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MO-QAP 3.2, "Review of Division Nonconformance Reports and Audit Deficiencies," revision 2 dated November 8, 1979 and revision 3 dated May 11, 1981

MO-QAP 3.4, "Reviews of Formal Appraisal Findings for Significance," revision 0 dated August 19, 1980

CONST-QAPP15, "Nonconforming Materials, Parts, or Components," revision 1 dated May 11, 1979

CONST-QAP15.1, "Control of Nonconformances," revision 5 dated
September 29, 1980

CONST-QAPP16, "Corrective Action," revision 1 dated May 14, dated May 14, 1979

CONST-QAP16.3, "Formal Responses to NRC," revision 0 dated November 10, 1980

CONST NCM Section 10.2, "Nonconforming Items," revision 15 dated January 15, 1981

CONST-QASP4.7, "Review of Significant NCR's for Action Required to Prevent Recurrence," revision 1 dated March 17, 1981

CONST-QASP5.3, NRC-OIE Replies," revision 0 dated November 1, 1978

CONST-QASP7.2, "Trend Analysis, revision 2 dated September 21, 1979"

CONST-QASP7.3, "Program Information Notices," revision 1 dated September 13, 1979

CONST-QASP7.4, "Stop Work," revision 1 dated March 28, 1981

EN DES-EP 1.26, "Nonconformances - Reporting and Handling by EN DES, revision 3 dated March 13, 1980

EN DES-EP 2.02, "Handling of Conditions Potentially Reportable Under Title 10 of Code of Federal Regulations, Parts 21 and 50.55(e), revision 4 dated July 24, 1980

BNP-QCP10.4, "Nonconforming Condition Reports," revision 8, addendum 2 dated March 9, 1981

BNP-QCP10.26, "Quality Control Investigaltion Reports," revision 3, addendum 1 dated July 24, 1980

BNP-QCP10.28, "Handling Allegations," revision 0, addendum 2 dated December 5, 1980

BNP-QCP10.33, "Stop Work Procedure," revision 3 dated March 9, 1981

BNP-QAU-SOP7.2-1, "Trend Analysis," revision 0 dated September 27, 1979

Memorandum from R. W. Dibeler to R. T. Hathcote dated July 17, 1980, "Program Information Notices," (CQA 800718 001)

QCRU Log of QCIR's - Books 1 through 7

QCRU Log of NCR's - Books 1 and 2

Bellefonte Nuclear Plant Quality Trend Analysis Reports BN-TA-80-01, BN-TA-80-02, BN-TAAI-80-01, BN-TAAI-80-02, BN-TAAI-80-03, BN-TAAI-80-04, BN-TAAI-81-01, BN-TASR-80-01, BN-TASR-80-02, BN-TASR-80-03, and BN-TSAR-80-04

7. <u>Training and Qualification of Personnel</u>

QAPP2, Quality Assurance Progra, revision 2 dated September 24, 1980

QAP2.2, Qualification/Certification of Inspection, Examination, and Testing Personnel, revision 4 dated January 21, 1980

QASP6.1, Qualification and Certification of Inspection, Examination, and Testing Personnel, revision 3 dated July 23, 1980

QASP6.2, Qualification and Certification of Audit Personnel, revision 2

QASP6.4, Qualifications and Duties of the Construction Quality Assurance Examiner, revision 2

BNP-QCP10.29, Quality Assurance Training Program, revision 3 dated March 9, 1981

BNP-QCP10.30, Craft Quality Assurance Training, revision 2 dated April 13, 1981

BN-QAU-SOP6.1-1, Standard Operating Procedure for Procedure Certification Testing, revision 4 dated May 27, 1980

QCP Master Examinations for QCP's 1.2, 1.3, 1.4, 2.1, 2.2, 3.12, 3.12

Memorandum from W. R. Dahnke to All Construction Employees, Bellefonte Nuclear Plant, on "Quality Assurance Program, dated ril 9, 1981

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BLN Standard Operating Procedures as follows:

CEU-SOP100, Unit Training, revision 4
EEU-SOP209, Quality Assurance Training Program, revision 1
IEU-SOP307, Quality Assurance Training Program, revision 2
MEU-SOP608, Unit Training, revision 2
STCU-SOP908, Quality Assurance Training Program, revision 1
WEU-SOP707, Quality Assurance Training PRogram, revision 1
MTU-SOP400 Quality Assurance Training Program, revision 2
QCRU-SOP015, Quality Assurance Training Program, revision 2

Auditor Qualification Records

Personnel Training Records

QA Audit Reports

8. Records and Document Control

ID-QAP 17.1, "Transfer of Quality Assurance Records, revision 0 dated August 8, 1978

ID-QAP 17.2, "Quality Assurance Records for Design and Construction," revision 0 dated December 31, 1980

CONST-QAPP6, "Document Control," revision 1 dated May 11, 1979

CONST-QAPP17, "Quality Assurance Records," revision 0 dated February 9, 1979

CONST-QAP 17.1, "Quality Assurance Records," revision 4 dated January 25, 1980

NCM Section 9.1, "Records," revision 12 dated July 7, 1980

BNP-QCP10.1, "Preparation of Quality Control Procedures," revision 5, addendum 1 dated April 24, 1981

BNP-QCP10.7, "Quality Assurance Records," revision 3, addendum 4 dated December 5, 1980

BNP-QCRU-SOP004, "QA Records Filing Procedures," revision 5 dated May 22, 1980

BNP-QCRU-SOP016, "Review of QA Records," revision 0 dated April 9, 1979

BNP-QCRU-SOP017, "BNP Quality Assurance Records Index/ Checklist," revision 1 dated May 7, 1980

BLN-QCRU-SOP019, "Encoding QA Records," revision 0 dated August 18, 1980

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Design Criteria No. N4-50-D743, "Quality Assurance Records Storage Vault"

BLN QA Unit Audit Report Nos. BN-G-77-05, BN-G-79-13, and BN-G-80-10, all dealing with QA records

Memorandum from R. W. Dibeler to J. T. Barnes dated August 24, 1979, "Temporary Storage of QA Records," (CQA 790824 003)

ANSUL Halon 1301 Fire Suppression System Inspection and Test Report dated August 13, 1980

Public Safety weekly checklist "Bellefonte Nuclear Plant Hazardous Area/Activity Report"

9. Procurement

QCP10.3, "Preparation of Field Procurement Documents," revision 9 dated July 15, 1981

QCP10.31, "Evaluation and Selection of Suppliers," revision 1 dated April 13, 1981

10. Facility and Equipment Controls

OEDC, "Program Requirements Manual," revision 14 dated June 30, 1981

CONST, "Quality Assurance Program Manual," revision 19 dated April 14, 1981

CONST-QAPP7, "Control of Purchased Material, Equipment, and Services," revision 2 dated October 15, 1979

CONST-QAPP8, "Identification and Control of Materials, Parts, and Components," revision 0 dated February 9, 1979

CONST-QAP9.1, "Release for Drilling, Chipping, Cutting of, or Welding to Permanent Structure or Components," revision 1 dated May 14, 1979

CONST-QAPP12, "Control of Measuring and Test Equipment," revision 0 dated February 9, 1979

CPNST-QAPP13, "Handling Storage and Shipping," revision 0 dated February 9, 1979

CONST-QAPP20, "Housekeeping," revision 0 dated February 29, 1980

BLN Construction Quality Assurance Manual, revision 2 dated February 15, 1980

BLN-QCP1.1, "Receiving Inspection," revision 7 dated October 8, 1980

BLN-QCP1.2, "Storage," revision 9 dated January 15, 1981

BLN-QCP1.3, "Maintenance," revision 1 dated February 20, 1981

BLN-QCP1.4, "Handling of Nuclear Components, "revision 1, Addendum No. 1, dated March 16, 1979

BLN-QCP10.6, "Work Release," revision 10 dated February 20, 1981

BLN-QCP10.11, "Calibration of Measuring and Test Equipment," revision 6, Addendum Nos. 1, 2, and 3, dated February 26, 1980

BLN-QCP10.12, Material Issue Control," revision 6, Addendum Nos. 1 and 2, dated April 23, 1980

BLN-QCP10.27, "Housekeeping," revision 3, Addendum Nos. 1 and 2, dated September 8, 1980

EN DES General Construction Specification G-33, "Handling Storage, and Disposal of Askael and othe PCB's," revision 2 dated March 1, 1979

EN DES General Construction Specification G-53, "Certification, Identification, Storage, and Tightening Requirements of Bolting Material for Nuclear Power Plants," revision 1 dated March 13, 1979

EN DES General Construction Specification G-54, "Receipt, Storage, and Maintenance During Storage of Electric Motors Rated 4kV and Above," revision 0 dated December 15, 1979

BLN-FCP1.3.1, Electrical Enclosure Cleanliness," revision 0 dated December 8, 1980

BLN-FCP3.1.1, "Selection of Conduit for Permanent Installations," revision 0 dated February 15, 1980

BLN-FCP3.4.2, "Protection of Electrical/Mechanical Equipment and Cables from Construction Activities," revision 6 dated March 27, 1981

11. Scheduling of Construction Activities

Project Control Engineering Manual

Bellefonte Construction Schedules

Task Diagrams

12. <u>ASME Section III QA Program</u>

Division of Construction, "Quality Assurance Program Manual," revision 21 dated July 29, 1981, all policies and procedures

Division of Construction, Quality Assurance Branch Manual," revision 23 dated August 5, 1981

Division of Construction, "Quality Assurance Training Plan," revision 5 dated February 26, 1981

N4M-870, "Field Fabrication, Installation, Examination, and Tests for Piping," revision 1 dated June 6, 1979

N4G-889, "Identification of Structures and Systems Covered by the Bellefonte Nuclear Quality Assurance Program," revision 0 dated August 23, 1978

BNP-QCP1.1, "Receiving Inspection," revison 7 dated October 8, 1980

BNP-QCP1.2, "Storage," revision 9 dated January 15, 1981

BNP-QCP6.8, "Pipe Bending," revision 2 dated September 14, 1977, including addendums 1 and 2

BNP-QCP8.1, "Weld Filler Material Control," revision 4 dated July 24, 1980

BNP-QCP10.2, "Drawing Control," revision 7 dated March 27, 1981

BNP-QCP10.3, "Preparation and Review of Field Procurement Documents," revision 9 dated February 20, 1981

BNP-QCP10.4, "Nonconforming Condition Reports," revision 8 dated June 5, 1980, including addendums 1 and 2

BNP-QCP10.5, "Field Fabrication Orders," revision 4 dated April 24, 1981

BNP-QCP10.6, "Work Release," revision 11 dated April 24, 1981

BNP-QCP10.7, "Quality Assurance Records," revison 3 dated July 10, 1979, including addendums 1, 2, 3, and 4

BNP-QCP10.8, "Control of Temporary Installations or Omissions, revison 2 dated February 26, 1980

BNP-QCP10.9, "Material Identification and Marking," revision 8 dated April 3, 1980, including addendums 1, 2, and 3

BNP-QCP10.12, 'Material Issue Control," revison 6 dated April 23, 1980, including addendums 1 and 2

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BNP-QCP10.13, "Weld Procedure Assignment," revision 5 dated August 16, 1979, including addendum 1

BNP-QCP10.18, "Weld Repair," revision 6 dated January 9, 1979, including addendum 1

BNP-QCP10.19, "Arc Strike Removal," revision 2 dated January 15, 1980

BNP-QCP10.24, "Welder, Welding Operator and Peening Operator Performance Qualification," revision 5 dated October 22, 1980, including addendum 1

BNP-QCP10.31, "Evaluation and Selectionof Suppliers," revision 1 dated January 15, 1981, including addendums 1 and 2

BNP-QCP10.32, "Construction Engineer's Organization," revision 2 dated April 13, 1981

Specification G-29M, "Fabrication, Welding and Examination Specifications and Procedures, "revision 16 dated November 28, 1980

Memorandum from H. H. Mull to Those Listed on "Hartsville and Subsequent Plants - Project Quality Assurance/Control Program Policy," dated September 15, 1978 (DOC 780919 004)

Memorandum from W. R. Dahnke to R. M. Hodges on "Bellefonte Nuclear Plant - ASME Code of Record for Bellefonte Nuclear Plant Systems," dated October 13, 1978 (BLN 781013 066)

MEU-SOP602, "APC," revision 7 dated February 7, 1981

WEU-SOP706, "Method for Handling Mechanical Pipe Hanger Weld Cards," revision 1 dated November 13, 1979

WEU-SOP707, "Quality Assurance Training Program," revision 1 dated September 22, 1980

WEU-SOP711, "Spot Radiography of ASME III, Class 3 and ANSI B31.1 Pipe Butt Welds," revision 3 dated March 16, 1981

WEU-SOP712, "Handling of Suspected Discrepant Conditions," revision 1 dated March 27, 1981

WEU-SOP713, "Engineering Acceptance of ASME III Weld Records," revision 0 dated August 27, 1980

WEU-SOP715, WEU Welding Inspector Training and Certification Program," revision 1 dated March 16, 1981

WEU-SOP716, "Audit of Visual Weld Inspections," revision 0 dated May 14, 1981

BNP-FCP8.1.1, "Handling of Weld Filler Material," revision 0 dated

BNP-FCP 8.2.1, "Post Weld Heat Treatment of ANSI B31.1 Welds," revision 0 dated September 5, 1980

BNP-FCP 6.8.1, "pipe Bending," revision 1 dated September 15, 1977

BNP-FCP8.1.1, "Handling of Weld Filler Material," revision 0 dated July 18, 1980

BNP-FCP10.6.1, "Work Release," revision 1 dated March 27, 1981

BNP-FCP10.7.1, "Automated Process Control, revision 0 dated June 28, 1976

BNP-FCP10.9.2, "Control and Identification of Fasteners," revision 1 dated

IQT Contracts Reviewed: 79K72-589854, 79KA1-589858, 79KA1-589861, 78KA1-589862, 80K74-607876, 81K74-607881, 81K74-607883, 81K74-607885, 81K74-607886, 81K74-607888

