ABB
ASEA BROWN BOVERI

June 25, 1993
LD-93-100

Docket No. 52-002

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

Subject: System 80+™ Information for Issue Closure

Dear Sirs:

The attachments to this letter provide material to close follow-on questions to DSER responses. Attachment 1 provides responses to questions on the fire and flood portions of the PRA. Attachment 2 contains responses to the Technical Evaluation Report for the Human Factors Engineering task analysis. Attachment 3 contains the operator response time assessment for the large break LOCA and MSLB with a common mode failure of digital safety equipment. Attachment 4 transmits the key indicator and credited operator response times for the analysis of accidents with a common mode failure of digital safety equipment.

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

Very truly yours,

COMBUSTION ENGINEERING, INC.

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ATTACHMENT 1

RESPONSES TO RAIs

INTERNAL FIRES

RAI 1:

Please include appropriate set of figures (plans) showing such information as fire areas, fire barriers and passageways, commensurate with available design details. Refer to such figures in your analysis. For areas where no adequate design details are available, please identify potential ITAAC, RAP and COL items.

RESPONSE TO RAI 1:

The appropriate set of figures showing such information as fire areas, fire barriers and passageways is included as Attachment 1. These figures are used and referenced in the fire risk assessment for the System 80+ design (CESSAR-DC 19.7.3.1.3.2). Accordingly, Table 19.7.3.1-6 will be updated to reflect the attached figure numbers instead of drawing numbers previously used.

RAI 2:

Please include the control room in your fire analysis. With the possibility of a fire in the control room affecting both multiple systems and the operator's ability to respond to the fire, this omission does not seem justified. Does the alternate shutdown panel have identical controls and displays of plant status information needed during accidents as the main control panel? If the answer is no, how would this affect accident modeling and operator actions? Could a fire in the control room affect the transferring of control to the alternate panel? Please explain.

RESPONSE TO RAI 2:

Sufficient instrumentation and controls are provided outside the control room to: 1) achieve prompt hot standby of the reactor, 2) maintain the unit in a safe condition during hot shutdown, and 3) achieve cold shutdown of the reactor through the use of suitable procedures using the Remote Shutdown Panel in the remote shutdown room. The remote shutdown room is located in a separate fire area at an elevation which is different from the control room.

The NUPLEX 80+ design provides fiber optic switches near each control room exit for transfer of control from the main control panel to the remote shutdown panel. The fiber optic switch interrupts the light being transmitted through the fiber optic cable. The light transmitters and receivers and their power sources are located in the appropriate component control cabinets in the channelized I&C equipment rooms. Actuation of the switches at either exit initiates transfer of each division of the ESF-CCS and each division of the Process-CCS to perform a soft transfer to deactivate the main control panel as a control interface and to activate the remote shutdown panel control interface. Transfer initiated from these switches is one way, they cannot transfer control back to the main panel. Transfer of control back to the main control panel can be performed using the Maintenance & Test Panels provided for each division of the ESF-CCS and Process-CCS in the channelized equipment rooms. Because the control room exits are located at different ends of the room, a fire in the control room will not affect the transferring of control to the remote shutdown panel.

The control panel specifications prohibit the use of neoprene, limit the use of PVC, and require that non-metallic materials used in control panels will neither ignite nor explode (from an electric spark, open flame, or from heating). The specifications also prohibit the use of material that would independently support combustion.

The energy sources coming into the control panels are limited to 120 VAC. The 120 VAC power is distributed within the panels to: 1) CRTs, 2) control panel multiplexers, 3) lamp and switch power supplies, and 4) electro-luminescent display power supplies. Approximately 95% of the power distribution and wiring within the panels is low voltage. Such limited energy sources practically eliminates potential ignition sources with the panels. Circuit breakers and fuses are provided for power distribution circuit protection within the panels. Power supplies are current limiting. This practically eliminates the possibility of a fault that provides an ignition source.

The use of lamp LEDs and a low voltage interface between control switches and panel multiplexers minimize the potential that any fault could generate sufficient heat to act as an ignition source. This also allows the use of smaller wire size for panel wiring, which results in reducing the amount of insulation (non-metallic material) in the panels. Power wiring (120 VAC) is run separately from the low voltage wiring (5 VDC - 24 VDC) within the panels. Where adequate separation distance cannot be maintained between power wiring and low voltage wiring, power wiring is run in conduit and separation is maintained to the maximum extent practicable. Conventional control rooms and panels bring high energy control circuits into the control room/panels. In NUPLEX 80+, over 98% of the control and indication signals are interfaced to the main control panel via fiber optic cables.

CRTs, electro-luminescent displays, multiplexers and power supplies are packaged (enclosed or at least partially enclosed) units which inhibit, if not prevent, the spread of fire within a panel if it occurred. Each panel section is provided with high temperature switches_to provide indication of both abnormal operating temperature and of a fire internal to the panel.

The overall NUPLEX 80+ approach reduces the number of indicators, switches and other control panel mounted devices, which further reduces combustible loading and potential problem areas related to fire.

The component control power interfaces through field multiplexers and does not enter the main control room panels. Therefore in the event of a panel fire, hot shorts in the component control power circuits cannot occur, eliminating the possibility of component damage.

Because of the design features of the control room as summarized above, ABB C-E believes that a fire in the control room is an extremely low probability event and inclusion of the control room in the qualitative fire risk assessment is therefore not warranted.

RAI 3:

One item of concern in the design of evolutionary plants is the migration of smoke, hot gases, or the fire suppressants into other fire areas where these elements could affect safe shutdown capabilities, including operator actions. The issue of the propagation of smoke, hot gases and fire suppressants was identified in SECY 90-016. While this issue was identified in the fire protection analysis (CESSAR DC Chapter 9), the analysis of severe accident fires does not address this issue. Please address this issue. For areas where no adequate design details are available, please identify potential ITAAC, RAP and COL items.

RESPONSE TO RAI 3:

The migration of smoke, hot gases, or the fire suppressants into other fire areas where these elements could affect safe shutdown capabilities, including operator actions, is considered to be an extremely low probability event. The control building ventilation system (CESSAR-DC 9.5.1.3.3) is provided with outside air intakes for the control room separate from the remainder of the control room complex and the remote shutdown room. Separate ductwork is utilized for the control room and the remote shutdown room to eliminate the migration of smoke, hot gases, or the fire suppressants between these two fire areas. The control complex has a smoke control system which utilizes dedicated smoke exhaust fans, smoke dampers and outside air supplies by the control complex air-handling units.

The main control room and the remote shutdown room are located at different elevations and in different fire areas. Since the main control room ventilation system is separate from the ventilation system for the remote shutdown room, and the stairwells connecting the main control room and the remote shutdown room are pressurized, smoke cannot migrate to these areas. Thus, the pathway to the remote shutdown room will be free of smoke originating from a control room fire. Safe shutdown can be accomplished from the control room or the remote shutdown room without accessing personnel into other areas of the plant.

If a fire occurs inside the control room, it is expected that the operator will leave the control room before conditions degrade to the point where the environment makes it impossible to transfer control to the remote shutdown room. A set of transfer switches (as described in response to RAI 2) are located near each exit door of the control room. It is expected that under the most adverse conditions, the operator will trip the reactor, walk to the control transfer switches, actuate the switches and exit the control room.

Because of the design features of the System 80+ control room, a control room fire would be an extremely low probability event and was therefore eliminated from the qualitative fire risk assessment.

RAI 4:

Some of the information provided in Table 19.7.3.2-1 indicates that fires in several areas could trip the plant but would not impact any systems used to bring the plant to a safe shutdown condition. Included in these areas are the intake structure and the transformer yard. A fire in either of these structures could also affect safety related support systems associated with the ultimate heat sink and power supplies. Also, a fire in the turbine building could affect system in addition to the feedwater system that would make the transient worse than a loss of feedwater. Please provide risk based analyses to assess the significance of fires in these areas. For areas where no adequate design details are available, please identify potential ITAAC, RAP and COL items.

RESPONSE TO RAI 4:

For the System 80+ design, a fire in the intake structure is assumed to cause the loss of one station service water pump and at most the loss of one division of component cooling water/station service water. For this event, the plant would respond in a manner similar to the loss of CCW/SSW to one division of ESF equipment. As such, the results of the scoping evaluation presented in the PRA will include the frequency of a fire occurring in the intake structure.

For the System 80+ design, the occurrence of a fire in the transformer yard will not affect the onsite safety related power supply (i.e., the emergency diesel generators). Each emergency diesel generator is located in a fire area which is different from the fire area of the transformer yard. The main and reserve transformers will be physically separated such that no fire or environmental effect will disable both offsite sources of power (CESSAR-DC 8.2.1.3). The frequency of loss of offsite power used in the PRA is due to all causes which include fires in the yard transformers. Therefore, a fire in the switchyard need not be addressed separately because it has been already covered in the loss of offsite power event (CESSAR-DC 19.4.8).

For the System 80+ design, the occurrence of a fire in the turbine building is assume to result in loss of main feedwater, startup feedwater, and possible the loss of one switchyard. Consequently, a plant trip would occur. For this event, the plant would respond in a manner similar to any other transient (Note that in the PRA the main feedwater system was not credited for other transients). Since a fire in the turbine building would initiate a plant trip and have no direct effect on any of the safety related systems, this event was included as a contributor to the frequency of other transients and need not be addressed separately.

RAI 5:

The internal fire analysis concludes that the worst possible scenario following an internal fire event would affect only one division of the Engineered Safety Features (ESF) equipment. The separation among divisions is a very important feature of the System 80+ design. For this reason, wall and fire barrier integrity as well as the requirements any penetrations in the wall separating the two divisions must meet should be addressed in the ITAAC.

RESPONSE TO RAI 5:

The requirements for fire barrier integrity and the penetrations in the wall separating the two divisions of ESF equipment are addressed in the ITAAC which are provided under a separate submittal.

ATTACHMENT 1

FIRE AREA DRAWINGS

(CESSAR-DC Figures 9.5.1-2 through 9.5.1-9)

19.7.3.2 Internal Fire Frequency Calculation

A quantitative assessment of the risk due to internal fire can not be made at this time because detailed design information for cable routing and the fire detection and fire suppression systems is not presently available. However, a scoping evaluation is performed to assess the risk due to internal fires in areas of the Nuclear Annex other than the containment or the control room. The containment and the control room were not considered for the reasons given in Section 19.7.3.1.

Two types of fires were considered in the scoping evaluation. One type is a fire in an area (or a room) which could disable safety-related equipment in that area and which has the potential for initiating a transient. For example, a fire in the Engineered Safety Features (ESF) pump room not only would disable the ESF pumps but could possibly initiate a transient due to the fire damage to the cables running through the room and the ESF equipment. It is assumed that all the ESF equipment in the effected division would be disabled as a result of this type of fire. The other type is a fire in an area which by itself could disable safety-related equipment but would require the penetration of a fire barrier in order to initiate a transient. For example, a fire in the diesel generator room would disable that diesel generator but would initiate a transient only if it penetrates the fire barrier to another area or room. Manual detection and suppression of the fire was not credited in the scoping evaluation.

The first type of fire was designated as type "a" and the second type as type "b". The fire event initiating frequency (IE_{fires}) is determined by using the equation:

$$IE_{fires} = [F_a * (F_d + F_s)] + [F_b * F_{fb} * (F_d + F_s)]$$

where

F_a = Fire occurrence frequency for type "a" fires (event/year),
F_b = Fire occurrence frequency for type "b" fires (event/year),
F_d = Failure rate of the (automatic) detection (failure/demand),
F_s = Failure rate of the (automatic) suppression (failure/demand), and
F_{fb} = Failure rate of the fire barriers (failure/demand)

Type "a" fires included fires in the Auxiliary Building, the Switchgear Room and the Cable Spreading Room, and type "b" fires included fires in the Diesel Generator Room and the Battery Room. Reference 45 (Table 1.2 in Attachment 10.3) provides the fire occurrence frequencies for types "a" and "b" fires. The ~~types of~~ fires and their occurrence frequencies are summarized in Table 19.7.3.2-1.

generic

areas

used in the System 80+ scoping assessment

$$F_a = F_{aux\ bldg} + F_{switchgear} + F_{cable\ spreading} + F_{SSWP\ bldg}$$
$$= 3.8E\text{-}02 + 1.5E\text{-}02 + 3.2E\text{-}03 + 3.8E\text{-}02$$
$$= \cancel{5.62E\text{-}02}\ 9.42E\text{-}02 \text{ event/year}$$

$$F_b = F_{diesel\ generator} + F_{battery}$$
$$= 2.84\text{-}02 + 3.2E\text{-}03$$
$$= 3.16E\text{-}02 \text{ event/year}$$

Reference 45 (Reference Table 2 in Attachment 10.3) provides failure rates for the suppression systems. Because detailed design information for System 80+ plant is not available at this time, the automatic suppression system reliabilities were averaged for the five types of the suppression system to generate an estimate of the System 80+ fire suppression system failure rate.

$$F_s = 4.2E\text{-}02 \text{ failure/demand}$$

A large range of values was found for smoke and thermal detector failure rates. The value used in this scoping evaluation was the mean value of the log-normal distribution for failure rates, defined using the lowest value of 4.0E-04 (Reference 102) as the 5% confidence lower bound and the highest value of 8.0E-01 (Reference 103) as the 95% confidence upper bound. The resulting value was 2.0E-01 failure per demand. That is,

$$F_d = 2.0E\text{-}01 \text{ failure/demand}$$

Reference 104 provides failure rates for the fire barriers. Barrier type 3 was used as the reference type since the System 80+ plant design utilizes 3-hour fire barriers between the fire zones. The (best) estimate value of 1.2E-03 failure per demand was selected as the mean. That is,

$$F_{fb} = 1.2E\text{-}03 \text{ failure/demand}$$

Hence, the fire event initiating frequency is

9.42E-02

$$IE_{fires} = [F_a * (F_d + F_s)] + [F_b * F_{fb} * (F_d + F_s)]$$
$$= [\ 5.62E\text{-}02 * (2.0E\text{-}01 + 4.2E\text{-}02)\ ] +$$
$$[\ 3.16E\text{-}02 * 1.2E\text{-}03 * (2.0E\text{-}01 + 4.2E\text{-}02)\ ]$$
$$= \cancel{1.36E\text{-}01 + 9.1E\text{-}06}\ 2.28E\text{-}02 + 9.1E\text{-}06$$
$$IE_{fires} = \cancel{1.36E\text{-}02}\ 2.28E\text{-}02 \text{ event/year}$$

Since the failure could occur in either of the two divisions, the fire event frequency would be

$$IE_{fires} = 2 * \cancel{1.36E\text{-}02}\ 2.28E\text{-}02 = \cancel{2.72E\text{-}02}\ 4.56E\text{-}02 \text{ event/year}$$

TABLE 19.7.3.1-6

DRAWINGS REFERRED TO IN THE FIRE ANALYSIS

1. CESSAR-DC Figure 9.5.1-2, ~~Drawing K200-01 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 50+0.
2. CESSAR-DC Figure 9.5.1-3 ~~Drawing K200-02 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 70+0.
3. CESSAR-DC Figure 9.5.1-4 ~~Drawing K200-02-01 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 81+0.
4. CESSAR-DC Figure 9.5.1-5 ~~Drawing K200-03 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 91+9.
5. CESSAR-DC Figure 9.5.1-6 ~~Drawing K200-04 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 115+6.
6. CESSAR-DC Figure 9.5.1-7 ~~Drawing K200-05 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 130+6.
7. CESSAR-DC Figure 9.5.1-8 ~~Drawing K200-06 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 146+0.
8. CESSAR-DC Figure 9.5.1-9 ~~Drawing K200-07 (4248-00-16019.7.00), Rev. 0~~, Nuclear Island Fire Barrier Locations Plan at Elevation 170+0.

