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

June 3, 1992

Docket No. 52-002

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

PROJECT: CE System 80+

SUBJECT: SUMMARY OF THE NUCLEAR REGULATORY COMMISSION/ABB-CE MEETING ON SYSTEM 80+ PROBABILISTIC RISK ASSESSMENT (PRA) HELD ON APRIL 28, 1992

On April 28, 1992, a meeting was held between the Nuclear Regulatory Commission (NRC) and ABB-CE staff at the ABB-CE offices in Windsor, Connecticut. Staff from the Risk Applications Branch/NRR and the Brookhaven National Laboratory (BNL) participated. The purpose of the meeting was: (1) to discuss the PRA update; (2) to discuss ABB-CE's answers to requests for additional information (RAIs); and (3) to discuss NRC's requirement for a new safety evaluation report (SER) format that emphasizes PRA-based insights about the design and on applications and uses of the PRA during design certification and beyond. Discussion of post-certification uses of PRA with ABB-CE revealed that ABB-CE is currently involved in a concerted industry "Action Management" effort aimed at providing guidance to operating plants and future combined licensing (COL) applicants in the areas of severe accident, PRA modeling, and PRA applications. Among the participants are the Nuclear Power Oversight Committee (NPOC), the Nuclear Management and Resources Council (NUMARC), the Institute of Nuclear Power Operations (INPO), and the Electric Power Research Institute (EPRI). The list of attendees and the ABB-CE presentation are enclosed.

The topics and a summary of the discussion are listed below:

- System 80+ PRA Update Schedule: ABB-CE staff presented a list of PRA tasks, their expected completion dates, and the relationship of these tasks to NRC RAIs. According to ABB-CE staff, draft reports and/or updates of previous write-ups will be produced in the period between May 31, 1992, and September 30, 1992. The updates will include Human Reliability Analysis, uses of PRA, level 2 phenomenology issues, and a plan for long-term maintenance of the System 80+ PRA. A complete updated PRA will be available by July 31, 1993, with the exact date depending on seismic PRA requirements. The NRC staff mentioned that, given this schedule, it will be too late to include any information from these PRA task reports in the draft safety evaluation report (DSER) (due June 23, 1992). In addition, the System 80+ updated PRA should be completed in time to meet the final safety evaluation report (FSER) deadline (May 26, 1993) for input to NRR Projects.

- Two Part Safety Evaluation Report (SER) Objectives: The NRC staff explained the process that is being adopted for use in reviewing advanced light water reactor (ALWR) PRAs and provided ABB-CE a copy of the draft outline for the ABWR FSER. The staff explained that a similar format will be followed in the forthcoming System 80+ PRA DSER. The PRA review, in addition to emphasizing the PRA quality and completeness, also emphasizes the objectives that are behind the requirement of a

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design certification PRA. These objectives are: (1) use of the PRA to develop a deeper understanding and insights about the design strengths and relative weaknesses; and (2) use of the PRA to support pre- and post-certification regulatory activities.

- <u>Guidance for COL Applicants</u>: The staff indicated that, in addition to the design certification PRA insights, ABB-CE should provide guidance to the COL applicant on how to use the information and results of the certification PRA. In particular, the certification PRA could be used to demonstrate that: (1) data and features associated with the selected site are enveloped by the external events analyses and (2) design details, not explicitly considered during certification, do not compromise the design-safety insights drawn from the PRA. ABB-CE indicated that they will be providing a draft program plan for long-term maintenance of the PRA by July 31, 1992. ABB-CE staff also mentioned that there is currently underway a concerted industry "action management" effort aimed at coordinating industry activities on severe accident guidelines and NRC interface for advanced reactor designs. This effort will address post-certification issues (including issues related to the development and maintenance of a "living PRA") and will provide guidance to the COL applicant.

- <u>Importance Analysis (IA) and Use of Uncertainty Analysis Results</u>: The NRC staff discussed the need for a more detailed IA in the PRA. The IA, as the first step in most risk-based applications, should provide priorities and focus for certification and follow-up regulatory, verification and operational activities (e.g., develop ITAAC, RAP, Technical Specifications (TS); aid in the Control Room design verification; etc.). This can be achieved by using appropriate and complementary "risk importance measures" in ranking systems, structures, and components (SSCs). The two general objectives of an IA are: (1) risk reduction by identifying the dominant contributors to "present" design risk (part of the design process); and (2) safety or reliability assurance by identifying the dominant contributors to maintaining the "present" risk level (to assure that risk does not increase and is as low as the PRA indicates it is). Both of these general objectives should be considered when choosing specific "importance measures" to rank SSCs. In addition, the PRA should provide some guidance on the use of IA insights in conjunction with insights from the uncertainty analysis to plan activities aimed at reducing or maintaining risk levels.

