50-397

ABB

January 15, 1999
LD-99-002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Report of a Defect Pursuant to 10 CFR 21 Concerning Incorrect Modeling of a BWR Lower Plenum Volume in BISON

The purpose of this letter is to notify the Nuclear Regulatory Commission of a defect under 10 CFR 21, "Reporting of Defects and Noncompliance." The defect concerns the Operating Limit for the Minimum Critical Power Ratio (MCPR) in Boiling Water Reactors (BWRs) analyzed using the BISON fast transient analysis code. Specifically, the defect involves incorrect modeling of the reactor vessel lower plenum volume in the BISON code, which could lead to the establishment of non-conservative MCPR Operating Limits in plant technical specifications.

The Enclosure summarizes the evaluation performed by ABB Combustion Engineering (ABB-CE). If you have any questions, please feel free to contact me or Virgil Paggen of my staff at (860) 285-4700.

Very truly yours,
COMBUSTION ENGINEERING, INC.

Ian C. Rickard, Director
Nuclear Licensing

Enclosure: As stated

cc: M. A. Barnoski (ABB-CE)

IE19

ABB Combustion Engineering Nuclear Power

Combustion Engineering, Inc.
P.O. Box 500
2000 Day Hill Rd.
Windsor, CT 06095-0500
Telephone (860) 688-1911
Fax (860) 285-5203

9901220193

(vi) In the case of a basic component which contains a defect or fails to comply, the number and location of all such components in use at, supplied for, or being supplied for one or more facilities or activities subject to the regulations in this part:

The BISON computer code is a proprietary ABB Atom AB program used by ABB-CE for performing BWR safety analyses for U.S. customers. The defect applies only to reload fuel assemblies currently in operation at WNP-2.

(vii) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action:

An evaluation of the defect has been performed by ABB-CE. The results of that evaluation show that there is no safety problem at WNP-2 for operation of the previous Cycles 12 and 13 and its current Cycle 14 operating state (Normal Scram Speed, Recirculation Pump Trip operable, and Turbine Bypass operable) prior to 5,000 MWd/MTU. That is, the established MCPR Operating Limits currently in the plant technical specifications for this operating state provide adequate protection. However, certain MCPR Operating Limits for other operating states prior to 5,000 MWd/MTU and all operating states after 5,000 MWd/MTU for WNP-2 Cycle 14 must be increased to accommodate the defect. The utility currently projects that WNP-2 will reach 5,000 MWd/MTU in early 1999. ABB-CE has notified the Washington Public Power Supply System of the necessary changes to the WNP-2 MCPR Operating Limits.

The BISON code model for WNP-2 has been revised to correct the error. All ABB-CE BWR fuel users have been notified of this condition.

(viii) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees:

The Washington Public Power Supply System has been notified of the defect and has been provided with revised MCPR Operating Limits for WNP-2.

ABB Combustion Engineering Nuclear Power
10 CFR 21 Report of a Defect or Failure to Comply

The following information is provided pursuant to the requirements set forth in 10 CFR 21.21(c)(4):

(i) Name and address of the individuals informing the Commission:

Ian C. Rickard, Director
Nuclear Licensing
Combustion Engineering, Inc.
2000 Day Hill Road
Windsor, CT 06095-0500

(ii) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect:

The activity for which this report is being filed is the establishment of non-conservative MCPR Operating Limits for the Washington Public Power Supply System Nuclear Project Unit 2 (WNP-2) nuclear power plant during Cycles 12, 13, and 14 operation.

(iii) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect:

Combustion Engineering, Inc.
2000 Day Hill Road
Windsor, CT 06095-0500

(iv) Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply:

The defect identified involves the incorrect modeling of the reactor vessel lower plenum volume in the BISON code. The defect caused the BISON code model to have a lower plenum volume approximately twice its proper size. This incorrect modeling has the potential to lead to calculation of non-conservative MCPR Operating Limits in some situations.

(v) The date on which the information of such defect or failure to comply was obtained:

ABB-CE determined that a defect in the BISON code existed on January 14, 1999.

General Information or Other (PAR) Event # 35271

Rep Org:	ABB COMBUSTION ENGINEERING	Notification Date / Time:	01/15/1999 14:15 (EST)
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State:	CT		
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HQ Ops Officer:	DOUG WEAVER		VERN HODGE NRR
Emergency Class:	NON EMERGENCY		
10 CFR Section:			
21.21	UNSPECIFIED PARAGRAPH		

PART 21 NOTIFICATION RELATED TO MINIMUM CRITICAL POWER RATIO (MCPR)

"The purpose of this letter is to notify the Nuclear Regulatory Commission of a defect under 10 CFR 21, 'Reporting of Defects and Noncompliance.' The defect concerns the Operating Limit for the Minimum Critical Power Ratio (MCPR) in Boiling Water Reactors (BWRs) analyzed using the BISON fast transient analysis code. Specifically, the defect involves incorrect modeling of the reactor vessel lower plenum volume in the BISON code, which could lead to the establishment of non-conservative MCPR Operating Limits in plant technical specifications.

"The Enclosure summarizes the evaluation performed by ABB Combustion Engineering (ABB-CE). If you have any questions, please feel free to contact me [Ian C. Rickard] or Virgil Paggen of my staff."

