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ADVANCED LIGHT WATER REACTOR PLANTS

SYSTEM 80+ ™ DESIGN CERTIFICATION PROGRAM

ANNUAL PROGRESS REPORT
for period October 1, 1992
through September 30, 1993

Prepared for:

UNITED STATES DEPARTMENT OF ENERGY
OFFICE OF NUCLEAR ENERGY
UNDER CONTRACT DE-AC03-91SF19090

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LEGAL NOTICE

Acronyms

ABB-CE	ABB Combustion Engineering
ADVs	Atmospheric Dump Valves
ALWR	Advanced Light Water Reator
AMG	Accident Managment Guidance
ARSAP	Advanced Reactor Severe Accident Program
BNL	Brookhaven National Laboratory
BOP	Balance-of-Plant
CCW	Component Cooling Water
CESSAR-DC	ABB Combustion Engineering Standard Safety Analysis Report for Design Certification
CFS	Cavity Flooding System
CMF	Common Mode Failure
D-RAP	Design Reliability Assessment Program
DE&S	Duke Engineering & Services, Incorporated
DSER	Draft Safety Evaluation Report
EFWS	Emergency Feedwater System
EOGs	Emergency Operating Guidelines
EPRI	Electric Power Research Institute
FDA	Final Design Approval
FRGs	Functional Recovery Guidelines
FSER	Final Safety Evaluation Report
FY	Fiscal Year
HCLPF	High Confidence of Low Probability of Failure
HFE	Human Factors Engineering
HMS	Hydrogen Mitigation System
I&C	Instrumentation and Controls
IRWST	In-Containment Refueling Water Storage Tank

ISI	In-Service Inspection
ISLOCA	Interfacing System Loss of Coolant Accident
IST	In-Service Testing
ITAAC	Inspections, Test, Analyses and Acceptance Criteria
ITP	In-Service Testing Progam
LBB	Leak-Before-Break
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCR	Main Control Room
MSSV	Main Steam Safety Valves
NCC	Natural Circulation Cooldown
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
P&ID	Piping and Instrumentation Diagram
PAGs	Protective Action Guidelines
PASS	Post Accident Sampling System
PRA	Probabilistic Risk Assessment
PSV	Primary Safety Valve
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAI	Request for Additional Information
RCGVS	Reactor Coolant Gas Vent System
RCS	Reactor Coolant System
RSB	Reactor System Branch
SAMDA	Severe Accident Mitigation Design Alternatives
SBLOCA	Small Break Loss of Coolant Accident
SGTR	Steam Generator Tube Rupture

SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SMA	Seismic Margins Assessment
SWEC	Stone & Webster Engineering Corporation
TMI	Three Mile Island
URD	Utility Requirements Document
V&V	Verification and Validation

A. Purpose

The purpose of this report is to provide a status of the progress that was made towards Design Certification of System 80+™ during the U.S. government's 1993 fiscal year. The System 80+ Advanced Light Water Reactor (ALWR) is a 3931 MW_t (1350 MWe) Pressurized Water Reactor (PWR). The design consists of an essentially complete plant. It is based on evolutionary improvements to the Standardized System 80 nuclear steam supply system in operation at Palo Verde Units 1, 2 and 3, and the Duke Power Company P-81 balance-of-plant (BOP) that was designed and partially constructed at the Cherokee plant site. The System 80/P-81 original design has been substantially enhanced to increase conformance with the EPRI ALWR Utility Requirements Document (URD). Some design enhancements incorporated in the System 80+ design are included in the four units currently under construction in the Republic of Korea. These units form the basis of the Korean standardization program. The full System 80+ standard design has been offered to the Republic of China, in response to their recent bid specification. The ABB-CE Standard Safety Analysis Report (CESSAR-DC) was submitted to the NRC and a Draft Safety Evaluation Report was issued by the NRC in October 1992. CESSAR-DC contains the technical basis for compliance with the EPRI URD for simplified emergency planning.

The Nuclear Steam Supply System (NSSS) is the standard ABB-Combustion Engineering two-loop arrangement with two steam generators, two hot legs and four cold legs each with a reactor coolant pump. The System 80+ standard plant includes a spherical steel containment vessel which is enclosed in a concrete shield building, thus providing the safety advantages of a dual containment.

