ATTACHMENT #1

SIMPLIFIED ANALYSIS

PROBLEM NO. ____________

PREPARED BY ____________ DATE ____________

CHECKED BY ____________ DATE ____________

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ATTACHMENT # 2

TVA

WATTS BAR NUCLEAR PLANT SIMPLIFIED ANALYSIS CHECKLIST	WBN-SAH-400

PROBLEM ____________________

WATTS BAR NUCLEAR PLANT

SIMPLIFIED ANALYSIS PIPING CHECKLIST

USING TPIPE PROGRAM

(CLASSES 2, 3, AND NONSAFETY ANALYSIS)

Analyst ____________________ Date ____________

Checker ____________________ Date ____________

I. INTRODUCTION

This check package is to assist the analyst in assuring that all calculations, qualitative decisions, analytical revisions, and physical representations are correct and accurate. The prerequisites for completing this check are that the checker be qualified and work independently, and that the program used is TPIPE in conjunction with the TPIPE SPECIAL POSTPROCESSOR.

TPIPE's extensive data checking program utilizes the computer to isolate as many errors in coding procedures as possible. However, there are many areas that cannot be checked by the computer and must be checked manually. The attached checklists enable an individual to systematically check the coding data for errors and omissions and to eliminate repeating checks made by the computer. These lists are to be followed and each item checked off as completed. NA may be marked beside those items which are not applicable and ABC for those which have already been checked.

This checklist is a guide, and the completion of this checklist does not relieve the preparer or checker of their responsibility to ensure a quality product. The supervisor is responsible for engineering and his approval ensures sound engineering.

TVA

WATTS BAR NUCLEAR PLANT SIMPLIFIED ANALYSIS CHECKLIST | WBN-SAH-400

PROBLEM ______________________

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USING TPIPE PROGRAM

(CLASSES 2, 3, AND NONSAFETY ANALYSIS)

Analyst ______________________ Date ______________

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________________ Date ______________

Checker ______________________________ Date ______________

II. CHECKLIST TOPICS

The following is a list of major topics covered by this checking procedure. The calculation package should include only those topics applicable to the analysis being checked. Those topics which are not applicable should be identified as such.

_____ 1. Geometry check - Checks the physical data and coding.

_____ 2. Load case determination - Determines the proper loading conditions.

_____ 3. Static check - Ensures that the static runs are correct.

_____ 4. Dynamic check - Ensures that the dynamic runs are correct.

_____ 5. Postprocessor check - Checks that the analysis limits are established correctly.

_____ 6. Anchor check - Ensures that the anchor load program, "-Anchor," is correct.

_____ 7. Equipment check - Hand calculations that ensure equipment nozzle qualifications.

_____ 8. Valve check - Checks valve accelerations and active valve qualifications.

_____ 9. Miscellaneous check - Checks branch line data, clearances, and special design requirements.

_____ 10. Isometric check - Checks isometric drawings for supports, dimensions, and accuracy.

_____ 11. Flange check - Checks that all flanges are qualified.

_____ 12. System calculation package - Checks that the calculation package contains all significant information.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date __________

Checker ____________________ Date __________

III. CHECKLIST 1 - GEOMETRY CHECK

A. General Requirements

This list is to be checked after the TPIPE check program (Tcheck or TTcheck) has been completed for the piping system. The system model must include, but not be limited to static weights, joints, and/or restraints at tentative support locations, dynamic mass point spacing, system anchors, and material and cross-section input.

B. Geometry Check

1. Data Tables

___ a. Check Table of Design Modes and Operating Conditions and ensure all modes are entered and that operational modes are provided in the design data section.

___ b. Check that valve/flange data is provided in the design data section.

___ c. Check that all sleeve locations are known and the type of seal is known. (Note: Link seals must be modeled as three way supports.)

___ d. Check Table of Pipe/Component Cross Section Data.

2. Programs

___ a. Check that the first page of computer printout has the following information: name of plant, name of system, system number, date, and problem number.

___ b. Check that there is one material card for each pipe mode, and check:

1. The operational modes match those in the operational modes design data section.

2. Young's modulus for the cold and hot condition of the material.

3. Poisson's ratio.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

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4. Coefficient of thermal expansion and temperature.

___ c. Check that the cross sections properties are in agreement with the Table of Pipe/Component Cross Sections.

___ d. Check that the Control Point Specification (CPS) card names, coordinates, and pipe radii (if applicable) are correct.

___ e. Check that modeling and coordinates of all pipe, branches, instrument lines, anchors, reducers, valves, etc., are correct.

___ f. Check that lump masses agree with the weights given in the design data section.

___ g. Check nonglobal coordinate system definitions.

___ h. Check that member list includes all members and that each node is used the proper number of times.

___ i. Check members for proper materials, cross section, SIF, and generation code.

___ j. Check that all warning messages are acceptable and that no correction should be made.

___ k. Check that all curved member angles are correct.

___ l. Check that member end releases, if included, are correct. (Probably on tieback models.)

___ m. Check the Member Table to ensure that all runs are in the proper direction and do not double back over other members. (This may not show up on the plot.)

___ n. Check that all revisions shown on the isometrics that did not require reanalysis have been incorporated in the new analysis.

___ o. Check that all DCRs, ECNs, FCNs, and FCRs are incorporated in latest revision.

___ p. Check that lap regions extend into next problem according to Policy Statements 1, 2, 9, and 10.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date __________

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___ q. Check that all revisions of physicals or HEU isometrics since the last analysis are incorporated in this analysis and that the problem boundaries are clearly identified.

___ r. Ensure that insulation requirements given in the design data section have been analytically considered.

___ s. Ensure that all changes in the calculation package which affect the analysis and have been evaluated since the last reanalysis are incorporated into this analysis.

Support Check

C. Support Code Checklist

___ 1. Check that all the supports are in the correct zone.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

IV. CHECKLIST 2 - LOAD CASE DETERMINATION CHECKLIST

A. General Requirements

This section is a check to determine that all required load cases are identified. The load case definition table in the design calculation needs to be checked at this time.

B. Load Case Determination

___ 1. A gravity load case is required. If the piping is sometimes full and sometimes empty, two load cases may be required.

___ 2. Check that all the thermal modes of operation (normal, upset, and faulted) are independently prepared and checked and placed in the design data section.

___ 3. If any type of bellows exist in the analysis refer to the technical supervisor in charge for analytical instructions.

___ 4. If a preload loadcase exists in the analysis refer to the technical supervisor in charge for analytical instructions.

