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W3F1-98-0027
A4.05
PR

February 18, 1998

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Updated Final Safety Analysis Report - Revision 9 Supplement

Gentlemen:

By letter dated January 29, 1998, (W3F1-98-0021) Waterford 3 submitted Revision 9 to the Waterford 3 Steam Electric Station Unit 3 Updated Final Safety Analysis Report (UFSAR) in accordance with 10CFR50.71(e) and 10CFR50.4(b)(6).

Subsequent to that submittal, Waterford 3 identified a text flow condition affecting the following pages: 9.5-111, 9.5-171, 9.5-181, 9.5-225, 15.4-3 and Table 10.4.9A-1 (pages 1 of 7 through 7 of 7).

Those pages have been corrected. Eleven copies of the corrected pages are enclosed along with revised instructions and a reprinted List of Effective Pages.

To update your copy of the UFSAR, please remove the affected pages and insert the enclosed replacements.

A053

Should you require further information, please contact me at (504) 739-6242 or T.J. Gaudet at (504) 739-6666.

Very truly yours,

T.J. Gaudet
for

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RAB 7B

1. Part height 1-hour rated walls with Class B fire doors constructed to isolate:

- Auxiliary Panel 3 (A/B) from Auxiliary Panel 2(SB)

a) Walls constitute a complete 1-hour barrier separation (including doors and seals) up to the height of the wall which extends well above the height of the safe shutdown equipment.

b) Detection and automatic suppression protection provided throughout the zone for all redundant equipment. An ionization smoke detection system is provided.

c) Overhead interference makes construction of complete 1-hour wall not feasible.

d) Since part height walls adjoin 3-hour boundary walls on both sides of the fire zone, there is no postulated fire capable of radiating two redundant auxiliary relay panels simultaneously, with the exception of the cabinet tops which protrude approximately 1 foot above the top of the walls. The relays are inside these cabinets and are therefore not exposed to the direct radiative affects of a fire. Due to restricted access in this zone, it is not postulated that a significant accumulation of combustibles would occur to support a fire of sufficient magnitude. At least one safe shutdown cable/conduit train is provided with a 1-hour wrap where essential redundant trains occur in each subzone of RAB 7.

2. Enclosure of essential redundant cable tray and conduit in the same 1-hour wrap for the following systems controlled from the auxiliary control panel (LCP-43):

- Shutdown Cooling System
- Chemical and Volume Control System

a) Regulatory Guide 1.75 separation criteria provides reasonable assurance that an internal tray or conduit fire will not propagate to a redundant tray or conduit.

1) This separation between one cable tray and a redundant cable tray or conduit within the same fire wrap is accomplished by providing the subject cable tray with either a metal tray cover or a 1/2 inch blanket cover (i.e., 30 minute fire rating). Where access and spatial separation between cable trays allow, a 1-1/2 inch blanket tray cover (i.e., 1-hour fire rating) is preferred over a 1/2 inch blanket.

b) One-hour wrap provides sufficient protection to redundant cabling until actuation of automatic suppression system or arrival of fire brigade.

RAB 7B

3. Lack of documented hourly fire rating based on ASTM E-119 for gap sealed walls.

   a) The gap sealed walls in this zone have sprinklers and detectors on both sides.

RAB 7C

I. Description of Fire Zone:

a. Building: REACTOR AUXILIARY Elev: +35.00 Ft. MSL
b. Space Name: RELAY ROOM (Isolation Panel)
c. Figure No.: 9.5.1-21 Approximate Coordinates: Cols, 11A, H-J
d. Floor Area: 150 sq ft
e. Subspaces Within the Fire Zone: None

II. Essential Equipment Within the Fire Zone:

a. Isolation Panel
b. Cables - SA, SAB, SB

III. In situ Combustible Material Loadings:

In situ combustible materials consist of, but are not limited to, cable insulation. The calculated fire duration does not exceed the fire resistance rating of the boundary barriers.