- <u>Response to Level II RAIs and Closure of Severe Accidents</u>: ABB-CE is preparing a severe accident section to be included in the FSAR discussing System 80+ mitigating systems and severe accident phenomenology. Plant damage states definition is being modified and containment event trees (CETs) are being restructured and will be requantified taking into account uncertainties. A reactor cavity pressurization analysis will be performed. Input from these analyses will not be available in time to be included in the DSER.

- Level II Uncertainty, Sensitivity, and Parametric Analyses: ABB-CE plans and the staff expectations for the level II analyses were discussed. There was general agreement that a full-scope NUREG-1150 type uncertainty analysis will not be performed. Rather, the approach for addressing uncertainties will be to represent, within the containment event trees, all credible issue outcomes, and to support the degrees of belief assigned to each outcome using existing data/analyses to the extent possible. ABB-CE indicated that they will not be relying exclusively on MAAP for this purpose, but will try to utilize other calculational tools, and expert opinion information developed in support of NUREG-1150. Additional sensitivity analyses will be performed using MAAP. The related EPRI guidance document will be generally followed, but ABB-CE plans to consider parameter ranges and issues beyond MAAP's limitations identified by the staff. ABB-CE indicated that they will be performing a reactor cavity pressurization analysis as part of a phenomenology section/chapter to be submitted on July 31, 1992. The staff committed to provide available information on a recent experiment in the beta test facility, which may have a bearing on this analysis.

- Distribution of Containment Failure Probability Versus Pressure: The applicability of containment strength uncertainty distributions for NUREG-1150 plants to the System 80+ design (RAI 722.17) was discussed. ABB-CE staff explained that this curve was used only for "early containment failure" challenges. Containment performance is not very sensitive to this curve since the pressure loads for early containment challenges are well below the estimated ultimate pressure capacity, and in fact are in the range of the service level "C" value for the System 80+ containment. For "late containment failure" challenges, it was conservatively assumed that containment failure would occur at the estimated containment ultimate pressure capacity (even though the pressure distribution curve would indicate a 0.5 probability at this value). ABB-CE indicated that a plant specific ultimate strength analysis, including hatches, will be performed by the COL applicant prior to plant operation.

- Evaluation of Design Alternatives: ABB-CE summarized the results of their assessment of several design alternatives including Severe Accident Mitigation Design Alternatives (SAMDAs). The PRA was used to perform a bounding evaluation of the costs and benefits associated with 11 design alternatives, involving both front end and back end (containment) features. The results indicated that none of these design alternatives are cost-effective. This analysis has been submitted to NRC.

- Agreement With EPRI's Key Assumptions and Groundrules (KAGs): ABB-CE staff mentioned that they intend to include a list of agreements and disagreements with EPRI's KAGs in the updated PRA.

- Human Reliability Analysis (HRA) Modeling and NUPLEX 80+ Control Room Design: ABB-CE staff stated that although the Control Room for the System 80+ is still being designed, it will not affect the completion of the PRA. The reason is that this new control room design is an evolutionary design (the difference is that it provides more information to the operator than the old System 80 design). In validating control room display, ABB-CE staff are using a simulator for operator training that envelopes operator actions modeled in the PRA. NRC staff indicated that because the detailed Control Room design will not be complete at the time the PRA/HRA is completed, ABB-CE will need to define a process by which the vendor/COL-applicant will reassess the adequacy of the HRA after the design details and the Control Room prototype have been developed. ABB-CE indicated that this issue can be addressed in the ABB-CE program plan for long-term maintenance of the PRA.

- Shutdown Risk Evaluation: Results from the quantitative evaluation of shutdown risk will be available in July 1992. Issues that are currently being investigated are: (a) alternative means of decay heat removal (e.g., bring a steam generator on line); and (b) ways that unborated water could get in the RCS (rapid boron dilution). Qualitative analyses of fires, flooding and spills during shutdown have been completed. ABB-CE intends to use the results of the shutdown PRA to produce guidance on developing administrative controls, tech specs, and procedures during shutdown.

- Fires and Internal Floods: The "FIVE" methodology, adapted to accommodate lack of "as-built" information, will be used for quantitative fire analysis of areas that were not screened out by the qualitative analysis (Appendix R requirements). ABB-CE expects to use similar methodology to evaluate internal floods. NRC staff stated that they will review these methodologies as they become available.

- External Events Analyses: NRC staff explained the current NRC position on submittal of beyond design basis external events analyses by ALWR vendors and COL applicants. Specifically, the staff does not intend to require a full external events PRA. Instead, ALWR vendors could choose to perform only a PRA-based margins analysis which provides insights on features that limit the plant capability to withstand external events challenges. ABB-CE staff stated that they intend to update their seismic PRA models to address NRC's main comments provided with the RAIs and use them to gain insights on the seismic plant capability.