Requests for Deliveries Reviewed: RD-812593 and RD-812607

OEDC Quality Assurance Manual for ASME Section III Nuclear Power Plant Components (NCM), revision 31

ASME Boiler and Pressure Vessel Code, 1974 Edition through Summer 1974 Addenda, inclusive; and 1980 Edition through Summer 1981 Addenda, inclusive

13. Special Process Controls

CONST-QASP3.5, "Quality Assurance Training Program Plan," revision 2 dated May 7, 1981

CONST-QASP6.1, "Qualification and Certification of Inspection, Examination, and Testing Personnel," revision 3 dated July 23, 1980

CONST-QASP6.3, "Qualifications and Duties of NDE Level III Personnel and NDE Instructors, "revision 5 dated August 3, 1981

CONST-QAPP2, "Quality Assurance Program," revision 2 dated September 24, 1980 and Implementing Procedures QAP2.2R4 and QAP2.3R6

CONST-QAPP5, "Instructions, Procedures, and Drawings," revison 2 dated October 7, 1980 and Implementing Procedure QAP5.1R1

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CONST-QAPP9, "Control of Special Processes," revision 0 dated February 9, 1979 and Implementing Procedure QAP9.1R1

Division of Construction, "Quality Assurance Training Plan," revision 5 dated February 26, 1981

OEDC Qualtiy Assurance Program Requirements Manual for Design, Procurement, and Construction (PRM)," revision 14 dated June 30, 1981

BNP-QCP6.8, "Pipe Bending," revision 2 dated September 14, 1977, including addendums 1 and 2

BNP-QCP7.1, "Radiography Examination," revison 2 dated July 25, 1979, including addendum 1

BNP-QCP7.2, "Ultrasonic Examination," revision 1 dated August 4, 1980

BNP-QCP7.3, "Magnetic Particle Examination," revision 4 dated August 8, 1980

BNP-QCP7.4, "Liquid Penetrant Examination," revision 3 dated June 4, 1979

BNP-QCP7.5, "Visual Examination of Weld Joints," revision 5 dated July 21, 1980

BNP-QCP7.6, "Hydrostatic Testing, revision 5 dated September 17, 1979, including addendum 1

BNP-QCP7.8, "Vacuum Box Leak Testing," revision 3 dated July 9, 1979, including addendum 1

BNP-QCP7.9, "Fitup and Cleanliness," revision 8 dated August 4, 1980, including addendums 1 and 2

BNP-QCP7.10, "Thickness Measurement by Ultrasonic Method," revision 1 dated February 15, 1980

BNP-QCP8.1, "Weld Filler Material Control," revision 4 dated July 24, 1980

BNP-QCP8.2, "Postweld Heat Treatment," revision 2 dated July 9, 1979, including addendum 1

BNP-QCP10.8, "Control of Temporary Installations or Omissions," revision 2 dated February 26, 1980

BNP-QCP10.9, "Material Identification and Marking," revision 8 including addendums 1, 2, and 3

BNP-QCP10.12, "Material Issue Control, revision 6 dated April 23, 1980, including addendums 1 and 2

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BNP-QCP10.13, "Weld Procedure Assignment" revision 5 dated August 16, 1979, including addendum 1

BNP-QCP10.18, "Weld Repair," revision 6 dated January 9, 1979, including addendum 1

BNP-QCP10.19, "Arc Strike Removal," revision 2 dated January 15, 1980

BNP-QCP10.24, "Welder, Welding Operator, and Peening Operator Performance Qualification," revision 5 dated October 22, 1980 including addendum 1

MEU-SOP602, "APC," revision 7 dated February 7, 1981

WEU-SOP706, "Method for Handling Mechanical Pipe Hanger Weld Cards," revision 1 dated November 15, 1979

WEU-SOP708, "Ultrasonic Thickness Measurement Utilizing the D-Meter," revision 1 dated December 31, 1980

WEU-SOP711, "Spot Radiography of ASME III, Class 3 and ANSI B31.1 Pipe Butt Welds," revision 3 dated March 16, 1981

BNP-FCP6.81, "Pipe Bending," revision 1 dated September 15, 1981

BNP-FCP8.1.1, "Handling of Weld Filler Material," revision 0 dated July 18, 1980

BNP-FCP10.6.1, "Work Release," revison 1 dated March 27, 1981

BNP-FCP10.7.1, "Automated Process Control," revison 0 dated June 28, 1976

BNP-FCP10.7.2., "Handling of APC (Weld) Cards and Weld Maps," revision 0 dated September 26, 1979

BNP-FCP10.92, "Control and Identification of Fasteners," revision 1 dated

ASME Boiler and Pressure Vessel Code, 1974 Edition through Summer 1974 Addenda, inclusive; 1977 Edition; and 1980 Edition through Summer 1981 Addenda, inclusive

American Welding Society (AWS), "Structural Welding Code," AWS D1.1-72

ANSI B31.1, "Power Piping," 1975 Edition through Summer 1974 Addenda

ANSI B31.7, "Nuclear Power Piping," 1969 Edition and all addenda

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SNT-TC-M, "Personnel Qualification and Certification in Nondestructive Testing," 1975 Edition

Specification G-29M, "Fabrication, Welding and Examination Specifications and Procedures," revision 16 dated November 28, 1980

Quality Control Investigation Report No. 10807 dated June 17, 1981

Memorandum from R. L. Harris to F. Gilbert, on "Main Coolant Pipe Welding Procedure Qualification and Welding Material Certification," dated July 18, 1977 (DOC 770719 003)

Memorandum from W. W. Aydelott to G. Farmer on "Bellefonte Nuclear Plant - Test Request for Main Coolant Pipe Weld Test Procedure," dated October 12, 1977 (BLN 771013 068)

APPENDIX A

Notes:

1. Appendix A summarizes the recommendations of this review. Positive aspects and areas that NSRS considered to be adequate have been omitted.

2. The item number corresponds to the recommendation number in the body of the report (section V) with the exception that the report number R-81-14-OEDC(BLN) has been omitted.

3. The column labeled "Details Section" gives the report section which contains details of the finding and the next column gives the page number where that section begins.

4. The column labeled "R/E" classifies the finding according to its basis. The (R) indicates that NSRS has concluded the recommendation is based on a regulatory requirement or a commitment. The (E) indicates NSRS has determined that the recommendation has no regulatory basis. It is considered an enhancement and based on subjective judgment.

5. The column labeled "PROG/IMP" further classifies the finding as either a problem with the written program (PROG) or a problem with the implementation (IMP) of the program.

6. Recommendations which in the opinion of NSRS should receive initial management attention are marked with an "X".

7. As asterisk, (*), indicates that an item appears more than once in this table. Only one response to this item is required, however.

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	A. OEDC					
	1. Management Controls					
1	Establish a program for attaining specific quality goals and objectives	V.A.1.a	42	E	PROG	X
	2. Quality Assurance					
2	Develop composite list of applicable codes and standards	V.A.2.a.(1)(a)	45	R	PROG	X
3	Issue commitment sheets for regulatory guides/standards	V.A.2.a.(1)(b)	45	R	IMP	
4	Revise MO-QAP's 2.1, 2.2, and 2.3 to require periodic review of the manuals	V.A.2.a.(1)(c)	46	R	PROG	
5	Revise appropriate documents to specify the requirements of ANSI N45.2.9 for temporary storage of QA records	V.A.2.a.(1)(d)	46	R	PROG	
6	Expedite the submittal of a revised Topical Report on QA for BLN to NRC and assure that it accurately describes the current QA program	V.A.2.a.(1)(e)	46	R	PROG	X
7	Increase the manpower resources for the Quality Requirements Section	V.A.2.a.(1)(f)	47	E	IMP	

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
8	Increase the manpower resources for the Compliance Section	V.A.2.a.(2)(a)	49	E	IMP	
9	Provisions should be made for storage of audit support records as required by ANSI N45.2.12 and N45.2.9	V.A.2.a.(2)(b)	49	R	PROG	
10	Comply with the present wording of the NCM OEDC Policy statement and committed code edition requirements or evaluate revising the statement to reflect: •That the NCM is maintained to the latest edition and addenda issued to the ASME Code •That the NCM may be used directly if section details are specific enough to eliminate a duplication effort •The intended degree to which the NCM is to govern when compared with PRM and ID-QAP requirements	V.A.2.b.(2)(a)	53	R	IMP	
11	•The OEDC QA Manager's office needs to reevaluate its ASME QA program responsibilities involving the establishment of minimum training, record retention requirements, and the auditing of EN DES and CONST Code activities	V.A.2.b.(2)(b) V.B.10.b.(1)	53	R	IMP	

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	•NEB needs to reevaluate its NCM maintenance responsibilties to ensure that when documents described in the NCM are changed, corresponding changes are also made in the NCM					
12	Specific program measures used to control the forming, bending, and aligning processes at TVA construction sites holding ASME Certificates of Authorization should be incorporated into the NCM	V.A.2.b.(2)(c)	56	R	PROG	
	3. Interface Control					
13	Upgrade the ID-QAP's to define organizational responsibilities and to provide instructions for management's control over the following interdivisional quality-related activities •design changes •control of vendor manuals •design review from the constructability and operability standpoint •review of plant operating procedures by EN DES	V.A.3.a.(1)	57	R	PROG	X

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
14	Establish EN DES and CONST interface responsibilities on DIS inputs and provide assurance that those responsibilities will be carried out	V.A.3.a.(2)	58	R	IMP	
	4. Training and Qualifications of Personnel					
15	QA should develop an administrative procedure delineating QA training requirements and provisions for management approval of QA auditor training program	V.A.4.b	61	E	PROG	
16	OEDC should effect timely resolution and keep NSRS informed regarding the resolution of deficiency No. 6, Audit Report M78-05	V.A.4.d	63	R	IMP	X
	B. EN DES					
	1. Management Controls					
	See Management Controls for OEDC					
	2. Design Process Controls					
17	Review the EP's to: •correct conflicting statements, inconsistencies, and overlaps •provide further guidance to designers and reviewers	V.B.2.a	65	R	PROG	X

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
18	Establish detailed QA policy statements and place in a central higher tier document similar to CONST QA Program Manual	V.B.2.b	70	R	PROG	
19	Delineate all management responsibilities dealing with the design, review, procurement, and quality assurance functions for nuclear power plants in a single source reference	V.B.2.c	71	R	PROG	
20	Establish a comprehensive, controlled list of all safety-related systems and components covered by the QA program, and establish provisions to ensure that this listing be kept up to date	V.B.2.d	72	R	PROG	X
21	Revise the review program to require NSSS vendor review of design criteria	V.B.2.e	73	R	PROG	X
22	Revise EP 3.01 to require that all affected organizations review and approve DIM's	V.B.2.f	74	R	PROG	
23	Devise and implement a method of documenting the complete and up-to-date design bases for each safety-related system for the life of the nuclear plant	V.B.2.g	75	R	PROG	
24	Institute a program to verify that actual as-constructed structural steel loadings in safety related buildings are within allowable tolerances	V.B.2.h	76	R	PROG	X

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/F	PROG/IMP	More Serious NSRS Concerns
25	Provide QA procedures for the cable routing and termination program including their generation, verification, and use	V.B.2.i	77	R	PROG	
	3. Design Changes					
26	•Expand ID-QAP's to cover FCR responsibilities •Review and revise EP's to ensure that FCR reviews called for can be done in the time provided •Ensure that adequate documentation of FCR reviews will be made •Provide tighter control of the implementation of FCR procedures	V.B.3.a	79	R	PROG	X
27	Review the program for ECN's, then revise procedures and train personnel to ensure that, as a minimum: •responsibilities are defined •reviews are documented •safety consequences of changes are evaluated	V.B.3.b	83	R	PROG	
28	ECN S1 design change method should be discontinued at BLN because of the abuse of the procedure and the stage of construction	V.B.3.c	84	R	IMP	
29	Written procedures should be established to provide guidance for evaluation and processing of CCN's and address provisions to ensure that auditing of associated documentation is possible	V.B.3.d	86	R	PROG	

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
30	Establish a program that provides for management review and approval of situations that result in cases where ECN's cannot be closed within a specified amount of time	V.B.3.e	86	E	PROG	
	4. Configuration Control					
31	Review and revise established written procedures to ensure that all vendor drawings used to perform work or to verify equipment configuration are included in the DIS	V.B.4.a	88	R	PROG	
	5. Quality Assurance					
32	Provisions should be made for storage of audit support records as required by ANSI N45.2.12 and N45.2.9	V.B.5.b.(1)	92	R	PROG	
33	Obtain additional auditors to accomplish the interal audit function	V.B.5.b.(2)	92	E	IMP	
34	Define group and individual responsibilities within QAB in a single document to maximize interfacing within QAB and with other organizations	V.B.5.b.(3)	92	E	PROG	

APPENDIX A
SUMMARY TABLE OF RECOMMENDATIONS
OEDC MANAGEMENT REVIEW, R-81-14-OEDC(BLN)

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	6. Corrective Action					
35	Take measures necessary to assure that requirements of EP 1.26 (generation of NCR's) are understood and followed by all EN DES employees	V.B.6.b.(1)(a)	96	E	IMP	
36	•EP 1.26 should be revised to require consideration of whether document changes are required as a result of NCR's •Statement should be required, pro or con, in dispositioning correspondence as to whether document changes are needed, and if so, who is responsible	V.B.6.b.(1)(b)	96	E	PROG	
37	EP 1 should be revised to incorporat e definition of significance contained in the latest revision of QAI-4	V.B.6.b.(1)(d)	97	E	PROG	
38	Results of the investigation of the performance record in meeting previous commitments to the NRC should be evaluated very closely by management	V.B.6.b.(3)	98	E	IMP	