IV. Sources of Radioactive Materials:

None

V. Fire Control:

a. Physical Containment:

1. Fire Zone Boundary Barriers:

Fire zone boundary design ratings are 1 hour between zones and 3 hours for the envelope perimeter.

Essential "A," "AB," and "B" train cable trays, conduits, cable air drops, and junction boxes required for safe shutdown are protected with one-hour fire rated barriers. Each compartment within the Isolation Panel is separated by a sheet metal wall sandwiched by 1/16 inch thick inorganic fiber insulation boards forming a fire retardant barrier.

2. HVAC Penetrations Through Boundary Barriers:

Duct Penetration Location	Function	Duct Size (Inches)	Fire Damper	Safety-Related	Non-Safety-Related
East Wall	R	24x12	Yes (FD-51)	X	

RAB 23

b) Protection of floor side (RAB 23) of hatch is accomplished by area wide smoke detection and automatic fixed suppression; additional modifications would be physically cumbersome to traffic flow during maintenance outages.

c) Low probability of a flammable liquid spill in vicinity of the hatch due to strict administrative controls.

d) There are no credible sources of ignition in the hatch vicinity.

e) The design of the hatch is such that only limited seepage of a liquid past the hatch-to-floor fitting can occur, thus acting as a flame arrester.

f) Smoke detection and automatic fixed suppression coverage below the hatch provide adequate compensation for any fire hazard associated with seepage past the hatch fitting.

6. Exception from 20' separation for redundant diesel fuel oil storage and day tank piping:

a) Subject piping is located in corridor south of diesel generator rooms (RAB 15 and 16) with a minimum separation of approximately 7 feet.

b) Negligible combustible loading in corridor.

c) Smoke detection and automatic fixed suppression in this corridor.

d) Piping meets seismic Category I and Safety Class 3 design and construction criteria.

e) Construction criteria and heat dissipating capability of Schedule 80 pipe filled with liquid provide adequate protection from radiative and convective effects of a postulated fire until actuation of smoke detection and automatic fixed suppression or arrival of the fire brigade.

7. Lack of a documented hourly fire rating based on ASTM E-119 for Stairwell Enclosure 7 at Column 3A, Stairwell Enclosure 6 at Column 8A and Stairwell Enclosure 10 at Column 8A.

a) The interiors of the stairwell enclosures contain negligible combustible loading.

b) For Stairwell Enclosures 6 and 10, the gap sealed walls have sprinklers and detectors on one side of the enclosure walls.

c) For Stairwell Enclosure 7, the gap sealed walls have sprinklers and smoke detectors on one side of the north and east enclosure walls. The south enclosure wall has no sprinklers on either side; however, the adjacent corridor and Boric Acid Concentrator Rooms have a very low combustible loading and contain no essential safe shutdown equipment or cables.

RAB 23A

I. Description of Fire Area:

a. Building: REACTOR AUXILIARY Elev: -15.00, -4.00, +21.00 and +35.00 Ft. MSL

b. Space Name: RADIOACTIVE PIPE CHASE

c. Figure No.: 9.5.1-3, 9.5.1-4, 9.5.1-5, 9.5.1-13, 9.5.1-14, and 9.5.1-21 Approximate Coordinates: Cols, 2A-6A, L

d. Floor Area: 380 sq ft

e. Zones Within the Fire Area: None

II. Essential Equipment Within the Fire Area:

a. Cables - SA, SB, NA

III. In situ Combustible Material Loadings:

In situ combustible materials consist of, but are not limited to, cable insulation. The calculated fire duration does not exceed the fire resistance rating of the boundary barriers.

IV. Sources of Radioactive Materials:

Radioactive materials may be present in waste and boron management, fuel pool, blowdown, safety injection and CVCS System piping.

V. Fire Control:

a. Physical Containment:

1. Fire Area Boundary Barriers:

Fire area boundary design rating is 3 hours.

2. HVAC Penetrations Through Boundary Barriers:

Duct Penetration Location	Function	Duct Size (Inches)	Fire Damper	Safety-Related	Non-Safety-Related
East Wall	S	10x10	Yes		X
North Wall	E	10D	Yes (FD-48A)	X	
East Wall	E	6D	Yes (FD-47A)	X	

All piping and tray penetrations are sealed.

