Docket File

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

February 8, 1993

Docket No. 52-002

APPLICANT: ABB-Combustion Engineering, Inc. (ABB-CE)

PROJECT: CE System 80+

SUBJECT: PUBLIC MEETING OF JANUARY 11, 1993, TO DISCUSS THE REVIEW STATUS OF THE CE SYSTEM 80+ DESIGN WITH SENIOR MANAGEMENT

On January 11, 1993, a public meeting was held at the ABB-CE facilities in Windsor, Connecticut, between senior management representatives of ABB-CE and the U.S. Nuclear Regulatory Commission (NRC). Enclosure 1 provides a list of attendees. Enclosure 2 is the material presented by ABB-CE.

ABB-CE opened the meeting with a status of the project and an overview of the System 80+ design process. ABB-CE also provided a progress report on ABB-CE responses to items from the CE System 80+ draft safety evaluation report (DSER) and an update of progress achieved for closure of issues from shutdown risk and severe accidents areas.

ABB-CE expressed concern over severe accident closure for the issue of core-concrete interaction (CCI) and containment performance goals under this condition. ABB-CE stated that they were attempting to quantify the appropriate heat transfer coefficient. ABB-CE was concerned that assumptions of very low heat transfer coefficients could result in the System 80+ containment not meeting the 24 hour containment performance goal. System 80+ design modifications were under consideration as a contingency. The staff reiterated that severe accidents are not a design-basis accident requirement for evolutionary advanced light water reactors, and the 24-hour criterion for containment integrity is a performance goal. ABB-CE should analyze what the CE System 80+ containment and reactor cavity can accommodate with respect to CCI.

During the presentation on the System 80+ probabilistic risk assessment (PRA), the staff commented that the results of the system importance studies must be analyzed and provided as input into the design reliability assurance program (D-RAP) and the operational reliability assurance program (O-RAP). Results from the importance studies would also be appropriate for roadmapping purposes.

The morning session concluded with a demonstration of the PASCE computer aided design (CAD) system. ABB-CE also demonstrated PASCE application for System 80+ design work and potential use for future plant owners/operators.

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The afternoon session consisted of discussions on several technical issues. ABB-CE made presentations on issues concerning reactor coolant pump (RCP) seal integrity and instrumentation and controls (I&C) diversity with the associated common mode failure analysis (CMF) (see Enclosure 2). The afternoon session concluded with a tour and demonstration of the Nuplex 80+ control room complex.

The following commitments were made during the meeting:

(1) ABB-CE committed to provide an agenda for the CE System 80+ inspections, tests, analyses, and acceptance criteria (ITAAC) industry review for the first week at the ABB-CE facilities in Windsor, Connecticut and the second week at Duke Engineering & Services, Inc. (DESI) in Charlotte, North Carolina.

(2) ABB-CE committed to address the potential for cracking in the reactor vessel upper head including the CRDs. The staff noted this due to operating experience review and the System 80+ vessel outlet temperature of 615 °F and the potential for high residual stress in the vessel material due to the fabrication process. ABB-CE should also address use of lower stress fabrication processes, alternate materials, temperature adjustments, and leakage detection equipment.

(3) ABB-CE committed to evaluate that additional pathways for interfacing system loss-of-coolant-accident (ISLOCA) (beyond the shutdown cooling system scope) are designed to the ultimate rupture strength criteria. Pathways such as the direct vessel injection lines, the emergency core cooling system (ECCS) lines outside containments, containment spray pumps, valves, etc. should be addressed.

(4) The staff will evaluate if the use of relief valves on the suction of the shutdown cooling system (SCS) is sufficient to meet Branch Technical Position 5.1 on low temperature overpressure protection (LTOP).

(5) For operating experience review, NRC and ABB-CE management both need to determine what process was used to give confidence that operating experience has been comprehensively addressed. ABB-CE will provide a copy of the operations and maintenance study report for System 80+.

(6) ABB-CE should address both TID [technical information document] source term and the new source term in the power upgrade analysis.

(7) ABB-CE committed to provide an insights document that provides:

(a) interfaces with CESSAR-DC Chapter 18 and the associated task analysis review for the Nuplex 80+ control complex (i.e., emergency procedure guidelines (EPGs) PRA, shutdown risk, and severe accident input to the control room).

(b) roadmap input document for ITAAC, D-RAP, O-RAP, EPGs, etc.

(8) For the NRC structural audit, ABB-CE committed to define loads and load combinations, identify limiting regions and members, and prioritize critical plant areas for CESSAR-DC. This action should be completed in the February/March time frame. ABB-CE committed to evaluate improvement of the schedule for identification and prioritization of critical areas in the System 80+ structures prior to the current schedule for identification of April 30, 1993. This action should identify and prioritize critical areas for which detailed analysis and audit are needed for CESSAR-DC and the final safety evaluation report (FSER).

(9) For Generic Issue 23, "Reactor Coolant Pump Seal Failures," the NRC committed to evaluate why the staff required a diverse-safety grade seal injection system in lieu of a diverse but reliable seal cooling system. The staff will determine what will be needed for diversity.

(10) For CMF analysis of the I&C systems, the NRC and ABB-CE management committed that their respective staff's conduct an integrated meeting with representatives from reactor systems, PRA, and the I&C disciplines. Both parties will need to identify potential hardspots for this issue in conjunction with the following CESSAR-DC accidents:

(a) reactor trip on main steamline break or feedwater line break accident.

(b) automatic safety injection (SI) on large break LOCA.

(11) For common mode position 4, ABB-CE proposed use of LC1 level output. LC1 is a digital switch which has a binary output (on/off) for a specific set of inputs from either group control or manual. The staff will evaluate that the use of a digital switch versus use of a relay is appropriate for this issue.

(12) ABB-CE committed to evaluate if there is an adequate set of diverse displays. ABB-CE stated during the meeting that for the diversity issue, the main control room will provide diverse display of 15 parameters, and the set was based on Regulatory Guide 1.97. ABB-CE should evaluate if this is the sufficient set of parameters with input from the CMF/diversity analysis.

(13) ABB-CE should be prepared to provide their definition of "to the extent practicable" for the ISLOCA issue during the January 21, 1993, meeting.

(14) The staff will provide ABB-CE a copy of the outline for development of the PRA portion of the advanced boiling water reactor FSER.

February 8, 1993

(15) The staff will contact ABB-CE for a discussion on leak-before-break methodology and application on the CE System 80+ design.

(16) ABB-CE will provide slides for an overview of the System 80+ program.

Sincerely,

(Original signed by T.V. Wambach for)

Michael X. Franovich, Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/o enclosures:
See next page

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cc w/o enclosure:

Mr. C. B. Brinkman, Acting Director
Nuclear Systems Licensing
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

Mr. C. B. Brinkman, Manager
Washington Nuclear Operations
Combustion Engineering, Inc.
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Stan Ritterbusch
Nuclear Systems Licensing
Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500

Mr. Daniel F. Giessing
U. S. Department of Energy
NE-42
Washington, D.C. 20585

Mr. Steve Goldberg
Budget Examiner
725 17th Street, N.W.
Washington, D.C. 20503

Mr. Raymond Ng
1776 Eye Street, N.W.
Suite 300
Washington, D.C. 20006

MEETING ATTENDEES

January 11, 1993

NAME	ORGANIZATION
M. Franovich	NRR/PDST
T. Wambach	NRR/PDST
W. Borchardt	NRR/PDST
D. Crutchfield	NRR/ADAR
W. Russell	NRR/ADT
T. Murley	NRR/DO
B. D. Liaw	NRR/DE
A. Thadani	NRR
R. Newman	ABB-CE
R. Matzie	ABB-CE
C. Brinkman	ABB-CE
S. Ritterbusch	ABB-CE
J. Longo	ABB-CE
F. Carpentino	ABB-CE
G. Davis	ABB-CE
L. Gerdes	ABB-CE
D. Finnicum	ABB-CE
R. Turk	ABB-CE
K. Scarola*	ABB-CE
A. Hyde*	ABB-CE
M. Cross*	ABB-CE
Mr. Windsor*	ABB-CE
D. Harmon*	ABB-CE
R. Schneider*	ABB-CE
P. Hansen	ABB-CE
P. Lang	U.S. DOE
T. Crom	DE & S
G. Hedrick	DE & S
S. Stamm	Stone & Webster

* Afternoon session only

Enclosure 1

System 80+™ Design Certification

NRC/ABB-CE Management Meeting

January 11, 1993

Windsor, CT

ABB
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Enclosure 2

Proposed Agenda NRC Senior Management Meeting January 11, 1993 – Windsor, CT

8:30	Welcome	R. Newman
8:40	Agenda Review	C. Brinkman
8:45	Introductory Remarks	T. Murley R. Matzie
9:00	Design Summary	R. Turk
9:45	Chapter 15 Analysis Technical Approach	F. Carpentino
10:00	Process for Road-Mapping PRA Insights	D. Finnicum
10:30	Break	
10:45	Submittal Schedule	S. Ritterbusch
11:15	Documentation of Issue Closure	S. Ritterbusch
11:30	Plant Information System Demonstration - Building 12	T. Crom

Proposed Agenda (cont'd)

12:15	Lunch in Conference Room	
1:00	Technical Issues Discussions	
	Structural Design Details	L. Gerdes
	RCP Seal Coolability	M. Cross
	I&C Diversity	A. Hyde
1:45	Status of Other Technical Issues	S. Ritterbusch
2:45	Depart for Nuplex 80+ Demonstration	
3:00	Nuplex 80+ Demonstration	D. Harmon
4:45	Wrap-up	
5:00	Adjourn	

Chapter 15 Analysis Approach

- Overview:
  - Resolved Open Items Regarding Methodology
  - Incorporate Additional Changes to Improve Margin and/or to Resolve Other Issues (e.g., LBB)
  - Adjust for 3% Increased Core Power

Safety Analyses Changes

- The Following Changes are Being Incorporated per NRC Staff Agreement:
  - Delete 3 Seconds Time Delay for Loss of Offsite Power
  - Retain DNBR Convolution for any Event Which Fails the DNBR SAFDL
  - Calculate Doses Using TiD-14844 and NUREG-1465 Source Terms

Safety Analyses Changes (cont'd)

- The Following Additional Design Input Changes Will be Included in the Safety Analysis as Well:
  - Erbium Burnable Poison
  - DVI Line Size Increase
  - PZR Surge Line L/D Increase
  - Decrease Maximum Charging Flow
  - Letdown Line K-Factor Increase
  - X/Q Increases (EPRI URD)