This event effects WNP-2 current operating cycle (Cycle 14).

****

CATEGORY 1

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RICHARD,I.C. ABB Atom, Inc. (formerly ASEA Atom, Inc.)
RECIP.NAME RECIPIENT AFFILIATION
Records Management Branch (Document Control Desk)

SUBJECT: Part 21 rept re incorrect modeling of BWR lower plenum vol in BISON.Decfect applies only to reload fuel assemblies currently in operation at WNP-2.BISON code model for WNP-2 has been revised to correct error.

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I. C. Rickard

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IE19 1/1
1-4-1 PT21C

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SUMMARY OF JANUARY 27, 2000 MEETING WITH ENERGY NORTHWEST REGARDING THE PROPOSED SECONDARY CONTAINMENT/STANDBY GAS TREATMENT SUBMITTA L.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 15, 2000

LICENSEE: Energy Northwest

FACILITY: WNP-2

SUBJECT: SUMMARY OF MEETING WITH ENERGY NORTHWEST REGARDING THE PROPOSED SECONDARY CONTAINMENT/STANDBY GAS TREATMENT SUBMITTAL

On January 27, 2000, the Nuclear Regulatory Commission (NRC) staff met with representatives of Energy Northwest to discuss the upcoming secondary containment/standby gas treatment license amendment submittal. The proposed changes would revise the following:

SR 3.6.1.3.10 increases the secondary containment bypass leakage from .74 standard cubic feet per hour (scfh) to .028 percent/day approximately 9.4 scfh.

SR 3.6.4.1.1 change the requirement to verify secondary containment pressure is less than .25 inch vacuum water gage to less than 0 inch vacuum water gage.

SR 3.6.4.1.4 increase the secondary containment drawdown time from 2 minutes to 20 minutes.

TS 5.5.7 increase the standby gas treatment system flow rate to 5000 cubic feet per minute (cfm).

Energy Northwest withdrew a similar amendment request in July 1999, when the staff identified a calculation error in determining containment release concentration. Due to the extensive review required of the previous submittal, both the staff and the licensee felt that a meeting to discuss the upcoming submittal would be beneficial in shortening the review and avoiding unnecessary requests for additional information.

In order to improve efficiency the same NRC staff members who were involved in the final review for the first submittal were at the meeting and will review the second submittal. Enclosure 1 is a list of the meeting participants. Enclosure 2 is a copy of the slides presented by Energy Northwest.

At the outset of the meeting, Mr. John Arbuckle of Energy Northwest presented an overview of the submittal. The emphasis was on how this submittal has changed from the previous submittal. This submittal includes meteorological data and atmospheric dispersion calculations (X/Q calculations). Leta Brown, of the NRC staff, suggested that it would be useful if the X/Q calculations and an electronic version of the meteorological data were provided for staff review.

Mr. David Studley of Scientech discussed the X/Q calculations including the four release points and the control room intakes. The discussion also covered the application of ARCON 96 Code and the assumptions that were made. The description of the site configuration and of the

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control room intakes were useful to the staff in understanding the assumptions that were made. The staff expressed a concern with the vent option used in the ARCON 96 Code that was used to calculate some of the control room atmospheric dispersion factors. It was suggested that an acceptable solution would be to recalculate the vent run cases as a ground release option using ARCON 96.

Mr. Bruce Boyum of Energy Northwest discussed the accidents that were analyzed specifically, main steam line break accident, fuel handling accident, control rod drop accident and loss-of-coolant accident (LOCA). Mr. Boyum stated that the LOCA was the bounding accident. Mr. Mark Blumberg, of the NRC staff, stated that it would be useful to include the calculations for the LOCA analysis and include sufficient information on other accidents so that it can be determined that the LOCA is the bounding accident.

Mr. Studley described the Axident Code, which is the dose analysis code used in the submittal. The Axident Code models the transport of radioactivity to the environment and to the control room. The major assumptions and the reasons for them were discussed.

Mr. Boyum also discussed the release pathways including the use of the Gothic Code to justify the 40 percent mixing assumed in secondary containment. Mr. Richard Lobel, of the NRC staff, said the proposed submittal should include a description of the derivation and use of the flow equation which is the basis for Figure 4 of Attachment 2 to the licensee's October 15, 1996, submittal. In addition, the staff may request input used in reactor building pressure drawdown calculations so that the staff may perform independent calculations. A final decision has not been made.

Mr. Boyum then discussed control room air flows and unfiltered control room in-leakage. Mr. Blumberg stated that licensees have had to verify their unfiltered in-leakage assumptions. The ASTM E741, "Standard Test Methods for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution," test method is acceptable to the staff as a verification test for unfiltered in-leakage. The licensee could describe their design and propose an alternative test method that would have to be reviewed and approved by the staff.

The NRC staff felt that Energy Northwest did a good job explaining their submittal and that they were receptive to suggestions from the staff.