TABLE 19.7.3.2-1

GENERIC FIRE IGNITION SOURCES AND FREQUENCIES ~~BY APPLICABLE FIRE AREAS~~ USED IN SYSTEM 80+

Fire Areas/Rooms	Fire Ignition Sources	Fire Type	Fire Frequency (Event/Year)
Auxiliary Building	Electrical Cabinets Pumps	a	1.9E-02 1.9E-02
Switchgear Room	Electrical Cabinets	a	1.5E-02
Cable Spreading Room	Electrical Cabinets	a	3.2E-03
Diesel Generator Room	Diesel Generators Electrical Cabinets	b	2.6E-02 2.4E-03
Battery Room	Batteries	b	3.2E-03
Reactor Building	Electrical Cabinets Pumps	x (see Note)	
Control Room	Electrical Cabinets	x	
Intake Structure	Electrical Cabinets ~~Fire~~ Pumps & Other	~~x~~a	1.9E-02 1.9E-02
Turbine Building	T/G Excitor, T/G Oil, T/G Hydrogen, Electrical Cabinets, Other Pumps, Boiler Main Feedwater Pumps	x	
Radwaste Area	Miscellaneous Components	x	
Transformer Yard	Yard Transformers	x	
Plant-Wide Components	Fire protection panels, Non-qualified cable run, junction box in qualified cable, Transformers, Battery Cahrgers, H_2 Tanks, Gas Turbines, Air Compressors, Ventilation Sub-systems, Dryers, etc.	x	

NOTE: For System 80+ plant, fires in these areas may initiate a transient but would not disable safety-related equipment and, therefore, are excluded from the scoping evaluation.

INTERNAL FLOODS

RAI 1:

Please identify, describe and characterize potential sources of internal floods by area, their capacity, the equipment that can be affected by the flood, flood barriers and potential passageways or penetrations through which floods can propagate to other rooms and buildings. Include figures (plans) showing such information, commensurate with available design details. For areas where no adequate design details are available, please identify potential input to ITAAC, RAP and COL items.

RESPONSE TO RAI 1:

A preliminary analysis was performed for the System 80+ Nuclear Island in the form of a calculation to demonstrate that internal flood sources would not cause flooding in the Nuclear Annex and Reactor Building Subsphere above elevation 70+0. The Nuclear Annex and Reactor Building Subsphere were divided into flood zones using the System 80+ flood barrier drawings. Attachment 2 contains figures detailing the different flood zones, including flood barriers, for the Nuclear Island. Using these figures and the System 80+ general arrangement drawing for elevation 50+0 (CESSAR-DC Figure 1.2-4), the applicable flood sources were determined for each flood zone.

Using the general arrangement drawing, the volume of each flood zone was calculated and adjusted to account for equipment volumes, internal walls, structures, etc. to determine the flood zone free volume. To account for equipment volumes, internal wall volumes, etc. a conservative assumption was made that 50% of the overall flood zone volume would be taken up by items internal to the flood zone. A major flood protection design feature of the System 80+design is the divisional wall which prevents flood waters from migrating to the opposite division. This prevents a flood in one division from affecting the other divisions of safety systems. The limited number of penetrations in this wall are sealed and no doors are provided up to elevation 70+0. Therefore, it is assumed the integrity of all flood barriers is maintained, preventing migration of flood waters to adjacent zones. Only one internal flood source release is assumed to occur at a time. The flood zone free volumes were compared to the volume of water from each of the expected internal flood sources applicable to the flood zone to demonstrate flooding above elevation 70+0 will not occur.

The following internal flood sources were determined to have the potential for release within the Nuclear Island:

Flood Source	Volume
Component Cooling Water System (CCWS)	24,700 ft^3
Incontainment Refueling Water Storage Tank (IRWST)	72,958 ft^3
Emergency Feedwater System (EFWS)	46,785 ft^3

Fire Protection System (FPS)		80,075 ft^3
Chemical and Volume Control System (CVCS)		161,075 ft^3
	TOTAL VOLUME	385,721 ft^3

The volumes of the internal flood sources were determined base on the following:

CCWS - This volume is the estimated volume of water contained in one division of CCWS

IRWST - This volume is based on the normal operating water volume of 545,800 gallons from CESSAR-DC Table 6.8-1.

EFWS - This volume is based on the volume of water contained in one EFW tank, 350,000 gallons, from CESSAR-DC Table 10.4.9-1.

FPS - This volume is base on the volume of the fire protection water supply tanks, 600,000 gallons, as given in CESSAR-DC Section 9.5.1. It is assumed the contents of the FPS piping does not significantly add to the volume of water contained in the tanks due to preaction valves which limit the amount of water contained within the system piping.

CVCS - This volume is based on the combined estimated volume of the Holdup Tank (525,000 gallons), Boric Acid Storage Tank (180,000 gallons), and the Reactor Makeup Water Tank (500,000 gallons). Actual internal volumes of these tanks as given in CESSAR-DC Table 9.3.4-4 are less than or equal to the estimated volumes. The volume of the water contained within CVCS piping was considered insignificant as compared to the combined volume of water in the tanks since it is highly unlikely that a pipe break in the CVCS would cause all three tanks to simultaneously drain to the Nuclear Island.

The following are the flood zone free volumes based on the general arrangement drawing of elevation 50+0 and assuming 50% of each flood zone's total volume is taken up equipment, internal walls, structures, etc.

Flood Zone		Volume
I.	Diesel Generator Area (DGA)	35,496 ft^3
II.	Control Complex Area (CCA)	134,071 ft^3
III.	Fuel Handling Area (FHA)	236,560 ft^3
IV.	Reactor Building Subsphere Area (RBSA)	61,512 ft^3
V.	Emergency Feedwater Pump Room (EFWPR)	5,014 ft^3
VI.	Chemical and Volume Control System Area (CVCSA)	284,109 ft^3

Areas identical to the above flood zones due to symmetry were not called out as separate flood zones since they have the same free volume and internal flood sources (i.e., DGA, CCA, RBSA).

To further illustrate the free volume of the Nuclear Island below elevation 70+0 by division, the volumes for each division are listed and totaled below:

Division I Free Volumes		Division II Free Volumes	
DGA	35,496 ft^3	DGA	35,496 ft^3
CCA	134,071 ft^3	CCA	134,075 ft^3
FHA	236,560 ft^3	CVCSA	284,109 ft^3
RBSA	61,512 ft^3	RBSA	61,512 ft^3
EFWPR (2)	10,028 ft^3	EFWPR (2)	10,028 ft^3
Total	477,667 ft^3	Total	525,216 ft^3

A description of each flood zone and the potential internal flood sources for each zone are described below:

Flood Zone I - Diesel Generator Area (DGA)

This zone contains one of the two emergency diesel generators and its associated diesel generator support systems.

The Component Cooling Water System is considered the only potential internal flood source for this zone since CCW is supplied to the diesel generator support systems within the DGA and is present during all modes of operation. Fire Protection System water is excluded from the DGA by preaction valves located outside the DGA flood barrier boundary. DGA flood barrier integrity prevents other flood sources from entering the DGA.

Comparing the free volume of the DGA (35,496 ft^3) and the volume of the CCWS flood source (24,700 ft^3) shows that the volume of flood water is much less than the free volume of this flood zone. Therefore, should there be a CCWS pipe break within the DGA, the resulting flood water will be contained within the affected DGA below elevation 70+0 an the emergency diesel generator in the opposite division will be unaffected by this flood.

Flood Zone II - Control Complex Area (CCA)

This zone contains the vital electrical distribution equipment, vital batteries, vital instrumentation, and Instrument Air System equipment.

The Component Cooling Water System is considered the only potential flood source for the CCA since CCW is supplied to the instrument air compressors with the zone and is present during all modes of operation. Fire Protection System Water is excluded from the CCA by preaction valves located outside the CCA flood barrier boundary. CCA flood barrier integrity prevents other flood sources from entering the CCA.

Comparing the free volume of the CCA (134,071 ft^3) and the volume of the CCWS flood source (24,700 ft^3) shows that the volume of flood water is significantly less than the free volume of this flood zone. Therefore, should there be a CCWS pipe bread within the CCA, the resulting flood water will be contained within the affected CCA below elevation 70+0 and the electrical equipment in the opposite division will be unaffected by this flood due to the integrity of the divisional wall.

Flood Zone III - Fuel Handling Area (FHA)

This zone include a portion of the Reactor Building Subsphere and the CVCS equipment area in Division I. Equipment located in this zone includes a CVCS charging pump and miniflow heat exchanger, a containment spray pump and heat exchanger, a safety injection pump, the containment cooler condensate pumps and tanks, the CVCS chemical addition package, two component cooling water pumps and various MCCs, MUXs, and panelboards.

The following are potential internal flood sources for the FHA zone:

CCWS - The CCWS is a large source of water present in the FHA, during all modes of operation. Based on the flood barrier arrangement and flood zone figure, the CCW pumps and suction lines are located within the FHA. In the event of a break in this moderate energy piping system within the FHA, the contents of one division of the CCWS has the potential to empty and drain to elevation 50+0.

EFWS - The EFW tank in this division is a large source of water present in the FHA during all modes of operation. In the event of a break in this moderate energy piping system within the FHA, the volume of the EFW Tank in this division ha the potential to empty and drain to elevation 50+0.

FPS - The FPS is a potential source of water which could enter the Nuclear Annex through FPS piping located in the FHA. In the event of a break in the FPS piping within the FHA the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.

IRWST - Should there be a break in the Containment Spray System or the Safety Injection System piping located in the Subsphere quadrant located within this flood zone, the contents of the IRWST has the potential to empty and drain to elevation 50+0.

Comparing the FHA flood zone free volume (236,560 ft^3) to each of the above potential internal flood source volumes shows that the FHA free volume is considerably larger than any of the flood source volumes. Therefore, flood water from either of the applicable flood sources will be contained within the flood zone below elevation 70+0 and the equipment in the opposite division will be unaffected by the flood.

Flood Zone IV - Reactor Building Subsphere Area (RBSA)

This zone excluded the turbine-driven emergency feedwater pump room. Equipment located within this zone include a shutdown cooling system pump, heat exchanger, and miniflow heat exchanger, a safety injection pump, electrical panels, MUXs, and MCCs.

The following are potential internal flood sources for the RBSA zone:

IRWST - Should the be a break in the Shutdown Cooling System or the Safety Injection System piping located in this subsphere quadrant, the contents of the IRWST has the potential to empty and drain to elevation 50+0

FPS - The FPS is a potential source of water which could enter the Reactor Building Subsphere through FPS piping located in the RBSA. In the event of a break in the FPS piping within the RBSA, the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.

Comparing the free volume of the RBSA (61,512 ft^3) to the internal flood source volumes for the IRWST (72,958 ft^3) and the FPS (80,203 ft^3) reveals that the water volume of both flood sources exceeds the free volume of the RBSA. Therefore, each of the flood sources will completely fill the flood zone. Although the RBSA is shown to completely fill with the flood sources, flood water is restricted to the flood zone by the flood barriers leaving the remaining three subsphere quadrants unaffected.

Flood Zone V - Emergency Feedwater Pump Room (EFWPR)

This zone contains a motor-driven emergency feedwater pump. The motor-driven EFW pump room was chosen as a flood zone since it is somewhat smaller in size to the turbine-driven EFW pump room. This flood zone will be considered typical of the four EFW pump rooms.

The following are potential internal flood sources for the EFWPR:

EFWS - The EFWS suction lines are located in each EFW pump room creating a path for the EFW tanks to empty into the room should there be a line break in this piping.

FPS - The FPS is considered a potential source of water which could flood the EFW pump room due to the presence of FPS piping within this flood zone.

CCWS - Component Cooling Water is provided to the motor-driven EFW pups and is therefore present within this flood zone during all modes of operation.

Comparing the free volume of the EFWPR (5,014 ft^3) to the internal flood source volumes shows that this flood zone will completely fill with each flood source considered. However, the flood barriers will confine flood waters to the room. The remaining EFW pump in the division as well as the two EFW pump in the opposite division will be unaffected.

Flood Zone VI - Chemical and Volume Control System Area (CVCSA)

This flood zone contains much of the CVCS equipment including one of the CVCS charging pumps and its miniflow heat exchanger. Also included with this zone a.·· the division II CCW pumps, a containment spray pump and heat exchanger, a safety injection pump, various electrical panels, MCCs, and MUXs.