Reanalysis Scope

- The Following Scope was Accepted by the Staff. It Excludes Repeat of Previous Sensitivity Studies Since They Remain Valid
  - Containment P&T (MSLB only) (6.2)
  - Large LOCA ECCS (Worst Break) (6.3)
  - Small LOCA ECCS (Worst Break) (6.3)
  - Excess Load (15.1)
  - Main Steam Line Break (15.1)
  - Loss of Condenser Vacuum (15.2)
  - Feedwater Line Break (15.2)
  - Locked Rotor (15.3)
  - Loss of Flow (15.3)

Reanalysis Scope (Cont'd)

- CEA Drop (15.4)
- CEA Withdrawal (15.4)
- CEA Ejection (15.4)
- Pressurizer Level Malfunction (15.5)
- Letdown Line Break (15.6)
- Steam Generator Tube Rupture (15.6)
- LOCA Offsite Doses (15.6)
- Fuel Handling Accident (15.7)

• In Addition to These Reanalyses, a RSB 5-1 Natural Circulation Cooldown Analysis is Being Done for Chapter 5

Process for PRA Road Mapping

- PRA is Integral Part of Design Process
  - PRA Insights from Previous PRAs Factored into Initial Design
  - Baseline System 80 PRA
    - Reviewed by Engineering
  - Baseline PRA Modified During Design Process
    - PRA Staff Attend Weekly Design Meetings
    - PRA Review of Design Changes

Process for PRA Road Mapping (Cont'd)

- PRA Comparative Analysis for Selected Issues
  - 4 DG vs 2 DG and CT
  - Bus Arrangement
  - CCWS/SSWS System Design
  - Cavity Flood System
- PRA Reflects Design Evolutions
- Revision 0, System 80+ PRA and Comparison Assumptions Document
  - Reviewed by Engineering
  - Submitted to NRC for Review

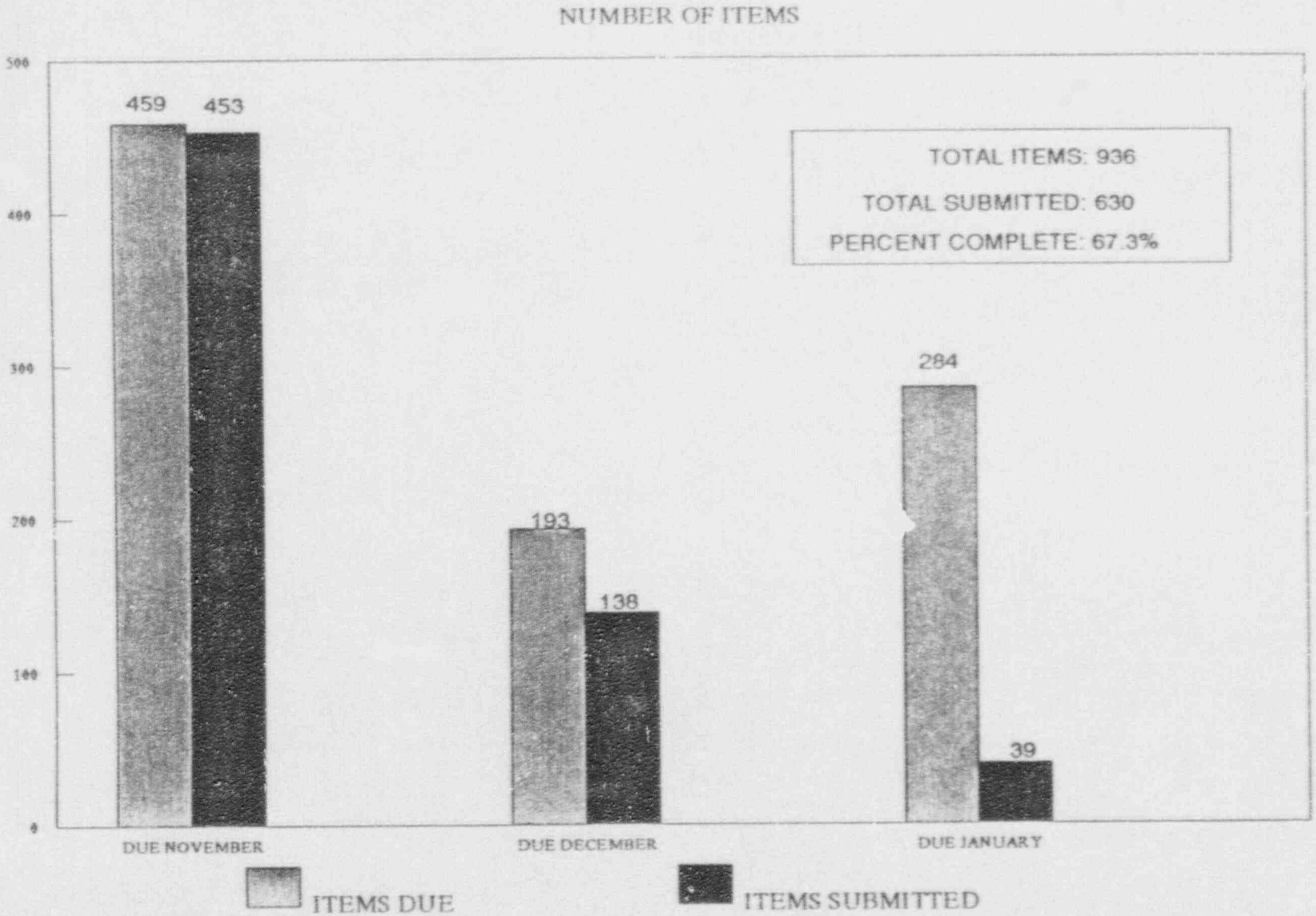
Process for PRA Road Mapping (Cont'd)

- Revision 1, System 80+ PRA In Progress
  - Address NRC Review Comments
  - Reflect Latest Design Charges
  - Review by Engineering
- All PRA Assumptions and Results Fully Documented in PRA Report
- Stand-alone Assumptions Document for Revision 0
- D-RAP Program Plan Prepared and Submitted to NRC
  - Covers Reliability Assurance During Detailed Design Phase

Process for PRA Road Mapping (Cont'd)

- PRA Assumptions and Insights Document Being Prepared for Final PRA
  - Replace Rev. 0 Assumption Document
  - By System
    - Design Assumptions
    - Operator Actions
    - Operation Assumptions
    - Risk Significant Components
  - Submit to Engineering for Review
    - Input to ITAAC
    - Cross Reference Items to ITAAC/CESSAR-DC

ABBIMS – DSER PROGRESS SUMMARY



DSER PROGRESS BY CHAPTER

CHAPTER	TOTAL	NOVEMBER 15TH.		DECEMBER 15TH.		JANUARY 15TH.		TOTAL SUBMITTED	REMAINING	PERCENT COMPLETE
		DUE	SUBMITTED	DUE	SUBMITTED	DUE	SUBMITTED			
1	2	1	1	0	0	1	0	1	1	50.0%
2	50	41	41	6	6	3	3	50	0	100.0%
3	145	61	61	47	30	37	10	101	44	69.7%
4	22	5	5	17	17	0	0	22	0	100.0%
5	78	39	39	23	16	16	4	59	19	75.6%
6	64	32	32	7	7	25	5	44	20	68.8%
7	29	9	9	4	0	16	0	9	20	31.0%
8	45	6	6	0	0	39	0	6	39	13.3%
9	166	112	111	29	27	25	5	143	23	86.1%
10	47	29	28	3	0	14	1	29	18	61.7%
11	25	10	10	0	0	15	2	12	13	48.0%
12	24	12	11	0	0	12	1	12	12	50.0%

DSER PROGRESS BY CHAPTER

CHAPTER	TOTAL	NOVEMBER 15TH.		DECEMBER 15TH.		JANUARY 15TH.		TOTAL SUBMITTED	REMAINING	PERCENT COMPLETE
		DUE	SUBMITTED	DUE	SUBMITTED	DUE	SUBMITTED			
13	12	0	0	4	3	8	0	3	9	25.0%
14	35	30	30	2	0	3	0	30	5	85.7%
15	22	11	10	10	10	1	0	20	2	90.9%
16	2	2	2	0	0	1	0	2	0	100.0%
17	23	6	6	6	6 (+2 repeats)	11	6	18	5	78.3%
18	17	5	4	2	0	10	0	4	13	23.5%
19	72	27	26	5	2	40	2	30	42	41.7%
20	56	21	21	28	14	7	0	35	21	62.5%
TOTALS	936	459	453	193	138	284	39	630	306	67.3%

NO. ITEMS	936	
NO. SUBM.	630	
TOTAL	630	[illegible]31% COMPLETE

SUBMITTALS BY BRANCH OF NRC

BRANCH	TOTAL	SUBM.	TO GO	PERCENT
ECGB	242	179	63	74.0%
EMCB	63	47	16	74.6%
EMEB	20	14	6	70.0%
EELB	48	10	38	20.8%
HICB	32	11	21	34.4%
HHFB	31	13	18	41.9%
OTSB	3	3	0	100.0%
PDST	3	3	0	100.0%
PEPB	5	1	4	20.0%
PRPB	31	19	12	61.3%
RPEB	55	45	10	81.8%
PSGB	29	21	8	72.4%
SPLB	252	195	57	77.4%
SPSB	75	31	44	41.3%
SRXB	47	38	9	80.9%
TOTALS	936	630	306	67.3%

Submittal Status and Schedule

• November 15	DSER Responses (48%)
• December 15	DSER Responses (63%)
• January 15	New Source Term Report
• January 21	DSER Responses (100%)
• January 28	ITAAC (~60%)
• February 15	Severe Accident RAI Responses
• February 15	Road Map - Draft
• February 28	Safety Analysis Results
• February 28	ITAAC (100%)

Submittal Status and Schedule (Cont'd)

• March 15	Severe Accident RAI Revisions
• March 15	Road Map - Complete
• March 31	Safety Analysis SAR Writeups
• March 31	ITAAC Industry Review Revisions
• April 1	Human Factors Engineering Submittal
• April 15	CESSAR-DC Amendment

Documentation of Issue Closure

- Issue:
  - Closure of Review After Submittal of Material
- Proposed Approach:
  - Document Review of Major Submittals via Internal Status Reports
  - Maintain Explicit Listings of Confirmatory and Open Item Status
  - Compile Status Reports and Confirmatory Item List and Release in a Mid-Summer Progress Report

Structural Design Detail

- Issue:
  - Present Sufficient Design Detail to Provide Staff Confidence in Adequacy of Structural Design in Their Safety Evaluations
- Objectives:
  - Identify Critical Areas for Design
  - Provide Detailed Design for Critical Areas
- Status:
  - Meeting Held with Staff to Reach Agreement on Requirements, Objectives, General Approach/Methodology, Preliminary List of Critical Areas and Schedule
  - Development of Governing Loads/Load Combinations and Preliminary Detailed Design of Selected Critical Design Areas are Proceeding in Parallel to Minimize Schedule