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
39	°Expedite implementation of the trend analysis program to fulfill commitments and provide a useful management tool °BLN FSAR description of who performs this trending function should be updated	V.B.6.b.(4)(a)	100	R	PROG	
	7. Training and Qualifications of Personnel					
40	Evaluate technical training being conducted in the branches and establish requirements at the division level to ensure technical training will be conducted in the EN DES branches	V.B.7.b	101	E	PROG	
41	QAB should develop an administrative procedure delineating QA training requirements and provisions for management approval of QA auditor training programs	V.B.7.c	102	E	PROG	
	8. Records and Document Control					
42	Provide protective or alternate storage location measures as required by ANSI N45.2.9 for the Codes and Standards stored in open shelves in the TIC and "backfile" documents being similarly stored in MEDS	V.B.8.a V.B.8.b	103	R	IMP	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
43	Document the review and approval cycle for the MEDS Handbook, preferably in or on the document itself	V.B.8.c	103	R	IMP	
44	Revise G-53 as required and assure that other specification revision notices have been incorporated into the appropriate General Specification within the timeframe of EP 3.04	V.B.8.d	104	R	IMP	
	9. Procurement					
45	Set up a program to record and retrieve information on past contracts, both QA and non-QA for the benefit of bid evaluators	V.B.9.a	105	R	PROG	
46	Specific and detailed instructions should be provided and the conditions under which optional methods of bid evaluation should be used	V.B.9.b	106	R	IMP	
47	Revise EP 5.01 to require review of requisitions to ensure the functions of interface review and special expertise review are accomplished	V.B.9.c	107	R	PROG	
48	OEDC should develop a procedure whereby actual contracts, not just requisitions, are reviewed before work is done using them	V.B.9.d	108	E	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	10. ASME Section III QA Program					
49	OEDC QA, EN DES QAB, and CONST QAB audit report distribution lists should be revised to reflect report transmittals to CSM are required when findings related to Code activities are contained therein. A similar revision should also be considered to NCR forms	V.B.10.b.(2)	111	E	PROG	
50	•Construction Specification N4M-870 should be revised to: (a) Indicate the proper revision level of Design Specifications BNP-DS-1935-2856, -2857, and -2858 (b) Delete its Section IX commitment to the 1974 edition of ASME Code •Review and revise all other documents which may contain the stated conflicts	V.B.10.b.(3)	112	R	PROG	
51	Apply the coorective action taken in response to EN DES internal audit 80-04, finding No. 2, to NCM section 3.5	V.B.10.b.(4)	112	E	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	11. Special Process Controls					
52	A general review and revision of G-29 is considered necessary to: •Pare down the size of G-29M by incorporating numerous addenda throughut •Categorize general welding requirements in G-29C for structures, systems, or components for easy access by CONST •Eliminate conflicts found between the -29M general index to corresponding welding procedures and process specification to code and standard requirements •Revise and compartmentalize or index individual G-29M processes to show plicable requirements for vintage or class of plant	V.B.11.a	115	E	PROG	
53	OEDC should evaluate which QA organization it considers best suited to concur in either the entire G-29M manual or each individual process specification prior to use and they should then do those reviews	V.B.11.b	118	R	PROG	
	C. CONST					
	1. Management Controls					
	See Management Controls for OEDC					

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	2. Construction Processes					
54	BLN CONST should establish an administrative control program to ensure that construction activities are adequately planned and controlled similar to CONST CEP 5.04, "Work Packages," in use at other sites	V.C.2.a	120	R	PROG	X
55	BLN CONST should review and revise the QCP's and indoctrinate personnel to ensure that, at a minimum, the following inadequacies are corrected: *•minor changes were not required to be incorporated into QCP's within a specific time period *•a matrix of documents referenced in QCP's was not provided *•uncontrolled copies of QCP's were in use •there was a problem with the tracking of FCR's •inspection checklists were not required •some safety-related activities were not covered in a QCP	V.C.2.b	121	R	PROG	
56	BLN CONST should review the current practice of grinding welds smooth and should establish inspection requirements to ensure that welds are inspected in the most informative condition	V.C.2.c	126	E	PROG	X

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
57	CONST should take the following steps to correct inadequate interfacing between BLN organizational units: •An engineering unit should be given overall responsibility to assure that all aspects of a job are correctly completed •A more complete checklist for concrete pours should be developed and implemented before pours begin •The use of area planners should be considered to reduce hanger interferences with field routed components •The construction superintendent's organization should be given the opportunity to supply input to the QCP's, FCP's, and SOP's	V.C.2.d	127	E	PROG	
58	Additional review will be required by NSRS to determine whether as-constructed seismic analysis is being accomplished on systems such as cable trays and electrical conduits	V.C.2.e	130			

Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	3. Design Changes					
59	CONST should incorporate the following information into the work plant forms: •Explanation of any remaining work required to complete an item •Specific list of the FCC drawings revised •Provisions for the generation of an FCR, if required •Generation of a "fire hazard evaluation" for the modification work	V.C.3.a	133	E	PROG	
60	CONST should establish and implement a mechanism for identifying safety-related/seismically qualified equipment to be excluded from the CCN change process	V.C.3.b	133	R	PROG	
	4. Configuration Control					
61	Implement the requirements of site procedures to ensure that the DIS provides the plant as-constructed configuration	V.C.4.a	136	R	IMP	
62	CONST should amend the procedures for cable installation slips to ensure that they receive configuration control as required by ID-QAP 6.1 similar to the drawings they supplement and amplify	V.C.4.b	136	R	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	5. Quality Assurance					
63	Provisions should be made to store audit support records in fire-rated cabinets as required by ANSI N45.2.12 and N45.2.9	V.C.5.a.(1)	138	R	PROG	
64	Assure that documentation of corrective actions taken in response to audit findings will always be accomplished	V.C.5.a.(2)	139	R	PROG	
65	Reemphasize the necessity for assuring that adequate corrective action has been completed prior to closure of audit deficiencies	V.C.5.a.(3)	139	R	IMP	
	6. Corrective Action					
66	CONST-QAP 15.1 and BNP-QCP's 10.4 and 10.26 should be revised to utilize the definition of significance, including examples, provided in the latest revision of QAI-4	V.C.6.b.(1)(a)	144	E	PROG	
67	Revise procedures to provide for an independent review of QCIR's as required by QAI-4	V.C.6.b.(1)(b)	144	R	IMP	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
68	The usefulness of the PIN system should be increased by using it to disseminate potentially generic information from other sources, such as OEDC QA findings	V.C.6.b.(2)	145	E	PROG	
69	The BNP-QCP's governing generation of QCIR's and NCR's should be revised to emphasize recording information as to the cause of problems. At present, this information is required only for significant NCR's	V.C.6.b.(4)(a)	147	E	PROG	
70	Trend analysis results should be factored into the site training program, thereby providing a more useful management tool	V.C.6.b.(4)(b)	147	E	PROG	
71	Management should evaluate the corrective action system to assure it is capable of identifying and correcting continual problems of a similar nature when those problems were individually categorized as nonsignificant	V.C.6.b.(4)(c)	148	E	PROG	X
	7. Training and Qualifications of Personnel					
72	CONST should establish requirements as opposed to recommended actions for the evaluation of the effectivness of craft supervision QA training	V.C.7.a	149	E	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
73	CONST should develop a written program which assigns responsibilities and specific training and retraining requirements for journeyman craftsman QA training	V.C.7.b	150	R	PROG	X
74	CONST should establish administrative procedures delineating apprentice training requirements, specific responsibilities, and how the program is to function	V.C.7.c	150	E	PROG	
75	CONST should develop a program to assure management that journeyman craftsmen who perform quality-related activities obtained from the union halls do possess the required skills	V.C.7.d	151	E	PROG	
	8. Records and Document Control					
76	The records vault Halon system weekly checklist should be revised to provide for determination of cylinder pressure and its acceptability. Also, adequacy of the cylinder pressure guage should be evaluated	V.C.8.a	153	R	PROG	
77	A time limit should be established within which changes per errata and addenda are physically incorporated into the QCP's and SOP's by revision of the base document	V.C.8.b	154	E	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
78	A matrix listing each QCP and the references contained in it should be developed such that changes in one document can be readily assessed in terms of impact on other documents	V.C.8.c	154	E	PROG	
79	Controlled copies of QCP's should be made available for the crafts and their use of them assured	V.C.8.d	155	E	PROG	
80	Management should evaluate the use of SOP's and FCR's. If they prescribe activities affecting quality and such instruction are not contained in QCP's or higher tier documents, formal control systems for their development, use, and maintenance should be instituted	V.C.8.e	155	R	PROG	
	9. Procurement					
	No recommendations in this area					
	10. Equipment and Facilities Control					
81	Devise and implement inspection requirements to ensure adequate protection of installed equipment form adjacent construction activities	V.C.10.a	158	R	PROG	
	11. Scheduling of Construction Activities					
	No recommendation in this area					

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Item No	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	12. ASME Section III QA Program					
82	Instructions should be provided in the NCM and BLN QCP's to control and identify nonconforming materials to be maintained within rod issue control centers to prevent their inadvertent use	V.C.12.b.(2)	164	R	PROG	
83	Identify on APC cards the applicable addenda of the BLN QCP used and the particular G-29 process specification, including revisions and addenda	V.C.12.b.(3)	164	R	PROG	
84	Since BLN interprets "monthly" contrary to standard technical specification guidance, definition of the terms "weekly," "monthly," etc., should be provided	V.C.12.b.(4)	165	E	PROG	
85	BLN should evalute conversion from their 31-category system to the 4-category system discussed in NCM section 9.1. Should BLN desire exemption from this requirement, it should be documented in the NCM section	V.C.12.b.(5)	166	R	PROG	

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Item No.	Recommendation	Details Section	Page No.	R/E	PROG/IMP	More Serious NSRS Concerns
	13. Special Process Controls					
86	P.S. 1.M.2.2.(b) should be revised to require more positive controls in ensuring welders have received additional training or practice prior to subsequent retest attempts following the failure of a welder to meet the requirements of a performance qualification test	V.C.13.d.(1)	168	E	PROG	
87	A positive means of verifying welder usage of a process, such as observation and documentation by welding inspection, should be instituted at BLN	V.C.13.d.(2)	170	E	PROG	
88	An ongoing training session/discussion should be instituted to continuously emphasize to the craft their role in meeting QA/QC requirements and that the inspector is only assuring that these requirements are being met	V.C.13.d.(3)	170	E	PROG	
89	BLN should evaluate measures to ensure qualification records for its NDE examiners are maintained complete and current	V.C.13.e.(1)	171	R	PROG	

APPENDIX B

MANAGEMENT EVALUATION TREE

One of the primary goals of this management review was to assess the management controls system within OEDC. To aid in the accomplishment of this objective, a fault tree was developed which NSRS believed would assist the reviewers in a systematic and uniform evaluation of the management system in each functional area. The fault tree which is attached to this appendix is entitled the "Management Evaluation Tree" and is commonly referred to as the MET chart.

The MET chart provided the reviewers with a structured approach to the assessment of the management systems that had been established for each functional area. By addressing each of the key elements of the MET chart, the reviewer should have been able to gain a good understanding of how business was being conducted in the area being reviewed. This management evaluation approach should have assured the following basic determinations.

1. If documented policy had been established to provide guidance in the management of the subject areas.

2. If a program had been developed and documented to successfully carry out the established policy in compliance with regulatory requirements, commitments, latest standards, and additional evaluation criteria.

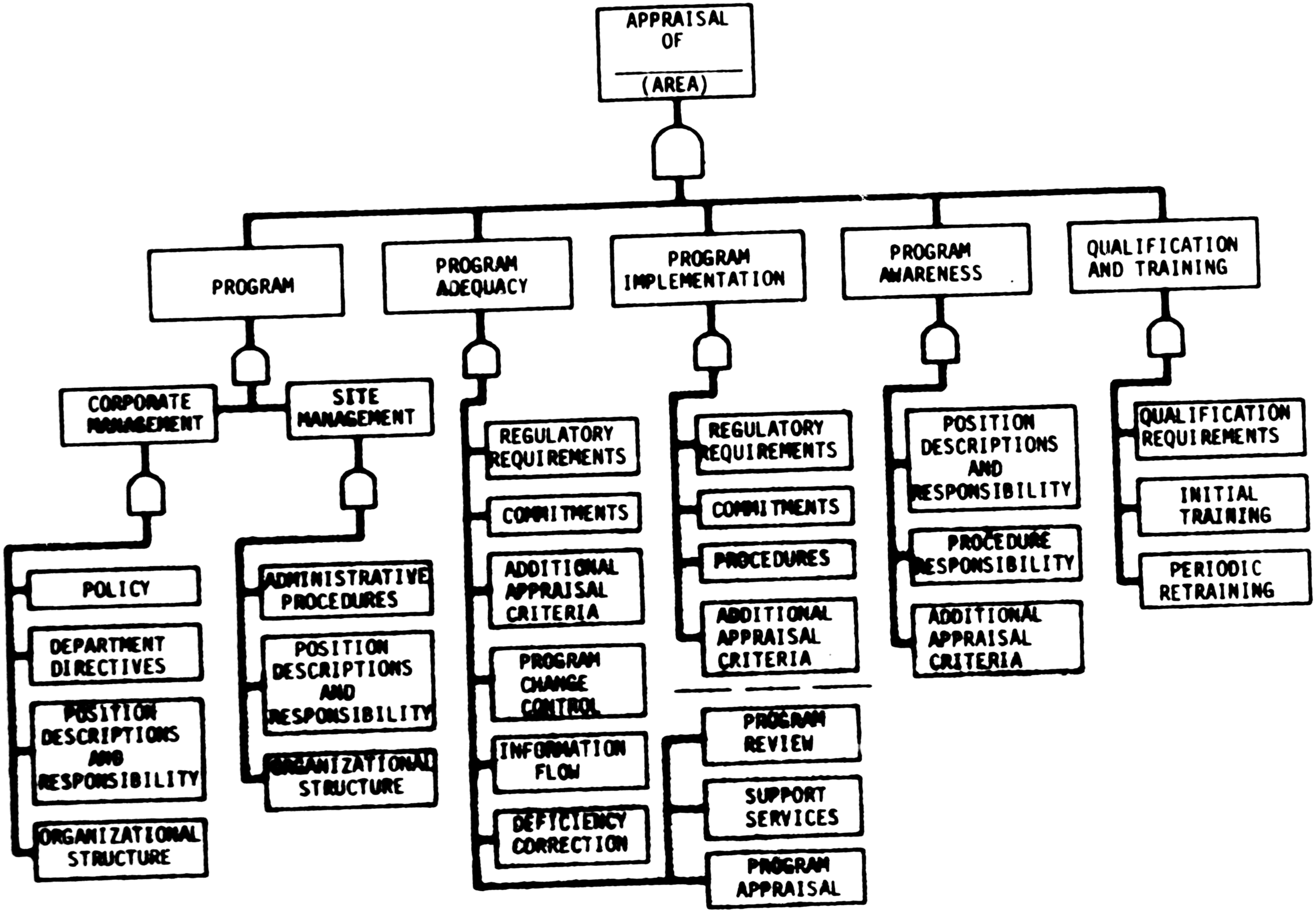
3. If the program was being implemented and implementing activities were being appropriately documented.