B. History and Status of Project

Since 1985, ABB Combustion Engineering (ABB-CE) and Duke Engineering & Services, Inc. (DE&S) have been developing the next generation of the pressurized water reactor plant for worldwide deployment. In 1990, Stone & Webster Engineering Corporation (SWEC) joined this team, thereby adding to the development of System 80+ an Architect Engineering (AE) firm's expertise. The result is a pre-licensed, standardized plant design that can satisfy the need for a reliable and economic supply of electricity for residential, commercial and industrial use. To ensure that this design is available to meet the utilities' needs , it has been based on proven technology and the most current licensing criteria. These requirements dictated the application of nuclear technology that is advanced, yet evolutionary in nature. This has now been achieved with the System 80+ Standard Plant Design.

In 1985, ABB-CE and DE&S joined forces under the aegis of the Electric Power Research Institute (EPRI) ALWR program to develop, with utilities, the design requirements for the next generation of nuclear power plants. With support from the U.S. Department of Energy, ABB-CE and DE&S again teamed the following year to design and license System 80+, an advanced PWR that responds to these utility requirements. The final version of the EPRI ALWR Utility Requirements Document was submitted to the U.S. Nuclear Regulatory Commission (NRC) in September 1990, and in May 1991 the complete 18-volume final safety analysis report for the System 80+ Standard Plant Design was officially docketed at NRC. A Draft Safety Evaluation Report (DSER) was issued by NRC on October 1, 1992. In 1993, after working on several commercial projects related to System 80+ development, SWEC became an approved subcontractor for Design Certification, taking on selected aspects of BOP licensing of the System 80+ Standard Plant Design. Other organizations involved in the technical development of System 80+ include Bechtel Power Corporation, Impell Corporation, RPK Structural Mechanics Consulting, United Engineers and Constructors and ABB-Atom.

C. Design Certification

Licensing in the United States is near completion, and has involved a new process called Design Certification, described in Title 10, Part 52, of the Code of Federal Regulations. This is complemented by the new nuclear regime in the U.S., which includes one-step licensing, where a single license is required to both build and operate the plant. This is codified by the National Energy Policy Act of 1992, and requires that only complete, standardized plant designs be licensed, and that all safety issues be resolved before construction begins, rather than after, as in the past. Opportunities for public participation are double, but are placed up front, where they are more effective, respecting both the public will and the public purse.

ABB-CE is currently pursuing final design approval (FDA) and design certification of System 80+ from the U.S. Nuclear Regulatory Commission. The NRC is scheduled to grant the FDA in August 1994, with public hearings to follow that will complete certification in December 1995.

As part of the design certification process, ABB-CE has received 4,469 Requests for Additional Information (RAI's) and follow-on questions from the NRC during their review of System 80+ over the past three years. This includes 1,590 RAI's before the DSER, 939 DSER open items, 1,138 follow-on questions to the DSER open issues, and 802 questions on System 80+ Inspections, Tests, Analyses and Acceptance Criteria (ITAAC), the NRC final checklist to ensure that the constructed plant is, in fact, the certified design and that its construction and testing prior to fuel loading has been properly carried out.

D. Licensing

The interactions with the NRC were continued intensively during fiscal year (FY) 1993. By the end of September 1993 (the close of FY 93), only 106 of the 4,469 items, excluding the ITAAC, were still open. The ITAAC items were reviewed by ABB-CE, and representatives from DOE and industry. This review culminated in the transmittal of responses tc the NRC on the scheduled date of September 27, 1993.

Included in the process of resolving the questions and comments for the NRC the following significant submittals were made during FY 1993:

1.	PRA Update - Level 1	10/15/92
2.	Sample ITAAC for Management Discussion	12/01/92
3.	CESSAR-DC Amendment "K"	12/21/92

4.	Draft System 80+ New Source Term Report	01/14/93
5.	DSER Open/Conf.irmatory Items Responses	01/26/93
6.	ITAAC Phase 1 Submittal Part 1	01/28/93
7.	ITAAC Phase 1 Submittal Part 2	02/01/93
8.	System 80+ New Source Term Report	02/16/93
9.	PRA Level 1 Report (Advance)	02/25/93
10.	Letter requesting NRC status of DSER Closeout	03/01/93
11.	Structural Analysis Data	03/02/93
12.	Responses to Shutdown Risk RAI's	03/03/93
13.	Letter Commenting on NRC Resource Allocation	03/04/93
14.	SBLOCA Realistic Evaluation Model Topical Report	03/04/93
15.	Human Factors Submittal	03/04/93
16.	CESSAR-DC Amendment "L"	03/05/93
17.	Prototype ITAAC	03/05/93
18.	ITAAC Supplementary Information	03/05/93
19.	Structural Audit Preliminary Review Package	03/10/93
20.	I&C Diversity Thermal Hydraulic Analysis	03/10/93
21.	ISI	03/17/93
22.	Supplementary Information for RSB Review	03/17/93
23.	Operating Experience Process	03/17/93
24.	Severe Accident Analysis Report	03/23/93
25.	QA Equipment Classification	03/23/93
26.	Nuclear Island Structure Prototype ITAAC	03/23/93
27.	CESSAR-DC Amendment "M"	03/25/93
28.	Revised Safety Analysis	03/25/93
29.	HFE Program Function Allocations	03/26/93
30.	CESSAR-DC Chapter 2& 3 Proposed Revisions	03/29/93