___ 5. If piping that is being analyzed is supported from two different structures, a seismic anchor point analysis is required. If the analysis attaches to another pipe and the attachment point is considered an anchor, SAM is required.

___ 6. Earthquake and rigid response analysis is required on all problems.

___ 7. For relief valves which relieve vapor to the atmosphere either directly or through a process pipe, check that the Functional Group has determined its significance. In cases where it is determined significant, the loading must be considered in the qualification.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date __________

Checker ______________________ Date __________

V. CHECKLIST 3 - STATIC CHECK

A. General Requirements

This list is to be checked after the geometry and all other physical characteristics of the system have been checked (see checklist 1) and corrected on final geometry TPIPE check run.

B. Thermal Check (Perform for each thermal mode of operation.) (T1, T2)

1. Tables

___ a. Check anchor movement calculations for the usage of the correct material, dimensions, temperatures, thermal expansion data, direction of movement, assumptions, and calculations.

2. Computer Program

Mode 1	Mode 2	Mode 3		
___	___	___	a.	Check that control card has thermal support configuration.
___	___	___	b.	Check material change deck for the correct E_{COLD}, coefficient of thermal expansion, T_{OP}, and E_{HOT} on all load case sets. Ensure there is one material card for each mode.
___	___	___	c.	Check that the correct temperature has been input on member cards.
___	___	___	d.	Check operating conditions. If more than one thermal run is made or an anchor movement load case (EDO, EDS) exist concurrently with a thermal case, then a thermal range calculation must be performed in the postprocessor.
___	___	___	e.	Check load case cards for appropriate load factors and support type (T).

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

___ ___ ___ f. Check that anchor movements are input correctly. Check magnitudes and directions of movements.

C. Gravity Load Case (DW)

___ 1. Check that the gravity support configuration (G) is used.

___ 2. Check that the weight factor is 1.0 and all other factors are zero.

D. Preload Load Case (PL)

___ 1. Consult the responsible technical supervisor for analytical considerations of preload.

E. Bellows Pressure Load Case (BL)

___ 1. Consult the responsible technical supervisor for analytical considerations of bellows pressure.

F. Relief Valve Load Case (VT)

___ 1. If significant consult the responsible technical supervisor for analytical considerations.

G. Seismic Anchor Point Movements - EDS and/or EDO

___ 1. Check that the dynamic support configuration (D) is used.

___ 2. Check that the largest applied movement for each direction for each structure applicable to this problem is determined.

___ 3. Check that the coordinate system for the SAM movement for the structure is identified.

___ 4. Check that all the movements are combined properly and applied to all points on the structure.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date __________

Checker ______________________ Date __________

___ 5. Check that the sign on the displacement is changed when attaching to the adjacent structure.

H. <u>Rigid Response Check</u>

___ 1. Check that rigid response loadcases were evaluated.

___ 2. Check that input acceleration values were chosen from the proper spectra curve at 33 Hz.

___ 3. Check that individual loadcases were evaluated for each orthogonal direction.

I. <u>Frequency Analysis Check</u>

___ 1. Check that cut off frequency is 33 Hz.

___ 2. Check that the frequency analysis runs to 33 Hz.

___ 3. In the frequency analysis section, check that the largest off diagonal term in the generated mass matrix is less than 0.1.

___ 4. In the frequency analysis section, check that the diagonal terms in the generalized mass matrix are 1.0.

___ 5. For subspace iteration, check for summary of warnings.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date ____________

Checker ____________________ Date ____________

VI. CHECKLIST 4 - DYNAMIC CHECK

A. General Requirements

This check is to be performed after system checks 1 and 2 have been made and the static and thermal support configurations are analyzed and proven.

B. Dynamic Check - Seismic

Check the dynamic response spectrum load cases for the following:

___ a. Check that the acceleration spectra curve data is from "-Frams" or correct curve headings are input for TPIPE versions 46D or later.

___ b. When the Envelop Program is used, check that each spectra file contains data and that the spectra is enveloped properly.

OBE		SSE		
XY	YZ	XY	YZ	
___	___	___	___	1. Check that the proper building spectra is used for conservative results.
___	___	___	___	2. Check that all spectra in one structure are enveloped for each direction. Documentation is required to show what spectra were enveloped.
___	___	___	___	3. Check that zones are established only for independent structures.
___	___	___	___	4. Check that the correct spectra is used in each zone in the proper direction.
___	___	___	___	5. Check that dynamic support configuration is requested by placing a D in column 2.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

OBE		SSE		
XY	YZ	XY	YZ	
___	___	___	___	6. Check that the correct spectra combination was used (vectorial sum).
___	___	___	___	7. Check that the GM option is used.

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WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

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VII. CHECKLIST 5 - POSTPROCESSOR CHECK

A. General Requirements for TPIPE Postprocessor

This list is to be checked after the analysis has been performed to ensure that the analysis limitations being evaluated have been met. The following items must be checked to assure consistent postprocessing results:

1. Those analyses utilizing any input restart tapes must have identical support configurations for all load cases. If multiple support excitation option is used check for TPIPE warning messages.

___ a. Warning messages are justified.

2. Those analyses utilizing two or more input restart tapes must have the following geometry data unchanged from one restart tape to another:

___ a. Node names.

___ b. Node locations.

___ c. Node name sequence.

___ d. Nonglobal coordinates.

___ e. Member name sequence.

___ f. Member connectivity.

B. Postprocessor Check

1. Computer Program

___ a. Check that the first sheet of the computer printout describes the plant, system, system number, and date.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

___ b. Check that the threshold stress ratio is 1.0 or less on MCC.

___ c. Check that the VS travel of 1.0" is input to change VS to CS on MCC.

___ d. Check that the correct class, design pressure, allowable stress, and minimum yield stress are being used for each design mode on materials cards.

___ e. Check that the following linear support stress constants are requested:

(1) SA8 = 1.0

(2) SA11 = 1.5

(3) SA9U = 1.0

(4) SAU9 = 1.5

(5) SA9E = 1.33

(6) SAE9 = 1.60

(7) SA9F = 1.33

(8) SAF9 = 1.6

(9) AVC = 0.76

___ f. Check that all unique load case titles are input so that the data is manipulated correctly.

___ g. Check that LG3 scale factors associated with LG4 and LG6 are correct.

___ h. Check that all support override cards are justified.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

___ i. Check that no snubber has design travel less than or equal to 1/16". If any do, other than existing supports, they should be changed to rigid support types; use support override card to do this. If snubber existed in previous analysis, disregard this check. Note: Where snubbers exist on valve operators, they must not be changed to RR support types in the postprocessor.

