Docket File

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

March 4, 1992

Docket No. 52-002

APPLICANT: Combustion Engineering, Inc. (CE)

PROJECT: CE System 80+

SUBJECT: SUMMARY OF MANAGEMENT MEETING HELD AT CE OFFICE IN WINDSOR, CONNECTICUT, ON JANUARY 21 AND 22, 1992

A meeting was held between senior management of the Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC) and CE, at the CE office in Windsor, Connecticut, on January 21, 1992. Two members of NRR project management remained for the morning of January 22, 1992, to complete discussions on some technical issues. Enclosure 1 lists the attendees at the meeting. Enclosure 2 provides the information presented at the meeting by CE except for the design information and PRA information which are contained in the CE safety analysis report (CESSAR-DC).

The purposes of the meeting were for CE to describe to NRC management the organization of CE, CE's goals and objectives with regard to System 80+, a design features overview of System 80+, and major issues that will require significant manpower resources to complete NRC evaluation. During the course of the presentations, there was an interchange of observations, and comments. The more substantive of these are addressed in this summary.

Reactor Trip Reduction Program

In response to questions about reactor trips induced by the secondary side of the plant and the goal of less than one reactor trip per year, CE responded that no effort was made to reduce trip initiators on the secondary side of the plant. They stated that they would provide a report on their trip reduction program work.

Steam Generator Design

EPRI Utilities' Requirements Document (URD) recommends 600°F hot leg temperature and System 80+ has 615°F. Part of EPRI's basis for lowering the temperature is to reduce susceptibility to steam generator (S/G) tube cracking and rupture. CE responded that System 80+ has Alloy 690 tubes which reduces the susceptibility for stress corrosion cracking and also a special thermal treatment of tubing to make it softer prior to installation in the S/G. NRC asked whether CE has considered a cost/benefit analysis to examine the desirability of increasing the S/G shell side pressure rating. The purpose is to prevent a steam generator tube rupture that could result in an external

DF03

release since the reactor primary pressure reduction following the event would reduce pressure below the S/G safety valve setting for the higher S/G design pressure. CE stated that an industry study was done on this in which CE participated in the mid-1980's. (Subsequently in a meeting between NRC and EPRI, EPRI stated they would send the report on this study to NRC.)

Human Factors Engineering

CE stated that human factors engineering was not applied in a formal sense outside the control room except for the remote shutdown panel. They indicated that they would review the emergency operating procedures to identify whether there were any operator actions required outside the control room. Additional information on the control room human factors engineering is being provided by the end of February. This issue remains open.

Reactor Coolant Pump (RCP) Seals

CE stated that the RCP seals do not fail catastrophically and that redundant seal cooling is provided by seal injection from the Chemical and Volume Control System (CVCS) and from the Component Cooling Water System. NRC asked that CE address any commonalities between these two systems. CE stated that it appeared to them that the NRC reviewer was objecting to the CVCS being a non-safety grade source of seal cooling. NRC indicated that this must be considered and justified.

Shutdown Risk

CE stated that they would provide an interim report on this subject in May 1992, and a final report in August 1992. NRC suggested a meeting in early March on this subject to shorten the schedule.

Seismic Analysis/Structural Design

CE stated that they were providing a response to the staff's concern on buckling of the containment that they believed would resolve the issue. CE also stated that the concern about the margin provided by the containment design pressure was an issue that was being driven by the operating basis earthquake (OBE) which may be resolved by the pending change to the regulations (10 CFR Part 100). NRC advised CE that an exemption should be applied for since the schedule of the change in the regulation would not support the System 80+ certification schedule.

Safety Analysis-Fuel Failure Criteria

NRC has previously approved the use of a convolution methodology for treating DNBR for fuel failure for the locked reactor coolant pump rotor accident and the control element assembly ejection accident. For System 80+, CE has also used this methodology for steamline break and loss of condenser vacuum with a single failure. This issue will have to be discussed expeditiously. To date, the NRC staff has resisted this extension, based on the probability of occurrence of the accidents being considered.

Time Delay for Loss of Off-Site Power

NRC has previously approved a three second time delay between reactor trip and loss of off-site power. CE uses this time delay for the locked reactor coolant pump rotor accident and steam generator tube rupture. The NRC has questioned continuing the use of this allowance for advanced plants. This is another issue that will have to be discussed expeditiously.

Severe Accidents

NRC gave to CE a copy of the questions on severe accidents that were sent to GE on the ABWR. Similar questions will be sent to CE.

ISLOCA

CE relies on PRA for not upgrading some systems. They perceive that the staff wants a system by system cost-benefit analysis.

Single Failure

CE assumes worst safety grade or non-safety grade failure. CE has previously been granted credit for redundant control grade equipment (e.g., turbine stop and throttle valves, alarms in CVCS for letdown line rupture). The NRC staff has concerns about both of these assumptions.

Both NRC and CE agreed that these issues and any others that are identified should not be stockpiled for the DSER but should create discussions as soon as possible with the objective of resolving as many as possible before the DSER is issued.

Original Signed By:

Thomas V. Wambach, Project Manager
Standardization Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. CE Presentation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	PDST R/F	PShea	DCrutchfield
NRC PDR	WTravers	CPoslusny	VMcCree
TMurley/FMiraglia	RNease	JNWilson	RPierson
TBoyce	RBorchardt	FHasselberg	THiltz
TKenyon	TWambach	JHWilson	JMoore, 15B18
EJordan, 3701	ACRS (10)	GGrant, EDO	CThomas, 10H3
WRussell, 12G18	AThadani, 8E2	AEl-Bassioni, 10E4	
JO'Brien, NLS217A	MMalloy		

92-046

OFC:	LA:PDST	PM:PDST TVW	SC:PDST
NAME:	PShea	TVWambach:sg	JNWilson
DATE:	03/4/92	03/4/92	03/4/92

OFFICIAL DOCUMENT COPY: CE121.TW

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Enclosure 1

CE SYSTEM 80+

Meeting Attendees

January 21, 1992

Name	Affiliation
Tom Wambach	NRC/NRR/PDST
C. B. Brinkman	ABB-CE
E. H. Kennedy	ABB-CE
R. E. Newman	ABB-CE
R. A. Matzie	ABB-CE
Stan Ritterbusch	ABB-CE
Rick Turk	ABB-CE
Peter M. Lang	DOE - Germantown
George A. Davis	ABB-CE
Kashmira Mali	DOE - Oakland
Mark Crump	ABB-CE
Bill Fox	Duke Engr. & Serv.
Cecil Thomas	NRC/NRR/DLPQ
Robert Pierson	NRC/NRR/PDST
William T. Russell	NRC/NRR/ADT
Thomas Murley	NRC/NRR
Ashok Thadani	NRC/NRR/DST
William Travers	NRC/NRR/DAR
Adel A. El-Bassioni	NRC/NRR/DREP