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. List of Meeting Participants
2. Energy Nothwest Slides

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J Cushing

Jack Cushing, Project Manager, Section 2
Project Directorate IV & Decomissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. List of Meeting Participants
2. Licensee's Slides

cc w/encls: See next page

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Mr. J. V. Parrish
Chief Executive Officer
Energy Northwest
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Richland, WA 99352-0968

ENERGY NORTHWEST

MEETING PARTICIPANTS

JANUARY 27, 2000

ENERGY NORTHWEST

Douglas Coleman
Bruce Boyum
John Bekahazi
Linda Woolsley
John Arbuckle

SCIENTECH-NUS

David Studley

NRC

Jack Cushing
Steve Dembek
Mark Blumberg
Richard Lobel
Leta Brown

Enclosure 1

ENERGY NORTHWEST

Secondary Containment/SGT Submittal Presentation

Introduction

(John Arbuckle)

Meteorological Data And X/Q Calculations

(John Arbuckle and Dave Studley)

Dose Analysis And Results

(Bruce Boyum and Dave Studley)

Summary

(Bruce Boyum)

INTRODUCTION

- Initial Problem
- Submittal History
- Analysis Problem/TS Retraction/JCO-FAO Impact
- Comparison of Key SGT Parameters
- Technical Specification Changes
- Submittal Content

INITIAL PROBLEM

- Under Certain Post-Accident Meteorological Conditions, WNP-2 could not Develop 0.25-inch Negative Differential Pressure Within 120 Seconds

- Therefore, a Revised Design Basis and Dose Analysis was Provided. JCO (FAO) Prepared and Submitted to Staff.

SUBMITTAL HISTORY

- **October 1996**

  Technical Specification Amendment Request

- **December 1997 – June 1999**

  Formally Responded to Three RAIs

ANALYSIS PROBLEM

Analysis Problem/TS Retraction/JCO-FAO Impact

<u>July 1999</u>

- Withdrew Technical Amendment Request – Discovery of a Nonconservative Error in Determining Containment Release Concentration During Resolution of Proposed RAI 4

- No Impact on JCO-FAO, Current Design Basis, Technical Specifications or Recent Analyses

COMPARISON OF KEY SGT PARAMETERS

Key Parameter	Original Design	Current Design (JCO-FAO)	Proposed Design
Drawdown Time	2 minutes	10 minutes	20 minutes
SC Leakage	2240 cfm	1475 cfm	2240 cfm
SGT Flow	4457 cfm	5385 – 5850 cfm	5000 cfm

TECHNICAL SPECIFICATIONS

- Proposed Technical Specification Changes

SR 3.6.1.3.10	Increase Secondary Containment Bypass Leakage from 0.74 scfh to 0.028%/day (9.4 scfh) Consistent [illegible]
SR 3.6.4.1.1	Change the Surveillance Requirement to Verify Every 24 Hours that the Pressure Within Secondary Containment is <0 inch (vs 0.25 inch) of Vacuum Water Gauge [illegible]
SR 3.6.4.1.4	Increase Secondary Containment Drawdown Time from 120 seconds to 20 minutes
5.5.7.2.A	Increase Standby Gas Treatment System Flow Rate from 4457 cfm to 5000 cfm

SUBMITTAL CONTENT

- Detailed History - Supersedes Previous Submittals
- Responses to RAIs Incorporated
- Design Basis Meteorology and X/Q Values [illegible] Tech. approved, impl. [illegible]
- Accidents Analyzed - LOCA in Detail, FHA, MSLB and CRDA
- GOTHIC Model and Benchmarking Efforts
- Discussion of Standby Gas Treatment System
- Evaluation of Significant Hazards
- Environmental Considerations Evaluation
- Marked-Up and Typed Technical Specifications
- Marked-Up Technical Specification Bases - For Info Only

METEOROLOGICAL DATA AND X/Q CALCULATIONS

- Description of Met Tower, Terrain, and Instrumentation
- Data Used (6yrs)
- ARCON96
  1. Description of Release Points and Control Room Intakes
  2. Application of ARCON96 at WNP-2
  3. Comparison With JCO-FAO and Power Uprate

DESCRIPTION OF MET TOWER, TERRAIN, AND INSTRUMENTATION

- Meteorological Tower Consists of a 240-ft Structure with a 5-ft Extension Mast

- The Tower is Triangular in Shape and of Open Lattice Construction to Minimize Tower Interference with Meteorological Measurements

- Wind Speed and Direction is Monitored by Separate Channels at the 33-ft and 245-ft Elevations

- A Single Channel Provides Air Temperature Difference Between 33-ft and 245-ft Elevations

DESCRIPTION OF MET TOWER, TERRAIN, AND INSTRUMENTATION

- Siting of Instrumentation with Respect to Meteorological Tower and Surrounding Vegetation is Very Good
- The Base of the Tower Maintained as Natural Vegetation
- Area Around the Tower is Open Terrain with no Natural or Man-Made Obstructions to Impact Data Being Collected

DATA USED (6YRS)

- Site-specific Meteorological (Temperature & Wind Speed) Data were used for a Six-year Span from January 1, 1984, to January 1, 1990
- A Corresponding Calculation Was Performed and a Curve was Generated which Encompasses a Minimum of 96.1 Percent of all WNP-2 Weather Conditions
- Data Checked for Reasonableness
- The Curve Excludes Approximately Four Percent as a Conservative Approximation of 95%/5%

DATA USED (6YRS)

- Winter Cases Yielded Longer Drawdown Times than Summer Cases

- Limiting Case Very Conservative - Atmospheric Temperature of 0°F, no Wind, Standby Service Water System Spray Pond Temperature 77°F, Division 2 Electrical Power (i.e., Division 1 Not in Operation), One Train of Standby Gas Treatment System in Operation, and 50% Room Cooler Efficiency