Postprocessor Run 1 Thermal Range Evaluation and Run 2 Support Load Calculations

Run 1 Run 2

___ ___ 1. Check that the output restart tapes are correct with the latest and correct load case runs.

___ ___ 2. Check that the material allowables are input correctly according to guidelines (not necessarily the true values).

___ ___ 3. Check that the correct data is suppressed.

___ ___ 4. Check that the threshold stress ratio is 1.0.

___ ___ 5. Check that the linear support stress constants agree with those in WBN-SAH-207.

___ ___ 6. Check that the VS to CS conversion travel is 1".

___ ___ 7. Check the new label definition for correct scale factor, usage of the correct files, the correct manipulation code and correct reversing or nonreversing input data to match the guidelines.

___ ___ 8. Check that the correct load groups are used. See WBN-SAH-207.

___ ___ 9. Check that the unique load case title is input so that the data is manipulated correctly.

___ ___ 10. Check that the LG3 scale factors associated with LG4, LG5, and LG6 are correct.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

VIII. CHECKLIST 6 - ANCHOR CHECK (For Simplified Analysis Side Only)

A. General Requirements

This list is to be checked after the analysis has been run to ensure that the anchor loads and type will be evaluated.

B. Anchor Check

1. Computer Run for Each Anchor to be Built

___ a. Check that the first sheet of the computer printout describes the plant, system, and problem number.

___ b. Check the joint identification card for:

(1) Joint name that the anchor is to be issued under.

(2) The correct output round option is used.

(3) The final results scale factor is correct.

(4) The comments describe the source of data of the other side of the anchor.

___ c. Check coordinate definition data.

(1) Check that each anchor is defined on both sides (comments or notes).

(2) Check that the direction cosines, plant global X, Y, and Z coordinates, and a point on the pipe axial to the anchor for each side have been correctly transfered from the TPIPE input.

___ d. Check the load source input data.

(1) Check that all load sources are considered for each anchor joint.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

(2) Check that the load entries have been correctly transferred from the TPIPE output.

(3) For the secondary load file cards, ensure a 1 is entered as the scale factor for the SCP, SCT, and SDM variables. Default is zero which will preclude thermal from being considered.

___ e. Check that all scale options are correct.

___ f. Check that anchor loads ≤ allowable loads for forged anchor only; REF. CEB 76-16.

___ g. Check that loads and stresses for each branch of the TEE anchor are ≤ 1/3 of the allowable. (If the branch is the last to be qualified, then the total load should be checked for revisions and qualifications.)

___ h. Check that the anchor load computer printout is placed with the analysis printout.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

IX. CHECKLIST 7 - EQUIPMENT CHECK

A. Equipment Check

This section includes the check of nozzle qualification and equipment.

1. Hand Calculations

___ a. Check that equipment and nodal points have been recorded.

___ b. Check that current and correct load source data is being used.

___ c. Check that (DW + TH) is the larger of /DW/ and /DW + TH/.

___ d. Check that all allowables and associated combinations are correct and source referenced.

___ e. Check that all load combinations are performed correctly.

___ f. Check that all allowables are met or justified. (May use Component Analysis Section. All nonretrievable correspondence must be included in calculation package.)

___ g. Source of allowables is referenced or included.

___ h. Check that the correct local or global coordinate system is shown.

___ i. Check that hand calculations are correct where CEB 82-1 was used to calculate nozzle allowables and/or to qualify nozzles.

PROBLEM ____________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date __________

Checker ____________________ Date __________

X. CHECKLIST 8 - VALVE CHECK

A. Valve Accelerations. (Tabulation required only if data extraction not used.)

___ 1. Check that valve accelerations are tabulated for each joint associated with a valve, valve operator, or flow element valves by joint name.

___ 2. Check that valve accelerations do not exceed the allowable limit, or are qualified by the Component Analysis Section (CAS).

B. Active Valve Qualification

___ 1. Check that the valve qualification table is filled out for the pipe that attaches to each end of every active valve.

___ 2. Check that an active valve stress ratio of 0.76 S_y is used for standard valve or a ratio of 1.0 S_y is used for swing check valve for each active valve on the pipe side of the valve.

___ 3. Check that stresses for active valves are within allowables, or approved by CAS.

C. Supports on Operators

___ 1. Check that loads for valve operator supports have been sent to the Component Analysis Section (CAS). Note: Support design loads which do not exceed three times the weight of the extended structure of the valve need not be sent to CAS.

___ 2. Ensure that documentation from CAS states valve is qualified.

PROBLEM ____________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date ____________

Checker ____________________ Date ____________

XI. CHECKLIST 9 - MISCELLANEOUS CHECK

1. Branch Line Data

___ a. Check that all branch connection data drawings are completed.

___ b. Check that new movements from this analysis, which effect any other interface, are used in evaluating the effects on that interface.

Note: This evaluation is a part of the documentation of the appropriate interface.

___ c. For B001s, check that all pipe stresses are within allowable stress. Check that instrument attachment node stress differences are greater than the following:

1800 lb/in^2 for eq 9U

3350 lb/in^2 for eq 9F

8150 lb/in^2 for eq 10

___ d. For B001s check that the valve and instrument attachment point accelerations do not exceed 2g vertical and 3g horizontal for SSE.

2. Pipe Clearances

___ a. Check that maximum pipe movement does not cause the pipe to come in contact with any sleeve. Assume pipe to be centered in the sleeve.

3. Maximum Pipe Movements

___ a. Check that maximum pipe movement does not exceed 1 inch. If pipe movements exceed 1 inch it must be checked by OE onsite for possible interferences.

4. Support Load and Anchor Load Table

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

___ a. Check that the correct support load types are shown on the correct support load table form.

___ b. Check that the title and unit number are correct.

___ c. Check that the direction on the load table exactly match the computer printout and isometric.

___ d. Check that the type of support is documented as instructed in WBN-SAH-301.

___ e. Check that all special design requirements are noted.

___ f. Check that all the support loads are revised to reflect the unrounded values.

___ g. Check that lap region support loads are enveloped with other problems.

___ h. For tieback supports, check that all tieback loads are included on the support load table, plant relative movements are tabulated, and a minimum section modulus requirement note is included on support load table.

___ i. If zones are used, the support load tables must identify what structure the support must be attached to, and the zone number used in the analysis.

___ j. Check that SPIPE runs used to generate tieback design loads are in the calc package or with the analysis computer printout.