Enclosure 1

CE SYSTEM 80+

Meeting Attendees

January 22, 1992

Name	Affiliation
Tom Wambach	NRC/NRR/PDST
C. B. Brinkman	ABB-CE
R. A. Matzie	ABB-CE
Stan Ritterbusch	ABB-CE
Rick Turk	ABB-CE
Peter M. Lang	DOE - Germantown
Kashmira Mali	DOE - Oakland
Mark Crump	ABB-CE
Bill Fox	Duke Engr. & Serv.
Robert Pierson	NRC/NRR/PDST
George A. Davis	ABB-CE

Enclosure 2

Meeting Between

NRC Office of Nuclear Reactor Regulation

and

Combustion Engineering, Inc. Nuclear Power Systems

January 21-22, 1992

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AGENDA - DAY 1

8:00	Organization and System 80+™ Marketing	R. E. NEWMAN
8:30	Agenda and System 80+™ Review Schedule	C. B. BRINKMAN
9:00	System 80+ Standard Plant Design Objectives and Development	R. A. MATZIE
9:30	Design Description and Comparisons	R. S. TURK/M. W. CRUMP
11:00	Improved Plant Arrangement and Computer Assisted Design (PASCE)	W. FOX
12:30	Lunch	
1:00	PRA Description and Results	D. J. FINNICUM
1:30	Summary of Significant Review Issues	S. E. RITTERBUSCH
3:15	Nuplex 80+™ Advanced Control Room	K. SCAROLA
4:00	Demonstration of Nuplex 80+ Mockup	D. L. HARMON
4:45	Complete Day #1	

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AGENDA - DAY 2

8:30	Continued Discussions as Needed	S. E. RITTERBUSCH
10:30	Wrap-up Session	C. B. BRINKMAN

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R. E. Newman

President

CE Nuclear Systems

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ABB Group Organization

ASEA Shareholders

BBC Shareholders

ASEA AB
Sweden

BBC Brown Boveri Ltd.
Switzerland

50%

50%

ABB Asea Brown Boveri Ltd.
Zurich, Switzerland

ABB Asea Brown Boveri Group
1,150 companies
50 Business Areas
in 8 Business Segments

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Asea Brown Boveri Ltd

World's largest
Electrical Engineering Group

- 220,000 employees
- 1,150 companies in
- 140 countries

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ABB NUCLEAR POWER PERSONNEL

ABB ATOM	1,200
ABB REAKTOR	250
ABB BADEN	25
ABB CE NUCLEAR POWER	3,600
	5,075

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ABB Combustion Engineering Nuclear Power

- S. T. Brewer, Chairman, ABB CENP
  - P. Van Nort, President, ABB CENP
    - CE Nuclear Systems — R. E. Newman, President
    - CE Nuclear Fuel — R. S. Siudek, President
    - CE Nuclear Services — W. S. Skibitsky, President
    - IMPELL — J. A. Famiglietti, President

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ABB Combustion Engineering Nuclear Systems

Organization

Robert E. Newman
President

- **Far East Operations**
  James W. Veirs
  Vice President
- **Nuclear Systems Development**
  Regis A. Matzie
  Vice President
- **Nuclear Systems Engineering**
  James D. Crawford
  Vice President
- **Domestic Projects**
  Michael F. Barnoski
  Vice President
- **Electro-Mechanics Total Quality**
  Donald F. Pedretti
  Vice President and General Manager
- **Newington Operations**
  Richard P. Adams
  Vice President and General Manager
- **Quality Assurance**
  John P. Pasquenza
  Director

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CRITICAL FACTORS: Proven Technology

Level of Experience in:	Evolutionary ALWR's	Passive ALWR's	Non–Water Advanced Reactors
■ Design Concept	Extensive	Extensive	Extensive
■ Detailed Design Implementation	Extensive	Moderate	Moderate
■ Projected Construction Schedule	Extensive	Moderate	Moderate
■ Engineering & Procurement Cost Estimates	Extensive	Moderate	Limited
■ Maintenance and Operating Cost Estimates	Extensive	Moderate	Limited
■ Sustained Reliable Operation	Extensive	Limited	Limited

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CRITICAL FACTORS: Regulatory Risk

	Evolutionary ALWR's	Passive ALWR's	Non–Water Advanced Reactors
■ Design Detail Available to Support Certification	Yes	No	No
■ Applicability of Current NRC Regulations	Yes	Partial	Minimal
■ Resolution of "New" Regulatory Issues	Known	Not Known	Not Known
■ Is Prototype Required for Certification?	No	Unlikely	Likely
■ Cost to Achieve Certification	10's of Millions	100's of Millions	More Than 1 Billion

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The U. S. Outlook for Advanced Reactors Mid 1990's

- Certification of Evolutionary ALWR's
- Large, Successful Nuclear Utilities Reenter Market
  - ➤ Individually, or
  - ➤ As a Major Member of IPP
- Order for Evolutionary ALWR and Application for Combined License
- Continued Development and NRC Review of Passive ALWR's and Non-Water Advanced Reactors

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The U.S. Outlook for Advanced Reactors (Con't)

Late 1990's

- Continued Deployment of Evolutionary ALWR's by Utilities and/or IPP's
- Certification of Passive ALWR's
- Continued Development of Non–Water Advanced Reactors

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The U.S. Outlook for Advanced Reactors (Con't)

2000 and Beyond

- Entry of Smaller Utilities Into Nuclear Market (Possibly IPP's?)
- Construction of "Lead" Unit Passive ALWR
- Plans for Construction of Non–Water Advanced Prototype

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OUTLOOK FOR OVERSEAS MARKETS

- Continued Commitment to Evolutionary ALWR'S
  - France
  - United Kingdom
  - Korea
  - Taiwan
  - Germany
  - Japan

- Slow Improvement in Acceptability of Nuclear Power
  - Europe: environmental concerns; oil dependence
  - Asia: load growth; lack of domestic resources

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C. B. Brinkman

Manager

Washington Nuclear Operations

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SYSTEM 80 +
DESIGN CERTIFICATION
PROGRAM MILESTONES

•	First SAR Submittal	November, 1987
•	Expanded Scope to Essentially Complete Plant	March, 1989
•	SAR Submittals Complete	March, 1991
•	Application Docketed	May, 1991
•	Draft SER (est)	September, 1992 (NRC) July, 1992 (NPOC)
•	Final SER/FDA (est)	November, 1993 (NRC) May, 1993 (NPOC)
•	Design Certification	May, 1995 (NRC) May, 1994 (NPOC)

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REMAINING SUBMITTALS

• Reliability Assurance Program Description	1/92
• Interface Requirements	2/92
• SRP Deviations	3/92
• Fire Hazards Analysis	3/92
• ITAAC	*5/92
• NEPA/SAMDA	5/92

Submittals do not raise new technical issues
Support draft SER date of 9/92

*Presumes GE lead examples resolved in Dec. 91.

