal have adequate margins and will operate with a high degree of reliability.

Based on the acceptable results of the portion of the seismic design margin review already completed as discussed in the Safety Evaluation Report Supplement No. 1, the NRC staff concludes that completion of the expanded design margin program within about 18 months is acceptable and that operating at full power need not be delayed pending completion of the reanalysis program. TVA agreed by letter dated August 11, 1980, to complete the expanded program in about 18 months. Our review has been performed consistent with the requirements of GDC 2 to 10 CFR 50 and Appendix A to the 10 CFR 100 as noted above.

2.6 Foundations

2.6.3 Foundation Evaluations

In order to determine compliance with the requirements of Section V(d)(5) of Appendix A to 10 CFR 100 and to General Design Criterion 2 of Appendix A to 10 CFR 50, it is necessary to establish the stability of subsurface materials and foundations.

In Section 2.6 of the February 1980 Safety Evaluation Report Supplement No. 1, the staff identified the need to review settlement records for certain Category I safety-related structures. We have completed our review of the settlement records recently provided informally by the applicant and conclude that the observed settlements have been minimal and have stabilized, except for the section along the ERCW conduit for Unit 2. Except for the ERCW conduit for Unit 2, the staff's concern for settlement on all safety-related structures is resolved.

The settlements recorded over a 125-foot length of ERCW conduit ranged from 0.50 inch to 1.0 inch. This settlement is considered significant enough to require further study to determine if allowable conduit stresses have been exceeded or will be exceeded at some time in the future. The applicant's position is that the involved settlement monuments were disturbed during recent construction activities in this area and that no significant settlement has occurred. Until it can be demonstrated that no significant settlement is occurring, the existing intake pumping station will be operational to provide essential raw cooling water (ERCW) for Unit 1 operation and will then act as a backup to the new ERCW pumping station after the new pumping station is put into service. In addition, Sequoyah has an auxiliary ERCW system capable of providing cooling water for Unit 1 which will also back up the ERCW system. Therefore, resolution of this issue has no bearing on the safe operation of Unit 1 during the interim period. The staff will pursue this concern for Unit 2 to full resolution which could ultimately result in the requirement for a Technical Specification to monitor future settlements.

TVA has agreed in a letter dated July 28, 1980, to continue monitoring the settlement markers in question to resolve this matter. On this basis, it is the judgment of the staff that if settlement is actually occurring to any significant degree, it will be detected in a timely manner. We expect the settlement issue to be resolved by January 1, 1981.

3.0 DESIGN CRITIERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.5 Missile Protection

3.5.1 Missile Selection and Description

TURBINE MISSILES

During November 1979, the NRC became aware of a problem of stress corrosion cracking in Westinghouse turbines. Meetings were held with Westinghouse to ascertain the probable extent and severity of the problem. Westinghouse was recommending early inspection of turbines that had long operating times, and particularly those machines with discs of marginal material properties or a history of secondary water or steam chemistry problems. Since then, inspections have been performed on about 18 more Westinghouse nuclear service turbines, with indications of cracking, some severe, found in most of them. Investigations are continuing.

The NRC staff considers that General Design Criteria 4 to Appendix A of 10 CFR 50 requires that this missile potential be evaluated for Sequoyah.

The main turbine for this facility is installed in a peninsular orientation (the axis of the turbine rotor is radial rather than tangential to the containment structure) and there is an intermediate structure between the containment building and the turbine building, thereby meeting Regulatory Guide 1.115 and affording considerable protection to safety-related equipment. Nonetheless, the NRC staff considers it prudent not to rely solely upon these facts to assure no damage to safety-related structures, systems, and components. As a matter of defense in depth, we also require assurance that the low pressure turbine rotor discs will not develop cracks which could result in the creation of missiles.

TVA has provided in a letter dated August 1, 1980, the material properties of the low pressure turbine discs, as well as the calculations of critical crack sizes. The method used by the TVA to predict crack growth rates is based on evaluating all the cracks found to date in Westinghouse turbines, past history of similar turbine disc cracking, and results of laboratory tests. This prediction method takes into account two main parameters; the yield strength (and stress) of the disc and the temperature of the disc at the bore area where the cracks of concern are occurring. The higher the yield strength of the material and the higher the temperature, the faster the crack growth rate will be.

We have evaluated the data submitted by TVA and, in addition, performed our own calculations for crack growth and critical crack size. We conclude that Sequoyah 1 may be safely operated at full power. However, we will require that the LP turbine discs be reinspected during the second refueling outage for continued assurance that turbine integrity would not be jeopardized. In accordance with the GDC 4 to

Appendix A of 10 CFR 50, the staff has evaluated missile potential for Sequoyah and finds that the pertinent sections of this requirement have been met.

3.9 Mechanical Systems and Components

3.9.2 Bolted Connections in Component Supports

As reported in SER Supplement No. 1, operating experience at other facilities indicated a potential generic problem with "as installed" bolted connections that could adversely affect safety-related items. This experience is described in IE Bulletin 79-02. Bolted connections that can affect safety-related items must meet the applicable requirements of General Design Criteria 1, 2, and 4 of Appendix A to 10 CFR 50.

Based on our review and data provided by TVA on July 5, 1979, January 2 and 16, 1980, and February 1, 1980, we conclude that the design of Seismic Category I pipe supports using concrete expansion bolts is based on conservative criteria and assumptions. The factor of safety actually obtained in the verification program exceeded those recommended in IE Bulletin 79-02. By memorandum dated February 26, 1980, the NRC Office of Inspection and Enforcement concluded that these bolts have been installed with construction practices which meet the requirements of IE Bulletin 79-02.

Based on the above evaluation, we have completed our review of the issue of design and installation of concrete expansion anchor bolts at the Sequoyah Nuclear Power Plant and find them acceptable for the issuance of a normal full-power operating license. Specifically, based on the results of the above reverification program, we affirm that they meet the requirements of GDC 1, 2, and 4 of 10 CFR 50 Appendix A.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.6 Inservice Inspection Program

5.2.6.1 Inservice Inspection of Pressure Isolation Valves

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA. Periodic leak testing of pressure ioslation valves shall be performed pursuant to Technical Specifications after all disturbances to the valve are complete. The licensee has categorized the Sequoyah pressure isolation valves, except for the boron injection system, as Category A or AC. These categorizations meet our requirements and we find them acceptable. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code except as discussed below. The allowable leakage rate shall either not exceed 1.0 gallon per minute (GPM) for each valve or the leak rate stated in the Technical Specifications.

TVA has not categorized the check valves in the boron injection system as Category AC but has agreed to leak testing these valves by the same method and criteria as those valves categorized as AC. We find this acceptable and will add these valves to the table for RCS pressure isolation valves in the Technical Specifications.

The staff's present position on leak rate criteria is that a leakage rate at or below 1 GPM will ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function, and give an indication of valve degradation over a finite period of time. Significant increases over this leak rate would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM may be considered acceptable if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case-by-case basis. The Technical Specifications currently specify a leak rate limit of 1.0 GPM for all pressure isolation valves. This limit will ensure that the integrity of the valve is maintained, that degradation of the valve can be quantitatively measured, and that the redundant pressure isolation function is sustained. Limiting Conditions for Operation (LCO) will be added to the Technical Specifications which require corrective action, i.e., shutdown or system isolation when the final approved leakage limits are exceeded. Also surveillance requirements, which state the acceptable leak rate testing frequency, will be provided in the Technical Specifications.

We conclude, subject to resolution of the above, that TVA's commitments to periodic leak testing of pressure isolation valves between the reactor coolant system and low pressure systems will provide reasonable assurance that the design pressure of the low pressure systems will not be exceeded and thus reduce the probability of an occurrence of an intersystem LOCA. The staff believes, with resolution of the above, that the pertinent sections of these requirements of GDC 55 will be met.

5.2.6.2 Pressurizer Relief Line Weld Repair

In October 1979, we were advised by the applicant that during hot functional testing at Sequoyah Nuclear Plant Unit 1 a hanger failed to slide and caused the 6-inch schedule 160 (6.625OD x 0.718 wall) pressurizer relief piping to undergo plastic deformation. To straighten the pipe, weld material was deposited in two adjacent grooves that extended from the outside pipe surface approximately two-thirds of the way into the pipe wall and 270° around the pipe circumference.

Following completion of the weld repair, TVA requested permission to use the Summer 1978 Addenda to Section XI of the ASME Code to eliminate the necessity for performing a hydrostatic test. The staff performed an initial evaluation of the weld repair to ensure that the repair did not degrade system integrity. Based on this initial evaluation, the staff agreed that a hydrostatic test was not necessary and required that the repair welds be included in the Sequoyah 1 inservice inspection program.

Following this initial evaluation, additional evaluations were performed in response to concerns raised by a member of the staff concerning the need for additional information. Staff evaluations took place from January to April 1980 in various meetings with TVA, including visits by NRR staff to the TVA engineering laboratories and by IE staff to the Sequoyah site.

Following these later evaluations, the NRR and IE staff in May and June 1980 finally concluded that the weld repair was acceptable.

On June 16, 1980, a member of the NRC staff submitted a differing professional opinion concerning the integrity of the repair weld in the pressurizer relief line. The differing professional opinion expressed the concern that the evidence available to support staff acceptance of the weld repair was inconclusive and that the draw-bead technique used in the repair could have caused sufficient sensitization of the piping material to make it susceptible to intergranular stress corrosion cracking (IGSCC).

To resolve the differing professional opinion, additional comments concerning the integrity of the weld repair were obtained from staff members in the Division of Engineering, NRR, including the staff member having the differing opinion. Additionally, a peer review group comprised of members from the Office of Nuclear Regulatory Research was formed to provide an independent assessment of previous staff evaluations and the differing opinion. Further, presentations of both viewpoints were made to the ACR

On July 15, 1980, the ACRS, in a letter to the Chairman of the Commission, stated that the Committee did not consider the weld repair to be particularly likely to present a serious hazard; but believed that the evidence on this point could be improved. The Committee recommended that through-wall metallographic examination be made of a mockup closely simulating the repair welding to improve the evidence. The Research peer review group completed a review, including review of additional work along the lines recommended by the ACRS. The peer review group concluded that the pressurizer drawbead weld repair did sensitize the piping material, making it susceptible to IGSCC in service, but that it did not penetrate the coolant pressure boundary, and, therefore, hydrostatic testing was not required. The peer review group recommended that an augmented inspection program should be implemented for the repair weld in accordance with that required for nonconforming, service-sensitive lines in BWRs as defined in NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

As resolution of this issue, NRR has accepted the recommendation of the Research peer review group to institute an augmented inservice inspection program consistent with that required for nonconforming, service-sensitive lines in BWRs. Although the pressurizer line is not service-sensitive and NUREG-0313 addresses BWR lines rather than PWR lines, we believe that implementation of the recommended augmented inspection is appropriate and will provide a high level of assurance of the continued integrity of this line during operation. The specific requirements for the augmented inservice inspections will be included in the Sequoyah Unit 1 Technical Specifications.

In addition to the metallurgical evaluation, an analysis was performed of a double-ended guillotine rupture of the pressurizer relief line for a UHI plant. This calculation was done by Sandia National Laboratories using the UHI version of RELAP4 to support the conclusion that this break will not result in unacceptable consequences. The analysis followed the requirements of Appendix K to 20 CFR 50. These include the worst single active failure and the use of 1.2 times the 1971 ANS-5 standard decay heat. The results showed that no core uncovery was predicted to occur. Consequently, no heatup of the cladding occurred, and the limits of 10 CFR 50.46 were not exceeded.

The NRC staff member having the differing professional opinion has reviewed this safety evaluation and concurs in its resolution.

This matter is directly a consideration for Regulation 10 CFR 50.55 a(g) and General Design Criteria 1, 14, 30, 31, and 32. The staff believes that the pertinent sections of these requirements have been met with respect to this weld repair.

5.3 Component and Subsystem Design

5.3.1 Steam Generator Tube Integrity

In Section 5.3.1 of SER Supplement No. 1, we noted that steam generators of the design used in the Sequoyah plants have experienced denting and cracking of the steam generators tubes. We required TVA to implement a water chemistry control program, but noted that although an effective secondary water chemistry control program can reduce the rate of tube degradation, there is no assurance that a 40-year steam generator lifetime can be obtained.

Since that time the staff has identified additional measures which can be taken to provide further assurance that operation of the steam generators will not constitute an undue risk to the health and safety of the public. These additional measures are discussed below.

Inspection Ports

For some forms of steam generator degradation which have occurred in units similar to the Sequoyah design, eddy current testing and tube gauging alone are not sufficient to assess and monitor tube and support plate conditions. In order to perform adequate assessment and monitoring of these areas, we require that inspection ports be installed in each steam generator. These ports should be installed just above the upper support plate and in line with the tube lane. At the upper support plate level, at least one inspection port is required which shall be large enough for visual observation of the tube lane.

Under the as low as reasonably achievable (ALARA) concept radiation exposure, NRC has been requesting that all possible steam generator modifications be made before the start of operations in order to minimize personnel exposure. Although installation prior to initial operation is perferable, we have determined that the potential installation exposure following the first cycle of operation is not significant enough to justify the delay of the initial startup of the plant to permit the installation of inspection ports. However, since secondary side contamination will increase as the operating time increases, we require that these ports be installed prior to startup after the first refueling.

In a letter dated July 8, 1980, TVA has stated its intention to design and test a camera device for remote inspection of tube supports as an alternative to additional ports. The camera inspection device would be inserted through the existing handholes located between the tubesheet and lower support plate in line with the tube lane. Should the NRR conclude that the device is unsuccessful or inadequate, TVA will be required to install the inspection ports before startup after the first refueling. The results of the inspection device should be available for NRR review by March 1, 1981. We consider this approach acceptable.

Row 1 Steam Generator Tubes

Operating experience has shown that the Row 1 tubes in the steam generators of Westinghouse design are particularly susceptible to an early onset of cracking because of their small bend radius. We do not currently require licensees to plug Row 1 tubes prior to startup or issuance of a full-power license. Westinghouse has committed (letter from R. M. Anderson to R. H. Vollmer, May 12, 1980) to a program to determine the particular susceptibility of Row 1 tubes to cracking. The program involves removing numerous tubes from the Trojan plant and subjecting them to nondestructive and destructive testing in an attempt to identify the cause of the cracking and thus develop a field inspection method capable of detecting potential leaking tubes. The results of this program are expected to be available in October 1980. We shall review the program results and decide at that time on the necessity to plug the Row 1 tubes. If necessary, we will require that these tubes be plugged prior to startup after the first refueling.

Summary

Although the possibility of tube and tube support plate degradation exists, we have concluded that, with the additional measures mentioned above and discussed further below, operation of the steam generators will not constitute an undue risk to the health and safety of the public for the following reasons:

1. Primary to secondary leakage rate limits and associated surveillance requirements in the Technical Specifications will be established to provide assurance that the occurrence of tube cracking during operation will be detected and appropriate corrective action, such as tube plugging, will be taken such that any individual crack present will not become unstable under normal operating, transient, or accident conditions.

2. Augmented inservice inspection requirements and preventative tube plugging criteria will be established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

Steam generator tube integrity as described above is directly a consideration for General Design Criteria 14, 15, 30, and 31 of Appendix A to 10 CFR 50. Staff believes that the pertinent sections of these requirements have been met with respect to the steam generators tubes.

5.3.2 Condenser Leaks

In Section 5.3.1 of SER Supplement No. 1, we stated that we would require the TVA to repair or plug a condenser leak within 96 hours of confirming the existence of a condenser leak in accordance with Branch Technical Position MTEB 5-3 appended to Standard Review Plan 5.4.2.1.

Subsequently we established the following alternate approach to condenser leak corrective action and discussed it with the TVA:

1. The hotwell pump discharge sample point along with continuous cation conductivity monitoring will be used as the control point for confirming a condenser leak and for initiating corrective action to locate and repair the leak.

2. Impurity-time operating limits for feedwater should be incorporated into the water chemistry program. The limits use feedwater pH and cation conductivity impurity-time limit values the same as used for steam generator blowdown limits.

TVA agreed to incorporate the above provisions into the Sequoyah secondary water chemistry control program and submitted confirmation of these changes by letter dated August 13, 1980.

We find this alternate approach to MTEB BTP 5-3 for condenser leak corrective action acceptable because:

a) it establishes a specific continuously monitored condensate sample point for confirming a condenser leak,

b) the incorporation of feedwater impurity-time operational limits provides earlier indication of impurities entering the steam generator before the entire steam generator secondary side reaches or exceeds its operational limits, and

c) it provides an effective integrated impurity-time limit to the quantity of impurities entering the steam generator.

This constitutes an acceptable basis for satisfying the pertinent sections of the requirements of General Design Criteria 14 of Appendix A to 10 CFR 50 with respect to interactions between condenser in-leakage and its impact on the ability to maintain an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross failure of the reactor coolant pressure boundary that exists across the steam generator tubes.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Containment Air Purification and Cleanup Systems

Auxiliary Building Gas Treatment System

In Section 6.2.3 of our SER, we stated the following: "The containment systems of Sequoyah Nuclear Plant also include the auxiliary building gas treatment system. The auxiliary building gas treatment system is used to maintain portions of the auxiliary building which contain emergency safeguards systems and fuel handling systems at a negative pressure of 0.25 inches of water gauge following a loss-of-coolant accident. Exhaust from the auxiliary building gas treatment system is filtered prior to release to the atmosphere."

The portion of the auxiliary building served by the auxiliary building gas treatment system is known as the auxiliary building secondary containment enclosure (ABSCE). TVA has defined an interim ABSCE to separate Unit 1 operations from Unit 2 construction during the interim period between startup of Unit 1 and the completion of construction of Unit 2. This interim ABSCE is smaller than the final ABSCE, and its boundary is generally inside that defined for the final ABSCE.

Since the issuance of our SER, TVA found by tests that some portions of the interim ABSCE could not be maintained by the auxiliary building gas treatment system at the required negative pressure of 0.25-inch water gauge.

Following notification of the staff, modifications were made and additional tests were run in July 1980, which demonstrated the ability of the entire interim ABSCE to be maintained at the required pressure, as described in Section 15.4.1 of this supplement. We, therefore, affirm that the interim ABSCE is acceptable for full-power operation.

When the final ABSCE is established, TVA will be required to demonstrate that a negative pressure of 0.25-inch water gauge can be maintained in the spent fuel storage area and in the ESF pump rooms by testing in the manner detailed in the Technical Specifications. This will demonstrate, for the final ABSCE configuration, that the ABGTS and final ABSCE are acceptable for two-unit operation of the Sequoyah plant. This review is in conformance with General Design Criteria 43.

6.2.4 Containment Isolation Systems

In the Safety Evaluation Report (SER) for the Sequoyah Nuclear Plant, Units 1 and 2, we concluded that:

"...the containment purge system may be used as frequently as necessary during the normal plant operating modes of startup, power, hot standby and hot shutdown, but in a manner consistent with the above dose consequence analysis; i.e., with only one pair of purge system lines open at a time. In the cold shutdown and refueling modes all purge systems may be used simultaneously. The Technical Specifications will reflect this requirement."

The NRC staff has recently determined that restrictions should be placed upon containment purging and venting during plant operation. Restrictions on purging during operation will decrease the likelihood of a LOCA occurring with the purge system lines open. Such open lines constitute a direct connection between the containment atmosphere and the outside environment, and failure of the redundant purge system isolation valves to close as required during a LOCA, though they may have been properly tested and qualified, would result in offsite doses far in excess of 10 CFR 100 guidelines.

Therefore, we require that for Sequoyah Units 1 and 2, TVA limit use of the containment purge and venting systems to a total of no more than 90 hours per year, per reactor unit, during the normal plant operating modes of startup, power, hot standby, and hot shutdown, with only one pair of purge system (or venting system) lines open at a time. Thus, the 90-hour limit applies to the total time in use of all venting lines and purge lines. In the cold shutdown and refueling modes, all purge and venting systems may be used simultaneously and without time limitation. The Technical Specifications will reflect this requirement. This conforms to the requirements of General Design Criteria 54, 55, 56, and 57 with respect to containment purging and venting.

6.3 Emergency Core Cooling System

PAD 3-3 Performance Code

This evaluation concerns the use of Westinghouse PAD 3-3 computer code in plant safety analyses. We find its use in the Sequoyah analyses acceptable for first cycle operating at full power. The evaluations presented below supersedes our earlier thermal performance analyses portions of Section 4.2 of the Sequoyah SER.

Thermal Performance Analysis

The new Westinghouse fuel thermal performance code PAD 3-3 is described in WCAP-8720, "Improved Analytical Methods Used in Westinghouse Fuel Rod Design Calculations," October 1976. This code contains a revision of an earlier fission gas release model and revised models for helium solubility, fuel swelling, and fuel densification.

The new Westinghouse code was approved with four restrictions as described in our safety evaluation of February 9, 1979 (Letter from J. Stolz, NRC, to T. Anderson, Westinghouse). Three of those restrictions deal with numerical limits and have been complied with. The fourth restriction relates to use of the PAD 3-3 code for the analysis of fission gas release from uranium dioxide (UO_2) for power increasing conditions during normal operation. This restriction applies to the safety analysis of Sequoyah Units 1 and 2. However, Westinghouse has stated that this restriction does not adversely affect the results of the safety analyses performed for Sequoyah. Although we believe that this is essentially correct for the planned operation of Sequoyah Units 1 and 2, Westinghouse has prepared and submitted a detailed evaluation of this fourth restriction in WCAP-8720, Addendum 1.

At this time, we have not completed our review of the Westinghouse evaluation of this restriction. However, our review has progressed to the point where the following conclusion can be made.

1. The Westinghouse evaluation of our restriction on the use of the PAD 3-3 code supports Westinghouse's earlier statement that the restriction does not adversely affect the results of the safety analyses performed for Sequoyah Units 1 and 2.

2. We continue to believe that this result is essentially correct and anticipate some additional information from Westinghouse to confirm this conclusion.

3. Because the restriction pertains to the release of fission gases from the fuel, any change in our conclusions would not have significant impact at low burnup (e.g., first-cycle operation), when the fission gas inventory in the fuel is low.

At this time we can therefore state that for the first-cycle operation at full power, the restriction for PAD 3-3 is not significant and the analyses as presently docketed for Sequoyah are acceptable. We anticipate a timely completion of our review of the Westinghouse evaluation prior to operation at extended burnup.

With respect to the thermal performance of the reactor fuel, this analysis conforms with the requirements set forth in 10 CFR 50 Appendix K and 10 CFR 50 Appendix A, General Design Criteria 10.

6.3.3 Evaluation

Functional Design

In Section 6.3.3 of SER Supplement No. 1, we discussed the design of the emergency core cooling system (ECCS) containment sump screen. We stated that we had not determined whether additional protection against containment debris entrained in the recirculating coolant needed to be provided. We also concluded that the low-power operation program could safely proceed while additional information was gathered and positions were developed. Since then we have visited the Sequoyah plant and have reviewed the overall issue of debris in the ECCS recirculation system. Our evaluation is presented below.

Housekeeping

We have evaluated housekeeping requirements (e.g., maintenance and inspection activities) within containment to preclude debris from non-LOCA sources.

The Sequoyah Nuclear Plant (SNP) quality assurance program establishes written guidelines for assuring that good housekeeping practices are followed during maintenance. The SNP Technical Specifications include surveillance requirements which are implemented pursuant to written procedures. The requirements include inspections to verify that no loose debris which could be transported to the sump remains in the containment, periodic inspections of the containment sump suction inlets to ensure that they are not blocked by debris, and inspection of the sump components (trash racks, screens, etc.) to verify that structural distress or corrosion is not present.

The SNP Technical Specifications (including required surveillance inspections) adequately address control of loose debris in the containment. The NRC's Office of Inspection and Enforcement will monitor TVA's compliance with the Technical Specification requirements.

We find the housekeeping provisions for the SNP to be acceptable.

Small Debris

We have considered materials capable of being transported to the sump which would have a tendency to form particles small enough to pass through the fine screens in the sump.

Virtually all the piping insulation in the containment and particularly in the lower containment regions is of the mirrored metal type. This material is not expected to float or to form small particles as a result of pipe whip or jet impingement. Also, see discussion below on Larger Debris on this type of material.

Foam glass insulation is used to cover the wall of a tunnel which is outside of the crane wall. This wall is between elevations 680 and 693 feet. The foam glass insulation is covered with sheet metal which reduces the likelihood of its being damaged as a result of a pipe break in the tunnel. TVA has verified that since the crane wall is sealed to prevent water leakage below elevation 693 feet, there are no available paths for foam glass insulation to be transported to the sump.

TVA has stated in a letter dated July 21, 1980, that sand or similar material is not used in the containment for purposes such as subcompartment blowout plugs or sand-filled tanks or sandbags for the reactor cavity annulus biological shielding.

With regard to other potential sources of debris (i.e., paint chips or other degraded material) periodic surveillance inspections are provided to detect occurrences of degraded materials.

Based on the above considerations, we conclude that the SNP design acceptably avoids the use of materials in the contaiment which would be likely to produce small-sized debris in significant quantities.

Larger Debris

We have considered the use of materials which would have the potential to block the containment sump screens if transported to the screens as a result of an accident. The present design of the containment sump has been modeled in one-quarter scale and successfully tested under conditions of potential sources up to 50% screen blockage.

Virtually all the piping insulation in the containment and particularly in the lower containment regions is of the mirrored metal type. No other larger debris were identified.

Based on the observations made during our site visit, we believe it unlikely that a significant quantity of mirrored metal insulation debris would be transported to the sump. This conclusion is based primarily on such aspects as the large number of obstructions in the form of piping of varying sizes, pipe hangers, snubbers, pipe support members, structural steel, platforms, cabling, motors, stairways, etc., to the passage of this insulation material to the sump.