4. If responsible personnel were being adequately trained and qualified.

5. If those individuals having gained responsibilities in the area being reviewed understood their roles in the accomplishment of activities within the area.

The various elements indicated by the MET chart were considered in some depth for each area reviewed. Additional detailed checklists appropriate for each specific area were also developed for use during the review.

MANAGEMENT EVALUATION TREE



APPENDIX C

ABBREVIATIONS AND ACRONYMS

ACE	Assitant Construction Engineer - Bellefonte
ADB	Architectural Design Branch - EN DES
AI	Administrative Instruction - EN DES
AISC	American Institute of Steel Construction
ANI	Authorized Nuclear Inspector
ANSI	American National Standard Institute
APC	Automated Process Controls
ASME	American Society of Mechanical Engineers
AWS	American Welding Society
B&W	Babcock and Wilcox Company
BFN	Browns Ferry Nuclear Plant
BLN	Bellefonte Nuclear Plant
BLP	Bellefonte Design Project - EN DES
Board	TVA Board of Directors
BPA	Blanket Purchase Order
CAQ	Condition Adverse to Quality
CCN	Construction Change Notice
CDB	Civil Engineering and Design Branch - EN DES
CEB	Civil Engineering Branch - EN DES
CEO	Construction Engineering Organization - Bellefonte
CEU	Office, Civil, and Material Engineering Unit - Bellefonte
Code	ASME Section III
CONST	Division of Construction
CONST QA	Quality Assurance Branch - CONST
CSM	Codes, Standards, and Materials Section - NEB
CSO	Construction Superintendent Organization - Bellefonte
DCC	Drawing Control Center

DCR	Design Change Request
DIM	Design Input Memorandum
DIR	Design Information Request
DIS	Drawing Information System
DWP	Detailed Weld Procedure
ECM&D	Engineering Construction Monitoring and Documentation
ECN	Engineering Change Notice
EDB	Electrical Engineering and Design Branch - EN DES
EDP	Environmental Design Project - EN DES
EEB	Electrical Engineering Branch - EN DES
EEU	Electrical Engineering Unit - Bellefonte
EN DES	Division of Engineering Design
EP	EN DES Engineering Procedure
ESS	Engineering Support Services - EN DES
FCLD	Functional Control Logic Diagram
FCP	Field Construction Procedure
FCR	Field Change Request
FSAR	Final Safety Analysis Report
G MGR	Office of the General Manager
HEU	Hanger Engineering Unit - Bellefonte
HPP	Hartsville & Phipps Bend Design Project - EN DES
H&S	Office of Health and Safety
HTA	Hartsville Plant "A"
HTB	Hartsville Plant "B"
ID-QAP	TVA Interdivisional Quality Assurance Procedure
IEB	NRC Office of Inspection and Enforcement Bulletin
IEU	Instrument Engineering Unit - Bellefonte
IPM	Interdivisional Quality Assurance Procedures Manual

I&T	Indoctrination and Training
IQT	Indefinite Quantity Term Contract
MOM	OEDC QA Manager's Office QA Manual
MO-QAP	OEDC QA Manager's Office QA Procedure
MDB	Mechanical Engineering and Design Branch - EN DES
MEB	Mechanical Engineering Branch
MEDS	Management and Engineering Data System
MEU	Mechanical Engineering Unit - Bellefonte
NCM	OEDC Quality Assurance Manual for ASME Section III Nuclear Power Plant Components
NCR	Nonconformance Report
NEB	Nuclear Engineering Branch - EN DES
NDE	Nondestructive Examination
NLS	Nuclear Licensing Section - NEB
No.	Number
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUC PR	Division of Nuclear Power
OEDC	Office of Engineering Design and Construction
OEDC QA	Office of Engineering Design and Construction Quality Assurance Staff
OGC	Office of the General Counsel
OSSD or D	Overage, Shortage, Substitution, Defect, or Damage
OWIL	Outstanding Work Items List
PBN	Phipps Bend Nuclear Plant
PERS	Division of Personnel
PIN	Program Information Notice
POWER	Office of Power
PQR	Procedure Qualification Record

PR	Procurement Request
PRM	OEDC QA Program Requirements Manual
P.S.	G-29 Process Specification
PSAR	Preliminary Safety Analysis Report
psig	Pounds Per Square Inch-Gauge
PURCH	Division of Purchasing
PWHT	Post Weld Heat Treatment
QA	Quality Assurance
QAB	Quality Assurance Branch - EN DES
QAP	Quality Assurance Procedure - CONST
QAPP	Quality Assurance Program Policy - CONST
QASP	Quality Assurance Staff Procedure - CONST
QAU	CONST Quality Assurance Unit - Bellefonte
QC	Quality Control
QCP	Quality Control Procedure - Bellefonte
QCRU	Quality Control and Records Unit - Bellefonte
QEB	Quality Engineering Branch - EN DES
QPM	OEDC Quality Policy Memorandum
RG	Regulatory Guide
SCCDL	System Configuration Control Drawing List
SOP	Standard Operating Procedure - Bellefonte
SPED	Architectural, Hydro, and Special Projects Engineering and Design Branches - EN DES
SQN	Sequoyah Nuclear Plant
S-R	Safety Related
STCU	Storage, Test, and Coordination Unit - Bellefonte
SWP	Sequoyah and Watts Bar Design Projects - EN DES
R	Revision level

TAS	Technical and Administrative Services Staff - EN DES
TB&A	Theodore Barry and Associates
TDP	Thermal Plants Design Project - EN DES
TIC	Technical Information Center
T E	Thermal Power Engineering Branches - EN DES
TPED	Thermal Power Engineering Design Projects - EN DES
WBN	Watts Bar Nuclear Plant
WEU	Welding Engineering Unit - Bellefonte
WPS	Welding Procedure Specification
WR	Work Release
YCN	Yellow Creek Nuclear Plant
YCP	Yellow Creek Design Project - EN DES
10CFR21	Title 10, Code of Federal Regulations, Part 21
10CFR50	Title 10, Code of Federal Regulations, Part 50

APPENDIX D

COMPARISON OF LISTS OF SAFETY RELATED
STRUCTURES, SYSTEMS, AND COMPONENTS FOR BLN

	Design Criteria	Lstng of SR Items (FSAR Tbl 17.1A-3)	Sfty Classification of Compnts in SR Fluid Sys or portion of SR Fluid System (FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-D754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM C9
Category I (Seismic structures)							
Reactor Building		X	NR[2]	NR	NR	X	
Shield building incl main steam and feed-water compartment	N4-3R-0701 N4-R3-D701 N4-R4-D701 N4-R5-D701 N4-R7-D701	X	NR	NR		X	
Containment vessel	N4-R6-D701 N4-9R-D701	X	NR	NR		X	
Auxiliary Building	N4-A3-D701		NR	NR	NR	X	X
Control Building	N4-C3-D701		NR	NR	NR	X	X
Diesel Generator Bldg	N4-D2-D701		NR	NR	NR	X	X
Intake Pumping Station	N4-K2-D701		NR	NR	NR	X	X
Primary & Secondary Containment Structures	N4-R5-D701 N4-R7-D701		NR	NR NR	NR NR	X X	X X
Reactor and Reactor Coolant System		X	X	X	X	X	X
Reactor pressure vessel	N4-50-D714 N4-NC-D740	X	X	X	X	X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Compots in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Reactor vessel internals	N4-NC-D740	X					)
Control rod drives and CR assemblies	N4-NR-D740	X	X	X	NR	X	X
Reactor coolant pumps	N4-NC-D740	X	X	X	X	X	X
Reactor coolant piping	N4-NC-D740	X	X	X	X	X	X
Steam generators	N4-NC-D740	X	X	X	X	X	X
Pressurizer	N4-NC-D740	X	X	X	X	X	X
Nuclear Fuel Assembly System	NR			NR	NR	NR	
Nuclear Startup Source System	NR			NR	NR	NR	
Relief valves, safety valves	N4-NC-D740	X	X	X	X	X	
<u>Systems involved in emergency core and reactor bldg cooling</u>							)
Core flood system	N4-NL-D740	X	X	X	X	X	X
Piping		X	X	X	X		
Decay heat removal (DHR) system (low-pressure injection system)	N4-ND-D740	X	X	X	X	X	X
Piping	N4-ND-D740	X	X	X	X	X	

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Makeup (high-pressure injection system)	N4-NV-D740	X	X	X	X	X	X
Piping	N4-NV-D740	X	X	X	X	X	
Reactor building spray system	N4-NS-D740	X	X	X	X	X	X
Piping	N4-NS-D740	X	X	X	X	X	
Reactor bldg cool-ing (RBC) system	N4-VH-D740	X	X	X		X	
Piping		X					
Post accident hydrogen removal system	N4-NO-D740	X	X	X		X	X
Reactor building Purge System	N4-VV-D740	X	X	X		X	
Secondary plant ANS Safety classed portion							
Main steam from steam generator through isolation valve	N4-SM-D740	X	X	X	X	X	X
Feedwater from steam generator through second isolation valve	N4-CF-D740	X	X Note restrc	X Note restrc	X	X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Auxiliary and Emergency systems							
Chemical addition and boron recovery system (Seismic Category I parts except piping)	N4-NB-D740	X	X	X	X	X	X
Piping		X	X				
Component Cooling Water System	N4-KC-D740	X	X	X	X	X	X
Piping		X	X				
Essential raw cooling water system	N4-KE-D740	X	X	X	X	X	X
Control rod drive Cooling water system	N4-KD-D740		X	X	X	X	X
Fire Protection systems (Seismic Category I Parts)	N4-GC-D740 N4-RY-D740	X	X	X	X	X	X
Auxiliary feedwater system	N4-CA-D740	X	X Incld prt of cndst storage	X	X	X	X
Piping		X	X	X	X	X	
Spent fuel cooling system	N4-NM-D740	X	X	X	X	X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Piping		X	X	X	X	X	
Control building air condition system	N4-VK-D740 N4-VL-D740 N4-VC-D740 Others	X		X		X	
Auxiliary building ventilation system	N4-VW-D740 N4-VD-D740 N4-VE-D740	X		X		X	X
Waste disposal		X	X	X	X	X	X
Radioactive waste systems (Seismic Class I parts except piping	N4-WE-D740 N4-WG-D740 N4-WL-D740 N4-NS-D740	X	X	X	X	X	X
Piping		X					
Radiation Monitoring system	N4-IR-D740	X		X	X	X	
6900v AC Power System	N4-RP-D775A			NR	NR	X	X
480v AC Power System	N4-RP-D775A			NR	NR	X	X
120v AC Power System	N4-EJ-D775 N4-ER-D775			NR	NR	X	X
125v Emergency DC Lighting	N4-50-D789			NR	NR	X	X
Auxiliary Control Room Panels	N4-50-D793			NR	NR	X	X