Design Methodology

- Design Details Will be Developed for the Following Critical Design Areas:
  - Containment Vessel
  - Nuclear Island Basemat
  - Shear Walls Including Connections to the Basemat
  - Side Walls to Include Dynamic Soil Pressures
  - Floor Slab Connections to Shield Building
  - Freestanding Portion of the Shield Building, Particularly in the Dome Region
  - Containment Vessel Evaluation and Reactor Cavity Walls, Including Static and Dynamic Pressure Capacity, for Severe Accident Loading Conditions
  - Steel Containment Vessel Embedment
  - Steam Generator Cavity
  - Non-Nuclear Island Safety Related Structures

Design Methodology

- Determine Critical Structural Members from Analysis Results
- Determine Critical Local Loadings That May Govern Design of Members (Wind, Tornado, Missile Impact, Equipment Loads, Hydrodynamic Loads, etc.)
- Develop Preliminary Reinforcing Schemes for Walls, Floors, Basemat and Connections for the Existing Member Sizes
- Apply Enveloped Global and Any Local Loads to Members From Analysis Results and Determine Maximum Shears, Forces and Moments in Members
- Check Adequacy of Members and Connections per Codes and Standards Identified in CESSAR-DC Table 3.8-4
- Redesign Members That do not Meet Code and Standard Requirements

Analysis Methodology

- Develop a Detailed 3-D Finite Element Model of the Nuclear Island and the Annex Structures for Global Load Analyses
  - Use Shell Elements to Model the Shear Walls and the Floor Diaphragms
  - Use Solid Elements to Model the Basemat
  - Use Spring Elements to Model the Soil
- The Containment Analysis Will be Based on a Separate Dynamic Response Spectrum Analysis of the SCV Alone
- The Finite Element Model Will be Used for Global Load Distribution of all Other Structures
- The Behavior of the Composite Static 3-D Model Will be Cross-Checked Versus That of the Stick Model Used in the Dynamic SSI Analysis
- Local Loads Will be Developed for Critical Design Areas

Schedule

1/30/93	Complete the 3-D Model of the Nuclear Island Structures
1/30/93	Identify Global Loads and Combinations
2/15/92	Reactor Cavity Capacity and Design for Severe Accidents
2/26/93	Complete Loading Analyses in Support of Wall/Floor Design
3/ /93	Meeting With NRC Staff to Discuss and Resolve Comments on Previously Submitted Material
3/15/93	Preliminary Details for Walls, Floors, Basemat and Connections for Critical Areas
3/31/93	Complete Loading Analysis in Support of Basemat Design

Schedule (cont'd)

4/30/93 Confirm Critical Areas

5/ /93 Meeting With NRC Staff

- Discussions on and Resolution of Questions on Previously Submitted Material
- Concurrence on Final Definition of Areas for Which Critical Design Details are Provided

6/30/93 Check Adequacy of Design for Critical Areas

6/30/93 Complete Criteria and Required Design Detail for Non-Nuclear Island Safety Related Structures

7/ /93 NRC Audit of Structural Design

Design Summary

Design Process

Plant Overview

Design Philosophy

- Maintain the advantages of ABB's well proven NSSS: System 80
  - An "Evolutionary" ALWR
- Assess design against ALWR goals
- Incorporate enhancements
  - EPRI Utility Requirements Document
  - Regulatory requirements
  - Start-up & operational experience
- Ensure integration of design
- Standardize design

Design Approach

- Start with System 80 NSSS & Duke Power's BOP
- Consider Changes Based on:
  - EPRI ALWR Requirements
  - NRC Mandated Changes
  - Operational Experience
- Assess Impact of Changes on:
  - Safety
  - Performance, Operations & Maintenance
  - Cost
- Incorporate Changes and Certify with NRC

EPRI ALWR Requirements Conformance

- System 80+ is firmly based on the EPRI ALWR Utility Requirements Document
- Currently only 23 nonconformances identified
- Working towards resolution of nonconformances by:
  - Resolving NRC open items
  - Revising System 80+ design
  - Proposing revisions to URD

ABB

System 80+ Design Objectives

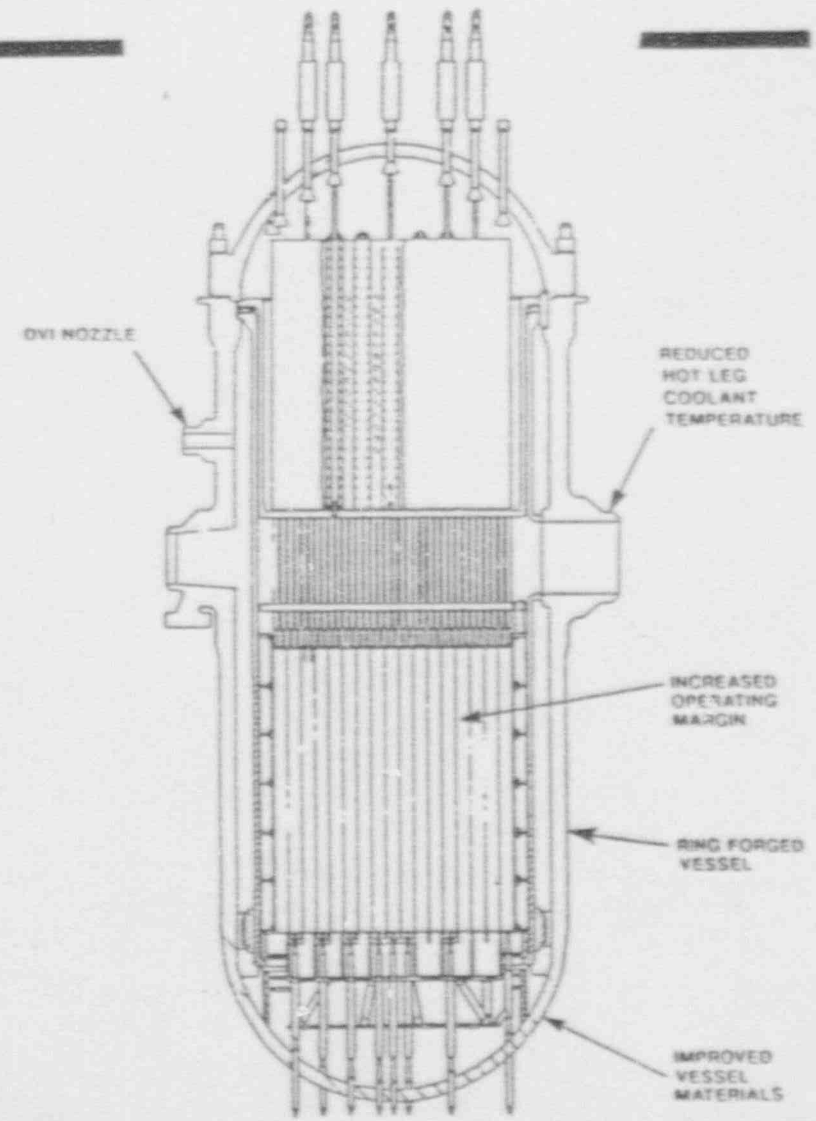
Area	Design Objectives	Major Changes from System 80
Reactor	• Maintain Proven Design • Meet Utility Performance Needs	• Very Few Changes • Part-strength Rods for load follow • Increased Core Margin
Reactor Coolant	• Improve Plant Margins	• Lower Operating Temperatures • Increased System Volumes • Improved Materials
Safeguards Systems	• Reduce Core Melt Frequency	• Increased Redundancy • Added Safety Depressurization System • Redesign in Very Close Conformance with EPRI ALWR Requirements

System 80+ Design Objectives

Area	Design Objectives	Major Changes from System 80
Auxiliary Systems	• Simplify Design	• Non-safety CVCS
Containment and Nuclear Annex	• Address Severe Accidents • Meet Utility Maintenance Needs	• Use Dual, Spherical Steel Design • Large Maintenance Access Areas • Specific Radiation Protection Features
Instrumentation and Control	• Provide State of the Art Human Factors Engineered Control Complex	• Nuplex 80+ Advanced Control Complex
Electric Distribution and Support Systems	• Improve Reliability Consistent with Safeguards Systems	• Greater Redundance and Diversity

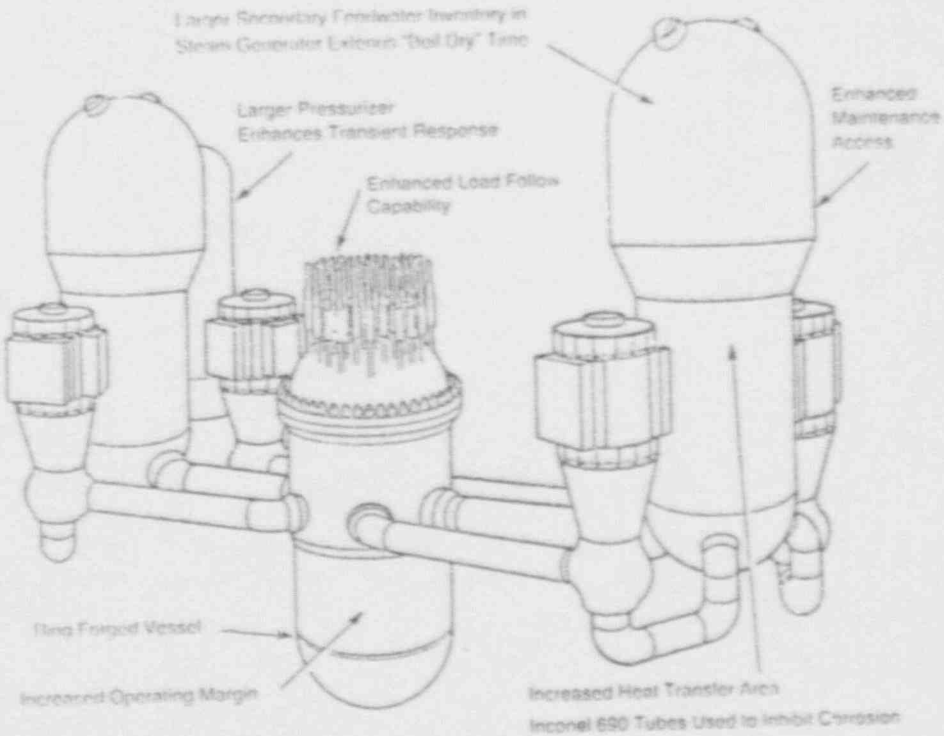
Reactor

- Objective: Maintain Proven Design
- Essentially No Changes
  - Part`Strength CEA's Provide for Manuvering without Change in Boron Concentration