ECCS Status

We have reviewed the adequacy of the information available to the control room operator to monitor the low pressure injection (LPI) system status during recirculating cooling. We conclude that sufficient information (e.g., flow rate, pump motor current, pump suction pressure, and pump discharge pressure) is available to the operator to detect LPI performance degradation. When the residual heat removal pumps are operating in a recirculation mode, TVA has stated in a letter dated

July 21, 1980, operating instructions require stationing of an operator in the control room with no other duties than to monitor RHR system performance.

This is supplemented by requiring the maintenance of an emergency administrative log. Also this log is complemented by reference information (e.g., pump curves, decay heat curves) available to be used to determine LPI performance. SNP operators are specifically instructed in the means and procedures for recognition and mitigation of LPI performance degradation. The SNP LOCA emergency operating procedures also include guidance to alert the operator to the symptoms of inadequate core cooling.

Based on procedures and operator training which address the potential for ECCS performance degradation, we find the above measures acceptable to monitor ECCS performance during the recirculation mode at Sequoyah.

Conclusion

Based on the considerations noted above with respect to housekeeping requirements, the avoidance of materials likely to form small size debris, the lack of an apparent mechanism for blockage of more than the previously tested value of 50% of the screen area by large debris, and the ability to monitor and control LPI system status, we conclude that the present design of the SNP provides reasonable assurance that the post-LOCA recirculation of core coolant will not be impaired by debris. Those requirements of 10 CFR 50.46 and Criterion 35 of the General Design Criteria given in 10 CFR 50 Appendix A which are applicable to sump debris are acceptably addressed.

6.3.5 Performance Evaluation

New Cladding Swelling and Rupture Model

The NRC staff has been generically evaluating three materials models that are used in ECCS evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) discussed our evaluation with vendors and other industry representatives (Reference 1), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference 2), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed.

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we have required (see Section 6 of SER Supplement No. 1) that the ECCS analyses be accompanied by supplemental calculations to be performed with the materials models of NUREG-0630. For these supplemental calculations only, we have accepted other compensatory model changes that are not yet approved by the NRC but that are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above.

Supplemental calculations have been provided by TVA in a letter dated July 31, 1980. TVA has also addressed a recently identified nonconservatism of the Westinghouse 1978 ECCS evaluation model. This new concern was discovered by Westinghouse and formally communicated to the NRC staff in November 1979.

Specifically, Westinghouse discovered that the February 1978 ECCS evaluation model was, in part, based on cladding burst tests which were conducted at relatively fast temperature ramp rates; whereas the LOCA analyses of actual plant heatup rates (including those of Sequoyah) were at relatively slow temperature ramp rates.

TVA assessed the impact of this calculational error to be offset by a corresponding peaking factor (F_Q) reduction of 0.012. A reduction of 0.015 in F_Q was also required to demonstrate conformance with the LOCA acceptance criteria of 10 CFR 50.46 if the material models of NUREG-0630 were employed in the analyses.

However, TVA identified a margin in F_Q available through the use of a reduction in pellet-temperature uncertainty. This margin was worth 0.015 in F_Q. Thus the net result is an F_Q reduction of 0.012. The Technical Specifications have been amended to reflect an overall peaking factor of 2.237 compared with the previous value of 2.25.

In order to ensure that the core peaking factor will not exceed 2.237 in normal operation of the power plant, the unrodded plane peaking factor Fxy(Z) has been reduced in the Technical Specification from 1.52 to 1.50. Should Fxy(Z) exceed the values of 1.50 at any unrodded elevation, the specification requires determination that there is margin in the 17-case peaking factor analysis to offset the excess at that elevation or a proportional power reduction be made. We find that this is acceptable to maintain operation within the bounds assumed as input to the LOCA analysis.

We therefore conclude that TVA has satisfied our 10 CFR 50.46 concerns related to the swelling and rupture issue.

6.5 Containment Pressure Boundary Fracture Toughness

The fracture toughness of the ferritic materials that constitute the contaiment pressure boundary of the Sequoyah Unit 1 nuclear plant was reviewed to assess compliance with the 10 CFR 50 Appendix A GDC-51, "Fracture Prevention of Containment Pressure Boundary." The Sequoyah Units 1 and 2 containment systems consist of a free-standing steel containment vessel with penetrations, such as the equipment hatch, personnel airlocks, and pipe penetrations. The fracture toughness requirements of GDC-51 apply to those ferritic steel parts of the containment pressure boundary which are not supported by concrete and are thus load-bearing. These materials have been applied in the Sequoyah containment pressure boundary in the design and construction of the containment vessel, equipment hatch, personnel airlocks, and penetrations.

TVA has stated in the Sequoyah FSAR that the ASME Code Section III, Subsection B, and Material Related Code cases 1413 and 1431 and the Winter 1968 Addenda were applied in the construction of the containment pressure boundary. We therefore conclude that compliance with the requirements of the ASME Code for the ferritic steel parts of the containment pressure boundary satisfies the 10 CFR 50 Appendix A GDC-51 requirements.

7.0 INSTRUMENTATION AND CONTROL

7.2 Reactor Trip System

7.2.2 Process Analog System

Environmental Qualifications for Safety-Related Electrical Equipment

In December 1979, the staff issued guidance for the environmental qualification of safety-related electrical equipment (NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"). By letter dated February 21, 1980, the staff requested TVA to review the environmental qualification documentation for each item of safety-related electrical equipment which could be exposed to a harsh environment so as to identify the degree to which the associated environmental qualification program complies with the staff's position as described in this NUREG. Further, where there are deviations, we requested the applicant to provide the basis for concluding that the associated environmental qualification program demonstrates that each item in question is environmentally qualified for its service conditions. In response to this request, TVA provided an environmental qualification submittal on June 16, 1980, which provides the results of the applicant's review. The results of this review essentially confirm our previous conclusion as provided in Supplement Number 1 (dated February 1980) to the Safety Evaluation Report: that the associated electrical equipment is adequately qualified for its expected service environments with the exception of deficiencies which were identified for 44 types of items.

Of these 44 types of items, 6 have been replaced with qualified equipment, 1 has been relocated to a less severe environment for which that type of item is environmentally qualified, and for 5 types additional confirmatory information has been obtained from the vendor which confirms that these types will perform their safety functions in the associated harsh environments. Environmental qualification information and documentation for these items are expected to be completed by November 1, 1980. For 20 of these types, analyses were performed, which in most cases included preliminary test data. The results of these analyses provide justification for interim operation until completed qualification information can be obtained (presently expected by November 1, 1980). These analyses support the conclusion that either the items in question will operate for the duration of the stated service environments or that the function which the item performs can otherwise be completed. For an additional 11 types, TVA has reviewed additional documentation which show that these items are environmentally qualified for their service environments. For the remaining item, TVA has committed to replacing this item with qualified components if the "in situ" instruments cannot be qualified. This item will be resolved before the plant exceeds 5% power.

We find these actions, i.e., TVA's review and subsequent relocating and replacing some type of items, additional documentation providing the stated justifications for others, and previous evaluations provided by the staff to be adequate bases for the operation of this station at increased power levels pending completion of the ongoing action below.

The Commissioner's Memorandum and Order dated May 23, 1980, directs the staff to complete its review of environmental qualification including the publication of the Safety Evaluation Reports for all operating reactors by February 1, 1981. Also, this order directs that by no later than June 30, 1982, all electrical equipment in operating reactors subject to this review be in compliance with NUREG-0588 or Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors. Accordingly, the staff intends to complete the environmental qualification review in accordance with these stated dates.

7.10 Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation

On November 30, 1979, the Office of Inspection and Enforcement issued IE Bulletin 79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation," to all power reactor facilities with an operating license and to those nearing licensing. This bulletin outlined actions to be taken to address control system malfunctions and significant loss of information to the control room operator as a potential consequence of the loss of 120-volt alternating current control power to these plant systems. Further, IE Information Notice 80-10, issued on March 7, 1980, provided information relating to a Crystal River Unit 3 event of February 26, 1980, in which a significant loss of information to the operator resulted from a loss of power to a portion of the plant instrumentation system.

As a result of these concerns for operating plants, TVA identified that the instrumentation and controls required to achieve safe cold shutdown are powered from eight vital buses and non-vital unit preferred bus. TVA performed a thorough evaluation of the effects of sustained loss of power to the instrument and control loads supplied by each of these nine buses and determined that no design modifications or administrative control changes are required to permit achieving cold shutdown upon loss of a single bus. The evaluation did indicate the need for a number of additions to the plant emergency procedures; updated procedures will be in place prior to full-power operation.

Based on our review of the TVA submittal and on a visit to the TVA engineering offices by members of the NRC staff on July 28, 1980, we find that TVA has satified those portions of Criterion 13 of the General Design Criteria that are amplified by IE Bulletin 79-27 that are applicable to Sequoyah Units 1 and 2.

7.11 Engineered Safety Feature (ESF) Reset Controls

On March 13, 1980, the Office of Inspection and Enforcement issued Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," to address the concern that the use of reset pushbuttons alone could permit certain engineered safety feature system components to revert to the normal (nonaccident) state following safety system actuation. As a result of these concerns, TVA provided a response to this bulletin for the Sequoyah plant by letter dated June 12, 1980.

The review of this matter by TVA identified a number of components for which those concerns are applicable. Upon further consideration of this matter, TVA, by letter dated July 24, 1980, provided an analysis for each instance identified in the initial Bulletin response. For the majority of the items identified, the system response to reset action does not involve a safety concern. For the remainder of the items, the concern for the unwanted system response following reset action would be applicable only for specific modes of operation which are prohibited by either Technical Specifications or operating procedures.

Tests are being conducted to verify that there are no other areas of concern. Further, a thorough review of the control schemes are being conducted to determine if other schemes would enhance equipment control or increase the plant safety margin following a reset of engineered safety features.

We will continue to follow those actions which TVA has scheduled to confirm this review and any subsequent proposed modifications to enhance plant safety, although we now believe, as stated above, that the plant can be operated safely. Based on our review, we find that the applicant has satisfied IE Bulletin 80-06, which amplifies the requirements of Section 4.16 of IEEE Std. 279 (as required by 10 CFR 50.55 a(h)).

IMAGE EVALUATION TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

8.0 ELECTRICAL POWER SYSTEMS

8.3 Onsite Power Systems

8.3.1 Diesel Generator Reliability

A report prepared for the NRC, NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," made specific recommendations on increasing the reliability of nuclear power plant emergency diesel generators. Information requests concerning these recommendations and also concerning the design of the fuel oil storage and transfer system were transmitted to TVA on January 17, 1980. TVA responded in a letter dated May 19, 1980.

We have reviewed this response and have determined that the Sequoyah diesels fully conform to all of our recommendations except those listed below:

Moisture in Air Start System, Turbocharger Gear Drive Problem, and Personnel Training (Partial)

On the basis of our review, we have concluded that there is sufficient assurance of diesel generator reliability to warrant plant operation through the first refueling period. However, to assure long-term reliability of the diesel generator installations, we require that the following design and procedural modifications be implemented prior to operation following the first refueling.

Moisture in Air Starting System: The air starting system at Sequoyah does not include air dryers or filters to remove moisture and contaminants such as oil carryover and rust. The system relies on manual blowdown valves on the receivers and a line strainer to reduce the moisture and remove coarse rust particles respectively. Operating experience has shown that accumulation of water and other contaminants in the starting system have been the most frequent causes of diesel engine failure to start on demand. To improve starting reliability we require that a filter be installed downstream of the air receiver and that the air be dried to a dew point of not more than 50°F when installed in a normally controlled environment, otherwise the starting air dew point should be controlled to at least 10°F less than the lowest expected ambient temperature. We also require that the present design of the air starting system be modified to include an air dryer and filters to provide clean and dry air to the diesel engine air start valves.

Turbocharger Gear Drive Problem: The Sequoyah diesel generators have a turbocharger mechanical drive gear assembly whose gear ratio is 18:1. This assembly has not been designed to operate at no-load or light-load conditions and full-rated speed for prolonged periods. To improve the reliability and availability of the deisel generators, we require the installation of a heavy duty turbocharger drive gear assembly. TVA states that the manufacturer (EMD) has developed a heavy duty turbocharger drive gear assembly which has a gear ratio of 17:9:1 that will be available in the near future. This new gear assembly will have the desired characteristics of the 16:8:1 gear assembly recommended in NUREG/ CR-0660 without reducing the engine rating as would be required with the 16:8:1 assembly. The gear ratio may be as recommended by NUREG/CR-0660 or it may be the new 17:9:1 drive gear assembly as recommended by EMD provided it is available for installation within the time noted above.

Personnel Training: Preventive maintenance, minor repairs, and trouble shooting for the emergency diesel generators will be performed by the plant's electrical and mechanical maintenance prsonnel, but no specific training concerning diesel generator maintenance and trouble shooting has been identified for these personnel. Although TVA has stated in general terms that training would be provided for the maintenance and operating personnel, we require that a complete formal training program be identified and implemented for all the mechanical and electrical maintenance, quality control, and operating personnel, including supervisors, who will be responsible for the maintenance and availability of the diesel generators. The depth and quality of this training program shall be at least equivalent to that of training programs normally conducted by major diesel engine manufacturers.

The present diesel generator design meets the requirements of Criteria 17 and 21 of Appendix A to 10 CFR 50.

Upon implementation of the above additions, modifications, and training, Sequoyah diesel generators and their auxiliary systems and their maintenance will also be in conformance with recommendations of NUREG/CR-0660 for enhancement of diesel generator reliability and the related NRC guidelines and criteria. We therefore conclude that this will provide reasonable assurance of diesel generator reliability through the design life of the plant. This review conforms with the requirements of General Design Criteria 17 and 21 of Appendix A to 10 CFR 50 and the recommendations of NUREG/CR-0660.

9.0 AUXILIARY SYSTEMS

9.5 Fire Protection System

The staff has reviewed the applicant's proposed fire protection program and fire hazards analysis against the guidelines of Appendix A to Branch Technical Position APCSB 9.5-1, supplemental staff guidelines dated June 14, 1977, and applicable NFPA standards. The staff concluded that the fire protection program meets GDC-3 and is acceptable for full-power operation.

The staff was under the belief that all the necessary fire protection modifications were completed prior to the issuance of the Fire Protection Program Safety Evaluation in February 1980 (NUREG-0011) and were in compliance with GDC-3. However, by letter dated June 11, 1980, the licensee informed us that the seven items discussed below would not be implemented by November 1, 1980. By letter dated July 17, 1980, the licensee requested an approval of schedule for these modifications or an exception of these seven items from GDC-3 of Appendix A to 10 CFR 50. The proposed implementation dates extended from April 30, 1981 to December 30, 1981.

Subsequently, by letter dated August 7, 1980, the licensee further discussed these seven items and proposed some changes.

The first three items deal with the essential raw cooling water (ERCW) supply. To protect this system from a fire, the licensee agreed to a) enclose the necessary exposed conduit with 1½-hour fire barrier, b) reroute train B ERCW pump cables and ERCW transformer power cables to obtain a minimum 20-ft. separation from train A, and c) enclose the ERCW junction box with 1½-hour fire barrier. Sequoyah also has an auxiliary essential raw cooling water (AERCW) system that is capable of bringing the plant to a cold shutdown. If the ERCW system is incapacitated from a fire or for any other reason, then AERCW will be used to bring the plant to a cold shutdown. If any one loop of either the AERCW or ERCW is lost, the plant has to be shut down according to the plant Technical Specifications. Therefore, we conclude that the loss of the ERCW from a fire will not affect the health and safety of the public. The modifications that the licensee has committed to are extensive, and our estimate is that it would require more than 10,000 man-hours to complete them. Based on this evaluation, we find the proposed June 1981 implementation date for these items reasonable.

The fourth item is the installation of five additional fire dampers. By letter dated August 7, 1980, the licensee has agreed to complete the installation of the fire dampers by November 1, 1980.

The fifth item concerned the coating of exposed surfaces of cables with flame retardant material. The licensee had completed the coating of all exposed surfaces of initially installed cable trays. However, the licensee had to make modifications (as a result of TMI and other requirements) requiring the pulling of additional cables. According to the plant Technical Specifications, the licensee is allowed to install up to 10 cables per tray before coating is required. This will be an ongoing condition during the life of the plant. Therefore, we find that the licensee is presently complying with this item.

The sixth item was to coat the metal barrier plate in cable tray penetration assemblies with Pyrocrete. By letter dated August 7, 1980, based on test penetration assembly results, the licensee requested that the metal barrier plates in these penetration assemblies not be coated with Pyrocrete. We have reviewed the construction features of the electrical penetration assemblies involving these plates and conclude that the licensee has provided acceptable documentation to demonstrate the requested fire resistability of the affected electrical penetration assemblies. Based on the results of these tests, we agree with the licensee that the metal barrier plates in the cable tray penetration assemblies need not be coated with Pyrocrete.

The last item concerns the additional sprinkler heads and the relocation of existing heads that are needed in the auxiliary building. On August 7, 1980, we visited the plant and found that 58 sprinkler heads need to be installed or relocated. We will condition the license requiring TVA to complete the installation or relocation of these 58 sprinkler heads by November 1, 1980. By letter of August 11, 1980, TVA has stated that the sprinkler heads will be in place by November 1, 1980.

On April 23, 1980, the Commission approved a proposed rule concerning fire protection. The rule and its Appendix R were developed to establish the minimum acceptable fire protection requirements necessary to resolve certain areas of concern between the staff and the licensees of plants operating prior to January 1, 1979. On May 23, 1980, the Commission issued a Memorandum and Order (CLI-80-21) related to the proposed rule and stated therein: "The combination of the guidance contained in Appendix A to BTP 9.5-1 and the requirements set forth in this proposed rule define the essential elements for an acceptable fire protection program at nuclear power plants docketed for Construction Permit prior to July 1, 1976, for demonstration of compliance with General Design Criterion 3 of Appendix A to 10 CFR Part 50," (p. 19). In the event that the rule, when it becomes an effective rule, has provisions which apply to Sequoyah Nuclear Plant Unit 1, such provisions will be implemented in accordance with the rule.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.0 In SER Supplement No. 1, we stated that we had not yet completed our review of TVA's Process Control Program (PCP) and that implementation of Technical Specifications 3/4-11.3 on Solid Radioactive Waste would be delayed until the PCP had been approved.

On July 18, 1980, TVA provided Revision 4 to the PCP. A demonstration test was successfully performed using boric acid evaporator bottoms and the solidified product checked for transportability. The demonstration test was observed by the NRC Resident Inspector. We find that the PCP when used at Sequoyah Nuclear Plant will result in acceptable solidification of radwaste and is therefore acceptable. Further, we find that the associated Technical Specifications may be implemented and that the PCP meets the applicable requirements and guidance set forth in Regulatory Guide 1.43 and General Design Criteria 1, 2, and 60 of Appendix A to 10 CFR 50.

13.0 CONDUCT OF OPERATIONS

13.2 Training Program

In SER Supplement No. 1, we stated that TVA-licensed operators augmented with experienced startup engineers were acceptable for low-power operation. We have reviewed this matter for operation beyond the low-power testing operation and have concluded that such augmentation of the TVA-licensed operators above 5% power should continue through the startup program up to and including 50% of full power. Continued presence of the special staff will be dependent upon plant status and staff development. We will so condition the full-power license (see also Section 22.2.I.B.1.1.).

This meets the requirements specified in the Technical Specifications and 10 CFR 50.57(a)(4). This matter is also directly a consideration of 10 CFR 55.33, renewal of licenses. The staff believes that the pertinent sections of these regulations have been met.

13.3 Emergency Planning

Our evaluation of emergency preparedness is discussed in 22.3.III.A.1.1.

15.0 ACCIDENT ANALYSIS

15.2 Normal Operation and Anticipated Operational Transients

ATWS

Section 15.3.8 of the Safety Evaluation Report (March 1979) addressed the background of the staff's concerns on ATWS and stated interim procedural and operator training requirements to reduce the risk from anticipated transients without scram. Further requirements may result from the Commission's rulemaking on ATWS. Section 15.2 of SER Supplement 1 stated that we had reviewed TVA's proposed procedures for ATWS and required that the procedures be modified in accordance with staff comments on them. The design features dealing with an ATWS situation are generally met for Sequoyah in accordance with the provisions of GDC 10, 15, 26, 27, and 29 of 10 CFR 50 Appendix A.

TVA revised the ATWS emergency procedure to incorporate staff comments. We reviewed the revised procedure and observed Sequoyah operators responding to an ATWS event on the TVA simulator as a part of our review of emergency procedures (addressed in Section 22.2.I.C.1 of this supplement). Using this procedure, the operators diagnosed the event and took appropriate actions to minimize its effects and bring the simulated plant to a safe shutdown condition.

The procedure describes the automatic responses of the plant as well as the operator's actions taken immediately after he diagnoses the ATWS and later when he attempts to bring the plant to a cold shutdown condition.

Based on our review and observations, we conclude, pending the outcome of the Commissioner's rulemaking on ATWS, that the emergency procedure and operator training on ATWS are acceptable for interim full-power operation of the Sequoyah Nuclear Plant Unit 1 in accordance with General Design Criteria 10, 15, 26, 27, and 29 of 10 CFR 50 Appendix A, based on our understanding of the plant response to postulated anticipated transients without scram events. The Commission will, by rulemaking, determine any future required modifications necessary to resolve ATWS concerns and the required schedule for implementation of such modifications.

Topical Reports Relating to Steam Line and Feedwater Line Breaks

In Section 15.2 of the Safety Evaluation Report, we stated that we required TVA to commit to provide prompt responses to additional information requirements regarding the review of Westinghouse transient analysis codes dealing with steam line and feedwater line break accidents.

The plant response analyses of postulated steam line and feedwater line breaks were evaluated with the use of the MARVEL computer program (WCAP-7909). MARVEL is a systems code designed to model transients which do not result in primary side two-phase separation. The primary system nodes are treated homogeneously. The MARVEL computer program is presently under review by the NRC staff. Due to some simplified assumptions used in the development of the code, the staff requires confirmation from Westinghouse of the steam line break and feedwater line break analytical methodology with a more detailed model, as provided in WCAP-9230, "Reactor Core Response to Excessive Secondary Steam Releases"; WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture"; and WCAP-9236, "NOTRUMP - A Nodal Transient Steam Generator and General Network Code." By letter dated August 8, 1980, TVA has agreed to participate in a confirmatory review of its steam and feedwater line break analyses, as part of the ongoing generic review of the Westinghouse topical reports. This review is intended to confirm that the analyses conducted for Sequoyah Units 1 and 2 were appropriate and conservative. TVA agreed to provide plant-specific inputs to NRC for an independent audit should the staff conduct one.

The analytical methods used for postulated transients and accidents are normally reviewed on a generic basis. Our review at this time indicates that there is reasonable assurance that the conclusions based on the SAR analyses will not be appreciably altered by the completion of the analytical methods review. If the final approval of the methods indicates revisions to the analyses are required, the licensee will be required to implement the results of such changes.

Based on previous acceptable analyses for Westinghouse plants, on a comparison with other industry models, on independent staff audit calculations, and on previous startup testing experience, we conclude that, with the exceptions noted above, the analytical methods used for Sequoyah Units 1 and 2 are acceptable for full-power operation.

15.4 Radiological Consequences of Accidents

15.4.1 Loss-of-Coolant Accident

In our Safety Evaluation Report (SER) of March 1979, we concluded that the combined radiological consequences due to leakage of post-LOCA recirculation water from a postulated seal failure of an ESF pump and those due to direct containment leakage would be within the guidelines of 10 CFR 100. This conclusion was based on the determination that the auxiliary building gas treatment system (ABGTS) will maintain a negative pressure of 0.25 inch of water gauge throughout the auxiliary building secondary containment enclosure (ABSCE), including the ESF pump rooms, and thus will prevent direct exfiltration of the airborne pump seal leakage to the environment. This negative pressure would assure that any release would be filtered prior to its discharge.

In Supplement 1 to our SER of February 1980, we restated this conclusion but also determined that TVA should demonstrate by test, and prior to our issuing a full-power operating license, that the ABGST can establish and maintain the specified negative pressure. The basis for this requirement was TVA's indication at that time that the ABGTS could not meet this requirement during the interim period until the completion of the Sequoyah Unit 2. The staff determined that in the absence of this negative pressure and with the assumption of direct exfiltration of the pump seal leakage to the environment, the combined radiological consequences could exceed the guidelines of 10 CFR 100 at full-power operation.

TVA provided in Amendment 64 (April 11, 1980) of FSAR Section 6.2.3.2.3 descriptive information on the interim ABSCE. TVA advised us on July 28, 1980, confirmed by letter of August 1, 1980, that a test had been performed to demonstrate the depressurization capability of the ABGST and provided us with the test results. Additional clarification on the test procedures was provided by telephone on July 31, 1980. The test results show that each of the two redundant trains of the ABGTS has the capability to reduce the pressure in the interim ABSCE to less than a negative pressure of 0.25 inch of water gauge in less than 2 minutes. This negative pressure was achieved at the operating floor of the spent fuel pool as well as in all ESF pump rooms. TVA also informed us that the major modifications, made since the earlier unsuccessful tests in February 1980, include the installation of dedicated and less restrictive exhaust ducts from the ESF pump rooms and the installation of additional and improved seals at doors and other openings.

We have evaluated the information provided by TVA in Amendment 64 and in our discussion of July 31, 1980. We conclude that the ABGTS has sufficient capability to achieve and maintain a negative pressure throughout the interim ABSCE and thus prevent the direct exfiltration of potential airborne radioactivity to the environment. We therefore reaffirm that the combined radiological consequences associated with an ESF pump seal failure and with the design basis accident are within the guidelines of 10 CFR 100 and therefore are acceptable.