Original Signed By:

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Enclosures:
1. List of Attendees
2. Presentation Slides

cc w/enclosures:
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Enclosure 1

ABB-CE/NRC MEETING ON SYSTEM 80+ PRA

APRIL 28, 1992

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ABB-CE/ NRC MEETING
ON
SYSTEM 80+ PRA UPDATE

APRIL 28, 1992

Enclosure 2

CE/NRC MEETING ON SYSTEM 80+ PRA
APRIL 28, 1992

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TOPICS FOR DISCUSSION

- SYSTEM 80+ PRA UPDATE SCHEDULE
- TWO PART DSER OBJECTIVES
- GUIDANCE FOR COL APPLICANTS
- IMPORTANCE ANALYSIS AND USE OF UNCERTAINTY ANALYSIS RESULTS
- LEVEL 2 UNCERTAINTY, SENSITIVITY AND PARAMETRIC ANALYSES
- CONTAINMENT FAILURE PROBABILITY DISTRIBUTION
- NEPA/SAMDA
- AGREEMENT AND DISAGREEMENT WITH EPRI KAG
- HUMAN RELIABILITY ANALYSIS MODELING AND NUPLEX 80+ CONTROL ROOM DESIGN
  - PRA Input to DAC
- SHUTDOWN RISK EVALUATION
- LATEST NRC STAFF POSITION ON EXTERNAL EVENTS ANALYSES
- FIRE AND FLOOD RISK ASSESSMENT
- PRA APPLICATIONS
- WALK THROUGH RAIs

SYSTEM 80+ PRA UPDATE

- SYSTEM 80+ PRA IS A DYNAMIC PRA
  - Will be Maintained and Updated Throughout Plant Life Cycle
- SYSTEM 80+ PRA CURRENTLY BEING UPDATED
  - Address NRC RAIs
  - Incorporate System Design Changes
- SYSTEM 80+ PRA WILL BE UPDATED FOR FIRST OF A KIND ENGENEERING
  - Confirmatory Review of Models and Analyses
  - Greater Design Detail
- SYSTEM 80+ PRA WILL BE UPDATED FOR THE AS BUILT PLANT
- SYSTEM 80+ PRA INSIGHTS WILL BE FED BACK TO DESIGNERS AND OPERATORS

TASKS FOR SYSTEM 80+ PRA UPDATE	
PRA TASK	FINISH DATE
2.1. LEVEL 1 PRA ANALYSES	
1. Convert System Models to CAFTA	4/17/92
2. Update System Models and Documentation	6/29/92
3. Review System Fault Tree Models	6/26/92
4. Revise Human Reliability Analysis	4/30/92
5. Revise Event Trees per NRC comments	5/15/92
6. Review Event Trees	5/15/92
7. Verify Success Criteria for Selected Sequences	6/26/92
8. Update Reliability Database with CE Operating Experience data	6/26/92
9. Requantify Core Damage Sequences	8/28/92
10. Identify and Evaluate Sensitivity Issues	8/28/92
11. Update and Expand PRA Report (Sections 2, 3, 4, 5, 6 and 8)	9/30/92
2.2. EXTERNAL EVENTS ANALYSES	
1. Update Seismic PRA	4/30/93
2. Update Tornado Strike Analysis	8/28/92
3. Incorporate Fire Hazards Evaluation	5/15/92
4. Incorporate Flood Hazards Evaluation	7/17/92
5. Update PRA Report (Section 7)	5/31/93
2.3. LEVEL 2 ANALYSES	
1. Revise Plant Damage State Definitions	5/15/92
2. MAAP Parameter File Update	4/10/92
3. Develop Severe Accident Phenomenology Section	7/31/92
4. Restructure CETs and SLMs	7/31/92
5. MAAP Analyses	9/26/92
6. Preliminary Quantification of CET	9/26/92
7. Level 1/Level 2 Linking	5/28/93
8. Uncertainty Analysis	6/25/92
9. Update Level 2 Documentation	6/25/93
2.4. LEVEL 3 ANALYSES	
1. Update and Expand Risk Calculations	6/25/93
2. Update Level 3 Documentation	7/30/93
2.5. PRESENTATION OF PRA RESULTS AND CONCLUSIONS	7/30/93