___ k. Check that lug stress for equation 11 has been calculated correctly. (See WBN-SAH-301)

5. Special Stress Qualifications

___ a. Check that the reducing elbow stress has been properly calculated and intensified. (See WBN-SAH-312)

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

XII. CHECKLIST 10 - ISOMETRIC CHECK

A. Isometric Drawings (HEU drawings or sketch)

___ 1. Check that all supports are present, dimensioned, and has a circle around its name. In lap regions, check that support is circled on one problem or the other.

___ 2. Check that the dimensioning of all supported joints shown are within ±1" of the computer program coordinates.

___ 3. Check that nonglobal coordinates systems are defined.

___ 4. Check that all support directions correspond with th[illegible] program and are shown dimensioned properly.

___ 5. Check that the type of support is documented as instructed in WBN-SAH-301.

___ 6. Check that a note denotes to which building or structure the support must be attached unless designated on the support load table.

___ 7. Check that unit 2 analysis detailing and references are clearly defined.

___ 8. Check drawing continuation notes for accuracy.

___ 9. Check that interface analysis reference note(s) are shown on the drawing and are correct.

___ 10. Check that special support requirements are noted.

___ 11. Check that terminal points which connect problems or systems are cross referenced.

___ 12. Check that all tiebacks are identified with a support circle and the directions for support are clearly defined. A note should also be added to the isometric identifying all tieback supports.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

___ 13. Check that direction reference vectors are included for global coordinate system.

___ 14. Check that all joints are shown.

___ 15. Check that all valves, flow elements, operators, instrument lines, etc., are shown and referenced.

___ 16. Check that mode numbers are shown and pipe size, schedule, elevation, and insulations are adequately identified.

___ 17. Check that wall sleeves, floor sleeves, and penetrations are shown. Reference sleeve number.

___ 18. Check that all lines are dimensioned correctly.

___ 19. Check that plant units are correctly identified.

___ 20. Check that equipment is identified and located by centerline elevation and dimensions.

___ 21. Check isometric against plot to ensure regions desired are present and correct.

___ 22. Check that for each penetration the number, azimuth, elevation, and radius is shown.

___ 23. Check that lap region is shown correctly on the isometric.

___ 24. Check that all skewed supports have their positive line of action indicated with a + symbol.

___ 25. Check that a node is shown on the isometric at the B001 intersection point.

___ 26. Check that all termination points are located by column lines.

___ 27. Check that redline isometrics are prepared for Construction and OASES groups with all proposed support locations clearly identified and all modifications to existing supports noted and circled in red.

PROBLEM ____________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ____________________ Date __________

Checker ____________________ Date __________

___ 28. Check that the isometric in the calculation package agrees with both Construction and redline isometrics.

___ 29. Check that the standard note has been placed on both the Construction and OASES redline isometrics: "See OE calculation package __________ for documentation and evaluation of this isometric."

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

XIII. CHECKLIST 11 - FLANGE(S) CHECK

A. General Requirements

The checking of flange stresses must be done manually. Using flange stress evaluation report CEB-ME1-77-5. This checklist is to be done after all flanges have been evaluated.

B. Data Tables

___ 1. Check that a flange stress worksheet is completed for all applicable flanges.

___ 2. Check that correct data is used and combined properly.

___ 3. Check that each table is completed and signed checked.

___ 4. Check that all flanges are qualified.

___ 5. Check that data is used from the same I or J end of the same member for each flange.

PROBLEM ______________________

WATTS BAR NUCLEAR PLANT CHECKLIST (Continued)

Analyst ______________________ Date ____________

Checker ______________________ Date ____________

XIV. CHECKLIST 12 - SYSTEM SUMMARY BOOK

A. Analysis Calculation Package Check

___ 1. Check that the system summary book contains the following components: table of contents, check lists, summary, correspondence section, table of pipe properties, loading conditions evaluated, hand calculations, design data section, and response spectra data.

___ 2. Check that each component of the summary book has been completed signed as being checked, and the current revision number is entered.

___ 3. Check that the summary book is clear and easy to understand.

___ 4. Check that all certification exceptions to the analysis (e.g. nozzle overload, valve accelerations etc.) are documented in the calculation package and data base.

___ 5. Check that all special procedures and analysis techniques are documented in the calc package.

___ 6. Check that a list of all the B001s are included in the calc package.

___ 7. Insure the ECNs for the revision are noted.

___ 8. In the revision log, add "Reanalyzed" if problem was reanalyzed or "Documentation Change" when reanalysis is not required.

NRC

TVA 64 (OS-9-65)
UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : S. Schum, QTC-ERT Program Manager, WBN CONST

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : October 4, 1985

SUBJECT: TRANSMITTAL OF ACCEPTED FINAL REPORTS

The following final reports have been reviewed and accepted by NSRS and are transmitted to you for preparation of employee responses:

IN-85-325-006 (I-85-222-WBN)

MKidd for
K. W. Whitt

Please acknowledge receipt by signing below, copying and returning this form to J. T. Huffstetler, E3B37 C-K.

Name Date

Attachments
cc: W. F. Willis, E12B16 C-K (4)
J. W. Coan, P-104 SB-K
H. N. Culver, W12A19 C-K
E. R. Ennis, WBN
Guenter Wadewitz, WBN

REP07:G5

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-222-WBN

Milestone 1 - Fuel Load

SUBJECT: ERT ITEM NO. IN-85-325-006 - VALVE CONFIGURATION CONTROL

INVESTIGATOR: J. B. Rollins ______ Sept. 30, 1985
J. B. Rollins — Date

REVIEWER: G. G. Brantley ______ Sept 30, 1985
G. G. Brantley — Date

APPROVED BY: M. A. Harrison ______ 10/1/85
M. A. Harrison — Date

FINAL

I. BACKGROUND

NSRS has investigated employee concern IN-85-325-006 which Quality Technology Company identified during the Watts Bar Employee Concern Program. The concern is worded:

> Inadvertent valve operation during Unit 1 hot functional testing, resulting in a nonradioactive water spill, would have caused a radioactive spill had the plant been in operation. It was expressed that valve control and operator training have not improved since the incident.

Additional infomration was requested from the Quality Technology Company. None was obtained. During this investigation the specific instance of valve operation was not identified.

II. SCOPE

Valve configuration control and related operator training improvement since Unit 1 hot functional testing was determined to be the primary concern. This concern was investigated by contacting applicable personnel and reviewing documentation relating to valve configuration control. NSRS reviewed reports, procedures/instructions, and training documents that had been issued since Unit 1 hot functional testing.

III. SUMMARY OF FINDINGS

Based upon a review of applicable documents and interviews with appropriate personnel, the specific findings listed below were identified.

