Weise

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA. TENNESSEE 37401
500A Chestnut Street Tower II

April 5, 1985

U.S. Nuclear Regulatory Commission
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW.
Suite 2900
Atlanta, Georgia 30323

Dear Dr. Grace:

WATTS BAR NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-390 - STATUS FOR FUEL LOADING

In my letter of February 20, 1985, TVA advised NRC that the design, construction, testing, and preparation for operation of Watts Bar Nuclear Plant unit 1 was essentially complete. The purpose of this letter is to update the enclosures to that letter and to respond to your March 18, 1985 letter to me requesting additional information on this subject. As you are aware, fuel loading for unit 1 is scheduled for April 23, 1985.

Enclosures 1 through 3 are updates of the same enclosures to the February 20, 1985 letter. Enclosure 4 is our response to your March 18, 1985 letter to me.

If you have any questions, please get in touch with R. H. Shell at FTS 858-2688.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

H G Parris

H. G. Parris
Manager of Power and Engineering

Enclosures (4)
cc: Director of Nuclear Reactor Regulation (Enclosures)
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

EXHIBIT 56

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
OUTSTANDING WORK ITEMS

A. 10 CFR 50 Appendix R

Appendix R modifications inside primary containment will be completed before the currently scheduled fuel load date for unit 1. TVA is attempting to complete all work outside unit 1 primary containment. However, if the present schedule cannot accommodate all of this work, the only portions which may remain outstanding at fuel loading will be cable tray wrap (approximately 600 feet), conduit wrap (approximately 600 feet), and replacement of fire doors (approximately two). This remaining work will be completed by start of mode 3.

During the interim period between fuel loading and mode 3, TVA will initiate the necessary fire patrols and fire watches as prescribed in draft technical specification sections 3.7.11.2 and 3.7.12.

B. Critical Safety System Components (CSSC) and Non-CSSC

There are approximately 100 unit-1 related work items requiring physical work which will be completed after fuel loading of unit 1. Eight of these items affect the as-built plant in conformance with the FSAR and/or affect a TVA licensing commitment. These eight items are listed in attachment I with item number, item description, and justification for each item extending past fuel loading. The remaining items are not listed uniquely as they are system enhancements, office space additions/changes, and noncritical safety system changes that do not affect the as-built plant as described in the FSAR and/or other licensing commitments. Each of the 100 work items is listed on the WBN outstanding work item list (OWIL) and will have a design safety evaluation made with the results documented in accordance with engineering design procedure (EP) 2.13 before unit 1 fuel loading.

OWIL Item Number	Description	Justification
ECN 2351	Additional diesel generator unit (ADGU)	The ADGU will be used as a backup in maintenance situations and therefore is not required to be operational from a safety standpoint during normal operation of the plant. The essential raw cooling water (ERCW) to the ADGU is installed. The valves which connect the ADGU to the ERCW are closed and an interface hold order in place on each valve. Electrical connectors which will allow use of the ADGU have been installed and an interface hold placed on their use. This system will not be used until NRC approval is given and all preops are completed satisfactorily.
ECN 4594	Complete heating, ventilating, and air conditioning (HVAC) controls work in low level radioactive waste (LLRW).	This modification will not be needed to process radwaste until after initial plant operations. This will be installed after material receipt and in advance of the need for the compactor in radwaste processing.
ECN 4816	Add temperature switch on auxiliary building standby HVAC coolers to make logic agree with schematics	The remaining change for this ECN is to install a temperature switch on auxiliary B standby HVAC cooler fans to prevent their operating when not required. Presently they operate simultaneously with primary fans. This change only eliminates their unneeded operation and does not affect safety. Therefore, deferral until after fuel loading is acceptable.
ECN 4884	VHF radio and paging	The only portion of this work required to meet TVA commitments or for safety is completion of communication commitments (in health physics area) under the radiological emergency plan (REP). REP work is not required until initial criticality since no radiological hazard exists until after that time.