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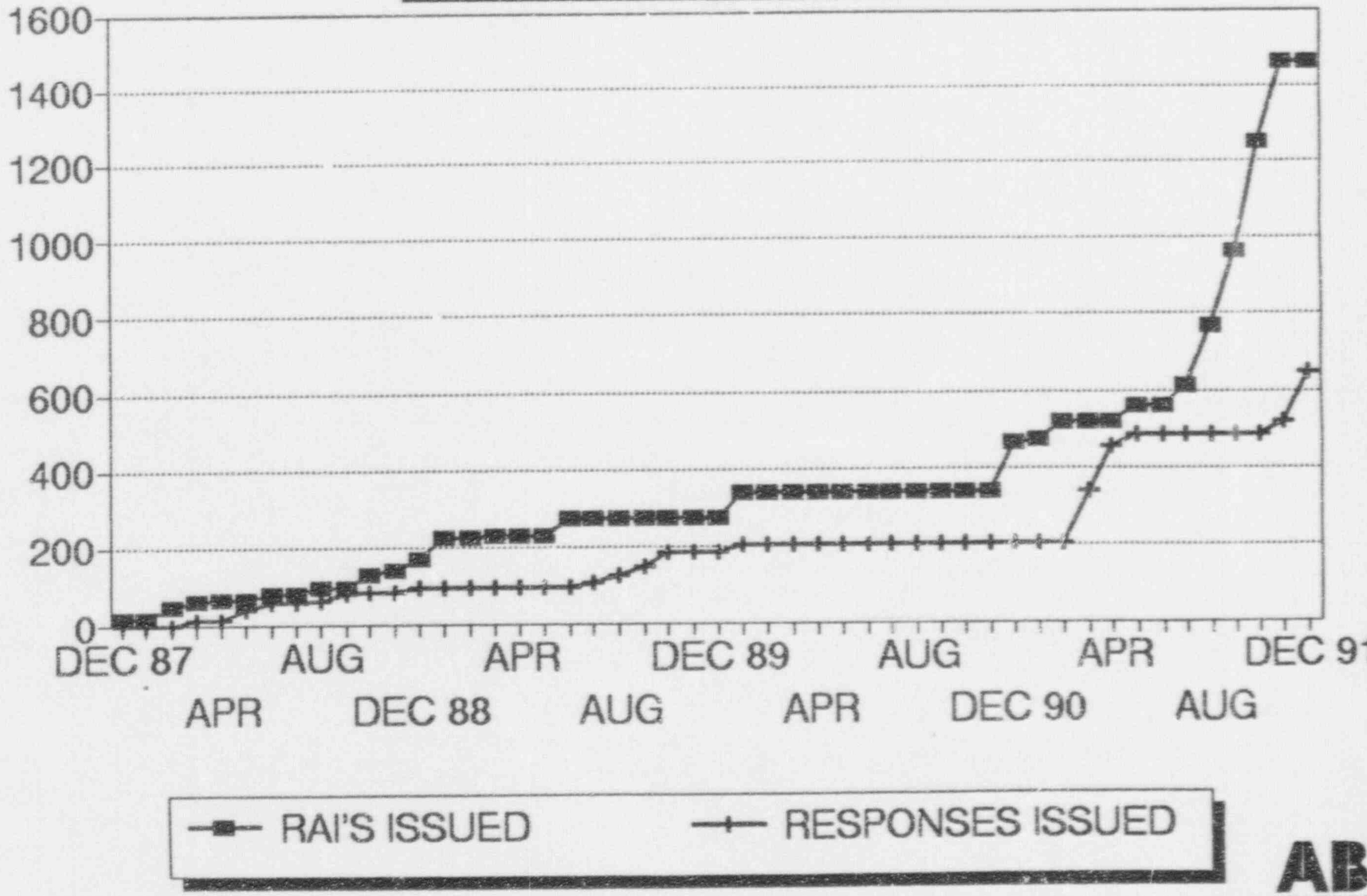
CESSAR-DC
STATUS OF NRC RAI'S



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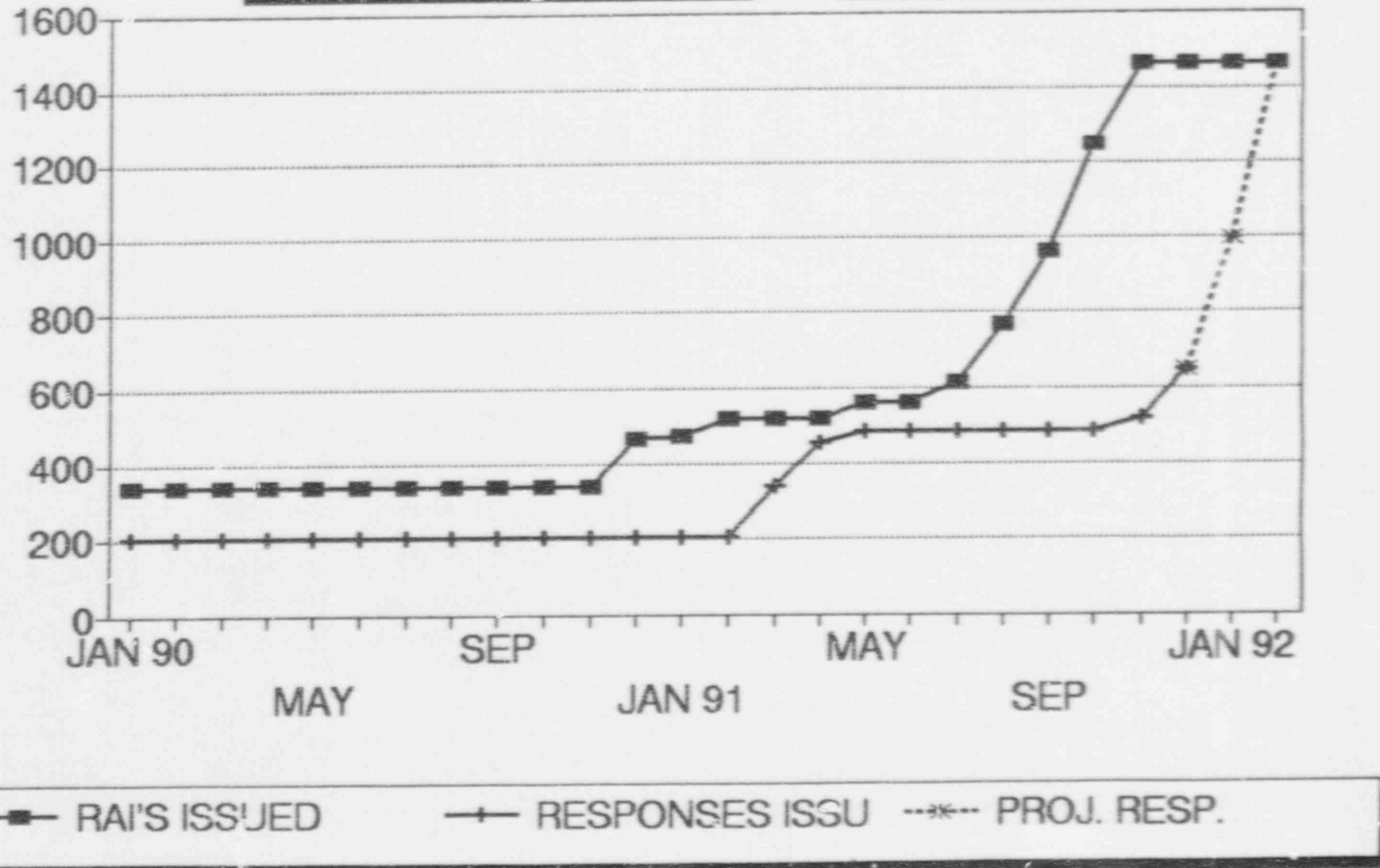
CESSAR-DC
PROJECTED RESPONSES TO RAI'S