17.0 QUALITY ASSURANCE

Our review of the quality assurance program description for the operations phase for this facility has verified that the criteria of Appendix B to 10 CFR 50 have been adequately addressed in Section 17.2 of the FSAR. This determination of acceptability included a review of the list of safety-related structures, systems, and components (Q-list) to which the quality assurance program applies. The results of a revised NRC staff procedure for conducting the Q-list review, that involved other NRR technical review branches and significantly enhances the NRC staff's confidence in the acceptability of the Q-list, have been discussed with TVA. Differences between the current Q-list and NRR requirements have been resolved by information provided in letters dated July 11, 1980 and July 31, 1980.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its letter of December 11, 1979, the Committee addressed the special low-power test program and several other aspects of the plant. The Committee stated that there was reasonable assurance that such a test program could be conducted without undue risk to the health and safety of the public. This letter is incorporated in Appendix D of Supplement No. 1 to the Safety Evaluation Report.

On July 9 and 11, 1980, the ACRS completed its review of the application of TVA for authorization to operate the Sequoyah Nuclear Plant, Units 1 and 2, at full power. The Committee's letter of July 15, 1980, stated that Sequoyah units can be operated at levels up to full power without undue risk to the public if due consideration is given to the following items:

1. Efforts on hydrogen control should be vigorously pursued.

2. The acceptability of the pressurizer relief line weld repair (refer to section 5.2.6.2) should be pursued further. The Committee suggested another mockup of the weld in question.

These matters have been pursued by the staff and by TVA. The efforts on hydrogen control are discussed in Section 22.2, "Full-Power Requirements" (II.B.7). The resolution of the weld repair is discussed in Section 5.2.6.2. Review by the ACRS of action to issue operating licenses to reactor facilities is mandated by the Atomic Energy Act of 1954, as amended, and by the Emergency Reorganization Act of 1974, as amended.

22.0 TMI-2 REQUIREMENTS

22.1 Introduction

In a letter dated June 26, 1980, we advised all applicants for construction permits and operating licenses of the Commission's guidance regarding the requirements to be met for current operating license applications. The requirements are derived from NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating Licenses."

The requirements discussed in NUREG-0694 were listed in four categories: those required for fuel loading and low-power testing requirements; those required for full-power operation; those requiring internal NRC action; and those required to be implemented by a certain date.

Since requirements for fuel loading and low-power testing were addressed in Part II of Supplement No. 1 to the Sequoyah Power Station Unit 1 Safety Evaluation Report, this supplement only addresses full-power requirements and dated requirements.

Each applicable full-power requirement and appropriate dated requirements are discussed below and follows the numbering sequence used in NUREG-0694.

22.2 Full-Power Requirements

I. Operational Safety

I.A.1 Operating Personnel and Staffing

I.A.1.3 Shift Manning

POSITION

Assure that the necessary number and availability of personnel to man the operations shifts have been designated by the licensee. Administrative procedures should be written to govern the movement of key individuals about the plant to assure that qualified individuals are readily available in the event of an abnormal or emergency situation. This should consider the recommendations on overtime in NUREG-0585. Provisions should be made for an aide to the shift supervisor to assure that, over the long term, the shift supervisor is free of routine administrative duties.

DISCUSSION AND CONCLUSION

Our requirements for shift manning of Sequoyah Nuclear Plant are described below.

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	*MODES 1, 2, 3 & 4	*MODES 5 & 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

*Refer to definition of operational modes (Table 22.2-1).

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1

SRO - Individual with a Senior Reactor Operators License on Unit 1

RO - Individual with a reactor Operators License on Unit 1

AO - Auxiliary Operator

STA - Shift Technical Advisor

TVA currently has 22 individuals holding senior operator licenses and 12 individuals holding reactor operator licenses in Unit 1 of the Sequoyah Nuclear Plant. The number of licensed SROs and ROs is more than adequate to meet the shift manning requirements without routine overtime. We consider the number of licensed operators sufficient to meet the manning requirements of Technical Specification 6.2, Minimum Shift Crew Composition, in all operating modes.

TABLE 22.2-1

DEFINITION OF OPERATIONAL MODES

MODE	REACTIVITY CONDITION, Keff	% RATED TERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4 HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

* Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

Note: These definitions are contained in the Sequoyah Technical Specifications.

I.B.1 Management for Operations

I.B.1.1 Organization and Management Criteria

POSITION

Assure that the applicant meets the requirements for onsite and offsite support personnel, both management and technical, that will assure safe operation of the plant during normal and abnormal conditions and provide the capability necessary to respond to accident situations.

Items to be considered include (a) competence of management and technical staff, both onsite and offsite; (b) size of offsite staff and degree of involvement in plant operations; (c) types of expertise needed; (d) pooling of resources among utilities; (e) organizational arrangements for both normal and accident situations; (f) training of management and technical personnel, both onsite and offsite, to assure full knowledge of plant operations and reactor safety; (g) staffing of control room personnel; (h) quality assurance program and staffing; (i) financial capability (in the event reliance is placed on outsid contractual assistance during the accident situation; (j) requalification program for management and technical personnel; (k) procedures for normal operations, accident conditions, surveillance, and maintenance; (l) special requirements for accident situations including control room access, onsite technical support center, and onsite operational support center; (m) status of pre-established plans for using available resources in the event of unusual situations; and (n) reporting of unusual events; (o) policy for the consideration at management levels of safety issues identified at all levels, but unresolved.

DISCUSSION AND CONCLUSIONS

This matter is discussed in Section 13.2 of this supplement. TVA is required to augment the control room staff for operations above 5% power through the startup program up to and including 50% of full power.

I.B.1.2 Safety Engineering Group

POSITION

An independent safety engineering group shall be established to increase the available technical expertise located onsite and to provide for continuing, systematic, and independent assessment of nuclear plant activities. This group, which shall consist of not less than five dedicated, full-time engineers, shall be physically located onsite, but shall report offsite to a high-level corporate official who is not in the management chain for power production. The function of this group shall be to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, Licensee Event Reports, and other appropriate sources which may indicate areas for improving plant safety. Where useful improvements can be achieved, it is expected that this group will develop detailed recommendations for revised procedures, equipment modifications, or other means of achieving the goal of improved plant safety. A principal function of the independent safety engineering group shall be to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as much as practical. The independent group shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

DISCUSSION AND CONCLUSION

TVA provided an independent safety engineering group during the special low-power test program in accordance with the staff position. We require that TVA maintain the independent safety engineering group on a continuing basis for full-power operations. TVA, by letter of August 11, 1980, agreed to maintain the independent safety engineering group. This requirement will be incorporated in the Technical Specifications.

I.C.1 Short-Term Accident Analysis and Procedure Revision

POSITION

Analyze the design basis transients and accidents including single active failures and considering additional equipment failure and operator errors to identify appropriate and inappropriate operator actions. Based on these analyses, revise, as necessary, emergency procedures and training. This requirement was intended to be completed in early 1980; however, some difficulty in completing this requirement has been experienced. Clarification of the scope and revision of the schedule are being developed and will be issued by July 1980. It is expected that this requirement will be coupled with Task I.C.9., Long-Term Upgrading of Procedures. See NUREG-0578, Sections 2.1.3b and 2.1.9 (Ref. 4), and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24).

DISCUSSION AND CONCLUSIONS

In Amendment 4 to License DPR-77 we stated that prior to operation above 5% power, we would observe a simulation of selected Sequoyah emergency procedures conducted by Sequoyah personnel and a walk-through of at least one emergency procedure in the control room. The objective was to verify that the emergency procedures adequately addressed successful mitigation of accidents and transients, as required in Section I.C.8 of NUREG-0660. With respect to the analysis for small-break accidents for UHI plants, this matter is discussed in II.F.2 of this supplement.

On July 23 and 24, 1980, a team of NRC and contractor personnel observed Sequoyah operators participating in the simulation of several transients and accidents on the Sequoyah simulator. The transients and accidents included loss-of-coolant accidents (LOCA) in a range of break sizes, steam generator tube rupture, loss of main feedwater, and recovery from inadequate core cooling. Some transients and accidents were run more than once and equipment failures such as loss of offsite power and failure of one emergency diesel generator, failure of scram breakers to open (ATWS), and failure of individual components in the emergency core cooling systems and auxiliary feedwater systems were included in the simulated events. During the simulation of the events and following each event, we discussed the operators' actions and the procedures with the operators.

On July 24, 1980, the team observed a walk-through of the Emergency Operating Instruction for a LOCA in the Sequoyah Unit 1 control room and discussed the procedure with the operators.

The procedures provided for our review have been revised to reflect the Westinghouse analysis of small-break LOCAs and inadequate core cooling in accordance with license requirement and Task Action Plan (NUREG-0660) Item I.C.1.

The procedures had been reviewed by the NSSS supplier, Westinghouse, and changes recommended by Westinghouse had been incorporated in compliance with Task Action Plan I.C.7(a).

Some procedural deficiencies were identified to TVA personnel during the simulator exercises and the control room walk-through. The necessary changes were made to drafts of the procedures. We require that these changes be made to the approved procedures and that the Sequoyah operators be briefed on the changes and their bases prior to their assuming operating responsibilities on Unit 1 after issuance of the full-power license. We also require that the remainder of the emergency operating instructions be revised in accordance with our comments on the procedures reviewed and that the operators be briefed on the revisions within 30 effective full-power days of operation. The Office of I&E will verify that these requirements are satisfied.

During the simulator exercises and the control room walk-through, we observed that several control board labels were not consistent with the equipment nomenclature in the procedures. TVA agreed to correct the control board labeling inconsistencies identified by the staff prior to full-power operation. I&E will verify that this action has been taken. We also observed that some instrumentation referred to in the immediate actions section of the procedures is located on a panel behind the control board (e.g., containment temperature and humidity indications). It is preferable that the operator not have to leave the main control board area to perform immediate operator actions; however, there is adequate manpower present in the control room to perform the task and operator training assures that the operators are aware of the location and significance of these instruments. We believe that, in the longer term, some control room modifications should be made. In a letter dated August 11, 1980, TVA agreed to make control room design changes. To ensure that these additional modifications are made in the most efficient and effective manner, we will not require their implementation until TVA has completed the detailed control room design review. We require that this review be completed and the corrective actions implemented consistent with the schedule of the TMI Task Action Plan as follows:

Based on our review of the emergency procedures and our observation of the procedures being implemented on the simulator and in the plant walk-through, we have concluded that the Sequoyah emergency procedures are acceptable for operation at power levels up to 100% of rated power, since the procedures assure that plant operators will perform the correct actions in a timely manner following an accident or transient. We have concluded that the actions called for in Task Action Plan, Items I.C.1.a(1), LOCA, I.C.1.a(2), Inadequate Core Cooling, I.C.7.a., NSSS Vendor Review of Procedure, and I.C.8, Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants, have been adequately completed. Future actions addressed by Task Action Plan Items I.C.1.a(3), Transients and Accidents, and I.C.9, Long-Term Program Plan for Upgrading of Procedures, may require future revisions to the emergency procedures. These revisions will be identified in the long-term program stipulated in Item I.C.9.

I.C.7 NSSS Vendor Review of Low-Power Test Procedures

POSITION

Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy.

This requirement must be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

The NSSS vendor, Westinghouse Electric Corporation, has reviewed the Sequoyah low-power and power-ascension test procedures and emergency procedures. The changes recommended by Westinghouse have been incorporated into the procedures. This review has been documented in letters to the staff dated March 27 and April 28, 1980. This satisfies Item I.C.7 of NUREG-0694.

I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants

POSITION

Correct emergency procedures as necessary based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steam-line break or steam-generator tube rupture).

This action will be completed prior to issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

During our review of these procedures, we met with TVA on July 14, 1980. We also observed the simulation of these procedures and observed a walk-through of one of them during a site visit the week of July 21, 1980.

We have reviewed the guidelines for small-break LOCA and inadequate core cooling for Westinghouse plants and conclude that TVA has revised its Sequoyah procedures to follow these guidelines. A more complete discussion of this item is in Section 22.2, I.C.1. This satisfies Item I.C.8.

I.D.1 Control Room Design Review

POSITION

Perform a preliminary assessment of the control room to identify significant human factors deficiencies and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies.

DISCUSSION AND CONCLUSIONS

In Section IV of Part II of Supplement No. 1 to the Safety Evaluation Report for Sequoyah Unit 1, we identified a number of corrective actions which we believed were necessary to improve operator effectiveness during emergency operations. TVA was required to implement a number of corrective actions prior to criticality and several corrective actions at a later date (prior to escalation beyond 5% of rated power). Accordingly, the low-power testing operating license for Sequoyah Unit 1 was conditioned to reflect these actions.

Corrective actions implemented by TVA prior to criticality were:

1. Dedicated panel telephones were installed to improve control room communications between operators.

2. Panel guardrails were installed to prevent inadvertent actuation of switches on vertical panels 1M1 through 1M6. Warning tape was installed at the base of vertical panels 1M7 through 1M15, 1M18, 0M12, 0M26 and 0M27 to designate off-limit areas to employees not performing a required task.

3. Arrangements were made to maintain procedures in a specific location in the control room and an index was added to assist operators in locating and accessing specific emergency procedures. Also, immediate action steps in emergency procedures were revised to eliminate references made to external documents.

4. Alarms important to safety were prioritized by color coding annunciators windows (tiles).

5. Panels which contain controls and displays unique to Units 1 and Unit 2 and common to Units 1 and 2 were modified to improve identification of displays and controls by using color coding and adding specific identification of each (i.e., 0 for common, 1 for Unit 1 and 2 for Unit 2).

6. A black border (bezel) was installed on all overhead annunciator display panels to improve contrast between annunciator windows and background.

7. Actions were taken to reduce the background noise level in the control room.

8. Control room procedures were revised to instruct operators to use the lamp test buttons on the status monitoring panels to verify that a lamp is burned out, to assure that a system is not available.

During the week of July 28, 1980, the Office of Inspection and Enforcement audited the measures implemented by TVA to meet the corrective actions required prior to escalation beyond 5% rated power. The audit was to verify implementation of the human factors improvements made to the control room which will serve to substantially improve the operator's ability to take effective control actions under stressful conditions. The corrective actions that TVA was required to implement prior to full-power operation are:

1. Bind emergency procedures stored in the control room so that each one can be individually removed when needed.

2. Install improved labels on all panels including panels which contain common controls and displays. Add improved labels to identify functional grouping, subsystems, systems, and panels.

3. Take additional actions necessary to further reduce the control room background noise level to a maximum of 65 dBa.

4. Blank out or otherwise identify unused windows (tiles) on status monitoring panels.

5. Correct problem associated with one process computer printer.

6. Improve operator capability for donning emergency equipment.

7. Improve the immediate operator action steps of emergency procedures in the following areas:

   (a) Provide one instruction per step.

   (b) Shorten and simplify instructions.

   (c) Include all necessary steps.

   (d) Provide cross references.

Based on an oral report from the I&E resident inspector (July 31, 1980), the licensee has satifactorily implemented all the above stated corrective actions, except Item 3. Further work is needed to meet the requirement of Item 3.

During our initial review the week of February 4, 1980, we identified a number of minor deficiencies which we believe offer no significant risk to full-power operation. TVA sould consider correcting these as time permits. However, for the longer term, in order to ensure that further modification to the control room are made in the most efficient and effective manner, we will not require correction of these minor deficiencies until TVA has completed the detailed control room design review to be required of all operating reactor licensees. As part of this design review, we will require the TVA to evaluate the benefits of installing data recording and logging equipment in the control room to correct the deficiencies associated with the trending of important parameters on strip chart recorders used in control rooms at most nuclear power plants.

Because of the similarity between Units 1 and 2, we will require that all corrective actions specified for Unit 1 also be implemented on Unit 2 as appropriate. In a letter dated August 11, 1980, TVA has documented proposed changes to the Unit 2 control room with confirmation that Unit 1 improvements will be made on Unit 2, explanation and justification of differences where Unit 1 improvements cannot be made on Unit 2, and identification of any additional Unit 2 improvements. NRR will review the TVA Unit 2 proposal and OI&E will be requested to audit its implementation.

CONCLUSIONS

Based on the TVA's implementation of the corrective actions identified by our control room design review and I&E's audit of the implementation of the human factors requirements stated herein and in SER Supplement 1, we believe that the control room design is such that Unit 1 can now be safely operated at full power. We conclude that the full-power requirements of Item I.D.1 of NUREG-0694 have been met for Unit 1.

I.G.1 Training During Low-Power Testing

POSITION

The TMI Task Action Plan states that applicants for operating licenses will perform a set of low-power tests to increase the capability of shift crews and ensure training in plant evolutions and off-normal events. Near-term operating license facilities will be required to develop and implement intensified exercises during the low-power testing programs. This may involve the repetition of startup tests on different shifts for training purposes.

DISCUSSION AND CONCLUSION

In a letter dated December 3, 1979, TVA proposed "pursuing certain limited activities in the case of those power plants where construction has been completed during the Commission's pause...." One of the activities proposed was a series of natural circulation tests to be performed at power levels up to 5% of full design power.

The NRC staff reviewed the low-power test program proposed by TVA using the following five criteria:

1. The tests should provide meaningful technical information beyond that obtained in the normal startup test program.

2. The test should provide supplemental operator training.

3. The tests should not pose an undue risk to the public.

4. The risk of damage to the nuclear plant during the test program should be low.

5. The radiation levels that will exist after the low-power test program is completed (including those from crud deposits) must not preclude implementation of requirements stemming from the NRR Lessons Learned Task Force, the Kemeny Commission, the Rogovin Commission, or the Task Action Plan (NUREG-0660).

On December 7, 1979, TVA submitted a document that briefly stated the purpose, listed the major initial conditions, and outlined the test method for each test. Subsequently, on January 7, 1980, TVA submitted a draft of the special operating procedures for each of the ten proposed tests. These special procedures included the objectives, prerequisites, precautions, special test equipment, instructions, and acceptance criteria for each test.

The staff conducted a review of the proposed test program and concluded that the test program as described in the special operating procedure and TVA safety evaluation could be conducted safely at Sequoyah Unit 1. A discussion of how TVA met the criteria listed above was included in paragraph I.G.1 of SER Supplement 1.

Approval of the TVA special test procedure, and TVA's safety evaluation of the special test program documented in Amendment 4 to the Sequoyah Unit 1 low-power test license, DRR-77.

The special low-power test program, as approved by the NRC, was conducted at Sequoyah Unit 1 starting on July 11, 1980. NRC staff representatives including the Sequoyah resident inspectors were present to observe these tests. At least one NRC representative was present during the first run of each of the ten tests, identified as follows:

1. Natural circulation test

2. Natural circulation with simulated loss of offsite power

3. Natural circulation with loss of pressurizer heaters

4. Effect of steam generator secondary side isolation on natural circulation

5. Natural circulation at reduced pressure.

6. Cooldown capability of the charging and letdown system

7. Simulated loss of all onsite and offsite ac power

8. Establishment of natural circulation from stagnant conditions

9A. Forced circulation cooldown

9B. Boron mixing and cooldown

Tests 6, 8, and 9A were each conducted only once. All other tests were repeated on each shift so that each operating crew gained "hands-on" experience for each test. Not repeating tests 6, 8, and 9A was acceptable to the staff because they have little training value. Two licensed operators were on scheduled vacation and one was on sick leave during conduct of the test program. These three operators, however, did receive simulator training on the special tests. We therefore conclude that TVA has met the requirement for operator training on the special low-power test program.

TVA has submitted a special startup test report which describes each test and presents test results and operator training, achieved during the tests. A brief summary of the results of each test follows.

Test #1 Natural Circulation Test

All reactor coolant pumps were tripped with the reactor at approximately 3% power. the initial loop delta T's were 1.5 to 2°F. As the flow coastdown started, delta T and pressurizer level and pressure began to rise. Stable natural circulation was achieved in 8 to 10 minutes with loop delta T's about 36°F during the transient, RCS pressure rose to the power-operated relief valve setpoint of 2335 psig then stabilized at about 2320 psig. The PORV momentarily lifted and then reseated. The pressurizer level rose about 8%. Thermocouples located in the reactor vessel upper head indicated that the upper head temperature followed the core exit thermocouple reading throughout the test. The plant responded as expected during this test and all test objectives were met.

Test #2 Natural Circulation with Simulated Loss of Offsite Power

With the reactor at approximately 1% power the simulated loss of offsite power was initiated. Average loop delta T's ranged from 18 to 19°F with a much greater variation between loops than was experienced in the tests that were run at 3% power. Data indicated that one or two steam generators tended to carry the majority of the heat load due to heavier additions of feedwater and a corresponding increase in natural circulation flow in those loops.

The blackout signal, automatic start of diesel generators, and vital load sequencing to emergency power buses went as expected. Equipment and components requiring manual actuation were successfully loaded onto the diesels, and the plant was maintained in a stable operating condition. The process was then reversed with the emergency loads returned to offsite power and the diesels returned to standby status. The plant responded as expected, and all test objectives were met.

Test #3 Natural Circulation with Loss of Pressurizer Heaters

With the reactor at approximately 3% power, the reactor coolant pumps and all pressurizer heaters were tripped and the reactor coolant system allowed to come to equilibrium conditions. The reactor coolant system charging and letdown flow rates were adjusted to maintain a constant pressurizer level and the pressurizer was allowed to slowly cool. Over the testing period, the cooldown rate averaged between 6 and 7°F/hr (approximately 100 psig/hr).

Once the depressurization rate had been determined, primary system charging was increased to verify that the margin to saturation could be controlled by increasing pressurizer level (and therefore system pressure). The response to a 40-gpm increase in charging flow was immediately noticeable in system pressure and saturation margin. The pressure increase averaged about 12 to 14 psig for each 1% increase in pressurizer level.

A slight increase in steam flow slowed the pressure increase due to cooling of the primary system, but the saturation margin continued to increase. The plant responded as expected during the test, and all test objectives were met.

Test #4, Effect of Steam Generator Secondary Side Isolation on Natural Circulation

With the reactor at approximately 1% power, the reactor coolant pumps were tripped and the reactor coolant system allowed to come to equilibrium conditions. Steam generator #3 was then isolated by closing its main steam isolation and feedwater valves. Delta T in this loop soon began to reduce with the cold leg temperature slowly rising toward the hot leg temperature. Delta T's in the three operating loops increased about 6 to 8°F. Delta T in the the isolated loop dropped 5°F to 15.6°F and stabilized. Then Steam Generator #4 was also isolated, the final Delta T in loop 3 dropped to 5°F, and to 6.9°F in loop 4. A summary of the final Delta T's for all three configurations are tabulated below.

LOOPS ISOLATED	LOOP 1 DELTA T (°F)	LOOP 2 DELTA T (°F)	LOOP 3 DELTA T (°F)	LOOP 4 DELTA T (°F)
0	15.9	20.6	20.3	18.5
Loop 3	22.7	26.2	15.6	28.4
Loops 3 & 4	36.8	42.4	5	6.9

Recovering from the isolated condition was slow, but no problems were encountered. The delta T on loop 4 began to increase as soon as the atmospheric relieve valve was opened. Again the increased natural circulation flow in loop 4 induced more flow in loop 3 as the delta T in loop 3 also began to increase even though the steam generator was still isolated.

Loop 3 was not completely isolated when the steam and feed valves were closed and the level in SG#3 was decreasing so feedwater was added to maintain the level. When loop 4 was isolated, for some reason valves isolating loop 3 closed tighter and the SG #3 level remained constant without feedwater addition. This accounts for the apparent effect on loop 3 Delta T of isolating loop 4.

The test was conducted without incident and all test objectives were met.

Test #5, Natural Circulation at Reduced Pressure

With the reactor at approximately 3% power, the reactor coolant pumps and all pressurizer heaters were tripped and the reactor coolant system allowed to come to equilibrium conditions. The reactor coolant system (RCS) charging and letdown flow rates were adjusted to maintain pressurizer level approximately constant. As the RCS depressurized, system parameters were recorded including saturation margin. The lowest indicating RCS system pressure and the highest indicating RCS temperature were used to determine saturation margins as pressure was reduced. The saturation margin was calculated using ASME steam tables and compared to the margin indicated on the plant saturation meter. The two values were found to be essentially identical. The plant responded as expected during this test, and all test criteria were met.

Test #6, Cooldown Capability of the Charging and Letdown system

With the reactor at zero power and one reactor coolant pump running, all steam generators were isolated. When charging and letdown flows were increased to their maximums, a cooldown rate of approximately 15°F/hr was observed. When charging and letdown lows were reduced to minimum flow, cooldown rate of approximately 7°F/hr was observed.

The plant responded as expected during the test, and all test objectives were met.

Test #7, Simulated Loss of All Onsite and Offsite AC Power

With the reactor at approximately 1% power, the reactor coolant pumps were tripped along with all normal auxiliary building lighting, vital instrument power, ventilation in the main control room, and turbine-driven auxiliary feedpump room, pressurizer heaters, reactor coolant system charging and letdown, the motor-driven auxiliary feedwater pumps, and the main feedpump in operation at that time. All motor- and air-operated valves were assumed inoperable and only vital instruments, powered by the emergency battery system, were used for the duration of the test.

Operators were sent to the auxiliary feedwater valves and to the steam generator power-operated steam relief valves to manually control steam generator pressure and level as required. After the trip, the air accumulators on the auxiliary feedwater valves allowed the valves to operate automatically for approximately seven minutes after the trip sequence began. There was little demand on the valves at the beginning of the test so the validity of this time is questionable. Eventually complete manual control was taken on the feedwater valves and a steady flow established to the steam generators as required. Again the tendency for uneven power distribution between loops at low-power levels was seen, as only one atmospheric relief valve had to be opened during the first hour of the test. Eventually all atmospheric relief valves and feedwater valves were manually opened and adjusted to maintain equilibrium condition. Loops 2 and 4 removed more heat with delta T's of approximately 21°F as compared with loops 1 and 3 delta T's of around 13-14°F.