31.	Fire Protection DSER Responses	04/02/93
32.	PRA Off-site Dose Results	04/02/93
33.	CESSAR-DC Amendment "N"	04/05/93
34.	ITAAC Submittal Schedule Commitment	04/05/93
35.	CESSAR-DC Amendment "O" Draft Revisions	04/15/93
36.	IST program-related P&ID's to BNL	04/15/93
37.	System 80+ Tech. Spec. Bases Update	04/20/93
38.	Piping Audit Calculations	04/21/93
39.	ITAAC Submittal #1	04/30/93
40.	Supplemental Information	04/30/93
41.	Completion of ITAAC Submittal #1	05/03/93
42.	Completion of Supplemental Information Package #1	05/03/93
43.	Chapter 2 & 3 Markups and Structural Details	05/14/93
44.	CESSAR-DC Amendment "O"	05/14/93
45.	I&C Diversity CMF Analysis for Limiting Faults	05/19/93
46.	Supplemental Information Package #2	05/20/93
47.	Comments on SECY-93-087	05/27/93
48.	HFE Program Plan Procedures Approach	05/27/93
49.	ITAAC Submittal #2	05/28/93
50.	Completion of ITAAC Submittal #2	06/04/93
51.	Completion of Supplemental Information Package #2	06/04/93
52.	Additional Information to Close Various Issues	06/11/93
53.	ISLOCA Report	06/14/93
54.	Source Term Analysis Time Delays	06/14/93
55.	CESSAR-DC Amendment "P"	06/15/93
56.	ITAAC Submittal #3	06/18/93
57.	Supplemental Information Package #2	06/18/93

58.	SAMDA Report	06/18/93
59.	Request for Status of NUPLEX 80+ Review	06/24/93
60.	I&C and PRA Supplemental Information	06/25/93
61.	Completion of ITAAC Submittal #3	06/29/93
62.	Completion of Supplemental Information Package #3	06/29/93
63.	Explanation of #3 ITAAC Not Submitted	06/29/93
64.	HFE Design Features Review Comment Responses	06/30/93
65.	Tier 1 Submittals (Protested) on D-RAP & ITP	07/01/93
66.	Shutdown Risk Information	07/01/93
67.	ITAAC Writer's Guide	07/01/93
68.	Detailed Arrangement Drawings (Proprietary)	07/14/93
69.	CESSAR-DC Amendment "Q"	07/16/93
70.	Supplemental Information on Piping and PRA	07/16/93
71.	Fire Hazards Assessment Revision	07/22/93
72.	Revised Feedwater Line Break Data	07/22/93
73.	RAI's Response	07/23/93
74.	Information Requested by Reactor Systems Branch	07/23/93
75.	Steam Generator Information Requested by NRC	07/26/93
76.	Response to Reactor Systems Branch Questions	07/29/93
77.	ITAAC Changes Due to Open Items and CESSAR Changes	08/13/93
78.	Summary of CESSAR-DC Amendment "Q" Changes	08/25/93
79.	Refutation of Delinquent Fee Payment Notice	08/25/93
80.	Transmittal of CESSAR-DC Amendment "R"	08/31/93
81.	Supplemental Information for Issue Closure	09/01/93
82.	Supplemental Information for Issue Closure	09/23/93

83.	HFE V&V Information	09/24/93
84.	Responses to NRC ITAAC Comments	09/27/93
85.	PRA & HFE Information	09/30/93

During FY 1993 a great deal of effort has been expended by the NRC, ABB-CE and its subcontractors to resolve many technical areas that have not previously been addressed in any licensing review. As a result the eventual design certification of System 80+ in the United States represents a major technical and licensing advances. The following is a summary of some significant areas that were worked in pursuit of an FDA and Design Certification.

