ABB
ASEA BROWN BOVERI

April 30, 1992
LD-92-063

Docket No. 52-002

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

Subject: System 80+™ Supplements to RAI Responses

Reference: ABB-CE Letter LD-92-038, Submittal Schedule, March 25, 1992

Dear Sirs:

Enclosed with this letter are a series of attachments which provide information to supplement previous RAI responses. This corresponds to the commitment in the reference letter to provide "miscellaneous RAI responses" as they are completed.

Attachment 1 includes a statement to confirm that the analysis of a ruptured reactor coolant pump seal cooler tube is bounded by the analysis of a ruptured steam generator tube.

Attachment 2 includes information on single failure assumptions for the analysis of a moderate-energy-line break. This information supplements the response to RAI 440.55(b).

Attachment 3 provides information to supplement the response to RAI 440.127(1). The most significant issue is that the System 80+ plant is designed to not have any adverse systems interactions. If any such interactions are found during the design process, they are corrected at that time. If any problems arise during construction, the constructor must resolve the problem with the designer to ensure that adverse interactions are not created during construction.

Attachment 4 provides supplemental information on emergency lighting, Section 9.5.3 of CESSAR-DC. The changes to CESSAR-DC provided herein will be included in the next amendment, currently scheduled for June, 1992.

Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500
Telephone (203) 688-1911
Fax (203) 285-9512
Telex 99297 COMBEN WSOR

D03 8

Attachment 5 provides resolution of issues discussed with the Structural and Geosciences Branch in an April 9, 1992 conference call. The attached CESSAR-DC revisions will be included in the next amendment.

Attachment 6 adds a footnote to Table 7.5-3 of CESSAR-DC to clarify compliance with Regulatory Guide 1.97 (R3), Category 1 criteria. This change will be included in the amendment to CESSAR-DC currently scheduled for September, 1992.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)

ATTACHMENT 1

In connection with the response to RAI 440.121, the following statement confirms that the High Pressure Seal Cooler (HPSC) tube rupture scenario is bounded by a restrictive area which is less than the area of a 0.75" steam generator tube: "The flow rate from a 0.938" I.D. HPSC tube rupture is bounded by more restrictive passages in series within the pump. The minimum flow area is through the clearance in the water lubricated bearing. This area is 0.231 in^2 compared to 0.348 in^2 for a 0.75" O.D. steam generator tube."

ATTACHMENT 2

Attachment to Letter ALWR-419

Response to NRC Review Comment on RAI 440.55

440.55 Supplement:

Comment:

(b) ABB/CE stated that a flood resulting from a moderate-energy line break in a SCS with a single failure will not affect the plant ability to safe shutdown. What are the single failures considered in drawing your conclusion? Provide a discussion (including a flood resulting in submerging instruments and valves in water) of why isolation of the flooded quadrant will not preclude safe operation of the SCS with a single failure.

Response:

(b) As previously [illegible] the response to RAI 440.55, a study was performed to investigate the effects of pipe breaks outside of containment associated with the shutdown cooling system when the plant is in a shutdown cooling mode.

In accordance with SRP criteria, the most adverse location for a pipe failure resulting in the maximum effects of fluid spraying and flooding was determined to be a through-wall leakage crack in the shutdown cooling system suction-line. The leakrate from the crack was determined to be 1988 gpm.

In order to maximize flood conditions, the study conservatively utilized the leak rate of 1988 gpm and no operator action for 30 minutes.

The shutdown cooling system piping, outside of containment, is located in the reactor building subsphere. The reactor building subsphere is divided into quadrants (reference CESSAR-DC Figure 1.2-4). Adequate design features exist such that any fluid leakage from the shutdown cooling system would be contained in the particular quadrant in which the leak occurs. Walls, flood doors, and curbs are utilized to prevent flood waters from migrating to adjacent quadrants or the nuclear annex. The reactor building subsphere and nuclear annex are divisionally separated by a wall with no unsealed penetrations up to elevation 70+0. This prevents a potential flood in one division from flooding into the other division. Analysis shows that the flood depth in one quadrant considering a continuous flood of 1988 gpm for 30 minutes would reach 1.4 feet. Therefore, the equipment and insurumentation located in one quadrant may become disabled due to submergence from the flood waters, however, equipment in the other quadrant of the same division and two quadrants of the opposite division are unaffected.