Temperatures in the auxiliary feed pump room rose steadily over the first 90 minutes of the test but stabilized at approximately 112°F. The main control room temperature continued to rise during the test but at the slow rate of about 4°F per hour. After two hours the main control room temperature was 84°F.

Temperature in the main steam valve room rose to approximately 165°F during the test. However, this did not preclude manual actuation of the steam relief valves during the test. TVA has initiated plant changes to limit temperatures in this area to insure access in the event of a loss of ac power. TVA indicated in oral discussions that all safety-related equipment in this area is designed for operation at temperatures in excess of 165°F (steam line break requirements).

The vital batteries maintained the emergency loads through the duration of the test. The output voltage on each bank was monitored closely and no detectable reduction was seen by the end of the 2-hour test. At the end of the 2-hour period, normal power was returned to the vital instruments and emergency lighting, and the plant was restored.

The plant responded approximately as expected, and all test objectives were met.

Test #8, Establishment of Natural Circulation from Stagnant Conditions

With the reactor at zero power, three reactor coolant pumps were tripped and allowed to coast down and then the fourth reactor coolant pump was tripped. This method of tripping the pumps induced reverse flow in three loops to more closely simulate stagnant conditions when the last pump was stopped. All steam generators were then isolated for approximately five minutes to allow flow to coast down and to avoid inducing any circulation due to cooling effects of feedwater or steam dump.

A slow reactor power increase was then started at approximately 0.13 decades per minute. Within about two or three minutes, signs of a small temperature rise across the core became evident as loop delta T's started to rise. Loop 1 delta T indicated that some reverse flow existed in loop 1 but loops 2, 3, and 4 showed no signs of this. From all indications, natural circulation started almost immediately following nuclear heating and increasing smoothly with reactor power. When steam dumps were opened, T cold dropped noticably and delta T increased. In loop 1 where reverse flow was evident, the flow reversed within 10 minutes after starting nuclear heating. Loop 1 has consistently shown lower delta T under natural circulation than the other loops, possibly due to the effects of charging flow into this loop. Core power was increased to approximately 2% power with almost no time lag in natural circulation flow. The core exit thermocouple temperatures came up steadily with power, indicating there is essentially no minimum core temperature rise required to induce natural circulation.

Test 8 was successfully completed, and the test objectives were met.

Test #9A, Forced Circulation Cooldown

With the reactor at approximately 3% power and all four reactor coolant pumps running, a slow cooldown was initiated using steam dump to the condenser. Periodically throughout the cooldown and the following heatup, reactor power measurements were made using the incore movable detectors, a primary steam calorimetric using best estimate flow rates, and the excore nuclear instrumentation. The incore movable detectors and the primary side calorimetric calculated powers were averaged at each temperature plateau and compared to the excore indicated power to determine an excore indicated power correction factor. The incore movable detector power measurements were assumed to be unaffected by the lower temperatures.

The calculated correction factor as a function of cold leg temperature was determined to be approximately .375% reduction in indicated excore power per 1°F cooldown in the cold leg. This correction factor was used in all natural circulation tests where T cold was reduced.

The test was successfully completed, and the test objective of determining the excore detector indicated power correction factor as a function of average cold leg temperature was accomplished.

Test #9B, Boron Mixing and Cooldown

With reactor at approximately 2.5% power and natural circulation established, a slow boration of the reactor coolant system was started (2.7 gal/min) and allowed to run for a 2-hour period. Core exit thermocouple maps were run periodically to determine if a nonuniform boron distribution would develop in the core. Along with the T/C maps, the incore movable detectors (6) were positioned in the core at varying radial and axial positions.

The time delay from the initiation of the boron addition until the negative reactivity effects were observed in the core was approximately the same as in forced circulation (4-5 minutes).

The traces from the incore detectors showed occasional indications of a slightly nonuniform distribution, but for the most part the flux levels recorded by the detectors trended consistently. The core exit thermocouple maps showed no indication of nonuniform distribution as the exit temperature distribution was slightly better than in the full flow case, as was indicated in previous tests.

After sampling the system to assure adequate mixing, a slow cooldown was started and again thermocouple maps were taken periodically to monitor temperature distributions during the cooldown. The overall temperature distribution remained very uniform throughout the cooldown with some indications of a slightly increased radial tilt. The thermocouple calculated tilt is not considered extremely accurate, but the trend of the tilt from 550°F to 450°F should be a relatively reliable indication of the direction of changes.

During the cooldown, temperatures in the upper head, as indicated by the upper head thermocouples, were monitored closely to determine if the upper head temperatures would drop with system temperature under natural circulation. The upper head temperatures lagged the core exit and hot leg temperatures but followed the cooldown very well, indicating that some natural circulation flow was reaching the upper head region.

The test was successfully completed, and all test objectives were met.

SUMMARY

All tests were conducted in substantial agreement with the test procedures as approved by the NRC staff prior to the test program. All tests were conducted in accordance with the requirements of low-power testing license DPR-77 and its appended Technical Specifications.

In summary, the special low-power test program conducted at Sequoyah Unit 1 satisfies all requirements of item I.G.1 of the TMI Task Action Plan, NUREG-0660. This conclusion is based on the following:

1. All operating crews received adequate training during the program by participating in each test except 6, 8, and 9A, which were deemed to have little training value. The three operators who were not available during the test program on Unit 1 were trained on the Sequoyah simulator.

2. Meaningful information was obtained on plant response to a variety of abnormal conditions.

3. At all times during the tests, the plant was under complete control and responded predictably.

4. Acceptance criteria for each test as specified in the test procedure were met.

II. Siting and Design

II.B.1 Reactor Coolant System Vents

POSITION

Provide a description of the design of reactor coolant system and reactor vessel head high point vents that are remotely operable from the control room and supporting analyses. This requirement shall be met before issuance of a full-power license. See letter of September 27 and November 9, 1979. (See Section 22.3 II-B.1 for dated requirements position, discussed herein.)

DISCUSSION AND CONCLUSIONS

By letter dated January 11, 1980, and as supplemented by letters dated May 8 and July 8, 1980, TVA provided its conceptual design for the TMI Task Action Plan requirement II.B.1 to install reactor coolant system vents. TVA has designed the vent system to be remotely controlled and monitored. TVA has committed that the design will be safety grade, seismically qualified, and single-failure proof. Finally, TVA has stated that the system design is to be such that a break in the vent line will be within the capacity of one charging pump makeup and will, therefore, be smaller than the definition of the smallest LOCA.

Our preliminary review of this information has concluded that this conceptual design adequately addresses the requirements of our November 9, 1979 letter on vents. However, a detailed evaluation of the design has not been completed. Some areas that will require further detail are vent system qualification to operate under accident conditions, system testability to satisfy the requirements of IEEE 279, piping design, procedural guidelines, and analyses.

Specifically, the criteria for venting initiation and termination have not been addressed. These guidelines for vent operation will address adequate core cooling and the potential for producing combustible mixtures in the containment. They must also provide methods and tests or analyses to assure adequate heat removal through the U-tubes of the steam generator. The guidelines are currently under development in a generic effort by Sequoyah's NSSS supplier, Westinghouse.

TVA and Westinghouse have concluded that the vent system should not be operated without an indication of vessel water level (vessel water level is required per TMI Task Action Plan requirement II.F.2). Components for the installation of vessel level instrumentation cannot be obtained before May 1981. TVA has proposed that the vent and level system be installed at the first refueling outage, early 1982. (NRC staff review of the vessel level system has concluded that the delay of installation beyond the January 1, 1981 deadline is acceptable given system installation at the earliest outage with sufficient time for installation. Procedural guidelines and bases will be submitted before January 1, 1981.) TVA estimated the required outage time for installation at one month for the vent system and somewhat less for the vessel level system. The earliest possible installation date is early 1981 for the vent system.

The reasoning that vent operation has a link to the vessel level indication is that venting should proceed only with a reliable means of determining both the location of noncondensibles (e.g., reactor vessel head) and when to terminate the venting. The venting operation should be controlled and monitored to assure no resultant (or additional) core damage due to loss of inventory. Therefore, to assure core cooling, Westinghouse has concluded that a direct, reliable indication of vessel level is needed to conduct the venting operation.

While TVA and Westinghouse have indicated that the vessel level instrumentation would be needed under all foreseen scenarios to operate the RCS vents, they did not preclude the potential for other scenarios where venting without vessel level may be desirable. However, it is our judgment that the extension of time would not significantly affect reactor safety.

We concur with the TVA and Westinghouse conclusion that reactor vessel level is important in the initiation and control of venting. However, we will require that procedural guidelines and analytical bases be submitted to us by January 1, 1981, and that the vent system be installed and functional before or at the early 1982 refueling outage, consistent with scheduled or forced outages which could accommodate vent installation.

On the forgoing bases, we conclude that the applicant has provided an acceptable description of the vent system conceptual design for full power in accordance with NUREG-0694, but that further detailed review will be necessary as outlined above.

II.B.2 Plant Shielding

POSITION

Provide (1) a radiation and shielding design review that identifies the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by radiation during operations following an accident resulting in a degraded core, and (2) a description of the types of corrective actions needed to assure adequate access to vital areas and protection of safety equipment.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.6b, and letters of September 27 and November 9, 1979.) (See 22.3-II-B.2 for dated requirements position, discussed herein.)

DISCUSSION AND CONCLUSIONS

By letters dated 11/21/79, 1/11/80, and 6/13/80, TVA has submitted commitments and documentation of actions to be taken at Sequoyah to implement short-term lessons learned items in NUREG-0578.

The Sequoyah radiation and shielding design review used source terms and criteria as contained in Regulatory Guides 1.4, 1.7, Technical Information Document (TID) 14844, and General Design Criteria (GDC) 19 of Appendix A to 10 CFR 50. The plant was designed so that access is not required outside the main control room for 30 days after an accident, except for limited access to the shutdown board room and structures away from the main complex. Areas evaluated as vital areas included the control room, technical support center, normal plant sampling station, and shutdown board room outside the main control room. Post-accident doses for the control room and the technical support center have been determined to be well within GDC 19 criteria, totaling 0.18 rem for the 30-day post-accident period. Brief access of a few minutes to the shutdown board room when required meets GDC 19 through access/stay-time restrictions.

The TVA analysis determined that the calculated dose in the control building habitability zone is due almost entirely to noble gas airborne radioactivity in that zone and in neighboring spaces due to the introduction of filtered outside air used for maintaining a pressurized condition in the habitability zone. Similarly, the access restriction in the shutdown board room is due primarily to airborne noble gas radioactivity. In its June 13, 1980 design review response, TVA has committed to an accessible post-accident sampling system.

A local TVA analysis code, "STP," has been used to determine source terms and activity transports and has provided results similar to other standard analysis codes. Two approaches were analyzed for Sequoyah; a large break LOCA where dilution of the reactor coolant was considered in the source term, and a "non-mechanistic" accident where fission product release was assumed to occur 3.6 hours after shutdown, with maximized auxiliary system involvement in accident effects. As a result of its review, TVA mapped and classified the plant areas into ten zones based on calculated dose rates from sources at 0.5 hour after a postulated accident. Also, rates beyond 30 days were calculated for these zones. Areas and systems evaluated as sources within the large LOCA and "non-mechanistic" criteria included the residual heat removal pump room and pipe chases, heat exchangers, the boron injection tank, seal water heat exchangers, the volume control tank, the letdown heat exchanger, the containment spray pump rooms, and the safety injection pump room. With the exception of shielding which may be required for post-accident sampling systems, no design changes requiring additional shielding have been identified for Sequoyah as a result of the TVA evaluation.

Shielding design reviews for Sequoyah have been completed. In a letter from TVA dated July 25, 1980, planned modifications for additional shielding installation for the primary sampling area have committed to be complete as a dated requirement by January 1, 1981, as required by NUREG-0694. (See 22.3.II.B.2 of this supplement.) Onsite verification of these shielding modifications will be reviewed during routine inspections.

II.B.3 Post-Accident Sampling

POSITION

Provide (1) a design and operational review of the capability to promptly obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples under degraded core accident conditions without excessive exposure, (2) a description of the types of corrective actions needed to provide this capability, and (3) procedures for obtaining and analyzing these samples with the existing equipment.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.8a, and letters of September 27 and November 9, 1979.) (See 22.3-II.B.3 for dated requirements position, discussed herein.)

DISCUSSION AND CONCLUSION

TVA has completed the above full-power requirements for this item, with respect to dated requirements (see 22.3.II.B.C). The licensee has estimated installation of the improved post-accident sampling system to be complete by January 1982 in lieu of the dated requirements date of January 1, 1981. Until the improved system can be installed, the licensee will continue to use the interim procedure for sampling and analysis. We find this acceptable.

In a letter dated July 25, 1980, TVA has committed to procure and install equipment and to implement the relevant procedures for operation of the equipment necessary to comply with the NRC staff's criteria, as set forth in NUREG-0578, in the letter of November 9, 1979, and in NUREG-0694. The staff finds the described equipment and procedures to be in compliance with the staff's criteria. TVA projects January 1982 as the date for installation of certain equipment necessary for safe operation of the improved post-accident sampling system. This may be dependent on delivery of components to the vendor by subcontractors.

The staff further finds that the dates scheduled by TVA for completion of actions show reasonable effort and intent on the part of TVA to comply with the staff's projected completion dates and are therefore acceptable.

II.B 4 Training for Mitigating Core Damage

POSITION

Complete the training of all operating personnel in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program shall include the following topics.

Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

Excore Nuclear Instrumentation (NIS)

1. Use of NIS for determination of void information; void location basis for NIS response as a function of core temperatures and density changes.

Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures:

   a. Determination of pressurizer level if all level transmitters fail.

   b. Determination of letdown flow with a clogged filter (low flow).

   c. Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primary water; time to failure.

Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector r . (over-ranged detector); expected accuracy of detectors at different loc ions; use of detectors to determine extent of core damage.

2. Methods c determining dose rate inside containment from measurements taken outs containment.

Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

DISCUSSION AND CONCLUSIONS

TVA has a training program that meets all the requirements stated above. This training program submitted on July 22, 1979, has been completed for all currently licensed Sequoyah Unit 1 personnel.

The program "Training for Mitigation Core Damage" was developed by TVA to ensure that all licensed operating employees are properly trained to use information available through installed plant systems to recognize, control, and mitigate an accident in which the core is severely damaged. This training supplements the existing training program and consists of 14 hours of classroom instruction followed by a 2-hour examination at the conclusion of the program.

Based on the foregoing, we have concluded that the Sequoyah Nuclear Plant has provided adequate training of all licensed operating personnel for Unit 1 in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

II.B.7 Analysis of Hydrogen Control

POSITION

Reach a decision on the immediate requirements, if any, for hydrogen control in small containments, and apply, as appropriate, to new OLs pending completion of the degraded core rulemaking in II.B.8 of the Action Plan.

DISCUSSION AND CONCLUSIONS

The generation and release of substantial amounts of hydrogen into the Sequoyah containment (for example, from a zirconium-water reaction like that which occurred at TMI-2) could, under certain assumptions, lead to containment failure. By contrast, a hydrogen release similar to the TMI-2 release in a conventional, large "dry" containment would probably not lead to containment failure. It is therefore necessary to consider whether scenarios leading to containment failure in ice condenser plants are sufficiently likely as to warrant additional protective plant features.

This subject has been discussed previously in the TMI Action Plan, with the ACRS, and in recent Commission papers and briefings (SECY 80-107 and its supplements). As previously stated, our conclusion is that the likelihood of a degraded core event that would produce large amounts of hydrogen has been made acceptably low such that no additional design features for ice-condenser containment plants are required pending completion of the degraded core rulemaking on this subject.

In connection with the recent licensing action on Sequoyah, the addition of a system of hydrogen igniters is an added measure of risk reduction. The hydrogen igniters, which could be activated on demand, would cause hydrogen to burn as it is released. Such burning would likely take place where it is released; namely, in the lower containment volume. Such controlled ignition would result in energy absorption by the ice itself (passive heat removal mechanism) and the upper compartment sprays (active heat removal system). Preliminary calculations by TVA and by the NRC indicate that such an igniter system would result in a reduction in the peak pressure in containment such that for most postulated accident sequences, the containment pressure would not exceed yield stress of the containment. We therefore believe this system offers considerable potentials for post-accident containment pressure reduction and presently have it under active review.

Our safety review of the igniter system will be completed in December 1980. Our safety review will focus on an assessment of the potential risk improvement to ensure the installation and use of igniters would not result in a decrease in safety margin (e.g., a failure mode occasioned by a postulated local denotation). By license condition we will require that TVA submit the necessary information for our safety review. In support of our review, the NRC is sponsoring confirmatory studies, and we and our consultants (BNL, BCL, Sandia, and LLNL) are doing independent calculations and experiments. Pending completion of the staff's review in December 1980, we will not authorize use of the igniter system.

In the period until the staff evaluation is completed in December 1980, TVA proposes to continue power escalation testing, up to 100% with routine power production to start after the 100% test mode (nominally October 1980). We have concluded that operation without igniters is acceptable during this period because of the acceptably low likelihood of a large hydrogen release from an accident during this short period and taking into consideration the lessons learned, and the associated safety improvements, flowing from the TMI-2 accident.

Upon completion of our review, we will issue an NRC supplemental Safety Evaluation Report. If our evaluation is favorable, the operation of the igniters in accordance with special NRC-approved procedures will be authorized. If the evaluation concludes that igniters are found unacceptable, we will continue to work with TVA on other suitable alternatives. These alternatives are the subject of a long-range program that TVA is conducting and include halon systems and a water fog system.

In addition, during the interim period of time until our evaluation is completed, we will require more stringent technical specifications on containment and core heat removal systems. Both the decay heat removal system in the recirculation mode and the containment spray system would be useful in removing excess heat generated by a zirconium-water reaction with subsequent burning of hydrogen. These additional more stringent technical specifications will provide additional safety margins during this period. These additional technical specification requirements for these systems will be set forth in Technical Specifications 3.4.1.3 and 3.6.2.1, as shown in the end of this section.

In summary, we have concluded that:

1. In the interim, the Sequoyah facility should be permitted to operate without any additional post-accident hydrogen control features.

2. The TVA proposed igniter system can be installed, but not authorized to operate, pending completion of the staff's safety review for igniters, scheduled for December 1980.

3. The staff's preliminary review indicates considerable promise that the igniter system will be found to be able to safely perform its function.

4. More stringent technical specifications will be required and provide additional safety margins during the interim.

5. TVA will be required to submit additional information supporting the distributed ignition system within 90 days.

Therefore we conclude that, with respect to H_2 control, operation should be authorized at full power pending completion of our safety review of the TVA proposal for a distributed ignition system. A supplement to this SER will be issued upon completion of this evaluation.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION (a)

3.4.1.3 a. At least three of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

4. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,

5. Residual Heat Removal Loop A,**

6. Residual Heat Removal Loop B.**

b. At least two of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

**The normal or emergency power source may be inoperable in MODE 5.

***All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided: 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

(a)These more stringent requirements shall remain in effect pending resolution of the hydrogen control matter.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION (a)

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 36 hours or be in COLD SHUTDOWN within the next 30 hours.

SUP .LANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 140 psig when tested pursuant to Specification 4.0.5.

c. At least once per 18 months during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

2. Verifying that each spray pump starts automtically on a Containment Pressure--High-High test signal.

d. \t least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

[a] These more stringent requirements shall remain in effect pending resolution of the hydrogen control matter.

II.B.8 Rulemaking Proceeding on Degraded-Core Accidents

POSITION

Reach a decision on the immediate requirements, if any, for hydrogen control in small containments and apply, as appropriate, to new OLs pending completion of the degraded core rulemaking in II.B.8 of the Action Plan.

Issue an advance notice of rulemaking or requirements for design and other features for accidents involving severely damaged cores.

These actions shall be completed before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

The first steps in the resolution of item II.B.8 of the TMI Action Plan include the issuance of an interim rule and an advance notice of rulemaking. The advance notice has been brought before the Commission (SECY-80-357, July 29, 1980). The interim rule has been prepared for amending Section 50.44 of 10 CFR, Part 50 and is expected to be ready for Commission consideration during August 1980. The interim rule, in summary, required the inerting of all Mark I and Mark II BWR containments and requires operation of (or applicants for) PWR plants or Mark III BWRs to study various methods of controlling hydrogen in substantially greater amounts than current designs provide for. Further, it requires implementation of TMI-2 lessons learned pertaining to (1) high point vents, (2) protection of safety equipment and vital areas, (3) in-plant iodine instrumentation, (4) post-accident sampling, (5) leakage integrity outside containment, (6) accident monitoring instrumentation, (7) detection of inadequate core cooling, and (8) training to mitigate degraded core accidents. In addition, the rule requires the capability for the installation of hydrogen recombiner systems at all light-water power reactors that currently require purging for hydrogen control following a LOCA. A public announcement is planned that would allow 30 days for comment.

In addition to the efforts related to rulemaking, the staff has requested that a Commission-sponsored research program be initiated to investigate the effects of degraded/melted core accidents for generic LWR plant designs and to investigate various safety systems for mitigating the effects of such accidents. As a part of this safety research, we have identified the evaluation of hydrogen control systems for ice condenser and BWR Mark III containments as priority items. Additionally, the staff will seek assistance to evaluate the effectiveness of distributed ignition sources within containment on an expedited basis; i.e., within 3 months. The use of ignitors within containment is currently regarded as a promising short-term hydrogen control device which would be adapted to current plant designs. The staff will, however, evaluate a spectrum of mitigation techniques to control hydrogen and reduce the impact of severely degraded core accidents as part of the safety research program discussed above.

II.E.1.1 Auxiliary Feedwater System Reliability Evaluation

POSITION

(1) Provide a simplified auxiliary feedwater system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFWS failure following a main feedwater transient, with particular emphasis on potential failures resulting from human errors, common causes single-point vulnerability, and test and maintenance outage.

(2) Provide an evaluation of the AFWS using the acceptance criteria of Standard Review Plan Section 10.4.9.

(3) Describe the design basis accident and transients and corresponding acceptance criteria for the AFWS.

(4) Based on the analyses performed, modify the AFWS, as necessary.

These requirements shall be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

I. Introduction and Background

In a letter dated March 10, 1980, our requirements regarding the Sequoyah Auxiliary Feedwater System were forwarded to the Tennessee Valley Authority (TVA). TVA provided responses in letters dated January 25, 1980, April 15, 1980, and May 1, 1980.

The following paragraphs present the results of our evaluation of the information provided by TVA to meet our requirements.

II. Implementation of Our Recommendations

A. Short-Term Recommendations

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFWS pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent ACTION time should be as required in current Technical Specifications; i.e., 72 hours and 12 hours, respectively.

In response, the licensee indicated in a letter dated January 25, 1980, that Sequoyah Appendix A Technical Specification 3.7.1.2 applies. This Specification limits the plant operation with one

AFWS train out of service to 72 hours and the subsequent ACTION time to 12 hours. We conclude that Technical Specification 3.7.1.2 satisfies Recommend   ı GS-1 and is therefore acceptable.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFWS pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFWS flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. (See the discussion below on Recommendation GL-2 for the long-term resolution of this concern.)

   In response to this recommendation, the licensee indicated in a letter dated January 25, 1980, that there is no single valve or multiple valves in series in the AFWS pump suction piping and single valves or multiple valves in series that can defeat the system. Alternate water sources to the pump suctions do not share the same flow path with any va ves in the normal water supply lines. In addition, the AFWS suction will automatically align to an alternate water source (essential raw cooling water) on low suction pressure in each AFWS pump. We have reviewed the licensee's response and conclude that the existing design meets this recommendation and is therefore acceptable.

3. Recommendation GS-3 - The licensee should verify that the AFWS will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFWS flow, the licensee should provide sufficient information to demonstrate that the required initial AFWS flow will not result in plant damage to water hammer.

   The licensee has stated that it throttles AFWS flow to avoid water hammer. The licensee was requested to reexamine the practice of throttling AFWS flow to avoid water hammer.

   In response to Recommendation GS-3, the licensee in a letter dated January 25, 1980, indicated that on automatic start the AFWS will deliver full flow until normal water level is established in the steam generators. AFWS flow will not be prevented or reduced to avoid water hammer. Instead, the steam generator feedwater ring headers in Sequoyah 1 and 2 have been modified by TVA to minimize the possibility of a water hammer.

We requested TVA to provide the design basis for AFWS flow requirements. This information was provided by TVA in a letter dated January 25, 1980. We have reviewed this response and have concluded that the design basis for the Sequoyah 1 and 2 AFWS flow requirements are acceptable.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFWS supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - Primary water supply is not initially available. The procedures should include any operator actions required to protect the AFWS system pumps against self-damage before water flow is initiated; and,

   - Primary water supply is being depleted. The procedure should provide for transfer to the alternate water sources prior to draining of the primary water supply.

   In response to this recommendation, TVA indicated in a letter dated January 25, 1980, that each AFWS pump has its own safety-grade instrumentation that will sense low pump suction pressure prior to draining of the normal water source and will automatically align a safety-grade alternate water source to the pump. Isolation from the primary water source also occurs automatically by closure of a check valve in each pump suction line due to back pressure in the valve upon alignment of the qualified alternate water source. This qualified alternate water source is the essential raw cooling water. We conclude that the above automatic features of TVA's Sequoyah 1 and 2 design adequately address Recommendation GS-4 and that additional emergency procedures for operator actions are not required to assure this transfer to an alternate source of AFWS supply.

5. Recommendation GS-5 - The as-built plant should be capable of providing the required AFWS flow for at least two hours from one AFWS pump train, independent of any alternating current power source. If manual AFWS initiation or flow control is required following a complete loss of a.c. power, emergency procedures should be established for manually initiating and controlling the AFWS under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on a.c. power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures

should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all a.c. power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in a manual on-off mode until a.c. power is restored. Adequate lighting powered by d.c. sources and communications at local stations should also be provided if manual initiation and control of the AFWS is needed. (See the discussions below on Recommendation GL-3 for the longer term resolution of this concern.)