- Case Resulted in a Drawdown Time of 711 Seconds (11.85 Minutes)

DATA USED (6YRS)

Zero Wind Speed and Zero Degrees Temperature used Because Temperature Impact on Differential Pressure is Prominent Factor, due to the Higher Differential Temperature Between Inside and Outside Temperatures

7.7

6.3

X/Q CALCULATIONS

- ARCON96

    1. Description of Release Points and Control Room Intakes

    2. Application of ARCON96 at WNP-2

    3. Comparison with JCO-FAO and Power Uprate

X/Q CALCULATIONS

- **ARCON96**
- ARCON96 used to Determine X/Q values for Three Control Room Intakes and Four Release Points
- Utilized the Same Time Period/plant Specific Data as Used in the Drawdown Analysis
- CR Intakes – Local and Two Remote Intakes

X/Q CALCULATIONS

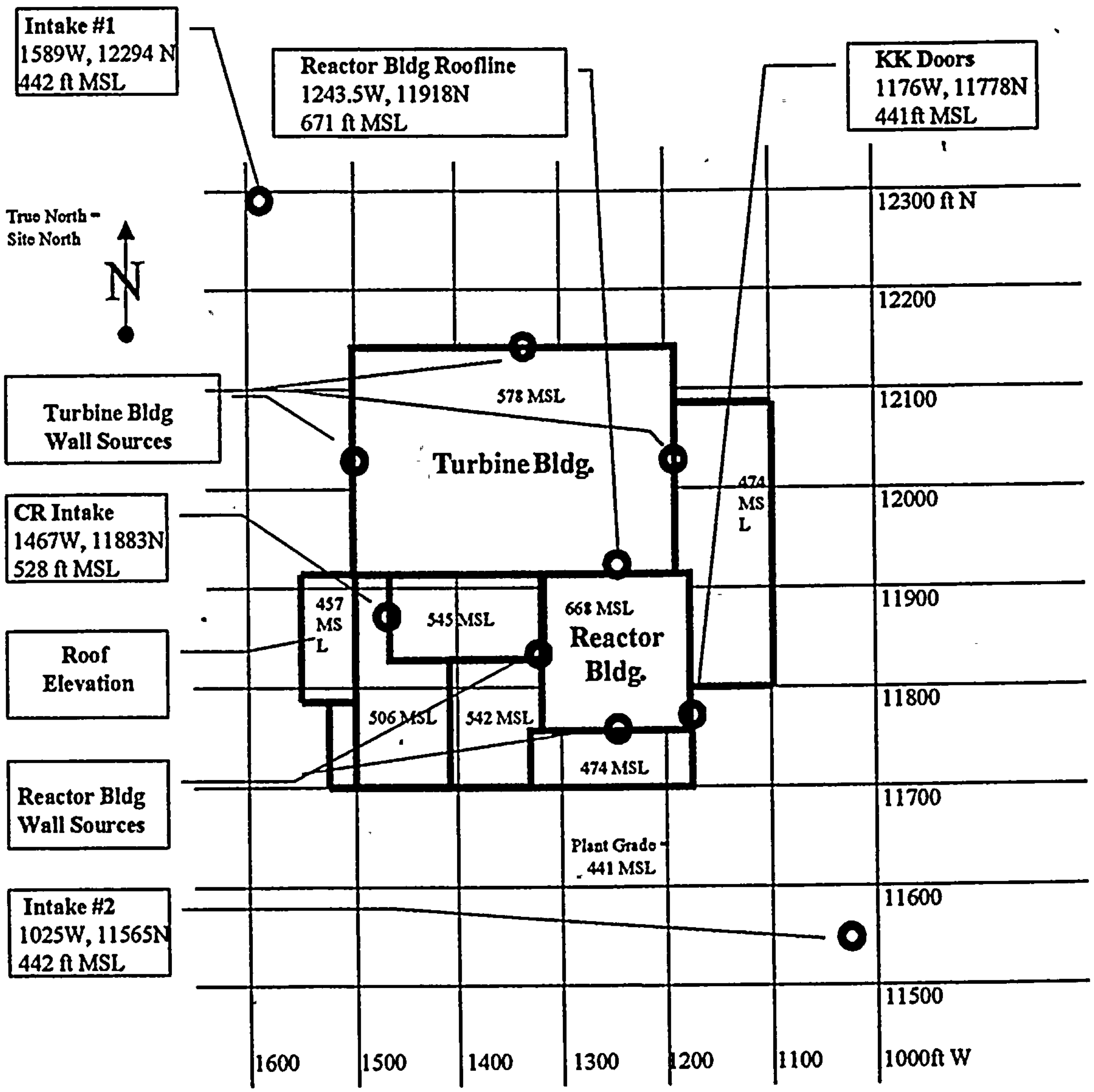
- Release Points Considered

- **Turbine Building Walls** – Release from Turbine Building Walls to be Used with Events such as CRDAs
- **Reactor Building Walls** – Release from Reactor Building Walls to be Used During Drawdown Period and Secondary Bypass Leakage
- **Reactor Building Roofline (Stack) Release** – Vent Release from Reactor Building Roof Used for SGTS Releases
- **Reactor Building (King Kong) Doors** - Release from Reactor Building Grade Area