3. Ingress and Egress:

Normal ingress and egress is not provided for this fire area. Access is only provided via removable block walls.

RAB 27

I. Description of Fire Area

a. Building: REACTOR AUXILIARY Elev: +7.00 Ft MSL
b. Space Name: MECHANICAL-ELECTRICAL, HVAC EQUIPMENT & HEALTH PHYSICS ENVELOPE
c. Figure No.: 9.5.1-10 Approximate Coordinates: Cols, 8A-12A, G-L
d. Floor Area: 7,503 sq ft
e. Subspaces Within the Fire Area: Communications Equipment Room (371 sq ft)
I&C Room (442 sq ft)
HVAC Equipment Room (1,683 sq ft)

II. Essential Equipment Within the Fire Zone:

a. Air Handling Units AH-30 (3A-SA), AH-30 (3B-SB)
b. CWS Control Valves - 3AC-TM137A, 3AC-TM161B
c. Intake Dampers - D-50(SA), D-50(SB)
d. Cables - SA, SB

III. In situ Combustible Material Loadings

In situ combustible materials consist of, but are not limited to, cable insulation and ordinary combustible material. The calculated fire duration does not exceed the fire resistance rating of the boundary barriers.

IV. Sources of Radioactive Materials:

None

V. Fire Control:

a. Physical Containment:

1. Fire Area Boundary Barriers:

A 3-hour fire boundary is provided for the envelope perimeter.

All vertical cable trays are provided with firebreaks at approximately 15 foot intervals. Essential "B" train cable trays, conduits, cable air drops, and junction boxes required for safe shutdown are protected with one-hour fire rated barriers.

2. HVAC Penetrations Through Boundary Barriers:

Duct Penetration Location	Function	Duct Size (Inches)	Fire Damper	Safety-Related	Non-Safety-Related
East Wall	S	22x24	Yes		X
East Wall	OAI	10x52	Yes		X
East Wall	OAI	48x20	Yes		X
East Wall	E	16x14	Yes		X
Ceiling	R	46x18	Yes (FD-16)	X	
Ceiling	R	46x18	Yes (FD-17)	X	
West Wall	S	24x24	Yes		X
Ceiling	S	16D	Yes (FD-29)	X	
Ceiling	S	16D	Yes (FD-25)	X	
Ceiling	S	16D	Yes (FD-26)	X	
Ceiling	S	16D	Yes (FD-27)	X	
Ceiling	S	16D	Yes (FD-28)	X	
Floor	S	8x6	Yes		X

All piping and tray penetrations are sealed.

3. Ingress and Egress

Ingress and egress for this fire area are provided through two "B" label fire doors (DI01 and D107) to enclosed stairs at columns K-12A and J-11A. These stairs lead to fire zone RAC 8C at El. 21 ft. and fire area RAB 30 at El.-4 ft. In addition, an access door, AD101, located on the east wall leads to fire area RAB 3A.

b. Detection:

Ionization smoke detectors are provided for this fire area, including pipe chase and cable riser shaft.

c. Fire Protection:

Primary:

A pre-action automatic sprinkler system is provided for this entire fire area, including the area above the drop ceiling. No sprinkler protection is provided for the communications equipment subspace. Portable fire extinguishers are provided in accordance with the guidelines of NFPA 10. Specific types and locations of portable fire extinguishers are indicated on General Arrangement drawings.

Secondary:

Class 1 hose stations installed in accordance with the guidelines of NFPA 14 are available for use in this zone. Specific locations of hose stations are indicated on General Arrangement drawings.

III. In situ Combustible Material Loadings:

In situ combustible materials consist of, but are not limited to, cable insulation, charcoal and lubricating oil. The calculated fire duration does not exceed the fire resistance rating of the boundary barriers.