	Design Criteria	Latng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3 2.2 4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Reactor Building Instrument Room Panels	N4-50-D793			NR	NR	X	
Electrical and control equipment		X	NR	NR	NR	X	
Emergency power systems	N4-RPD775A						
Diesel generator system		X	X			X	X
DC power supply system	N4-EU-D775	X		NR	NR	X	
Power distribution cables and busses		X		NR	NR	X	
Transformers		X		NR	NR	X	
Instrumentation and controls	N4-50-D793	X		NR	NR	X	
Reactor protection and control systems	N4-IL-D775 N4-11-D775 N4-50-D791	X					
Shutdown boards and switchgear		X	NR	NR	NR	X	
Vital ac instrumentation and control supply system		X	NR	NR	NR	X	X
Essential Air	N4-RJ-D740		X	X	X	X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Compnts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Emergency Diesel Fuel Oil System	N4-FD-D740 N4-FF-D740		X	X	X	X	X
Miscellaneous Systems							
Environmental Control Systems	Several(1)		X	X	X	X	X
Hydrogen System	N4-GS-D740		X	X	X	X	X
Nitrogen System	N4-GT-D740		X	X	X	X	X
Startup & Recirculation	N4-CR-D740			X	X	X	X
Gasoline Storage & Transfer	N4-FG-D740			X	X	X	X
Fuel Handling System	N4-MF-D740			X		X	X
Condensate Storage	N4-CS-D740		X (Partly in Aux Feed)		X		X
Heat Rejection System	N4-KH-D740			X	X	X	X
Containment Isolation & Leak Test System	N4-NI-D740		N4-50-D747			X	X
Traveling Water Screens for Intake Pumping Station	N4-KED-740A			X		X	X
Revolving Platform			NR	X		X	
Laboratory Compressed Gas	N4-ML-D74C			X	X	X	X
Reactor Bldg - D wall 10 ton crane	N4-MMD740D		NR	X		X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Reactor Bldg Polar Crane	N4-MMD740A		NR	X		X	X
Reactor Coolant Drains & Vents	N4-NK-D740			X			
Main Control Room Ceiling (Physcially Removed)				X			
Demineralized Water Storage & Transfer System				X	X	X	X
Service Air	N4-RH-D740			X		X	X
Control Air System	N4-RH-D470			X	X	X	X
Compressed Air	N4-KK-D740			X	X	X	
Roof Drains	N4-RR-D740			X	X	X	X
Auxiliary Bldg ESF Zone Environmental Control System	N4-VW-D740			X			X
Raw Cooling Water System	N4-KW-D740				X		X
Borated Water Storage & Transfer System	N4-ND-D740	X	X	X	X	X	
Raw Service Water System	N4-RS-D740			X	X	X	X
Potable Water Distribution System				X	X	X	X
Makeup Demineralizer System				X	X	X	X

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-839)	N-OQAM CSSC
Sampling & Water Quality System	N3-YQ-D740			X	X	X	X
Auxiliary Steam	N4-SA-D740			X	X	X	X
Equipment & Floor Drainage System	N4-WE-D740			X	X	X	X
Sodium Hypochlorite System	N4-YC-D740						X (Nonsafety related)
Diesel Generator Lubricating Oil System						X	X
Pipe & Cable Tunnel Manways	N4-Y7-D701		NR	NR	NR	X	X
Secondary Containment Isolation System	N4-NJ-D740						X
Screen Wash System	N4-KE-D740A			X		X	X
Conduit & Grounding System			NR	NR	NR		X
Cable Tray System	N4-2R-D70 N4-50-D728		NR	NR	NR		X
Plant Lighting System	N4-50-D789		NR	NR	NR	X	X
Secondary Containment Air Cleanup System	N4-VX-D740			X		X	X
Reactor Building Vacuum Relief System	N4-ZR-D740			X	X	X	

	Design Criteria	Lstng of SR Items (FSAR Tbl 171A-3)	Sfty Classification of Componts in SR Fluid Sys or portion of SR Fluid System FSAR Tbl 3.2.2-4)	Ident of Mech SR Sys and Compnts (Design Criteria N4-50-D744)	Classification of Piping, Pumps Valves & Vessels (Design Criteria N4-50-754)	Ident of Strs & Sys Covered By the BLN QA Prog (CONST Spec N4G-889)	N-OQAM CSSC
Leakage Detection	N4-50-D747						
Post Accident Sampling	N4-50-D764(TBI)[3]						
Cable Tray Supports	N4-50-D728		NR	NR			
RCS Supports	N4-4R-D701		NR	NR			
Pipe Supports	N4-50-D717		NR	NR			
Lighting Supports	N4-50-D719		NR	NR			

(1)There are several heating, ventilating, and environmental control systems. All are not listed separately on this list.

(2)NR indicates that the list is not required to include that item.

(3)TBI indictes that criteria is to be issued.

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '81 1105 051

TO : G. H. Kimmons, Manager of Engineering Design and Construction, W12A9 C-K

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : November 4, 1981

SUBJECT: BELLEFONTE NUCLEAR PLANT - REVIEW OF HIGH ENERGY PIPE BREAKS OUTSIDE CONTAINMENT - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-81-22-BLN

Attached for your information and action is the final report on the subject review. The report concludes that, in general, the pipe break evaluation meets or exceeds regulatory requirements. The report contains three recommendations which have been denoted with an (R) or an (E). Items denoted by an (R) are based on regulatory requirements or commitments while those denoted by an (E) are not based on requirements but are considered enhancements.

You are requested to provide us with your resolutions or your plan for resolving all recommendations within 30 days of the date of this memorandum. If you have any questions regarding this report, please contact B. F. Siefken at extension 6860 in Knoxville.

M V Siefken for

H. N. Culver

BFS:LML
Attachment
cc (Attachment):
MEDS, 100 UB-K
M. N. Sprouse, W11A9 C-K

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-81-22-BLN

SUBJECT: BELLEFONTE NUCLEAR PLANT
ROUTINE REVIEW OF HIGH-ENERGY PIPE RUPTURES
OUTSIDE CONTAINMENT

DATE
OF REVIEW: October 1981

REVIEWER: Henry L. Jones 11/4/81
DATE

REVIEWER: Bruce G. Siefken 11-4-81
DATE

APPROVED BY: Marvin V. Sinkule 11/4/81
DATE

TABLE OF CONTENTS

I. SCOPE

This routine review covered the analysis of postulated high energy pipe ruptures outside containment at Bellefonte Nuclear Plant (BLN). The review was undertaken to determine whether this analysis met NRC requirements, TVA commitments, and industry standards. Specific areas investigated included the classification of piping as high or moderate energy, the analysis performed to determine the acceptability of a postulated break, and the protective measures taken to mitigate these events.

II. BACKGROUND

The NRC requirements for the design of nuclear plants to withstand postulated pipe ruptures evolved during the design effort on BLN. Criterion 4 of Appendix A to 10CFR50 requires that systems, structures, and components important to safety be protected against the effects of pipe ruptures. The AEC clarified this requirement in letters in December 1972 and July 1973. In 1975 the NRC issued Branch Technical Positions (BTP) MEB 3-1 and APCSB3-1 which gave more detailed guidance for implementing the requirements of Criterion 4. One of the important clarifications that was made during this period was in the definition of a high energy pipe. In the earliest NRC correspondence, the maximum operating pressure and temperature were used as the criteria for determining the energy classification of a pipe. In later correspondence (BTPs) an additional criteria was added. This new criteria allowed a pipe to be classified as moderate energy if the time period that it operated at temperatures or pressure in the high energy classification did not exceed 2 percent of the time the system operated within the moderate energy criteria (2 percent rule).

ANSI standard N176 was developed in the early 70s and a draft version of this standard was released in May 1975. This draft contained a similar criteria allowing classification of a pipe as moderate energy. The basis for the classification was also operating time, but the criteria was 1 percent of the plant operating life span rather than 2 percent of the time that the system operates at moderate energy conditions. This is often called the 1 percent rule. This was considerably more liberal than the NRC criteria and allowed several systems at BLN to be classified as moderate energy (e.g., auxiliary feedwater, and startup and recirculation). This draft standard also allowed the option of using the 2 percent rule. In an October 1976 draft, this standard was modified to allow only the use of the 2 percent rule.

TVA requirements in the area of pipe breaks have also evolved. BLN design criteria N4-50-D720, R0, dated November 1973 required use of the 1 percent rule for classifying piping as moderate energy. Sequoyah Nuclear Plant (SQN) was designed earlier using the 1 percent rule. (Refer to design criteria SQN-DC-V-1.1.11)

NSRS became concerned about postulated high energy pipe ruptures at BLN during the major management review of OEDC. During the onsite review, the startup and recirculation pumps were observed to be located in proximity to several safety-related pumps. A preliminary investigation by the reviewers indicated that further work would be required to adequately address this issue. Furthermore, it was decided to expand the scope of the investigation to include all postulated breaks in high energy piping outside containment for the sake of completeness.

III. CONCLUSIONS AND RECOMMENDATIONS

The pipe break evaluation for high energy pipe breaks outside containment for BLN generally appears to be adequate to meet regulatory requirements and TVA commitments except as discussed below. Some areas exceed regulatory requirements and contribute an added degree of safety to the BLN design.

R-81-22-BLN-1, Incomplete Documentation of Pipe Break Evaluation

The design basis for features which actively mitigate the consequences of pipe breaks is not adequately documented in the pipe break evaluation to ensure that future design changes do not invalidate the pipe break evaluation. ANSI N45.2.11, section 3.1, requires that design bases be documented.

Recommendation

EN DES should document the basis for concluding that a particular pipe break results in acceptable consequences to facilitate evaluation of future design changes. All design features that are specifically provided to actively mitigate the consequences of a pipe break should be identified with respective design criteria. Refer to section IV.A for details. (R)

R-81-22-BLN-2, Inadequate Justification of Exceptions to Regulatory Guidance

The BLN FSAR, section 3.6, commits TVA to document and justify less conservative criteria than those given in standard review plan sections 3.6.1 and 3.6.2. Contrary to the above, no justification for using the 1 percent rule was found.

Recommendation

EN DES should provide justification of the above exception to regulatory guidance as committed to by the BLN FSAR. Refer to section IV.B. for details. (R)

R-81-22-BLN-3, Minor Improvements to Pipe Break Evaluation

The following items are improvements which should be made to the pipe break evaluation.

1. CEB report 77-10 mentioned several areas which at the time the report was issued were unresolved. The design has now advanced to the point where these areas have been resolved. The resolutions should be documented in the CEB report.

2. CEB report 77-10 did not clearly define when spurious operation of equipment is assumed.

Recommendation

The following clarifications should be made to improve the pipe break evaluation.

1. CEB report 77-10 should be updated to reflect the resolution of several items which were open when the report was issued. Refer to section IV.C.1 for details. (E)

2. CEB report 77-10 should clarify when spurious failures were assumed in more detail. A listing of the conditions under which spurious operation is assumed should be included in the report. Refer to section IV.C.2 for details. (E)

IV. DETAILS

Criterion 4 of Appendix A to 10CFR50 requires that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions of postulated accidents. This requirement was further clarified by the AEC in letters to TVA dated December 18, 1972 and July 12, 1973. Further clarification of these requirements was provided in the NRC's Standard Review Plan (SRP) sections 3.6.1 and 3.6.2 dated November 24, 1975. Branch Technical Positions APCSB3-1 and MEB3-1 are included in these sections of the SRP as attachments.

The BLN FSAR commits TVA to meet the requirements of AEC's July 12, 1973 letter and states that those sections of the SRP referenced above would be used for guidance. TVA Design Criteria N4-50-D720, "Design Criteria for Evaluating the Effects of a Pipe Failure Inside and Outside Containment," provides the designer with detailed guidance to follow in evaluating postulated piping failures. The Civil Engineering Branch (CEB) has issued report CEB 77-10, "Evaluation of the Effects of Postulated Pipe Ruptures Outside Containment," which contains the preliminary results of this evaluation for BLN. This report states that the criteria and methods applied to the evaluation were developed based on 1) AEC letter of December 18, 1972, 2) Regulatory Guide 1.46, May 1973, 3) those sections of the SRP referenced above.

The NSRS review of this area consisted of an examination of the following areas:

- The program for performing the pipe rupture analysis

- The classification of systems as high or moderate energy

- The analysis performed to determine the acceptabilty of a postulated rupture

- The mitigative measures taken to correct identified problems

These areas were evaluated by comparison of the TVA documentation to regulatory requirements and TVA commitments. Additionally, meetings were held with involved employees to strengthen the reviewers' understanding of the pipe break evaluation process. A detailed listing of the persons contacted is given in section V of this report and a detailed listing of the documents reviewed is given in section VI.

In general, EN DES has done an adequate evaluation and had an adequate program in place to control the evaluation process. In some areas regulatory requirements were exceeded but in other areas, however, NSRS concludes that regulatory requirements were not met.

There were two areas where it appeared that the EN DES evaluation exceeded NRC requirements. These are the so called special protection requirements and the evaluation of lines smaller than one inch in diameter. The special protection concept requires that arbitrary pipe failures be postulated if this could result in a complete loss of a safety function. Mitigative measures are then required to ensure that the reactor can be safely shutdown. NSRS feels that a complete loss of a safety function is a serious event as stated in a draft policy statement (NSR 800125 107) and commends EN DES for striving to eliminate these events. NRC does not require that lines smaller than one inch in diameter be evaluated for potential pipe break consequences. EN DES, however, will consider these lines as potential wetting sources and evaluate these during the field evaluation phase of the pipe break study.

A. R-81-22-BLN-1, <u>Incomplete Documentation of Pipe Break Evaluation</u>

A complete record of the pipe break evaluation for BLN had not been developed. There were several documents containing portions of the pipe break study including the following:

- CEB report CEB 77-10

- EN DES technical information document TI-743

- Design criteria N4-50-D720

- Design criteria N4-50-D741

- BLN FSAR section 3.6

The following deficiencies in the documentation were found:

(1) CEB report CEB 77-10 contains the major portion of this study. This report describes those aspects of the pipe break evaluation for which CEB has responsibility. These areas are blowdown force, jet impingement, and pipe whip. The report does

not discuss the results of environmental analyses (e.g., flooding, pressurization, and temperature) and does not discuss design changes made to mitigate pipe breaks before the CEB evaluation was done.

(2) The calculations performed to evaluate subcompartment pressurization after a postulated pipe break are documented in accordance with EP 3.03, "Design Calculations." The corrective actions taken to limit post-rupture pressurization to an acceptable level, however, are not well documented. The ECN process is used to ensure that the changes are made in the design, but the design basis for the change is not well documented. NSRS is concerned that a change made to the plant after it has been operating may not receive an adequate safety review because of this poor documentation. An example of this poor documentation is the addition of check valves in the startup and recirculation system to limit pressurization effects.