X/Q CALCULATIONS - ARCON96 - SITE CONFIGURATION

SEE NEXT SLIDE

X/Q Calculations - ARCON96 - Site Configuration



X/Q CALCULATIONS

- Application of ARCON96
- Analyzed each of the four Release Paths for each of the Three Intakes (i.e., 12 Scenarios)
- Analyzed all 12 Scenarios for all 6 Years
- The Maximum Value of the 6 Years was Chosen for each Time Period (i.e., not just the Maximum Year but the Maximum Value of each Time Step)

  conservative worst pt in each time step [illegible]

- Local Intake not Used for LOCA due to the Presence of an Automatic Isolation Signal

X/Q CALCULATIONS

- Application of ARCON96

- Between 0 to 3 Hours, no Credit for Operator Action – Used Average of the two Remote Intakes *credit [illegible] [illegible] [illegible] there for manual action.*
- From 3 Hours to the 30 Days, Credit for Operator Action – Used the Lower of the two X/Qs Calculated for the Remote Intakes *Operator action in [illegible] period – twice. Due to operator [illegible] [illegible] performing action*
- Very Conservative Treatment – With the Presence of two Remote Intakes, the Plant will be able to Switch to the Upwind Intake and in Effect Preclude the Introduction of Activity During the Accident

X/Q CALCULATIONS

SEE NEXT SLIDE

X/Q Calculations

ARCON96 Results and Comparison with Original CR X/Qs for UFSAR

Time Period	X/Q Calculated with ARCON 96 (s/m^3)	X/Q Used for UFSAR Analysis (s/m^3)	Ratio of ARCON 96 Value to UFSAR Value
Ground Release			
0 - 2 hr	9.94E-5	2.17E-4	45.81%
2 - 3 hr	9.91E-5	5.43E-5	182.50%
3 - 8 hr	7.16E-5	4.49E-5	159.47%
8 - 24 hr	4.37E-5	3.55E-5	123.10%
1 - 4 days	2.35E-5	1.67E-5	140.72%
4 - 30 days	1.56E-5	1.67E-5	93.41%
SGTS Release			
0 - 2 hr	2.54E-4	3.77E-4	67.37%
2 - 3 hr	1.70E-4	9.43E-5	180.28%
3 - 8 hr	8.15E-5	7.80E-5	104.49%
8 - 24 hr	3.20E-5	6.17E-5	51.86%
1 - 4 days	2.26E-5	2.90E-5	77.93%
4 - 30 days	1.90E-5	2.90E-5	65.52%

DOSE ANALYSIS AND RESULTS

- Accidents Analyzed for Radiological Consequences
- AXIDENT Code
  1. How Applied
  2. Where used Before
- Summary of LOCA Major Assumptions
- Changes from Original Design and Previous Submittal
- LOCA Results

Bruce Boyum

ACCIDENTS ANALYZED

- Main Steam Line Break (MSLB)
- Fuel Handling Accident (FHA)
- Control Rod Drop Accident (CRDA)
- Loss Of Coolant Accident (LOCA)

AXIDENT CODE

Dave Stradley Scientech

- **Dose Analysis Code** – Radiological Consequences of the Spectrum of Design Basis Accidents were Analyzed using the SCIENTECH-NUS AXIDENT Code

- **Code Description** - AXIDENT Models the Transport of Radioactivity to the Environment and to the Control Room. This Code Includes the time Dependent Effects of Containment Sprays, Recirculation, Purge and Intake Filters, Atmospheric Dispersion, Natural Decay, etc. The Code is Based on the Explicit Solution of the Integrated Activity in a Receptor Volume.

AXIDENT CODE

- **Industry Experience/Usage of the AXIDENT Code –**

- Developed in the early 70's to Support the Licensing and Licensing Reviews of both U.S. and International Commercial Nuclear Power Plants. Developed to fill the Void in Codes Available to Assess the Emerging Issue at the time – Control Room Habitability.

- <u>General Industry Usage</u>
  - Used to Support Licensing Submittals in the 70's
  - Used for a Number of Plants in Support of the Post-TMI Action Item
  - Used Throughout the 80's and 90's to Resolve Control Room Habitability and Accident Analyses Issues

AXIDENT CODE

SCIENTECH Experience

- Successfully Benchmarked the Previous Results Generated by other Codes which Compute the Integrated/exact Solution (i.e., Bechtel's LOCADOSE and S&L's PostDBA)

- Over the Years, Thousands of Cases have been run on the AXIDENT Code. In all Cases, the Results have Trended as Expected. The Results have also Consistently been in Close Agreement with the Steady State Murphy/Campe Equations.