Regis A. Matzie

Vice President

Nuclear Systems Development

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SYSTEM 80+ DESIGN DEVELOPMENT

- Assess Market Requirements
- Design Development
  - Design Certification
- Pre-Order FOAKE
  - Order
- Post-Order FOAKE
- Site-Specific Engineering
  - Combined License
- Construction Reconciliation/Procurement Engineering

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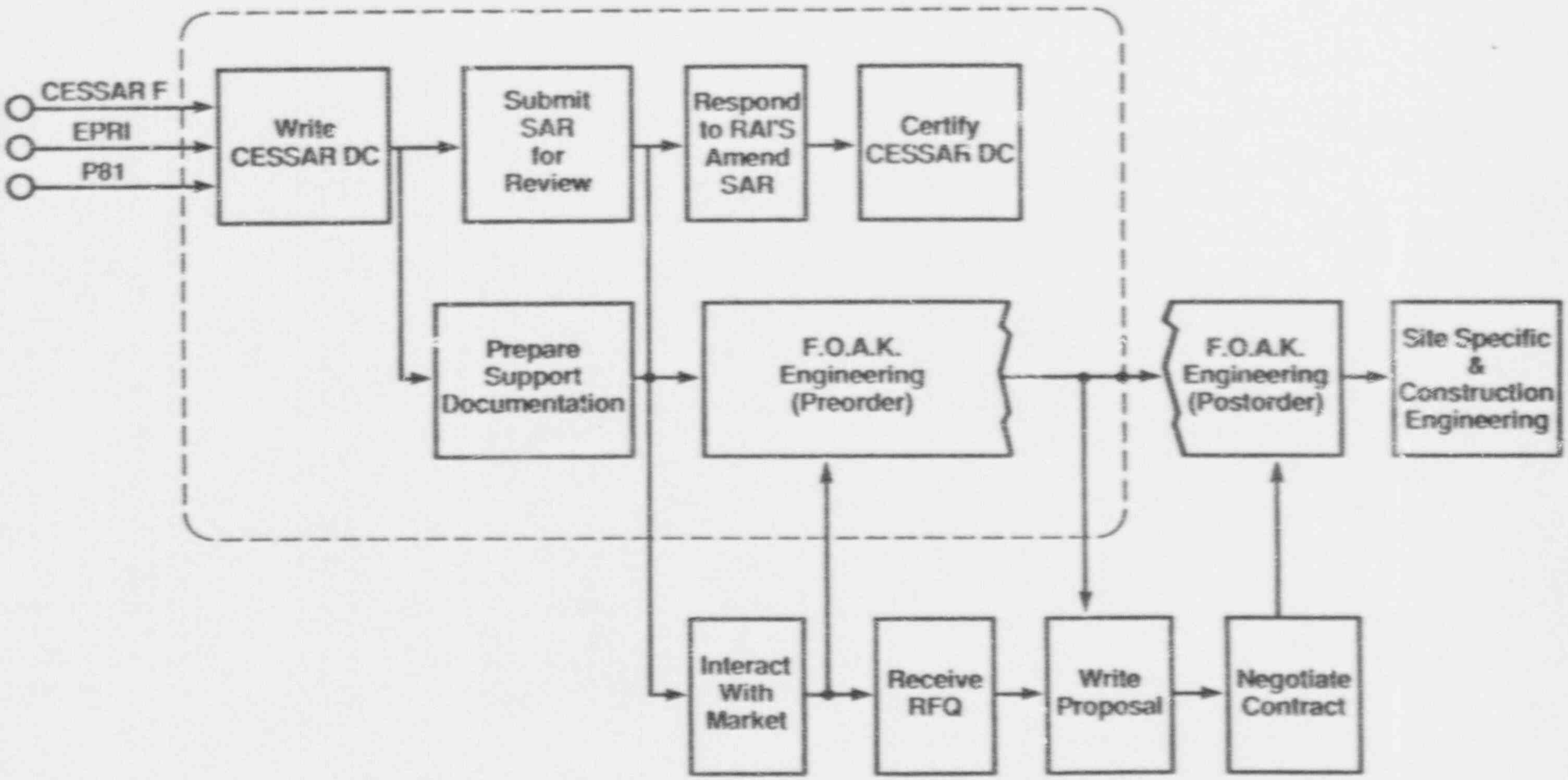
Model for System 80+ Design Development



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UTILITY PARTICIPATION IN SYSTEM 80+ DESIGN DEVELOPMENT

- EPRI Utility Requirements Document
- Feedback from System 80 operations (Palo Verde)
- System 80+ Executive Advisory Committee
- Duke Power Company Experience (thru Duke Engineering & Services)
- Design Reviews/Work Shops, eg:
  - Plant Arrangements
  - Operational Support Information
  - CESSAR-DC Integrated Review
  - Control Room Human Factors

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SYSTEM 80+ EXECUTIVE ADVISORY COMMITTEE MEMBERS

MEMBER	AFFILIATION
John Board PWR Project Group	Nuclear Electric, plc
William F. Conway Exec. Vice President	Arizona Public Service Co.
William Counsil Vice Chairman	TU Electric
Jerome Goldberg President, Nuclear Division	Florida Power & Light
Martin Hall Senior Manager	British Nuclear Fuels, plc
Donald Mazur Managing Director	Washington Public Power Supply System
Christian H. Poindexter Vice Chairman	Baltimore Gas & Electric Co.
Harold B. Ray Vice President, Safety & Licensing	Southern California Edison Co.
Cordell Reed Vice President	Commonwealth Edison Company
Richard B. Priory Senior Vice President Generation & Information Services	Duke Power Company
Huh, Sook General Manager Nuclear Power Construction	Korea Electric Power Corp.
Mark Sanford Manager New Construction	Tennessee Valley Authority