(3) Another area that was poorly documented was the basis for concluding that some breaks were acceptable. CEB 77-10 does not discuss the basis for concluding that some breaks were acceptable without design changes. Thus, there did not appear to be adequate documentation to ensure that future plant modifications would not invalidate the pipe break evaluation. This is similar to the concern discussed above concerning pressurization.

(4) An analysis to evaluate the building environment after a pipe break was not found as a part of the pipe break study during the review. Pressurization studies had been made, but temperature profiles, flooding effects, and humidity profiles had not been done and documented. In response to NUREG-0588, temperature and humidity profiles were being calculated to ensure that the environmental qualification of equipment needed to mitigate the postulated pipe rupture would not be exceeded. This analysis should have been done as part of the original pipe break evaluation. The response to NUREG-0588 should provide the additional design input to complete the evaluation. Flooding effects from pipe breaks at BLN are accommodated by a combination of isolation and emergency drains. Although the design appears adequate, the documentation of the design is poor; for example, design calculations documented in accordance with EP 3.03 were not found during this review. Based on the interviews held, there also appeared to be some confusion as to which group within NEB was responsible for this analysis.

NSRS concludes that the current pipe break study is incomplete and poorly documented. The documentation needs improvement so that all the design features that actively mitigate pipe breaks are identified in a manner which would facilitate the evaluation of future design changes. It is not the intent of NSRS to require documentation of all postulated pipe breaks. The large number of such breaks would result in a large workload which would have a large impact on

EN DES. Rather, only those breaks which require an active means of mitigation need additional documentation. In this report active mitigation means that a mechanical action must be made by a component required to mitigate the consequences of the break. Isolation valves, check valves, sump pumps, and relief valves are a few examples of components which make mechanical actions to mitigate events and should be considered as active for the purpose outlined above. NSRS feels that the passive means of mitigating pipe breaks are adequately documented in existing design criteria.

B. R-81-22-BNL-2, Inadequate Justification of Exceptions to Regulatory Guidance

The NRC classifies piping as either high energy or moderate energy depending upon the maximum operating temperature and pressure of the contained fluid. The NRC recognizes that some piping may contain high energy fluids for only a short time and allows this piping to be classified as moderate energy for the pipe break evaluation. The NRC has defined a "short time" as 2 percent of the time that the system operates as a moderate energy system. This definition appears in Branch Technical Position MEB3-1 as footnote 6 on page 3.6.2-14 of the Standard Review Plan. The BLN pipe break evaluation uses 1 percent of the plant operating life to define a "short time." The BLN definition is much more liberal than the NRC defnition and allows several systems to be classified as moderate energy instead of high energy. These systems include the auxiliary feedwater system and the startup and recirculation system.

During the review, the basis for using the 1 percent rule was investigated. Refer to section II for a history of the 1 percent and 2 percent rules.

The 1 percent rule was used in the early design of BLN. Although the 2 percent rule was issued by NRC late in 1975, serious consideration of using this rule for the BLN pipe break study did not begin until early 1977. The auxiliary feedwater system (AFWS) and the startup recirculation system (SRS) would have been reclassified as high energy systems if the 2 percent rule had been adopted. The existing design of BLN would have mitigated the consequences of high energy pipe breaks in the AFWS but significant design changes would have been required to mitigate the consequences of high energy breaks in the SRS. At the time the decision was made to continue the use of the 1 percent rule at BLN, it appeared to EN DES management that the cost of the modifications necessary to meet the 2 percent rule outweighed the licensing risk of not meeting the standard review plan requirements. It was felt that a technical case could be made to justify use of the old 1 percent rule.

The area of NSRS concern here is due to the lack of documentation that an exception to the standard review plan was taken. The rules used for the pipe break evaluation were well documented, but the fact that an exception to the NRC rules was taken was not documented or justified. This is contrary to the BLN FSAR, section 3.6, which states that "any less conservative criteria will be adequately justified and fully documented for each case."

C. R-81-22-BLN-3, Minor Improvements in the Pipe Break Evaluation

A review of CEB 77-10 resulted in NSRS concluding that several deficiencies exit in that report as follows:

1. There are several areas that the pipe break evaluation report, CEB 77-10, indicates are not yet resolved. These areas need to be resolved and the CEB report updated to reflect the resolutions reached. These areas are listed below.

   a. Several unacceptable pipe ruptures in several systems are mentioned as being under study on page 6-2. These pipe ruptures and the changes made to accomodate them have not been documented in the report.

   b. Alternate solutions for protecting the essential air system are mentioned on page 6-7. The alternative has not been documented in the report.

   c. Alternate protection schemes for the control building are discussed on page 6-13. The alternative has not been documented in the report.

2. CEB report CEB 77-10 states that spurious operation of equipment is not assumed unless specific reasons are shown to exist. This statement needs to be clarified to list examples of such reasons and to give the designer more definitive guidance.

NSRS feels that these items represent deficiencies in the pipe break evaluation which should be corrected. NSRS also understands that a revision to CEB report 77-10 is in the process of being written which will correct some of the above concerns.

V. PERSONS CONTACTED

R. D. Adkison
*C. P. Baxter
E. G. Beasley
*R. H. Bryan
J. C. Carter
W. A. English
*P. A. Evans

*B. B. Neely
*H. G. O'Brien
*D. C. Phung
T. C. Price
C. Sohn
L. Warrix

*Also attended exit meeting.

VI. REFERENCES

A. Proposed American National Standard, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," ANSI N-176, December 1979

B. Proposed American National Standard, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture," ANSI N176, Draft 7, May 1975

C. Letter to J. E. Watson from A. Giambusso, AEC Deputy Director, dated December 18, 1972

D. Letter to J. E. Watson from J. F. O'Leary, AEC Director, dated July 12, 1973

E. BLN FSAR Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping"

F. Design Criteria for Steam Generator Startup and Recirculation System, N4-CR-D740, R1, dated June 6, 1977

G. Design Criteria for Evaluating the Effects of a Pipe Failure Inside and Outside Containment, N4-50-D720, R4, dated November 8, 1979

H. Branch Technical Position APCSB3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," dated November 24, 1975

I. Branch Technical Position MEB3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," dated November 24, 1975

J. NRC Standard Review Plan, Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," dated November 24, 1975

K. CEB report CEB 77-10, "Bellefonte Nuclear Units 1 and 2 Evaluation of the Effects of Postulated Pipe Rupture Outside Containment," R0, dated May 4, 1977

L. Nuclear Shielding and Analytical Design Section Technical Information, TI-743, "Overpressurization Effects of High Temperature Pipe Ruptures Outside Containment," R1, dated May 5, 1978

M. Engineering Change Notice, ECN-451, on Bellefonte Nuclear Plant dated July 24, 1978 (BLP 780724 005)

N. Design Criteria N4-50-D740, "Physical Separation Outside of Primary Containment," R0, dated November 22, 1974

TVA 64 (OS-9-65)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS '82 0629 050

TO : M. N. Sprouse, Manager of Engineering Design, W11A9 C-K

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : June 28, 1982

SUBJECT: ALL NUCLEAR PLANTS - REALIGNMENT OF OPEN TEST LINES - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-82-04-NPS

Please refer to your memorandum to me dated August 31, 1981 (NEB 810831 293).

In previous correspondence regarding safety systems required to mitigate design basis events, there seemed to be good agreement that required monthly testing of safety systems should not prevent the systems from performing intended safety functions. In your August 31, 1981 response you requested specific examples where NSRS considered potential problems existed in the design.

In the attached report two specific cases are identified that we believe exist at both the Sequoyah and Watts Bar Nuclear Plants. We are not aware of any situations at Browns Ferry or at Bellefonte; however, in the case of Bellefonte, a complete review has not been completed.

Our examination has been limited to the identification of situations where we believe potential problem areas exist. Although we have recommended automatic realignment of valves, the NSRS review was not expanded to determine if alternate solutions are possible.

If you have any questions, please call the NSRS contact in this matter--Bruce F. Siefken at extension 6860.

H N Culver

H. N. Culver

HNC:BFS:LML

Attachment

cc (Attachment):

G. F. Dilworth, E12D46 C-K

MEDS, W5B63 C-K

H. G. Parris, 500A CST2-C

NSRS FILE

TENNESSEE VALLEY AUTHORITY

NUCLEAR SAFETY REVIEW STAFF

REVIEW

NSRS REPORT NO. R-82-04-NPS

SUBJECT: ALL NUCLEAR PLANTS - REALIGNMENT OF OPEN TEST LINES

REVIEWER: Bruce F. Siefken 6/28/82

BRUCE F. SIEFKEN DATE

APPROVED BY: James A. Crittenden 6-29-82

JAMES A. CRITTENDEN DATE

TABLE OF CONTENTS

I. SCOPE

This routine review investigated the ability of TVA nuclear plants to mitigate an accident while an accident-mitigation system was being tested. The review included Browns Ferry (BFN), Sequoyah (SQN), Watts Bar (WBN), and Bellefonte (BLN) Nuclear Plants. The review for BLN, however, was limited since a complete set of surveillance instructions had not been prepared at the time of the review. Standby fluid systems needed to mitigate the consequence of accidents were reviewed to determine the manner in which these systems are tested and if features exist which would automatically place the system into its accident-mitigation mode.

II. BACKGROUND

On June 23, 1980, NSRS wrote EN DES concerning the need to realign open test lines. Refer to H. N. Culver's memorandum to M. N. Sprouse dated June 23, 1980 (GNS 800625 002). This memorandum outlined the reasons why NSRS felt that TVA nuclear plants should be designed such that the testing of safety systems would not impact the ability of the plant to withstand the effects of a single failure while mitigating an accident. This original request was expanded to include any system realignment required by monthly testing procedures in H. N. Culver's memorandum to M. N. Sprouse dated January 20, 1981 (GNS 810120 002). EN DES replied to these concerns in M. N. Sprouse's memorandum to H. N. Culver dated August 31, 1981 (NEB 810831 293). In this reply it was stated that an indepth study had not been made of NSRS's concern and that EN DES would investigate specific examples of NSRS's concerns. In the EN DES memorandum it was indicated that EN DES believes the design is such that the plants can withstand an open test line and a single failure. This NSRS review was undertaken to do a more thorough review of the design of TVA nuclear plants.

III. CONCLUSIONS AND RECOMMENDATIONS

The NSRS review indicated that the design of two safety systems (containment spray and safety injection) at WBN and SQN are such that during surveillance testing of portions of these systems, the intended mitigating features of these systems would not be immediately available to mitigate design basis accidents assuming a single failure in the safety system or its essential support systems. This is contrary to the intended design of these systems as indicated in the EN DES response dated August 31, 1982 (NEB 810831 293).

A. R-82-04-NPS-1, <u>Containment Spray Test Line at Sequoyah and Watts Bar Nuclear Plants</u>

Sequoyah Surveillance Instruction (SI)-037, "Containment Spray Pump Test," and Watts Bar SI-4.0.5.72-P, "Containment Spray Pumps," require that the recirculation line to the Refueling Water Storage Tank (RWST) be open during testing of the containment spray pumps and allow this test to be run with the reactor at power. This situation could result in the containment spray system being inoperable after a single failure when one train is out of service for testing.

Recommendation

NSRS recommends that automatic isolation of this test line at SQN and WBN be provided to isolate this line if an accident occurs whenever the containment spray system is required to be operable. Refer to Section IV.A for details. (E)

B. R-82-04-NPS-2, Safety Injection Pump Operability Test at Sequoyah and Watts Bar Nuclear Plants

SQN SI-129, "ECCS Safety Injection Pump Operability," and Watts Bar SI-4.0.5.63.P, "Safety Injection Pumps," require that the cold leg injection line from the tested pump be closed during the test. There are no automatic signals to open this valve in the event of an accident. The situation could result in the ECCS being inoperable if single failure occurs coincident with testing.

Recommendation

NSRS recommends that the cold leg injection line at SQN be automatically realigned in the event of an accident requiring safety injection while the safety injection pumps are being tested. Refer to section IV.B for details. (E)

IV. DETAILS OF THE REVIEW

This review was performed by reviewing a number of documents and drawings including surveillance instructions, flow diagrams, design criteria, design criteria diagrams, and logic diagrams. Refer to section V for a more detailed list of the documents reviewed. The process used in reviewing the design for potential testing unavailability concerns consisted of first reviewing the surveillance instructions for several safety-related systems on a particular plant. The flow or design criteria diagram was then reviewed to determine whether the surveillance instruction aligned the system under test in such a manner that it would be unavailable to mitigate an accident. Logic diagrams were then consulted to establish whether an accident signal would realign the system to its accident mitigation configuration. Finally, design criteria or FSAR descriptions were reviewed to confirm the accident-mitigation role of the system under review. A concern was determined to potentially exist if the following conditions were found.

°The safety system was required immediately after an accident occurred.

°The portion of the safety system under test conditions was not immediately operable to mitigate accident conditions as required.

°A single failure in the remaining part of the safety system or its essential support systems (e.g., electrical power) prevented the safety system from performing its function as required to control the accident consequences.

The following concerns were identified using the process described above.

A. R-82-04-NPS-1, <u>Containment Spray Test Line at Sequoyah and Watts Bar Nuclear Plants</u>

Surveillance instruction SI-37 and SI-4.0.5.72.P require that the containment spray pumps be shown to be operable at least every 31 days while the reactor is at power. This testing could be performed with the reactor in any mode and was to be performed by recirculating borated water through the RWST via the eight-inch containment spray test line. This line contains three manual valves which are locked closed, two of which must be opened to test a containment spray pump. These valves are numbered 72-502, 72-503, and 72-504. Thus, while a containment spray pump is being tested it is unavailable to mitigate an accident since the flow of borated water is back to the RWST, not to the containment spray headers. The containment spray system is assumed to be immediately operable in the accident analyses of Chapter 15 of the FSAR. With the present method of testing and the present system design, the containment spray system would not be able to perform its intended safety function following an accident. Thus, this system is identified as one example of the concern previously raised.