IV. Sources of Radioactive Materials:

Sources of radioactive materials are present within the various components located in the Reactor Containunent Building during normal operation and station shutdown periods.

V. Fire Control:

a. Physical Containment:

1. Fire Area Boundary Barriers:

Fire area boundary design rating is 3 hours.

Vertical cable trays are provided with firebreaks at approximately 15 foot intervals. The conduits for one division of Shutdown Cooling System Isolation Valves are wrapped wherever there is less than a 20 foot separation between redundant cables. The Shutdown Cooling System Isolation Valves' electrical penetrations are protected by radiant energy shields.

2. HVAC Penetrations Through Boundary Barriers:
All penetrations are protected with containment isolation valves. All piping and tray penetrations are sealed.

3. Ingress and Egress:

Ingress and egress for the Reactor Containment Building are provided at the locations described below:

a) A personnel lock (center line located at EL. +11.00 ft msl and 18 degrees east of Column 7A) which discharges into fire area RAB 32 at EL. -4.00 ft msl.

b) A 14 ft diameter equipment and maintenance hatch (center line EL. located at +26.5 ft msl and 7-1/2 degrees north of Column 0) which discharges into a walkway providing access to the Cooling Tower Area and Fuel Handling Building.

c) An escape lock (center line located at EL. +25.00 ft msl and 37-1/2 degrees west of Column 7A).

d) Access to the containment Shield Building (annulus) is provided through an access lock (center line EL. at +25-00 ft msl and 52-1/2 degrees west of Column 7A). Both of the locks mentioned in items 3 and 4 discharge into a walkway providing ingress to the cooling towers and Fuel Handling Building at EL. +21.00 ft msl.

b. Detection:

Ionization smoke detectors are provided for the electrical penetration area, the cable assembly area (EL. +21.00 ft msl), and for the cable trays on either side of the reactor at approximate floor Elevation +46 ft msl. This also provides partial area detection for the reactor and the steam generators. Continuous thermistor wire detectors are provided for the reactor coolant pump areas. Charcoal filter enclosures are provided with thermistor wire heat detectors. Photoelectric smoke detectors are provided for the annulus and are located on the duct of the annulus negative pressure system.

c. Fire Protection:

Primary:

A multicycle automatic sprinkler system is provided for the protection of the reactor coolant pump lube oil systems. A manually activated water spray system is provided over charcoal beds in filter units E-13 (3A-SA) and E-13 (3B-SB).

Seconday:

Class 1 hose stations installed in accordance with the guidelines of NFPA 14 are available for use in this area. Portable fire extinguishers are provided in accessible locations in accordance with the guidelines of NFPA 10. Specific types and locations of portable fire extinguishers and hose stations are indicated on General Arrangement drawings.

d. Smoke Venting:

The normal ventilation system may be used for smoke removal. Portable smoke ejector equipment is provided for use by the fire brigade.

e. Drainage:

Adequate drainage of water used for fire extinguishment will be provided by the containment sump pumps. The pumps discharge into the waste tanks.

VI. Analysis of Effects of Potential Fires:

In the Reactor Containment Building, in situ combustible materials include moderate amounts of charcoal in filter enclosures, lubricating oil in RCP pumps, and cable insulation. Transient materials are administratively controlled to limit the amount of such materials to that required for operations and maintenance purposes.

TABLE 10.4.9A-1 (Sheet 1 of 7)

EVALUATION OF THE WATERFORD SES UNIT NO.3 EMERGENCY FEEDWATER SYSTEM VERSUS THE REQUIREMENTS OF STANDARD REVIEW PLAN (SRP) 10.4.9 AND BRANCH TECHNICAL POSITION (BTF) ASB 10-1