Attachment to Letter ALWR-419

Response 440.55(b) Cont'd:

Since the pipe break is assumed to occur in the shutdown cooling suction line the division in which the break occurred is unavailable for performing shutdown cooling (i.e. the suction line in that division is isolated to prevent continued leakage from the crack). However, the other division is available for continued shutdown cooling. Adequate equipment is available in this division to allow continued shutdown cooling even with a single failure. For example, should a failure of the shutdown cooling pump occur, the containment spray pump, which is identical, can be manually aligned to perform the shutdown cooling function (reference CESSAR-DC Figures 6.3.2-1A and 1B). Two component cooling water pumps, two component cooling water heat exchangers, and two station service water pumps are provided in each division (reference CESSAR-DC Figures 9.2.1-1 and 9.2.2-2). A single component cooling water pump, component cooling water heat exchanger, and service water pump are adequate for removing decay heat from the shutdown cooling heat exchanger during the shutdown mode. Therefore, a single failure of a component cooling water pump, component cooling water heat exchanger, or service water pump can be tolerated.

Two Class 1E electrical busses are provided for each division (reference CESSAR-DC Figure 8.3.1-1 provided in RAI 430.13). The containment spray pumps and shutdown cooling pumps are supplied electrical power from a different bus. In addition, the control power for starting each identical pump is from a different control channel. Therefore, a single failure of an electrical bus or control channel can be tolerated. Finally, should there be a loss of offsite power concurrent with the pipe break and the diesel generator of the available division fails or is unavailable, the alternate AC source (combustion turbine) can be aligned to supply power to either or both Class 1E busses in the available division.

ATTACHMENT 3

PREVENTION OF ADVERSE SYSTEM INTERACTIONS

System 80+ is currently arranged to prevent adverse system interactions from water intrusion, internal floods, seismic, pipe ruptures, etc. (reference System 80+ General Arrangement Drawings, CESSAR-DC Figures 1.2-1 through 1.2-12). Several examples of arrangement features which prevent adverse system interactions are listed below:

Electrical equipment is located in areas away from areas containing fluid systems which could flood such electrical equipment should a pipe break or leak occur.

Flood barriers are provided on elevation 50+0 to protect electrical equipment on this elevation from potential floods that could occur in the mechanical areas.

A divisional barrier is provided which totally separates the safety related mechanical and electrical divisions. There are no unsealed penetrations in this barrier up to elevation 70+0. Therefore, a flood in one division cannot migrate to the other division.

The reactor building subsphere is further divided into quadrants. Flood barriers are provided to prevent flooding from one quadrant to the other within a division.

The divisional wall is also a three hour fire barrier for complete divisional fire separation. The quadrant walls are also three hour fire barriers.

HVAC systems are separated into divisions even though they may be non-safety such that there are no duct penetrations through the divisional wall except for the control room ventilation system. This prevents smoke migration between divisions should a fire occur in one division.

.ne divisional and quadrant walls also prevent divisional and quadrant interaction due to seismic, pipe whip or jet impingement.

To ensure the piping, HVAC duct and electrical cables are appropriately routed, designed and analyzed in later detailed design beyond Design Certification, a System 80+ Distribution Systems Design Guide is currently being developed. This guide will specify the appropriate routing guidelines and design and analysis requirements which will maintain the System 80+ arrangement philosophy to prevent adverse system interactions. Applicable portions of this design guide will be included as Design Acceptance Criteria in the Inspection, Test, Analysis Acceptance Criteria

(ITAAC). This will ensure that piping, HVAC duct and electrical cables are appropriately laid out in later detailed design such that there are no adverse systems interactions. All piping, HVAC duct and electrical cabling will be routed on design drawings for appropriate erection. There will be no field routed piping, HVAC duct or electrical cabling in the Nuclear Island.

Any modifications made during the construction of the plant will require design approval through the quality assurance program. The design organization will have the responsibility of ensuring that the construction modification does not create adverse system interactions.