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APPROACH FOR DEVELOPING SYSTEM 80+ STANDARD DESIGN

- Start with Current System 80 (CESSAR-F) and Duke Power's Cherokee/Perkins BOP

- Consider Changes Due to

  - EPRI ALWR Requirements
  - NRC Mandated Changes (Primarily to Address Severe Accidents)
  - C-E Desired Changes (as a Result of Operational Feedback)

- Assess Impact of Changes on

  - Safety
  - Performance
  - Operability
  - Maintainability
  - Cost

- Incorporate Changes Using

  - PRA
  - Cost/Benefit

- Revise Standard Design (System 80+/CESSAR-DC)

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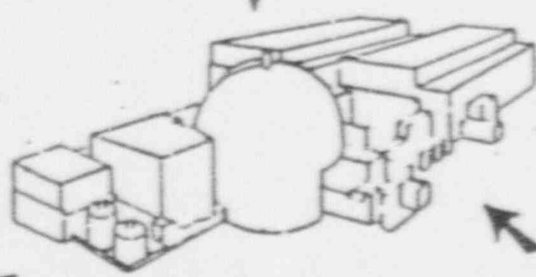
INTEGRATED DESIGN APPROACH

Maintenance

Safety/Non-Safety Separation

PRA



Security

Human Factors

USI Resolutions

Modularization

ALARA

SYSTEM 80+ SAFETY GOALS

- Core Damage Frequency $< 10^{-5}$ Events/Yr
- Severe Accident Release $< 10^{-6}$ Events/Yr for Occurence of Doses Greater than 25 Rem at Site Boundary

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SYSTEM 80+
IMPROVED OPERATION & MAINTENANCE

- 60-Year Design Life
- Availability >87%
- Outage Time <30 Days/Yr, Including Refueling Time, <50 Days/Fuel Cycle
- Unplanned Trips <1/Yr
- Personnel Exposure <100 Man-Rem/Yr
- Improvement Maintainability:
    - Self-Testing Features
    - Reduced ISI
    - Increased Work Space
    - Separation of Safety/Non-Safety Systems

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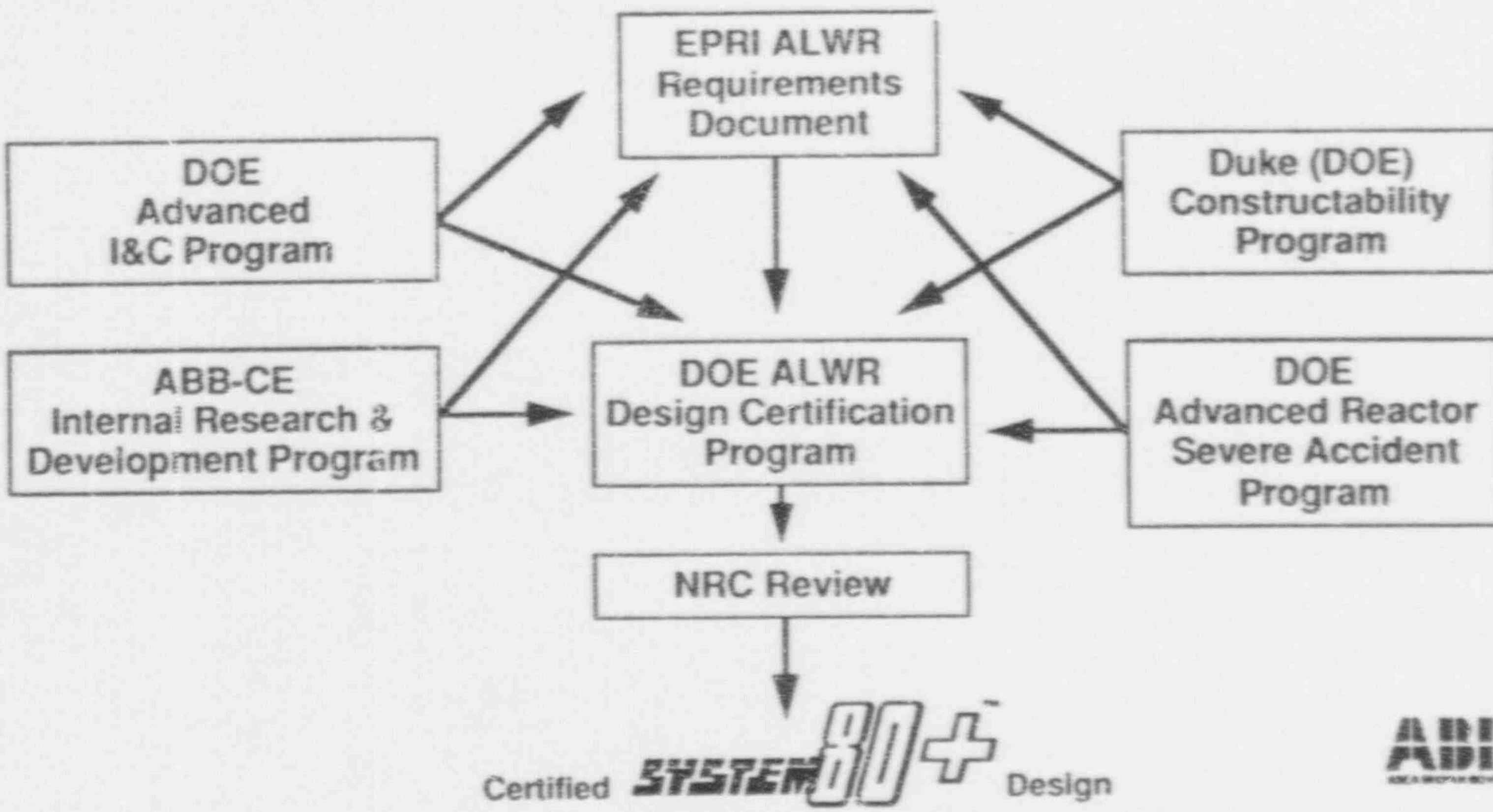
ABB C-E Evolutionary ALWR Program