OWIL Item Number	Description	Justification
ECN 4884 (Continued)		Remaining work is an enhancement of present facilities and is therefore justified for completion later.
ECN 4978	Spray shields on hydrogen igniters in upper compartments	These shields have not been approved for use by NRC at this time. Implementation will be contingent on NRC approval.
ECN 5070 and 5355	Technical support center (TSC) and safety parameter display system (SPDS) modifications other than computer room hardware	These modifications have been committed to be installed by startup after the first refueling outage. The justifications for this are found in TVA's response to NUREG-0737, supplement 1.
PT 18279	Terminate cables for reactor vent and condensate vacuum vent radiation monitors to the TSC computer	This is the only computer room hardware item not expected to be complete by fuel loading. New cables have been pulled to match new radiation monitors but require connector installation before termination. The information provided by these monitors will be available in the main control room (MCR). Also, operation of the TSC computer is not required until the first refueling outage. Therefore, deferral of this work until the first refueling outage is acceptable.
Items concerning NUREG-0612, FS-439, and ECN 4411	Replacement and/or upgrade cranes, slings, or other lifting devices	This is for completion of remaining work on lifting devices falling under NUREG-0612. TVA has determined that the devices in question are not needed to perform safety-related lifts until the beginning of the first refueling outage. Any unexpected lifts will be analyzed to conform with NUREG-0612. Therefore, this work is not needed before first refueling outage.

C. Interfaces Between Units and the Use of Temporary Alterations (TA)

The interface program uses two basic tools to control the interfaces between the licensed and unlicensed units. These two are (1) interface hold orders and (2) interface Temporary Alteration Control Forms (TACFs). The points of interaction between the licensed and unlicensed unit's systems or components may be temporarily altered by wire lifts, jumpers, damper or valve removal, drain capping, and installation of temporary blanking plates or cross-tie lines. An example of a typical interface point would be the jumper in the protection racks which disables the unit 2 safety injection signal's ability to start all of the diesel generators.

Temporary alterations to the physical facilities of the plant are controlled under AI-2.15. TACFs in effect at fuel load will have an unreviewed safety question determination (USQD) performed per AI-2.18 or will have an unimplemented design item evaluation (UDIE) performed by the Office of Engineering (OE). If a TACF to a system configuration is to remain on an operable system longer than 30 days, all controlled copies of the "as-constructed" drawings shall be marked. Copies of existing TACFs and their Safety Reviews are onsite and available for review.

D. Operational Instructions (EOIs, SOIs, SIs)

The following Watts Bar Surveillance Requirements will not have a Power Operations Review Committee (PORC) reviewed SI before fuel load. All required operating instructions will be written and approved.

SR 4.4.5.0 through 4.4.5.5 - The ASME Section XI augmented inservice inspection program (IIP) covers steam generator samples, tube sample selection and inspection, inspection frequencies, acceptance criteria, and reports. The Nuclear Services Division is preparing these instructions and will submit the inservice inspection program six months before the first refueling outage. This meets SER sections 5.2.4 and 6.6 requirements of the IIP being submitted to the NRC staff for approval before the first refueling outage.

SR 4.4.10 - Inspection of each reactor coolant pump flywheel per C.4.b. Regulatory 1.14 (R1) (August 1975) will be submitted in conjunction with the inservice inspection program above.

SR 4.0.5 - (SI 4.0.5.0) - The SI for ASME Code Class 1 leakage test (SI-4.0.5.0) will be in place at fuel load. Since the code does not require a similar leakage test for Classes 2 and 3 components until after commercial operation, the SIs for these tests will not be written and in effect until after fuel load. Unique 10-year interval requirements, such as system hydros, will be written and approved well ahead of required performance. Other portions of this SR will be implemented before fuel load.