PRA UPDATE TASKS VERSUS NRC RAIs		
TASK NO.	TASK DESCRIPTION	RELATED NRC RAIs
2.1.1	Convert System Models to CAFTA	(No Specific RAIs)
2.1.2	Update System models and Documentation	720.9, 720.31, 720.36, 720.37, 720.65, 720.66, 720.67, 720.68, 720.72
2.1.3	Review System Fault Tree Models	(No Specific RAIs)
2.1.4	Revise Human Reliability Analysis	720.4, 720.9, 720.28, 720.61, 721.1, 721.2, 721.3, 721.4, 72[illegible].5, 721.6, 721.7, 721.8, 721.9, 721.10, 721.11, 7[illegible].12, 712.13, 721.14, 721.15, 721.16, 721.17
2.1.5	Revise Event Trees per NRC Comments	720.3, 720.4, 720.8, 720.9, 720.10, 720.14, 720.18, 720.19, 720.24, 720.26, 720.28, 720.31, 720.98
2.1.6	Review Event Trees	(No Specific RAIs)
2.1.7	Verify Success Criteria for Selected Sequences	720.3, 720.10, 720.26, 720.31
2.1.8	Update Reliability Database	720.32, 720.37, 720.98
2.1.9	Requantify Core Damage Sequences	720.32
2.1.10	Identify and Evaluate Sensitivity Issues	720.97, 720.98, 722.90, 722.91
2.1.11	Update and Expand PRA Report	720.10, 720.32, 720.36, 720.53, 720.97, 720.98
2.2.1	Update Seismic PRA	720.49, 720.50, 720.54, 720.59, 720.61, 720.64, 720.81, 720.82, 720.89, 720.91, 720.97
2.2.2	Update Tornado Strike Analysis	720.53, 720.61, 720.97
2.2.3	Incorporate Fire Hazards Analysis	720.52, 720.97
2.2.4	Incorporate Flood Hazards Analysis	720.52, 720.76, 720.97
2.2.5	Update Chapter 7 of PRA Report	(No Specific RAIs)
2.3.1	Revise Plant Damage State Definitions	722.3, 722.14, 722.35, 722.36, 722.43
2.3.2	MAAP Parameter File Update	(No Specific RAIs)
2.3.3	Develop Severe Accident Phenomenology Section	722.3, 722.12, 722.14, 722.17, 722.19, 722.20, 722.21, 722.65, 722.66,

PRA UPDATE TASKS VERSUS NRC RAIs		
TASK NO.	TASK DESCRIPTION	RELATED NRC RAIs
2.3.4	Restructure CETs and SLMs	722.3, 722.12, 722.14, 722.17, 722.20, 722.21, 722.36, 722.42, 722.43, 722.48, 722.51, 722.56, 722.57, 722.58, 722.65, 722.66, 722.71, 722.73,
2.3.5	MAAP Analysis	722.3, 722.14, 722.36
2.3.6	Preliminary Quantification of CET	722.3, 722.12, 722.14, 722.17, 722.19, 722.20, 722.21, 722.36, 722.43, 722.51, 722.56, 722.57, 722.58, 722.65, 722.66, 722.71, 722.73
2.3.7	Level 1/Level 2 Linking	720.97, 722.3, 722.36, 722.43, 722.51, 722.56, 722.57, 722.58, 722.73
2.3.8	Uncertainty Analysis	720.97, 722.69, 722.71, 722.73, 722.90, 722.91
2.3.9	Update Level 2 Documentation	720.97, 722.25, 722.30, 722.31, 722.35, 722.56, 722.67, 722.78
2.4.1	Update and Expand Risk Calculations	720.97, 722.21, 722.88
2.4.2	Update Level 3 Documentation	720.97, 722.88
2.5	Presentation of PRA Results and Conclusions	722.89

SYSTEM 80+ PRA UPDATE SCHEDULE

DATE	TASK/PRODUCT
5/31/92	- Draft Writeup-Human Reliability Analysis Methodology and Description of Operator Actions - Draft Report on Use of PRA in the System 80+ Design Process
7/31/92	- Draft Writeup on Level 2 Phenomenology Issues as They Pertain to System 80+ - Draft Program Plan for Long-Term Maintenance of the System 80+ PRA
9/30/92	- Draft Update of the Level 1 PRA Analyses for Internal Events
7/31/93	- Completed Update of System 80+ PRA (Schedule Dependent on Seismic PRA Requirements)

SYSTEM 80+ SHUTDOWN PRA

METHOD: FOLLOW EPRI REQUIREMENTS DOCUMENT
USING SIMPLIFIED EVENT TREE ANALYSIS
EMPHASIS ON LEVEL 1 ASPECTS
EVENT TREE EVALUATED FOR EACH PLANT OPERATION STATE

STATUS: REVIEWED EXISTING PRAs
ESTABLISHED TRANSIENT INITIATING EVENTS
PRESENTLY CONSTRUCTING EVENT TREES FOR EACH MODE

INITIATING EVENTS:

1) LOSS OF RHR, LOCA, STATION BLACKOUT, REACTIVITY EVENTS
2) FREQUENCIES BASED ON BNL LETTER REPORT 11/13/91
3) FREQUENCY MODIFIED FOR C-E DATA (RCP SEAL, OUTAGE SCHEDULE)