- Steam Generator Tube Rupture (SGTR) with Containment Bypass

The NRC has required that the System 80+ design, as an advanced LWR, include improvements that would reduce the potential for containment bypass via lifting the main steam safety valves (MSSVs) following multiple tube ruptures. If actuated, the MSSVs could result in a bypass of the containment and ultimately radioactive release to the atmosphere, after a postulated severe accident which resulted in fuel damage. It should be noted that the severe accident does not cause the tube rupture. These events are considered to be independent.

Transient analyses were performed for eleven proposed design changes that would add equipment and/or automate the SGTR mitigative actions. A report evaluating these changes was submitted to the NRC along with ABB-CE's recommendations on design changes. With the recommended changes, operator action may be delayed for over four hours following rupture of one tube and over one half hour for five tubes without lifting the MSSVs.

Two changes were made to the System 80+ design. The first change added instrumentation to monitor nitrogen-16 gamma radiation on one steam line from each steam generator. These monitors provide a timely, sensitive and unique indication for tube rupture (and are alarmed in the main control room). The second change revised the component cooling water (CCW) system to assure continued cooling is provided to the instrument air compressors following a Safety Injection Actuation Signal (SIAS). Instrument air is required to operate the steam bypass valves so that steam bypass to the condensers is available to remove heat following reactor trip. Employment of the bypass system avoids the alternative steam blowdown (and radiological release) to atmosphere via the atmospheric dump valves or the MSSVs. The NRC has concurred in these design and documentation changes and this issue is resolved.

CESSAR-DC and related documents were revised to incorporate the design changes and associated procedural changes. Revised documents included CESSAR-DC Chapters 7, 9, 11 and Appendix A, Technical Specifications, Emergency Operations Guidelines, and the ITAAC.

- Interfacing System LOCA (ISLOCA)

The interfacing system LOCA is an event postulated to occur outside containment in a system connected to the reactor coolant system, causing an indirect loss of primary coolant. For the System 80+ design the likelihood of an interfacing system LOCA is already less than 3×10^{-9} events per reactor-year. Nevertheless, design changes include an increased pressure rating for additional equipment and systems, added instrumentation and equipment that terminate and moderate such events, and a simplified design that eliminates unnecessary systems where the same function can be performed by existing equipment. Further, pressure isolation valves are now leak tested, with indication and control from the main control room.

- Standard Technical Specifications

The System 80+ technical specifications have been developed to incorporate the streamlined standard specification issued by NRC for ABB-CE plants in NUREG-1432. System 80+ is the first ABB-CE plant design to take advantage of the reduced requirements and extended time periods incorporated, and has received a comprehensive review by NRC. System 80+ technical specifications have also benefited from specific features of the Nuplex 80+ advanced control complex, including comprehensive application of self-testing and displays that reduce surveillance testing requirements and, consequently, enhance operability. The use of an alternative AC power source (a combustion turbine) to supplement the emergency diesels has also simplified requirements for safety system configuration during certain modes of operation.

- Standard Emergency Operating Guidelines

Emergency operating guidelines have been developed for the System 80+ standard plant design to incorporate operating guides for the plant's advanced safety features. These guidelines represent the supplier's model for effective operator response to postulated transients and accidents. The emergency operating guidelines are based on ABB-CE's standard emergency procedure guidelines and have been reviewed by NRC. The System 80+ guidelines include the general philosophy for and approach to design basis accident mitigation developed for ABB-CE operating plants, plus management of emergencies during shutdown and severe accident conditions.

- Natural Circulation Cooldown Testing

The NRC has required increased assurance of the ability of the operator to perform a natural circulation cooldown (NCC) as presented in Appendix 5D of CESSAR-DC. This analysis was performed within the restrictions of US NRC Branch Technical Position RSB 5-1 which allows only the use of safety grade equipment and assuming the worst single failure. Hence, the emergency feedwater system (EFWS) and the atmospheric dump valves (ADVs) are used for decay heat removal. The safety injection system (SIS) is used for reactor coolant system (RCS) inventory and reactivity control. The reactor coolant gas vent system (RCGVS) is used for RCS venting and pressure control. The NRC felt that testing was required to validate the controllability and process interactions using these systems. The NRC also requested information about the amount of operating margin and the consequences of delayed operator actions. The NRC was especially interested in the use of the RCGVS which was not part of the System 80 design and, therefore, not used during previous NCC testing.