In response to this recommendation, TVA indicated in a letter dated January 25, 1980, that the turbine-driven pump can run for two hours using only battery power for control and a battery-powered room fan to remove heat from the pump room.

The licensee further indicated that emergency procedures have been established to cover the event of loss of all a.c. power sources. We conclude that the provisions available in the existing AFWS in Sequoyah already meet the requirements outlined in Recommendation GS-5 and are therefore acceptable.

6. Recommendation GS-6 - The licensee should confirm flow path availability of an AFWS flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFWS valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFWS water source to the steam generators. The flow test should be conducted with AFWS valves in their normal alignment.

In a letter dated January 25, 1980, TVA stated that procedures are in place which require the operator to determine if AFWS valves are properly aligned when performing Technical Specification surveillance (testing or maintenance) and when performing any maintenance. The licensee does not require a second operator to directly verify AFWS valve alignment. Instead, the Technical Specifications require that AFWS valve alignment be verified every 7 days. Additionally, each Sequoyah unit has a status monitoring system with control board indication lights. The status monitoring system automatically checks AFWS valve alignment and alarms if the alignment is not correct.

The licensee requires an AFWS flow test to be performed with the AFWS valves aligned for emergency operation. During any reactor startup, until steam generators are filled, the AFWS motor-driven pumps are used to supply water from the condensate storage tank to the four steam generators. The AFWS automatic level control system is used to maintain steam generator level until the reactor is at a power level (approximately 5% power) sufficient to transfer to the main feedwater pumps. During AFWS supply to each steam generator, AFWS flow and steam generator level must be monitored to assure that adequate flow criteria have been met for each steam generator and to assure adequate performance of each AFWS motor-driven pump. We have reviewed TVA's response and conclude that Sequoyah 1 and 2 meet the requirements of this recommendation.

7. Recommendation GS-7 - The licensee should verify that the automatic start AFWS signals and associated circuitry are safety-grade. If this cannot be verified, the AFWS automatic initiation system should be modified in the short term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety- grade requirements as indicated in Recommendation GL-5. (See the discussion below on GL-5.)

   a. The design should provide for the automatic initiation of the AFWS flow.

   b. The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of AFWS function.

   c. Testability of the initiation signals and circuits shall be a feature of the design.

   d. The initiation signals and circuits should be powered from the emergency buses.

   e. Manual capability to initiate the AFWS from the control should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

   f. The a.c. motor-driven AFWS pumps and valves should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

g. The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In response to this recommendation, TVA stated in a letter dated January 25, 1980, that the Sequoyah AFWS is designed for automatic start. There are three safety-grade automatic start modes provided for the AFWS; loss of offsite power, safety injection actuation, and low-low steam generator level. We have reviewed TVA's response and conclude that Sequoyah 1 and 2 meet the requirements of this recommendation.

8. Recommendation GS-8 - This recommendation does not apply to Sequoyah as TVA stated in a letter dated January 25, 1980, that the Sequoyah AFW system is already designed for automatic start.

B. Additional Short-Term Recommendations

1. Recommendation - The licensee should provide redundant-level indication and low-level alarms in the control room for the AFWS primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low-pump suction pressure condition from occurring. The low-level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

As stated above under Recommendation GS-4, TVA has indicated that its main line of defense against low pump suction pressure is an automatic transfer to an alternate source of water. Each AFWS pump has its own safety-grade instrumentation that will sense low pump suction pressure and automatically switch the pump suction to the alternate source. Additionally, TVA has stated that there is a level indicator in the main control room for each condensate storage tank. Level alarms for each tank are actuated in the main control room for both "low" level and "low-low" level. The "low-low" level alarm will warn the operator of imminent tank emptying and will occur when 2.5 feet of water remains in the tank. This amount of water is sufficient to supply two motor-driven pumps and the turbine-driven pumps and the turbine-driven pump at full flow, 1760 gpm, for 176.6 minutes.

We have reviewed TVA's response and conclude that Sequoyah meets the requirements of this recommendation and is therefore acceptable.

2. Recommendation - (This recommendation has been revised from the original recommendation in NUREG-0611). The licensee should perform a 48-hour endurance test on all AFWS pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 48-hour pump run, the pumps should be shut down and cooled down and then restarted and run for 1 hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

   In response to this recommendation, TVA by letters dated January 25, 1980, and May 1, 1980, indicated that it will perform the recommended AFW pump tests, if they have not been previously conducted (TVA is presently reviewing existing records), prior to exceeding 5% of full power. TVA further indicated that a test with a summary of the test conditions and the results of the tests will be provided within 30 days after all tests are completed.

   Based on the above commitment from TVA, we conclude that the response to this recommendation is acceptable. However, we intend to evaluate the AFW pump test results to confirm that the Sequoyah AFWS pumps are acceptable. If the test results are not acceptable to NRC, we will then require modifications and will issue a safety evaluation regarding the tests and modifications.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   > "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

   This is a dated requirement which must be completed by January 1, 1981. Our evaluation of the Auxiliary Feedwater Indication (2.1.7.b - NUREG-0578) regarding the ability of the design to satisfy the control-grade requirements specified in the NUREG position and clarifications was presented in Part II of SER Supplement No. 1.

TVA has indicated that while the flow indication has not been classed as safety grade, it was the same type of transmitters that are used in other safety-grade circuits. The transmitters are mounted on two separate, seismically qualified panels and powered from power sources connected to the emergency power system. The cables are in low-level signal trays and are kept separate from all power cables. The requirements of this recommendation must be implemented by January 1, 1981. We will evaluate this design regarding its ability to satisfy the safety-grade requirements in time to allow TVA to implement any design modifications by the January 1, 1981 date.

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room, this operator would realign the valves in the AFWS train from the test mode to the operational mode alignment.

   In response to this recommendation, TVA by letter dated January 25, 1980, indicated that there are three AFW trains. During the periodic tests of the AFWS, only one flow control valve in the AFWS train being tested would be affected; there would still be two AFW trains available. As a result of the licensee's testing lineup, we conclude that this recommendation is not applicable to Sequoyah.

C. Long-Term NUREG-064 Recommendations

1. Recommendation GL-1 - This recommendation does not apply to Sequoyah as TVA stated in a letter dated January 25, 1980, that the Sequoyah AFW system is already designed for automatic start.

2. Recommendation GL-2 - Licensees with plant designs in which all (primary and alternate) water supplies to the AFWS pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   In response to this recommendation, TVA by letter dated January 25, 1980, indicated that the alternate water supplies to the AFWS pump suctions do not share the same flow path with any valves in the primary water supply lines. We have reviewed TVA's response to Recommendation GL-2 and conclude that it is acceptable.

3. Recommendation GL-3 - At least one AFWS pump and its associated flow path and essential instrumentation should automatically initiate AFWS flow and be capable of being operated independently of any a.c. power source for at least 2 hours. Conversion of d.c. power to a.c. power is acceptable.

   In response to this recommendation, TVA indicated in a letter dated January 25, 1980, that the turbine-driven AFWS pump is capable of operating for 2 hours without a.c. power, using only battery power for control and a d.c.-powered room fan to remove heat from the pump room. One potential concern has arisen regarding the AFWS level control during a station blackout. During such an event, steam generator level control would be accomplished by manipulation of the level control valves using air from accumulators near the valves and some d.c. power. The steam generator level control valves using air from accumulators were tested under the Low Power Test Program (Test No. 7), and they were found to be acceptable.

4. Recommendation GL-4 - Licensees having plants with unprotected normal AFWS water supplies should evaluate the design of their AFWSs to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure, or upgrading the normal source of water to meet the seismic Category I and tornado protection requirements.

   In response to this recommendation, TVA by letter dated January 25, 1980, stated that the Sequoyah AFWS design already provides for automatic transfer to an alternate water source on low suction pressure at the intake to the pumps. The alternate water source is the essential raw cooling water.

We have reviewed TVA's response and conclude that it meets Recommendation GL-4 and therefore is acceptable.

5. Recommendation GL-5 - The licensee should upgrade the AFWS automatic initiation signals and circuits to meet safety-grade requirements.

   In response to this recommendation, TVA indicated in a letter dated January 25, 1980, that the present AFWS automatic initiation signals are safety-grade. We will review this aspect of the Sequoyah 1 and 2 design in detail, and our evaluation will be contained in a supplement to this SER. Implementation of modifications, if appropriate, will be required as a dated item by January 1, 1981. (See 22.3 II.E.1.1)

III. Conclusions

On the basis of the above considerations, we have concluded that the Sequoyah auxiliary feedwater system meets the Section II.E.1.1 full-power requirements of NUREG-0694 and therefore is acceptable.

II.E.3.1 Emergency Power For Pressurizer Heaters

POSITION

Install the capability to supply from emergency power buses a sufficient number of pressurizer heaters and associated controls to establish and maintain natural circulation in hot standby conditions.

The requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.1, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

Westinghouse Owners group analysis has determined that to maintain natural circulation in a four-loop plant with a pressurizer volume of 1800 cubic feet, a heater of 150-kW capacity should be available within four hours. The Sequoyah 1 and 2 pressurizer heaters consist of four 485-kW heaters per unit. The heaters are powered and controlled from redundant Class IE sources (two per division). The motive and control power interfaces with the emergency buses are qualified in accordance with safety-grade requirements. All four heaters will trip on a safety-injection signal when in the normal mode. After safety-injection reset and level recovery in the pressurizer, one heater will operate automatically. The other heaters will not come on automatically but are manually activated. In the event of a loss of offsite power and safety-injection signal, two redundant heaters can be manually activated by hand switches in the main control room and connected to the diesel generator power source 90 seconds after emergency power becomes available. Procedures are in force to instruct the operator in manual use of the pressurizer heaters to establish and maintain natural circulation.

We have reviewed the Sequoyah 1 and 2 design for emergency power for pressurizer heaters. Based on our review, we conclude that the existing design for emergency power for pressurizer heaters meets the Section II.E.3.1 full-power requirements of NUREG-0694 and therefore is acceptable.

II.E.4.1 Dedicated Penetrations

This item has been resolved as not applicable to the Sequoyah plant. See section II.E.4.1 of Supplement No. 1 to the Sequoyah SER for details. See also Section 22.3-II.E.4.1 of this supplement.

II.E.4.2 Containment Isolation Dependability

POSITION

Provide (1) containment isolation on diverse signals, such as containment pressure or ECCS actuation, (2) automatic isolation of nonessential systems (including the bases for specifying the nonessential systems), (3) no automatic reopening of containment isolation valves when the isolation signal is reset.

These requirements shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.4 (Ref. 4), and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24).

DISCUSSION

The Sequoyah containment isolation system provides diversity in the parameters sensed for the initiation of containment isolation, and the isolation signals satisfy safety-grade requirements. The isolation system is designed to operate in two stages: Phase A and Phase B. Phase A isolates all nonessential systems and Phase B isolates all "desirable" systems (see below for definition of this categorization).

Phase A isolation can be initiated manually and is initiated by automatic or manual safety injection (SI) actuation. The SI is derived from (1) high steam line flow coincident with low steamline pressure or low-low average reactor coolant temperature, (2) high steam line differential pressure between loops, (3) low pressurizer pressure, or (4) high containment pressure. Phase B isolation can be initiated manually or automatically on a high-high containment pressure signal. In addition, isolation valves in the primary containment ventilation system actuate on manual initiation of Phase A, Phase B, or SI and automatically on SI or high radiation signals (see Table 22.2 II.E.4.2-1).

TVA has performed an evaluation of essential and nonessential systems. The containment isolation system is designed to prevent the release of radioactive material to the environment after an accident while ensuring that systems important for post-accident mitigation are operational. Table 22.2 II.E.4.2-1 shows the different isolation signals and the parameters that initiate each signal.

Isolation is provided on the following three levels, as classified by the applicant:

1. Nonessential systems - These systems are not required for post-accident mitigation and are isolated automatically upon receipt of a Phase A isolation signal.

2. Essential systems - This group consists of the emergency core cooling systems, the containment spray system, and post-accident hydrogen monitors. These

systems are not automatically isolated in the event of an accident. Remote manual valves are provided to permit isolation of these lines from the main control room if necessary.

3. Desirable systems - Systems that, while not required, significantly increase the plant's ability to cope with a small steam line break or LOCA. The systems are isolated automatically upon the receipt of a Phase B isolation signal (Table 22.2 II E.4.2-1). The systems falling into this category are the essential raw cooling water to the reactor coolant pumps (RCP) and containment coolers, component cooling water to the RCPs, and control air.

Each line penetrating primary containment has been reviewed by TVA to ensure that (1) isolation of the line was based on its need to be in service post-accident and (2) that each containment isolation valve received the proper isolation signal.

Certain systems, while not engineered safety feature (ESF) systems required by design for accident mitigation, may nonetheless be considered important to post-accident plant safety and valuable in accident mitigation. Such systems may be deemed essential insofar as not requiring diversity in the parameters sensed for the initiation of containment isolation. The "desirable" systems listed above fall into this category, as will be shown below.

Component Cooling Water to the Reactor Coolant Pumps

The staff has determined that post-accident operation of the reactor coolant pumps is highly desirable. Component cooling water, through the thermal barrier heat exchanger, cools the reactor pumps seals. Although component cooling water to the reactor coolant pumps is not a required system for the safe shutdown of the plant, isolation of this system could cause reactor coolant pump seal damage and subsequent loss of reactor coolant, and also loss of reactor coolant pump operability. Therefore, diverse automatic isolation of this system is not required.

This system is automatically isolated by a Phase B isolation signal.

Reactor Coolant Pump Motor Cooling Water

This system provides essential raw cooling water to the reactor coolant pump motor. As discussed above for the component cooling water to the reactor coolant pumps, post-accident operation of this system is desirable in order to maintain operation of the reactor coolant pumps, and therefore diverse automatic isolation of this system is not required.

The reactor coolant pump motor cooling water supply and return lines ar automatically isolated by a Phase B isolation signal.

Essential Raw Cooling Water to Containment Coolers

Although not required for the safe shutdown of the plant, the operation of normal containment coolers would be desirable during a small steam line break or LOCA.

They would cool the containment atmosphere before the containment sprays or ice condenser begin operation. Essential raw cooling water (ERCW) is required for containment cooler operation, and so diverse, automatic isolation of ERCW to the containment coolers is not required. Automatic isolation would occur upon the receipt of a Phase B isolation signal.

Control Air

Control air may provide several desirable post-accident functions, including controlling various valves inside containment. Although not required for safe plant shutdown, these functions are desirable, and so diverse, automatic isolation of control air is not required. Automatic isolation would occur upon the receipt of a Phase B isolation signal.

Post-Accident Hydrogen Monitors

The following is justification for including this system in Category 2 (essential systems).

The post-accident hydrogen monitors are used for continuous sampling of the containment atmosphere to measure hydrogen concentration after an accident. These are essential systems which must operate to provide a continuous indication to the control room of hydrogen concentrations in the containment atmosphere. Thus, automatic isolation is not required.

Manual isolation capability is provided. Each monitor and its sample and return line is a closed system outside of containment, and each line penetrating containment has a remote manual isolation valve in the line, inside containment. Also, the systems are located inside the secondary containment (annulus between the primary containment and the shield building), so that post-accident leakage from the systems would be contained and processed by the emergency gas treatment system before release. These isolation provisions are acceptable.

The isolation of ventilation lines and lines that carry potentially radioactive fluid outside containment during power operation received special consideration by the applicant. The ventilation lines receive high radiation signals in addition to the Phase A or B isolation signals (Table 22.2 II.E.4.2-1). At present, the isolation of fluid lines that carry potentially radioactive material outside containment occurs upon the receipt of a Phase A signal. This isolation signal should preclude the type of releases of radioactive material that occurred at TMI. However, to provide an additional margin of safety as identified in the TVA's Nuclear Program Review as a result of TMI, TVA is adding radiation monitors that will automatically isolate each of these lines in the event of high radiation in the line. In a letter dated June 23, 1980, TVA has committed to have these monitors and associated isolation logic changes in the plant by May 1981.

Based on our review, we conclude that TVA is in compliance with Position 1 above.

TVA has identified all essential and nonessential systems penetrating containment. TVA has also identified certain systems as being "desirable." Based on our review, we conclude that TVA is in compliance with Position 2, above.

All nonessential systems receive automatic diverse containment isolation signals in accordance with Position 3, above. Also, all nonessential systems and "desirable" systems, without exception, are automatically isolated by a Phase B isolation signal, if not already isolated by an earlier signal. (Phase A always exists if Phase B exists.)

The containment isolation system is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. The initiating signal must be reset and each automatic valve individually opened by the operator. Resetting of an initiating signal will not cause a containment isolation valve to change position. We find this to be in accordance with Position 4, above.

CONCLUSION

Based on our review, we conclude that TVA is in full compliance with the requirements for containment isolation dependability given in section 2.1.4 of NUREG-0578 and that the full-power requirements of Item II.E.4.2 of NUREG-0694 have been met.

TABLE 22.2.II.E.4.2-1

ISOLATION SIGNALS

Phase A Initiation

Manually	-	1 of 2 hand switches, or
Manually	-	SIS switch, or
Automatically	-	SIS auto-initiation

SIS Initiation

Manually	-	1 of 2 hand switches, or
Automatically	-	on 2 out of 3 high containment pressure, or
	-	2 out of 3 logic on any of 4 sets of differential pressure between steam lines, or
	-	low pressurizer pressure on 2 out of 3 channels,
	-	coincident high steam line flow with low steam line pressure or low-low average RCS temperature. Each loop has two high flow meters. One pressure and temperature instrument are provided per loop. At least 2 of the 4 loops must reach the instrument setpoints to initiate the SIS.

Phase B Initiation

Manually	-	2 of 4 hand switches, or
Automatically	-	2 of 4 high-high containment pressure

Containment Ventilation Initiation

Manually	-	Phase A manual initiate, or
	-	Phase B manual initiate, or
	-	SIS manual initiate, or
Automatically	-	SIS auto-initiate, or
	-	high radiation 1 sensor (train A only), or
	-	high radiation 1 sensor (train B only), or
	-	high purge exhaust radiation 1 of 2 sensors

II.K.3 Final Recommendations of B&O Task Force (Item C.3.3)

POSITION

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in an annual report.

This requirement shall be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

TVA will be required to prepare a Technical Specification to ensure that all failures or challenges of the PORVs or safety valves are identified, recorded, and promptly reported to the NRC. The Technical Specification also requires documentation of all PORV or safety valve challenges.

On this basis we consider the Section II.K.3, Item C.3.3, full-power requirements of NUREG-0694 have been met.

III. EMERGENCY PRFPARATIONS AND RADIATION PROTECTION

III.A.1.1 Upgrade Emergency Preparedness

POSITION

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments) except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

This requirement shall be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

We have reviewed TVA's revised emergency plan and find that it is in substantial compliance with NUREG-0654 and meets the requirements of 10 CFR 50, Appendix E.

The basis for this finding is summarized in our Emergency Preparedness Evaluation Report and is presented in Appendix E to this report. An exercise was conducted at the Sequoyah Nuclear Plant in June 1980 and Federal observers reported that it adequately exercised the TVA, State, and local plans.

The Federal Emergency Management Agency reviewed the State and local emergency plans, and their findings and determinations are in Appendix E. Based on their findings and our evaluation, we conclude that TVA meets the full-power license requirements of Item III.A.1.1 of NUREG-0694 for Sequoyah.

III.B.2 Implementation of NRC and FEMA Responsibility

We have also concluded that TVA's revised emergency plan and TVA's commitments adequately respond to the deficiencies to be corrected for a full-power license listed in the Supplement No. 1 to the SER.

The findings and determinations made by the Federal Emergency Management Agency on the State and local emergency response plans are an attachment to Appendix E of this supplement, "Emergency Preparedness Evaluation Report." Based on the information in Appendix E of this supplement, we find that the II.B.2 requirements of NUREG-0694 are satisfied.

III.D.1.1 Primary Coolant Sources Outside Containment

POSITION

Reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels, measure actual leak rate, and establish a program to maintain leakage at as-low-as-practical levels and monitor leak rates.

This requirement shall be met before issurance of a full-power license.

DISCUSSION AND CONCLUSIONS

The staff has reviewed TVA's submittal of July 11, 1980, in which TVA provided leak testing procedures for the waste gas system and the results of the initial tests conducted under the leak reduction program for liquid and gas systems outside the containment.

Based on this review, we conclude that the Section III.D.1.1 full-power requirements of NUREG-0694 have been met.

III.D.2.4 Offsite Dose Measurements

POSITION

The NRC will place approximately 50 thermoluminescent dosimeters (TLDs) around the site in coordination with the applicant's and State's environmental monitoring program. This action shall be completed prior to issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

IE states that 41 TLDs have been placed around the plant site. A program has been established as part of the State environmental program to collect the TLDs quarterly and send them to NRC for processing.

Based on the above, we conclude that the Section II.D.2.4 full-power requirements of NUREG-0694 have been met.

III.D.3.4 Control Room Habitability

POSITION

Identify and evaluate potential hazards in the vicinity of the site as described in SRP Sections 2.2.1, 2.2.2, and 2.2.3, confirm that operators in the control room are adequately protected from these hazards and the release of radioactive gases as described in SRP Section 6.4, and, if necessary, provide the schedule for modifications to achieve compliance with SRP Section 6.4.

This requirement shall be met by issuance of a full-power license.

DISCUSSION AND CONCLUSION

In the Safety Evaluation Report for Units 1 and 2 of the Sequoyah Nuclear Plant (NUREG-0011) issued in March 1978, the staff concluded that the control room is adequately protected against accidents involving airborne radioactivity or an accidental release of chlorine. No outstanding issues concerning control room habitability were identified.

In a letter dated May 7, 1980, TVA was advised by the NRC of "Five Additional TMI-2 Related Requirements to Operating Reactors," the fifth item of which required a response to item III.D.3.4 "Control Room Habitability" of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."

Additional clarification regarding the sche e for implementation of III.D.3.4 was provided in NUREG-0694, as stated in the ove position.

In a letter dated July 24, 1980, TVA stated that

> "TVA has identified and evaluated potential hazards in the vicinity of the Sequoyah Nuclear Plant as described in Standard Review Plan (SRP) sections 2.2.1, 2.2.2, and 2.2.3. The results of these evaluations have been reported in section 2.2 of the Sequoyah Final Safety Analysis Report (FSAR). The Sequoyah operators are adequately protected from these hazards.
>
> TVA has evaluated the main control room habitability system for protection from radioactive and toxic gas releases as described in SRP sections 2.2 and 6.4. The results of these evaluations have been reported i sections 2.2., 6.4, and 15.5 of the Sequoyah FSAR. The Sequoyah operators are adequately protected from these airborne hazards."

In the letter dated July 24, 1980, TVA affirmed that it has complied with the control room habitability provisions of the NRC letter of May 7, 1980, described above.

In summary, as a result of the staff review conducted in accordance with Standard Review Plan sections 2.2.1, 2.2.2, 2.2.3, and 6.4, and Regulatory Guides 1.78 and 1.95, and General Design Criterion 19 of Appendix A to 10 CFR Part 50, the staff reaffirms, as previously reported in NUREG-0011, that the control room at Sequoyah meets our requirements for a full-power license.

On the basis of this information provided by the applicant, the staff concludes that the requirements of Section III.D.3.4 of NUREG-0694 have been met.

IV. PRACTICES AND PROCEDURES

IV.F.1 Power-Ascension Test

POSITION

IE will monitor the power-ascension test program to confirm that safety is not compromised because of the expanded startup test program and economic costs of the delay in commercial operation.

This action shall be taken during the startup and power-ascension program.

DISCUSSION

IE will monitor the power-ascension test program.

22.3 Dated Requirements

With respect to TMI-2 dated requirements, we state in NUREG-0694 that "Experience with implementation of the dated requirements on operating reactors is indicating to NRR that the January 1, 1981 deadline may be too tight in some cases to allow reasonable time for completion of the work required. This experience may prove to be the case for some of the dated requirements for NTOLs. The staff would intend to allow case-by-case exceptions to the deadlines if good cause is shown. The dated requirements are not preconditions for licensing of new plants. That is, if a completion deadline falls later than the operating license date for a new plant, then that requirement need not be met by the newly licensed plant until the completion deadline. If in the future a completion deadline fails before an operating license issuance date, then that requirement is a prerequisite for the new operating license, except when a good cause is shown for exception."

Among the factors the staff will consider in its determination of whether good cause has been shown for exceptions are problems associated with the specification, development, procurement, delivery, and installation of components and other factors beyond the control of the applicant.

In a letter dated July 25, 1980, TVA submitted a mid-year status of design and installation of Category B (dated requirement items identified in NUREG-0694) modifications and the proposed schedule for implementation of modifications at Sequoyah. TVA indicates that installation dates for some of the Category B items are after the NUREG-0694 specified implementation date of January 1, 1981. As indicated above, the staff has determined that implementation delays caused by the problems discussed above can provide a sufficient basis for finding that good cause for delaying implementation has been established. The factors applicable to each of the dated items and our conclusions concerning the acceptability of these factors are addressed in our discussion of each of the dated items.

A meeting was held on July 23, 1980, with the Virginia Electric Power Company, Public Service Electric and Gas Company, and the Tennessee Valley Authority to discuss the dated requirements and the bases for any exceptions that would be required to meet the implementation dates specified in NUREG-0694.

If good cause is established on certain items, an exception may be granted. Good cause was defined above as establishing that the applicant has made reasonable effort to complete the dated requirements and could not do so due to circumstances beyond his control, such as those discussed above.

We also require that the applicant demonstrate that extending the implementation date will not cause any significant risks to the health and safety of the public. This has been done.

The following section presents an evaluation of each of the dated requirement items, including justification for extending the implementation dates where required.