The following are potential flood sources for the CVCSA:

CCWS - The CCWS is a large source of water present in the CVCSA, during all modes of operation. Based on the flood barrier arrangement and flood zone figure, the CCW pumps and suction lines are located within this flood zone. In the event of a break in this moderate energy piping system within the zone, the contents of one division of CCWS has the potential to empty and drain to elevation 50+0.

EFWS - The EFWS Tank in this division is a large source of water present in the CVCSA during all modes of operation. In the event of a break in this moderate energy piping system within this flood zone, the volume of the EFW Tank in this division has the potential to empty and drain to elevation 50+0.

IRWST - Should there be a break in the Containment Spray System or the Safety Injection System piping located in the Subsphere quadrant within this flood zone, the contents of the IRWST has the potential to empty and drain to elevation 50+0.

PFS - The FPS is a potential source of water which could enter the Nuclear Annex through FPS piping located in the CVCSA. In the event of a break in the FPS piping within this flood zone the volume of the two fire protection water supply tanks has the potential to empty and drain to elevation 50+0.

CVCS - The volume of the CVCS Tanks is assumed to be a potential source of water which could enter the Nuclear Annex through CVCS piping located throughout the CVCS Area, including upper elevations. For conservatism it is assumed the volume of the three largest tanks of the CVCS (Holdup Tank, Boric Acid Storage Tank, and Reactor Makeup Water Tank) are combined and drained to elevation 50+0 in the CVCS Area.

Comparing the free volume od the CVCS Area (284,109 ft^3) to each of the potential internal flood source volumes shows that the flood source volumes are significantly less than the free volume of this flood zone. Therefore, flood water from either of the applicable flood sources will be contained within the flood zone below elevation 70+0 and the equipment in the opposite division will be unaffected.

In looking at the estimated total free volume of each System 80+ division below elevation 70+0 (approximately 477,667 ft^3 in division I and 525,216 ft^3 in division II), and comparing it to the total volume of the potential flood sources (385,521 ft^3), it is evident that the Nuclear Island is of sufficient size to contain any of the postulated internal flood sources within a single division below elevation 70+0.

The COL applicant will provide a detailed flood analysis to verify the assumptions and results of this preliminary analysis.

RAI 2:

Please address the possibility that some floods, if not mitigated, could impact multiple systems. The Nuclear Island structures ITAAC shows very little separation above the lowest level in the reactor building. If this is the case, the possibility of a large flood source affecting both divisions of the safety systems may not be insignificant. It may be necessary to use event trees and appropriate system fault trees to model flood detection, mitigation (automatic and manual), and propagation. Sufficient design information should be available for this activity.

RESPONSE TO RAI 2:

Refer to response to RAI 1.

RAI 3:

The flood analysis should address the loss of all non-safety systems (due to the flood) credited in the PRA such as the feedwater system. The loss of component cooling water (CCW) event is assumed to bound the risk for all potential internal floods and flood areas. ABB C-E must provide justification for this assumption.

RESPONSE TO RAI 3:

In assuming that loss of one division of CCW bounds the risk for all potential internal floods and flood areas, the non-safety systems credited in the PRA are addressed accordingly. For example the charging pumps depend on CCW to remove heat from the pump motor coolers, and each charging pump is cooled from a different division of CCW. The effect on the charging system due to loss of one division of CCW is addressed and accounted for in the PRA. Other non-safety related systems such as the Instrument Air System are addressed and accounted for in a similar manner because of the divisional separation of these systems. The feedwater system is not credited in the PRA as a mitigating system. Therefore, loss of the feedwater system is already accounted for in the scoping analysis for internal floods.

As shown in CESSAR-DC Figures 9.5.1.2 through 9.5.1.9, the divisions of safety related equipment are physically separated. Within the division, certain equipment are further separated in various rooms or enclosed compartments. Because of the separation, it is assumed that the flood would most likely disable the one of the two pumps, and in the extreme both pumps. For this type of internal flooding, the plant would trip and respond to the initiator in a manner similar to loss of One division of the CCWS/SSWS. Therefore, the accident sequences identified for loss of one division of CCW/SSW were used to estimate the scoping value for core damage frequency due to internal floods.

RAI 4:

The statement is made that a flood in the turbine building would not affect any other system/area and therefore does not need to be analyzed. Please provide a bounding analysis showing that the probability of an unisolable leak in the service water system or the circulating water system propagating to the control or any other building housing safety related equipment is negligible. If this cannot be shown by a bounding analysis, please provided analysis, models and results. Show all assumptions made. Identify any potential areas that can be included as ITAAC, RAP and COL requirements.

RESPONSE TO RAI 4:

The Condenser Circulating Water System (CWS) is an out of scope item which shall be provided by the applicant. As such, interface requirements are provided (CESSAR-DC 10.4.5.1.2) to ensure adequacy with the System 80+ design. The following interface requirements pertains to flooding of the Circulating Water System:

1. "Means shall be provided to prevent or detect and control flooding of safety related areas so that the intended safety functions of a system or component will not be precluded due to leakage from the Condenser Circulating Water System. Malfunction or failure of a component or piping of the system shall not have unacceptable adverse effects on the functional performance capabilities of safety related systems or components."

2. "Flooding protection shall be provided to ensure that large leaks from circulating water piping do not result in the loss of all circulating water pumps."

Flooding due to a failure of a condenser water box expansion joint or circulating water piping (CESSAR-DC 10.4.1.3) would be limited to the Turbine Building which contains no safety related equipment. The excess water would be released from the Turbine Building to the yard through openings, doors, and additional openings caused by shearing of building siding bolts. The ground level of the Turbine Building is located above the finished plant grade level to ensure water drainage away from the Turbine Building. All Turbine Building interconnections and miscellaneous pipe tunnels to safety related structures will either be above the maximum internal flood level (associated with failure of the circulating water piping or condenser water box expansion joint), or sealed to prevent back-flooding. Flooding of safety related plant structures from Turbine Building sources is precluded since the plant grade is sloped away from safety related structures for proper drainage. Since the Turbine Building contains no safety related equipment, and no other building is affected by the Turbine Building flooding, the impact of internal flooding from the Turbine Building is limited to non-safety related equipment in the Turbine Building.

The major piping (main and feedwater) enters the Turbine Building above grade and if any interconnecting tunnels are required they will be sealed to prevent flooding propagation into the Nuclear Annex.

The Turbine Building Service Water System (TBSWS) removes heat from the Turbine Building Cooling Water System (TBCWS) and rejects the heat to the cooling towers. The TBSWS uses pumps to circulate circulating water from the cooling towers through the TBCWS heat exchangers and then discharge back to the CWS at a point between the main condenser cooling water outlet and the cooling tower inlet.

Based on the above summary of the CWS and TBSWS, flooding due to failure of TBSWS or CWS piping will not propagate to the control or any other building housing safety related equipment because these equipment are not located in the Turbine Building, and the safety related equipment are protected by safety related building flooding barriers and penetration seals. An unisolable leak in the CWS due to elevation (penstock effect) is a practical impossibility if the site is to meet the possible maximum flooding criteria. A flood in the Turbine Building is assumed to cause a loss of main and startup feedwater followed by a plant trip. For this event, the plant would respond in a manner similar to any other transient and is therefore included in this category of initiating events which is addressed in CESSAR-DC 19.4. (Note that the main feedwater system is not credited in the PRA as a mitigating system.) Since the safety related equipment are not affect by a flood in the turbine building, ABB C-E believes that a bounding analysis showing the probability of an unisolable leak in the TBSWS and the CWS is not warranted.

RAI 5:

The service water system (SWS) has the potential to be nearly unlimited flood source. This depends upon the configuration used, the relative position of the ultimate heat sink (i.e., at a higher elevation than the plant), and the design of the isolation and detection systems. Please provide information on the provisions made to ensure that this flood source can be isolated. The potential for this flood source flooding the entire NSW pump structure or the CCW heat exchanger building should be addressed. The ability to isolate the SWS, as well as the circulating water system, can have a significant impact and should be addressed in the internal flood analysis.

RESPONSE TO RAI 5:

The station service water intake structure contains the SSWS pumps and and the CCW heat exchanger structure contains the CCW heat exchangers. The pumps and heat exchangers in division I are completely separated from the pumps and heat exchangers in division II. The station service water and the CCW heat exchanger structures is constructed to limit the effects of a station service water break to one division of the CCW heat exchanger structure only. The tunnel connecting the CCW piping from the Nuclear Annex to the CCW heat exchanger structure is sealed at the entrance to the pipe tunnel such that even if the CCW heat exchanger division fills to the top of the structure and overflows, no service water will enter the CCW pipe tunnel and flow toward the Nuclear Annex. A break in the CCW piping in the connecting tunnel will be limited to the contents of one division of the closed loop system. If the tunnel is relatively long due to placement in the yard, it is likely that the entire contents will be contained in the tunnel itself. If the tunnel is short, due to placement adjacent to the Nuclear Annex, then the effect will be on that one division only, since the bottom elevation volume of that section of the Nuclear Annex is more than adequate to contain the CCW division contents, and is sealed by internal flood barriers from propagating to the other division. Flooding of the station service water structure to include all divisions requires a violation of the possible maximum flooding and siting criteria, since each service water division is flood protected from external as well as credible internal flooding. Drowning the service water pump motors with an internal flood is an effective way of shutting off the motive power to the source of the flooding of that division, and it will not affect the other division. As mentioned in the response to RAI 4, an unisolable flood source due to elevation (penstock effect) is a violation of the siting criteria for possible maximum flood.

RAI 6:

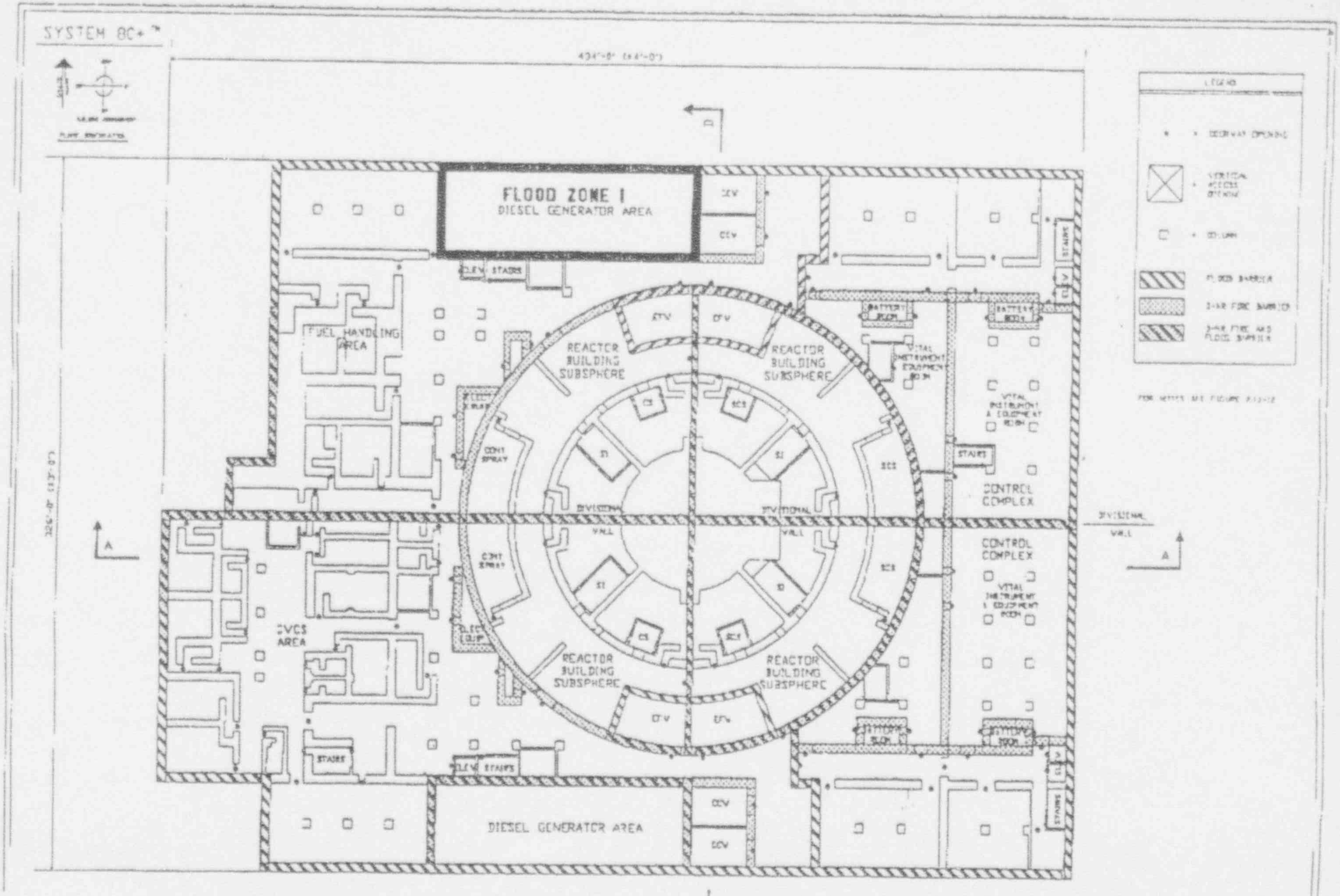
The internal flood analysis concludes that the worst possible scenario following an internal flood event would affect only one division of the Engineered Safety Features (ESF) equipment. This separation among divisions is a very important feature of the System 80+ design. For this reason, wall and flood barrier integrity as well as the requirements for any penetrations in the wall separating the two divisions must meet should be addressed in the ITAAC.

RESPONSE TO RAI 6:

The requirements for flood barriers between the two divisions of ESF equipment are addressed in the ITAAC which are provided under a separate submittal.