FEATURES FOR SHUTDOWN RISK REDUCTION

1) IMPROVED INSTRUMENTATION IN SHUTDOWN MODE FOR TEMPERATURE AND COOLANT LEVEL
2) IPSO PANEL WILL GIVE EQUIPMENT STATUS IN SHUTDOWN MODES
3) CONTAINMENT SPRAY PUMPS CAN BE USED TO REMOVE RESIDUAL HEAT
4) 4 TRAIN SI PUMPS COULD BE USED FOR FEED AND BLEED
5) ALTERNATE AC GAS TURBINE FOR ADDITIONAL AC REDUNDANCY
6) HOT LEG BETTER DESIGNED FOR MID LOOP OPERATION
7) SEAL TABLE IS AT UPPER HEAD LEVEL
8) NO TEMPORARY SEALS AT SEAL TABLE
9) CONTAINMENT HATCH MORE QUICKLY CLOSED
10) RHR SYSTEM RELIEF VALVES SIZED FOR SI FLOW

SHUTDOWN RISK REDUCTION THRU OPERATION

1) LESSONS LEARNED FROM PAST EVENTS AND PRAs INCORPORATED

2) TECHNICAL SPECIFICATIONS ADDRESS SHUTDOWN MODES

3) MID LOOP OPERATION MINIMIZED

4) EMERGENCY PROCEDURES FOR SHUTDOWN MODES WILL EXIST

5) RESTRICTIONS PLACED ON INSTRUMENTATIONS, DG MAINTENANCE, CONTAINMENT CLOSURE

SYSTEM 80+ SHUTDOWN RISK EVALUATION STATUS

- O PROCEEDING TO LOOK AT LOSS OF DHR
  - DIFFERENT MODES AND PLANT CONFIGURATIONS
  - OPERATOR ACTIONS
  - OFFSITE DOSES
  - POTENTIAL FOR FUEL FAILURE
  - OVERPRESSURIZATION
- O RAPID BORON DILUTION
  - LOOKING AT SOURCES OF UNBORATED WATER
  - PRELIMINARY REACTIVITY CALCULATIONS
- O FIRE PROTECTION: COMPLETE
- O ECCS RECIRCULATION CAPABILITY: COMPLETE
- O EFFECT OF PWR UPPER INTERNALS
  - PRELIMINARY RESULTS INDICATE A LACK OF CIRCULATION BETWEEN REFUELING POOL AND CORE
  - ALTERNATE DHR PATH MAY BE AVAILABLE
- O FUEL HANDLING AND HEAVY LOADS: COMPLETE

SYSTEM 80+ SHUTDOWN RISK EVALUATION STATUS

- O FLOODING AND SPILLS: COMPLETE
- O PRA: E. R. SIEGMANN
- O CESSAR-DC CHAPTER 15:
  - EVALUATIONS FOR ACCEPTABLE CONSEQUENCES OF MANY EVENTS: COMPLETE
  - ANALYSES TO FOLLOW IN JUNE
- O CESSAR-DC CHAPTER 6 LOCA ANALYSES
  - MOST LOCAs CAN BE CONSIDERED BOUNDED BY CESSAR-DC
  - OTHER LOCAs UNDERGOING FURTHER EVALUATION
- O CESSAR-DC CHAPTER 6 CONTAINMENT ANALYSES/ EVALUATIONS: COMPLETE
  - BOUNDED BY CESSAR-DC
- O DESIGN FEATURES REDUCING STRESS ON THE OPERATOR, IMPROVING OUTAGE PLANNING AND TRAINING: COMPLETE
- O PROCEDURES AND TECHNICAL SPECIFICATIONS: IN PROGRESS - HIGHLY DEPENDENT ON OTHER TASKS

SYSTEM 80+ SHUTDOWN RISK EVALUATION STATUS

- O REVIEWED MAJOR EVENTS AT U.S. PWR OCCURRING IN 1976 - 1990 TIME FRAME

  - TOTAL OF 122 EVENTS SELECTED

  - EVENTS GROUPED INTO 10 CATEGORIES

    1. LOSS OF SHUTDOWN COOLING
    2. LOSS OF ELECTRIC POWER
    3. LOSS OF REACTOR COOLANT
    4. CONTAINMENT INTEGRITY
    5. OVERPRESSURIZATION
    6. FLOODS AND SPILLS
    7. BORON AND REACTIVITY EVENTS
    8. FIRE PROTECTION
    9. HEAVY LOADS AND FUEL HANDLING
    10. MODE CHANGE EVENTS