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

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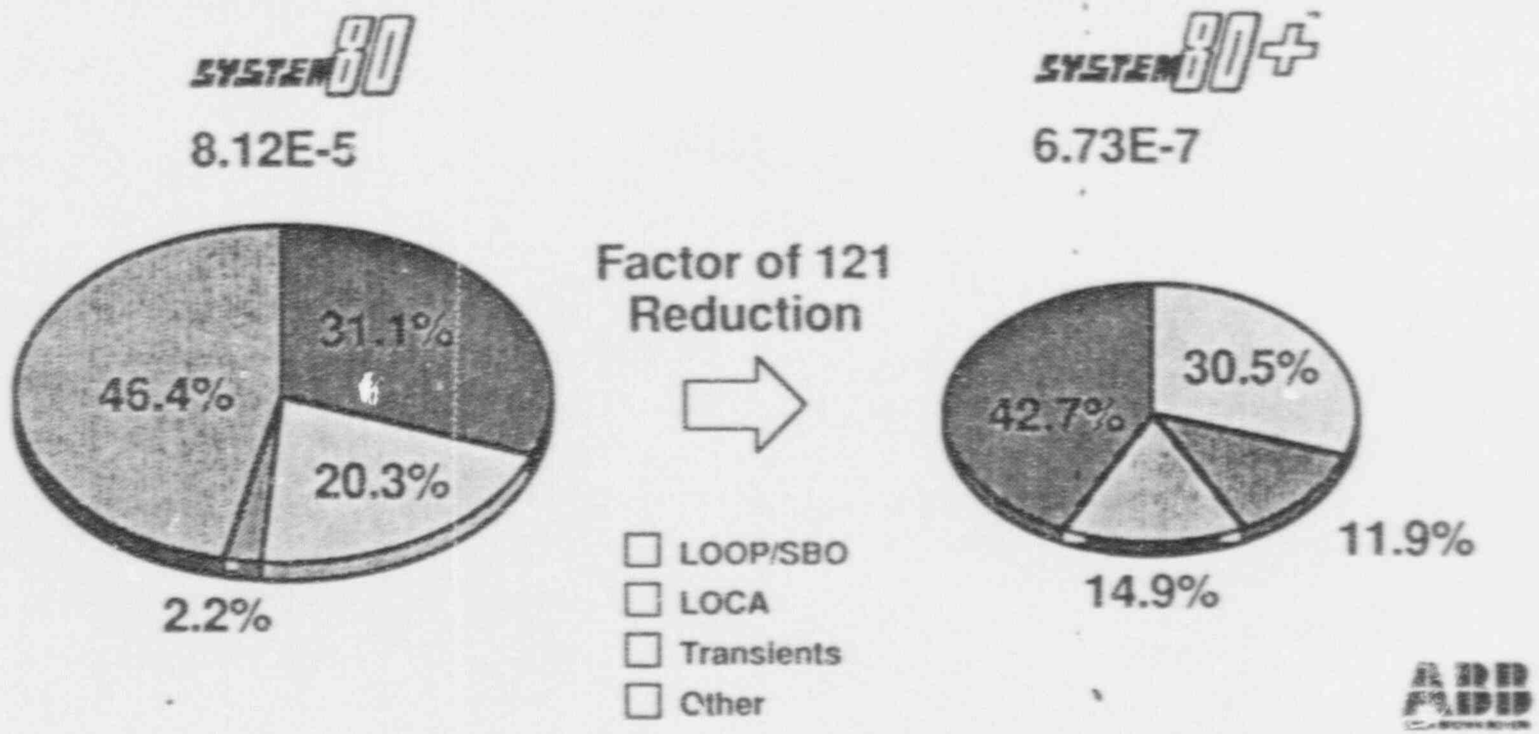
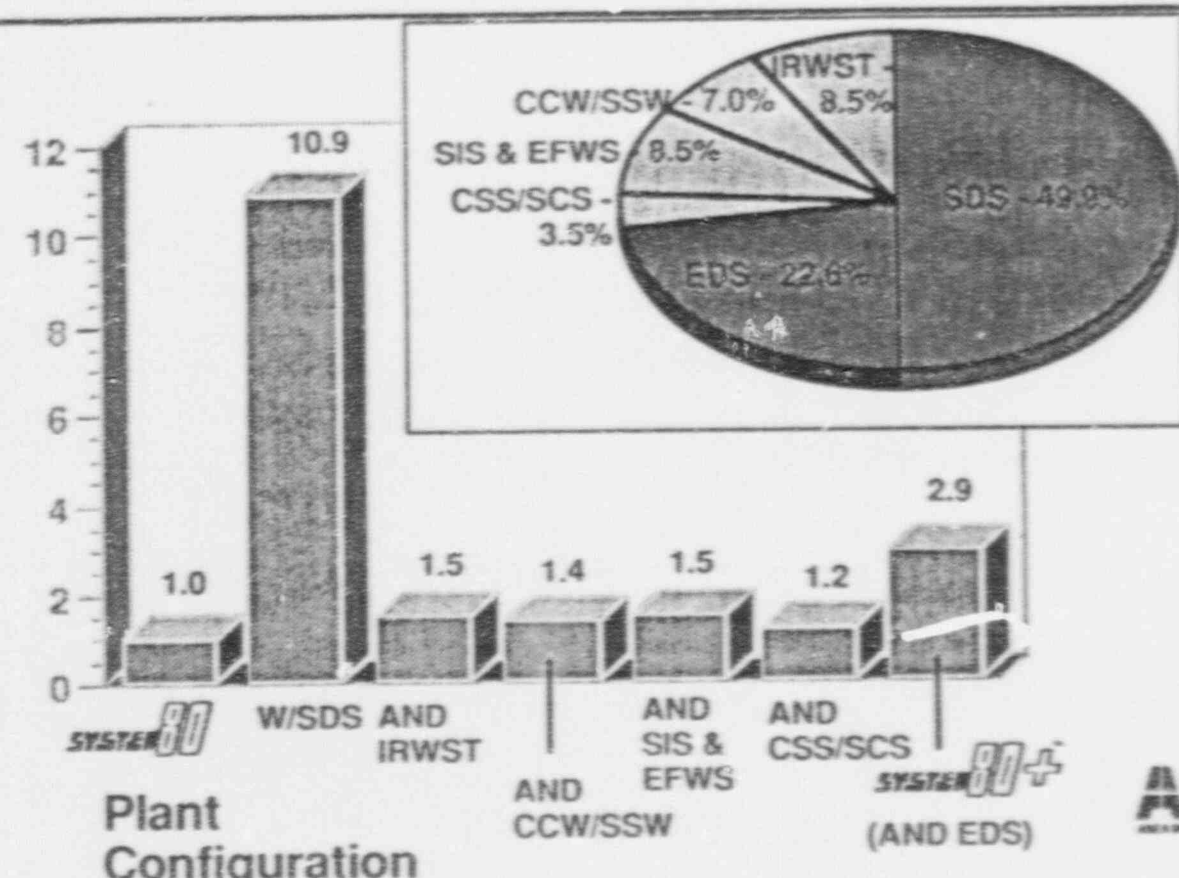
Dominant Contributors to Severe Accident Risk
(Core Damage Frequency, Internal Events)