A. SRP 10-4.9

ACCEPTANCE CRITERIA	COMPLIANCE	FSAR REFERENCE
Acceptability of the design of the Auxiliary Feedwater System, as described in the applicant's Safety Analysis Report (SAR), is based on specific general design criteria and regulatory guides. Listed below are the specific criteria as they relate to the AFS.		
1. General Design Criterion 2, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.	The EFS is designated seismic Category I. EFS pumps their controls, and the Condensate Storage Pool are located inside the RAB and are thus protected from site related phenomena such as tornadoes, hurricanes and floods. The EFS isolation valves and the Turbine Steam Supply Valves are located in the open area on top of the RAB and are protected by the RAB walls from flooding and direct tornado and hurricane winds. Additional makeup water for the EFS may be obtained from the Wet Cooling Tower (WCT) basins, which are designated seismic Category I and designed to withstand the effects of tornadoes, hurricanes and floods.	10.4.9.3.2, 9.2.5.3.3, Table 3.2-1 NRC Q.371-11
2. General Design Criterion 4, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet	The EFS pumps, their controls, and the Condensate Storage Pool are located inside the RAB and are are thus protected from external missiles; the EFS isolation and impingement forces associated with pipe breaks. Turbine Steam Supply Valves located in the open area on top of the RAB and are protected from tornado missiles by grating installed above the valves.	10.4.9.3.2, 3.5, Table 3.5-3
	The EFS is designed to withstand the effects of internally generated missiles. This subject is discussed in Section 3.5 and summarized for the EFS in Table 3.5-3.	10.4.9.3 3.5

TABLE 10.4.9A-1 (Sheet 2 of 7)

ACCEPTANCE CRITERIA	COMPLIANCE	REFERENCE
	The EFS is protected from the effects of pipe rupture. Each motor driven pump is located in its own room; the turbine driven pump is located away from high from high energy systems. The EFS isolation valves and the Turbine Steam Supply Valves are located adjacent to the break exclusion area portion of the main steam and feedwater piping. Otherwise, design features, such as separation, pipe whip restraints and barriers ensure that pipe break, will not impair the functional capability of the EFS.	3.6, 10.4.9
3. General Design Criterion 5, as related to the capability of shared systems and components important to safety to perform required safety functions.	Not Applicable	
4. General Design Criterion 19, as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.	The EFS is automatically initiated by an Emergency Feedwater Actuation Signal (EFAS) as described in Section 7.3. The EFS can also be started manually from the Main Control Room and the auxiliary control panel. Safety related display information located in the main control room provides the operator with sufficient information to perform the required safety functions. See Section 7.5.	10.4.9, 7.3.1.1.6, 7.5
5. General Design Criterion 44, to assure:		
a. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.	The EFS supplies sufficient cooling water to the steam generators following loss of normal feedwater or main steam or main feedwater line break to provide cooldown of the Reactor Coolant System to the temperature and pressure at which the Shutdown Cooling System can be placed in operation	10.4.9

TABLE 10.4.9A-1 (Sheet 3 of 7) Revision 9 (12/97)

ACCEPTANCE CRITERIA	COMPLIANCE	FSAR REFERENCE
b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)	The EFS is designed to perform its safety functions assuming a single active component failure coincident with a loss of offsite power. The turbine driven EFS pump or both motor driven pumps together have been designed to provide 100% of the flow necessary for residual heat removal over the entire range of reactor operation including all postulated design basis accidents. However, it has been determined that under realistic conditions, any one EFS pump can supply adequate flow for decay heat removal to one (400 gpm required) or both (450 gpm total required) steam generators.	10.4.9
c. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.	The EFS isolation valves are powered from Redundant Class 1E dc buses to ensure that the emergency feedwater flow to the affected steam generator would be shut off, and at least one path would remain open to admit emergency feedwater to the intact steam generator when required, assuming a single active failure.	10.4.9
6. General Design Criterion 45, as related to design provisions made to permit periodic inservice inspection of system components and equipment.	Provisions have been made in the design and layout of the EFS to allow for compliance with the inservice inspection requirements of ASME Code Section XI. The inservice inspection program will be submitted to the NRC at least six months prior to first refueling.	6.6

TABLE 10.4.9A-1 (Sheet 4 of 7)