- O INITIATORS REVIEWED AND COMPARED TO SYSTEM 80+ FEATURES

RESPONSE TO LEVEL 2 RAI'S

- O PREPARE A SEVERE ACCIDENT SECTION IN FSAR DISCUSSING SYSTEM 80+ MITIGATING SYSTEMS AND SEVERE ACCIDENT PHENOMENOLOGY
- O MODIFY PLANT DAMAGE STATES DEFINITION
- O RESTRUCTURE CETs AND SLMs
- O REQUANTIFY THE CET
- O PROPAGATE UNCERTAINTY THROUGH THE CET

SYSTEM 80 + SEVERE ACCIDENT PHENOMENOLOGY

- SYSTEM 80+ SEVERE ACCIDENT MITIGATING FEATURES AND DESIGN BASIS

- EARLY AND LATE CONTAINMENT FAILURE PROCESSES/MECHANISMS AND THEIR RELATIONSHIP TO THE SYSTEM 80+ DESIGN AND ROLE IN THE PRA

- EXPERIMENTAL BASIS SUPPORTING PRA POSITIONS

- REPRESENTATIVE RCS/CONTAINMENT SYSTEM 80+ RESPONSE CHARACTERISTICS

- PLANT SENSITIVITY TO MAJOR DESIGN FEATURES AND MODELING ASSUMPTIONS

Phase I Fire Screen - Qualitative Analysis

- Identify Fire Areas
- Identify Major Equipment Items in Each Area
- Determine Safe Shutdown Items
- Evaluate Qualitative Risk in Each Area Containing Safe Shutdown Equipment and Screen Areas Based On:
  - Complete Loss of Equipment in Area Due To Fire
  - Existence of Redundant Equipment or Functionality in Separate Area
- Identify Fire Areas Adjacent To Each Fire Area
- Identify Potential Fire Propagation Paths Between Areas and Provisions for Preventing Propagation of Fire Between Areas
- Evaluate Potential for Propagation of Fire to One or More Adjacent Areas to Develop Extended Fire Areas
- Evaluate Qualitative Risk in Each Extended Fire Area Based On:
  - Complete Loss of Equipment in All Areas of Each Extended Area
  - Existence of Redundant Equipment or Functionality in Separate Areas

Fire Hazards Analysis Methodology - System 80+

- Generally Follows EPRI "FIVE" Methodology
- "FIVE" Methodology Adapted to Accommodate Lack of "As-Built" Information
- General Design Information Qualitatively Analyzed to Evaluate Fire Risk and Screen Out Low Risk Areas
- Assumptions Used in Qualitative Analysis Will Be Documented For Use in First-of-a-Kind Engineering
- Qualitative Analysis Updated During FOAK to Incorporate Detailed Design Information to Verify Screening Remains Valid
- Quantitative Analysis of Fire Risk Will be Performed For Fire Areas Not Screened Out

Table 1 - Safe Shutdown Effect of Loss of Major Components In Each Fire Area

Fire Area	Description	Equipment In Area	Safe Shutdown Effect from Loss of All Equipment in Area to Fire
1	**Division I Channel A Vital Instrument & Equipment Room**	**Safe Shutdown Equipment** ESF-Component Control System Inverter Battery Charger Batteries Power Panel Boards & Distribution Center RTSG **Equipment Not Required for Safe Shutdown** Plant Protective Cabinets CPC Cabinets Multiplex Cabinets Recirculation Coolimg Units APC Manual By-pass Transfer Switches MCC A Incore/PAMI	In the event of loss of equipment in this area due to fire, safe shutdown can be reached and maintained by the use of Division II systems

Table 2 - Potential for Fire Propagation From Each Fire Area to Adjacent Fire Areas

Fire Area	Description	Adjacent Fire Areas	List of Penetrations	Potential for Fire Propagation to Adjacent Ar
1	Division I Channel A Vital Instrument & Equipment Room	3	None	None - Hard interdivisional wall between areas
		21	Cable Penetrations	Minimal - 3 hour fire ratin for penetrations
		2	Fire Door	Minimal - 3 hour fire ratin for door
		33	2 Double Fire Doors	Minimal - 3 hour fire ratin for door
		19	Fire Door	Minimal - 3 hour fire ratin for door
		23	None	None - hard concrete wall between areas

Fire Area	Description	Equipment In Area	Safe Shutdown Effect from Loss of All Equipment in Area to Fire
33	Division I Personnel Aisle (El. 50, Col. C-R and 17-25)	**Safe Shutdown Equipment** Recirculating Cooling Unit Reactor Makeup Pumps CCW Chemical Addition Tank Safety-Related Cable Provides Personnel Access to Equipment **Equipment Not Required for Safe Shutdown** Floor Drain Pumps	This is an extended area required for Division I equipment on this elevation. A localized fire in this area may prevent access to one or more Division I safe shutdown components, however, access to the remaining components may continue to be available through the use of either of the Division I stairwells. Access to redundant Division II safe shutdown components would continue to be available, and safe shutdown could be achieved and maintained through the use of Division II systems.