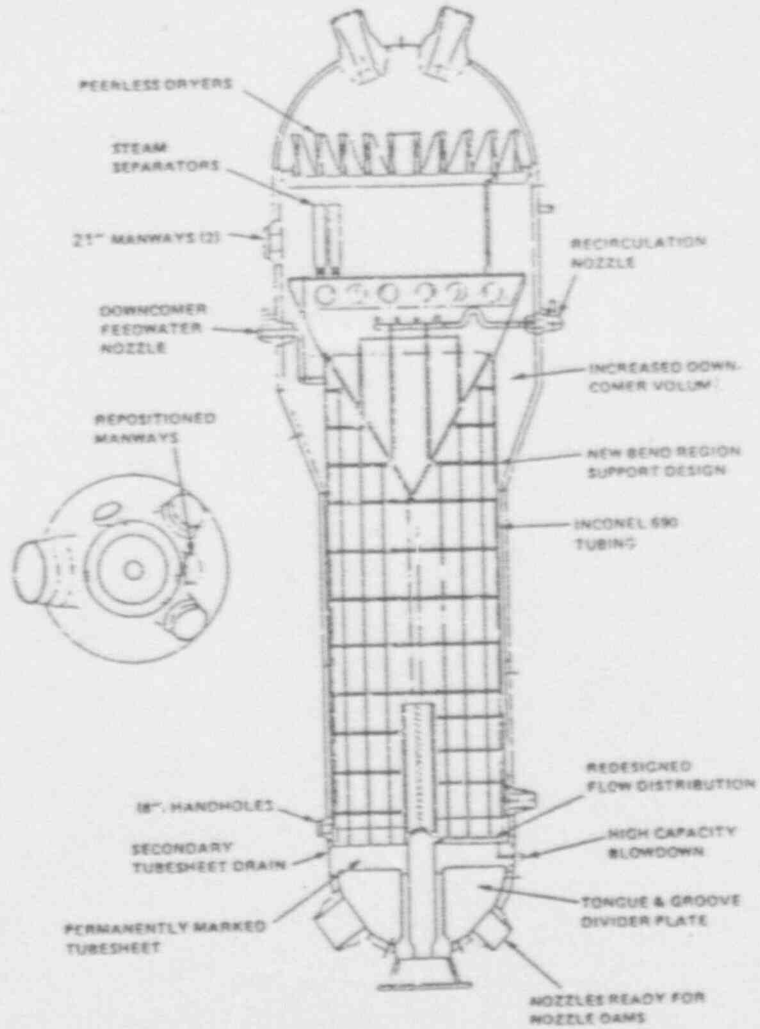
Reactor Coolant System

- Objective: Increase Margins
  - Lower Operating Temperature
  - Increased System Volumes
  - Ring Forged Vessels
  - Improved Steam Generator Design



Improved Steam Generator Design

- Increased Downcomer Volume
- Inconel 690 Tubes
- 10% Tube Plugging Margin
- Improved Maintenance Access
- Improved Dryers



Safety Systems

- **Objective: Meet PRA Goals**
  - Added Safety Depressurization System
  - Improved Safety Injection performance
  - 4 Train Emergency Feedwater
  - Alternate AC Power Source (Gas Turbine)

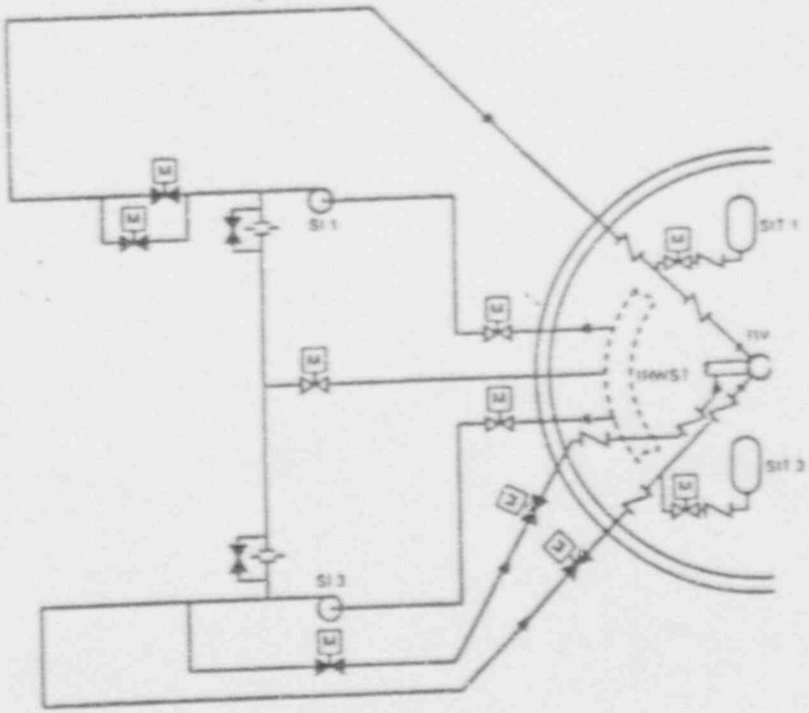
Safety System Approach

- Two division redundancy & separation at system level
- Four train redundancy & separation at component level
  - Two Emergency Diesel Generators
- "non-safety grade" diversity for all safety functions
  - e.g. , Gas turbine generator

Safety Injection System

- 4 Train redundancy for small break
- Direct Vessel Injection
- IRWST - no automatic recirculation actuation
- No fuel damage for breaks less than 10 in.
- Full flow testing

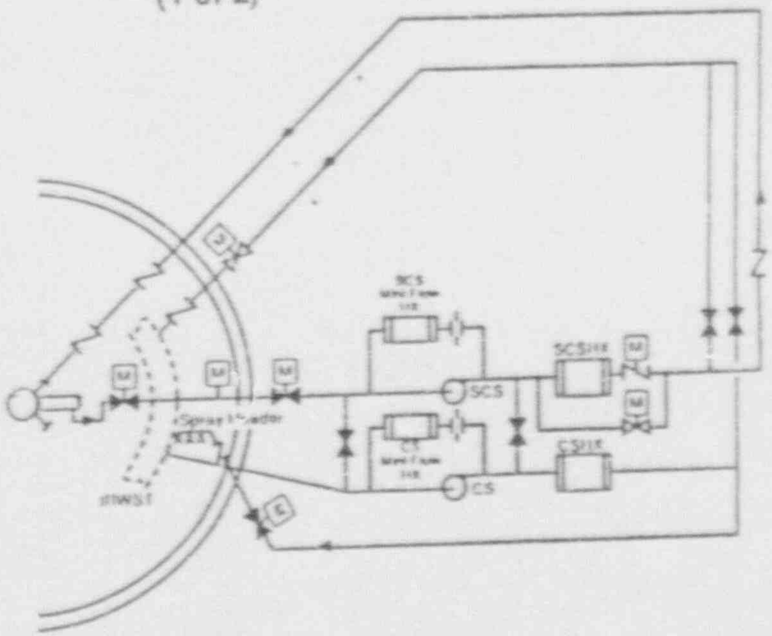
Safety Injection System
(1 of 2)



Shutdown Cooling & Containment Spray

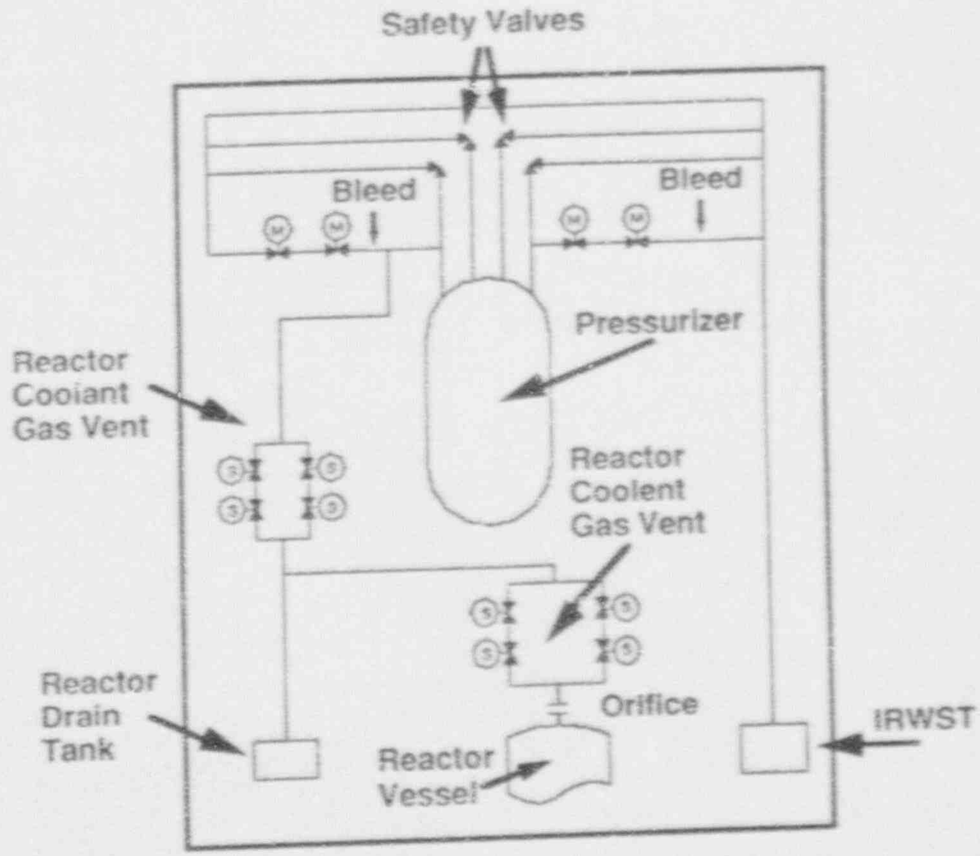
- Increased design pressure
- Dedicated system configurations
  - 2 additional heat exchangers
  - No realignment from injection
  - Pumps & heat exchangers interchangeable
- Full flow testing

Shutdown Cooling &
Containment Spray System
(1 of 2)