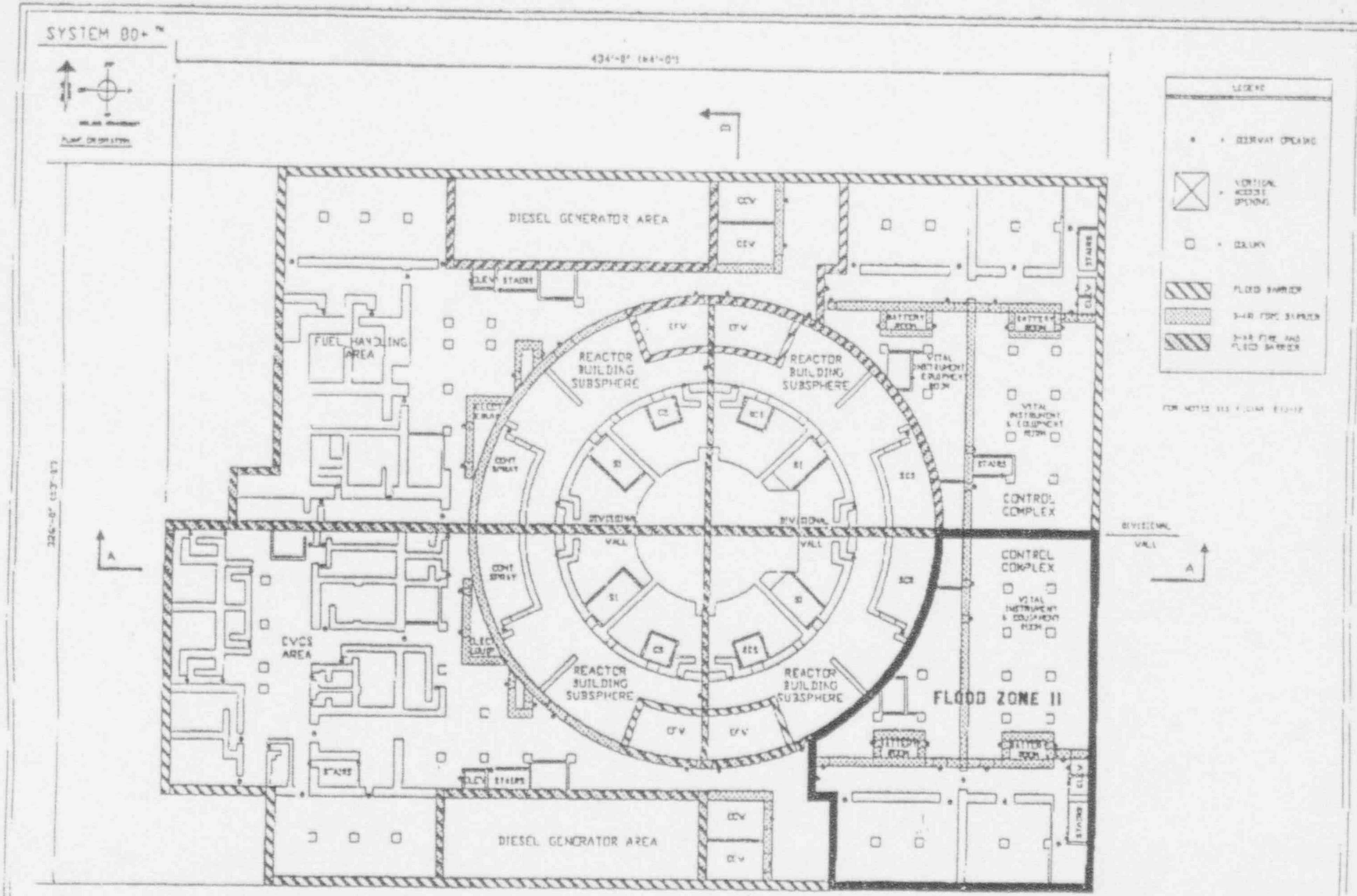
ATTACHMENT 2

FLOOD ZONE DRAWINGS



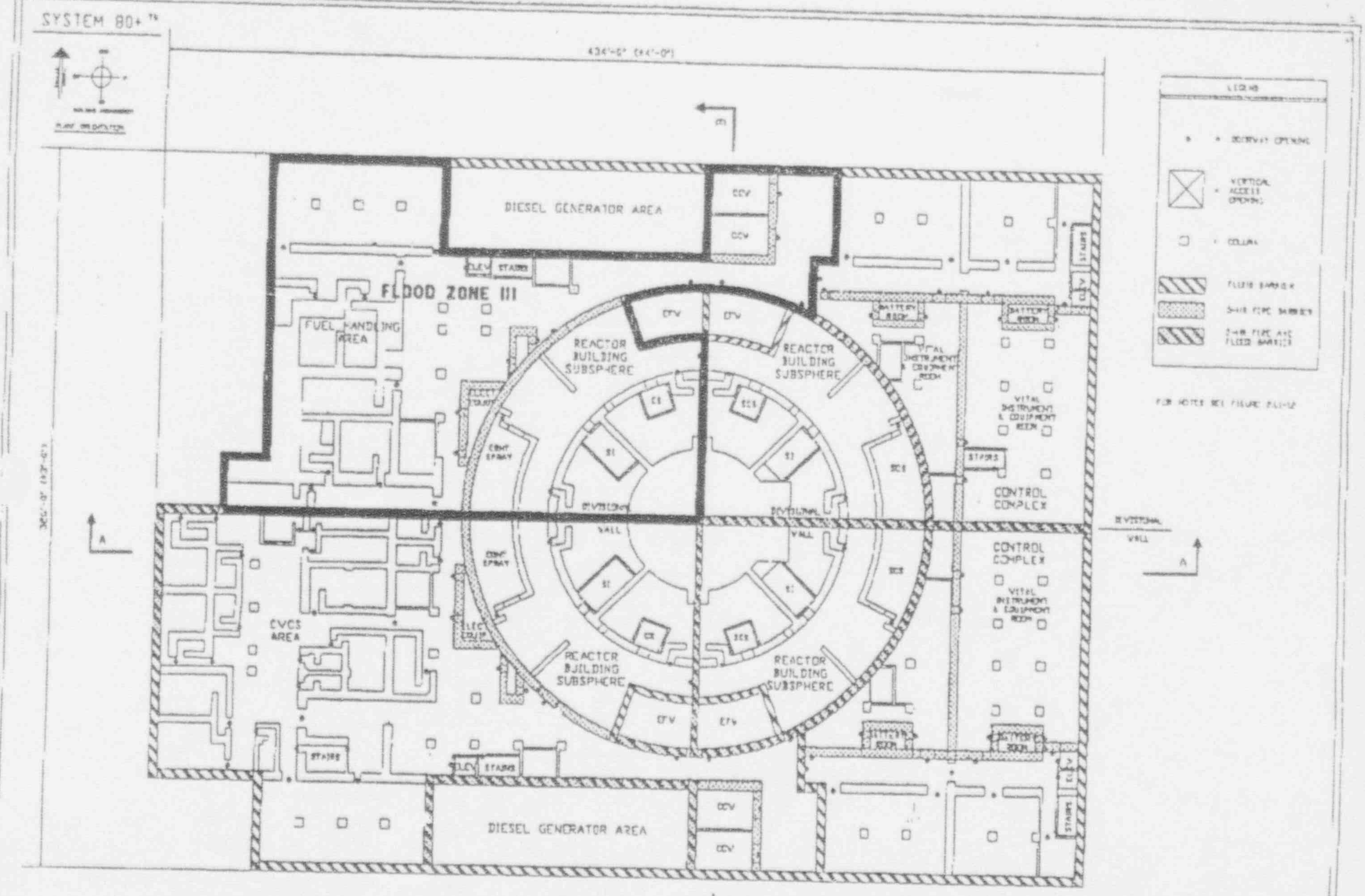
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 211-3



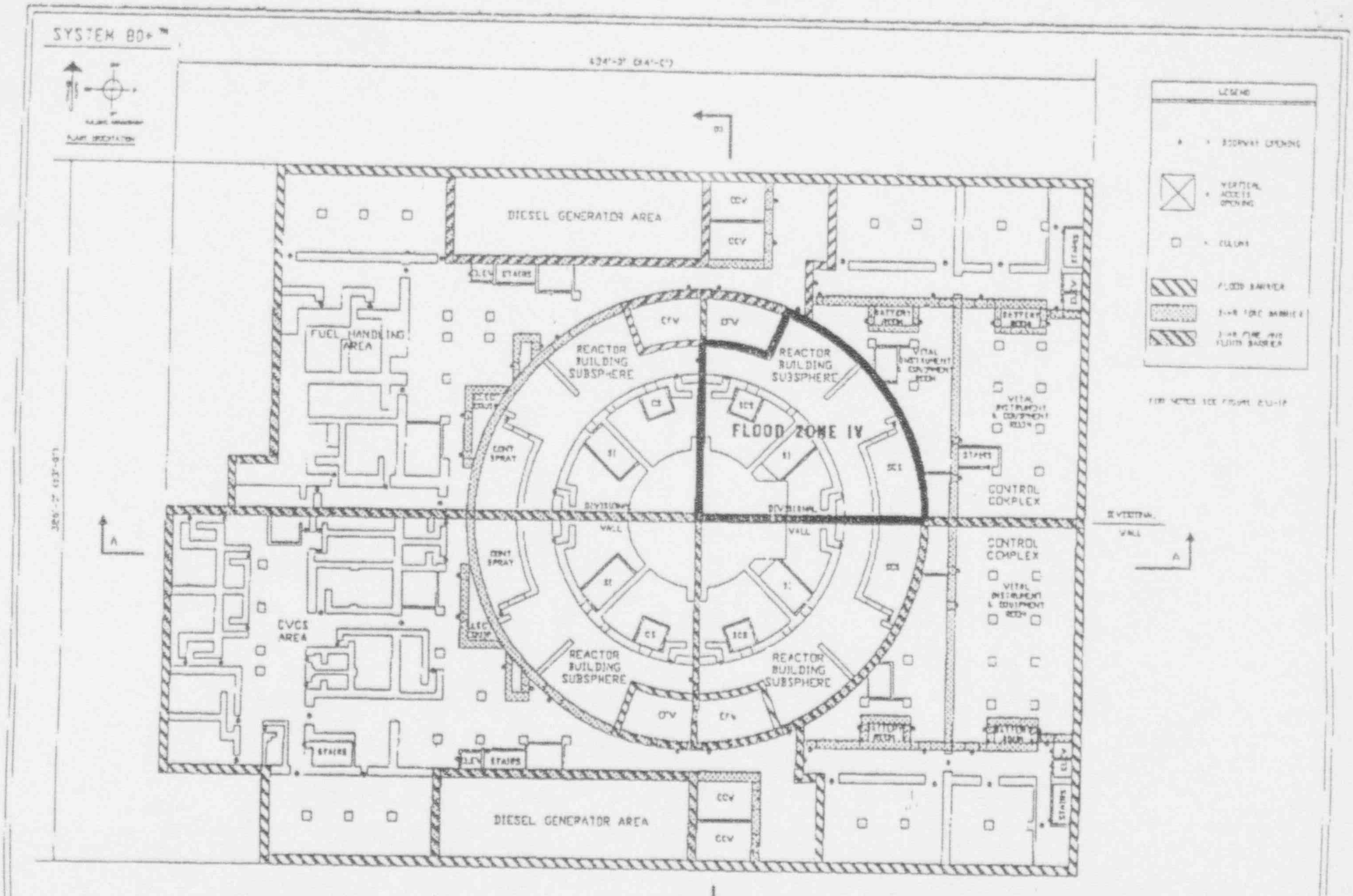
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 2.11-3



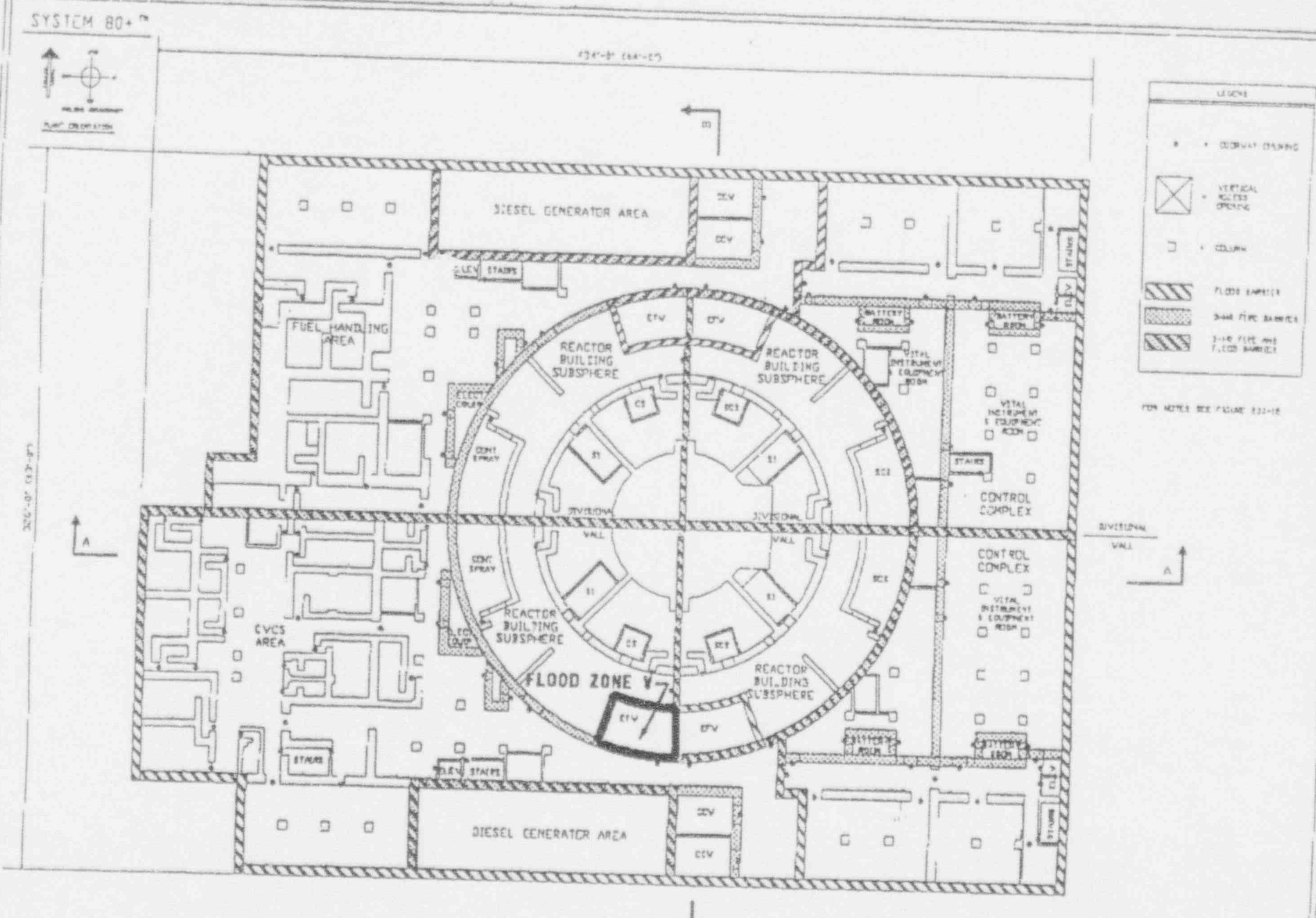
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 2.11-3