ATTACHMENT 4

ATTACHMENT TO ALWR-419

REQUEST FOR ADDITIONAL INFORMATION SECTION 9.5.3

Question

Part 1

Identify the vital areas and hazardous areas where emergency lighting is needed for safe-shutdown of the reactor and the evacuation of personnel in the event of an accident or a fire. Tabulate the lighting system provided in your design to accommodate those areas so identified. In addition, describe your lighting system provided for the main control room.

Part 2

Paragraph 9.5.3.2.3 states that for all emergency conditions, both emergency lighting systems shall be considered operational except in emergencies involving some loss of Class 1E power. Please explain what is meant by this statement and list all areas that would be affected by loss of Class 1E power.

Part 3

A recent event at Palo Verde revealed that a burned-out light bulb, inadvertently repositioning of the emergency lighting fixtures, and inadequate illumination contributed to the difficulty in operators performing their required activities. In addition, there were deficiencies in the licensee's maintenance and testing of emergency lighting. This incident has raised concerns regarding the adequacy of the lighting system design.

Provide a discussion as to how your lighting system design and your testing program of the emergency lighting will ensure that operators can perform the required tasks in the event of a loss of all normal lighting systems.

Response:

Part 1

Vital areas are identified through performing hazards analysis and development of plant emergency procedures. Detailed hazards analysis can not be completed until detailed piping, HVAC and electrical routings are complete. Several known vital areas can be identified at this time; however, vital areas will continue to be added and revised throughout the design process as hazards analysis are completed and plant emergency procedures are finalized.

ATTACHMENT TO ALWR-419

A fire hazards assessment has been sent to the NRC for review. This document describes how the fire hazards assessment will evolve through detailed design, construction, and startup of the plant into a fire hazards analysis. This document also assesses fire areas where major equipment has been located which is required for safe-shutdown of the plant. As additional design detail is completed the fire hazards assessment will identify additional vital areas for safe-shutdown, personnel egress and fire brigade access following a fire which require emergency lighting. These areas will be identified in later detailed design beyond certification.

CESSAR-DC Section 9.5.1.6.1 discusses the Emergency Lighting for safe-shutdown following a fire and identifies vital areas known at this time. Section 9.5.1.3.5 discusses egress routes following a fire as well as emergency lighting requirements for these routes. Clarification of emergency lighting requirements were included in the response to RAI 280.9. A revised copy of these sections are attached for your review. Section 9.5.1.6.1 states emergency lighting will be located as determined in the fire hazards analysis.

The following will be added to Section 9.5.3:

"Emergency lighting is located in vital areas throughout the plant as identified in Emergency Procedures and Hazards Analysis for safe-shutdown of the plant following an accident or hazard. Included in the vital areas will be the Control Room, Technical Support Center, Operations Support Center, the Remote Shutdown Panel Room, the stairway which provides access from the Control Room to the Remote Shutdown Panel room, Sample Room, Hydrogen Recombiner Rooms, routes for personnel passage and egress, and other areas where operator access is required post-accident or hazard."

Requirements for control room lighting are discussed in Section 18.6.6.1.B.

Part 2

The emergency discussed here is the Station Blackout event or loss of all AC power including the emergency Class 1E diesel generators, or an event where there is a failure of one Class 1E electrical division where the Emergency Procedures may call for local operator action to ensure reliability from two divisions. As discussed above, all vital areas will not be identified until the Emergency Procedures are developed, but areas most likely to be identified are the diesel generator rooms, steam-driven emergency feedwater pump rooms and pathways from

the control room to these rooms.

Part 3

Section 9.5.3.2.3 states that the emergency lighting system achieves illumination units of at least 10 foot-candles in those areas of the plant where emergency operations are performed which could require reading of printed or written material or the reading of scales and legends. In other areas of the plant, the emergency lighting achieves a minimum illumination level of 2 foot-candles. The 10 foot-candles will assure adequate lighting to accomplish the necessary tasks, particularly those which require reading. Lower levels in other parts of the plant provide adequate light for egress without requiring additional equipment and power supply capability.

Section 9.5.1.6.1 discusses the test requirements for the Emergency Lighting specified per the Fire Hazards Analysis. These same test requirements will be added to Section 9.5.3.3. See attached markup of this section.