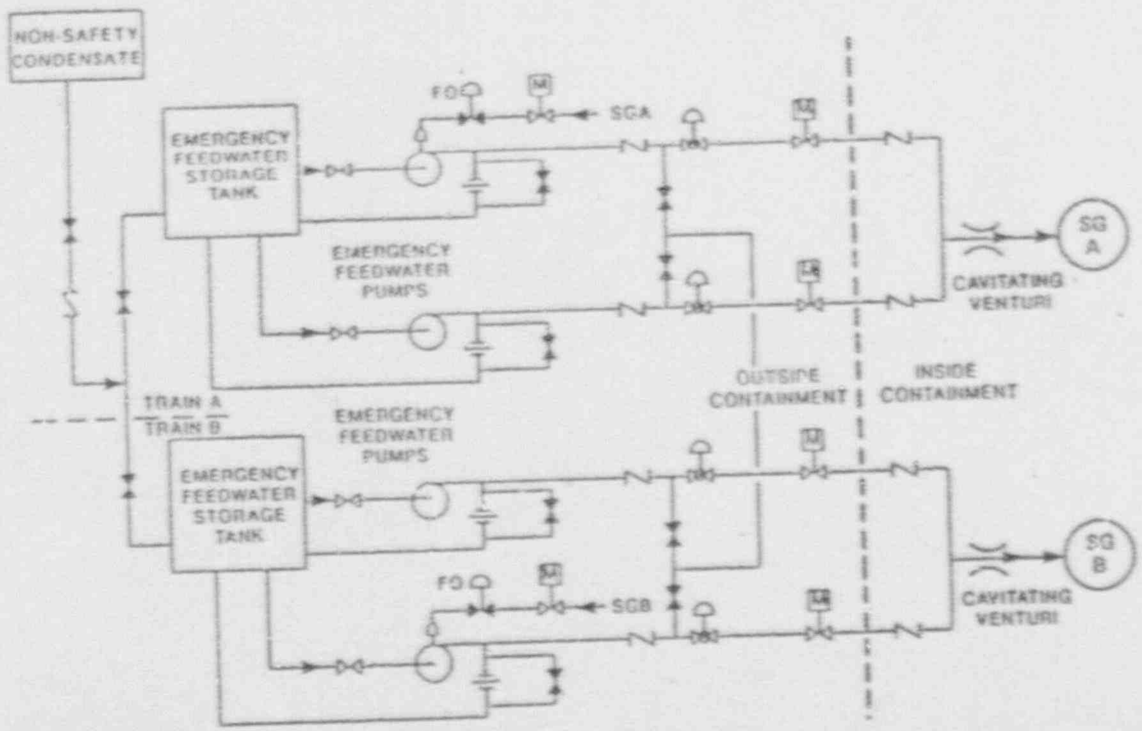
Safety Depressurization System

- Venting noncondensibles
- Alternative if Pressurizer spray is unavailable
- Depressurization to initiate feed & bleed
- Depressurization during severe accident



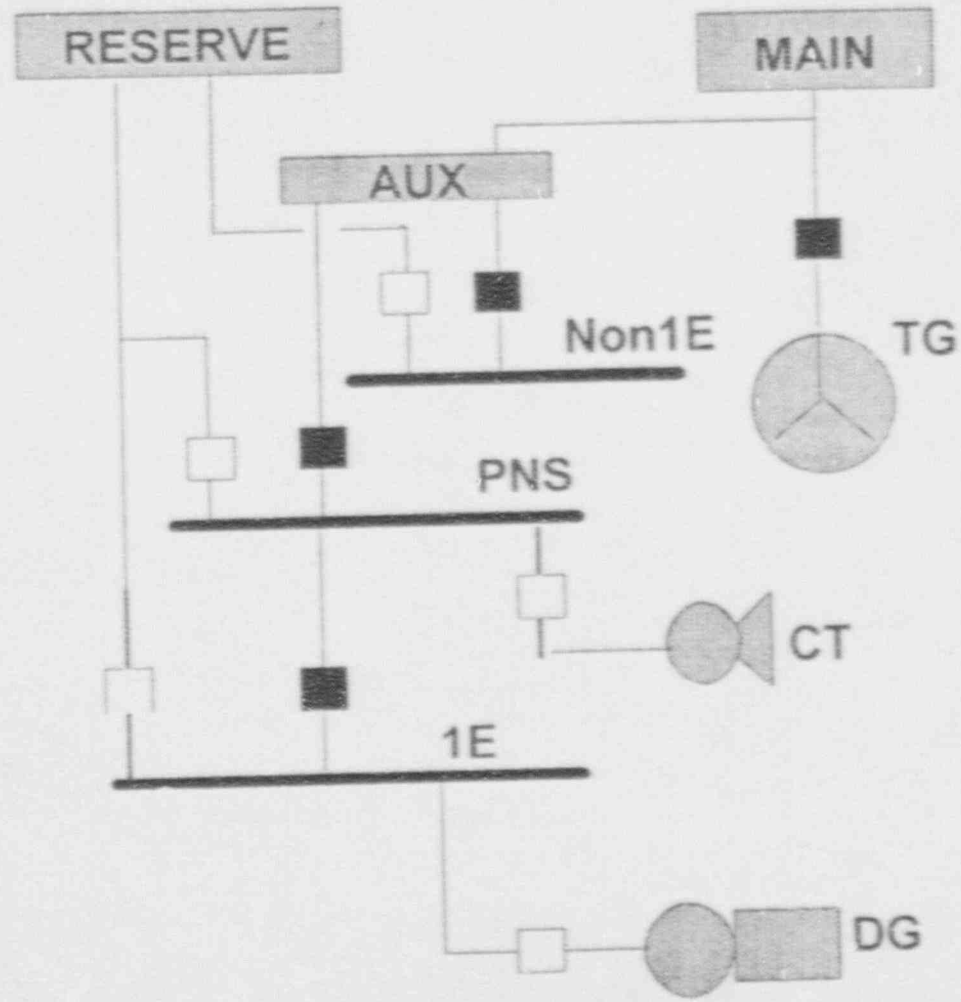
Emergency Feedwater System

- Dedicated safety system
- Motor & turbine driven pump for each SG
- Cavitating venturi eliminates isolation logic
- Redundant storage tanks



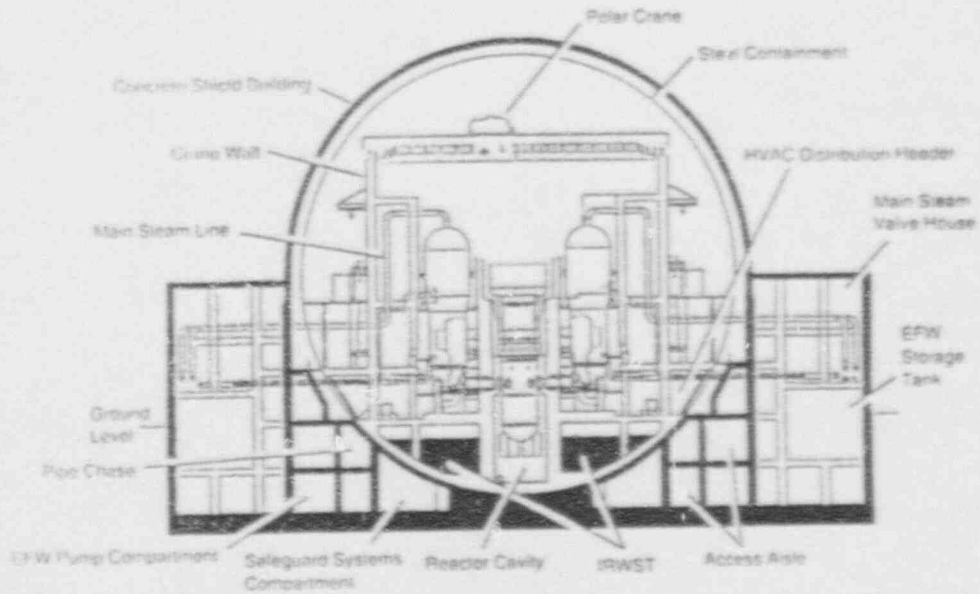
Electrical Distribution System

- TG Breaker
- 3 Tier onsite power
  - Non 1E
  - Permenant non-safety
  - 1E
- 100% increase in onsite capacity



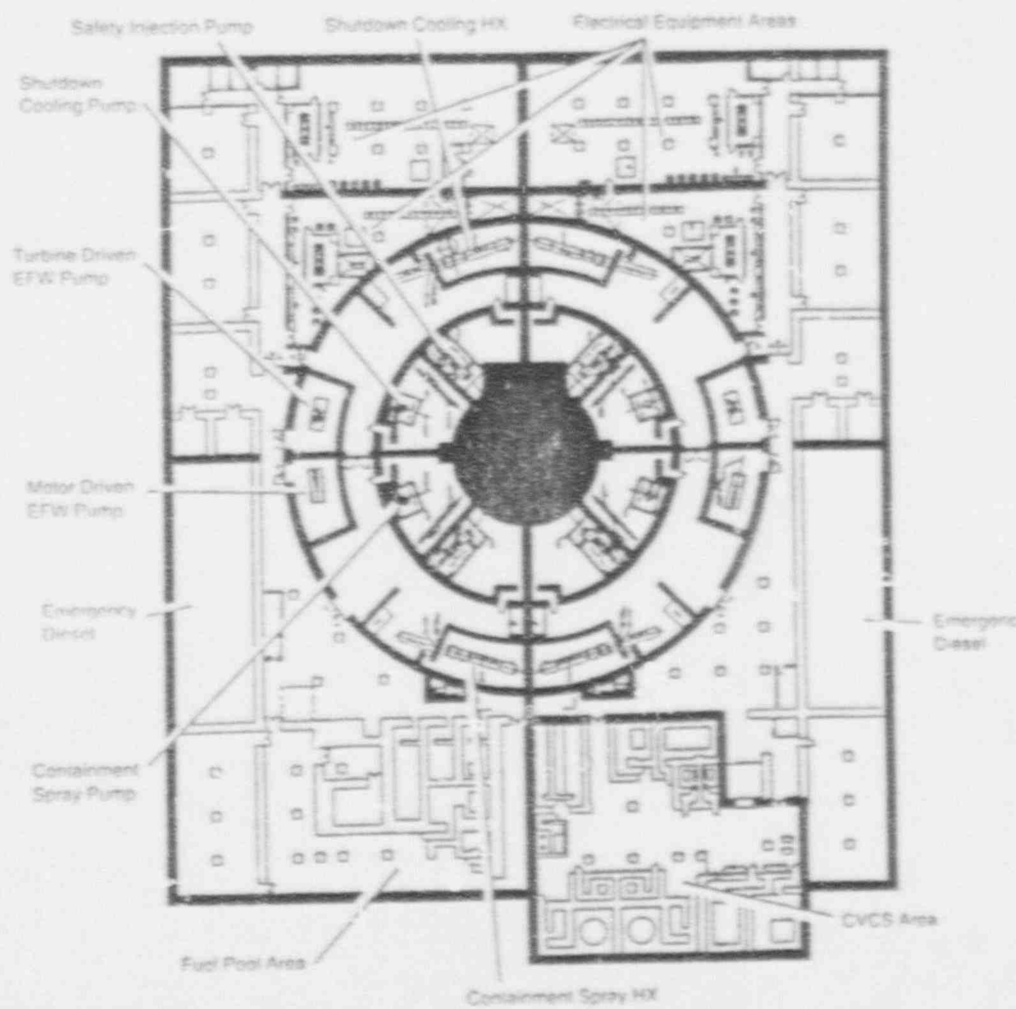
Containment

- 200 Ft. Diameter Steel Sphere
- 3 Ft Thick Shield Building
- Increased Space for Maintenance Access
- Designed to Mitigate Core Damage
- Subsphere Houses Safety Systems