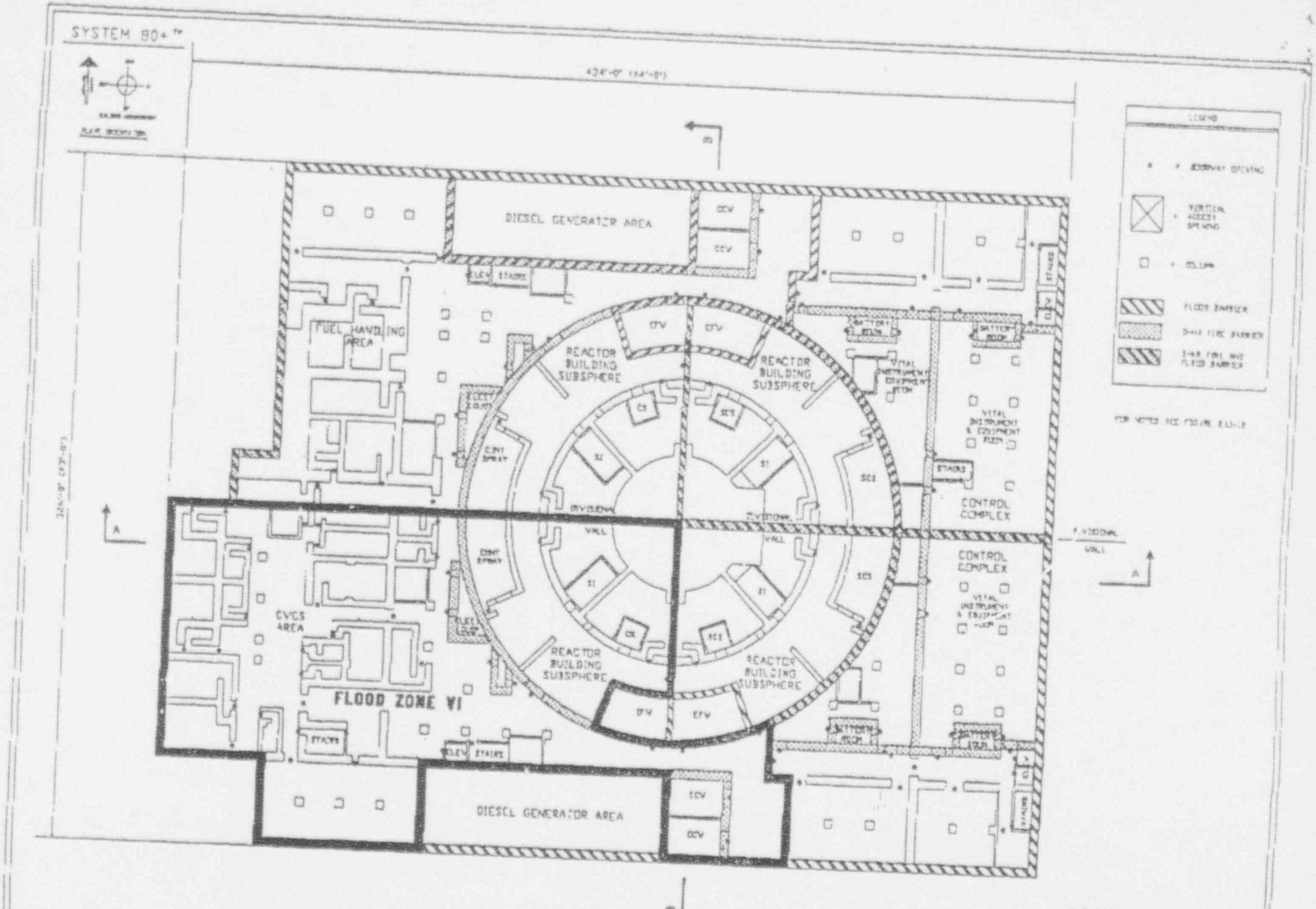
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 2.1-3



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 21.1-3



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 2.1.1-3

ATTACHMENT 2

TASK ANALYSIS REVIEW

Overview of Issues

<u>GOAL</u>: Approve TA methodology as part of System 80+ certification review (per 28 Sept 92 Meeting)

<u>SUMMARY OF DSER FINDINGS</u>:

SCOPE - 1. Specify greater breadth of TA (more scenarios)
- 2. Commit to comprehensive TA
- 3. Address maintenance issues

METHOD - 4. Address workload
- 5. Address crew characteristics & interactions
- 6. Address critical TA

<u>RESOLUTION ACTIVITIES</u>:

- 10/11 Sept 92 Meeting (commitments) documented by NRC
- ABB-CE submissions & responses
- Draft Technical Report for NRC by BNL: Comments on System 80+ TA

TASK ANALYSIS REVIEW

Table 1: Scope of Analyzed Scenarios

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
1.1	1	A(1). Full range of op modes should incl LP & AOPs (more breadth)	- More depth & detail desired (p7 4.1.2/1b)	- Requested breadth added in C-DC 18.5.1.5.1 (draft) of LD-93-005, 18.7-1, 1/18/93
1.2	1	B(2). Justify lack of LP ops in RCS TA		- S/U included; - Mid-loop ops & Physics tests N/A to RCS panel tasks
1.3	1	E. Single failure of DPS, DIAS, IPSO	- OK (p4 4.1.1/E)	- Required Availability is redundant between systems; - Committed to TA in C-DC 18.5.1.5.1; - Committed to V&V in NPX80-IC-VP790-03

ABB-CE Position: Commitments are satisfied by responses.

TASK ANALYSIS REVIEW

Table 2: Documentation of Findings

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
2.1	4	B. Document the resolution of RCS TA (NPX80-IC-DP790-02) App I findings	- Documentation not received (p3 & 4, 4.1.1/B)	- Findings obviated by redesign of DPS navigation scheme; - Issues in TOI to ensure closure; - W/L & time response treated iteratively by TA

ABB-CE Position: Item is confirmatory.

TASK ANALYSIS REVIEW

Table 3: HRA/Critical Tasks

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
3.1	6	C. Commit to identify critical tasks via HRA for inclusicn in TA	- Methodology does not identify specific critical tasks (p9 4.2.2/E & 4.2.3/2)	- Committed to HRA-ID'd critical tasks in A-1.4 of HFPP - Specification of tasks is PRA/HRA result, not methodology

ABB-CE Position: Commitments are satisfied by responses.

TASK ANALYSIS REVIEW

Table 4: TA Data Model (page 1 of 3)

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
		D. Address details of PRM Criterion 3:		
4.1	2	(a) Information requirements	- OK (p10 4.2.3/3a)	- Addressed in RCS TA and C-DC 18.5
4.2	2	(b) Decision making requirements	- Complex decision making, errors, K-based behavior not addressed (p8 4.2.1/3&5; p10 4.2.3/3b)	- Decision making within structure of tasks (18.5.1.1.D, E, F); - Hi W/L ==> ELSs, more analysis (18.5.1.1.H); - Errors screened via HRA (Item C above); - K-based behavior is non-observable, generally below grain of TA
4.3	2	(c) Response requirements	- Use of Middleman in RCS TA in-appropriate (p8 4.2.1/4; p10 4.2.3/3c-1) - Interactions/ overlap not addressed (p11 4.2.3/3c-2) - Machine limits not addressed (p11 4.2.3/3c-3) - Body movements not addressed (p11 4.2.3/3c-4)	- Use of human perf models revised (18.5.1.4) "slowman" to be default - Additivity assumed (18.5.1.1.F), final acceptability validated; - I&C system responses not limiting in screening model - Movements unrestricted given anthropometry specs, DPS flexibility

TASK ANALYSIS REVIEW

Table 4: TA Data Model (page 2 of 3)

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
		D. Address details of PRM Criterion 3:		
4.4	2	(d) Feedback requirements	- OK (p11 4.2.3/3d)	- Addressed in RCS TA and C-DC 18.5
4.5	2, 4	(e) Workload requirements	- Consider concurrent tasks, multi-person tasks, complex decision-making (p11 4.2.3/3e)	- Cognitive W/L addressed in RCS TA & C-DC 18.5 by response time evals; - Physical W/L considered non-limiting for CR tasks - Additivity & Staffing assumptions (18.5.1.1) treat multi-person tasks - See also (b) above
4.6	2	(f) Support requirements	- Describe/justify degree to which support rqmts will/won't be addressed (p5 4.1.1/G; p12 4.2.3/3f)	- Addressed by analysis (RAI response 620.28); - Residual issues treated as TA miscellany*, where significant

* <u>TA Miscellany</u> refers to items of potential significance, whose frequency as a substantive issue is too low to justify making it a separate data category. Issues categorized as miscellany should be outside the main focus of the TA. However, substantive findings beyond any particular data model should be addressed, and TA is a typical source of such findings.

TASK ANALYSIS REVIEW

Table 4: TA Data Model (page 3 of 3)

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
		D. Address details of PRM Criterion 3:		
4.7	2	(g) Workplace factors	- Address workspace envelope, movements btwn panels (p12 4.2.3/3g)	- Addressed by analysis (RAI response 620.28), HSI design guidance, V&V; - Discretionary link analyses TBD for benefit of design; - Residual issues treated as TA miscellany, where significant; - See also (c), (e) above
4.8	2, 5	(h) Staffing & communication requirements	- Complex interactions of steps, personnel not addressed (p12 4.2.3/3h)	- 10CFR50.5[illegible] rqmts accommodat[illegible] facie - C-DC 18.[illegible] F, I; - 2.3 [illegible] - C-DC 1[illegible] - Validation activities - Plasticity indicates COL applicant option - See also (c), (e) above
4.9	2	(i) Hazard Identification	- OK (for CR); not addressed outside CR (p13 4.2.3/3i)	- Residual issue treated as TA miscellany, where significant

ABB-CE Position: Commitments are satisfied by response : add requisite assumptions to 18.5, make item confirmatory.

TASK ANALYSIS REVIEW

Table 5: Position Descriptions

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
5.1	5	F. Provide position descriptions for CR personnel	- OK (p.4, 4.1.1/F)	- Addressed in LD-92-065 & LD-93-005

ABB-CE Position: Commitments are satisfied by responses.

TASK ANALYSIS REVIEW

Table 6: Crew Interaction

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
6.1	2, 5	G. Justify no TA for: (a) crew interactions in CR; (b) interactions between CR & plant	Has not provided: - analyses to indicate CR supports coordinated activity of crew, or min staffing assumptions; - detailed task descriptions addressing staffing & communication requirements;	- W/in CR, crew interaction below grain of screening TA, not limiting; - Btwn CR & plant, a residual issue treated as TA miscellany, where significant - Acceptability of <u>design</u> subject to Validation - Plasticity indicates COL applicant option; - See D(c, e, g, & h) above
6.2	2, 5	H. Discuss position on TA for 1, 3, and 6-person crews	- analysis of panel transitions to evaluate W/L (p5 4.1.1/G&H)	- General position: Minimum staff held to be limiting W/L case for TA; - Min & augmented CR staffing subject to Validation; - HFPP 2.3 & C-DC 18.6.2 gives bases for staff lvls; - HFPP A-3.4.2.5 identifies rqmts for TA staffing assumptions; - C-DC 18.5.1.1.I addresses method

ABB-CE Position: Commitments are satisfied by responses.

TASK ANALYSIS REVIEW

Table 7: Treatment of Maintenance Issues

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
7.1	3	I. Commit to address CR indication of equipment availability & impact of maintenance/ unavailability on operations	Described in LD-93-005 (p6 4.1.1/I), but: - Does not describe how crew will track status of OOS equipment... should consider how to present info to crew; - Contradiction...in statements that W/O processing is performed in/outside of CR; - Clarify where & by whom W/O tracking & tagout handled; - Detail how TA will address W/O generation & equipment tagout (p6 4.1.2/1a); - Address issues via TA or other means;	- Addressed in LD-93-005; - Indication of status of unavailable equipment explicitly provided through DPS & SPMS; - Work Order (W/O) & tagout activities virtually excluded from CR controlling workspace (not from MCR facility); - W/O & tagout processing otherwise admin'ly controlled by COL applicant; out-of-scope for Nuplex 80+ design; - HSI design and V&V (but not TA) are forums to eval means of display
7.2	3	G(e). Justify no TA for maintenance, inspection, & test activities in CR	- Describe how HF program addresses maintenance on CR equipment (p6 4.1.2/1a)	- Redundant info & control capabilities; no need to repair in abnml ops (see E above); - Committed to V&V for surveillances in V&V plan

ABB-CE Position: Existing design and process satisfies original DSER concern.

TASK ANALYSIS REVIEW

Table 8: Residual Issues

Table Ref #	DSER Area	Sept 10/11th Mtg Commitment	Draft Tech Report Comment	ABB-CE Response
		G. Justify no TA for:		
8.1	2	(c) Supporting materials		- See Table D(f) above (Table Ref # 4.6)
8.2	2	(d) Needed info unidentified by generic process	- Did not address (p5 4.1.1/G)	- Design & analysis is standard, not generic
8.3	2, 5	(f) Training program input		- Interface item for COL applicant via OSIP

ABB-CE Position: Commitments are satisfied by responses.