Impact of System 80+ Design Features on Severe Accident Risk (Core Damage Frequency, Internal Events)



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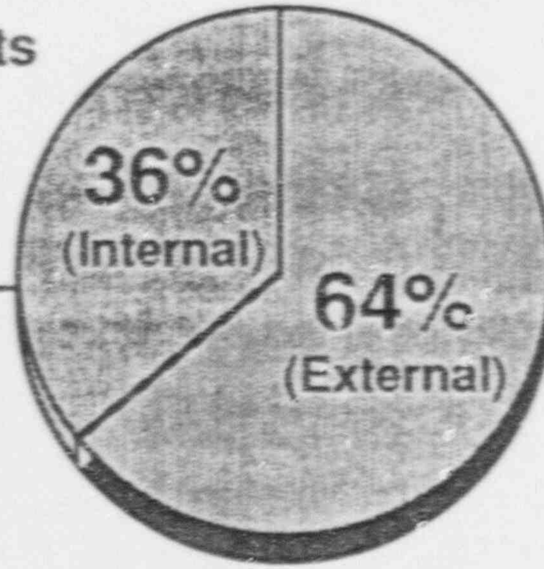
Total Core Damage Frequency

Internal Events	6.73×10^{-7}
Tornadoes	1.21×10^{-8}
Earthquakes	1.19×10^{-6}
Total	1.87×10^{-6}



Other external events were judged to have inconsequential impact due to specific System 80+™ design features.

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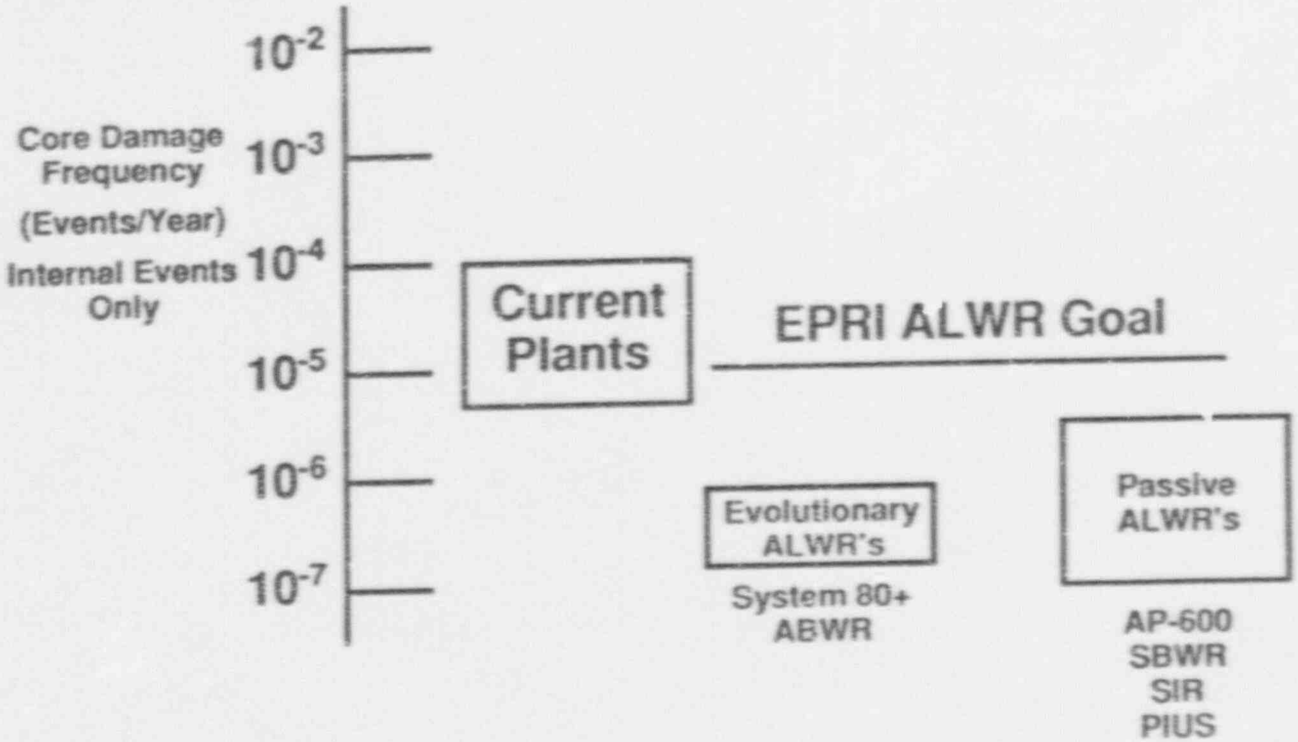
Safety Levels



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Stan Ritterbusch

Nuclear Systems Licensing

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ISSUES FOR DISCUSSION

- Human Factors Engineering
- I&C Software Reliability
- Reactor Coolant Pump Seal Coolability
- Shutdown Risk (Operational Guidance, Deterministic Analysis, PRA)
- Seismic and Structural Design
- Piping Design
- Leak Before Break
- Safety Analysis Fuel Failure Criterion and LOOP Time Delay
- Severe Accident Design and Analysis

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OTHER POTENTIAL ISSUES

- Reliability Assurance Program
- Interface Requirements Summary
- Standard Review Plan Deviations Summary
- Fire Hazards Analysis
- Severe Accident Mitigation Design Alternatives
- Inspections, Tests, Analysis, and Acceptance Criteria
- Operational Support Information
- Probabilistic Risk Assessment
  - Fire Methodology
  - Flood Methodology
  - MAAP Analysis Assumptions and Methodology
- Intersystem LOCA
- Shielding Analysis Methodology
- Inservice Inspection and Testing
- Safety Analysis Methodology
  - Treatment of Single Failures
  - Crediting Redundant Control Grade Equipment
  - Source Term Revisions
  - Anticipated Transients Without Scram
  - Analysis with Emergency Procedures

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HUMAN FACTORS ENGINEERING

- Major Review Topics:

  - Criteria and Process for HFE review
  - Design Acceptance Criteria

- Progress to Date:

  - Meetings to discuss HFE review
  - Agreement with staff on the approach for: 1) A HFE program plan description; 2) Revisions to RAI responses submitted previously
  - Meeting to discuss Design Acceptance Criteria

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- Design Acceptance Criteria Submittal:

  - Final design of remaining control room panels
  - Demonstration of acceptability of man-machine interface from a human performance standpoint
  - DAC not proposed for makeup of the design team or the design process

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I&C SOFTWARE RELIABILITY

- Increased use of software in protection systems perceived to have an adverse effect on reliability
- Major Review Topics:
  - Extent of V&V prior to D.C.
  - Common mode failures in software
  - V&V of commercial software
  - Degree of independence of V&V reviews
- Approach to Resolution:
  - Field proven software products
  - V&V and Configuration Management Programs
  - Maximum level of standardization
  - Minimum level of diversity
  - Deterministic software design
  - Functional segmentation of software
  - Sabotage protection