<u>ENCLOSURE 2</u>

WATTS BAR NUCLEAR PLANT UNIT 1
PREOPERATIONAL TESTING

All of the Preoperational Test Program open items that have been identified as being important to safety will be satisfactorily tested before fuel loading as identified by the Office of Engineering (OE). The FSAR Chapter 14, Table 14.2-1, identifies the preoperational tests that will not be completed until after fuel load. These tests are included as attachment I.

There are remaining open items on preoperational tests that were previously identified as required for fuel load. Attachment II identifies items which will not be completed until after fuel load.

W-1.5	AFL**	By Hot Shutdown
W-1.6	AFL	By Hot Shutdown
W-1.8	AFL	By Hot Shutdown
W-1.9	AFL	By Hot Standby
W-5.1	AFL	Before Initial Criticality
W-5.2	AFL	Before Initial Criticality
W-5.3	AFL	Before Initial Criticality
W-5.4	AFL	By Hot Shutdown
W-7.5	AFL	
W-8.1	AFL	15 to 30% Power
W-8.2	AFL	50% Power
W-8.4	AFL	Hot Shutdown
W-8.5	AFL	0 to 6% Power
W-9.3	AFL	Escalation to Full Power
W-9.4	AFL	Escalation to Full Power
W-9.5	AFL	
W-9.6	AFL	Escalation to Full Power
W-9.7	AFL	Before Initial Criticality
W-9.11	AFL	75% Power
W-10.2	AFL	
W-11.2	AFL	By Hot Standby
TVA-10	AFL	
TVA-11A & B	AFL	
TVA-12A	AFL	
TVA-22	AFL	By 10% Power
TVA-23A & B*	AFL	
TVA-29	AFL	
TVA-55	AFL	
TVA-70	AFL	
TVA-71	AFL	

*The thermal expansion of the feedwater system will be checked at the various power plateaus during power ascension testing. Shimming will be performed as required to meet inspection criteria.

**After fuel loading

(a). TVA-12D - Offsite Power System (reactor coolant pump (RCP) boards) - The test is written to test the reactor coolant pump boards for both units 1 and 2. The remaining open item is on the unit 2 RCP boards.

(b). TVA-16B - Vital dc Power System - Battery Load Verification - Sections 5.4 and 5.5 of this test will be performed during Startup Tests SU-4.9B, Loss of Offsite Power, and SU-6.2, Trip from 100-percent Power. These steps observe the Vital dc system behavior during the two startup tests.

(c). TVA-18B - Essential Raw Cooling Water Flow Balance - This item has been completed on the existing carbon steel valves. The temperature control valves on upper and lower containment vent coolers, control rod drive mechanism (CRDM) coolers, and RCP motor coolers will be retested after ECNs 2756 and 4845 replace the valve bodies. This will improve system integrity.

(d). TVA-24 - Fire Protection - Ventilation System - An exception is made on testing of several dampers because construction of the Hot Shop Area has not been completed. Testing will be performed immediately after hot shop completion. Testing these fire dampers in the shop area after fuel load will have no effect on unit 1 operation. The hot shop area is not used until after initial criticality, and the dampers will be tested before area use.

(e) TVA-28 - Sampling System -

(1). A sampling capability check of the upper head injection (UHI) surge tank and water accumulator cannot be done until the accumulator is pressurized.

(2). The gross failed fuel detector (GFFD) samples cannot be obtained until the system has been placed in service. The GFFD system is not needed until initial criticality.

(3). Distillate and concentrate samples at the boric acid evaporator will be tested after construction work is complete on ECN 3815. It should be noted that plant operation without this modification will result only in an economic penalty to TVA due to increased boric acid use and waste processing requirements.

(4). Testing relative to the sodium analyzer can only be done when the steam generator reaches pressure and temperature during startup.

All of the above listed items will be completed before initial criticality.

(f). TVA-31A - Proce . Radiation Monitoring - Two ra. .tion monitors on the boric acid evaporator packages will be tested when the evaporators are operational after the scheduled modifications are completed. (See item (e) concerning boric acid evaporator.)