There are 15 dated requirements that should be met. TVA will meet all of these requirements except for five for which good cause has been shown which supports the staff determination that an extension to the dates given in NUREG-0694 should be allowed. These are summarized below:

Item	Title	Date (0694)	Date (TVA)
II.B.1	Reactor Coolant System Vents	Jan. 1, 1981	1/82
II.B.3	Post-Accident Sampling	Jan. 1, 1981	1/82
II.F.1(d)	Containment Radiation Monitors	Jan. 1, 1981	1/82
II.F.1(e)	Noble Gas Effluent Monitor	Jan. 1, 1981	1/82
II.F.2	Reactor Coolant Vessel Water Level	Jan. 1, 1981	1/82

The extensions beyond January 1, 1981 for the above items are based upon several factors that support good cause; i.e., procurement and installation. Backup capability in the form of either alternate hardware or procedures are available for short-term operations.

I. Operational Safety

I.A.1 Operating Personnel and Staffing

I.A.1.1 Shift Technical Advisor

POSITION

The Shift Technical Advisor shall have a technical education which is taught at the college level and is equivalent to about 60 semester hours in basic subjects of engineering and science and specific training in the design, function, arrangement, and operation of plant systems in the expected response of the plant and instruments to normal operation, transients, and accidents including multiple failures of equipment and operator errors.

This requirement shall be met by January 1, 1981. See NUREG-0578, Section 2.2.1b and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSION

In a letter dated February 7, 1980, TVA has agreed to provide this office with a description of their STA training program and their plans for requalification training by November 1, 1980. This description will indicate the level of training which the STAs will have attained by January 1, 1981. The description will also compare the licensee's STA training program with an INPO document entitled, "Nuclear Power Plant Shift Technical Advisor Recommendations for Position Descriptions, Qualifications, and Education and Training," or will demonstrate the adequacy of the licensee's alternate training requirements. We agree this will provide sufficient time for compliance with the January 1, 1981 requirement.

I.A.2.1 Immediate Upgrading of Operator and Senior Operator Training and Qualification

POSITION

Applicants for SRO license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at the specific plant) and no more than 2 years shall be academic or related technical training.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation.

Revise training program to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

DISCUSSION AND CONCLUSION

In addition to their previous experience for cold license eligibility, all licensed senior operators, commensurate with their positions, have participated in the initial fuel loading and the special low-power test program.

Applications which have been recently submitted are signed by the Director of Nuclear Power for TVA.

On July 31, 1980, TVA submitted programs which included initial training in heat transfer, fluid flow, thermodynamics, and plant transients. We conclude that TVA has satisfied the requirements of Section I.A.2.1 of NUREG-0694.

I.A.2.3 Administration of Training Programs for Licensed Operators

POSITION

Training instructors who teach systems, integrated response, transient, and simulator courses shall successfully complete an SRO examination, and instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems, and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

DISCUSSION AND CONCLUSION

There are currently five licensed SROs on the Sequoyah Training Staff at the TVA Power Production Training Center. In addition, one SRO is assigned to the plant training staff. All licensed personnel and those assigned as instructors are required to participate in the station requalification program. Based on the foregoing, we have concluded that TVA has complied with NUREG-0694 in regard to this item.

I.A.3.1 Revise Scope and Criteria for Licensing Exams

POSITION

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations. Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operation shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations (to be submitted by August 1, 1980).

DISCUSSION AND CONCLUSION

TVA has a policy which requires its licensed operator applicants to grant permission to the NRC to inform TVA management regarding the results of examination.

In the letter of July 31, 1980, TVA submitted its outline of the training in heat transfer, fluid flow, thermodynamics, and mitigation of accidents for their requalification program. Also included was the revised examination criteria for accelerated training consistent with new passing grades for issuance of licenses.

Modifications to the requalification program which revised specific reactivity control manipulations for startup, normal, abnormal, and emergency operations have been submitted.

Based on the information submitted by TVA at this time, we conclude that TVA has satisfied these requirements of Section I.A.3.1 of NUREG-0694.

I.C.1 Short-Term Accident Analysis and Procedure Revision

Analyze the design basis transients and accidents including single active failures and considering additional equipment failures and operator errors to identify appropriate and inappropriate operator actions. Based on these analyses, revise, as necessary, emergency procedures and training.

This requirement was intended to be completed in early 1980; however, some difficulty in completing this requirement has been experienced. Clarification of the scope and revision of the schedule are being developed and will be issued by July 1980. It is expected that this requirement will be coupled with Task I.C.9., Long-term Upgrading of Procedures. See NUREG-0578, Section 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSIONS

Our evaluation of this matter is addressed in Section 22.2, Item I.C.1, of this supplement.

II. Siting and Design

II.B.1 Reactor Coolant System Vents

POSITION

Install reactor coolant system and reactor vessel head high-point vents that are remotely operable from the control room.

This requirement shall be met before January 1, 1981. (See Enclosure 4 to letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSION

The staff's review of TVA's response to this position is included in the full-power requirement, reactor coolant system vents, Section 22.2, Item II.B.1 of this supplement.

Projected Completion Date

On the basis given in Section 22.2, Item II.B.1 of this supplement, the NRC staff has concluded that the delay of installation beyond the January 1, 1981 deadline to January 1, 1982 should be allowed for good cause shown.

II.B.2 Plant Shielding

POSITION

Complete modificaton to assure adequate access to vital areas and protection of safety equipment following an accident resulting in a degraded core.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.6b and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

Our evaluation of the radiation and plant shielding report which is required prior to full-power operation is presented in Section 22.2, Item II.B.2 of this report.

Planned modifications for additional shielding installation for the primary sampling area have been committed to be complete by January 1, 1981.

II.B.3 Post-Accident Sampling

POSITION

Complete corrective actions needed to provide the capability to promptly obtain and perform radioisotropic and chemical analysis of reactor coolant and containment atmosphere samples under degraded-core conditions without excessive exposure.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.8a. and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

Our evaluation of the post accident sampling system which is required prior to full-power operation is presented in Section 22.2, Item II.B.3 of this report.

The NRC staff has concluded that the delay of the installation of the improved system beyond January 1, 1981 to January 1982 should be allowed for good cause shown.

II.D.1 Relief and Safety Valve Test Requirements

POSITION

Complete tests to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met by July 1, 1981. (See NUREG-0578, Section 2.1.2, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

TVA has stated that they will participate in the EPRI/NSAC program to conduct performance testing of PWR relief and safety valves and associated piping and supports. TVA has referenced the proposed EPRI program ("Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," dated December 13, 1979) for the performance testing of these valves.

A description of the test program was provided to the NRC by EPRI in December 1979. We will review this program and schedule to ensure that the NUREG-0578 requirements are met. Preliminary discussions with EPRI also indicate that meeting the clarified requirements of NUREG-0578 is feasible.

This commitment provides adequate assurance that the requirement for performance testing of relief and safety valves will be satisfied. Our basis for accepting this commitment is, first, that the preliminary discussions with EPRI indicate that the EPRI proposed test program will meet the requirements of NUREG-0578, and second, that we will review the test programs and schedule to confirm acceptability of the program and applicability to the applicant's facility. We will report on our review of this program and associated schedule in a supplement to this evaluation.

TVA's response to the performance testing requirement for PWR relief and safety valves is acceptable. The staff will perform a detailed review of the program proposed by EPRI and of the applicability of the program to all PWRs, including Sequoyah. We will report the final results of that review in a supplement to this evaluation.

In a letter dated April 2, 1980, TVA has committed to meet these requirements by the due date of July 1, 1981.

II.E.1.2 Auxiliary Feedwater Initiation and Indication

(a) Initiation

POSITION

Upgrade, as necessary, automatic initiation of the auxiliary feedwater system to safety-grade quality.

This requirement shall be met by January 1, 1981.

DISCUSSION AND CONCLUSIONS

The staff's review of TVA's response to the auxiliary feedwater initiation requirement is included in Section 22.2, Item II.E.1.1, Paragraph II A-3 of this supplement.

We conclude that this dated requirement has already been met.

(b) Indication

POSITION

Upgrade, as necessary, the indication of auxiliary feedwater flow to each steam generator to safety-grade quality.

DISCUSSION AND CONCLUSIONS

The staff's review of TVA's response to the indication of auxiliary feedwater flow to each steam generator to safety-grade quality is included in Section 22.2, Item II.E.1.1, Paragraph II B-3 of this supplement.

We conclude that this dated requirement has already been met.

II.E.4.1 Containment Dedicated Penetrations

POSITION

Inst·ll a containment isolation system for external recombiners or purge systems for post-accident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion. This requirement shall be met before January 1, 1981.

DISCUSSION AND CONCLUSION

Our discussion and conclusion regarding the need for dedicated penetrations for hydrogen control at Sequoyah were given in Section II.E.4.1 of Supplement No. 1 to the SER for hydrogen recombiner use following an accident that results in a degraded core and a release of radioactivity to the containment, and has determined that they are now adequate. Also, there are no shielding requirements or personnel exposures involved in operating the existing recombiners since they are located inside containment and are remote manually controlled from the main control room.

Therefore, we conclude that the Sequoyah plant complies with the provisions of this position.

II.F.1 Additional Accident Monitoring Instrumentation

POSITION

Install continuous indication in the control room of the following parameters:

a. Containment pressure from -5 psig to three times the design pressure of concrete containments and four times the design pressure of steel containments;

b. Containment water level in PWRs from (1) the bottom to the top of the containment sump, and (2) the bottom of the containment to a level equivalent to 600,000 gallons of water;

c. Containment atmosphere hydrogen concentration from 0 to 10 volume percent;

d. Containment radiation up to 10^8Rad/hr;

e. Noble gas effluent from each potential release point from normal concentrations to 10^5 μCi/cc (Xe-133).

Provide capability to continuously sample and perform onsite analysis of the radionuclide and particulate effluent samples.

This instrumentation shall meet the qualification, redundancy, testability, and other design requirements of the proposed revision to Regulatory Guide 1.97.

This requirement shall be met by January 1, 1981.

DISCUSSION AND CONCLUSION

a. Containment Pressure Indication

Four qualified, continuous indications of the containment pressure are presently provided in the main control room. The 5 psig negative pressure requirement is not applicable to Sequoyah since qualified vacuum relief of the containment maintains the pressure at greater than negative 0.5 psig. The existing pressure indicators have a range of -1 to 15 psig. Redundant, continuous containment pressure indication with a range up to four times the design pressure (0 to 50 psig) of the steel containment will be provided. The monitors will be installed and operational by January 1, 1981, in accordance with TVA letter dated July 25, 1980. We conclude that TVA's response is acceptable and is in compliance with this portion of NUREG-0694.

b. Containment Water Level Indication

The floor of the reactor building serves as the sump for the containment. It is instrumented with four separate, qualified, and continuous level instruments which indicate in the main control room. The range of the instruments is from 3.75 inches above the floor up to 200 feet above the floor.

If 600,000 gallons of water were introduced into containment in addition to the fluid volume of the reactor coolant system, safety injection accumulators, and total ice melt, the containment water level would not exceed the 20 ft. range of the level instruments. A small pump suction pocket (about 120 cubic feet) in the reactor building floor serves as a collector for the recirculation piping exiting the containment and does not require qualified level instrumentation.

The normal containment equipment sump is monitored by the narrow range sump level instruments. They cover the required range. These instruments are installed and operational.

The licensee states that the wide range sump level instrument meets the applicable requirements for qualification, redundancy, and testability in accordance with Sequoyah's commitment to IEEE 323-71, which is acceptable. However, TVA has stated that the narrow range sump level instrument meets the appropriate requirements of Regulatory Guide 1.45, not Regulatory Guide 1.89. A recent discussion with a representative of TVA indicates that TVA believes this is adequate, and that the adequacy of its design will be determined by the staff under NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." This determination must be made before this design provision may be found acceptable.

c. Containment Hydrogen Indication

Redundant safety-grade hydrogen analyzers are located in the annulus between the primary containment and the shield building. These analyzers are installed and operational. These monitors provide continuous indication in the main control room. The range of these monitors is from 0 to 10 percent hydrogen concentration from negative 2 psig to positive 50 psig pressure.

The analyzers monitor the containment through stainless steel tubing coming from one point in the upper compartment and one point in the lower compartment. These lines are equipped with an isolation valve identical to those on the incoming lines. Because the analyzers are in the annulus, the accident environment for them is a temperature of 150°F and a radiation dose of 5×10^7 rads. The analyzer internals are designed to process containment atmosphere at 56 psig, 300°F, and 100% relative humidity. Hand switches, indicators, and alarms are located in the main control room. The analyzer electronics are located in the auxiliary building. The system is seismically qualified.

When the system is actuated, containment atmosphere is continuously drawn through a series of sample conditioners before entering the analyzer, including a trap, moisture separator, and filter. The atmosphere from the upper and lower compartments is mixed before entering the analyzer. As a result of the analyzer capability and the mixing afforded by the hydrogen collection system which draws from compartments within the containment lower compartment and the containment dome, a true indication will be given of the hydrogen concentration within containment. The analyzers are calibrated to measure hydrogen concentrations between zero and ten percent with an accuracy of plus or minus one-tenth of one percent.

In addition to the above information, the hydrogen analyzers meet the applicable requirements for qualification, redundancy, and testability in accordance with Sequoyah's commitment to IEEE 323-71.

We conclude that the presently installed system meets the requirements of this dated item, and is therefore already acceptable.

d. Containment Radiation

In letter of August 8, 1980, TVA has committed to provide Sequoyah with high-range in-containment monitors. Two General Atomic gamma monitors of 10^7 R/hr range (which is an acceptable alternative to 10^8 rads/hr) with continuous control room indication and recording will be provided. Power supply will be vital bust power. Sensitivity is adequate to measure low energy Xe^{133} gamma radiation. Four monitors will be located in each plant, with two each in the upper and lower containment areas in locations not protected by massive shielding. Seismic and environmental qualification per Regulatory Guides 1.100 and 1.89 will be completed before installation. Calibration will be performed during refueling periods in accordance with manufacturers' instructions. Until these monitors are operable, the current containment radiation monitor, located out of contamination in the auxiliary building, will be supplemented temporarily by a second high-range monitor to be installed outside of a containment personnel hatch. This would provide indirect accident radiation level monitoring capability for the Sequoyah containment. TVA's commitments meet the design and operation positions in NUREG-0578 regarding high-range in-containment radiation monitors but do not meet the NUREG-0578 implementation dates listed in NUREGs 0660 and 0694, Item II.F.1.

TVA has identified several reasons for the delayed commitment: acceptable and qualifiable equipment was not available in the early part of the year; General Atomic only recently offered equipment acceptable to TVA; the recently completed shielding and environmental analysis results indicated the desirability of in-containment monitoring. Thus, the possible delays are attributable to TVA's prior commitment to out-of-containment radiation monitors and the late commitment of TVA to install in-containment monitors. The revised commitment dates identify March 1981 as delivery date and the first refueling outage, now scheduled for early 1982, as the installation date. This review completes the evaluation of Item 2.1.6.b/II.F.1, with onsite verification of installation and operation to be completed during a future routine inspection.

In a letter dated July 25, 1980, TVA has committed to install adequate high-range in-containment radiation monitors at Sequoyah 1 and 2 by early 1982. This does not meet the January 1, 1981 date of NUREG-0694. TVA has proposed an alternate method of monitoring radiation levels in containment by using monitors outside containment until the in-containment monitors are installed. It is our position that TVA establish procedures to correlate the out-of-containment monitor readings with in-containment radiation levels. In a letter dated August 11, 1980, TVA agreed to provide these procedures by August 20, 1980.

Projected Completion Date

It is our position that for good cause shown, TVA should be allowed to install the monitors during the first forced or scheduled outage of sufficient length to allow installation after delivery of the monitors. TVA projects a completion date of early 1982.

e. Noble Gas Effluent

In a letter of July 18, 1980, TVA agreed to provide Sequoyah with high-range in-containment monitors. Two General Atomic gamma monitors of 10^7 R/hr range (which is an acceptable alternative to 10^8 rads/hr) with continuous control room indication and recording will be provided. Power supply will be vital bus power. Sensitivity is adequate to measure low energy Xr^{133} gamma radiation. Four monitors will be located in each plant, with two each in the upper and containment areas in locations not protected by massive shielding. Seismic and environmental qualification per Regulatory Guides 1.100 and 1.89 will be completed before installation. Calibration will be performed during refueling periods in accordance with manufacturers' instructions. Until these monitors are operable, the current containment radiation monitor, located out of contamination in the auxiliary building, will be supplemented temporarily by a second high-range monitor to be installed outside of a containment personnel hatch. This would provide indirect accident radiation level monitoring capacity for the Sequoyah containment. TVA's commitments meet the design and operation positions in NUREG-0578 regarding high-range in-containment radiation monitors but do not meet the NUREG-0578 implementation dates listed in NUREGs 0660 and 0694, Item II.F.1.

TVA has identified several reasons for the delayed commitment: acceptable and qualifiable equipment was not available in the early part of the year; General Atomic only recently offered equipment acceptable to TVA; the recently completed shielding and environmental analysis results indicated the desirability of in-containment monitoring. Thus, the possible delays are attributable to TVA's prior commitment to out-of-containment radiation monitors and the late commitment of TVA to install in-containment monitors. In letter of July 25, 1980, TVA provided revised dates which identifed March 1981 as delivery date, and the first refueling outage, now scheduled for early 1982, as the installation date. TVA proposed in letter of July 18, 1980, an acceptable temporary alternative until monitors are installed in containment. Also, TVA agreed in letter of August 11, 1980, to provide a procedure to correlate the out-of-containment monitor readings with in-containment radiation levels by August 20, 1980. This review completes the evaluation of Item 2.1.6.b/II.F.1, with onsite verification of installation and operation to be completed during a future routine inspection.

Projected Completion Date

It is our position that for good cause shown, TVA should be allowed to install the monitors during the first forced or scheduled outage of sufficient length to allow installation after delivery of the monitors. TVA projects a completion date of early 1982.

TVA has stated that a noble gas effluent monitor will be installed on all identified release paths at Sequoyah Unit No. 1. The monitors will meet the staff criteria established in NUREG-0578.

II.F.2 Instruments for Inadequate Core Cooling

POSITION

Install, if required, additional instruments or controls needed to supplement installed equipment in order to provide unambiguous, easy-to-interpret indication of inadequate core cooling.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.3b, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSION

Staff Evaluation of ICC Instrumentation and Procedure for Full-Power Operation

In NUREG-0011, SER Supplement No. 1, February 1980, the staff concluded that the subcooling meter committed by TVA using the plant computer is acceptable for full power. It is the staff position that prior to January 1, 1981, the Sequoyah instrumentation to monitor adequacy of core cooling should meet the provisions of Regulatory Guide 1.97 Revision 2. This applies to the subcooling meter, the incore thermocouple, and to the computer used to process instrument signals.

With respect to our review of emergency procedures for inadequate core cooling in a letter of July 10, 1980 (reference 1) to the Westinghouse Owners Group (which includes TVA), we stated that:

> "The question of the influence of UHI on the core exit thermocouple (T/C) indications of inadequate core cooling (ICC) must be resolved before full power operating guidelines depending on core exit thermocouple indications can be found acceptable."

The basis of this concern is that cold water from the UHI accumulators might inject during periods of ICC and affect the core exit thermocouple readings. The operator may then mistakenly believe that the core is being adequately cooled. To address this concern we requested Westinghouse (Ref. 1) to perform detailed calculations of T/C behavior during ICC conditions.

In response to this request, Westinghouse provided the results of several calculations using NOTRUMP (Reference 2). A one-inch break without any high pressure safety injection was analyzed to examine conditions of inadequate core cooling in a UHI plant. Several periods of UHI were predicted to occur. These injection periods were of relatively short duration compared to the total accident time. One injection period did occur during a period of core uncovery and heatup (ICC). In this case, the core uncovery reduces the steam in the core which resulted in system depressurization and consequent UHI injection. The injected fluid drained into the core and generated additional steam which repressurized the system and terminated UHI after about 100 seconds. During and immediately following this calculated injection period, the core and upper plenum remained superheated. Only for a very short time did the upper head contain a two-phase mixture. Thus, there is a short-lived potential for UHI water to drain through the support column and pass the T/C's to cause T/C temperature indications lower than expected. Even if the operator were to terminate ICC procedures based on this indication, the T/C's would quickly return to a superheated condition soon after UHI termination. Also, low T/C readings soon after UHI do not necessarily mean a misinterpretation of the existence of ICC since the core is being partially cooled by the UHI water during the injection period.

To further explore this problem, Westinghouse performed calculations to determine the effect of steady slow injection as opposed to the relatively rapid injection described above. These calculations showed that an optimized depressurization rate of 0.07 psi per second generates a maximum UHI delivery rate of about 5 lbs/sec. This rate is so small that most of the water is predicted to be boiled by heat from the metal and would have almost no effect on ICC indication.

Therefore, it appears that there is no prolonged effect of UHI that would invalidate use of the core exit T/Cs as an indicator of ICC. Even if the operator should terminate ICC procedures based on a temporary drop in T/C temperature indications, the operator would quickly become aware that ICC conditions were either still present or re-established, and emergency procedures would require that he reinstate ICC procedures.

Westinghouse also provided an analysis of a 4-inch break without high pressure safety injection. For this case, the UHI stop valves permanently terminate UHI injection long before ICC conditions would be achieved. Therefore, the problem does not exist for the larger end of the small break spectrum which do not significantly rely on the steam generator to remove decay heat.

Without providing supporting analyses, Westinghouse has contended that three dimensional effects would always show that some T/Cs would reflect ICC conditions when they exist. Also the effect of non-equilibrium behavior has not been explored. It is not known, a priori, if this would further confound indications of ICC. We therefore intend to pursue confirmation of these conclusions both with Westinghouse and through our own audit calculations. However, we conclude that the Westinghouse analysis contained in Reference 2 provides adequate assurance that core exit T/Cs can be used as an indicator of inadequate core cooling.

TVA, in their letter dated July 25, 1980, states that installation cannot be completed on schedule due to delays in development and delivery. TVA estimates that the installation will be completed by January 1981.

The staff has been monitoring the progress of other applicants and licensees in meeting schedule requirements of II.F.2 and has had meetings with suppliers of various level measurement systems to review the design and development progress and the equipment procurement situation. Based on our continuing review of this situation, we conclude that the applicant is making a good faith effort to procure this system as early as feasible. Therefore, we find the Sequoyah 1 compliance with TMI II.F.2 to be acceptable for full-power operation. However, we will require that the procedure guidelines for use of the proposed equipment, the analysis used in developing these procedures, an updated schedule giving the development and procurement status, and any available test data be submitted for staff review by January 1, 1981. Barring unforeseen circumstances which preclude the acceptability of this system, we require that it be installed at the earliest feasible date consistent with scheduled or forced plant outages, and that in-service testing, calibration, and implementation proceed on a schedule acceptable to the staff.

References

1. Letter from P. S. Check to Cordell Reed, WOG, transmitting review of WCAP-9639 dated July 10, 1980.

2. Letter from T. M. Anderson to T. D. Speis, NS-TMA-1179 dated July 24, 1980.

III. Emergency Preparations and Radiation Protection

III.A.1.2 Upgrade Emergency Support Facilities

POSITION

Provide radiation monitoring and ventilation systems, including particulate and charcoal filters, and otherwise increase the radiation protection to the onsite technical support center to assure that personnel in the center will not receive doses in excess of 5 rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site emergency operations facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center and direct display of radiological and meteorological parameters.

This requirement shall be met by January 1, 1981, although the safety parameter information requirements will be staged over a longer period of time. (See NUREG-0578, Sections 2.2.2b and 2.2.2c, and letters of September 27 and November 9, 1979 and April 25, 1980.

DISCUSSION AND CONCLUSIONS

The requirements stated above have been revised. This revision has been approved by the Commission. The licensee will be required to meet the requirements of NUREG-0696, "Functional Criteria for Emergency Response Facilities," to be published for comment in July or August 1980. NUREG-0696 provides the details needed to design and implement a Technical Support Center (TSC) and Emergency Operations Facility (EOF). A revised schedule for implementation of a total requirements package is also under development.

The Emergency Preparedness Evaluation Report (Appendix E to this Supplement) describes the Technical Support Center, Operations Support Center, and Emergency Operations Facility established on an interim basis. Therefore, we conclude as a result of our review and licensee's commitments by letter of July 28, 1980, as well as FEMA's findings of August 7, 1980 (see Appendix E), that these facilities are adequate for full-power operation.

III.D.3.3 Inplant Radiation Monitoring

POSITION

Provide the equipment, training, and procedures to accurately measure the radioiodine concentration in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before January 1, 1981. See NUREG-0578, Section 2.1.8.C, and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSION

By letters dated 11/21/79 and 1/11/80, TVA has submitted commitments and documentation of actions to be taken at Sequoyah to implement short-term lessons learned items in NUREG-0578.

The Sequoyah plant has portable low-volume air monitoring equipment with charcoal filters and silver zeolite filters available to sample for radioiodine. Analysis equipment includes a Nuclear Data 6620 system with three Ge(Li) detectors in the radiochemical laboratory, with an Eberline SAM-2/NaI detection system as backup. Alternate counting facilities, with Nuclear Data 6620 systems and 2 Ge(Li) detectors, are located onsite in the training facility and offsite at the nearby Watts Bar Nuclear Plant. The alternate facilities give a capability to promptly and accurately analyze samples under low background conditions. The Sequoyah Radiation Control Instruction Manual and Health Physics Laboratory Instruction Manual contain the necessary procedures for radioiodine sampling and analysis, and training in procedures and instrumentation is required for plant health physics technicians. The Sequoyah plant has adequate post-accident iodine sampling and analysis capability and meets our positions in NUREG-0578.