Flame spread, fuel contribution, and smoke development are measured in accordance with UL 84, "Steiner Tunnel Test." Critical radiant flux is measured in accordance with UL 251, "Fire Tests of Building Construction and Materials."

If it is necessary to select a specific material which does not meet or has not been tested to the above qualifications (in the installed configuration), an engineering annlysis will confirm that the General Design Guidelines are met and there is no reduction in the fire safe quality of the plant.

9.5.1.3.5 **Means of Egress**

Personnel egress in the Nuclear Annex is arranged to meet provisions of NFPA 101, "Life Safety Code" or NFPA 101m, Alternative Approaches to Life Safety."

There are stairs in each quadrant of the Nuclear Annex enclosed by three-hour fire rated walls. Each stair tower is pressurized by a dedicated fan mounted at the top of the tower. Personnel access/egress corridors are arranged to assure an unobstructed path of travel. Three-hour fire rated doors are installed along the corridors to assure a maximum of 200 feet travel to an exit or area of refuge. I

There are three stair towers in the Reactor Building Subsphere. Each is located in a separate quadrant of the Reactor Building. Access/egress into the Containment Building is through two personnel air locks, one located on elevation 115+6 and one located on elevation 146+0.

Sealed beam, battery powered emergency lighting units are installed to illuminate emergency egress paths in accordance with standards of the American Illuminating Society.

9.5.1.4 __Safe Shutdown Following Fire__

The System 80+ plant arrangement and layout provides inherent separation of safety related systems, equipment and components, divisions and channels. The plant arrangement permits the unit to be taken to cold shutdown following a fire without the need to implement repairs or for operators to perform extraordinary manual actions outside of the control room or remote shutdown panel room.

In the Nuclear Annex, each division and redundant channels of safety related equipment are separated by three-hour fire rated

Sealed beam, 8-hour minimum battery powered emergency lighting units are provided for all areas and access to areas that must be occupied for safe shutdown of the plant following a fire.

ATTACHMENT TO ALWR-419

Failure of the fire detection and alarm system would not affect operation of other plant systems.

Fire detectors, control panels, and manual pull stations are Underwriter's Laboratory Listed or Factory Mutual Approved for fire protection service.

9.5.1.6 System Interfaces

9.5.1.6.1 Emergency Lighting

Sealed beam, battery powered lights are located, as determined by the Fire Hazards Analysis, for personnel egress in accordance with NFPA 101, "Life Safety Code," as well as in the control room, Technical Support Center, Operations Support Center, the Remote Shutdown Panel Room, ~~Transfer Switch Room~~, and the stairway which provides access from the control room on elevation 115+6 to the Remote Shutdown Panel, ~~and Transfer Switch Rooms~~ on elevation 70 ~~65~~+0, and ~~to elevation 50'+0 where the reactor trip switchgears are located.~~

Batteries of these emergency lights are designed for eight hours continuous operation following loss of station auxiliary power. Bulbs are located so that adequate illumination is provided and is not obstructed by plant equipment and components.

Battery powered, emergency lighting units are Underwriter's Laboratory Listed.

9.5.1.6.2 Ventilation Systems

Fire and smoke control are recognized as important elements of the overall fire protection program. The ventilation systems are designed in accordance with NFPA 90A, "Air Conditioning and Ventilation Systems" and NFPA 204M, "Smoke Control Systems."

Ventilation Systems are division-specific so that fire or smoke in an area containing a safety related division of equipment cannot migrate through the ventilation ducts to an area containing the redundant division of safety related equipment. Fire dampers are installed in fire rated barriers and have the same fire resistance rating as the barrier. Exceptions are the Containment Purge and Pressure Control Systems and Annulus Ventilation System which must function following some plant design basis accidents to prevent release of radioactivity. Fire dampers are not installed in these systems because failure or spurious actuation would interfere with system safety function. Portions of the Nuclear Annex Control Complex Smoke Control System motor operated smoke control dampers are installed in lieu of thermally operated, automatic closing fire dampers as described below. I

ATTACHMENT TO ALWR-419

The circuits to the individual lighting fixtures are staggered as much as possible to ensure some lighting is retained in a room in the event of a circuit failure.