(g). TVA-51 - Flood Mode Protection System - The test is written for both units 1 and 2. All of the unit 1 testing is complete. Unit 2 spool pieces for the HPFP connection to the unit 2 auxiliary feedwater pumps will be tested later.

(h). TVA-53 - Equipment for Replacement of Radwaste Filter Elements - Test portions concerning the spent filter storage pit will be tested when the Westinghouse drum grab lift and drum shield are available. Testing of unit 2 filter equipment will be done prior to unit 2 fuel loading. This equipment is used only in the storage area snd performs no safety function.

(i). TVA-55 - Loose Parts Monitoring System - Baseline signature data on steam generation No. 4 and the reactor coolant pumps was not obtained during Hot Functionals due to minor sensor cable problems. The cable problems have been resolved, and the signature data will be obtained during plant startup.

(j). W-1.1 - Reactor Coolant System (RCS) Heatup for Hot Functional Testing -

(1). Steam Generators 1-4 reference level controllers (L-3-231, 232,233,234) automatic response failed to meet test acceptance criteria. The controllers will be recalibrated and the data submitted to OE for review. No additional retesting is expected.

(2). The Chemical and Volume Control System (CVCS) letdown pressure control valve PCV-62-81 valve trim will be changed out because it is not properly sized (C_V is too large) to control RCS pressure automatically with the RCS system water solid. The valve trim will be replaced before solid system operation, and all testing will be complete by initial criticality.

(k). W-1.2 - RCS Heatup for Hot Functional Testing -

(1). The damaged AVMS annunciator relay (K1 on TEC Model 134 annunciator card) will be replaced and retested. Relay PRVAR which caused relay K1 to fail has already been modified by ECN 5169.

(2). Foxboro transmitters for LT-68-325C and LT-68-326C (pressurizer level) are being replaced with transmitters that have a maximum working pressure of at least 2500 psi. These transmitters will be installed and calibrated before heatup to initial criticality.

(3). Pressurizer pressure transmitter testing will be completed at normal RCS operating temperature.

(4). The air temperatures inside the steam generator enclosures will be rechecked at normal RCS operating temperature. The ventilation system ductwork has already been modified to redirect the air flow at the top of the system generator enclosures.

Test items (2) through (4) above cannot be completed until the unit reaches operating pressure and temperature but will be completed before initial criticality.

(l). W-1.7 - Reactor Coolant System Thermal Expansion - The reactor coolant pump loop 2 tie rod gaps PS 9 and PS 10 will be measured during heatup to 557°F and shims will be installed as necessary.

(m). W-7.1B - Reactor Protection System Time Response Measurement - The response time for the turbine driven auxiliary feedwater pump to reach full flow will be measured during REC heatup at normal operating temperatures before initial criticality. The response time of the containment pressure transmitters will be measured before initial criticality.

(n). W-3.1A3 - Safety Injection System (SIS) - Integrated Check Valve Flow and Integrity Test - Document the changing of FI-63-76 range according to FDR-WATM-10261. The existing flow indicator has been tested; however, it is scheduled for replacement. This will increase the ability to more accurately measure the SIS check valve back leakage.

(o). W-9.2 - Incore Thermocouple System - Performed at 250°F, 350°F, 450°F, and 557°F during heatup. Collect resistance data from [illegible] ire resistance temperature detectors (RTDs) and temperature data from incore thermocouples. The test will determine T_{avg} from data collected from RTDs and compute correction factors for incore thermocouples. This test is to be reperformed due to the replacement, as a system enhancement, of the incore thermocouples and modifications to the cables and reference junction boxes before criticality.

(p). W-10.8 - UHI - The remaining after fuel load testing consists of filling, venting, and pressurizing the system and verifying that the hi-lo level and pressure alarms function properly. This system is not required to be operational until mode 3.

(q). TVA-25-C - Fire Detection System - Six additional fire detectors are being added to the North and South valve rooms by ECN 5613. Installation of these additional detectors will not be completed until after fuel loading. Testing of these detectors will be completed before entry into mode 4.