A. Audits and Reports

1. NRC Report 50-390/84-59, item 05, issued November 8, 1984, identified that procedures which implemented system configuration control and independent verification were not clearly written and that training was inadequate for operations personnel using the applicable procedures. In response to that item configuration control procedures were revised, and operations personnel were trained on the revised procedures. Upon review of the corrective action taken, the NRC closed this item.

2. NSRS reported problems with configuratin control and independent verification in NSRS Report R-84-15-WBN, item 03, issued December 27, 1984. The report recommended that Operation Section Instruction Letter (OSL) A-2 and Administration Instruction 2.19 be reviewed with all operations staff. This review was completed by the Operations Section and training for operations personnel was conducted.

3. Quality Audit Branch (QAB) Audit Report QWB-A-85-006, issued March 14, 1985, discussed the audit of instructions for establishing and maintaining system status. No deviations were identified; however, areas for instruction improvement were discussed with the operations staff. QAB personnel contacted about this report stated that this area would be audited annually for the next few years.

B. WBNP Procedures/Instructions

1. A review of WBN instruction OSLA-2, "Maintaining Cognizance of Operational Status," identified that four revisions to improve clarification and implementation had been made since Unit 1 hot functional testing.

2. A review of WBN instruction AI-2.19, "Independent Varification," identified that three revisions were made to identify criteria and performance provisions, clarification improvement, and to expand coverage of the instruction after Unit 1 hot functional testing.

C. Interview Information

1. Operations management personnel stated that problems with valve configuration control had been brought to their attention due to increased regulatory and audit activity in this area. The plant operations section requested the Office of Quality Assurance (OQA) to provide assistance in this area by working with the plant staff to identify and resolve procedural and inplementation problems. Procedures were revised as necessary, and applicable training was conducted.

2. Operations training personnel stated that required operations personnel had received training on the revised procedures and further training on procedures implementation was ongoing through shift safety meetings. NSRS reviewed the training lesson plans and records of the training sessions which appeared to be adequate.

3. As a result of a request from the Operations Section the Plant Quality Assurance (PQA) organization performed a survey of valve configuration control. Deviations were identified, and corrective actions were initiated, as applicable. PQA personnel informed NSRS that a followup survey of system configuration control is scheduled for the near future.

IV. CONCLUSIONS/RECOMMENDATIONS

Conclusion

The employee concern was not substantiated. Since Unit 1 hot functional testing, the WBN staff had taken definite actions to identify and correct problems with valve configuration control and to improve the quality of applicable plant procedures. Training had been conducted for operations personnel in the use of the improved procedures.

Additionally, future QAB audits and PQA surveys of these activities are scheduled.

Recommendation

None.

NRC

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Guenter Wadewitz, Project Manager, OC-WBN

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : October 8, 1985

SUBJECT: NUCLEAR SAFETY REVIEW STAFF INVESTIGATION REPORT TRANSMITTAL

Transmitted herein is NSRS Report No. I-85-398-WBN

Subject Fire Protection System Hydrostatic Test

Concern No. IN-85-534-005

and associated recommendations for your action/disposition.

It is requested that you respond to this report and the attached recommendations by November 1, 1985. Should you have any questions, please contact M. A. Harrison at telephone 6328-K; 3715-WBN.

Recommend Reportability Determination: Yes ____ No X

M Ridd for

Director, NSRS/Designee

cc: W. F. Willis, E12B16 C-K (4)
J. W. Coan, P-104 SB-K
H. N. Culver, W12A19 C-K
E. R. Ennis, WBN
QTC/ERT, CONST-WBN

--Copy and Return--

To : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

From:

Date:

I hereby acknowledge receipt of NSRS Report No. ____________

Subject ______________________________

for action/disposition.

Signature ____________ Date ____________

(Please copy entire page for return)

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

NSRS INVESTIGATION REPORT NO. I-85-398-WBN

Milestone 1 - Fire Protection System Hydrostatic Test

SUBJECT: ERT ITEM NO. IN-85-534-005

INVESTIGATOR: J. C.Catlin 10/2/85
J. C. Catlin Date

REVIEWER: P. B. Border 10/2/85
P. B. Border Date

APPROVED BY: M. A. Harrison 10/2/85
M. A. Harrison Date

DRAFT

I. BACKGROUND

An investigation was conducted to determine the validity of an employee concern received by QTC on August 15, 1985. The concern stated, "The Unit 1 fire protection hydro was conducted improperly. The test pressure was maintained throughout the test by running the pump. This happened three years ago."

II. SCOPE

The scope of the investigation included reviews of documents and personnel interviews to determine applicable requirements, specifications, and procedures; and similar reviews and interviews to determine test methods and results. Since no specific fire protection block was designated, the investigation included both the sprinkler systems and the fire protection water feed.

III. SUMMARY OF FINDINGS

A. Requirements and Commitments

1. Regulatory and FSAR Requirements

a. 10CFR50.48, Fire Protection

b. 10CFR50, Appendix A. Criterion 3, Fire Protection

c. FSAR, Paragraph 9.5.1.1, Criterion 8 (includes NFPA Codes by reference)

2. a. The sprinkler system has been designed in compliance with National Fire Codes Specification NFPA 13 - Standard for the Installation of Sprinkler Systems - 1976 Edition. It has been designed as a dry-pipe system.

b. The fire protection system upstream of the shutoff valve is a wet-pipe system. It also has been designed in compliance with NFPA 13, 1976, as described by paragraph 1-3.

3. TVA has issued WBNP-QCT-4.37 as a procedure for hydrostatic testing of piping systems at Watts Bar Nuclear Plant.

a. According to some of the test reports, QCT-4.37 R0 was in effect for testing Unit 1. The actual tests were performed to QCT-4.37 R2 or R3, depending on the timeframe.

b. QCT-4.37 has included by reference the requirements of ASME Section III and ANSI B31.1, Power Piping Code, as well as NFPA 13.

c. QTC-4.37 R3, paragraph 6.2.6.6.7 states that "Fire suppression systems to be buried or embedded are hydrostatically tested prior to backfilling or embedment."

4. By requiring the strictest specified pressures and durations for hydrostatic tests from the three codes noted, TVA has specified that all fire protection system piping be tested at 1.5 times design pressure, and for a period of two hours. The requirement is that there shall be no external leakage under these conditions.

B. Findings

1. Sprinkler Systems

a. The basic design of a dry-pipe sprinkler system has made it very difficult to contain internal leakage at the hydrostatic test pressure. The test criteria, accordingly, permit internal leakage through valves and other components.

b. The method used for performing the sprinkler systems hydrostatic tests was to fill the lines and to pressurize the system at operating pressure using the system line pump. The hydrostatic test pressure was reached and maintained using an air-turbine-driven reciprocating auxiliary pump.

c. There was no lower level test procedure which included or described utilizing this pump.

d. There is no restriction in any of the applicable codes against maintaining hydrostatic test pressure with an auxiliary pump.