Plant Layout

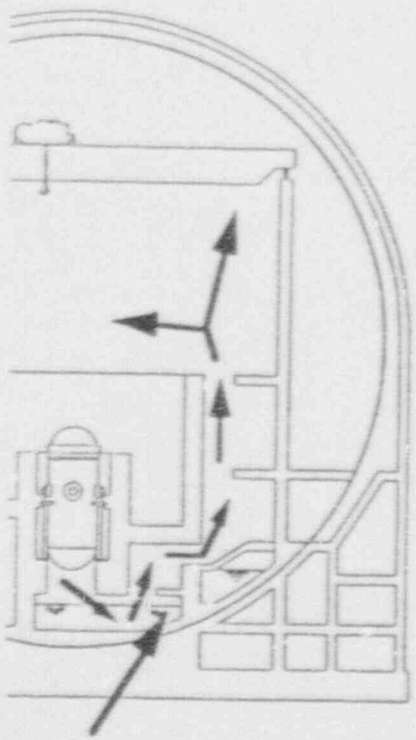
- 4 Quadrant Separation
- Common Basemat
- Safety Systems in Sub-sphere



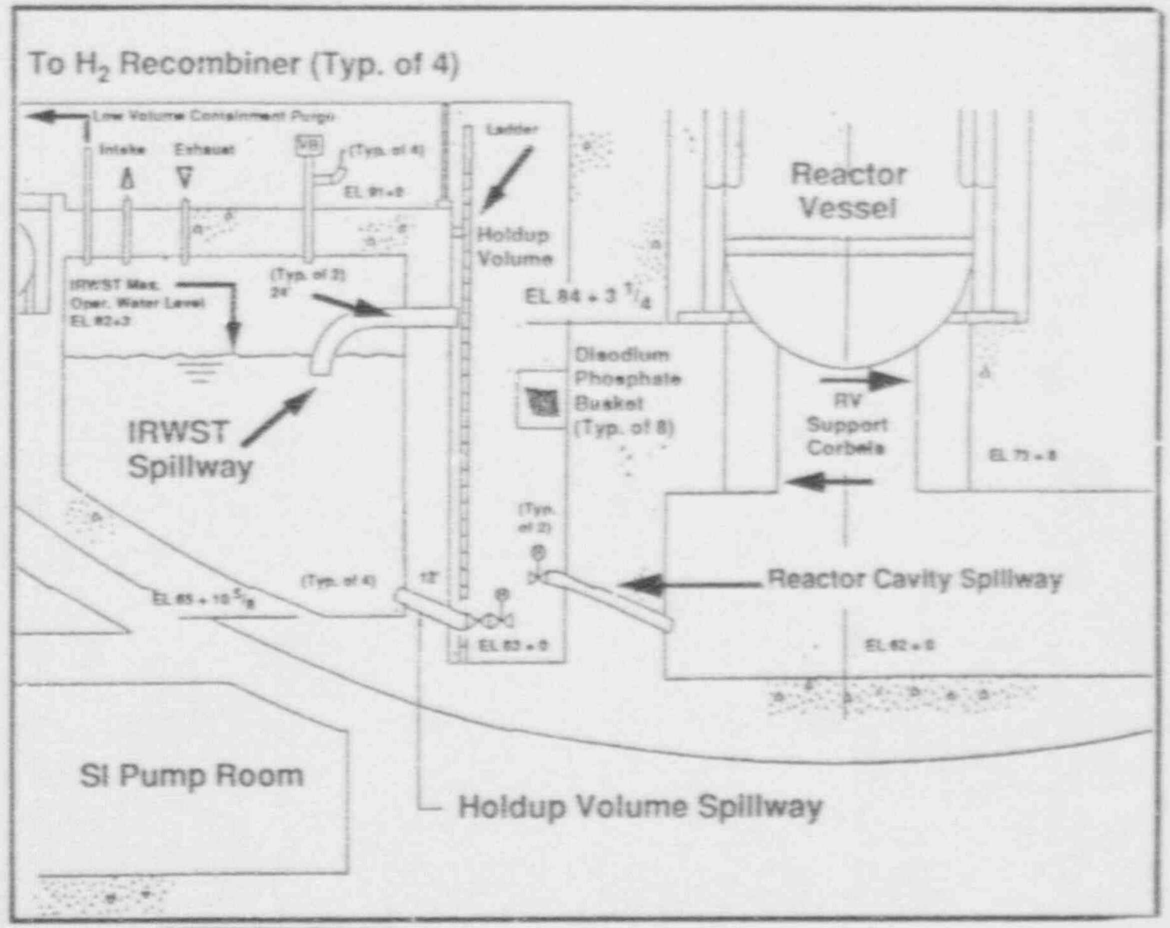
Severe . ccident Mitigation

- System 80+ includes features to provide additional Safety Margin Beyond Deterministic Licensing Bases
- Severe Accident Features
  - Cavity Design to Permit Core Spreading & Retention
  - Provisions for Cavity Flooding
  - Hydrogen Control System
  - Containment Performance Evaluation
- Reliability Assessment Program will Provide Monitoring of Systems Important to Safeaty

Severe Accident Features



CORE DEBRIS CHAMBER



RCP SEAL COOLING

- o NRC DOES NOT AGREE WITH ABB-CE DESIGN APPROACH

- o ABB-CE DESIGN APPROACH HAS REDUNDANT MEANS OF COOLING THE REACTOR COOLANT PUMP SEALS

- o NRC WANTS A SEAL ASSEMBLY THAT WOULD NOT LEAK EXCESSIVELY DURING A STATION BLACK OUT. A SATISFACTORY RESOLUTION INVOLVES:

  A DEMONSTRATION TEST

  OR

  DIVERSE, SAFETY GRADE, SEAL COOLING

- o ABB-CE IS INVESTIGATING A DEMONSTRATION TEST, IF AGREEMENT TO THE TEST ACCEPTANCE CRITERIA CAN BE ACHIEVED WITH THE NRC

- o JANUARY 21, 1993, ABB-CE WILL BE MEETING WITH THE NRC STAFF TO DEFINE THE DEMONSTRATION TEST CRITERIA

RCP SEAL INJECTION / CVCS DESIGN

SEAL INJECTION (SI) PROVIDED BY TWO INDEPENDENT & REDUNDANT CVCS DIVISIONS TO ASSURE RELIABILITY, REDUNDANCY AND AVAILABILITY.

- CHARGING/SI PORTION OF CVCS ASME III - SAFETY CLASS 3 DESIGN
- TWO CENTRIFUGAL CHARGING PUMPS - SAFETY CLASS 3 DESIGN
- CHARGING PUMPS POWERED FROM NON-SAFETY RELATED BUSES
- EACH DIVISION CAN PROVIDE COMPLETE CHARGING FLOW RANGE (44-132 GPM)
- FOR STATION BLACKOUT (SBO) EVENT CHARGING PUMPS POWERED FROM ONSITE ALTERNATE AC (AAC) POWER SUPPLY
- FOR SBO EVENT CONTINUED SEAL COOLING ASSURED BY SI AND AAC
- CVCS/SI SYSTEM DESIGN MEETS DRAFT RG 1008 REQUIREMENTS FOR AN INDEPENDENT POWERED SYSTEM

RCP SEAL COOLING / CCWS DESIGN

COOLING WATER IS PROVIDED BY THE NON-ESSENTIAL COOLING WATER HEADER.

- o CCWS CONSISTS OF TWO TRAINS
- o EACH TRAIN SUPPLIES ESSENTIAL AND NON-ESSENTIAL COOLING WATER
- o EACH TRAIN INCLUDES REDUNDANT CCW PUMPS
- o ESSENTIAL CCW LOOP IS COMPOSED OF SAFETY CLASS 3 PIPING AND COMPONENTS
- o THE NON-ESSENTIAL CCW LOOP IS COMPOSED OF NON-NUCLEAR SAFETY PIPING AND COMPONENTS

SEAL COOLING

- o SEAL COOLING PROVIDED BY INDEPENDENT AND REDUNDANT COOLING SYSTEMS

  - SEAL INJECTION (SI) WATER (6.6 GPM EACH PUMP AT 120°F) INTRODUCED INTO SEAL COOLING CIRCUIT.

  - COMPONENT COOLING WATER (CCW) WHICH COOLS SEAL WATER BY HIGH PRESSURE SEAL COOLER (HPSC) AND THROTTLE SEAL COOLERS (TSC).

- o SEALS CAN OPERATE INDEFINITELY WITH:

  - LOSS OF SEAL INJECTION (SI) WATER WITH COMPONENT COOLING WATER AVAILABLE.

  - LOSS OF CCW WITH SI AVAILABLE.