Table 14.2-1 in the FS identified several tests whicl ave portions that are performed both before and after fuel load. The following open items have been identified on the portion of those tests which were to be performed before fuel load:

(a). TVA-22 - Auxiliary Feedwater System - The turbine drive auxiliary feedwater pump (AFWP 1A-S) flow data did not meet the acceptance criteria for full flow with the trip and throttle valve (FCV-1-51) throttling and pump speed less than 3950 RPMs. Recirculation and full flow time response data was questionable due to suspected problems with test equipment calibration. AFWP 1A-S will be tested before criticality at 557°F in the RCS to take additional data for pump performance and time response. At this time, the AFWP 1A-S steam seals will be observed to determine why water is leaking from the turbine shaft seals when FCV-1-51 is cracked open before pump starts. A leaky union on the AFWP 1A-S steam drain line will be replaced with a straight section of pipe. This work will be completed before the operation of the AFWP during heatup.

Our schedule testing of the AFWP 1A-S at an RCS temperature of 557°F (mode 3) is in accordance with the requirements of Technical Specifications. The Technical Specifications require that the pump be operable at the entry into mode 3 but also states that surveillance requirement 4.0.4 is not applicable to this pump for entry into mode 3. The pump performance surveillance cannot be performed with less than 1000 psig in the secondary side of the steam generators. Our planned testing activities will satisfy these requirements.

(b). TVA-23-A - Thermal Expansion of Piping Systems - As a result of modifications that were made on the main steam line hangers 01A-1MS-V128 and V64 after mini-hot functionals, these lines will have to be monitored during heatup. Hangers on the steam dump lines will have to be checked during heatup and low power operations since there was not an adequate steam supply during Hot Functionals to sufficiently heat the lines.

(c). TVA-29 - Steam Generator Blowdown (SGBD) - Listed below are actions that were taken to address problems encountered in testing during Hot Functionals. A retest will be performed during startup to ensure that the listed actions adequately address and resolve the problems.

(1). The control loop for 1-FCV-15-18 was recalibrated to allow for proper flash tank level control.

(2). The Δ P trip setpoint for the SGBD pumps 1A and 1B was revised to allow the pumps to trip out on low Δ P.

(3). Level Switch 1-LS-15-12B was recalibrated to turn off the SGBD pumps when a low level in the flash tank occurs.

(4). Flow orifi   for 1-F1-15-213 was installed   allow an accurate flow measurement to be obtained.

(5). The 2-329 control loop was recalibrated to allow for adequate flow through the stacked heat exchanger to obtain an acceptable Δ P across the stacked heat exchanger.

(6). The suction piping for the SGBD pumps was increased in size to allow for adequate flow rates.

(7). Flow recorder 1-FR-15-25 has been set up properly to allow flow rates to be recorded.

(8). 1-FCV-15-212 was repaired so proper SGBD flow isolation could be obtained.

(9). Pressure indicators 1-P1-15-5 and 20 were changed to a high range to keep them from pegging out and breaking. A general exception was on the remaining portions of the test which could not be completed due to the problems that were encountered during testing. The portions of the test that are required to be completed by initial criticality will be completed by initial criticality. The other remaining portions will be completed by their scheduled power levels.

(d). W-8-4 - Initial Turbine Roll - A test deficiency was written when an unacceptable vibration level was measured at bearing numbers 4 and 6 during Hot Functionals. Rebalance of the turbine rotor at bearings 4 and 6 has been completed. Acceptable vibration levels will be verified during the turbine roll after criticality.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT UNIT 1
10CFR50.55(e) REPORTS (CONSTRUCTION DEFICIENCY REPORTS)