ATTACHMENT 2

Justifications of ABB Positions
Requested for Closure
of Task Analysis

TA REVIEW ISSUE 1.1

NRC Comment: The full range of operating modes should include "low power" operations and AOPs to provide a representative set of events for Task Analysis (TA).

ABB Response: The TA Methodology was revised to increase its scope in Amendment N of CESSAR-DC (Section 18.5.1.5.1). A future amendment will add explicit reference to a Loss of SCS for a Midloop scenario, and Loss of Offsite Power without Emergency Diesel Generators, as requested by the Staff. This response was agreed to comprise an acceptable resolution to the issue.

TA REVIEW ISSUE 1.2

NRC Comment: The lack of low power operations in the RCS TA should be justified.

ABB Response: Several distinctly different "low power" events receive treatment in the System 80+ TA, but the tasks for which TA data were generated using the RCS panel were limited to those with substantial RCS panel interactions. One such "low power" event is Plant Startup; this was included in the RCS TA work. Other low power events, such as Physics Testing and Midloop operations, are less relevant to the RCS Panel because they do not rely on the RCS Panel equipment (e.g., RCS pumps are secured at midloop). In addition, all scenario data are entered as updates to a common TA database. Thus, complete coverage of all panels by all scenarios occurs via successive iterations of the TA performed for the remaining panel designs. Impact of information gained through successive iterations will be factored into the RCS panel design. This response was agreed to comprise an acceptable resolution to the issue.

TA REVIEW ISSUE 1.3

NRC Comment: The full range of operating modes should include Single Failure of the Data Processing System (DPS), the Discrete Indication and Alarm System (DIAS), and the Integrated Process Status Overview (IPSO), in the representative set of Task Analysis (TA) events.

ABB Response: The TA Methodology was revised as requested to increase its scope to include design basis failures of the DIAS and DPS, in Amendment N of CESSAR-DC (Section 18.5.1.5.1). To perform design basis operations with the hypothesized single failure, DIAS provides sufficient minimum redundancy with (i.e., is an information subset of) DPS. Commitments in the V&V plan ensure that this Availability will be Verified, and the scenario performance Validated.

The IPSO display hardware is not itself required because the same data presentation exists in the DPS. Therefore this scenario was not included for analysis in the TA.

This response was agreed to comprise an acceptable resolution to the issue.

TA REVIEW ISSUE 2.1

NRC Comment: The findings of the RCS Task Analysis (TA) in Appendix I of NPX80-IC-DP790-02 should be resolved.

ABB Response: These issues have received three treatments, of which each alone should be sufficient for resolution. First, following the RCS TA, the DPS access scheme was totally revised to improve access performance. Thus, the general display access problem has been addressed, and the specific TA results have been obviated. Second, the Appendix I findings were entered in the TOI database to ensure closure. This material has been presented to the staff. Finally, workload and time response are treated iteratively by the TA; thus, the original testing will be repeated for the improved design. This response was found acceptable by the Staff.

TA REVIEW ISSUE 3.1

NRC Comment: Commit to identify HRA critical tasks via PRA for inclusion in the Task Analysis (TA).

ABB Response: This commitment already could be found in Section A-1.4 (Task Analysis) of the System 80+ Human Factors Program Plan. In addition, the TA Methodology in Section 18.5.1 of CESSAR-DC will be modified in a future amendment to include the following statement:

> "In addition to the representative event sequences in Section 18.5.1.5.1, the System 80+ PRA and associated Human Reliability Analyses (HRA) are used to identify "Critical Tasks." These are operator tasks indicated by PRA to make a significant contribution to total plant risk. Critical Tasks are incorporated as separate event sequences in the Task Analysis database. Findings from the associated HRA and TA are dispositioned through the formal documentation and tracking mechanisms of the Human Factors Program Plan"

See Attachment 3,
Section 18.5.1.5.2

The draft Technical Evaluation Report commented that the Critical Tasks identified thus far are not identified in the TA Methodology of 18.5.1. However, since these comprise results rather than methodology, and as such are subject to revision with the PRA, the Staff found this to be a reasonable position that completes an acceptable response to the overall issue.

TA REVIEW ISSUE 4.1

NRC Comment: Address details of <u>information requirements</u> per PRM Criterion 3.

ABB Response: The draft Technical Evaluation Report considered these to be adequately addressed in the present TA Methodology (CESSAR-DC Section 18.5) and the RCS TA Report (NPX80-IC-DP790-01).

TA REVIEW ISSUE 4.2

NRC Comment: Address details of <u>decision-making requirements</u> per PRM Criterion 3.

ABB Response: Three sets of issues were identified in the comments: 1) Complex decision making; 2) Errors; and 3) Knowledge-based behavior. The Staff's questions were addressed to their satisfaction as follows:

1) The TA deals with decision making as it is structured by the procedures and the operating sequences. This is explicitly addressed by Assumptions D, E, and F in Section 18.5.1.1 of CESSAR-DC. Although information requirements necessary to evaluate all procedural decisions (including contingencies) will be addressed, the event sequences and time response evaluations will not necessarily exercise each possible decision contingency. However, care will be taken in the development of TA scenarios to ensure that they address a range of complexity in terms of demands on operator performance, and are not limited to straight-forward or low-demand cases.

2) Although the TA addresses correct rather than erroneous behavior, there are three independent methods by which the TA methodology will specifically address and reduce the likelihood of human errors. One method is through workload analysis: scenario sections producing high workload in the screening analysis are considered error-likely situations, and receive a more detailed assessment. A second method is through the PRA (see the Response to Issue 3.1). A third method is through the observations and remarks of the task analyst (see the Response to Issue 4.6).

3) The present TA methodology is focussed on rule-based (e.g. procedural) rather than knowledge-based (e.g., reasoning) behavior. This reflects the purpose of the TA, which is to support design. A training-oriented TA would place more emphasis on knowledge-based behavior; however, this is a COL applicant issue (see Response to Issue 8.3). In turn, training will tend to reduce this material to rules, where possible. In the tasks addressed by the present TA, the plant designer wishes to <u>minimize</u> the need for operators to engage in complex knowledge-based behavior (it is the most creative, but the most unreliable). For these reasons, few such tasks are found in procedures.

<u>**TA REVIEW ISSUE 4.3**</u>

NRC Comment: Address details of <u>response requirements</u> per PRM Criterion 3.

ABB Response: Three sets of issues were identified in the comments: 1) Workload & response time measurement; 2) Body movements; and 3) Interactions & overlap of responses. The Staff's questions were addressed to their satisfaction as follows:

1) Improvements were made to the use of human performance models and Loading Criteria in Amendment N of the TA Methodology. Also, explicit commitments to apply conservative criteria for human and machine performance are being incorporated in Section 18.5.1.4 (see Attachment 3). Due to their relatively brief duration, machine responses are not limiting in the proposed screening model, but may be significant in (and are incorporated by) the more detailed model.

2) Body movements are virtually unrestricted in the control room. Anthropometry specs ensure acceptability of intra-workstation movement requirements. Inter-workstation movement requirements are bounded by those of conventional control rooms, but their number and magnitude will be significantly reduced by DPS flexible access, IPSO and DIAS visibility, and the more compact and integrated layout. Loss of DPS is tentatively considered the limiting scenario for movement requirements, and is analyzed in both TA and V&V activities.

3) Task response overlap and interactions are dealt with by assuming additivity of task resources and requirements (18.5.1.1.F). This is acceptable for the purpose and granularity of the TA methodology and the inclusive human performance models, and is consistent with similar applications throughout the literature. Acceptability of the final design and its resulting task interactions is evaluated by Validation testing.

TA REVIEW ISSUE 4.4

NRC Comment: Address details of feedback requirements per PRM Criterion 3.

ABB Response: The draft Technical Evaluation Report considered these to be adequately addressed in the present TA Methodology (CESSAR-DC Section 18.5) and the RCS TA Report (NPX80-IC-DP790-01).

TA REVIEW ISSUE 4.5

NRC Comment: Address details of workload requirements per PRM Criterion 3.

ABB Response: Cognitive Workload (W/L) is addressed in the RCS TA and the TA Methodology by response time evaluation. This is a typical and acceptable approach for formal W/L evaluation. Physical W/L is considered non-limiting for CR tasks performed by normal (i.e., 5th-95th percentile) healthy members of the general working-age population. Evaluation of local tasks shall incorporate adjustments to the methodology as appropriate. Also, an additional margin of conservatism is added to the time response criteria for local tasks (see Section 18.5.1.4.A of CESSAR-DC). Predicted and subjective workloads will also be assessed as part of Validation.

Other comments in the draft Technical Evaluation Report were highly correlated with issues addressed elsewhere in these responses. Concurrent tasks are addressed via the assumption of Additivity (see Issue 4.3, Response item 3, above). Multi-person tasks are further dealt with via the inclusion of Staffing assumptions (see Issues 4.8, 5.1, 6.1, and 6.2). Complex decision-making is addressed in Issue 4.2. Collectively this comprised an acceptable treatment of Issue 4.5 for the Staff.

TA REVIEW ISSUE 4.6

NRC Comment: Address details of support requirements per PRM Criterion 3.

ABB Response: Some support requirements have received formal analytic treatment (e.g., procedure storage, in RAI Response 620.28). Residual issues will be treated as TA Remarks, in keeping with the following revision (see Attachment 3) to the TA Methodology in CESSAR-DC:

> "Remarks accommodate extra notations or miscellaneous task requirements from data categories with infrequent significance. In the present task analysis, these issues could include, for example, specific workplace

suitability issues, task support requirements, communications requirements, crew interaction, or hazard identification."

Section 18.5.1.3.3

This response was agreed to comprise an acceptable resolution to the issue.

<u>TA REVIEW ISSUE 4.7</u>

NRC Comment: Address details of <u>workplace factors</u> per PRM Criterion 3.

ABB Response: Workplace factors are treated by a multitude of mechanisms in the design. The design of systems and components must conform to the HFE Standards, Guidelines, and Bases. The adequacy of the results are evaluated in design review, and during V&V (particularly within Suitability Verification.) Some formal assessment has also been performed (e.g., procedure storage, in RAI Response 620.28). Also, a commitment to link analysis for the Loss of DPS scenarios (limiting design basis cases for movement in the controlling workspace) has been added to Section 18.5.1.5.5 of CESSAR-DC.

Other comments in the draft Technical Evaluation Report were highly correlated with issues addressed elsewhere in these responses. Workspace envelopes and movements between panels are addressed under Issue 4.3, Response item 2. Staffing levels are addressed under Issue 4.8. Residual issues of significance will be treated under "Remarks" in the TA data (see Issue 4.6). Collectively this comprised an acceptable treatment of Issue 4.7 for the Staff.

<u>TA REVIEW ISSUE 4.8</u>

NRC Comment: Address details of <u>staffing & communication requirements</u> per PRM Criterion 3.

ABB Response: Staffing and communications requirements are addressed with a variety of mechanisms. The minimum staffing requirements of 10CFR50.54 are accommodated <u>prima facie</u> by the proposed design. Assumed staffing levels are treated as workload capacity by the TA Methodology (Assumption I of Section 18.5.1.1). Bases for the assumed staffing are provided both in Section 18.3.2 of CESSAR-DC, and Section 2.3 of the Human Factors Program Plan.

Choices for the control room configuration that considered the communication environment are also discussed in Section 18.6. Communications systems themselves have specified design (CESSAR-DC, Section 9.5.2) and applicable Human Factors Standards, Guidelines, and Bases (Section 6). Since the interaction of staff and their communications is not the focus of the present TA, the TA Methodology is not tailored to this purpose. In the screening model, for example, crew interaction within the control room is generally below the grain of the analysis. However, staff interactions will be observed in Validation activities, and the results made available to the COL applicant. The high flexibility of organizational and staffing decisions, communications protocols, and other programmatic system components suggest that there are multiple options to be pursued in these areas at the COL applicant's discretion.

Other comments in the draft Technical Evaluation Report were highly correlated with issues addressed elsewhere in these responses. Complex interactions are addressed via the assumption of Additivity (see Issue 4.3, Response item 3). Residual issues of significance will be treated under "Remarks" in the TA data (see Issue 4.6). Collectively this comprised an acceptable treatment of Issue 4.8 for the Staff.

<u>TA REVIEW ISSUE 4.9</u>

NRC Comment: Address details of <u>hazard identification</u> per PRM Criterion 3.

ABB Response: The draft Technical Evaluation Report considered this to be generally inapplicable to control room tasks (and therefore, acceptably treated by the TA Methodology) but questioned its treatment for tasks outside the control room. Since there will be relatively few tasks outside the control room in the TA, identified hazards will be residual issues and treated as TA Remarks (see Issue 4.6). Specific treatment of hazards by the TA has also been incorporated in Assumption J. This comprised an acceptable response to Issue 4.9 for the Staff.

TA REVIEW ISSUE 5.1

NRC Comment: Provide position descriptions for CR personnel.

ABB Response: The responses provided in LD-92-065 and LD-93-005 were considered acceptable by the draft Technical Evaluation Report.

TA REVIEW ISSUE 6.1

NRC Comment: Provide justification that no TA is performed to address a) crew interactions in the control room, or b) interactions between the control room and plant staff.

ABB Response: Issue 4.3 addressed the details of Response Requirements including workload & response time measurement, body movements, and interactions & overlap of responses. Staffing and Communication Requirements were addressed in Issue 4.8. Collectively these comprised an acceptable treatment of Issue 6.1 for the Staff.

TA REVIEW ISSUE 6.2

NRC Comment: Discuss position on TA for 1, 3, and 6-person crews.

ABB Response: In general, design basis minimum staffing is held to be the limiting workload case (i.e., <u>over</u>loading) for TA. This is consistent with the stated requirements for TA treatment of staffing and workload in the Human Factors Program Plan (Section A-3.4.2.5). Additional staff are considered to provide desirable workload capacity, within the design basis occupancy limits for the controlling workspace. Both minimum and augmented control room staffing levels will be subject to Validation. This, and the related Issues 4.3, 4.5, and 4.8, collectively comprised an acceptable treatment of Issue 6.2 for the Staff.