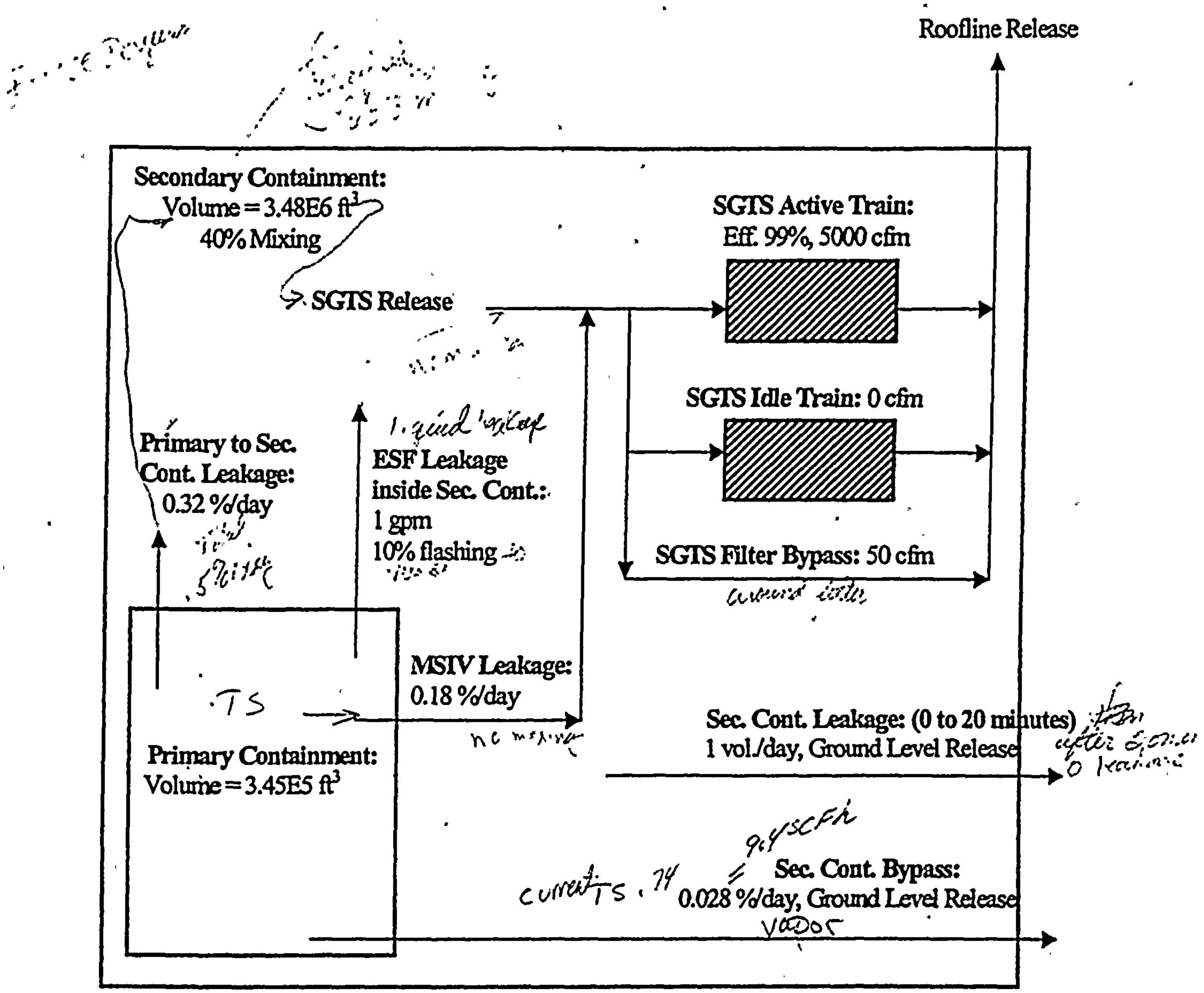
33

AXIDENT CODE

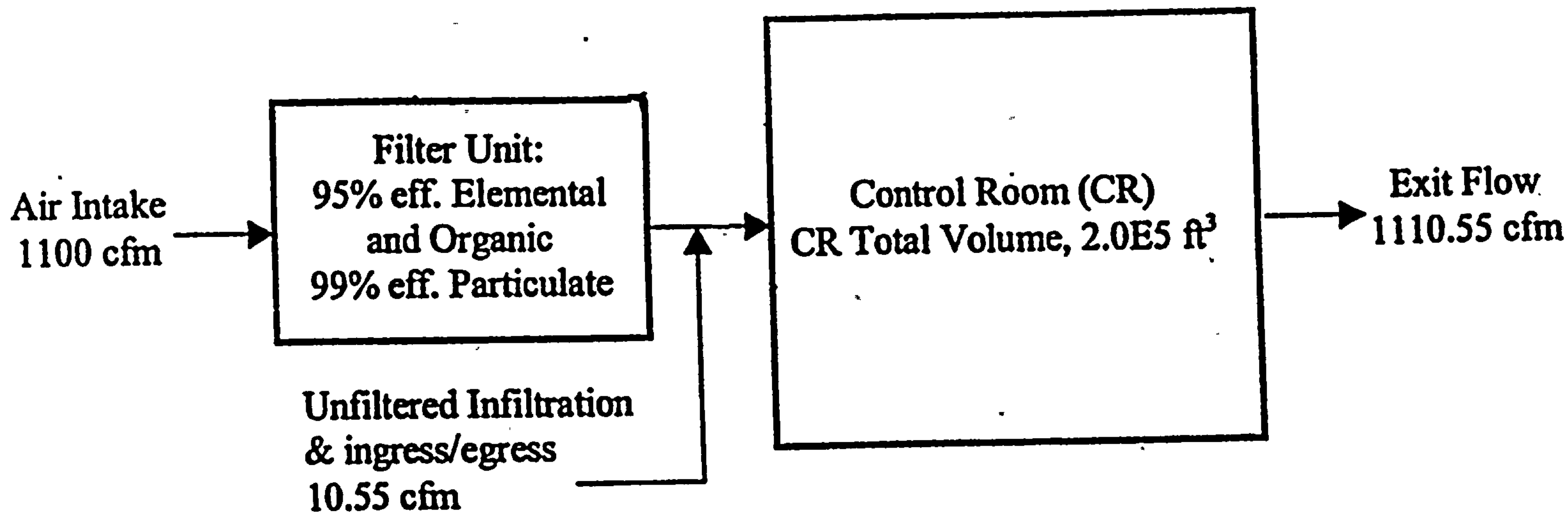
Recent Calculations Submitted to the NRC Include:

- Control Room and Offsite Dose Calculations to Support the FPC Crystal River Restart
- CRDA and MSLB Analyses for the CP&L Brunswick Station's Power Uprate Effort
- Reanalysis Effort for the Cooper Station – Pending Review

RELEASE PATHWAYS



CONTROL ROOM AIR FLOWS



MAJOR ASSUMPTIONS

Bruce

- AXIDENT Code used for Dose Analysis
- Source Term Release Per TID-14844 (102% Power Level)
  - 100 % Noble Gases
  - 25 % Halogens (50% ESF Leakage)
    - 91% Elemental
    - 5 % Particulate
    - 4 % Organic
- Dose Conversion Factors in Accordance with ICRP 30

Bruce Boyer

MAJOR ASSUMPTIONS

- Instantaneous Mixing in Primary Containment

- Release from Containment of .5%/day Total
  - .32%/day Containment Leakage
  - .18%/day MSIV Leakage

- No Suppression Pool Scrubbing Credited

MAJOR ASSUMPTIONS

- SGT Filter Efficiency
  - 99% Efficient for Halogens
  - 0 % Efficient for Noble Gases
- SGT Flow:
  - 5000 cfm Single Train
  - 50 cfm Bypass Leakage
- Control Room Filter Efficiency
  - 95% Efficient for Elemental and Organic Iodine
  - 99% Efficient for Particulate Iodine

MAJOR ASSUMPTIONS

- Secondary Containment Drawdown Time of 20 Minutes
- During Drawdown Time:
  - No SGT Filtration Credited
  - Secondary Containment Leakage at a Rate of 1 Volume/day
- 40% Mixing in Secondary Containment for Entire Scenario

MAJOR ASSUMPTIONS

- ESF Leakage Into Secondary Containment:
  - 1 gpm
  - 10% Flashing Fraction
  - 50% Core Iodine Source Term
- Control Room Unfiltered Inleakage:
  - 10 scfm Ingress/egress
  - .55 scfm Infiltration
- Secondary Containment Bypass Leakage of .028%/day (9.4 scfh)
- New X/Q Values

DOSE ANALYSIS METHODOLOGY MAJOR CHANGES

ITEM	CURRENT DESIGN	PREVIOUS SUBMITTAL	PROPOSED DESIGN
Drawdown Time	5 Min	20 Min	20 Min
Sec. Ctmt Mixing	None	40% Vol.	40% Vol.
ESF Leakage	---	None	1 GPM
SGT Filter Bypass	14 cfm	14 cfm	50 cfm
Bypass Leakage (% Day)	.00209 .74	.054* 18 [illegible]	.028 9.4

LOCA ANALYSIS RESULTS

	CALC. (REM)	LIMIT (REM)
CR Whole Body	0.4	5
CR Thyroid	28.1	30
CR Beta	6.8	30
EAB Whole Body	3.7	25
EAB Thyroid	56.6	300
LPZ Whole Body	3.4	25
LPZ Thyroid	131	300

LOCA DOSE BY PATH

RELEASE PATH	CONTROL ROOM THYROID	CONTROL ROOM W. BODY	LPZ THYROID	LPZ W.BODY
Sec. Ctmt Bypass	21	.02	101	0.5
Sec. Ctmt Leakage	.02	8E-5	0.2	4E-3
Sec. Ctmt SGT Rel.	4.2	.16	16	1.2
ESF Leakage Sec.Ctmt	0.3	1E-5	1.2	2E-3
MSIV Leakage	3.0	.18	13	1.7
TOTAL	28.1	0.4	131	3.4

CONTAINMENT RELEASE IMPACT

(.04%/day Sec Ctmt Bypass Leakage)

RELEASE PATH	CR THYROID DOSE (REM) .32%/DAY	.5%/DAY
Sec Ctmt Bypass	29	29
Sec Ctmt Leakage	.02	.03
Sec Ctmt SGT Rel.	4.2	6.5
ESF Leak (Sec Ctmt)	0.3	0.3
MSIV Leakage	3.0	3.0
TOTAL	36.9	39.2

EFFECT OF RELEASE ELEVATION

DESCRIPTION	DOSE (REM)
Ground Release = 100% Roofline Release = 0%	39.2
Ground Release = 60% Roofline Release = 40%	39.6
Ground Release = 50% Roofline Release = 50%	39.7
Ground Release = 0% Roofline Release = 100%	39.2

- Based on Thyroid Dose in Control Room Assuming .5%/day Containment Leakage and .04%/day Secondary Containment Bypass