The Sequoyah plant meets the staff position for this item.

23.0 CONCLUSIONS

Based on our evaluation of the application as set forth in our Safety Evaluation Report issued in March 1979 and Supplement No. 1 and our evaluation as set forth in this supplement, we conclude that, subject to resolving matters related to hydrogen control as discussed in Section 22.2 Item II.B.7, the operating license can be issued to allow power operations at full rated power (megawatts thermal) subject to license conditions which will require further Commission approval and license amendments before the stated condition can be removed.

We conclude that the construction of the facility has been completed in accordance with the requirements of Section 50.57(a)(1) of 10 CFR Part 50 and that construction of the facility has been monitored in accordance with the inspection program of the Commission's staff.

Subsequent to the issuance of the operating licenses for full rated power for Sequoyah Nuclear Plant, Units 1 and 2, the facilities may then be operated only in accordance with the Commission's regulations and the conditions of the operating license under the continuing surveillance of the Commission's staff.

We conclude that the activities authorized by the licenses can be conducted without endangering the health and safety of the public, and we reaffirm our conclusions as stated in our Safety Evaluation Report and its supplement.

APPENDIX A

CHRONOLOGY FOR RADIOLOGICAL

SAFETY REVIEW

November 19, 1979	Letter to TVA concerning ~~implementation of Resident Inspection Program~~.
December 21, 1979	Letter to TVA concerning environmental monitoring for direct radiation.
December 21, 1979	Letter to TVA concerning change to current regulation on radiological emergency response plans.
December 26, 1979	Letter to TVA ref 12-3-79 letter to consider permitting TVA to conduct activities including fuel loading, zone power physics testing, special testing and operator training at Sequoyah.
December 26, 1979	Letter to TVA concerning request for information regarding evacuation times.
December 27, 1979	Letter to TVA concerning ATWS procedures for procedures for Sequoyah.
December 31, 1979	Letter to TVA ref their letter of 11-21-79 on steps taken re IE Inspection report.
January 3, 1980	Letter from TVA re upgraded emergency plans.
January 4, 1980	Letter from TVA forwarding responses to our December 3, 1979 questions.
January 7, 1980	Letter concernng preacceptance safeguard site visit at Sequoyah.
January 7, 1980	Letter to TVA concerning Physical Security Plans.
January 7, 1980	Letter from TVA forwarding list of significant outstanding items required for plant operation.
January 7, 1980	Letter to TVA concerning test program at Sequoyah.
January 7, 1980	Letter from TVA forwarding revisions to revised response to Lessons Learned Task Force requirements.

January 10, 1980	Letter from TVA responding to our 12-27-79 letter re concerns about abnormal operating instruction.
January 11, 1980	Letter from TVA forwrding revision re relief and safety valves testing program.
January 11, 1980	Letter from TVA forwarding revised responses to 10-5-79 questions on water level measurement system inside Government.
January 11, 1980	Letter from TVA re hole formation in rodded guide thimble tubes.
January 17, 1980	Letter to TVA concerning implementation of recommendations of NUREG-0660.
January 18, 1980	Letter from TVA forwarding revisions to revised response to NUREG-0578 re TMI Lessons Learned Task Force short-term requirements...
Jauary 23, 1980	Letter from TVA with final deficiency report re ice condenser heat loads exceeding design heat loads.
January 23, 1980	Letter from TVA....forwarding responses to Geosciences Branch.
January 24, 1980	Letter from TVA re welds on pressurizer relief piping.
January 25, 1980	Letter from TVA re review of procedures and operator training program.
January 25, 1980	Letter from TVA forwarding utility response to Rogovin report.
January 25, 1980	Letter from TVA forwarding revised response to short-term Lessons Learned Task Force requirements.
January 28, 1980	Letter to TVA re installation of some NRC-sponsored instrumentation.
February 1, 1980	Letter from TVA re concrete expansion anchors.
February 4, 1980	Letter from TVA forwarding proposed revisions to FSAR.
February 4, 1980	Letter from TVA forwarding proposed tech specs re surveillance requirements for diesel generator batteries and vital battery banks.
February 5, 1980	Letter to TVA concerning issuance of NUREG-0588.
February 7, 1980	Letter from TVA on enhancement of onsite emergency diesel generator reliability.
February 7, 1980	Letter from TVA re methods used to ensure integrity of auxiliary building secondary containment enclosure.
February 11, 1980	Letter from TVA concerning Short-Term Lessons Learned Requirements.

February 12, 1980	Letter from TVA discussing operational procedures for degraded core conditions.
February 13, 1980	Letter from TVA advising that employees with experience in operation of PWRs will be provided on each shift during fuel loading and low power testing.
February 14, 1980	Letter from TVA re seismic and environmental qualification of Class IE electrical equipment.
February 14, 1980	Letter from TVA re utility position on items identified as open or in progress re organization and management criteria....
February 14 1980	Letter from TVA advising of utility commitment to implement Lessons Learned Task Force short- and long-term changes.
February 21, 1980	Letter from TVA re shift manning and operator retraining.
February 21, 1980	Letter to TVA concerning qualification of safety-related electrical equipment.
February 21, 1980	Letter from TVA re onshift coverage by persons with experience in loading large PWR plants.
February 26, 1980	Letter from TVA forwarding Test Report, "Reactor Vessel Nozzle Inspection."
February 27, 1980	Letter from TVA re human engineering of control room panels.
February 29, 1980	Letter to TVA transmitting license DPR-77...License for Fuel Loading & Low Power Testing.
March 5, 1980	Letter to TVA concerning metallurgical examination of pressurizer relief pipe mockup.
March 5, 1980	Letter to TVA concerning fuel load procedure change.
March 5, 1980	Letter from TVA re pile design for waste packaging & condensate demineralizer waste evaporator bldg.
March 7, 1980	Letter from TVA....forwarding executed Amendment 6 to Indemnity Agreement B-82.
March 7, 1980	Letter to TVA concerning emergency operating instructions.

March 11, 1980	Letter to TVA concerning change of submittal date for evacuation time estimates...
March 17, 1980 ENVIRO	Letter from TVA forwarding amended fish entrainment operational monitoring plan.
March 19, 1980	Letter to TVA concerning potential design deficiencies in bypass, override & reset circuits.
March 20, 1980	Letter to TVA transmitting 15 copies of Supplement 1 to SER.
March 20, 1980	Letter from TVA forwarding evacuation time estimates provided.
March 24, 1980	Letter to TVA concerning request for additional information on PCP.
March 25, 1980	Letter from TVA on Reactor Coolant System-Low Pressure System Isolation Valves.
March 26, 1980	Letter from TVA on Low Pressure Turbine Inspection.
March 26, 1980	Letter from TVA forwarding utility visual inspection procedure.
March 27, 1980	Letter from TVA on the Special Test Program.
March 28, 1980	Letter from TVA with application for amendment to License DPR-77.
April 1, 1980	Letter to TVA concerning safeguards contingency plans.
April 2, 1980	Letter from TVA re preacceptance safeguards visit.
April 2, 1980	Letter from TVA forwarding revisions to revised response to NUREG-0578.
April 9, 1980	Letter from TVA forwarding "Master Plan for Special Low Power Test Program."
April 10, 1980	Letter to TVA concerning low power test programs.
April 11, 1980	Letter from TVA discussing revision to abnormal operating instruction for reactor trip.
April 11, 1980	Letter from TVA on the ATWS Procedure.
April 11, 1980	Letter from TVA transmitting Amendment 64 to FSAR.
MONTHLY REPORT	Letter 4-11-80 from TVA re utility participation in ORNL on-line analyzer program.

April 13, 1980	Letter from TVA on Emergency Operating Procedures.
MONTHLY REPORT	Letter 4-14-80 from TVA proposing amendment to License DPR-77, changing tech specs.
MONTHLY REPORT	Letter 4-15-80 from TVA discussing facility low-pressure turbine preservice inspection.
MONTHLY REPORT	Letter 4-15-80 from TVA forwarding procedure to be used to conduct water hammer test on auxiliary feedwater system.
April 15, 1980	Letter from TVA forwarding draft revisions to emergency operating instructions.
April 21, 1980	Letter to TVA concerning information request on Category I masonry walls.
April 22, 1980	Letter from TVA on PCP
MONTHLY REPORT	Letter 4-24-80 from TVA re adequacy of piles supporting waste packaging area & condensate demineralizer waste evaporator bldg.
April 24, 1980	Letter from TVA re changes to engineered safety features.
April 24, 1980	Letter from TVA on the WPA and CDWEB Foundations.
April 25, 1980	Letter from TVA forwarding Rev. 9 to preservice baseline inspection and inservice inspection program.
April 25, 1980	Letter 4-25-80 to TVA concerning clarification of NRC requirements for emergency response facilities at site.
April 28, 1980	Letter from TVA re control rod guide turbine pin cracking.
MONTHLY REPORT	Letter 4-28-80 from TVA forwarding evaluation of auxiliary feedwater system.
April 29, 1980	Letter from TVA concerning incorrect pipe stresses.
MONTHLY REPORT	Letter 5-1-80 from TVA forwarding revised Radiological Emergency Plan.
May 1, 1980	Letter from TVA on the Radiological Emergency Plan.
May 1, 1980	Letter from TVA on the Technical Support Center.
May 1, 1980	Letter from TVA on the seismic analysis of the WPA and CDWEB.

May 1, 1980	Letter from TVA informing of relocation of Central Emergency Control Center.
May 1, 1980	Letter from VA advising that utility will perform auxiliary feedwater pump endurance tests.
May 1, 1980	Letter from TVA re status of operator training for degraded core conditions.
May 1, 1980	Letter from TVA notifying that seismic analysis of waste packaging area and condensate demineralizer waste evaporator bldg. has been completed.
May 5, 1980	Letter from TVA forwarding 2 additional requests for relief from ASME Section XI preservice and inservice inspection requirements.
May 5, 1980	Letter to TVA on low power test program and procedures.
May 8, 1980	Letter from TVA forwarding new test program, "Seals Between Ice Condenser and Containment Vessel for Reactor Buildings."
May 8, 1980 PROP INFO	Letter from TVA forwarding revision to revised response to NUREG-0578.
May 8, 1980	Letter from TVA re short-term Lessons Learned Task Force requirements.
May 8, 1980	Letter to TVA on the Reactor Coolant System Head Vent.
May 8, 1980 PROP INFO w/o PROP	Letter from TVA transmitting Prop version of revisions to facility radiological emergency plan.
May 9, 1980	Letter from TVA re safety evaluation of special test program.
May 12, 1980	Letter from TVA forwarding cross reference between evaluation criteria in NUREG-0654 and utility radiological emergency plan.
May 12, 1980	Letter from TVA re facility special tests.
May 13, 1980	Letter to TVA concerning water hammer tests for aux. feedwater system.
May 15, 1980	Letter from TVA on environmental qualification of Westinghouse supplied electrical equipment.
May 16, 1980 PROP INFO w/o PROP	Letter from TVA Safeguards Contingency Plan.

May 19, 1980	Letter from TVA re facility diesel generator system...
May 19, 1980	Letter from TVA forwarding response to ACRS questions transmitted in our letter.
May 19, 1980	Letter from TVA re yard drainage pond effluent. ENVIRO.
May 20, 1980	Letter from TVA transmitting "Flood Insurance Study."
May 20, 1980	Letter from TVA on the Special Test Program and Procedures.
May 24, 1980	Letter from TVA requesting modification of tech specs.
May 27, 1980	Letter from TVA forwarding proposal for seismic margin program, required by NUREG-0611.
May 28, 1980	Letter from TVA with application for amendment to License DPR-77.
June 2, 1980 PROP INFO w/o PROP	Letter from TVA forwarding information on training & qualification plan.
June 6, 1980	Letter from TVA forwarding Amendment 65 to FSAR.
June 9, 1980	Letter from TVA forwarding review of requests for relief from ASME Section XI preservice and inservice inspection requirements.
June 11, 1980	Letter from TVA revising schedule on actual finish dates in revision 4 of TVA responses to Auxiliary System Branch Fire Protection Review Questions.
June 13, 1980	Letter from TVA revising FSAR Question 5.9.A on service inspection program.
June 13, 1980	Letter from TVA responding to NRC on Barton Lot 2 transmitters.
June 13, 1980	Letter from TVA requesting amendment to license DPR-77 to change the Technical Specifications of Unit 1.
June 16, 1980	Letter from TVA forwarding radiation and shielding design review report and assessment of electrical equipment qualification program.
June 16, 1980	Letter from TVA forwarding results to Sequoyah, Watts Bar, Bellefonte, and Yellow Creek Hydromotor Actuator Deficiencies.
June 17, 1980	Letter from TVA with revised response to question 12.16.
June 19, 1980	Letter from TVA providing addditional information on the newly revised Radiological Emergency Plan for Sequoyah.

June 19, 1980	Letter to TVA requesting information on shielding review.
June 20, 1980	Letter from TVA forwarding change in Unit 1 Technical Specifications.
June 20, 1980	Letter from TVA on Inoperable Hanger RCH-118.
June 23, 1980	Letter from TVA forwarding amendment to license DPR-77.
June 23, 1980	Letter from TVA requesting that Unit 2 be exempted from the preoperational reduced pressure containment integrated leak test.
June 23, 1980	Letter from TVA responding to question 031.27 and additional seismic qualification report TS-1091, also responding to questions 040.60 and 031.129.
June 23, 1980	Letter from TVA submitting formal response to NUREG-0578 clarification Items, Rev. 0.
June 23, 1980	Letter to TVA requesting commitment for providing information on steam generator tubes.
June 24, 1980	Letter from TVA forwarding amendment to license DPR-77 to change Technical Specifications.
June 25, 1980	Letter from TVA on Special Test Program with regard to Rod Withdrawal Events.
June 26, 1980	Letter from TVA providing fuel load dates for Sequoyah Unit 2 and Watts Bar Units 1 and 2 and probable completion dates.
July 1, 1980	Letter from TVA providing additional information concerning the operator training.
July 1, 1980	Letter from TVA stating position with respect to a near-site EOF, prompt notification and shift manning.
July 3, 1980	Letter from TVA responding to TMI-2 Lessons Learned Task Force Status Report on on Short Term Recommendations.
July 3, 1980	Letter from TVA on NUREG-0578, Item 2.1.4, Containment Isolation.
July 3, 1980	Letter from TVA revising their commitment in the Fire Protection Program Reevaluation.
July 8, 1980	Letter from TVA transmitting diesel generator drawings.

July 8, 1980	Letter from TVA containing recommendations regarding the Sequoyah Nuclear Plant generator.
July 8, 1980	Letter from TVA responding to Final Safety Analysis Report Question 2.78.
July 9, 1980	Letter from TVA responding to Question on Category I Masonry Walls at Sequoyah.
July 11, 1980	Letter from TVA on the CSSC list.
July 11, 1980	Letter from TVA responding to FSAR questions.
July 11, 1980	Letter from TVA responding to TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations.
July 11, 1990	Letter from TVA on NUREG-0578, Item 2.1.6a, Leak Reduction.
July 15, 1980	Letter to NRC ACRS recommending full power operation of Sequoyah with certain considerations.
July 17, 1980	Letter from TVA discussing hydrogen ignitors.
July 17, 1980	Letter from TVA forwarding 41 copies of the documentation requested during the TVA-NRC meeting concerning the fire protection modification schedules for the Sequoyah Nuclear Plant Unit 1.
July 17, 1980	Letter from TVA regarding additional contractor support to provide operational experience on each shift during the low-power test program.
July 18, 1980	Letter from TVA providing additional information on the Process Control Program for Sequoyah Nuclear Plant.
July 18, 1980	Letter from TVA providing additional information regarding shielding design radiation monitor, access control of areas adjacent to spent fuel transfer tubes and containment sump debris.
July 18, 1980	Letter from TVA responding to questions on the Reactor Vessel Head Venting System.
July 18, 1980	Letter from TVA concerning performance of Special Test 7 "Simulated Loss of All Onsite and Offsite AC Power".
July 18, 1980	Letter from TVA responding to shift manning licensing examinations, licensee dissemination of operating experience, LOFW and small break LOCA generic matters, and control room habitability.
July 21, 1980	Letter from TVA on NUREG-0654 and the Radiological Emergency Plan.

July 21, 1980	Letter from TVA on Shielding Design Review, HIgh Range Containment Radiation Monitors, Spent Fuel Transfer Tube, and Containment Vent Sump Debris.
July 22, 1980	Letter from TVA regarding training program to mitigate core damage.
July 23, 1980	Letter from TVA concerning review of power-ascension test and emergency procedures before full power licensing.
July 24, 1980	Letter from TVA on Main Control Room Habitability
July 24, 1980	Letter from TVA providing preliminary list of equipment that could change position upon reset of an ESF signal.
July 25, 1980	Letter from TVA providing their evaluation of design and equipment delivery schedules for plant modifications required as dated items in NUREG-0694.
July 28, 1980	Letter from TVA providing revised responses to questions SNP Q2.84 and SNP Q2.73 on settlement.
July 28, 1980	Letter from TVA discussing the results of a preliminary study on the probability of barge traffic impacting the ERCW.
July 28, 1980	Letter from TVA giving updated status of responses to NUREG-0588.
July 28, 1980	Letter from TVA discussing the location of the emergency control center.
July 28, 1980	Letter from TVA discussing commitments to correct items identified in the main control room design review.
July 29, 1980	Letter from TVA providing information on temperature control for the main steam valves.
July 29, 1980	Letter from TVA providing test results of the Plant Auxiliary Building Gas Treatment System.
July 29, 1980	Letter from TVA providing the Sequoyah Special Startup Test Report.
July 31, 1980	Letter from TVA providing additional documentation of several items on the Appendix A, Critical Structures, Systems, and Components List, to the Operational Quality Assurance Manual.
July 31, 1980	Letter from TVA describing the methods used at Sequoyah to maintain steam generator tube integrity.
July 31, 1980	Letter from TVA providing information on the operator training program.
July 31, 1980	Letter from TVA providing results of ECCS Performance Analysis with Rate Dependent Burst Model.

July 31, 1980	Letter from TVA on the Inservice Inspection Program.
August 1, 1980	Letter from TVA responding to NRC questions on TVA Radiological Emergency Plan for Sequoyah Nuclear Plant.
August 1, 1980	Letter from TVA providing information on the test of the Interim Auxiliary Building Secondary Containment Enclosure and the Auxiliary Building Gas Treatment System.
August 1, 1980	Letter from TVA providing a table of turbine material characteristics. (withheld from public disclosure).
August 1, 1980	Letter from TVA commiting to revise the nomenclature for six labels on the Main Control Room Panels by September 1, 1980.
August 5, 1980	Letter from TVA clarifying their position on the use of the new ERCW pumping station between Unit 1 and Unit 2 operations.
August 5, 1980	Letter from TVA on operator qualification
August 5, 1980	Letter from TVA discussing Westinghouse analyses to show that the steam line and feedwater line break analyses were conservative.
August 5, 1980	Letter from TVA providing the Requalification Training Program and documentation of compliance with NUREG-0694, Item I.A.3.1.
August 5, 1980	Letter from TVA providing requested information on the Sequoyah Nuclear Plant Containment Analysis.
August 7, 1980	Letter from TVA on fire protection modifications.
August 8, 1980	Letter from TVA on RV head vent design.
August 8, 1980	Letter from TVA on vented filter containment.
August 11, 1980	Letter from TVA on containment radiation monitors.
August 11, 1980	Letter from TVA on MCR design review.
August 11, 1980	Letter from TVA on the ERCW pumping station foundation.
August 11, 1980	Letter from TVA on the Seismic Margin Program.
August 11, 1980	Letter from TVA on equipment qualification.

August 11, 1980	Letter from TVA on fire protection sprinkler heads.
August 11, 1980	Letter from TVA on containment hydrogen analysis.
August 11, 1980	Letter from TVA on Onsite Safety Review Group.

APPENDIX D

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

GENERIC MATTERS AND LETTERS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1980

The Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REPORT ON THE SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2

Dear Dr. Ahearne:

During its 243rd meeting, July 10-12, 1980, the Advisory Committee on Reactor Safeguards completed its review of the application of the Tennessee Valley Authority (hereinafter referred to as the Applicant) for authorization to operate the Sequoyah Nuclear Plant, Units 1 & 2 at full power. The Committee had considered aspects of the application during its 242nd meeting, June 5-7, 1980; 236th meeting, December 6-8, 1979; 229th meeting, May 10-12, 1979; and 228th meeting, April 5-7, 1979. A tour of the facility was made by members of the Subcommittee on January 24, 1976 and the application was considered at Subcommittee meetings on July 9, 1980; June 2, 1980; November 5, 1979; and March 12, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, the Westinghouse Electric Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on interim low power operation of Unit 1 on December 11, 1979 and on a construction permit for this plant on February 11, 1970.

In its letter of December 11, 1979 the Committee addressed the proposed special low power test program, to be carried out on Unit 1, the seismic reevaluation of the Sequoyah plant, actions on recommendations resulting from the review of the accident at the Three Mile Island Station, Unit 2, and actions on various generic problems. These generic problems were further discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 7," dated March 21, 1979. The Committee's recommendations in its December 11, 1979 letter are also applicable to Unit 2 except that the special low power test program will not be repeated on Unit 2.

The special low power test program has been reviewed by Westinghouse Electric Corporation and by the NRC Staff. The Applicant began these tests on July 11, 1980 and the Applicant, Westinghouse, and the NRC Staff will review the results of these tests. It is expected that the additional operator training and operator experience will prove to be beneficial.

The Committee has reviewed and reported on NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident," Draft 3. The status of the Applicant's compliance with the NTOL licensing requirements as well as a number of non-TMI-related items were reviewed during its 243rd meeting. There are a number of both non-TMI and TMI-related requirements not fully resolved. Both the NRC Staff and the Applicant expect that the complete resolution of these outstanding items is essentially a procedural or documentary matter which will be completed within a very few weeks. These items should be resolved to the satisfaction of the NRC Staff. The Committee wishes to be kept informed. The Committee believes that the implementation of the Action Plan as it will be realized at Sequoyah is adequate to assure the safe operation of this plant.

The Committee, in its March 11, 1980 report on the NTOL items, recommended that the licensees develop reliability assessments for their plants and that design studies of possible hydrogen control and filtered vented containment systems be required. The Applicant has conducted studies of a number of means for hydrogen control, and as an interim measure, has proposed installation of a distributed array of ignition sources which it expects to have in place by the fall of 1980. The Applicant has concluded that by this means the containment would be able to cope with the pressure resulting from the combustion of hydrogen released by the reaction with water of up to about 70% of the zirconium in the core. This compares with the 25% which the containment could cope with without any additional control measures and the 30 to 50% estimated to have reacted in the accident at TMI. The NRC Staff plans to review the proposed system in detail to assure itself of its efficacy and that all safety aspects have been taken into account. The Committee wishes to be kept informed of the further conclusions reached by the Staff and the Applicant in their continuing consideration of these matters. The Applicant has conducted reliability assessments of some features of the plant and has considered some aspects of the effects of a possible filtered vented containment. Though the work accomplished to date is limited in scope, these studies are definitely responsive to the Committee's recommendations on these points. The Applicant proposes to continue studies of this nature and to extend the range of their application. While these efforts, as well as those concerned with hydrogen control, should be vigorously pursued, in view of the commitments made by the Applicant, it is the opinion of the Committee that their present incomplete status need not delay the issuance of a full power operating license.

Early this year a differing professional opinion was advanced by a member of the NRC Staff concerning the acceptability of a particular weld repair in the piping to a pressurizer relief valve of Sequoyah Unit No. 1. All other qualified and responsible members of the NRC Staff, as well as professional personnel on the staff of the Applicant, take the position that the weld should be regarded as acceptable since there is no evident reason why it should not be at least as capable as other (more standard) welds which would be considered acceptable. The differing opinion is not that the weld is demonstrably less capable than it need be, but 1) that the evidence available is inconclusive on this point, and 2) that more specifically relevant information could be obtained without serious difficulty. This could be done by constructing a mock-up of the weld in question using material and procedures as similar as possible to those which apply in the actual case and subjecting the mock-up to a through-wall metallographic examination. The results of this examination could then (for example) be compared with those from a full penetration weld in the same material, which has been performed in the standard fashion and deemed acceptable based on satisfactory operational experience with which the majority opinion has compared the present weld. This has not been done. The Committee does not consider it to be particularly likely that this weld repair presents a serious hazard; but it does believe the evidence on this point could be improved. The Committee believes that, in the interest of resolving the question that has been raised to the maximum extent readily possible, steps of the nature outlined should be taken.

The Committee believes, that if due consideration is given to the items mentioned above, the Sequoyah Nuclear Plant, Units 1 and 2 can be operated at levels up to full power without undue risk to the health and safety of the public.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:
1. Tennessee Valley Authority, "Final Safety Analysis Report, Sequoyah Nuclear Power Plant," Volumes 1-13, and Amendments 1-63.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," NUREG-0011, March 1979.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2," Supplement No. 1, NUREG-0011, February 1980.
4. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, May 1980.
5. U.S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," NUREG-0694, June 1980.

APPENDIX E

EMERGENCY PREPAREDNESS

EVALUATION REPORT

INTRODUCTION

The Tennessee Valley Authority (hereinafter referred to as the Licensee, The Company, TVA) filed with the Nuclear Regulatory Commission a revision to the Sequoyah Nuclear Plant Emergency Plan dated July 21, 1980, as amended (hereinafter referred to as the Plan). The Commission's staff conducted a review of this Plan. The staff's review also included a site visit to the facility and a public meeting, and observation of an emergency exercise involving TVA, Tennessee State and local county agencies on June 16-17, 1980.

The Plan was reviewed against the criteria of the sixteen operator Planning Objectives in Part II of the "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (For Interim Use and Comment)," NUREG-0654.