9.5.3.2.2 Security Lighting System

The security lighting system is considered part of the permanent non-safety systems and is fed from an uninterruptible power supply connected to a non-safety battery. The security lighting system, therefore, remains energized as long as power from an offsite power source, a standby non-safety source, or a non-safety battery is available.

The security lighting system is designed to provide a minimum illumination of 0.2 foot-candles when measured horizontally at ground level.

Add insert 1

9.5.3.2.3 Emergency Lighting

The emergency lighting system achieves illumination units of at least 10 foot-candles in those areas of the plant where emergency operations are performed which could require reading of printed or written material or the reading of scales and legends. These areas are typically control rooms or local control stations. In other areas of the plant, the emergency lighting achieves a minimum illumination level of 2 foot candles.

The emergency lighting is accomplished by two systems:

A. Conventional AC fixtures fed from class 1E AC power sources, and

B. DC self contained, battery-operated lighting units.

Both systems are qualified Class 1E. For all emergency conditions both systems are considered operational except in emergencies involving some loss of Class 1E power, adequate illumination in those areas which could be involved in recovery, e.g., electrical distribution control panels and the emergency generators and their contents, depend only on the DC self-contained battery operated lights.

The DC self contained, battery-operated light units meets the following requirements:

A. The battery life is at least 8 hours at rated load.

B. The loading is not greater than 80% of the rated capacity with additional derating for temperature variations, where appropriate.

E

C. A time delay is provided so that the lights turn off on the resumption of power only after there is adequate time for the normal lighting to restart.

D. Provision is made to lock the power supply breakers which supply the units in the "energized" position.

9.5.3.3 Inspection and Testing Requirements

All lighting systems are inspected, checked, and tested for operability after installation to assure proper operation and coverage. Continuous use of the Normal Lighting System provides the bases for testing the system. E

Add Insert 2

ATTACHMENT TO ALWR-419

Insert 1:

Emergency lighting is located in vital areas throughout the plant as identified in Emergency Procedures and Hazards Analysis for safe-shutdown of the plant following an accident or hazard. Included in the vital areas will be the Control Room, Technical Support Center, Operations Support Center, the Remote Shutdown Panel Room, the stairway which provides access from the Control Room to the Remote Shutdown Panel room, Sample Room, Hydrogen Recombiner Rooms routes for personnel passage and egress and other areas where operator access is required post-accident or hazard.

Insert 2:

A. Acceptance Tests

1. Emergency lighting units are inspected to assure that each bulb is properly directed and unobstructed.

2. Ten percent of the DC powered emergency lighting units are tested utilizing battery power to assure operation for the designated duration.

B. Annual Tests

1. Emergency lighting units are inspected to assure that each bulb is properly directed and unobstructed.

2. Ten percent of the DC powered emergency lighting units are tested utilizing battery power to assure operation for the designated duration.

ATTACHMENT 5

Attachment. ALWR-430

3.5.1.3 _ -bine Missiles

1.0E-4

The probability of turbine missile generation and adverse impact effects on Seismic Category I systems and components is assured to be less than 1.0E-3 events per turbine-year by a combination of the following measures:

A. Reliable turbine overspeed protection provisions (see Section 10.2.2 for details).

B. Adequate assurance of turbine disc integrity by design and inspection (see Sections 10.2.2 and 10.2.4 for details). I

C. Placement and orientation of the turbine generator (described below).

D. Consideration of the protection provided by plant structures not explicitly designed as barriers that may limit missile penetrating capabilities to less than the capability of Seismic Category I structures. D

The turbine generator placement and orientation for the System 80+ Standard Design, and the corresponding low-trajectory missile strike zones, are illustrated in Figure 1.2-1. The placement and orientation of the turbine generator provides adequate protection against low trajectory turbine missiles by excluding safety-related structures, systems, and components from the low trajectory turbine missile strike zones in accordance with the guidelines of Regulatory Guide 1.115.

