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TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
500A Chestnut Street Tower II

NRK -

February 20, 1985

prior to licensing

~~PORC~~
CADR -
Radcon - ?
~~Security plan~~,
CDRS
SRS

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

This is to certify that, to the best of my knowledge, the design, construction, testing, and preparation for operation of Watts Bar Nuclear Plant unit 1 have essentially been completed in accordance with descriptions contained in the Watts Bar Final Safety Analysis Report (FSAR) and other licensing documents.

The status of completion of Watts Bar unit 1 has been reviewed, and all items have either been resolved or are being tracked to achieve appropriate resolution. It is anticipated that certain activities will not be completed at the time of fuel loading. We have identified these items, evaluated their potential impact on the safety of facility operation, and provided justification for their scheduled completion. Based on the planned resolution of the incomplete items as scheduled, including those anticipated to be incomplete at the time of fuel loading, we have concluded that unit 1 is ready for licensing and operation.

Incomplete construction activities are discussed in enclosure 1, sections A, B, and C. Incomplete surveillance instructions being written to implement the technical specification surveillance requirements are discussed in enclosure 1, section D. Incomplete preoperational testing is discussed in enclosure 2. Finally, incomplete corrective action for construction deficiency reports is addressed in enclosure 3.

We have concluded that the status of these items, including related compensatory actions, should not preclude issuance of an operating license.

The TVA Quality Assurance organization has made independent reviews of program activities in the engineering, construction, and operation organizations relative to readiness for licensing and operation of the unit, and all items identified have been resolved or entered into a tracking system that ensures completion before associated milestones. Based on limited reviews by the Nuclear Safety Review Staff of all program activities in engineering and construction and a series of operational readiness reviews, one item (involving material verification for modifications after fuel load) was identified requiring resolution before fuel load. Pending resolution of identified items as scheduled, both organizations have indicated no reservations that the unit is ready for licensing and operation.

EXHIBIT 55

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 (3)
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

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
OUTSTANDING WORK ITEMS

A. 10 CFR 50 Appendix R

Appendix R modifications will be completed inside primary containment on Watts Bar Nuclear Plant (WBN) before unit 1 fuel loading and outside primary containment by start of mode 3. The work outside of unit 1 primary containment consists of the following physical work activities: cable tray wrap (approximately 1000 feet), conduit wrap (approximately 2000 feet), fire detector installation (approximately 60 each), replacement of fire doors (approximately 7), and completion of seismic supports on fire protection sprinkler piping (approximately 100 supports in auxiliary building).

Sound justification exists for excluding the high pressure fire protection (HPFP) seismic supports from those items required for fuel load. The probability of experiencing a design basis seismic event during the approximately 6-week period between fuel load and heatup (mode 3) is judged to be extremely small. Furthermore, the boron dilution event is the only accident which could have significant safety consequence during this period. TVA believes that a seismic event postulated to occur during this time period which could potentially cause the HPFP piping in the auxiliary building to fall in the precise manner so as to initiate an uncontrolled boron dilution event and also disable the neutron source range indication is highly unlikely.

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 section 3.7.11.2 and 3.7.12.

The purpose of the fire detectors is to indicate undesirable combustion in the monitored areas. The safety function provided by the fire detectors can be achieved by establishing a fire patrol in the affected areas as prescribed in section 3.3.3.8 of the technical specifications.

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

There are approximately 120 unit-1-related work items on WBN to be completed after fuel loading of unit 1. Ten of these items affect the as-built plant in conformance with the FSAR and/or affect a TVA licensing commitment. These 10 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 120 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.
Weise Shyenbeb ECN 5246	Reanalyze residual heat removal (RHR) relief valve discharge lines inside containment and add hangers as necessary. - analyses done prior to fuel	This work is to analyze the RHR relief valve discharge lines downstream of the RHR relief valves for a higher temperature than currently analyzed. The relief valves can adequately discharge water up to 212°F in the current configuration. Higher temperatures would cause flashing in the discharge lines and introduce line movements which could cause potential valve damage. Since temperatures exceeding 212°F occur only as the reactor goes into mode 4, it is acceptable to defer completion of this work until initiation of mode 4.

OWIL Item Number	Description	Justification
ECN 5320	Modify Foxboro racks for addition of automatic low power feedwater control	This addition is for automatic low power control of the bypass feedwater valves. This is not a safety-related function. Therefore, deferral of this modification until after fuel loading is acceptable. prior to
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 Betwee: nits and the Use of Temporary . terations (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 a USQD performed per AI-2.18 or will have a safety evaluation 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 PORC-reviewed SI before fuel load. All required operating instructions will be written and approved.

SR 4.11.1.1.3 (SI-11.25) - Once per 12 months perform a gamma isotopic analysis on at least one sample of sediment from the holding pond. Since this is a check of the concentration of radioactive material released in liquid effluents to unrestricted areas, it is not required and not meaningful until some time after plant startup.

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 section 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 Regulatory Guide (Aug. 1975) will be submitted in conjunction with the inservice inspection program above.

SR 4.0.5 (SI-4.0.5.0) - This is only part of the ASME Section XI implementation. SI-4.0.5.0 is a visual inspection of pressurized piping at system conditions following startup after an outage and will not be written 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.

Additional technical specification items are still being discussed between TVA and Nuclear Reactor Regulation (NRR). Any additional surveillance instructions which result from the resolution of these issues will be prepared to allow performance before entry into required modes.