Additionally, Sequoyah surveillance instruction SI-51, "Weekly Chemistry Requirements," required that the RWST boron concentration be sampled weekly per Technical Instruction (TI)-16, method B.86. This method required that the contents of the RWST be recirculated for at least 24 hours before sampling. If the containment spray system pump is used to provide this recirculation, the containment spray system would be unavailable for accident mitigation. Since the containment spray system is needed before operator action could be assumed (10 minutes) NSRS considers this another example of the concern previously identified.

NSRS is cognizant that EN DES has become aware of a need to isolate these valves for different reasons since their August 31, 1981 response and has requested NUC PR to write a design change request to add this automatic isolation feature to SQN in modification request transmittal 1-181. Refer to J. A. Raulston's memorandum to R. W. Cantrell dated November 2, 1981 (NEB 811102 270). NSRS supports the EN DES request and urges NUC PR to comply. Similarly, an ECN (3334) is being processed for Watts Bar.

B. R-82-04-NPS-2, <u>Safety Injection Pump Operability Test at Sequoyah and Watts Bar Nuclear Plants</u>

Sequoyah surveillance instruction SI-129 and Watts Bar SI-4.0.5.63.P required that the safety injection pumps be tested every 31 days when the plant is at power. This testing was to be done by running a safety injection pump on recirculation flow to the RWST via the mini-flow line for the pump. This instruction further requires that the pump discharge line to the cold leg injection points be isolated by closing either valve 63-152 or 63-153. There were no automatic signals to open these valves resulting in the safety injection pump being unavailable to mitigate accidents. Thus, a single failure and a safety injection pump under test could result in a situation where no safety injection would be available to mitigate an accident.

Since the safety injection system is needed before operator action can be assumed, NSRS considers this another example of the concerns previously raised.

V. DOCUMENTS REVIEWED

A. Bellefonte Nuclear Plant Surveillance Instructions

1. BLSI 4.1.2.5-1, Decay Heat Removal Pump 1ND-MPMP-001-A Monthly Operability Test

2. BLSI 4.1.2.5-2, Decay Heat Removal Pump 1ND-MPMP-002-B Monthly Operability Test

3. BLSI 4.6.2.1.6-1, Reactor Building Spray Pump 1NS-MPMP-001-A Monthly Operability Test

4. BLSI 4.6.2.1.6-2, Reactor Building Spray Pump 1NS-MPMP-002-B Monthly Operability Test

5. BLSI 4.6.2.3.9-1, RB Cooler VJ-MCCR-031B Operational Test

6. BLSI 4.6.2.3.9-2, RB Cooler VJ-MCCR-032A Operational Test

7. BLSI 4.6.2.3.9-3, RB Cooler VJ-MCLR-033B Operational Test

8. BLSI 4.7.1.2.A.1-1, Auxiliary Feedwater Pump 1CA-MPMP-001-A Monthly Operability Test

9. BLSI 4.7.1.2.A.1-2, Auxiliary Feedwater Pump 1CA-MPMP-002-B Monthly Operability Test

10. BLSI 4.7.1.2.9.2-2, Auxiliary Feedwater Turbine Driven Pump 1-CA-MPMP-003-Q Bearing Temperature Test

B. Browns Ferry Nuclear Plant Surveillance Instruction

1. 3.1.1, Core Spray Pump Performance

2. 3.1.2, Residual Heat Removal Pump Performance

3. 3.1.3, Residual Heat Removal Service Water Pump Performance

4. 3.1.4, Emergency Equipment Cooling Water Pump Performance

5. 3.1.5, High Pressure Coolant Injection Pump Performance

6. 3.1.6, Reactor Core Isolation Cooling Pump Performance

7. 3.1.7, Standby Liquid Control Pump Performance

C. Sequoyah Nuclear Plant Surveillance Instructions

1. SI-015, R7, Emergency Core Cooling System Residual Heat Removal Loop 4 Reactor Coolant System Isolation and Containment Sump

2. SI-037, R9, Containment Spray Pump Test

3. SI-046, R7, Component Cooling Water Pumps
SI-051, R13, Weekly Chemistry Requirements

4. SI-068, R2, Functional Test of Containment Spray Pumps and Associated Valves

5. SI-118, R5, Auxiliary Feedwater Pump and Valve Automatic Actuation

6. SI-118.01, R3, Turbine-Driven Auxiliary Feedwater Pump and Valve Automatic Actuation

7. SI-119, R3, Essential Raw Cooling Water Auto Actuation from an SI Signal

8. SI-128, R12, Emergency Core Cooling System Residual Heat Removal Pumps

9. SI-129, R11, Emergency Core Cooling System Safety Injection Pump Operability

10. SI-130.01, R2, Turbine-Driven Auxiliary Feedwater Pump

11. SI-130.02, R2, Motor Driven Auxiliary Feedwater Pumps

D. Watts Bar Nuclear Plant Surveillance Instructions

1. SI 4.0.5.3.P, R3, Auxiliary Feedwater Pumps

2. SI 4.0.5.63.P, R2, Safety Injection Pumps

3. SI 4.0.5.72.P, R3, Containment Spray Pumps

4. SI 4.0.5.74.P, R2, Residual Heat Removal Pumps

5. SI 4.1.2.6.a.1, R1, Weekly Reactivity Control Systems Boric Acid, Boron Injection, and Refueling Water Storage Tanks Boron Determination

E. Sequoyah Nuclear Plant Technical Instruction TI-16, R15, Sample Points and Sampling Methods

F. Watts Bar Nuclear Plant Technical Instruction TI-16, R13, Sampling Methods

G. Bellefonte Drawings

1. Design Criteria Diagram - Reactor Building Spray System 3BW0615-NS-01, R8

2. Design Criteria Diagram - Decay Heat Removal System 3BW0612-ND-01, R7

3. Functional Control Logic - Diagram - Decay Heat Removal System, 2GW0900-ND

4. Functional Control Logic - Diagram Reactor Building Spray System, 2GW0900-NS

5. Design Criteria Diagram - Auxiliary Feedwater System, 3BW0618-CA

6. Functional Control Logic Diagram - Auxiliary Feedwater System, 2GW0900-CA

H. Browns Ferry Drawings

1. FSAR Figure 6.4-1, High Pressure Coolant Injection System Process Diagram

2. FSAR Figure 6.4-2, Core Spray System Process Diagram

3. FSAR Figure 6.4-3, Residual Heat Removal Process Diagram

4. FSAR Figure 10.6-1a, Reactor Building Closed Cooling Water System Piping and Instrumentation Diagram

5. FSAR Figure 10.9-1a, RHR Service Water System Flow Diagram

6. FSAR Figure 10.10-1a, Emergency Equipment Cooling Water Flow Diagram

I. Sequoyah/Watts Bar Drawings

1. Flow Diagram Auxiliary Feedwater System, 47W803

2. Logic Diagram Auxiliary Feedwater System, 47W611-46

3. Flow Diagram Safety Injection System, 47W811

4. Logic Diagram Safety Injection System, 47W611-63

5. Flow Diagram Containment Spray System, 47W812

6. Logic Diagram Containment Spray System, 47W611-72

TVA 64 (OS-9-81)

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

GNS 83 0225 050

TO : H. G. Parris, Manager of Power, 500A CST2-C

FROM : H. N. Culver, Director of Nuclear Safety Review Staff, 249A HBB-K

DATE : February 25, 1983

SUBJECT: SPECIAL PROGRAM MANAGEMENT REVIEW OF THE OFFICE OF POWER WATER QUALITY PROGRAM - NUCLEAR SAFETY REVIEW STAFF REPORT NO. R-82-08-NPS

References:

1. My memorandum to you dated June 11, 1982 (GNS 820614 051), "Special Review of the Office of Power Water Quality Program - Nuclear Safety Review Staff Report No. R-82-08-NPS"

2. My memorandum to H. A. Taff dated August 26, 1982 (GNS 820826 050), "Special Review of the Office of Power Nuclear Water Quality Program - Nuclear Safety Review Staff Report No. R-82-08-NPS"

Attached is the NSRS report of a special program management review conducted in accordance with the plans delineated in the referenced memoranda. The review dealt with POWER's programs and procedures for implementing its responsibilities in the area of water quality control and associated chemical activities. This review has involved a significant number of your key staff members and has required a closely coordinated effort by your staffs at the Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants; at the Power Operations Training Center; at the Central Laboratory; and at the NUC PR Central Office. Your staff provided excellent cooperation with a professional attitude which allowed us to complete our review in a meaningful manner. Particularly commendable is our observation that in some cases corrective actions for deficiencies identified at the exit meetings were promptly initiated by those facilities and organizations reviewed.

The details of the review findings are somewhat extensive. Only a representative number of the identified weaknesses have been discussed in the details and these have been combined into only 10 NSRS positions (recommendations) that require resolution.

This report represents a new NSRS format for reporting review findings with the eventual objective of less formality and more flexibility in terms of resolution of perceived problems and the time frame for resolution. Through this approach NSRS believes a more thorough consideration of meaningful resolutions of broader problem areas can be provided by the line organization and a better information exchange between NSRS and the line can be achieved through informal discussions. To aid in this approach, a formal written response to the NSRS positions as detailed in section IV of this report is not required. NSRS will perform a follow-up review in approximately six months to reassess the water quality program and the corrective action taken. In addition, the NSRS reviewers (including those that are

NSRS FILE

GNS '83 0225 051

TENNESSEE VALLEY AUTHORITY
NUCLEAR SAFETY REVIEW STAFF
SPECIAL PROGRAM MANAGEMENT REVIEW
NSRS REPORT NO. R-82-08-NPS

SUBJECT: REVIEW OF THE OFFICE OF POWER WATER QUALITY PROGRAM

DATES OF REVIEW: JULY 19 THROUGH SEPTEMBER 16, 1982

FACILITIES EVALUATED: NCO, BFN, SQN, WBN, POTC, CLS, NSRB, OPQA&A

TEAM LEADER: Robert C. Sauer — ROBERT C. SAUER — 18 Feb 83 DATE

REVIEWERS: Leonard F. Blankner — LEONARD F. BLANKNER — 18 Feb. '83 DATE

Gerald G. Brantley — GERALD G. BRANTLEY — 18 Feb. 83 DATE

APPROVED BY: K. W. Whitt — KERMIT W. WHITT — 2/22/83 DATE

I. BACKGROUND

Appendix A to 10CFR50 establishes minimum requirements to be used as principal design criteria in constructing water-cooled nuclear power plants. These criteria have been established to provide assurance that structures, systems, and components important to safety are designed and constructed with sufficient margin to ensure that the facility can be operated without undue risk to the health and safety of the public. Of these criteria, several require assurance that heat transfer and containment barriers be established and maintained along with reactivity, fuel, and radioactivity controls to ensure:

- The reactor coolant pressure boundary will have minimal probability of gross rupture or rapidly propagating failure (criteria 14 and 31).

- The reactor coolant system and auxiliary support cooling and seal water systems are designed with sufficient margin to ensure their heat transfer function of removing excess heat from the reactor core and from structures, systems, and components important to safety, to an ultimate heat sink will occur during any condition of normal operation, including anticipated operational occurrences (criteria 15, 34, 35, and 44).

- Variables and systems that can affect the reactor coolant pressure boundary, the containment and its associated systems will be monitored by instrumentation over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety (criterion 13).

- Core reactivity changes will be reliably controlled during planned, normal power changes to preclude exceeding fuel design limits and to ensure the capability to cool the core can be achieved under postulated accident conditions whenever utilizing soluble poison shim controls in conjunction with or independent of the use of control rods (criteria 26, 27, and 28).

- Radioactive material releases when made to the environment from normal operations, including anticipated operational occurrences and from postulated accidents will be controlled and maintained (criteria 60 and 64).

- Fuel and radioactive onsite waste storage systems will be monitored by instruments for surveillance and to detect loss of residual heat removal capability and shielding (criteria 61 and 63).

In addition to the Appendix A criteria, NRC has required, in part, through Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," that measures are to be established to control those activities which could affect the safety-related functions of those structures, systems and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. Therefore, quality assurance criteria are to be established for all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

To implement those criteria that deal with assuring that pressure boundaries of the reactor coolant system or other structures, systems, and components important to safety do not fail as a result of the mechanisms of general and stress corrosion or fouling, a water quality program is to be instituted from the time of construction and continue through all operations until final shutdown (see as examples NRC Regulatory Guides (RG) 1.37, 1.39, and 1.56). To implement those criteria that deal with reactivity, fuel, and radioactivity controls, the water quality program should be expanded to reflect these criteria at the time the station applies for an operating license and continue through all operations until the plant is decommissioned and decontaminated. The water quality program therefore is required to assure maintenance of high water quality at all times, through all phases of plant evolution; to reduce those impurities present which induce corrosion, fouling, and plant radiation to the lowest levels acceptable through state-of-the-art treatment practices; and to ensure plant effluents meet environmental and regulatory requirements.

II. SCOPE

The NSRS review was an evaluation of the administrative controls and implementation practices of the Office of Power's (POWER) Water Quality Program and related chemical activities being conducted within the Division of Nuclear Power (NUC PR) Central Office (NCO), Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), Watts Bar Nuclear Plant (WBN), the POWER Operations Training Center (POTC), and Central Laboratories Services (CLS). Discussions were also held with representatives from the NUC PR Steam Generator Task Force, the Nuclear Safety Review Board (NSRB), and the Office of Power Quality Assurance Staff as to their involvement in TVA's overall water quality control program.