ABB-CE felt that the results presented in CESSAR-DC were similar to those already observed during the NCC testing performed at Palo Verde (which used the auxiliary spray instead of a pressurizer vent). ABB-CE performed additional analyses to demonstrate the similarity of using the auxiliary spray versus the RCGVS to depressurize the reactor coolant system. The resulting system depressurization rates and the overall system response were very similar and quite acceptable. ABB-CE also investigated the amount of operating margin and the effects of delayed operator actions during a NCC. The results demonstrated that the Appendix 5D NCC analysis is flexible and allows the operators a large window of opportunity for all cooldown activities. ABB-CE demonstrated that all the operator actions utilized in the NCC will be tested separately during plant startup to validate the analyses. The instrumentation assumed to be available in the NCC analysis was also shown to be safety grade and consisting of multiple channels.

The results of the additional ABB-CE analysis and investigation were discussed with the NRC staff and incorporated into CESSAR-DC. Appendix 5D will be modified to include a detailed discussion of the plant operating margins, the consequences of delayed operator actions, a description of the instrumentation used and a comparison of the auxiliary spray and RCGVS methods to depressurize the RCS. The tests will be discussed in Appendix 5D and the test descriptions of Chapters 6 and 14 will be clarified, as necessary, to reflect their support of the NCC. The Emergency Operating Guidelines (EOGs) will also be reviewed to ensure consistency with the operator actions during an NCC analysis and will be discussed in Appendix 5D.

- Post Accident Sampling System

The ABB-CE post accident sampling system (PASS) design was finalized for NRC certification during this past year. At the beginning of the year, the ABB-CE PASS design, as described in CESSAR-DC, complied with NRC Requirements for post accident sampling as specified in NUREG-0737, "Clarification of TMI Action Plan Requirements". NUREG -0737 had required a PASS design with the capability for:

- sampling for radionuclides in the reactor coolant containment atmosphere that may be indicators of the degree of core damage,
- sampling for hydrogen concentrations in containment, and
- sampling for dissolved gas, chloride and boron concentration in the reactor coolant.

NRC documents (*Letters of Clarification* included) which followed the issuance of NUREG-0737 reiterated the NUREG-0737 requirements.

EPRI, however, had proposed a much simpler design which was based on the elimination of certain analytical requirements. EPRI's PASS design is described in Volume II, Chapter 3: *Reactor Coolant System and Reactor Non-Safety Auxiliary Systems* of the EPRI Utility Requirements Document. While the NRC had accepted some of EPRI's simplifications, the Staff had rejected the EPRI position which would eliminate pressurized reactor coolant sampling. The NRC's requirement for pressurized reactor coolant sampling resulted from consideration of the TMI-2 accident. The NRC maintained that an RCS dissolved gas measurement, following severe accidents which do not involve an early depressurization of the RCS, would provide the operator with information on the presence of gases which could impede natural circulation and decay heat removal. EPRI claimed that advanced light water reactors have the capability for venting to remove gases accumulated in the reactor vessel, resulting in prompt depressurization and cooling. Such prompt depressurization and cooling would minimize corrosion due to chloride and oxygen.

SECY-93-087, issued on April 2, 1993, presented the NRC recommendations for design compliance with regulations on PASS. Although analytical requirements for total dissolved gas measurements remained, the NRC eliminated the requirement for RCS chloride analysis, and relaxed the time frame for sampling activity and boron measurements. Since compliance with both the EPRI URD requirements and with the NRC requirements on total dissolved gas measurements is not possible, ABB-CE held further discussions with the NRC during the Summer and Fall of 1993.

Although ABB-CE would have preferred that the Commission not require analysis for total gas content, ABB-CE eventually concluded that the requirement was not excessively burdensome to the design. ABB-CE focused on PASS design changes which would make the design consistent with staff requirements for total dissolved gas sampling.

The present ABB-CE approach for the PASS design includes providing a functional description of PASS capabilities within CESSAR-DC. The functional description commits the System 80+ final design to meeting current regulatory criteria for post accident sampling, and gives flexibility in the design details of the final system.

- *Structural Design & Analyses*

Seismic, structural, and mechanical design and analysis required for certification proceeded on schedule to support issuance of the Final Safety Evaluation Report (FSER) by the NRC. Methodology and results were documented and appropriate level of detail included in CESSAR-DC amendments as tasks were completed. In addition to regular telephone and facsimile correspondence between the NRC staff and ABB-CE and its subcontractors (ABB-Impell, DE&S, and SWEC), numerous audit meetings were held for the purposes of presenting status and results and of closing open issues.

Design for an envelope of site conditions greatly increased the complexity and degree of structural and mechanical design and analysis effort required for certification compared to previous nuclear plant contracts. Specifically, the use of thirteen site conditions and three control motions for definition of the seismic design envelope required thirty-nine separate seismic analysis evaluations. This in turn presented the task of developing a standard design appropriate for potential nuclear plant sites, including relatively high seismic (i.e., 0.4g) sites (e.g., Taiwan), without incorporating excessive design conservatism which would not be cost effective for future applicants with lesser seismic requirements.