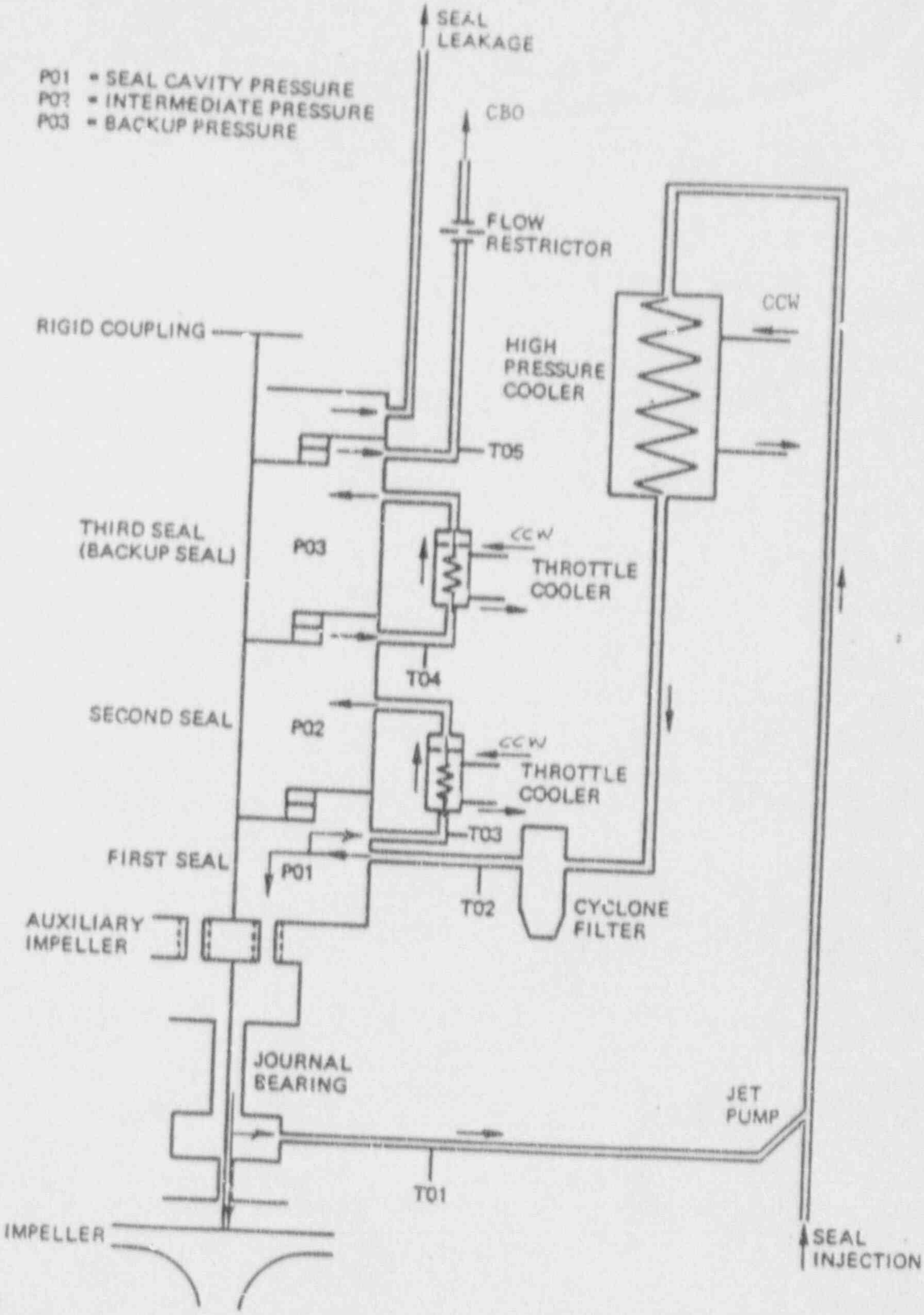


Figure 2 Flow Diagram for Hydrodynamic Shaft Seal System.
Normal Operation - with CCW & SI

RCP SEAL INTEGRITY

- o PROVEN BY MULTIPLE SHOP TESTS ON PALO VERDE RCP's
- o TEST SIMULATED ALL POSSIBLE LOSS OF SEAL COOLING EVENTS
  - LOSS OF CCW AND SEAL INJECTION WITH PUMP OPERATING
  - STATION BLACKOUT (SBO) CONDITIONS (LOSS OF CCW AND SEAL INJECTION WITH PUMP IDLE IN HOT TEST LOOP)
- o TEST RESULTS SHOWED THAT SEAL TEMPERATURE LIMITS WERE NOT EXCEEDED
- o INTEGRITY PROVEN BY ACTUAL PLANT OPERATING EXPERIENCE AT PALO VERDE
  - SEVERAL LOSS OF SEAL COOLING EVENTS INCLUDING 3 HOUR EQUIVALENT SBO EVENT
  - SEALS LAST FOR AT LEAST TWO FUEL CYCLES, THIRD CYCLE BEING CONSIDERED

TECHNICAL ISSUE: I&C DIVERSITY

RECOMMENDATIONS FOR ACHIEVING COMPLIANCE
WITH NRC POSITION ON
DIVERSITY AND DEFENSE IN DEPTH.

PRESENTED AT NRC-CE
SENIOR MANAGEMENT MEETING

JANUARY 11, 1993

Nuplex 80+™ DEFENSE-IN-DEPTH

Critical Function		Success Path	
		Non-safety	Safety
Reactivity Control		Rod Control, CVCS (Boration)	Safety Injection System, Reactor Trip Breakers
Vital Auxiliaries	AC	Main Transformer, Alternate A/C	Emergency Diesels, Aux Transformers
	DC	Station Battery	Station Battery
RCS Inventory Control		CVCS (Charging/Letdown)	Safety Injection System
RCS Pressure Control		Heaters/Spray, CVCS (Charging)	Safety Injection System, Safety Depressurization System
Core Heat Removal		Forced Circulation	Natural Circulation
RCS Heat Removal		Main Feed	Emergency Feed, Shutdown Cooling & Safety Injection System
Containment Isolation		Control Valves	Isolation Valves
Containment Environment		Fan Coolers, H_2 Ignitors	Containment Spray, Recombiners
Radiation Emission		Monitor and Control Radiation Release Paths	Isolation of Release Paths

ABB
ASEA BROWN BOVERI

RESULTS OF DIVERSITY & DEFENSE IN DEPTH EVALUATION

FOR THE 28 CHAPTER 15 EVENT INITIATORS WHICH APPLY TO SYSTEM 80+:

1) FOR 19 EVENTS, THE ABB EVALUATION DETERMINED THE DIVERSITY IN NUPLEX 80+ TO BE ADEQUATE:

   16 - MODERATE FREQUENCY

   2 - INFREQUENT

   1 - LIMITING FAULT

2) FOR 4 EVENTS, THE ABB EVALUATION DETERMINED THAT ADDITIONAL ANALYSIS WOULD BE NEEDED TC DEMONSTRATE THAT THE DIVERSE SYSTEMS WOULD MEET CHAPTER 15 CRITERIA:

   LOSS OF OFFSITE POWER - MODERATE FREQUENCY

   RCP SHAFT SEIZURE - LIMITING FAULT

   RCP SHAFT SHEAR - " "

   CEA EJECTION - " "

(CONTINUED)

RESULTS OF DIVERSITY & DEFENSE IN DEPTH EVALUATION

(CONTINUED)

IT IS LIKELY THAT FURTHER DISCUSSION WITH THE STAFF WILL DETERMINE THAT ADDITIONAL ANALYSIS IS NOT REQUIRED TO ESTABLISH THAT THE DIVERSE SYSTEMS ADEQUATELY ADDRESS THESE EVENTS (BASED ON RELAXED ACCEPTANCE CRITERIA).

3) FOR 5 LIMITING FAULT 1 & 3 EVENT INITIATORS, ABB'S ANALYSIS DETERMINED THAT THE DIVERSE SYSTEMS DO NOT PROVIDE ADEQUATE PROTECTION:

LIMITING FAULT EVENTS	ADDITIONAL PROTECTION REQUIRED
LETDOWN LINE BREAK	CLOSURE OF LETDOWN ISOLATION VALVE.
STEAM GENERATOR TUBE RUPTURE	CLOSURE OF THE MSIVs.
MAIN STEAM LINE BREAKS	ACTUATION OF REACTOR TRIP AND MSIVs.
FEEDWATER PIPE BREAKS	ACTUATION OF REACTOR TRIP AND MSIVs.
LOSS OF COOLANT ACCIDENT	ACTUATION OF REACTOR TRIP, SAFETY INJECTION, CONTAINMENT SPRAY, AND CONTAINMENT ISOLATION.

ABB RECOMMENDATIONS

1) WORK WITH THE NRC STAFF TO ESTABLISH ADEQUACY OF DIVERSE SYSTEMS FOR ALL MODERATE AND INFREQUENT EVENT INITIATORS.

2) MITIGATION OF LIMITING FAULT EVENTS SHOULD BE VIA MANUAL ACTUATION OF ESF SYSTEMS USING CONTROLS IMPLEMENTED TO COMPLY WITH DEFENSE IN DEPTH POSITION 4.

   JUSTIFICATION

   - MANUAL ACTUATION SHOULD PROVIDE ADEQUATE PROTECTION RELATIVE TO SITE RELEASE CRITERIA, GIVEN THE IMPROBABILITY OF A COINCIDENT COMMON MODE FAILURE.

   - LIMITING FAULT EVENTS SHOULD NOT BE CONSIDERED IN THE DESIGN BASES FOR THE DIVERSE SYSTEMS BECAUSE THE PROBABILITY OF A COINCIDENT COMMON MODE SOFTWARE FAILURE IS NOT CREDIBLE.

   - MODIFYING THE DIVERSE SYSTEMS TO INCLUDE LIMITING FAULT EVENTS IN THE DESIGN BASIS WILL ADD COMPLEXITY, (THEREFORE REDUCE RELIABILITY) AND INCREASE SUSCEPTABILITY TO SPURIOUS ACTUATION (THEREFORE INCREASED CHALLENGES TO PLANT SAFETY).

ABB RECOMMENDATION FOR COMPLIANCE WITH DEFENSE IN DEPTH POSITION 4

1. MANUAL CONTROLS

   - MANUAL ACTUATION PROVIDED FOR 1 TRAIN OF EACH ESF PRIMARY FLOW PATH

   - CONTROL IMPLEMENTATION BYPASSES ALL MULTIPLEXING AND ALL COMPUTERS WITH LARGE SOFTWARE APPLICATIONS (I.E., HARDWIRED)

     THE LAST PLC PROVIDING DIRECT CONTROL TO THE CONTROLLED COMPONENTS IS RETAINED.

   JUSTIFICATION

   - SMALL PLCs ARE MORE RELIABLE THAN EQUIVALENT MECHANICAL RELAYS.

   - THE SOFTWARE APPLICATION IS SUFFICIENTLY SMALL AND SIMPLE, SUCH THAT IT IS 100% TESTABLE.