DESIGN ALTERNATIVES
FOR THE
SYSTEM 80+ NUCLEAR POWER PLANT

APRIL 30, 1992
REV. 00

COMBUSTION ENGINEERING NUCLEAR, INC.
WINDSOR, CONNECTICUT

ABB
ASEA BROWN BOVERI

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2.0 SUMMARY AND CONCLUSION

The System 80+ design is an Evolutionary Advanced Light Water Reactor design with improved design features to reduce the risk of core damage and mitigate the consequences if core damage should occur. The design process was integrated with the PRA to ensure that the risk was very low and distributed over all of the safety related systems (i.e., no single system carries a disproportional responsibility for plant safety). The design insured that no single accident sequence dominated the plant risk and the lessons learned from previous PRAs were addressed.

Eleven design alternatives were evaluated. These were selected based on the Design Alternatives evaluated for the Limerick plant[2] and the results from the System 80+ PRA performed by C-E. The Design Alternative analysis used a bounding technique. It was assumed that each Design Alternative worked perfectly and completely eliminated the accident sequences that the Design Alternative was to address. This approach maximizes the benefits associated with each Design Alternative. The benefits were the reduction in risk in terms of whole body person-rems per year received by the total population around the ALWR site. Using $1,000 per person- rem, and a levelized capital cost rate of 17.9%, this risk reduction was converted to a maximum capital benefit that was compared with capital costs.

Table 2-1 summarizes the results of the Design Alternative analysis. The first column, is the percent of the total person-rem/year reduction for each design alternative. The next column, labeled capital benefit, is an equivalent present worth of the annual dose reduction. It is also the maximum amount that could be spent in capital to be cost beneficial. The third column is a rough capital cost estimate for the design alternatives. The net benefit (capital benefit - capital cost) is given in the last column.

The System 80+ plant was designed to meet the stringent design goals in the EPRI Advanced Light Water Reactor (ALWR) Utility Requirements Document. The System 80+ design has a core damage frequency approximately two orders of magnitude lower than existing plants. Therefore, the benefits of improving the existing design are significantly lower than predicted for the Limerick Plant[2]. The analysis presented in this report overestimated the benefits of the Design Alternatives by assuming that they would work perfectly to eliminate the type of accident they are designed to address. Because of the small initial risk associated with the System 80+ design, none of the Design Alternatives are cost beneficial.

TABLE 2-1

SUMMARY OF THE RISK REDUCTIONS OF THE DESIGN ALTERNATIVES

	Design Alternative	PERSON-REM REDUCTION	CAPITAL BENEFIT*	CAPITAL COST	NET CAPITAL BENEFIT
1	CONTAINMENT SPRAY	90%	$27,600	$1,500,000	-$1,472,400
2	FILTERED VENT	86%	$26,300	$10,000,000	-$9,973,700
3	DC BATTERIES & EFWS	69%	$21,100	$2,000,000	-$1,978,900
4	RCP SEAL COOLING	16.5%	$6,034	$100,000	-$93,966
5	PRESSURIZER AUXILIARY SPRAY	6.7%	$2,050	$5,000,000	-$4,997,950
6	ATWS VALVES	4.2%	$1,290	$1,000,000	-$998,710
7	CONCRETE COMPOSITION	2.5%	$765	$5,000,000	-$4,999,244
8	REACTOR VESSEL EXTERIOR COOLING	2.5%	$765	$5,500,000	-$5,499,244
9	H2 IGNITORS	0.1%	$31	$1,000,000	-$999,969
10	HIGH PRESSURE SAFETY INJECTION	0.004%	0	$20,000,000	-$20,000,000
11	RCS DEPRESSURIZATION	0.002%	0	$500,000	-$500,000