In addition, a larger number of NRC questions and concerns were received than had been originally anticipated. The major factor contributing to this was that the requirements for performing seismic analysis of all Seismic Category I structures and/or providing detailed structural design for certification were not fully recognized until after receipt of the DSER. These tasks which would typically be performed over a two to three year span were compressed into a one year schedule. Consequently, comments relating to those major work scope efforts which would normally be received as RAI's and DSER open issues were all received as post DSER questions over a compressed time period. Another contributing factor which led to additional questions was use of advanced state-of-the-art technology, such as that required for demonstrating leak-before-break on the main steam line, in the System 80+ certified design.

Design and analyses of the steel containment were completed for design requirements, evaluations performed for severe accident conditions, and seismic capacity calculated for assessing beyond design basis seismic excitations. Independent audit analyses were performed by a contractor to the NRC. Comparisons of design and audit analyses results were exceptionally good.

State-of-the-art methodology was developed and used in several areas during the design process in addition to that used in demonstrating leak-before-break (LBB) for the main steam line and in analyses of the steel containment. This includes development of piping evaluation diagrams to use as a design tool to assure satisfaction of LBB requirements, use of non-linear dynamic analyses in evaluating structural sliding for both design and seismic margin assessments, methods used in evaluating structure-to-structure interaction effects for the envelope of site conditions, procedures used in calculating dynamic soil pressures, methods used in demonstrating minimal basemat uplift when including the effects of buoyancy which precludes the need for a permanent dewatering system for any site conditions, and development and use of a three dimensional finite element model of the Nuclear Island for calculating gross structural loads used in subsequent detailed design.

The seismic, structural, and mechanical analysis and design methods used for System 80+ provide a robust standard design with an appropriate level of conservatism which is capable of being sited at most potential nuclear plant sites.

__- Use of New Source Term in the Post Accident Dose Analyses__

As a result of NRC/Industry initiatives (EPRI ALWR/DOE ARSAP "Physically based Source Term", 10CFR100/10CFR50 rulemaking, Issuance of Draft NUREG 1465, and most specifically, NRC letter to ABB-CE, dated October 19, 1992, "Staff position - Use of Revised Source Term for CE System 80+"), ABB-CE was interested in the impact of the new design basis radiological source term in connection with its design of the System 80+ Standard Plant. A key impetus behind the application of new design basis source term in the design of System 80+ was to increase the accuracy of the information that is being used to determine and establish plant capabilities in the area of accident mitigation. The greater the degree of accuracy in the design basis source term specification, the more capable the plant design will be to protect public health and safety in the unlikely event of an accident with the potential for the release of radioactivity to the plant environs.

The new source term guidance provided in Draft NUREG 1465 was incorporated in the post accident site boundary and control room dose analyses for the Loss of Coolant Accident (LOCA) as well as all other accidents which have fuel damage, i.e., steam line break, control element assembly ejection, fuel handling accident, locked rotor and feedwater line break. The dose analysis supporting the Protective Action Guidelines (PAGs) was also revised based on Draft NUREG 1465. Since there were no computer codes available in the industry to support the modeling of the Draft NUREG 1465 time delay in the core releases as well as the varying amounts of the 9 isotope groups released from the core following a LOCA, a personal computer based computer code, "PERC", was developed. To maintain consistency with the new source term concept, several safety and process analyses were also revised; i.e., development of post LOCA airborne particle size distributions, spray lambda's, containment mixing rates, mass energy releases for the hot leg double ended break with delayed emergency core cooling, etc.

Additionally, the analysis developing the "worst case" post LOCA radiation environmental zones to support equipment qualification was also revised to reflect the new source term. Since there are some differences in the treatment of equipment qualification from the approach taken by the industry currently, ABB-CE's position on equipment qualification was formally transmitted to NRC on September 30, 1993. CESSAR-DC and related documents were revised to reflect Draft NUREG 1465 and the analyses discussed above. As a result of the above analyses certain changes were implemented in the plant design.

- With the exception of the Main Control Room (MCR) intake/recirculation, no credit was taken for charcoal filters. Therefore, safety grade maintenance of all other charcoal filters is no longer required.

- MCR air intakes were relocated to the northwest/southwest corner of the turbine building.