     (I.E., NO COMMON MODE FAILURES)

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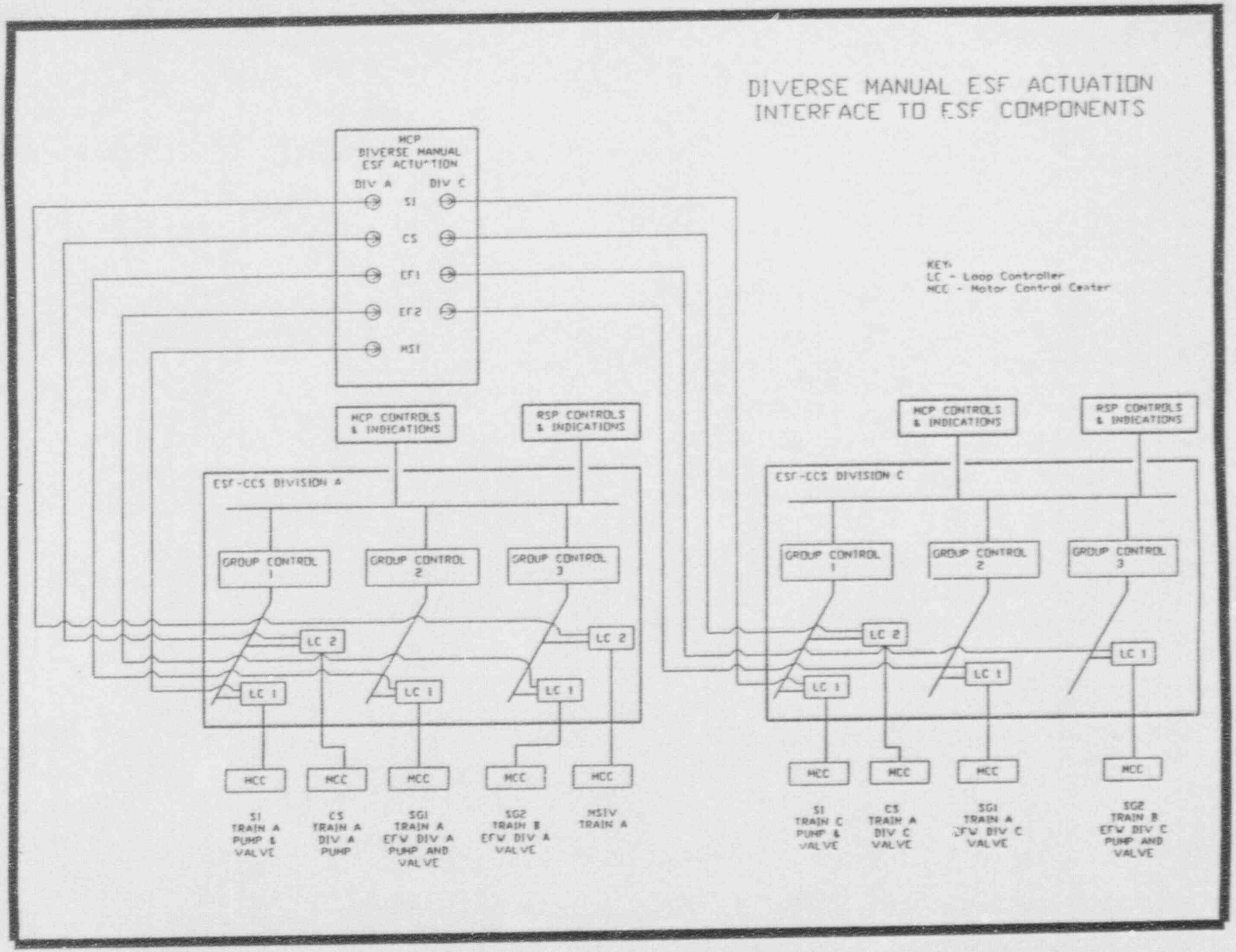

DIVERSE MANUAL ESF ACTUATION
INTERFACE TO ESF COMPONENTS
MCP
DIVERSE MANUAL
ESF ACTUATION
DIV A
DIV C
SI
CS
EF1
EF2
MSI
KEY:
LC - Loop Controller
MCC - Motor Control Center
MCP CONTROLS & INDICATIONS
RSP CONTROLS & INDICATIONS
ESF-CCS DIVISION A
GROUP CONTROL 1
GROUP CONTROL 2
GROUP CONTROL 3
LC 2
LC 2
LC 1
LC 1
LC 1
MCC
MCC
MCC
MCC
MCC
SI TRAIN A PUMP & VALVE
CS TRAIN A DIV A PUMP
SG1 TRAIN A EFW DIV A PUMP AND VALVE
SG2 TRAIN B EFW DIV A VALVE
MSIV TRAIN A
MCP CONTROLS & INDICATIONS
RSP CONTROLS & INDICATIONS
ESF-CCS DIVISION C
GROUP CONTROL 1
GROUP CONTROL 2
GROUP CONTROL 3
LC 2
LC 1
LC 1
LC 1
MCC
MCC
MCC
MCC
SI TRAIN C PUMP & VALVE
CS TRAIN A DIV C VALVE
SG1 TRAIN A EFW DIV C VALVE
SG2 TRAIN B EFW DIV C PUMP AND VALVE

SYSTEM 80+ TM

ABB RECOMMENDATION FOR COMPLIANCE WITH DEFENSE IN DEPTH POSITION 4

(CONTINUED)

2. INDICATORS

   - INDICATORS ARE PROVIDED FOR REG. GUIDE 1.97 CATEGORY 1 PARAMETERS.

   - CONTINUE TO USE COMPUTERS FOR INSTRUMENTATION WHICH IS IMPLEMENTED USING COMPUTERS ON ALL CURRENT PLANTS, E.G.:

     CORE EXIT THERMOCOUPLES

     REACTOR VESSEL LEVEL MONITORS

     SUBCOOLED MARGIN MONITOR

   - ONE CHANNEL OF OTHER INSTRUMENTS ARE HARDWIRED TO A DEDICATED DISPLAY DEVICE, WHICH USES A SMALL COMPUTER.

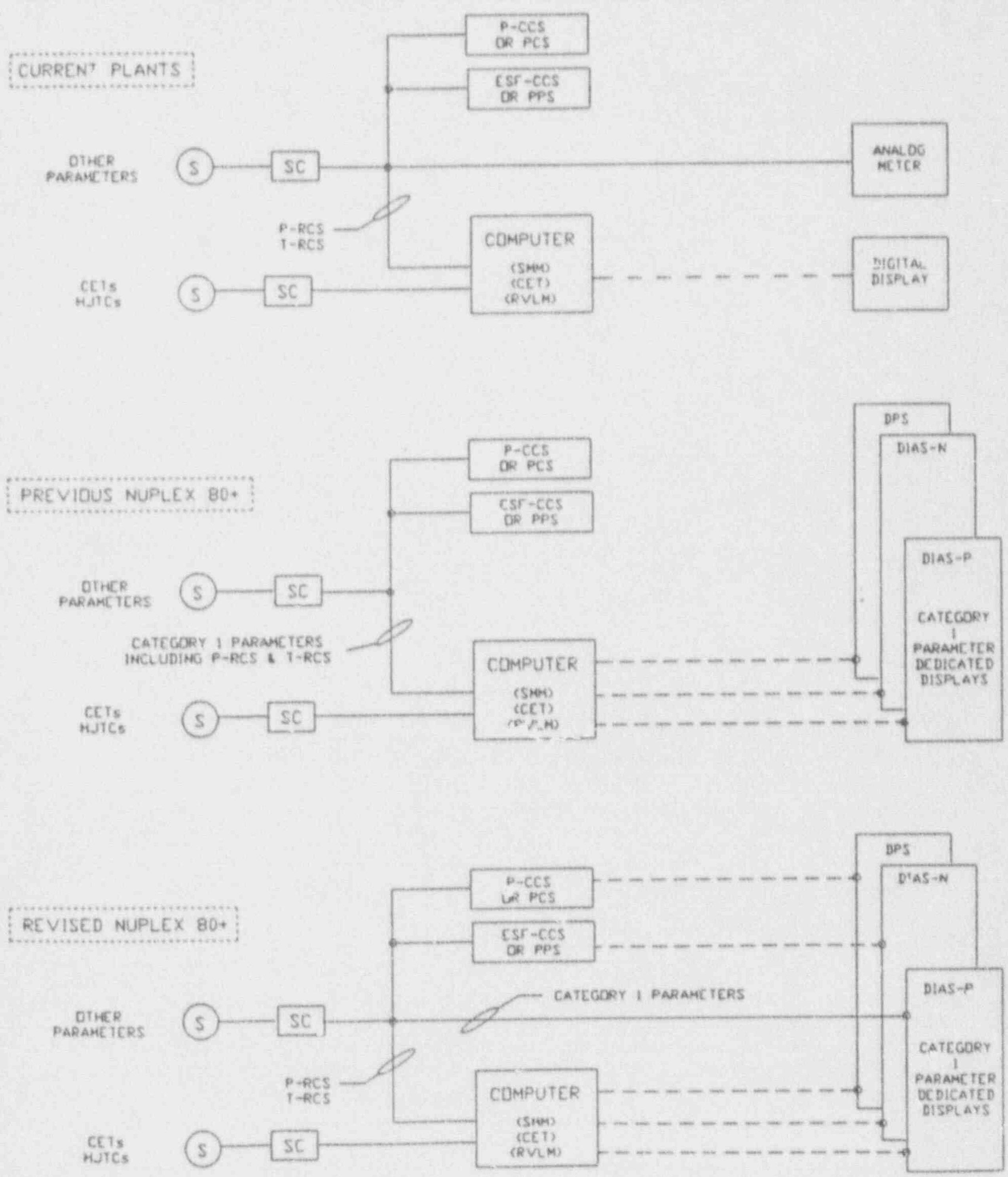
   <u>JUSTIFICATION</u>

     - COMPUTER BASED DISPLAYS HAVE DEMONSTRATED HIGHER RELIABILITY THAN THEIR ANALOG PREDECESSORS.

     - THE SOFTWARE APPLICATION IS SUFFICIENTLY SMALL AND SIMPLE, SUCH THAT IT IS 100% TESTABLE.

       (I.E., NO COMMON MODE FAILURES)

CHRONOLOGY OF DIGITAL DISPLAY IMPLEMENTATION FOR REG. GUIDE 1.97 CATEGORY 1 PARAMETERS IN SAFETY RELATED DISPLAY INSTRUMENTATION



Safety System Software Development Program

- For the 1/15/93 submittal, ABB-CE will provide the following for implementation in Tier 2:

    - Software Safety Plan Description
    - Software Program Manual, consisting of
        - Software Quality Assurance Plan
        - Software Verification & Validation Plan
        - Software Configuration Management Plan
        - Software Operations & Maintenance Plan

- ABB-CE has identified how safety software should be treated in the ITAAC:

    Software in a safety system is satisfactorily developed according to the software plans, as evidenced by successful audits of the software development programs against those plans.

- Some reviewers are not satisfied that safety software will be adequately assured through this ITAAC approach.

- Although ABB-CE is reviewing other approaches, no satisfactory alternative has been identified by either NRC or ABB-CE.

- ABB-CE and NRC are working constructively to resolve this problem.