	ACCEPTANCE CRITERIA	COMPLIANCE	FSAR REFERENCE
7.	General Design Criterion 46, as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and, leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown and accident conditions.	The EFS shall undergo preopoperational tests to ensure its ability to function as intended. A description of this test is contained in Section 14.2.12.2.68.	14.2.12.2.68
		The EFS shall also undergo periodic functional testing of the system and its components to ensure the system functions as intended. These tests shall be conducted in accordance with the surveillance requirements of the plant technical specifications.	Technical Specification 4.7.1.2.
8.	Regulatory Guide 1.26, as related to the quality group classification of system components.	The design and fabrication of the EFS meet the requirements of Regulatory Guide 1.26. The safety class 2 and 3 portions of the EFS meet the Regulatory Guide 1.26 Quality Group B and C standards respectively.	3.2.2 Table 3.2-1
9.	Regulatory Guide 1.29, as related to the seismic design classification of system components.	The EFS meets the requirements of Regulatory Guide 1.29. It is designated as Seismic Category I and will thus perform its design functions following the SSE. The pertinent quality assurance requirements of Appendix B to 10CFR50 are applied.	10.4.9, 3.2.1 Table 3.2-1
10.	Regulatory Guide 1.62, as related to design provisions made for manual initiation of each protective action.	The EFS meets the requirements of Regulatory Guide 1.62. The operator may manually initiate the Emergency Feedwater Actuation Signal (EFAS) from an easily accessible location in the control room (RTG Board). Manual initiation will ensure that protective action goes to completion.	7.3.1.1.6, 7.3.2.1.2
11.	Regulatory Guide 1.102, as related to the protection of structures, systems, and components important to safety from the effects of flooding.	The Waterford 3 Nuclear Plant Island Structure (NPIS) is a reinforced box structure with solid exterior walls and is flood protected up to elevation +30 ft. MSL. This is sufficient to accommodate at the Design Basis Flood identified in Regulatory Guide 1.59. The technical specification requirements of Regulatory Guide 1.02 position C.2 are therefore optional.	2.4.10 Technical Specification 3/4.7.5

TABLE 10.4.9A-1 (Sheet 5 of 7)

ACCEPTANCE CRITERIA	COMPLIANCE	FSAR REFERENCE
	However, due to the nature of potential flooding, Technical Specification 3/4.7.5 which meets the intent of that Regulatory Guide position has been established. In regard to position C.3 of the guide it has been determined that all roofs housing safety related structures can safety store the maximum possible ponding reresulting from the PMP. See also item I above.	
12. Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles	The EFS is protected from the effects of tornado as described in item 2 above.	
13. Branch Technical Position ASB 3-1 and MEB 3-1, as related to breaks in high and moderate energy piping systems outside containment.	The EFS is protected from the effects of pipe rupture as described in item 2 above.	
	The EFS itself is used only for emergency shutdown of the reactor when the Main Feedwater System is inoperative. Therefore in accordance with the APCSB 3-1, it is not analyzed as a high energy system. However, the EFS has been analyzed as a moderate energy system in accordance with the Branch Technical Position MEB 3-1.	3.6 3.6A
14. Branch Technical Position ASB 10-1, as related to auxiliary feedwater pump drive and power supply diversity.	See part B	

TABLE 10.4.9A-1 (Sheet 6 of 7)

B. BTP ASB 10-1 (DESIGN GUIDELINES FOR AUXILIARY FEEDWATER SYSTEM PUMP DRIVE AND POWER SUPPLY DIVERSITY FOR PRESSURIZED WATER REACTOR PLANTS)