Item No.	Justification
WBNMEB8107, et. al	Although completion of the equipment qualification program is required before fuel loading, specific components may be exempted from full compliance with the rule until November 30, 1985. The final rule concerning environmental qualification of electrical equipment allows the licensee to complete its equipment qualification program after plant licensing provided that adequate justification is given in accordance with the requirements specified by 10 CFR 50.49(i). This justification was submitted to the NRC by letters dated August 19, 1983, July 24 and December 20, 1984, and February 19, 1985.
WBNNEB8335	This item involved an error in the peak containment temperature analysis which did not consider superheated steam release due to uncovering steam generator tubes. Westinghouse has informed TVA that proposed analytical techniques being used for Duke's Catawba Nuclear Station (CNS) yield containment temperatures that are less than those currently used in the design basis of the plant. Since WBN is essentially the same as CNS, TVA intends to apply the same analytical techniques used for CNS at WBN and expects the same results. TVA, in conjunction with Westinghouse, does, however, plan to complete a plant-specific analysis for WBN approximately two weeks after the NRC-NRR Containment Systems Branch approves the CNS analysis. Based on the TVA review of the Westinghouse program on the CNS and the analyses performed to date, TVA does not expect this condition to require any modifications to the existing plant design. As such, it is our position that initial fuel loading and power ascension (up to full power) of WBN unit 1 can proceed without any unnecessary or undue risks to the health and safety of the public.
WBNNEB8403	This item involves an error in the Westinghouse main steam line break (MSLB) analysis for peak main steam valve room temperature which did not consider superheated steam release due to uncovering steam generator tubes. TVA has completed an analysis to revise the environmental profiles resulting from a postulated MSLB in the main steam line valve rooms utilizing data from Westinghouse. TVA is confident that the application of the release rates for an MSLB inside containment conservatively bounds the release rates for a valve room MSLB, and because of the design similarities between CNS and WBN, TVA is also confident that the use of the CNS data for WBN is appropriate.

TVA's evaluation of the postulated MSLB in the valve vault indicated that (1) the structural steel will remain intact even though some localized yielding will occur, (2) the valve vault concrete may undergo some localized damage with some spalling but its structural integrity will not be affected, and (3) all safety-related mechanical and class 1E electrical equipment (with the exception of postaccident monitoring (PAM) instruments and associated cabling) will perform all their required functions before temperatures rise to levels which invalidate the environmental qualification threshold of the equipment. Additionally, it was determined that equipment which could fail when its environmental qualification temperature is exceeded will fail in a position not adversely affecting plant safety. The PAM instrumentation and cabling which is required to function during and after the event will be protected from the valve vault environment by thermal installation and will be complete by fuel loading.

Because this problem has been determined to be generic to Westinghouse plants and a number of these plants have requested the same analysis that TVA has, Westinghouse will be resolving this issue through the Regulatory Response Group and expects to issue a final report on the matter by May 1985.

Even though TVA believes that its in-house analysis is conservative and the Westinghouse information will not contradict the analysis results nor require any additional corrective action, TVA will defer submittal of a final report on this matter to NRC until after review of the Westinghouse information. TVA expects this review to be complete and the final report to NRC to be issued by June 14, 1985.

WBNEEB8425

This item involves a deficiency with field-installed electrical cables. Specifically, field wiring that terminates within the housing of two solenoid valves has insulation which is not rated for temperatures which could possibly be generated within the valves. This condition was originally identified in NRC-OIE Information Notice 84-68.

In general, the use of electrical cable with inadequate temperature-rated insulation within a high ambient temperature valve body could cause the insulation to degrade prematurely. This could result in cable insulation failure and could possibly result in a failure of the ability of an affected valve to perform an intended safety function. With regard to WBN, TVA has determined by analysis that only two Target Rock valves, TVA Nos. 0-FSV-32-61-A and 0-FSV-32-87-B, would be significantly affected by the 280°F temperature which could be reached inside the valve housing. These valves control the flow of cooling water to the control air compressors. At their rat current, these cables can oper for 8000 hours (0.913 yea.) at 280°F. In addition, these valves fail open and they can be manually isolated and bypassed, so the potential for these valves to adversely affect the safe operation of the plant is minimal. The field wiring is scheduled to be replaced by September 13, 1985.