* THE MAXIMUM CAPITAL COST ASSUMES NO MAINTENANCE OR TESTING COSTS FOR THE ADDITIONAL EQUIPMENT

$ 1000 per man-rem basis

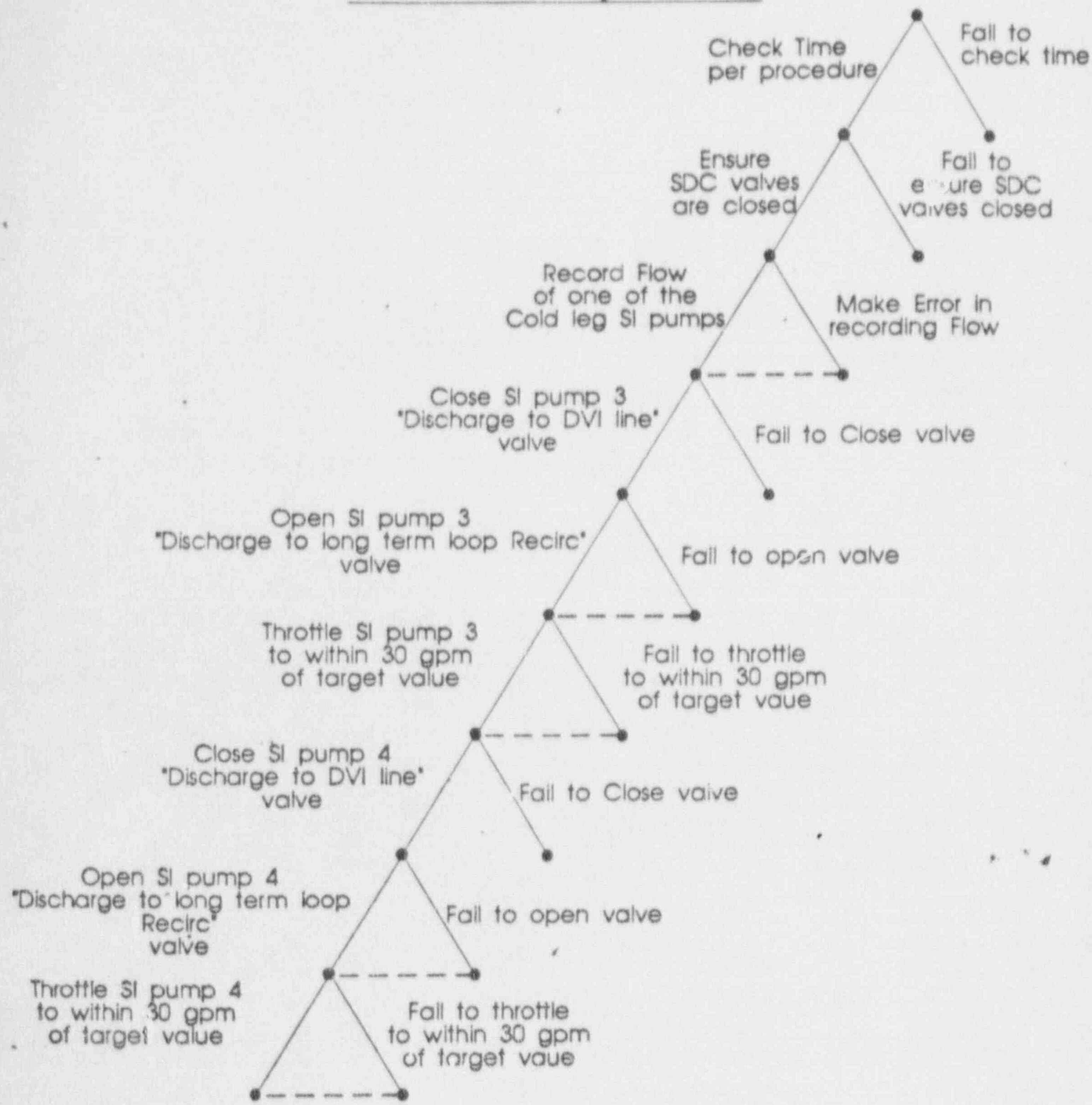
METHODOLOGY FOR THE HUMAN RELIABILITY ANALYSIS FOR SYSTEM 80+

Step 1 (Definition)	From the logic trees, the HRA team qualitatively describes the human interactions more comprehensively.
Step 2 (Screening)	The human interactions are screened, using expert judgement, in order to identify the most important human interactions that directly mitigate, or contribute to, core damage.
Step 3 (Breakdown)	Each key interaction, identified at step two, is broken down into tasks and subtasks which are required to achieve the goal.
Step 4 (Representation)	The subtasks and tasks are then explicitly modelled to identify the actions that the operator may take, the errors of commission and omission, and the associated factors that may impact that error, such as level of knowledge, performance shaping factors etc.
Step 5 (Impact Assessment)	The impact of these errors is then evaluated and added to the system logic trees, similar to step 2
Step 6 (Quantification)	The HEPs are evaluated and added to the PRA

STATUS TO DATE OF THE HUMAN RELIABILITY ANALYSIS FOR SYSTEM 80+

STEP 1	-	COMPLETE	
	-	OUTPUT	Qualitative description of system and Operator Action and reaction for the initiators
STEP 2	-	COMPLETE	
	-	OUTPUT	Finite set of "Critical Operator Actions" which is the input to steps 3 And 4
STEP 3	-	BEGUN	
	-	OUTPUT	example, Initiation of simultaneous Hot Leg/Direct vessel injection
STEP 4	-	BEGUN	
	-	OUTPUT	example, Initiation of Simultaneous Hot Leg/Direct vessel injection

Initiation of Simultaneous Hot Leg and DVI injection



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