- Each MCR air intake now has redundant radiation monitors (instead of just one per intake) and on detection of high radiation, the air flow will be automatically (not manually) diverted through filter banks. Additionally, component control logic is provided such that the more contaminated intake is automatically closed, and this logic is active for the duration of the accident.

- The MCR intake dampers are now battery backed to ensure that air intake selection capability is available on loss of a train.

- The Tag Out Area entrance will be considered the primary entrance to the MCR. All other entrances will be administratively locked closed.

- The containment low and high purge HEPA filters will be purchased, installed, and maintained to meet the intent of Reg. Guide 1.52 (previously, this requirement was only applicable to the high purge filters).

- The containment low and high purge exhaust ductwork up to the HEPA filters will be classified Seismic I (previously they were Seismic II).

- Probabilistic Risk Assessment

In October 1992, the NRC staff issued the DSER for System 80+. The DSER contained a total of 70 PRA and severe accident related open items. Responses were provided for all 70 open items by January 15, 1993. One response included a commitment to perform a Seismic Margins Assessment to replace the seismic PRA in CESSAR-DC. In the process of reviewing the responses to the open items, the NRC staff issued a number of additional questions that ranged from requests for additional clarifications and requests for additional substantiating information to requests for specific changes in the analyses. The major areas of specific concern were:

- The failure rate for selected motor operated valves,
- The treatment of human error and aggressive secondary cooldown for Steam Generator Tube Rupture (SGTR) and aggressive secondary cooldown for small LOCAs,
- The treatment of Primary Safety Valve (PSV) challenges in the transient event trees,
- Seismic Margins Assessment,
- Severe Accident Mitigation Design Alternatives, and
- The deterministic severe accident analyses for issues such as hydrogen mitigation and core concrete interaction.

ABB-CE performed a series of transient analyses to confirm that aggressive secondary cooldown was feasible for both small LOCAs and SGTRs. ABB-CE also performed a transient analysis to confirm that for the smallest medium LOCA, decay heat removal could be maintained without secondary side heat removal and without core damage. After a series of discussions with the NRC staff on the treatment of human errors in responding to an SGTR, ABB-CE agreed to revise the model for RCS pressure control to clarify the event flow and to provide treatment of additional human error.

After discussions with the NRC staff, the failure rate was increased for the Rapid Depressurization Valves and other motor operated valves that are tested every eighteen months.

In response to an NRC RAI's, ABB-CE performed a series of best estimate transient analyses to determine which transients had the potential for challenging the PSVs. These analyses demonstrated that only the loss of condenser vacuum transients had the potential to challenge the PSVs . Based on this information, ABB-CE deleted the PSV challenge from the loss of off-site power event tree and added it to the transient event tree for loss of condenser vacuum transients.

Resolution of the above issues resulted in changes to about one third of the Level 1 sequences and a requantification of the entire Level 1 PRA. The overall core damage frequency did not change significantly as a result of these changes. To maintain consistency, the Level 2 PRA analyses and the Level 3 PRA analyses also had to be requantified. The overall probability of exceeding 25 rem at 1/2 mile did increase slightly due to the changes in the SGTR evaluations but the results were still well within the specified ALWR goal.

In lieu of revising the original System 80+ seismic PRA to use the Lawrence Livermore National Laboratory (LLNL) hazard curves, ABB-CE performed a Seismic Margins Assessment (SMA) in accordance with the NRC guidance using System 80+ specific floor spectra. The base SMA for the limiting rock site was submitted in July 1993. This SMA will be updated in November 1993 to provide comparison results for a selected spectrum of soil sites. The SMA demonstrated a plant High Confidence of Low Probability of Failure (HCLPF) of 0.6g for System 80+ which exceeds the NRC goal of 0.5g.

- Severe Accidents

The establishment of criteria for Severe Accidents and agreement on their implementation represents one of the major safety advancements made in the

Design Certification Process. The top level severe accident policy positions were established by NRC Policy Issue letters SECY-90-016 and SECY-93-087 and their associated Staff Requirements Memorandums. As a result the System 80+ design has incorporated the following features:

- Core Debris Chamber
- Core Debris spreading area
- Reactor Cavity Flooding System
- Hydrogen Igniter System

In September 1992, ABB-CE provided a Severe Accident Mitigation Design Alternatives (SAMDA) report for System 80+. In response to a series of NRC RAI's and follow-on questions, ABB-CE revised the SAMDA report three times to add evaluation of additional design alternatives, to change the cost benefit evaluation basis and data, and to restructure the report.