ENCLOSURE 2

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. The FSAR Chapter 14, Table 14.2-1, identifies 30 preoperational tests that are not scheduled to be completed until after fuel load. These tests are included as attachment I.

We have identified open items on 13 preoperational tests that will not be resolved until after fuel load. Attachment II identifies these tests and the open items associated with each test. Testing currently in progress could also identify additional items which will not be tested 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.

(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. when?

(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.

CV
PA
installed
but can't calibrate

(e) TVA-28 - Sampling System - The hydrogen analyzers will be installed and calibrated before initial criticality. A sampling capability check of the upper head injection (UHI) surge tank and water accumulator cannot be done until the accumulator is pressurized. 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. 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. Testing relative to the sodium analyzer can only be done when the steam generator reaches pressure and temperature during startup. All these tests will be completed before initial criticality. CV

(f). TVA-31A - Process Radiation Monitoring - Two radiation 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 - Equ ient for Replacement of Radwast 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 and performs no safety function.

(i). W-1.1 - Reactor Coolant System (RCS) Heatup for Hot Functional Testing - Steam Generators 1-4 reference level controllers (L-3-231,232,233,234) automatic response failed to meet test acceptance criteria. OE is evaluating calibration data submitted by Preop. No additional retesting is expected, and these items will be closed before initial criticality. The chemical and volume control system (CVCS) letdown pressure control valve PCV-62-81 will be changed out because it is not properly sized (Cv is too large) to control RCS pressure automatically with the RCS system water solid. With the present configuration, the RCS pressure can be controlled manually when it is water solid. The valve will be replaced and the solid system response retested when the new valve is supplied by Westinghouse.

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No pot. challe to pressure integrity

(j). W-1.2 - RCS Heatup for Hot Functional Testing - (1) Drawings will be revised to show the actual location of the acoustic valve monitoring system (AVMS) accelerometers for the Pressurizer relief valves (PORVs) PCV-68-334,340A upstream of the valves instead of downstream. This is a documentation item only and does not affect plant operation. (2) Calibration of the pressurizer level transmitter LT-68-320 will be reverified; the scaling calculations and the calibration procedures for the pressurizer level transmitter LT-68-325C and -326C will be checked for correctness; and static alignments will be performed on pressurizer level transmitters LT-68-325C and -326C. (3) 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. (4) Pressurizer safety valve bench testing data must be obtained from the vendor and included in the Preop test results package. (5) The Foxboro and Barton pressurizer level transmitters will be checked for agreement within their required accuracies during RCS heatup to normal operating temperatures. (6) Pressurizer pressure transmitter testing will be completed at normal RCS operating temperature. (7) 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 steam generator enclosures. Test items 2-6 above cannot be completed until the unit reaches operating pressure and temperature but will be completed before initial criticality. Test item 7 has been completed but will be rechecked after fuel load.

prior to FL

(k). 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.

(l). W-9.2 - Incor Thermocouple System - Performed at 250°F, 350°F, 450°F, and 557°F during heatup. Collect resistance data from spare 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 re-performed due to the replacement, as a system enhancement, of the incore thermocouples and modifications to the cables and reference junction boxes before criticality.

(m). W-10.4 - Liquid Waste Processing - All of the required testing associated with unit 1 is complete. The two remaining open items on this test will be cleared when the modifications on the waste and auxiliary waste evaporator packages are complete under ECN 3713. The modifications will relocate the waste condensate tanks and install a new hyperfiltration system. The new equipment is not required to be operational until unit 2 operation.

(n). 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.

ENCLOSURE 3

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

Item No.	Justification
WBNMEB8107, et. al CDR 81-66	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 10CFR50.49(i). This justification was submitted to the NRC by letters dated August 19, 1983, July 24, 1984, and December 20, 1984. NRR
WBNNEB8208 Ralph Shell Delete last sentence 2/25/85 , CDR 82-36?	This item involves the accuracy of wide-range RCS pressure transmitters. Westinghouse has advised TVA that all hardware required for installation of the new instrument channels cannot be procured, delivered, and installed before fuel loading of WBN unit 1. New RCS transmitters will be installed outside containment before initial criticality. ~~TVA's WBN emergency operating instructions take into account the wide-range pressure transmitter error and will allow safe operation of the plant in the interim period between fuel loading and the first refueling outage~~.
WBNNEB8335 84-03	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.
NRR	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.

5760, 5761, WBNEEB8422, & WBNMEB8430	All of these items deal with inadequate separation of redundant safe shutdown circuits (i.e., 10CFR50, Appendix R, specifications were not met). TVA is rerouting or otherwise protecting these circuits to comply with 10CFR50, Appendix R. These safe shutdown circuits must be protected from fire damage to assure a controlled safe shutdown from operating conditions in the event of a fire. Before entry into mode 3, the need to protect safe shutdown circuits is moot since the plant is already shut down and no fission products exist in the core or reactor coolant. Therefore, TVA will complete these modifications before proceeding into mode 3 during initial plant startup. For additional details refer to the discussion of outstanding Appendix R items in enclosure 1.
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.

CDR 84-29

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.

CV where are valves?

can manual action be taken if high temp cond exists?

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 rated current, these cables can operate for 8000 hours (0.913 year) 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 9/13/85. TS