TA REVIEW ISSUE 7.1

NRC Comment: Commit to address control room indication of equipment availability, and impact of maintenance/unavailability on operations.

ABB Response: These issues were addressed in LD-93-005 with the following statement:

> "Tasks relating to maintenance work order tracking and tagout are not in the task analysis because these tasks will not be performed in the controlling workspace and have no impact on the control room HSI design. A separate facility to support maintenance work tag-out is provided adjacent to the main control room in the System 80+ design. Work order and tag-out tracking will be controlled by COL applicant administrative procedures. Nuplex 80+ HSI (both controls and monitoring systems) continuously [make available for] display the status of tagged-out components as "unavailable" [i.e., for operation] based on manually input data. Data input will not be required by the operators in the controlling workspace but by other personnel in the MCR. Data Processing System success path monitoring algorithms determine the impact of unavailable components on safety and non-safety success paths and respond to resulting unavailability conditions with alarms."

Section 18.7.1.1.8 of CESSAR-DC describes Nuplex 80+ manual information input capabilities, including component tag-out.

Clarifications requested in comments from the draft Technical Evaluation Report (identified in Table 7 of Attachment 1) are addressed here.

Work order and tagout activities have minimal direct impact on control room operations because these activities have been virtually excluded from the controlling workspace. The controlling workspace is that area bounded within the workstation consoles in the Main Control Room. Other workspace facilities are provided for tagout processing, within the control room, but outside the controlling workspace, to avoid interference with the panel operators.

In general, work order and tagout processing are otherwise administratively and programmatically controlled by the COL applicant, and are thus beyond the scope of System 80+ design and certification.

In sum, this comprised an acceptable treatment of Issue 7.1 for the Staff.

TA REVIEW ISSUE 7.2

NRC Comment: Justify no TA for maintenance, inspection, and test activities in the control room.

ABB Response: In general, the focus of TA is on rule-based plant operations. Maintenance forms a separate domain with a different set of problems and concerns. To deal with this divergence, an effort has been made to minimize the effects of maintenance on control room operations. One mechanism is the redundant indication and control capabilities provided by the multiple I&C systems (e.g., DIAS and DPS). This redundancy ensures that a single failure of such HSI components does not result in an immediate need to make repairs. A preferred time can be established to minimize interference with, and maximize safety of, plant operations.

In addition, a commitment to TA for surveillance activities (i.e., those potentially impacting operations) has been made in the TA Methodology (see Item O in Section 18.5.1.5.1 of Attachment 3). This comprised an acceptable treatment of Issue 7.2 for the Staff.

TA REVIEW ISSUE 8.1

NRC Comment: Provide justification for not completing a task analysis for equipment, documentation, and supplies required to support personnel during normal, abnormal, and emergency operations.

ABB Response: The Response to Issue 4.6 comprised an acceptable treatment of Issue 8.1 for the Staff.

TA REVIEW ISSUE 8.2

NRC Comment: Provide justification for not completing a task analysis for information needed for completing tasks or for reconstructing an event that may not be explicitly identified in the generic procedures.

ABB Response: Revision 3 of the Emergency Procedure Guidelines (EPGs) are generic in the sense that they applied to a set of similar (i.e., CE Owners) but non-standardized plants. The generic nature of the document changed with the revision to the System 80+ Emergency Procedure Technical Guidelines. This includes addition of any details found necessary to ensure the safety of the plant per its design bases and relevant analyses.

The task analyst elaborates the finer details of procedural tasks as a basis to assess the sufficiency of the available procedures, and to revise them if necessary to achieve sufficiency. Thus, TA does incorporate information that is not "explicitly identified" in the procedure guidelines. However, this should not be taken to mean that everything analyzed belongs in the procedures. This would lead logically to an upward spiral of textual volume, which itself leads to degraded usability.

Rather, the focus remains on the primary purpose of the TA (a design tool) and the procedure guidelines (a repository for technical operating information). Having each served successfully in the past, it is felt that evolutionary upgrades to both the TA methodology and the procedure guidelines form an appropriate and acceptable strategy for the Standard Design. The Staff has accepted this treatment of Issue 8.2.

TA REVIEW ISSUE 8.3

NRC Comment: Provide justification for not completing a Task Analysis (TA) for input to personnel training programs.

ABB Response: Although the present TA will be a useful input to the COL applicant training program (and will be so provided by ABB-CE), the purpose of the TA is to serve as a design tool. Thus, it remains a discretionary COL applicant issue as to how the TA database would be best enhanced to support training. This may depend on other aspects of the COL applicant's training program. It is in any case out of ABB-CE scope for design certification. The Staff has accepted this treatment of Issue 8.3.

<u>**TA REVIEW ISSUE A.1**</u> (Supplemental comment, 6/18/93)

NRC Comment: Indicate how new and modified functions (i.e., Rapid Depressurization, Hydrogen Ignitors, Alternate Generator, Startup Feed) will be addressed by the Task Analysis (TA; deferred from PRM Element 4 to 5).

ABB Response: The necessary uses of new and modified functions are specified in the procedure guidelines and operating sequences employed in the TA. The analytic scope of the TA will exercise the new and modified functions, extend the specified details of the operators' role from the function to the task level, identify human task performance requirements, and assess the resulting task loadings. Excessive loadings will result in further evaluation and formal resolution of the resulting allocation and design issues.

ATTACHMENT 3:

Revision to Section 18.5.1 of
CESSAR-DC, TA Methodology

18.5 FUNCTIONAL TASK ANALYSIS

Functional Task Analysis (FTA) is performed for the System 80+ plant as a formal part of the Nuplex 80+ ACC design process and Human Factors Program Plan (HFPP). FTA is a means to ensure that necessary operator tasks can be successfully performed. The FTA approach functionally decomposes the physical plant and its operations so that procedural tasks and decision processing can be analyzed independent of particular hardware implementations. The completed FTA provides the following analytic results for the design:

A. Procedure Guideline-based Information and Controls Requirements (PGICRs) for the control room human-system interface;

B. Operator task loading evaluations identify high workload situations for subsequent resolution;

C. Data on information and control usage by operators that supports the arrangement of physical components on the control panels.

The FTA methodology is based on the approach of References 1 and 2; details are provided in Section 18.5.1.

18.5.1 METHOD

The System 80+ FTA is based on the methodology used for the Combustion Engineering Owners Group (CEOG) Generic Operator Information and Control Requirements Review (References 1 and 2). This approach, developed and utilized to support the formal review of existing control rooms, has been modified to support a design process for the Nuplex 80+ ACC, particularly by incorporation of workload measures and criteria. The FTA methodology is presented in six major steps:

A. Establish assumptions and bases

B. Review input and design documentation

C. Establish task decomposition and data framework

D. Establish loading criteria

E. Perform analyses

F. Document results and conclusions

Details on each of these steps are provided in the remainder of Section 18.5.1.

18.5.1.1 Assumptions and Bases

The assumptions on which the FTA is based are specified as follows:

A. Evolutionary Design

The System 80+ design activities are being conducted to produce the next generation of Combustion Engineering's nuclear power plant. It is an evolutionary enhancement of a proven design - System 80. The functions and features of the Combustion Engineering System 80+ design are incremental revisions to this proven design, incorporating technological improvements and operating experience through a systematic design process.

B. Operator's Role

As a corollary of Assumption A, Nuplex 80+ is an advanced I&C implementation of existing MMI functions. Changes to the operators' role, and the tasks required to perform that role in support of operations, are minimal. Where changes have occurred, they aim to 1) resolve known problems, 2) retain successful aspects of existing control rooms, and 3) avoid new problems.

C. PGICRs

Procedure Guideline-based Information and Control Requirements resulting from the FTA will be afforded by the systems-based instrument and controls inventory and will be verified to be available in the control room, per HFPP requirements.

D. Event Sequences

Event sequences are representative examples of normal, abnormal, and emergency operating scenarios. Event sequences are generic cases based on the combined operator requirements of expected plant responses and proceduralized operating strategies (i.e., excluding complex interactions, error propagation, and sabotage.) The analysis of generic cases provides adequate data for the FTA's evaluation of operator workload and behavioral requirements. Selected event sequences are specified in Section 18.5.1.5.1; these sequences will be incorporated in validation activities per HFPP requirements.

E. Level of Detail

Event sequences are detailed by evaluating the necessary operator tasks per the applicable procedure guidelines (e.g., Reference 5) along a time line. Event sequences identify decision points and basic decisions, but do not pursue variations of these basic decisions into multiple contingencies.

F. Simple Additivity

The FTA will consider task elements to be additive and serially processed, unless otherwise noted. No general consideration is given to complex interactions of steps or personnel in the FTA. Formal evaluation of interactions will occur as part of human factors Validation activities.

G. Physical and Mental Workload

Regarding workload, the main concern in the FTA is with mental tasks in control center activities. The associated physical tasks are within the capabilities of the 5th percentile female operator. Exceptions to this assumption, such as might occur for a locally performed task, are documented in the data.

H. Workload and Human Error

The FTA evaluates operator workload as a comparison of the time available for a task, and the time required to perform it. Loadings which violate the acceptability criteria (see Section 18.5.1.4) are considered error-likely situations to be documented and resolved.

I. Staffing

The FTA considers staffing to be a form of workload capacity. Consistent with the concern for excessive workload, staffing is conservatively assumed to be at the design basis minimum level specified for each event sequence. However, staffing level will not impact the analysis unless a detailed evaluation (per Criterion B of 18.5.1.4) is made.

J. Environmental Hazards

The workspace environments in the Main Control Room and/or Remote Shutdown Room remain habitable for all design basis events and scenarios. However, local control stations included in the FTA shall be individually evaluated for personnel hazards as part of the evaluation of the specified operating sequences and tasks.

18.5.1.2 Input and Design Documentation Review

System 80+ includes design enhancements and improvements to address experience gained from earlier plant designs, criteria provided by the Advanced Light Water Reactor (ALWR) Requirement Document, and guidance from the NRC's Severe Accident Policy and Standardization Rule. Documentation for the System 80+ design has been reviewed to identify the plant processes, configurations, and modes of operation.

In particular, CESSAR-DC, supported by system descriptions, technical specifications, and training materials for System 80, provides the baseline for describing the operating role of the revised systems in System 80+ and extrapolating their operations for revised procedure guidelines and the FTA. A list of these basic purposes of the plant systems and configurations is maintained as part of the System 80+ FTA data base.

18.5.1.3 Task Decomposition and Data Framework

The following hierarchical structure was used as the framework to decompose event sequences into components:

I) Gross functions/subfunction
  A) Task
    1) Element

Each of these levels is detailed as follows.

18.5.1.3.1 Gross Function/Subfunction Level

Gross functions are high level statements of the operator's general purpose in performing a related set of tasks. They specify a basic operating goal (e.g., "Maintain RCS Heat Removal") from the operator's perspective. Each gross function (or subfunction) statement represents one or more tasks with a single main purpose, and may be comprised in different situations by different sets of tasks. Functions appear within sequences in a generic order of performance, per vendor operating procedure guidelines. Subfunctions are identified if a gross function has multiple purposes; otherwise, the two levels of description are similar.

18.5.1.3.2 Task Level

This level analyzes operator behaviors in terms of a generic, closed-loop information processing model. It utilizes a simple but comprehensive data framework that can accommodate a large variety of specific tasks. The model views a task as falling into one of four basic categories:

A. Input (Perception) - Collect or obtain needed information.

B. Process (Cognition) - Evaluate, plan, calculate, decide (etc.) on a result or course of action based on collected or otherwise known information.

C. Output (Action) - Perform the act or manipulation specified.

D. Feedback - Monitor the results of output actions and transmit the results back to the input; this either verifies success or cues further processing and corrective action.

Tasks in a sequence tend to cycle through these categories, although well-designed and skillfully performed tasks do not necessarily show four distinct components. The benefit of this framework is that it directs the analyst's attention to the necessary components of deliberate, rule-based (i.e., procedural) behavior (Reference 6).

A single task is expressed by a task statement. A task statement includes two basic parts, which are 1) a verb from the defined verb taxonomy (listed in the FTA data base), and 2) the object of the verb, (a parameter, component, etc.) For example:

Collect Pressurizer Pressure
(verb) (object)

These task statements then serve as the centerpiece around which the remaining task element data are organized and documented.

18.5.1.3.3 Element Level

The element level of this analysis specifies critical details that may be associated with each task statement. These data complete the FTA picture of task behavioral requirements; i.e., of how the task must be performed. The additional data include:

A) Cues - Cues are conditions, prompts, alerts, or similar items directing that the task statement should be executed. (A typical default cue for a task statement is its position within a procedure sequence.)

B) Criteria - Criteria are qualitative or quantitative values or limits which are necessary references for correct evaluation or execution of the task statement.

C) Time Allowed - The time allowed is the period of time required, as assumed by the analysis, for the execution of the multiple elements comprising the task statement. The initial screening value of time allowed to perform each task statement is one minute (see Section 18.5.1.4).

D) Location - Location is the place or position at which a given task is expected to be performed. Location data provide a basis to perform link analysis.

E) Remarks - Remarks accommodate extra notations or miscellaneous task requirements from data categories with infrequent significance. In the present task analysis, these issues could include, for example, specific workplace suitability issues, task support requirements, communications requirements, crew interaction, or hazard identification.

Elements are the lowest level of the FTA decomposition. An example of a task element data form is shown in Table 18.5.1-1. Additional analyses of the data performed as part of the FTA (i.e., information and controls requirements, and time profile/workload evaluations) are described in Sections 18.5.1.5.2 and 18.5.1.5.3.

18.5.1.4 Loading Criteria

Workload is evaluated on the basis of comparisons between estimates of time available for, and time required by, the elements of a task. Time criteria are as follows:

A. A conservative criterion based on ANSI/ANS 58.8 (Reference 8) provides a minimum of one minute for each required manual manipulation (i.e., task element). This is an initial screening criterion to identify potentially excessive loadings.

B. If task requirements exceed the limits of the screening criterion in 18.5.1.4.A, more detailed evaluation of the human performance requirements is performed, based on a cognitive processing model presented in Reference 7. These evaluations will utilize explicitly stated conservative assumptions, model parameters, and criteria for human and equipment response time performance. Example calculations will be provided.