Critical structures (i.e., those housing safety-related equipment) and exterior equipment are located in line with, or within close proximity to, the longitudinal axis of the turbines. This makes the potential for turbine-generated missiles to strike these targets negligibly small. I

The System 80+ design follows the guidelines of Regulatory Guide 1.115 by placing and orienting the turbine such that all safety-related structures, systems, and components are excluded from the low trajectory turbine missile strike zones or if site characteristics make this impossible, safety-related targets will be placed and shielded such that the combined strike and damage probability for the safety-related targets in these zones is less than 10^{-3} per turbine failure.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado-generated missiles are the limiting natural hazard and, as such, are a part of the design basis for Seismic Category I structures and components. Table 3.2-4 lists those structures, E

Attachment - ALWR-430

(from Section 3.1.44. Criterion 51 - Fracture prevention of Containment Pressure boundary)

RESPONSE:

SA-537 Class 2

The material selected for the containment vessel is ~~carbon steel normalized to refine the grain which results in improved ductility. In addition,~~ The actual mechanical and chemical properties of the material are within the limits of minimum ductility defined in the 1989 ASME Code Material Specifications Part A-SA-537/SA-537M.

The containment vessel is built to Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. I

The design of the vessel reflects consideration of all ranges of temperature and loading conditions which apply to the vessel during operation, maintenance, testing and postulated accident conditions.

All seam welds in the vessel are 100 percent radiographed, and the acceptance standards of the radiographs ensure that flaws in welds do not exceed the maximum allowed by the ASME Code.

Steady state and transient stresses are calculated in accordance with accepted methods (see Section 3.8). D

3.1.45 CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

RESPONSE: D

The containment vessel is designed so that integrated leak rate testing can be performed at design pressure after completion and installation of penetrations and equipment in accordance with the requirement of Appendix J of 10 CFR 50 (see Section 6.2.6).

3.1.46 CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows. D

Attachment ALWR-430

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

The containment vessel materials are in accordance with Article NE-2000 of Subsection NE, "Class MC Components," of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

The containment plate material is ASME SA537 Class 2. This material is exempt from post-weld heat treatment requirements through thicknesses of 1.75 inches.

Fabrication and erection of the containment vessel are in accordance with Article NE-4000 of Section III of the ASME Code. This includes welding procedures, procedure and operator performance qualifications, post weld heat treatment and tolerances.

Nondestructive examination of welds and materials is in accordance with Article NE-5000 of Section III of the ASME Code. I

DELETE THIS PORTION

~~Coatings used on the containment vessel meet the requirements of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" (Reference 17). This includes the coating materials, surface preparation and application.~~

3.8.2.6.2 Quality Control

The general provisions of the overall Quality Assurance program are outlined in Chapter 17. These are supplemented by the special provisions of the ASME Code for quality control as applicable to Class MC Components. The containment vessel is ASME Code stamped. Therefore, the ASME Code requirements for quality control have priority over those outlined in Chapter 17 in case of any conflict.

3.8.2.6.3 Special Construction Techniques

The steel containment vessel may be assembled in sections in an area of the construction yard and then lifted and moved into place with a walker crane. This procedure allows the assembly of the containment to begin when the plates forming the lower hemisphere are delivered to the site. In this manner the containment assembly can proceed on a parallel path with the construction of the concrete subsphere region. The following

Attachment ALWR-430

14. Regulatory Guide 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants".

15. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

16. Code of Federal Regulations, Title 10, Part 50.

~~17. ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities~~". I

17. ~~18.~~ Regulatory Guide 1.84, Design and Fabrication Code Case Acceptability ASME Section III Division 16 .

Attachment ALWR-430

as a three dimensional thin shell using the finite element method of analysis. The stresses and deflections produced in the shell under the applied loads are calculated with the ANSYS computer program (Reference 2). The ANSYS mathematical model used to represent the containment vessel is shown in Figure 3.8-3.

Seismic stresses and deflections are calculated using the response spectrum method. The frequencies of vibration and corresponding mode shapes are determined using the normal mode method. Modal responses are combined as described in Regulatory Guide 1.92 (Reference 15). The appropriate damping level for the applied response spectra is defined in Regulatory Guide 1.61 (Reference 14).

C. Buckling

The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional linear bifurcation analysis using an ANSYS finite element model similar to that constructed for the static and dynamic analyses. These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284 (Reference 5). Code Case acceptability is in concurrence with Regulatory Guide 1.84 (Reference ~~16~~ 17). I

D. Ultimate Capacity

The maximum pressure capacity of the containment vessel is evaluated by a large displacement elastic-plastic nonlinear analysis. The vessel is modeled with axisymmetric shell finite elements using the ANSYS computer program. The ANSYS model is shown in Figure 3.8-4.