2. Feed Piping

a. Fire protection piping upstream of the shutoff valve normally was hydrostatically tested using an auxiliary pump only to build up test pressure and holding test pressure for two hours without leaks.

b. Hydrostatic testing of the underground portion of the fire protection system was conducted prior to embedment.

IV. CONCLUSIONS AND RECOMMENDATIONS

A. Conclusions

1. The stated allegation of concern was not substantiated for the following reasons:

   a. Test Procedure WBNP-QCT-4.37 used for this testing references Fire Protection Code NFPA-13, ASME Section III, and ANSI-B31.1.

      (1) ANSI B31.1 requires pressurization to 1.5 times design pressure for an established minimum period of time.

      (2) NFPA-13 requires that the system be pressurized for two hours.

      (3) All three codes require visual inspection with any external leakage unacceptable.

   b. Internal leakage is permissible during this hydrostatic test.

   c. Running a pump to maintain system pressure is permissible under all applicable codes and standards and may be necessary due to internal leakage.

   d. The Unit 1 fire protection system test was conducted properly by maintaining system pressure throughout the test by running an auxiliary pump.

2. During the investigation, other items not relevant to the allegation were identified concerning the test procedure QCT-4.37 revision level and lack of detail in the procedure concerning the use of an auxiliary pump. These generated the following recommendations.

B. Recommendations

<u>I-85-398-WBN-01</u>

Construction should generate an addendum to QCT-4.37 which shows the procedure for the use of an auxiliary pump to maintain system test pressure during certain hydrostatic tests.

<u>I-85-398-WBN-02</u>

Hydrostatic test reports should be reviewed and corrected to reflect the correct revision number of WBNP-QCT-4.37 used to conduct the test.

NRC

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : S. Schum, QTC-ERT Program Manager, WBN CONST

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : October 8, 1985

SUBJECT: TRANSMITTAL OF ACCEPTED FINAL REPORTS

The following final reports have been reviewed and accepted by NSRS and are transmitted to you for preparation of employee responses:

WI-85-055-001

WI-85-056-001

K. W. Whitt

Please acknowledge receipt by signing below, copying and returning this form to J. T. Huffstetler, E3B37 C-K.

Name Date

Attachments

cc (Attachments):

J. W. Coan, P-104 SB-K
H. N. Culver, W12A19 C-K
E. R. Ennis, Watts Bar Nuclear Plant
G. Wadewitz, Watts Bar Nuclear Plant
W. F. Willis, E12B16 C-K (4)

REP07:G5

QUALITY
TECHNOLOGY
COMPANY

P.O. BOX 600
Sweetwater, TN
37874

ERT INVESTIGATION REPORT

CONCERN NO: WI-85-055-001
WI-85-056-001

CONCERN: See Details Below

INVESTIGATION
PERFORMED BY: William Kemp, Jr.

DETAILS:

WI-85-055-001

The welder certification test presently being administered to welders at Watts Bar, in the recertification efforts following a recent stop work order (No. 25) is not in compliance with Code (ASME Section 9).

WI-85-056-001

CI was told, (by welders who are in the process of retesting), that they are being tested on flat plate, in the flat position, for welding pipe using the TIG & SMAW processes. This is not in accordance with ASME code requirements.

DOCUMENTATION/CODES/REQUIREMENTS REVIEWED

FSAR 3.1
ASME Code Section III & IX
Stop Work Authority # 25 Rev. 1 issued 8/23/85
NCR 6277 issued 8/26/85
NRC Letter Docket 50-390/50-391
Watts Bar Nuclear Plant Informal Memo dated 8/28/85
Letter L44-85-0910-804 Response to confirmation of action letter dated September 11, 1985, to Dr. J. N. Grace, Region II NRC, from H. G. Parris
Memorandum C01-85-0903-004 Confirmation of Action Letter Welder Certification Program, from G. Wadewitz to J. W. Coan
ANI SIS Report, dated 9/18/85
Memo Watts Bar Code Welding (TOO 850823 916) dated August 23, 1985 stated: "Temporary suspend all code welding" IE: Critical Structure Systems Components welding

PERSONNEL CONTACTED Confidential

SUMMARY OF INVESTIGATION

These concerns are substantiated.

On 9/13/85, Mr. M. Harrison of NSRS contacted ERT and assigned WI-85-055-001 to ERT for investigation. While performing the investigation, a related concern, WI-85-056-001, was also authorized by NSRS for immediate investigation. Both concerns were investigated as one issue.

The purpose of the investigation was to determine if the actions taken by TVA in recertifying the welders in response to Stop Work Authority # 25, met FSAR requirements. In reviewing the program established by TVA and the associated documentation concerning the recertification of welders, it has been determined that the ASME and AWS codes were not met, and that the status of welders recertified under this new program is indeterminate.

FINDINGS:

The following documents were reviewed to determine what corrective action measures TVA had committed to and taken in recertifying their welders; Stop Work Authority # 25, Nonconformance Report 6277 and TVA's commitment letter L44-85-0910-804 to USNRC, Region II.

The following are excerpts taken from those documents listed above:

Non-conformance Report 6277, dated 8/26/85, references Stop Work Authority # 25, and in the corrective method block it states that: "A review of the program will be conducted."

Stop Work Authority # 25, issued 8/23/85, (Rev. 1) states: "Adequacy of some aspects of the welder certification program is indeterminate. Welding will be curtailed until adequacy of the program can be re-evaluated."

CONCERN: WI-85-055-001
WI-85-056-001

DETAILS, continued

TVA's commitment letter to NRC, L44-85-0910-804, dated September 11, 1985, states under the corrective action heading that: "All initial welder certifications older than 90 days have been rescinded...(reference Attachment "B" for details)". Attachment B states that: "Stop Work Authority # 25 was issued August 23, 1985, and all welder certifications were revoked effective August 26, 1985, with the exception of 30 welders". Attachment "B" goes on further to state that "a renewal qualification test program was initiated on August 28, 1985, for all welders whose certifications were revoked".

The information provided in TVA's letter to the NRC was a reiteration of the commitments made in TVA memorandum C01-85-0903-004 dated 9/3/85.

(G. Wadewitz to J. W. Coan). This memorandum C01-85-0903-004 states that the welders certifications had been "rescinded" and "revoked".