EFFECT OF SGT FLOWRATE

DESCRIPTION	DOSE (REM) THYROID	W.BODY
4K cfm for 30 Days	37.3	.357
5K cfm for 30 Days	36.9	.369
10K cfm for 30 Days (20 Min Drawdn)	37.3	.409
10K cfm for 30 Days (10 Min Drawdn)	37.3	.409
10K cfm for 1 Hr Then 4K cfm 30 Days	37.3	.362

Based On .32%/day Ctmt Leakage and .04%/day Sec Ctmt Bypass Leakage

SUMMARY

- Dose Analysis Meets 10CFR50 and 10CFR100 Limits
- No Hardware Changes Necessary Beyond Those Completed in Support of the JCO-FAO
- Tech Spec Submittal to be Made in February
- FSAR Changes will be Implemented Following Approval of Tech Spec Submittal
- FSAR Changes Include:
    - Accident Analysis and Doses
    - Control Room Habitability Analysis
    - Secondary Containment Description
    - Description of SGTS and REA
- JCO-FAO will be Closed Following Approval of Tech Spec Submittal

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

NOV 15 1999

Mr. J. V. Parrish (Mail Drop 1023)
Chief Executive Officer
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: INVITATION TO DISCUSSION OF PILOT PROGRAM RESULTS

Dear Mr. Parrish:

On May 30, 1999, we initiated a pilot of the risk-informed baseline inspection program at Cooper Nuclear Station and Fort Calhoun Station. The pilot program is scheduled to end November 27. On December 1, 1999, the NRC will meet with Omaha Public Power District and Nebraska Public Power District to discuss their perceptions and lessons learned gathered during the pilot program. You will find the details of the meeting documented in the enclosed meeting notice.

Although the meeting is between the NRC and members of the pilot plant staffs, we will be discussing topics that may be of interest to your organization. This meeting will be open to public observation. Following the meeting, the discussion will be opened for comments and questions from observers. We would entertain and welcome your participation at that time. Should you have any questions on this inspection plan, please contact Charles Marschall at (817) 860-8185 or David Loveless at (817) 860-8161.

Sincerely,

Charles S. Marschall

Charles S. Marschall, Chief
Project Branch C
Division of Reactor Projects

Docket No.: 50-397
License No.: NPF-21

Enclosure:
As Stated

PDR ADOCK 99328 0249

IE45

IA3

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NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

November 15, 1999

NOTICE OF LICENSEE MEETING

Name of Licensee:	Nebraska Public Power District Omaha Public Power District
Name of Facility:	Cooper Nuclear Station Fort Calhoun Station
Docket:	50-285 50-298
Date and Time of Meeting:	December 1, 1999 1 - 3 p.m. (CDT)
Location of Meeting:	Fort Calhoun Station Auditorium Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Purpose of Meeting:	Discussion of the licensee's perceptions and lessons learned from the pilot inspection program
NRC Attendees:	K. Brockman, Director, Division of Reactor Projects C. Marschall, Chief, Branch C, Division of Reactor Projects A. Madison, Transition Task Force Leader, Office of Nuclear Reactor Regulation W. Jones, Senior Reactor Analyst, Division of Reactor Safety L. Wharton, FCS Project Manager, Office of Nuclear Reactor Regulation L. Burkhart, CNS Project Manager, Office of Nuclear Reactor Regulation D. Loveless, Senior Project Engineer, Branch C, Division of Reactor Projects
Licensee Attendees:	W. Gary Gates, Vice President, Fort Calhoun Station S. Gambhir, Division Manager, Nuclear Operations J. Chase, Division Manager, Nuclear Assessments M. Tesar, Division, Manager, Nuclear Support Services R. Phelps, Division Manager, Nuclear Engineering J. Solymossy, Manager, Fort Calhoun Station M. Frans, Manager, Nuclear Licensing R. Jaworski, Manager, Revised Reactor Oversight Project B. Hansher, Supervisor, Station Licensing

993280267

H. Hackerott, Supervisor, Systems Analysis
J. McDonald, Plant Manager, Cooper Nuclear Station
P. Caudill, General Manager, Technical Services
G. Smith, Nuclear Projects Manager
J. Sumpter, Licensing Supervisor
D. Robinson, QA Assessment Manager
R. Wachowiak, Risk Management Supervisor

NOTE:
(1) This meeting is open to attendance by members of the general public.

(2) NRC personnel, not listed above, that desire to attend this meeting should notify C. S. Marschall at 817/860-8185 by COB on November 19, 1999.

(3) A comment session will be conducted at the end of the meeting for participation of all Region IV operating reactor licensees.

Approved By: Charles S. Marschall

Charles S . Marschall, Chief
Project Branch C
Division of Reactor Projects

cc:
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FJM	A/D/NRR
BWS	Acting, ADT/NRR
BAB2	Acting ADP/NRR
LJB	L. Burkhart, Project Manager, NRR
LRW	R. Wharton, Project Manager, NRR
OEMAIL	D/OE
WCW	W. Walker, Senior Resident Inspector
JAC	J. Clark, Senior Resident Inspector
CAH	C. Hackney, RSLO
GFS	G. Sanborn, EO
BWH	B. Henderson, PAO
JAC1, CJG	RA Secretaries
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