To accomplish this task, the programs for management control of water quality-related activities were reviewed for compliance with regulatory requirements and commitments, to the latest standards which relate to chemistry program management controls, and to good quality and safety practices established by industry.

The review was intended to be broad in scope with a depth commensurate with the degree NSRS perceived necessary to adequately assess the reviewed area's effectiveness and was based on a concept that incorporates a number of key elements important to a successful water quality program.

Nine functional review area topics containing idealized criteria established from NRC requirements, TVA commitments, and current industry practice were selected. The functional review area topics included:

A. Training and Qualification of Personnel
B. CSSC Water Chemistry Specifications and Surveillance/ Action Requirements
C. Chemical and Radiochemical Laboratory Analytical and Sampling Procedures
D. Laboratory Quality Control
E. Control of Bulk and Reagent Grade Chemicals
F. Nonconformance and Corrective Action Controls
G. Special Chemistry Considerations
H. Raw Water Treatment Practices
I. Chemical Measuring and Test Equipment (M&TE)

III. MANAGEMENT SUMMARY

This review of POWER's Water Quality Program was conducted by NSRS to provide an independent assessment of whether a satisfactory level of nuclear safety/quality had been provided in the area of water quality control. The intent of the review was to determine whether a written program had been established to satisfy TVA policy, regulatory requirements, and TVA commitments; whether procedures used to conduct the program are adequate; whether the program was being implemented effectively; whether the cognizant personnel throughout the responsible organizations were aware of their responsibilities and authority in carrying out the program; and whether personnel were qualified to perform their assigned activities.

Many positive aspects of the nuclear water quality program were observed during the review. Some examples are as follows:

- The concept of a central office Chemical Engineering Group (CEG) to disseminate program requirements, provide guidance and assistance to ensure adequate chemistry control, to monitor station activities, and to establish methods for resolving underlying causes of problems.

- The concept of issuing guidelines for nuclear plant water quality control in the form of division directives to promote consistency and efficiency.

- The level of related technical training of those personnel involved in the NUC PR water quality program is impressive and a valuable TVA resource.

- NUC PR plans to provide centralized controls and services for procurement, calibration, and maintenance of radiochemical laboratory nuclear counting equipment should promote efficiency and provide overall program improvement.

- The development and implementation of the POTC phase of the Radiochemical Laboratory Analyst Training Program is impressive.

- The Radiochemical Laboratory Analysts are making positive contributions to the water quality control program.

- The Radiochemical Laboratory Manual employed at BFN and POTC are good concepts.

- Some of the prompt and progressive management techniques now being employed by the Chemical Unit supervisor at SQN to facilitate program improvement following the startup phase of station operations are commendable. These techniques include:

  - Increased supervisor involvement at all levels of the Chemical Unit activities.

  - Establishment of good communications with Chemical Unit employees providing for good feedback for program improvement.

  - Establishment of internal unit reviews to assess degree of program implementation and to identify deficiencies.

- The high regard for quality work at Central Laboratory Services (CLS) is reflected by the attitude of the facility management and personnel and the involvement of their QA/QC manager in the development and implementation of their programs.

The review also identified a significant number of deficiencies or weaknesses as described in section V. These deficiencies were evaluated for root cause and on the basis of this evaluation a number of conclusions and NSRS positions are discussed in section IV.

NSRS believes that a large number of the concerns and weaknesses identified during this review can be attributed to decentralized control of the water chemistry program. Effectiveness of the NUC PR Chemical Engineering Group (CEG) is drastically limited by an apparent decentralization that inhibits cooperation and communication between the CEG and the site chemical units. Organizationally, the CEG appears to be established such that it is responsible for providing guidance and assistance as needed to ensure adequate water quality control at the nuclear plants and for assuring effective program implementation. CEG has made an attempt at providing appropriate measures in establishing a meaningful, consistent, and verifiable nuclear water quality program through issuance of various NUC PR Division Procedure Manual (DPM) directives. However, CEG has not met the goals they set for themselves in establishing criteria in that sections of the base division directive for a water quality program have not been issued and are overdue. In addition, the CEG does very little to assure proper or consistent implementation of any program requirements and did not appear to have adequate knowledge of how the water chemistry programs were being implemented at the plants. While a number of weaknesses were identified that were attributable to the CEG, the perceived problem of decentralization between the NCO and plants was not considered by the reviewers to result from a lack of interest or a willing reluctance to accept responsibilities on the part of the CEG. Rather, it appeared to be a condition that had evolved over a period of time that was accepted by the plants and NCO management or possibly from a conscious decision by the plant staffs and NCO management to limit the participation of the CEG in plant activities. The results of this condition are that problem areas and their underlying root causes are not systematically identified by NUC PR and methods for evaluation and corrective actions are not developed and implemented until identified by outside organizations. Realistically, outside organizations cannot identify for resolution all the problems that exist in the time frames available for reviews and audits. Therefore, it is probable that additional major program deficiencies exist throughout the program.

NSRS believes that many of the identified deficient conditions could have been prevented or mitigated by an involved CEG with sufficient authority, initiative, and credibility to establish comprehensive criteria and monitor as well as assist plant actions to implement the program. It was the expressed opinion of the NCO managers interviewed that an autonomous plant management method was supported by upper-division management and the CEG lacked authority to become involved in day-to-day plant operation. NSRS realizes that a few weeks of review and observations cannot compare with the continuous involvement of POWER management with the multifaceted management processes that are interrelated with the single area with which this review dealt.

In summary, in examining the nine functional areas identified in this report it was determined that nuclear safety/quality in the area of water quality control needs program improvement. Written programs had not been established in all cases to satisfy TVA policy, regulatory requirements, and TVA commitments. In some cases the procedures being used to conduct the water quality program were inadequate. Procedures had been issued from the NCO in the form of division directives containing mistakes without first being tested and proven correct (qualified). Some procedures were found badly out of date at some of the facilities. Where written programs had been established there were problems with the details of implementation. CEG personnel were somewhat confused as to their responsibilities and authority as related to the role they were to play in support of nuclear plant activities. The Chemical Unit personnel at the stations were well aware of their responsibilities and authority as defined by their station management but were unsure of their responsibilities as related to the implementation of division directives as issued by the NCO and of their relationship with the CEG. The personnel involved with the NUC PR water quality program were found to be technically well qualified to perform their assigned activities. However, the cognizant CEG and plant personnel were found to be unfamiliar with some major regulatory requirements, TVA commitments, and station requirements. Overall each plant's chemical unit was conducting activities in a autonomous manner without effective technical guidance from the CEG. The results are that no theme or master plan has been established to identify program requirements, needs, and deficiencies except for the attempt made by CEG which has not been effective.

IV. CONCLUSIONS AND NSRS POSITIONS

The following paragraphs contain conclusions followed by NSRS positions to correct perceived weaknesses in the POWER Nuclear Water Quality Program. Specific findings, positive as well as negative, are presented in section V for each area evaluated. Additionally, where the finding has indicated deficiencies, recommendations have been presented. The report details are provided for the awareness of the affected facility and to indicate the bases for the overall conclusions presented in sections III and IV. The conclusions in this section identify programmatically what the specific weaknesses listed in the details have indicated.

A. Chemical and Radiochemical Program Controls

1. R-82-08-NPS-01, Requirements/Needs/Activities Matrix

The N-OQAM itself is not self-sufficient but relies on division directives (DPMs) to provide supplemental detailed requirements where necessary. Discussions held with NUC PR CEG personnel disclosed that they were not comprehensively aware of what commitments and requirements had been made on behalf of water quality control or how these had been incorporated into directives and procedures except for the requirements of the specific plant technical specifications.

NSRS Position

A requirements/needs/activities matrix should be developed by CEG to identify and tabulate all requirements and TVA commitements; all the necessary program needs, such as qualifying analytical procedures, chemicals, personnel, etc.; and all respective activities that should be controlled. The requirements of the matrix should be indexed to the source control documents for tracking and verification of implementation. A similar position involving plant personnel awareness of plant requirements and commitments was previously identified to NUC PR as a recommendation (R-81-08-BFN-38) in NSRS report No. R-81-08-BFN dated May 15, 1981. The NUC PR response dated October 13, 1981 indicated that action was being taken on that recommendation. This recommendation expands the previous one to cover plant requirements and commitments by activity.

2. R-82-08-NPS-02, Quality Assurance Program for Chemistry Activities

The POWER quality assurance program was found devoid of controls required to be placed over safety-related chemistry activities. As a result, chemical and radiochemical program controls were not established to the degree warranted.

NSRS Position

Safety-related chemistry activities should be included in the POWER QA program.

B. Organization and Responsibilities

1. R-82-08-NPS-03, Chemistry Program Organization and Responsibility Review

NUC PR's functionally decentralized organization that has evolved has bred autonomy among its organizational elements. Presently there are six project control staffs for chemical and radiochemical control within NUC PR. These include BFN, SQN, WBN, BLN (not reviewed), CEG, and POTC. Each staff has duplicated the efforts of the other, developing chemical and radiochemical analytical procedures, laboratory equipment calibration, etc. Confusion exists as to the responsibility relationship of the CEG with the plants and POTC.

NSRS Position

POWER should reexamine the assignment of authority and responsibility for chemical and radiochemical control to assure that authority, accountability, and responsibility is specifically defined and delegated.

C. Chemical and Radiochemical Program Administration

1. Division of Nuclear Power

a. NCO

(1) R-82-08-NPS-04, Procedural Controls for Conducting Safety/Quality Affecting Activities Within CEG

NSRS review of CEG activities found that no procedural controls had been formulated to accomplish the safety/quality affecting activities being performed by this group. As a result, certain actions taken by the group circumvented normal administrative controls for assuring safety/quality objectives were maintained.

NSRS Position

CEG should develop procedural controls to formalize its activities.

(2) R-82-08-NPS-05, Program Improvement

CEG had become a reactionary group rather than a forward-thinking group. Part of this development had occurred as a result of CEG being confined to some degree to the central office. This confinement may have resulted indirectly by self-imposition due to the relationship between the CEG and the plants.

NSRS Position

CEG needs to be given strong management support which allows its analysts to perform their prescribed functions.

(3) R-82-08-NPS-06, Internal Review and Feedback Process

CEG had no internal review mechanism to apprise the group of administrative and program weaknesses; to identify which of its activities need to be more formally controlled; to verify through onsite reviews the implementation of its directives; to perodically advise management of overall chemistry program status and effectiveness; and to recommend corrective action when CEG activities fail to comply with POWER/NUC PR-approved procedure or regulatory requirements.

NSRS Position

Responsibility should be established within CEG to conduct internal reviews of its activities and assess the degree of implementation of NCO-issued division directives. These reviews should be performed by qualified persons who do not have responsibility for performing or directly supervising work activities being reviewed.

(4) R-82-08-NPS-07, Verification of Onsite RLA Training

After RLA trainees leave the training center following their 12-week orientation in chemical and radiochemical principles, administrative and regulatory requirements, and program indoctrination, they undergo an additional 21-month program of inplant training. Though the NCO Traning Branch is charged with the responsibility for preparing, administering, and directing training programs, no onsite involvement or program effectiveness appraisals are being accomplished in the area of RLA training. Sites train and certify RLAs under their own program implementation scheme of the division training plans. There is some indication that the training program breaks down at the plant level.

NSRS Position

The Training Branch should assess onsite RLA training activities at periodic intervals.

b. POTC

(1) R-82-08-NPS-08, Calibration and Radiochemical Laboratory Program Documentation

The Laboratory and Training Unit of POTC performs both a germanium detector calibration and a radiochemical laboratory analysis function in support of TVA's nuclear program. NSRS review of these activities indicated that major controls of the facility's operation had not yet been prepared or were fragmented into instruction letters or a partially completed radiochemical laboratory manual. Neither document receives upper-tier approval or plant concurrence of these operations.

NSRS Position

Coalesce all chemistry, radiochemistry, and calibration procedures and program descriptions into a QA program document to define the POTC QA responsibilities to the licensed plants. This manual should receive upper-tier approval.

2. Central Laboratories Services

a. R-82-02-08-NPS-09, Integrated Calibration and Chemical Program Development

Though CLS has established controls for its M&TE calibration program, formal controls for the safety-related chemical support functions have not been established.

NSRS Position

CLS should expand the formal controls to cover the chemistry and other quality affecting activities to define the laboratory's QA responsibilities to NUC PR.

D. Technical/Regulatory Issues

1. R-82-08-NPS-10, Items Requiring Management Attention for Resolution

NSRS' review of the POWER chemical and radiochemical control program identified three significant conditions adverse to quality which deserve management attention.

a. BFN Regulatory Guide 4.15 Program and Laboratory Quality Program

NSRS review of the BFN RG 4.15 QA program identified weaknesses such as failure to provide adequate written procedures required for count room equipment calibration and use; no formal intralaboratory quality control program in effect; and failure to take prompt corrective action to correct a condition adverse to quality.

b. <u>BFN Technical Specifications for Dose Equivalent I-131</u>

BFN technical specifications for determining coolant dose equivalent I-131 activity is deficient in that it does not require special surveillance sampling following transients when the equilibirum value as determined once a month is less than 0.032 uc/gm. The technical specification as presently written does not provide the assurances indicated in the "Bases" and the "Bases" do not provide a technical bases for assuring that following one or more transients the activity level will not exceed 3.2 uc/gm. NSRS considers the technical specifications to be deficient and believes they should be rewritten or proper justification for the existing technical specifications provided.

c. <u>Issuing of Directives Contrary to TVA Commitments</u>

Because of a lack of internal control procedures, NCO-CEG had issued directives which had resulted in chemical control parameters being exceeded and regulatory administrative requirements being violated.

<u>NSRS Position</u>

The program weaknesses described above should receive management attention to assure compliance with TVA commitments.