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REACTOR COOLANT PUMP SEAL COOLABILITY

- System 80+ includes two independent, continuously operating system for RCP seal cooling (CCW and Seal Injection)
- The RAIs indicate potential staff desire for safety grade independent cooling system (beyond draft Regulatory Guide)
- The seal injection system uses a CVCS pump which is powered by the alternate AC source.
- Response to RAIs will be provided next month

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SHUTDOWN RISK

- Major topics:
  - Procedures
  - Technical Specification Improvement
  - Mid-Loop Operation
  - Loss of Decay Heat Removal Capability
  - Primary/Secondary Containment Capability and Source Team
  - Rapid Boron Dilution
  - Fire Protection
  - Instrumentation
  - ECCS Recirculation Capability
  - Effect of PWR Upper Internals
  - Fuel Handling and Heavy Loads
  - Potential for Draining the Reactor Vessel
  - CESSAR-DC Chapter 15 - Non-Loca Events/Loca Dose
  - CESSAR-DC Chapter 6 Loss of Coolant Accidents
  - CESSAR-DC Chapter 6 - Containment Analysis
  - Probabilistic Risk Assessment

- Final Report Submittal Date:

  Preliminary - May, 1992
  Final - August, 1992

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SEISMIC AND STRUCTURAL DESIGN

- Major Review Topics:
  - Seismic envelope
  - Separation of OBE from SSE
  - Enough detail so SSI/Seismic models not affected by design completion
  - Response Spectra (R.G. 1.60)
  - Containment Buckling
  - Margin in Containment Design Pressure
- Status:
  - Responses to 170 RAIs in January and February
  - Meeting after staff reviews responses

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PIPING DESIGN

- STAFF POSITION

  - Provide Detailed Piping Layout and Plant Arrangement Drawings

  - Provide Data Required to Select Postulated Break Locations:

    -- Stress Intensities
    -- Cumulative Usage Factors
    -- Calculated Stress Ranges

  - Provide Locations of Postulated Pipe Rupture:

    -- Longitudinal and Circumferential Break Locations
    -- Restraint Locations
    -- Structural Barriers

- ABB-CE PROPOSED APPROACH

  - Level of Detail Requested Requires Specific Components and Resulting Final Design

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PIPING DESIGN (Cont'd)

- Provide Design Criteria, Design Basis and Acceptance Criteria to Allow for Detailed Design and Analysis in CESSAR-DC

- Prepare a Distribution Systems Design Guide:

  -- Systems/Equipment Interfaces
  -- Civil/Structural Interfaces
  -- Routing
  -- Leak Before Break
  -- Postulated Pipe Rupture

- Prepare Set of Sample Piping Layouts and Analyses:

  -- Layouts for Surge Line, Main Feedwater Line, and Main Steam Line
  -- Pipe Break Analyses for Main Feedwater
  -- LBB Analyses for Surge Line

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PIPING DESIGN (Cont'd)

- STATUS
  - Responses to RAIs Relating to Piping in Final Review
  - Detailed Outline for Distribution Systems Guide in Preparation
  - Follow-Up Meeting with Staff Proposal for Late February

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LEAK BEFORE BREAK

- STAFF POSITION
  - LBB Analysis for Specific Piping Systems Must be Reviewed and Approved by the Staff Before Dynamic Effects can be Excluded from the Design Basis
  - Analysis Should Be Based on Specific Plant Data
  - LBB Procedure not Pre-Approved by the Staff

- ABB-CE PROPOSED APPROACH
  - Apply LBB to Five Piping Systems:
    - RCS Main Loop
    - Surge Line
    - Safety Injection
    - Shutdown Cooling
    - Main Steam

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LEAK BEFORE BREAK (Cont'd)

- Provide Acceptance Criteria and Methodology in CESSAR-DC
- Provide Guidelines in Distribution System Guide
- Perform Detailed Sample Calculation for Surge Line
- STATUS
  - Responses to RAIs Relating to LBB in Final Review
  - Follow-Up Meeting with Staff Proposed for Late February

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SAFETY ANALYSIS FUEL FAILURE CRITERION

- CESSAR-F Chapter 15 safety analyses used the statistical convolution method to calculate fuel failure for seized rotor/sheared shaft and CEA ejection

- Other CESSAR-F events assumed fuel failure for all fuel pins with minimum DNBR below the SAFDL

- CESSAR-DC used statistical convolution

  - Fuel pins with DNBR less than the SAFDL do not necessarily experience DNB

  - Probability of experiencing DNB is a function of DNBR

  - All pins in DNB assumed to fail

- Staff requested the more conservative method; all pins with DNBR less than the SAFDL in DNB and fail

- Statistical convolution can be applied to all events

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SAFETY ANALYSIS LOSS OF OFFSITE POWER TIME DELAY

- Chapter 15 safety analyses used a 3 second time delay for loss of offsite power after turbine trip: previously used for System 80
- Staff RAI requested use of no time delay: future plants expected to be safer than current generation
- Response to RAI submitted November 1991
  - Time delay used conservative grid characteristics bounding contiguous 48 states
  - Plant generating capacity with respect to grid site specific

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CONTAINMENT PERFORMANCE

- Probabilistic Approach Originally Proposed:
  - 10% containment conditional unreliability (Including Seismic and Overpressure challenges)
- Uncertainties:
  - PRA methodology
  - Seismic Hazard Data
- Deterministic Approach Being Considered by Staff
  - Seismic Margins Assessment
  - Containment Overpressure Calculation
  - Additional interaction with staff needed to assess viability of the deterministic approach for containment overpressure

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HYDROGEN CONTROL

- Control-Grade System:
  - Two trains of igniters
  - Powered from emergency diesels, batteries, alternate AC source
- Igniters located by engineering judgement; distributed globally (not detailed analysis of hydrogen behavior)

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HIGH PRESSURE CORE-MELT EJECTION

- Safety Depressurization System
- Reactor Cavity Open to Containment Atmosphere
- Cavity Design for Debris De-Entrainment
  - Debris Chamber
  - Labyrinth Vent Path
- Adequacy of Design Based on Judgment

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CORE-CONCRETE INTERACTION

- Larger cavity floor size to enhance debri spreading
- Manually-controlled cavity flooding
- Five feet sacrificial concrete
- Large containment volume
- Adequacy of design based on engineering judgement, not detailed calculations or complex experiments

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SEVERE ACCIDENT METHODOLOGY

- MAAP analyses (Best-Estimate):
  - Probabilistic Risk Assessment
  - Containment Overpressure
- Uncertainty in severe accident phenomena requires use of judgement in evaluating assumptions and methods

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