As a result of public comments, staff comments, and development of the final rule on emergency planning, NUREG-0654 will be revised. The Plan will be reviewed against the revised criteria and a supplement to this report will provide our review results and conclusions.

Attachment 1 to this appendix provides the findings and determinations of the Federal Emergency Management Agency on the State and local emergency response plans.

TVA's letter of July 21, 1980, stated their intention to implement the Radiological Emergency Plan as submitted upon receipt of a full-power license for Sequoyah Nuclear Plant.

A. Assignment of Responsibility (Organization Control)

PLANNING OBJECTIVE

To assure that primary responsibilities for emergency response in nuclear facility operator, State and local organizations within the Emergency Planning Zones have been assigned, that the emergency responsibilities of the various supporting organizations have been specifically established, and that each principal response organization is staffed to respond and to augment its initial response on a continuous basis.

DISCUSSION

The Shift Supervisor for each unit of the Sequoyah Nuclear Plant is designated as the Site Emergency Director. When an abnormal condition arises, it is his responsibility to determine if the abnormality meets any of the emergency classifications specified in the Plan and to implement the Plan, if necessary. There is 24-hour a day communication capability between the station and Federal, State, and local response organizations to ensure rapid transmittal of accurate notification information and emergency assessment data.

Responsibility for overall performance of the emergency response organization is vested in the Site Emergency Director who is responsible for the overall direction of the plant emergency organization. Qualified members of the station staff who report directly have been assigned specific responsibilities for the major elements of emergency response.

Updated written agreements with appropriate agencies and organizations are maintained by TVA's Division of Occupational Health and Safety.

B. Onsite Emergency Organization

PLANNING OBJECTIVE

To assure that on-shift facility operator responsibilities for emergency response are unambiguously defined, that adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, and timely augmentation of response capabilities is available, and that the interfaces among various onsite response activities and offsite support and response activities are specified.

DISCUSSION

The Shift Supervisor assumes the responsibilities of the Site Emergency Director until relieved by plant superintendent or his designated alternate. The authorities and responsibilities of the Site Emergency Director have been clearly specified, including those that cannot be delegated. The Site Emergency Director can immediately and unilaterally declare an emergency and make offsite notifications.

Station staff emergency assignments have been made and the relationship between the emergency organization and normal staff complement are shown in the Plan. Positions and/or titles and qualifications of shift and plant staff personnel both onsite and offsite who are assigned major emergency functional duties are listed. Table B-1 of NUREG-0654 describes proposed minimum staffing requirements. TVA states that it is required by the existing operating license to provide shift staffing as outlined in Table B-1, with the exception of (1) the Mechanical Maintenance/Rad Waste Operator and (2) Electrical Maintenance/Instrument and Control (I&C) Technician. These two positions are normally manned 16 hours per day. TVA states that provisions will be made for these two additional staff to be made available onsite within 30 minutes. In a letter dated August 5, 1980, TVA has committed to revise the TVA-REP to include a summary of shift manning as described above by August 15, 1980.

TVA management personnel will supply support services utilizing any necessary manpower and equipment. A long-term emergency organization framework is in place, headed by the Central Emergency Control Center Director. Interfaces between and among the TVA staff, station staff, governmental and private sector organizations, and technical and/or engineering contractor groups have been specified along with services to be provided.

C. Emergency Response Support Resources

PLANNING OBJECTIVE

To assure that arrangements for requesting and effectively using assistance resources have been made, that arrangements for State and local staffing of the operator's Emergency Operations Facility have been made, and that organizations capable of augmenting the planned response have been identified.

DISCUSSION

TVA established its Central Emergency Control Center (CECC) in Chattanooga, Tennessee. The stated purpose of the CECC and its staff was to provide the facilities and manpower for evaluating, coordinating, and directing the overall activities involved in coping with a radiological emergency at any TVA reactor. The location of the CECC is 16-1/2 miles frcm the Sequoyah reactor; however, in response to NRC's concern over the distance from the CECC to the reactor, TVA has now agreed to establish a near-site Emergency Operations Facility for Sequoyah at the TVA Training Center located about one mile from the Sequoyah Plant. The Chattanooga facility will be used as a backup facility. State and local authorities can be accommodated at both of these locations. (Note: TVA believes that their philosophy of a central emergency response center requires additional discussion and reserves the right to pursue this issue with the NRC Commissioners and staff.)

When the Emergency Plan is activated, the Shift Engineer on duty at the reactor control room notifies the Operations Duty Specialist in Chattanooga, who notifies Tennessee State authorities, the TVA Information Office, and the Directors of four TVA Emergency Response Centers, and in the case of a General Emergency also makes a direct parallel notification to local County Emergency Control Centers.

The four TVA Emergency Response Centers are:

(1) The Central Emergency Control Center in Chattanooga;

(2) The Division of Nuclear Power Emergency Center (DNPEC), located immediately adjacent to the Central Emergency Control Center. The DNPEC provides organizational, material, personnel, and technical support from Nuclear Maintenance, Reactor Engineering, Controls and Test, and Outage Boundaries;

(3) The Knoxville Emergency Control Center, which provides the technical support of the Division of Engineering Design personnel and the personnel and equipment support of the Division of Construction; and

(4) The Muscle Shoals Emergency Control Center, which provides staff and equipment for performing environmental radiological monitoring and dose assessments and for recommending procedure actions for the public.

The TVA Radiological Emergency Plan has also identified facilities and staff at Oak Ridge at the George C. Marshall Space Flight Center, Tennessee State and local police organizations, all available for assistance in an emergency.

D. Emergency Classification System

PLANNING OBJECTIVE

To assure that a standard emergency classification and action level scheme is in use by the nuclear facility operator, including facility system and effluent parameters; and to assure that State and local response organizations will rely on information provided by facility for determinations of initial offsite response measures.

DISCUSSION

TVA utilizes the following emergency classifications:

(1) Notification of Unusual Event

(2) Alert

(3) Site Emergency

(4) General Emergency

The TVA Radiological Emergency Plan (TVA-REP) states that this system of classification is consistent with the systems used by State and local emergency organizations. The initiating conditions used for recognizing and declaring the emergency class are based on specific measurable values or observable conditions defined as Emergency Action Levels (EALs). The specific instrument readings and parameters required for determination of these EALs are detailed in plant operating instructions and will be used as thresholds for determining the emergency classifications. The NRC staff position is that the pertinent instrument readings, parameters, and equipment status should be specified in the Emergency Plan itself, and TVA has committed to revise the TVA-REP to include such information by August 15, 1980 (see letter dated August 1, 1980).

E. Notification Methods and Procedures

PLANNING OBJECTIVE

To assure that procedures have been established for notification, by the facility, of State and local response organizations and for notification of emergency personnel by all response organizations; to assure that the content of initial and followup messages to response organizations and the public have been established; and to assure that means to provide early warning and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

DISCUSSION

Procedures have been established for notification of State and local response organizations in case of emergency. The Site Emergency Director has been given authority and responsibility to initiate prompt notification to these agencies. TVA notifies the local authorities through the permanently manned position of the Operations Duty Specialist (ODS). The ODS, upon notification from the affected plant of a General Emergency, is responsible for notifying the local Civil Defense Agency at once, prior to any other action. Notification of the appropriate State agency is the ODS's next action.

In a letter dated August 1, 1980, TVA commits to a prompt notification system having the design objective capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. TVA will expedite procurement of a prompt notification system to be installed and operational in accordance with the following estimated dates:

(1) Order equipment (bid award) - 15 Nov - 15 Dec 1980

(2) Receive equipment - 15 April - 15 May 1981

(3) Install equipment - 15 May - 15 June 1981

(4) Operational - 1 July 1981

This notification system will meet requirements of the final rule on emergency planning with regard to prompt notification of the public.

F. Emergency Communications

PLANNING OBJECTIVE

To assure that provisions exist for prompt communications among principal response organizations, to emergency personnel and to the public.

DISCUSSION

The station communications system is designed to provide secure, redundant, and diverse communications to all essential onsite and offsite locations during normal operations and under accident conditions. Within-station systems are comprised of a public address system, two-way radio systems, a private automatic (PAX) exchange, and a sound-powered telephone system. Offsite systems are comprised of both commercial and leased telephone lines, a microwave system, and two-way radio systems. A Bell Telephone ring down system is dedicated as the primary means of communication between plant and offsite emergency control centers.

These telephones plus other systems located in plant areas are manned 24 hours a day. The Site Emergency Director will, in emergency situations, communicate directly with the TVA Operations Duty Specialist who is responsible for providing initial notification to the appropriate State emergency organization. In the event of a General Emergency, he is required to notify the appropriate local response agency. These offices are manned 24 hours a day.

Communications between the Control Room and the Technical Support Center, Operations Support Center, and Emergency Operations Facility (i.e., PAX telephone) are available. Tests of the systems are held weekly.

G. Public Information

PLANNING OBJECTIVE

To assure that accurate and timely information is provided to the public on how they will be notified and what their initial actions should be; to assure that the principal points of contact with the news media for dissemination of information (including physical location or locations) are established in advance; and to establish procedures for coordinated dissemination of information to the public.

DISCUSSION

An information brochure has been mailed to all residents within 10 miles of the Sequoyah reactor that describes how people will be notified, where they should go, and what to do in an emergency. The brochure includes a map indicating the various sectors around the plant, major evacuation routes, traffic control points, and Shelter Information Points. Enclosed with the brochure was a questionnaire with a request to return the questionnaire to the Hamilton County Civil Defense Office. The purpose of the questionnaire was to ascertain needs such as transportation or a place to stay in an evacuation; and this information is on file and will be available for use by local agencies in an emergency. TVA has committed, in coordination with appropriate State Agencies, to mail such a brochure to each residence in the 10-mile Emergency Planning Zone annually.

The Tennessee Governor's Press Secretary, or the Governor's designated representative, is the Emergency Information Officer and is responsible for coordinating and supervising the release of all public information in disaster conditions. An annual orientation will be conducted to acquaint news media with points of contact for release of public information.

H. Emergency Facilities and Equipment

PLANNING OBJECTIVE

To assure that adequate emergency facilities and equipment to support the emergency response are provided.

DISCUSSION

Emergency facilities needed to support an emergency response have been provided including a Technical Support Center and an Operations Support Center. Each will be activated for an Alert or higher emergency classifications. The Technical Support Center has been established in the relay room in the power house control bay inside the protected area. The Technical Support Center contains a complete set of functional plant drawings and necessary technical information.

In regards to the Emergency Operations Facility, TVA has agreed to provide an interim facility in a letter from L. M. Mills to H. R. Denton, dated July 28, 1980. The TVA Radiological Emergency Plan will be revised to include a description of this facility. This revision will include a description of the facility, location, communications, and manning requirements.

The Operations Support Center (assembly area) is located in the station's locker and lunch room and will be the assembly point for unassigned personnel. Equipment and supplies are available, if needed (respiratory protective devices, protective clothing, portable lighting).

Stored equipment is inspected and inventoried and replaced, if in need of calibration or repair. Sufficient equipment exists to ensure a minimum inventory in case of replacement delay. Portable monitoring instruments are stored in the Health Physics area. Calibration of equipment is carried out at intervals recommended by the supplier of the equipment.

The meteorology equipment at the site meets the criteria of Regulatory Guide 1.23, "Onsite Meteorological Programs," dated February 17, 1972. A backup meteorological measurements program with redundant power sources is also available. In letters dated August 1 and 5, 1980, TVA has committed to provide an upgraded Technical Support Center at Sequoyah Nuclear Plant by December 31, 1981. This facility will have the capability of providing real-time meteorological data to offsite locations. TVA has committed to provide for the remote interrogation of meteorological data by the NRC (at the incident response center) and other emergency organizations that require it.

TVA has models available for use during accidental atmospheric release of radioactivity to provide initial estimates and detailed dose information.

These models provide the calculational capability for generating real-time, site-specific estimates of atmospheric transport and diffusion and radiation doses for the major exposure pathways.

Provisions for offsite monitoring equipment have been made through TVA's Muscle Shoals Emergency Control Center (MSECC). Clerical support, dose accessment, personnel dosimetry, radioanalytical laboratory services, and coordination of field activities are provided. The field activities include ensuring the availability and transport of health physics personnel and equipment and the direction of environmental moritoring teams.

I. Accident Assessment

PLANNING OBJECTIVE

To assure the adequacy of methods, systems, and equipment for assessing and monitoring actual or potential offsite consequnces of a radiological emergency condition.

DISCUSSION

Onsite capability and resources to provide initial and continuing assessment throughout the course of an accident includes process, effluent, and area monitors that read out in the control room; post-accident sampling capability and containment monitoring. TVA has provided information on these capabilities in response to NRC letter dated October 30, 1979, relative to the Lessons Learned Program designated in NUREG-0578. TVA will review the Radiological Emergency Plan to reflect these capabilities.

The TVA-REP states that TVA is prepared to assess the consequences of potential or actual releases offsite. TVA has transmitted to the staff a copy of the Muscle Shoals Emergency Control Center Implementing Procedures. These procedures include emergency actual and predictive dose assessment procedures for atmospheric and liquid releases of radioactivity from the Sequoyah Plant.

In addition, one additional sampling team can be at the plant within two hours of notification. A third team will be dispatched from the Muscle Shoals facility. An all-weather helicopter is provided for use by the MSECC team.

J. Protective Response

PLANNING OBJECTIVE

To assure that a range of protective actions is available for the plume exposure pathway for emergency workers and the public, guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in use, and that protective actions for the ingestion exposure pathway appropriate to the locale have been developed.

DISCUSSION

The TVA-REP includes a table of recommended protective actions including shelter and evacuation, with special consideration for children and pregnant women to reduce whole body and thyroid dose from exposure to a gaseous plume, that are consistent with both the State of Tennessee and USEPA guidelines. The recommended actions include the statement that officials may implement low-impact protective actions at lower values in keeping with the principle of maintaining radiation exposures as low as reasonably achievable.

TVA commits to make recommendations for the ingestion exposure pathway to State and local agencies, but these agencies are responsible for the decision to act upon such recommendations. The State of Tennessee Radiological Emergency Response Plan states that protective actions will be based on the protective action guides developed by the USEPA and the U.S. Food and Drug Administration.

K. Radiological Exposure Control

PLANNING OBJECTIVE

To assure that means for controlling radiological exposures, in an emergency, are established for emergency workers and the affected population.

DISCUSSION

Implementing procedures have been developed to prevent or minimize exposure to radiation for onsite individuals. These procedures include evacuation, accountability, radiological monitoring, and decontamination of nonessential personnel. Respiratory protective equipment and protective clothing are provided for essential plant personnel who would remain onsite. The TVA-REP includes emergency guidelines for doses during an extreme emergency that are consistent with EPA Emergency Worker and Life Saving Activity Protective Action Guides. The REP states that personnel must be made aware of possible consequences of such exposures and must be selected on a voluntary basis unless they are members of an emergency team and have previously consented to receive the exposure.

The State of Tennessee and local agencies are responsible for implementing action to protect the health and safety of the public offsite. TVA will recommend protective actions, but the State and local governments are responsible for deciding if any actions are needed and what they should be. Potential choices for dose reduction for the public are: shelter; evacuation; closing of public water supplies; confiscation of crops, food and dairy products; placing milk animals on uncontaminated stored feed and use of potassium iodide. TVA has potassium iodide available for public utilization at State direction, according to specific agreements with the various States.

L. Medical and Public Health Support

PLANNING OBJECTIVE

To assure that arrangements are made for medical services for contaminated individuals.

DISCUSSION

TVA has made arrangements with Erlanger Medical Center, Chattanooga, Tennessee, to provide medical assistance to site personnel injured or exposed to radiation and/or radioactive material. This facility has an emergency plan, staff training program, and adequate equipment and supplies for receiving radiologically contaminated patients. The Oak Ridge Hospital in Oak Ridge, Tennessee, is available for the care and treatment of radiation accident victims from TVA and is used as a backup hospital. Based on the quality of the facilities at Erlanger Medical Center and Oak Ridge Hospital, we find the arrangement acceptable.

Emergency medical equipment such as stretchers, respirators, etc., are strategically located throughout the plant, and approved trauma kits and other specified equipment are readily available for use by the medical emergency response teams.

TVA has a Health Station that contains the normal complement of first aid supplies and equipment necessary to treat injuries not involving hospitalization or medical services. A TVA ambulance is available at the site to transport injured workers to hospitals. An agreement is maintained with Rhea County Ambulance Service to transport injured workers when necessary.

The Division of Medical Services provides training to selected onsite personnel to qualify them in first aid and emergency medical care. These personnel then serve as members of the plant medical emergency response teams.

M. Recovery and Reentry Planning and Post-Accident Operations

PLANNING OBJECTIVE

To assure that general plans for recovery and reentry are developed.

DISCUSSION

The Site Emergency Director and the site emergency organization will direct the recovery and reentry operations following an incident. All other TVA resources plus other governmental and vendor suport will be available through the TVA corporate organization to aid the Site Emergency Director in developing, evaluating, and implementing specific recovery and reentry operations.

The decision to downgrade an incident will be made by the Site Emergency Director after consultation with his plant technical and operations staffs and will be coordinated with the Central Emergency Control Center Director. Coordination of recovery activities will be handled through the Central Emergency Control Center, Chattanooga, Tennessee. In a letter dated August 1, 1980, TVA has agreed to provide a more detailed writeup of their recovery operations in the next revision of TVA's Emergency Plan, due January 1, 1981.

N. Exercises and Drills

PLANNING OBJECTIVE

To assure that periodic exercises are conducted to evaluate major portions of emergency response capabilities, that the results of exercises form the basis for corrective action for identified deficiencies, and that periodic drills are conducted to develop and maintain key skills.

DISCUSSION

The purpose of an exercise is to test the integrated capability of TVA, Tennessee State, and local emergency response organizations. It is designed to test a major portion of the basic elements existing within emergency preparedness plans and organizations.

An exercise was conducted at the Sequoyah Nuclear Plant in June 1980 and NRC observers reported favorable comments at the critique. TVA's Division of Occupation Health and Safety evaluates deficiencies disclosed in the critique and coordinates corrective actions throughout TVA and follows up to ensure completion.

A combined exercise involving TVA, State and local personnel will be held annually. The scenario for the exercise will be mutually agreed to and rotated each year to ensure that all major elements of the Emergency Plan are tested over a five-year period. At least once every six years, an exercise will be scheduled for each of the off-shifts.

Drills based on Emergency Conditions will be held at least annually for response components (e.g., fire, medical, health physics, communications) to ensure maximum effectiveness of the plan.

Each scenario is forwarded to the Radiological Emergency Planning Group, Radiological Hygiene Branch, no later than two weeks prior to conduct of the drill. This group shall approve the scenario and forward copies to the Quality Assurance Staff, Radiological Hygiene Branch, and the Quality Assurance and Audit Staff, Office of Power. Each drill will be conducted and critiqued by an independent TVA organization.

O. Radiological Emergency Response Training

PLANNING OBJECTIVE

To assure that radiological emergency response training is provided to those who may be called upon to assist in an emergency.

DISCUSSION

TVA provides training in emergency procedures to all permanent plant personnel and all nonplant personnel expected to be onsite for longer than one week. This training is such that each of these individuals will have a working knowledge of the emergency plan and his responsibilities and actions upon declaration of an emergency. Training consists of initial training classes and annual retraining, drills, and activation of the alarms to maintain familiarity with the features of the emergency plan. Training and annual retraining is provided to those offsite agencies who may be involved during an emergency and will include procedures for notification, basic radiation protection, their expected roles, and site access procedures, as applicable.

Offsite agencies who potentially will be called upon to participate in the plan have concurred with the responsibilities assigned their agency in the plan by executing a Letter of Agreement with TVA.

P. Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

PLANNING OBJECTIVE

To assure that responsibilities for plan development, review and distribution of emergency plans are established, and that planners are properly trained.

DISCUSSION

The Emergency Plan and Emergency Plan Implementing Procedures are formally reviewed annually for adequacy and applicability by the Division of Occupational Health and Safety (OCHS), noting any required changes. OCHS issues controlled revisions and assures that all holders receive all changes.

The qualifications of TVA staff responsible for radiological emergency planning include academic training in engineering or science. Several staff members have graduate training in nuclear engineering or health physics, and all have at least two years' experience in engineering, health physics, or emergency planning.

The Quality Assurance Staff, Radiological Hygiene Branch, and Office of Power, Quality Assurance and Audit Staff, audits the plan yearly for compliance with existing regulations and TVA's own internal requirements. These Quality Assurance organizations are responsible for offering recommendations on overall plan improvement. The results of audits are documented, reported to appropriate organizational management, and retained in the respective Quality Assurance files for a period of five years. TVA has agreements with outside organizations for radiological emergency support to furnish specific services. Copies of the letters documenting these agreements are forwarded to the Division of Occupational Health and Safety. These letters are updated every two years by the TVA organizations requiring these services.

CONCLUSONS

Based on our review of the Plan as submitted and the commitments made by TVA for further revisions, we have concluded that the Plan and the commitments meet the Planning Objectives as applicable to the licensee (operator) of the "Criteria for Preparation and Evaluation and Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (For Interim Use and Comment)," NUREG-0654. We have also concluded that the Plan and the commitments adequately respond to the Deficiencies to be Corrected for a Full-Power License listed in the NRC Safety Evaluation Report for Sequoyah Units 1 and 2, NUREG-0011, Supplement No. 1, February 1980.

The findings and determinations of August 7, 1980, on the State and local emergency response plans for Sequoyah were received on August 7, 1980, from the Federal Emergency Management Agency.

Based on the FEMA findings and our evaluation, we believe Sequoyah meets the emergency response plan requirements for a full-power license.

Attachment 7

FEDERAL EMERGENCY MANAGEMENT AGENCY
Washington D.C. 20472

August 7, 1980

Mr. William J. Dircks
Acting Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

In accordance with the proposed Federal Emergency Management Agency (FEMA) Rule 33 CFR 350, Review and Approval of State Radiological Emergency Plans and Preparedness, and the provisions of the new NRC Rule on emergency preparedness, I have prepared a certification of FEMA Findings and Determination (enclosed) with respect to the Tennessee Multi-Jurisdictional Radiological Emergency Response Plan for Tennessee Valley Authority's Sequoyah Nuclear Power Facility.

I find that I am able to approve the Plan with the following conditions:

(1) By July 1, 1981, the public alerting and notification system meets FEMA/NRC criteria, and;

(2) That the deficiencies detailed in the State findings are adequately resolved in accordance with the State schedule submitted to the Regional Director on August 1, 1980.

I will monitor the progress made by the State in correcting these deficiencies on the schedule defined in their findings relating to the exercise and report to you in accordance with §350.13 of the proposed FEMA Rule.

I am notifying Governor Alexander of my action as well as publishing the enclosed certification in the Federal Register.

Sincerely yours,

Frank A. Camm

Frank A. Camm
Associate Director for
Plans and Preparedness

Enclosure
as stated

Attachment I

United States of America

Federal Emergency Management Agency

In the matter of

THE TENNESSEE MULTI-JURISDICTIONAL RADIOLOGICAL EMERGENCY RESPONSE PLAN FOR TENNESSEE VALLEY AUTHORITY'S SEQUOYAH NUCLEAR POWER FACILITY

Docket No. FEMA-4-TN-1

CERTIFICATION OF FEMA FINDINGS AND DETERMINATION

In accordance with the FEMA rule, 44 CFR Part 350 (proposed), on June 20, 1980, the State of Tennessee submitted its plan to the Director of FEMA Region IV for review and approval. The Regional Director has forwarded his statement of findings and determination to the Associate Director for Plans and Preparedness on the subject plan and the associated facility, dated August 4, 1980, in accordance with §350.12 of the proposed rule. Included in these findings and determination is an evaluation of the State plan and the associated local plans for the Sequoyah facility, an evaluation of the exercise conducted on June 16-17, 1980, at the Sequoyah facility in accordance with §350.9 of the proposed rule, and a report of the public meeting held on June 10, 1980, to explain the site specific aspects of the State and local plan in accordance with §350.10 of the proposed rule.

Based on this statement and the review of FEMA headquarters staff, the Associate Director finds and determines that subject to the conditions stated below the State plans and preparedness including the local plans and preparedness for the Sequoyah facility are adequate to protect the health and safety of the public living in the vicinity of the Sequoyah facility by providing reasonable assurance that appropriate protective measures can and will be taken off-site in the event of a radiological emergency and are capable of being implemented. The Associate Director further finds and determines with respect to the joint criteria NUREG-0654/FEMA-REP-1 that:

a) the public alerting and notification system does not meet the requirements of Appendix 3, and

b) certain weaknesses were noted during the exercise for which the State has scheduled corrective action.

Therefore, the Associate Director approves the State plan and associated local plans for Sequoyah facility subject to conditions that:

a) by July 1, 1981, the public alerting and notification system meets FEMA/NRC criteria, and;

b) that the deficiencies detailed in the State findings are adequately resolved in accordance with the State schedule submitted to the Regional Director on August 1, 1980.

Attachment I

FEMA will continue to review the status of preparedness of the State and the local jurisdictions associated with Sequoyah facility in accordance with §350.13 of the proposed rule.

For further details with respect to this action, refer to the FEMA docket file maintained by the Regional Director at 1375 Peachtree Street, N.E., Atlanta, Georgia 30309.

For the Federal Emergency
Management Agency

Frank A. Camm

Frank A. Camm
Associate Director for
Plans and Preparedness

Dated August 7, 1980
Washington D.C.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

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Docket Nos. 50-327 and 50-328

16. ABSTRACT (200 words or less)
Supplement No. 2 to the Safety Evaluation Report of Tennessee Valley Authority's application for licenses to operate its Sequoyah Nuclear Plant, Units 1 and 2, located in Hamilton County, Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the status of certain items that had not been completely resolved at the time of publication of the Safety Evaluation Report, and defines the requirements that must be met for full power operation.

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