WBNNEB8419 This item involves problems with verifying that commitments made to NRC had been successfully implemented. All actions to prevent recurrence have been completed with the exception of revising TVA's Nuclear Quality Assurance Manual (NQAM) Part V, Section 16.2, "Control and Tracking of Licensing Commitments by OE and Office of Construction (OC)," to include controls for commitments to be implemented by TVA's Office of Nuclear Power (NUC PR). This is scheduled for completion by May 1, 1985. However, since all corrective actions necessary to provide reasonable assurance that the plant conforms to licensing commitments will be completed before fuel load, it is our position that initial fuel loading and power ascension can proceed without risk to the health and safety of the public.

WBNMEB8513 This item deals with the failure to document the basis for design changes for fire dampers. As part of the corrective actions for this item, TVA committed to issue affected system descriptions by September 1, 1985. However, the operating instruction has been revised to reflect the design basis to assure proper damper operation. In addition, Quality Information Release (QIR) No. QIR MEB 85005 was issued to serve as the design basis document until the system descriptions are issued. Therefore, it is our position that initial fuel loading and power ascension can proceed without risk to the health and safety of the public.

ENCLOSURE 4

RESPONSE TO NRC
REQUEST FOR ADDITIONAL INFORMATION
DATED MARCH 18, 1985
WATTS BAR NUCLEAR PLANT UNIT 1

NRC Position:

a. Based on significant technical inadequacies in surveillance instructions identified by a regional inspection team February 18-22, 1985, TVA should complete a review of every surveillance instruction for technical adequacy and resolve all identified deficiencies.

TVA Resolution/Completion:

Watts Bar Nuclear Plant (WBN) has completed a technical adequacy review of every surveillance instruction (SI) which implements the surveillance requirement of Appendix A technical specifications except those SIs identified in Enclosure 1 of this letter and three instrumentation SIs which are scheduled to be complete by April 5, 1985. This review was completed in conjunction with the review performed for the technical specification certification. Technical changes to the surveillance instructions resulting from the technical adequacy review are currently being incorporated and are scheduled to be completed by April 8, 1985.

NRC Position:

b. TVA licensed Operations staff should complete a thorough walkthrough of all Abnormal Operation Instructions and resolve all procedural deficiencies. Deficiencies in this area were identified during the regional inspection discussed in a. above.

TVA Resolution/Completion:

A walkthrough of all Abnormal Operating Instructions (AOIs) has been performed by licensed personnel on shift. This walkthrough was completed March 25, 1985. All procedural deficiencies identified as a result of the walkthrough are currently being evaluated. The results of this evaluation will be incorporated into all AOIs by April 8, 1985.

NRC Position:

c. During the resident inspectors' review of those human factors concerns generated by TVA's NUREG-0700 Phase I and II programs, significant technical deficiencies were identified which were not identified by TVA as requiring correction prior to licensing. TVA should re-review the human factors concerns and identify and resolve those deficiencies which constitute main control board design deficiencies or where control room alarms/indications are not consistent with the draft technical specifications.

TVA Resolution/Completion:

At the request of WBN management, the Control Room Design Review (CRDR) team accelerated completion of phases I and II (operator experience review and survey of the main and auxiliary control room) of the Detailed Control Room Design Review (DCRDR). These phases were completed on February 1, 1985 with the identification of approximately 1600 Human Engineering Concerns (HECs). Each of 1600 HECs were individually reviewed by the team member who identified the HEC for safety significance in accordance with the following criteria:

1) Prevent operator from performing timely action when immediate operator action is required for the safe shutdown of the unit

2) Result in inappropriate operator action or lack of appropriate operator action necessary for safe shutdown of the unit

3) Cause the operator to take inadvertent action which would lead to an unplanned release of radioactive material from the plant

As a result of the above review, 12 HECs were identified which, in the opinion of the reviewer, represented an area of potential safety significance. These 12 HECs were subsequently evaluated by plant management to establish a plan of action to resolve such items. All work identified as being required before fuel load has been completed except for movement of a nonseismically qualified clock which will be moved before fuel load.