Failure to meet both criteria A and B above indicates the need for further design assessment and formal resolution. Such findings are entered into the Tracking Open Items (TOI) database, per the requirements of the HFPP.

18.5.1.5 Analyses

18.5.1.5.1 Scope

The following event sequences comprise a representative cross section of operations for the Nuplex 80+ control room FTA, including all Emergency Procedure Guidelines (Reference 5):

A. Startup with Steady State and Transient Power Operations

B. Shutdown with Shutdown Decay Heat Removal

C. Design Basis Shutdown from the Remote Shutdown Area

D. Mid-loop (including Loss of SCS) and Refueling Operations

E. Reactor Trip and Recovery

F. Loss-of-Coolant-Accident

G. Steam Generator Tube Rupture

H. Excess Steam Demand

I. Loss of Feedwater

J. Loss of Offsite Power (LOOP)

K. Station Blackout (LOOP without DGs)

L. Anticipated Transient Without Scram

M. Design Basis Failures of DPS and DIAS

N. Selected Abnormal Operating Procedures

O. Selected Tech Spec Surveillances

18.5.1.5.2 PRA and Critical Tasks

In addition to the representative event sequences in Section 18.5.1.5.1, the System 80+ PRA and associated Human Reliability Analyses (HRA) are used to identify "Critical Tasks." These are operator tasks indicated by PRA to make significant contribution to total plant risk. Critical Tasks are incorporated as separate event sequences in the Task Analysis database. Findings from the associated HRA and TA are dispositioned through the formal documentation and tracking mechanisms of the Human Factors Program Plan (Reference 10).

18.5.1.5.3 Information and Control Requirements

The evaluation of PGICRs summarizes the procedure-based parametric requirements for display and control variables identified by the FTA. Summaries are sorted from the FTA database for each variable. For example, characteristics for 'pressurizer pressure' are summarized for each distinct gross function where pressurizer pressure is utilized. Characteristics include the following areas:

A. Device type

A recommendation for display/control type for each variable is provided. These recommendations are based on the FTA results, operating experience, human performance characteristics, and human factors guidance.

B. Range

The required upper and lower value limits for the variable as required for operations. This range is determined by evaluation of transient performance figures.

C. Accuracy

The instrument accuracy required for each variable based on operator need and transient performance figures is provided.

D. Units

The recommended unit of measure for each variable is provided. These recommendations are based on historical operational records.

18.5.1.5.4 Time Profile/Workload Evaluation

The event sequences identified in Section 18.5.1.1 are analyzed and reviewed by experts in plant operations. Event time profiles are then plotted on time lines and sectioned into discrete evaluation intervals (to minimize unnecessary calculation, fewer activities may be summed within longer intervals.) Process time estimates are derived by evaluating data from specific event profiles, based on operator experience and process transient response models. The time profile evaluation considers:

A) The time into the event sequence at which the operator is expected to be queued to perform the tasks in an interval

B) The time available to perform the tasks in the interval (i.e., plant process constraints)

C) The time required to perform the tasks in the interval (i.e., human performance constraints)

D) Whether time required exceeds time available for specified task intervals

Criteria for time required to perform tasks are specified in Section 18.5.1.4.

18.5.1.5.5 Link Analysis

Link analysis evaluates the distribution and interactions of the operators' panel transitions in a given scenario. Link analysis is performed for design basis normal operations and plant shutdown during a loss of the DPS. This is considered to be a limiting case in terms of its impact on the necessary movement of operators within the controlling workspace. link data shall be precise to the nearest half-panel.

18.5.1.5.6 Identification of Overload Situations and Recommendations

If time required exceeds time available per Criterion A of Section 18.5.1.4, then task loading is a concern. In such cases, the task sequence is reevaluated incorporating more refined timing assumptions per Criterion B. If this detailed evaluation continues to show that more time is required than is available for operator action, the issue is identified in the results, and must receive formal assessment and resolution per the design process and HFPP.

18.5.1.6 Results Documentation

The FTA data are stored on a personal computer database system to allow manipulation and updating of information. As additions are made to the database, existing portions of the analysis will be updated to reflect any changes to the FTA methodology. This will ensure internal consistency of the final FTA results, and of those results with the System 80+ design.

TABLE 18.5.1-1

Sequence:

Gross Function:

Subfunction:

Date:

Page:

Task/ Element #	Cue	Verb	Object	Criteria	Time Required	Location	Remarks

Task Element Data Form

ATTACHMENT 3

RESPONSE TO RAI 440.224 ON THE COMMON MODE FAILURE EVALUATION FOR THE SYSTEM 80+ DESIGN

Question 440.224

To enhance defense in depth, ABB-CE is requested to provide an evaluation of the LBLOCA and MSLB inside the containment events and to identify the time available for the operator to respond to these events.

Response 440.224

For pipes which are 12 in. or larger in diameter, a detectable leak would occur significantly in advance of a major rupture. Thus, the operator would have sufficient time to shut down and depressurize the plant prior to a large break occurrence. However, assuming that a large break LOCA (LBLOCA) did occur prior to this operator action, the following applies.

For the LBLOCA event with a common mode failure of the protection system software, the limiting time related to peak clad temperature is the time the operator manually starts the Safety Injection (SI) pumps. In the CESSAR-DC Chapter 6 analysis, SI pump flow is required prior to when the Safety Injection Tanks (SITs) empty. This is necessary in order to keep the downcomer filled to the reactor vessel inlet nozzle and, consequently, maintain the core reflood rate calculated in the Chapter 6 analysis. If SI pump flow does not start before the SITs empty, then, when the SITs empty, there will be no SI flow to the reactor vessel and the water level in the downcomer will begin to decrease. This will, in turn, lead to a decrease in the core reflood rate and an increase in cladding temperatures.

For the limiting break reported in Chapter 6, the SITs are calculated to empty at 95 seconds. To provide SI pump flow prior to 95 seconds, assuming offsite power is available, the operator would need to manually initiate SI about 15 seconds earlier, or in less than 80 seconds after the break. If the 1.0 DEG/HL break SIT emptying time of 85.7 seconds is used, then the operator would need to initiate SI prior to 70.7 seconds.

Following a LBLOCA, the limiting times related to radiological dose are the times to close the containment purge valve and to initiate containment spray. The attached Table 440.224-1 summarizes the time sequence required to limit the release from containment so that the 10CFR100 dose limits are satisfied. The operator would need to manually initiate the purge valve closure by approximately 145 seconds and manually initiate containment spray at approximately 360 seconds in order to satisfy the dose limits (300 REM thyroid and 25 REM whole body dose).

For the full power MSLB inside containment, the operator action times needed are reactor trip, MSIS, and spray actuation. An evaluation with none of these actions shows the containment pressure reaching 130 psig (Level C stress limit) at 135 seconds. The delays between manual actuation and plant response are short for the reactor trip and MSIS compared to the 28 seconds delay for spray delivery. For this scenario, operator action earlier than 100 seconds is needed for containment considerations. The offsite doses are bounded by the outside containment steam line break analysis presented in LD-93-080, May 19, 1993.

In summary, large break LOCAs and steam line breaks inside containment cannot be mitigated by manual operator actions since the actions would be required too quickly to be reliable. However, the leak detection capability of System 80+ obviates the need to consider these large breaks since the plant would be shut down and depressurized long before the large break can occur.

TABLE 440.224-1

TIME SEQUENCE TO LIMIT RADIOLOGICAL DOSE FOLLOWING A LOCA WITH A COMMON MODE FAILURE OF THE PROTECTION SYSTEM SOFTWARE

Time(Sec)	Event	Value
0	LOCA occurs, RCS coolant* release phase begins, offsite power available, purge valve open, maximum air flowrate through valve, cfm	16,000
30	End of coolant release phase, fuel claddings begin to release gap inventory (5% of total core inventory assumed to be linearly released during the following 30 minutes per NUREG-1465)	-
<70.7	Assumed operator initiates Safety Injection Actuation Signal	
<85.7	HPSI begins delivery of fluid to the core to prevent peak clad temperature from reaching limit	2200°F
145	Assumed operator initiates containment isolation and purge valve closing	
177	Containment purge valve completely closed; percent of total core inventory released to containment	0.4
300	Assumed operator initiates annulus ventilation system operation	
360	Operator initiates containment spray system operation	
390	Containment sprays begin spraying; percent of total core inventory released to containment	1.0
410	Annulus vent system operational	

TABLE 440.224-1

TIME SEQUENCE TO LIMIT RADIOLOGICAL DOSE FOLLOWING A LOCA WITH A COMMON MODE FAILURE OF THE PROTECTION SYSTEM SOFTWARE

Time(Sec)	Event	Value
1830	End of gap release; percent of core inventory released to containment	5.0

* Assumed Technical Specification I-131 concentration (w/o spiking)

ATTACHMENT 4

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
1. Total Loss Of Flow		
Reactivity Control: Alarms: not req'd. DPS Ind: not req'd.	Operator action not required to trip reactor: CEAs will drop into the core automatically upon loss of power to the CEA drive mechanisms resulting from the loss of offsite power associated with this event.	Operator action to trip reactor not credited.
RCS Heat Removal: Alarms: not req'd. DPS Ind: not req'd.	Operator action is not required to achieve RCS heat removal in the near term: Automatic actuation of Emergency Feedwater by the APS results in continued heat removal via the steam driven emergency feedwater pumps and dumping of steam through the main steam safety valves. Longer term heat removal could be accomplished using local manual control of the shutdown cooling system.	Operator response is not credited during the first 30 minutes of this event.

Key:
+ Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
2. RCP Shaft Seizure		
Reactivity Control:	Reactor Trip - 2 min	30 min
Alarms: low RCS flow high RCS T RCP trouble core heat removal + DPS Ind: RCP status RCS flow T-hot * + T-cold * + T-avg + core power rod bottom lights IPSO: no reactor trip		
RCS Heat Removal:	Subsequent to the reactor trip, heat removal can be continued using indications and controls not affected by the CMF.	No operator actions were credited prior to 30 minutes.

3. RCP Shaft Break

Same as RCP Shaft Seizure.

Key:
+ Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
4. CEA Ejection		
Reactivity Control:	Reactor Trip - 2 min	30 min.
Alarms: high core power low przr P low przr L reactivity control + RCS inventory control + RCS pressure control + DPS Ind: core power * + przr P * + przr L * + rod bottom lights IPSO: no reactor trip		
RCS Inventory Control:	SI - 15 min	after 30 min.
Alarms: low przr P low przr L RCS inventory control + RCS pressure control + DPS Ind: przr P * + przr L * +		

Key: + Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
5. Letdown Line Break		
RCS Inventory Control:	Isolate Letdown - 15 min	30 min.
Alarms: (Within a few seconds): letdown line low P nuc. annex high radiation " " high T " " high humidity (Within a few minutes): low przr L nuc. annex high sump L VCT low level RCS inventory control +		
DPS Ind: przr L * + letdown flow		

Key:
+ Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
6. SG Tube Rupture Reactivity Control: Alarms: low przr L low przr P RCS inventory control + RCS pressure control + DPS Ind: core power * + przr P * + przr L * + rod bottom lights IPSO: no reactor trip	Reactor Trip - 15 min	30 min
Radiological Emmissions Control: Alarms: high SG blowdown act. high main steam act. high air ejector act. DPS Ind: przr P * + przr L * + subcool mar. * + SG P * cont. P * SG blowdown act. main steam act. air ejector act.	Isolate SG after 30 min The postulated CMF does not significantly diminish the indications available to support this operator action.	after 30 min

Key: + Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
7. Main Steam Line Break		
Reactivity Control:	Reactor Trip - 2 min	APS trip at 17 min
Alarms: high power low SG P low przr P reactivity control +		
DPS Ind: rod bottom lights core power * + SG P * przr P * + IPSO: no reactor trip		
RCS Pressure Control:	SI - 15 min	Operator action to initiate SI not credited.
Alarms: low przr P RCS pressure control +		
DPS Ind: przr P * +		
RCS Heat Removal:	Close MFW valves - 17 min	Operator acion to close MFW valves not credited.
Alarms: low SG P low RCS T RCS heat removal +	Close MSIVs - 20 min	30 min.
DPS Ind: SG P * T-cold *		

Key: + Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
8. Feedwater Pipe Break		
Reactivity Control: Alarms: not req'd. DPS Ind: not req'd.	Operator action not required to trip reactor: APS trip on high pressurizer pressure.	Operator action to trip reactor not credited.
RCS Heat Removal: Alarms: low SG P high RCS T high cont P RCS heat removal + steam/feed conv. + DPS Ind: SG P * SG L * + T-hot * T-cold * cont. P * cont. T przr P * + przr T * +	Close MFW Valves - 16 min Containment Spray - 17 min MSIVs - 18 min	- 30 min - 30 min - 30 min

Key: + Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.

KEY INDICATORS & CREDITED OPERATOR RESPONSE TIMES FOR CMF EVALUATION

Event & Associated Indications Available	Estimate Of Reasonable Operator Response	Response Time Credited In Evaluation
9. Loss Of Coolant Accident		
Alarms: low przr P low przr L DPS Ind: przr P * + przr L * + core power * + rod bottom lights IPSO: no reactor trip	reactor trip - 3 min	Reactor trip not credited. Moderator reactivity alone credited to shutdown the core.
Alarms: low przr P low przr L DPS Ind: przr P * + przr L * +	safety injection - 15 min	16 min.
Alarms: low przr P DPS Ind: przr P * +	Trip 2 RCPs - 16 min	**For 6 inch pipe break:** Trip all 4 RCPs - 23 min (i.e., credited to occur after the latest time for actuation of all 4 RCPs.
Alarms: low przr P DPS Ind: przr P * +	Trip remaining RCPs - 22 min	**For 3 inch pipe break or .04 ft2 break:** Trip all 4 RCPs - 17 min (i.e., credited to occur no earlier than 1 minute after SI is credited.)

Key: + Indication or alarm provided on IPSO as well as DPS.
* Indication provided on DIAS-P as well as DPS.