The stresses in the containment vessel due to combustible gas loadings are calculated using a static linear elastic analysis. The vessel is represented by a three-dimensional thin shell finite element model with the ANSYS computer program. This model is similar to those used for the seismic analysis and buckling evaluation.

E. Nonaxisymmetric and Localized Loads

The containment is not divided into compartments (see Section 6.2.1.2) so there are no nonaxisymmetric loads applied to the containment vessel during a Design Basis Accident.

Attachment ALWE-430

overlay cladding, used in the fabrication of components of the engineered safety features system, is controlled to 5FN-20FN. The delta ferrite content of each lot and/or heat of weld filler metal used for welding of austenitic stainless steel code components shall be determined for each process to be used in production. Delta ferrite determinations for consumable inserts, electrodes, rod or wire filler metal used with the gas tungsten arc welding process, and deposits made with the plasma arc welding process may be determined by either of the alternative methods of magnetic measurement or chemical analysis described in Section III of the ASME Code. Delta ferrite verification should be made for all other processes by tests using the magnetic measurement method on undiluted weld deposits described by Section III of the ASME Code. The average ferrite content shall meet the acceptance limits of 5FN to 20FN for weld rod or filler metal.

For submerged arc welding processes, the delta ferrite determination for each wire/flux combination may be made on a production or simulated (qualification) production weld. D

6.1.1.1.5 Ferritic Steel Welding

The recommendations of Regulatory Guide 1.50, 'Control of Preheat Temperature for Welding of Low-Alloy Steel' and Article D, Section III of the ASME Code are followed. I

Moisture control on low hydrogen materials shall conform to the requirements of the ASME Code and/or AWS D1.1, 'Structural Welding Code'.

6.1.2 ORGANIC MATERIALS

6.1.2.1 Protective Coatings

Many coatings which are in common industrial use may deteriorate in the post-accident environment and contribute substantial quantities of foreign solids and residue to the containment sump. Consequently, protective coatings used inside the containment, excluding components limited by size and/or exposed surface area, are demonstrated to withstand the design basis conditions and meet the intent of ANSI N101.2 (1972), "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Any particulate debris of appreciable size that does occur will either settle to the bottom of the Holdup Volume Tank or will be trapped by the filter screen at the bottom of the in-containment refueling water storage tank (IRWST). The screen size is smaller than the line piping I

(Insert sentence from following page)

Attachment ALWE-430

<u>Insert as third sentence to Section 6.1.2.1</u>

In addition, the selection of coatings are based on ASTM Standards D3842-86 "Standard Guide for Selection of Test Methods for Coatings for use in Light Water Nuclear Power Plants", ASTM D3911-89, "Standard Test Method for Evaluating Coatings used in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions", and ASTM D3843-89,"Standard Practice for Quality Assurance for Protective Coatings Applied to Nuclear Facilities".

ATTACHMENT 6

TABLE 7.5-3 (Cont'd)

(Sheet 7 of 7)

POST-ACCIDENT MONITORING INSTRUMENTATION

Add to all pages

Parameter		Number of Sensed Channels(5)	Sensor Ranges(6,3,7)	Indicated Range(7)	Location(1,2)	Reg. Guide 1.97 Category(8)

NOTES:

1. See Chapter 18 for type of readout.
2. All Category 1 variables are also recorded via the DPS Historical Data Storage and Retrieval program.
3. Post-accident monitoring instrumentation is qualified for the appropriate environmental conditions (refer to Section 3.11). D
4. Degree of subcooling is calculated from RCS pressure, RCS temperature and core exit temperature parameters.
5. MCBDs are provided in appropriate sections of Chapter 7.
6. Post-accident channel accuracy is a time dependent function of post-accident environmental conditions.
7. Ranges given are typical. Actual ranges are site dependent based on the equipment procured. Therefore, the site specific SAR shall make appropriate adjustments as necessary. I
8. Where multiple categories are listed for a single parameter, the instrumentation is designed and qualified to Category 1 requirements.

Add