A subsequent review of selected HECs by the resident inspector and TVA revealed that several of the HECs were items which were not true human factor items but rather items which could be entered into WBN's maintenance/corrective action program for resolution. To determine the number of maintenance-type items, three members of the CRDR team, including an Operations section representative, were requested to review the 1600 identified HECs using the criteria presented below:

1) HEC which can or should be corrected as part of a normal good maintenance program or plant instruction improvement program

2) HEC which identified an obvious design or construction error that would adversely affect plant operation

As a result of the above review, 288 HECs were identified for further evaluation by plant management to determine the corrective action required. TVA made a decision to require resolution of eight additional items before fuel load and one additional item to be completed by mode 4, the operational mode required by the technical specifications. Four of these items involve recorder chart paper corrections and four items involve changes to abnormal range or setpoint indications on control room panel instruments. The one remaining item has been completed. Corrective action scheduled for the remaining HECs is based on the nature of the item and the material available.

In those cases where the HEC constitutes a main control board design deficiency that is not consistent with the draft technical specifications or where the control room alarm/indicators are not consistent with the draft technical specification, TVA will take action to resolve the HEC before fuel load. As previously noted, there were only 21 HECs at this time which require some action before fuel load or the operational mode required by Technical Specifications to ensure consistency with the draft technical specifications.

To ensure completion of the corrective action, the items have been scheduled and will be tracked and verified to completion. The tracking program will be kept current and will be available for review and inspection until completion or resolution of the above HECs.

NRC Position:

d. Due to the recent increase in potential 10 CFR 50.55(e) reports from TVA and due to TVA's recent request for extensions for submittal of corrective action resolution on outstanding 10 CFR 50.55(e) reports, Region II requires additional technical information in order to evaluate the Watts Bar unit 1 physical readiness for licensing. Certain reports and potential reports raise questions on the adequacy of plant construction in those areas where problems have been identified. Sufficient information must be provided to Region II in order to allow the inspection staff to evaluate the significance and impact of these deficiencies.

TVA Resolution/Completion

As of March 28, 1985, our site records indicate approximately 21 10 CFR 50.55(e) issues that have not been previously reviewed and closed by NRC-Region II. Of these 21 items, 14 presently require completion of field work before fuel loading, one item has been reevaluated as nonreportable, one item is under evaluation, and five items have been scheduled for completion after unit 1 fuel loading. Justification for not finishing work on these issues before fuel loading has previously been provided to NRC-Region II. These remaining items will be completed by TVA in a timeframe that supports our current fuel loading schedule.

It has always been TVA's intention to submit complete and accurate reports to NRC-Region II on any condition reported in accordance with 10 CFR 50.55(e). We are confident that upon closure of the 50.55(e) issues requiring work by TVA before fuel loading, the plant will be physically ready for licensing and subsequent operation. TVA will continue actively working with NRC-Region II personnel in the review and closeout of remaining 50.55(e) issues.

NRC Position:

e. TVA should revise appropriate Administrative Instructions and Standard Practices to reflect conduct of safety-related activities onsite under the Site Director organization.

TVA Resolution/Completion

Administrative Instructions and Standard Practices have been reviewed to ensure they reflect the TVA reorganization. Any changes necessary for the conduct of safety-related activities onsite under the Site Director organization are scheduled to be completed by April 8, 1985.

NRC Position:

f. TVA should complete calibration procedures and calibrate and test all radiation monitors which will be required by technical specifications to be operable at all times.

TVA Resolution/Completion:

A total of 17 calibration Surveillance Instructions (SIs) are required to verify operable radiation monitors which are required operable by technical specifications at time of licensing. These monitors will be calibrated as the SI revision is approved subsequent to the technical adequacy review. All required calibrations are scheduled to be completed by April 11, 1985.