The NRC Staff required more detailed reviews/evaluations of hydrogen mixing and burn test data to determine the number and locations of hydrogen igniters to be installed in the System 80+ containment. The staff further required detonation calculations within the containment This requirement resulted in detailed MAAP analyses for hydrogen distribution within the System 80+ containment and the development of hydrogen placement guidelines. The NRC has approved the ABB-CE igniter placement for the System 80+ design and has agreed to setting the number of igniters at 80. Half of these must be backed by the Class 1E batteries.

The original vent size of the In-Containment Refueling Water Storage Tank (IRWST) cover was based on a design basis accident requirement of minimizing the pressure differential across the cover and the severe accident related requirement of venting IRWST hydrogen into the upper portion of the containment. These requirements resulted in an IRWST vent size of 150 ft^2. The NRC Staff requested the IRWST vent size be reviewed for venting hydrogen during a severe accident. MAAP analyses have indicated that a 200 ft^2 IRWST vent area and four hydrogen igniters located and functioning within the IRWST would result in an acceptable hydrogen concentration within the IRWST. Furthermore, a larger IRWST vent size reduces equipment laydown area and increases the potential for collection of debris within the IRWST. After several meetings and discussions with NRC, an IRWST vent size of 200 ft^2 was approved by the NRC.

NRC has requested that instrumentation and other equipment required for severe accident mitigation and recovery be identified and designed to survive the severe accident environment. Both "in-vessel" and "ex-vessel" recovery need to be considered. ABB-CE provided the staff with a minimum set of instrumentation and equipment required for severe accident mitigation and recovery. Also identified was the survivability requirements in terms of RCS and containment environment. The staff has requested that this instrumentation and equipment be designed to survive a hydrogen burn equivalent to that due to a 100% metal water reaction. ABB-CE is working towards identifying the survivability requirements for the minimum set using MAAP analyses and available test data.

The NRC Staff requested specific severe accident management guidance for the System 80+ design. In response, ABB-CE prepared a separate Accident Management Guidance (AMG) document. This document provides guidance when accident recovery cannot be achieved using the information contained in the Functional Recovery Guidelines (FRGs) of the Emergency Operating Guidelines (EOGs) document.

The NRC Staff requested that both the Cavity Flooding System (CFS) and the Hydrogen Mitigation System (HMS) be specified in Technical Specifications to ensure that these systems would be available during a hypothetical severe accident for accident mitigation and recovery. Technical Specifications for both the cavity flooding valves of the CFS and the hydrogen igniters of the HMS have been developed and accepted by the NRC. Testing of the CFS and HMS components are required only during refueling outages.

All of the PRA-related open items dealing with severe accident issues, with the exception of two, have been technically resolved. All changes to the PRA have been documented in Chapter 19 of CESSAR-DC. An AMG document has also been prepared and submitted to the NRC.

E. ITAAC

A large effort at ABB-CE on System 80+ Design Certification was focused on the generation of Design Descriptions and the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). The Design Descriptions and ITAAC constitute the certified design material for System 80+ and therefore, will be part of the Design Certification Rule.

ABB-CE submitted ITAAC, Design Descriptions and supporting material for approximately 80 systems to the NRC in June 1993. The NRC responded with their comments on September 6. ABB-CE and its subcontractors drafted responses to the 802 NRC comments and a management review was performed prior to submitting these responses. The review team consisted of ABB-CE management, DOE, ABB-CE subcontractors and utility representatives. The responses were submitted to the NRC on the scheduled date of September 27, 1993.

F. Summary

The progress towards an FDA and design certification that has been made during FY-1993 can be credited to teamwork. The teamwork of ABB-CE and its subcontractors, NRC, DOE and the nuclear utility industry is evidenced in the resolution of RAI's and follow-on questions. During the process of resolving issues all parties have contributed to the development of the System 80+ design.

As stated earlier, the Final Design Approval for System 80+ is scheduled for August 1994. This goal can be met as long as the same high level of effort and cooperation is maintained in FY-1994. In fact, as of the writing of this report all but a handful of items have been resolved with the NRC, which provides confidence that the schedule for the FDA can be met.

G. Complete Bibliography of System 80+ and Related Publications

For information purposes, a complete bibliography of System 80+ publications is provided below. This list covers all public information provided on System 80+ since February 1985, when development began, through May 1994 (papers to be presented). Together, papers marked with an asterisk (*) give a fairly comprehensive and up-to-date picture of the System 80+ design, including the status of licensing and commercial igniter efforts. These are recommended for anyone wishing to obtain a basic understanding, without having to digest the entire collection of 104 publications.

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