On 9/13/85, after reviewing TVA's commitment to "rescind/revoke welders certifications", Mr. W. Kemp and Mr. O. Thero met with G. Wadewitz, Project Manager of Watts Bar, to discuss the rescinding/revoking of welders' certifications. It was stated by ERT that revoking welders certification would require requalification of all welders, not renewal of certification. Mr. G. Wadewitz contacted Mr. S. Stagnolia of ENDES who stated that ERT was talking semantics. By the conclusion of the meeting, the subject of rescinding/revoking of welders' certification had not been resolved.

On 9/13/85, a discussion was held with the site ANI to establish code compliance. It was stated by the ANI that there was a problem with those recertifying under the recertification program, who failed the test and were retested with only one test conducted not two as required by ASME IX QW 321. However, the ANI was not aware that TVA had stated that the welder certifications were rescinded/revoked.

On 9/13/85, Mr. M. Harrison of NSRS was contacted by Mr. O. Thero and it was stated that these concerns were substantiated.

From 9/14/85 to 9/16/85 a further review of related recertification documentation identified the following:

1) There was a 15% failure rate for the welders on the first retest and a 4% failure rate for the second retest.

2) Welders' cards stated that their certifications were rescinded on 8/26/85.

3) There were "special" procedures issued for these tests. These tests were not intended for initial certification, but for recertification only.

CONCERN NO: WI-85-055-001
WI-85-056-001

DETAILS, continued

4) The initial issuance of NCR 6277, block 1A stated: "The welder recertification program...". A corrected copy of NCR 6277 (8/26/85) stated: "The welder certification program:...". The copy of NCR 6277 received on 9/20/85 which is attached to Stop Work Authority # 25 again states: "The welder recertification program..."

5) TVA commitments to NRC as documented in L44-85-0910-804 states that the welders certifications were "rescinded/revoked". In reviewing the ASME and AWS code requirements, welders certifications are no longer valid based upon the revcking of their certifications. This according to ASME/AWS would require initial qualification of all welders. Reference ASME Section IX, QW 461.9.

On 9/17/85 Mr. G. Wadewitz stated that the NRC had given verbal approval for the welders to return to work, however, although the craft were being called back, no work would commence until the release was received in writing. It was inquired if TVA's position on rescind/revoke remained the same. It was stated that the verbage would stand as is.

On 9/18/85, Stop Work Authority # 25 was lifted.

On 9/18/85, the authorized inspection agency issued a SIS report raising the same code compliance questions as noted in this investigation.

On 9/19/85, Mr. G. Wadewitz and Mr. S. Stagnolia stated to ERT that the wording of rescinding/revoking would be changed to administrative withdrawal or some wordage as to that effect. At the time of this report the wording has not been changed.

9/20/85 - The offices of AWS, ASME and the National Board were contacted to request their opinions on the rescinding/revoking of welders certifications. It was the combined opinions that if qualifications (certifications) were revoked/rescinded, then the welder(s) must be initially qualified to position(s), material and process, just as if it was an initial qualification.

AWS -	Mr. D. Seal	Florida
NB -	Mr. M. Hoyle	Columbus, Ohio
ASME -	Welding Dept.	NY, NY

CONCLUSION

These concerns are substantiated. Based on the investigation of the concerns it is concluded that the recertification program does not satisfy ASME/AWS code requirements.

CONCERN NO: WI-85-055-002
WI-85-056-001

DETAILS, continued

This conclusion is based on the following:

* The revoking/rescinding of certifications (i.e. initial qualification versus requalification)

* Retesting of failures is not in compliance with code/standards.

Report Reviewed + Accepted.
[signature] 9/30/85
NSRS

PREPARED BY [signature] 9/23/85
Date

REVIEWED BY [signature] 9/23/85
Date

Note: No new recommendations are made. The issues cited in this report will be addressed in a supplemental response to IN-85-113-003, as a result of NSRS evaluation dated 9/20/85, per discussion w/ G. Wadewitz, 1105 9/30/85. [signature] 9/30/85

REQUEST FOR REPORTABILITY EVALUATION

1. Request No. WI-85-055-001 ______________________
(ERT Concern No.) (ID No., if reported)

2. Identification of Item Involved: Recertification of welders
(Nomenclature, system, manuf., SN, Model, etc.)

3. Description of Problem (Attach related documents, photos, sketches, etc.)

The welder recertification program test presently being administrated to welders at Watts Bar in the recertification efforts following a recent stop work order (No. 25) is not in compliance with code (ASME Section 9).

4. Reason for Reportability: (Use supplemental sheets if necessary)

A. This design or construction deficiency, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.

No ____ Yes X If Yes, Explain: Unqualified/uncertified welders conducting welding activities to safety related systems, components, items.

AND

B. This deficiency represents a significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B.

No ____ Yes X If Yes, Explain: ASME III NA 4000, ASME Section IX QW 322, AWS D1.1 Section 5, 10 CFR 50 Appendix B Criteria IX.

OR

C. This deficiency represents a significant deficiency in final design as approved and released for construction such that the design does not conform to the criteria bases stated in the safety analysis report or construction permit.

No X Yes ____ If Yes, Explain: ______________________

OR

D. This deficiency represents a significant deficiency in construction or or significant damage to a structure, system or component which will require extensive evaluation, extensive redesign, or extensive repair to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function.

No ____ Yes X If Yes, Explain: To assure the verification of welders certifications.

OR

E. This deficiency represents a significant deviation from performance specifications which will require extensive evaluation, extensive redesign, or extensive repair to establish the adequacy of the structure, system, or component to perform its intended safety function.

No X Yes ____ If Yes, Explain: ____

IF ITEM 4A, AND 4B OR 4C OR 4D OR 4E ARE MARKED "YES", IMMEDIATELY HAND-CARRY THIS REQUEST AND SUPPORTING DOCUMENTATION TO NSRS.

This Condition was Identified by: [signature] 365-4464
ERT Group Manager — Phone Ext.

[signature] 365-4414
ERT Project Manager — Phone Ext.

Acknowledgment of receipt by NSRS

[signature]
Signature

Date 9/30/85 Time 1108

REQUEST FOR REPORTABILITY EVALUATION

FINAL

1. Request No. WI-85-050-001 (SG 9 ... 2/26/85)
(ERT Concern No.) ______ (ID No., if reported)

2. Identification of Item Involved: Recertification of welders
(Nomenclature, system, manuf., SN, Model, etc.)

3. Description of Problem (Attach related documents, photos, sketches, etc.)

Welders are retesting Sman/TIG on plate for pipe welds.

4. Reason for Reportability: (Use supplemental sheets if necessary)

A. This design or construction deficiency, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.

No ____ Yes X If Yes, Explain: Recertification when revoked/rescinded call for initial qualification & retesting to two test pieces or all positions.

AND

B. This deficiency represents a significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B.

No ____ Yes X If Yes, Explain: Violation of ANS D.1 Section 5 ASME III NA 400 & ASME IX on 322 & 10 CFR 50 Appendix 50, Criteria IX.

OR

C. This deficiency represents a significant deficiency in final design as approved and released for construction such that the design does not conform to the criteria bases stated in the safety analysis report or construction permit.

No X Yes ____ If Yes, Explain:

OR

ERT Form M

D. This deficiency represents a significant deficiency in construction of or significant damage to a structure, system or component which will require extensive evaluation, extensive redesign, or extensive repair to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function.

No X Yes ____ If Yes, Explain: ____________________

OR

E. This deficiency represents a significant deviation from performance specifications which will require extensive evaluation, extensive redesign, or extensive repair to establish the adequacy of the structure, system, or component to perform its intended safety function.

No X Yes ____ If Yes, Explain: ____________________

IF ITEM 4A, AND 4B OR 4C OR 4D OR 4E ARE MARKED "YES", IMMEDIATELY HAND-CARRY THIS REQUEST AND SUPPORTING DOCUMENTATION TO NSRS.

This Condition was Identified by: ____________ 365-4464
ERT Group Manager Phone Ext.

____________ 365-4414
ERT Project Manager Phone Ext.

Acknowledgment of receipt by NSRS

____
Signed

Date 9/30/85 Time 1108

NRC

TVA 64 (OS-9-69)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Craven Crowell, Director of Information, E12A4 C-K

FROM : K. W. Whitt, Director of Nuclear Safety Review Staff, E3A8 C-K

DATE : October 4, 1985

SUBJECT: REPORTS SUBMITTAL FOR "NUCLEAR SAFFTY UPDATE"

Attached is one copy each of the following final reports of investigation or evaluation of employee concerns for your use, summarization, and publication in Nuclear Safety Update. All have been reviewed and accepted by NSRS.

Concern No.	Investigation Performed by	Concern No.	Investigation Performed by
IN-85-411-001	NSRS		
IN-85-465-001	NSRS		
IN-85-554-001	NSRS		
IN-85-021-002	ERT		

M Middleton for
K. W. Whitt

Attachments

Please acknowledge receipt by signing, copying, and returning this transmittal form to J. T. Huffstetler at E3B37 C-K.

Name

Date

Repo4A:B

cc: J. W. Coan, P-104 SB-K
H. N. Culver, W12A19 C-K
E. R. Ennis, WBN
Guenter Wadewitz, OC-WBN
QTC/ERT, CONST-WBN

EMPLOYEE CONCERN DISPOSITION REPORT

CONCERN NO. IN-85-411-001

DATE OF PREPARATION: 9-16-85

CONCERN: Individual had a concern about the safety hazard to the public and equipment at El. 729 Lines T15, T16, F&G Line on a platform of a small tank. A ladder leads to that platform with a swinging gate. When the gate opens, it strikes the valve which makes the valve open. It could be dangerous to the equipment and could damage the valve.

INVESTIGATION PERFORMED BY: TVA NSRS

FINDING(S): The valve was identified as 2-HCV-6-1679A which was an isolation valve for level switch (LS) 6-92B on moisture separator reheater (MSR) C-2 on the unit 2 side of the turbine building. The valve has a normally required open position and was located directly below LS-6-92B acting as the bottom isolation valve for the level column.

The swinging gate was the entrance to the platform below the C-2 MSR belly drain tank and struck the valve directly on the top of the handwheel whenever the swinging gate to the platform was opened fully. This could cause possible damage to the handwheel or valve stem if the force applied for opening was severe enough. Since LS-6-92B was located directly above the valve, the level switch might also be affected causing an inadvertent annunciation in the control room or valve misoperation.

CORRECTIVE ACTION(S) A chain has been installed on the gate to restrict its travel and prevent the gate from striking the valve.

CLOSURE STATEMENT: This concern was substantiated in that the swinging gate struck the valve, however, the impact did not appear to cause the valve to open nor did there appear to be a danger to public safety as stated in the employee concern.

ERT Form Q

PRELIMINARY **FINAL**

REQUEST FOR REPORTABILITY EVALUATION

1. Request No. IN-85-411-001 ______________________
(ERT Concern No.) (ID No., if reported)

2. Identification of Item Involved: Safety Hazard
(Nomenclature, system, manuf.,SN, Model, etc.)

3. Description of Problem (Attach related documents, photos, sketches,etc.)
SAFETY HAZARD TO THE EQUIPMENT AND PUBLIC BECAUSE OF CONSTANTLY STRIKING A SWINGING GATE ON THE VALVE OF A SMALL TANK ON THE PLATFORM.

4. Reason for Reportability: (Use supplemental sheets if necessary)

A. This design or construction deficiency, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant.

No __X__ Yes _____ If Yes, Explain:______________________________

AND

B. This deficiency represents a significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B.

No _X___ Yes _____ If Yes, Explain:______________________________

OR

C. This deficiency represents a significant deficiency in final design as approved and released for construction such that the design does not conform to the criteria bases stated in the safety analysis report or construction permit.

No __X__ Yes _____ If Yes, Explain:______________________________

OR

ERT Form M

PRELIMINARY **FINAL**

REQUEST FOR REPORTABILITY EVALUATION

D. This deficiency represents a significant deficiency in construction of or significant damage to a structure, system or component which will require extensive evaluation, extensive redesign, or extensive repair to meet the criteria and bases stated in the safety analysis report or construction permit or to otherwise establish the adequacy of the structure, system, or component to perform its intended safety function.
No __X__ Yes _____ If Yes, Explain: ____________________

OR

E. This deficiency represents a significant deviation from the performance specifications which will require extensive evaluation, extensive redesign, or extensive repair to establish the adequacy of the structure, system, or component to perform its intended safety function.
No __X__ Yes _____ If Yes, Explain: ____________________

IF ITEM 4A, AND 4B OR 4C OR 4D OR 4E ARE MARKED "YES", IMMEDIATELY HAND-CARRY THIS REQUEST AND SUPPORTING DOCUMENTATION TO NSRS.

This Condition was Identified by: W. F. Aubach [signature] 365 4478
ERT Group Manager Phone Ext.

William S. [signature]
ERT Project Manager Phone Ext.

Acknowledgment of receipt by NSRS

[signature] Date 9/24/85 Time 1424
Signed

ERT Form M