	BRANCH TECHNICAL POSITION	COMPLIANCE	FSAR REFERENCE
1.	The auxiliary feedwater system should consist of at least two full-capacity, independent systems that include diverse power sources.	The EFS consists of two motor driven pumps, each with a design flow capacity of 395 gpm (45 gpm recirculation included and powered from separate, redundant Class 1E 4.16 KV buses, and one 780 gpm (80 gpm recirculation included) turbine driven pump. The turbine driven pump or both motor driven pumps together have been designed to provide 100% of the flow necessary for residual heat removal over the entire range of postulated design basis accidents. However it has been determined that under realistic conditions, any one EFS pump can supply adequate flow for decay heat removal to one (400 gpm required) or both (450 gpm total required) steam generators.	10.4.9
2.	Other powered components of the auxiliary feedwater system should also use the concept of separate and multiple sources of motive energy. An example of the required diversity would be two separate auxiliary feedwater trains, each capable of removing the afterheat load of the reactor system, having one train powered from either of two ac sources and the other train wholly powered by steam and dc electric power.	The turbine driven pump, system safety controls, and turbine steam supply valves are powered from the SA/B Class 1E 125V dc bus. The EFS isolation valves are also fail-open and powered from Class 1E 125 dc buses. This ensures that flow to the depressurized steam generator can be terminated and at least one path to the intact steam generator will be available assuming any single active failure.	
3.	The piping arrangement, both intake and discharge, for each train should be designed to permit the pumps to supply feedwater to any combination of steam generators. This arrangement should take into account pipe failure, active component failure, power supply failure, or control system failure that could prevent system function. One arrangement that would be acceptable is crossover piping containing valves that can be operated by remote manual control from the control room, using the power diversity principle for the valve operators and actuation systems.	The piping arrangement is designed to permit the pumps to supply feedwater to any combination of steam generators. Waterford 3 uses the crossover piping scheme. This is described in FSAR Section 10.4.9 and shown in the schematic of Figure 10.4-8 The EFS exceeds this BTP criteria in that the pipelines to each steam generator are isolated by four pneumatically operated fail open isolation valves. The power diversity designed into the system is summarized above. The design of the EFS thus ensures that in the event of a single failure it can supply water to one or both steam generators, and that water could be prevented from entering the ruptured line during the postulated main steam feedwater line break.	10.4.9

TABLE 10.4.9A-1 (Sheet 7 of 7) Revision 9 (12/97)

BRANCH TECHNICAL POSITION	COMPLIANCE	FSAR REFERENCE
4. The auxiliary feedwater system should be designed with suitable redundancy to offset the consequences of any single active component failure; however, each train need not contain redundant active components.		
5. When considering a high energy line break, the system should be so arranged as to assure the capability to supply necessary emergency feedwater to the steam generators, despite the postulated rupture of any high energy section of the system, assuming a concurrent single active failure.	The Waterford 3 EFS is not used during normal reactor startup, hot standby or shutdown. It is only used for emergency shutdown when main feedwater is inoperative. In accordance with the definition and criteria of APCSB 3-1, the EFS is not a high energy system. However, the EFS has been analysed as a moderate energy system in accordance with Branch Technical Position MEB 3-1.	3.6, 3.6A

15.4.1.1.4 Barrier Perfomance

a) Mathematical Model

The mathematical model used for evaluation of barrier performance is identical to that described in Subsection 15.4.1.1.3.

b) Input Parameters, Initial Conditions and Results

In Subsection 15.4.1.2.4, it was determined that the most adverse CEA withdrawal event, in terms of degradation in barrier performance, is one initiated from low power conditions such that the reactivity addition rate combined with the natural plant feedback mechanisms result in a new steady state, not tripped condition.

The fuel performance conditions of the CEA withdrawal from subcritical conditions is combined with barrier performance and its associated steam releases to determine the radiological consequences.

15.4.1.1.5 Radiological Consequences

The radiological consequences due to steam releases from the secondary system are less severe than those from the inadvertent opening of the atmospheric dump valve, Subsection 15.1.1.4.

15.4.1.2 Uncontrolled CEA Withdrawal from Low Power Conditions

15.4.1.2.1 Identification of Causes and Frequency Classification

The estimated frequency of a control element assembly (CEA) withdrawal from low power conditions classifies it as a moderate frequency incident as defined in Reference 1 of Section 15.0. An uncontrolled withdrawal of CEAs is assumed to occur as a result of a single failure in the control element drive mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), Reactor Regulating System, or operator error. This analysis was done for Cycle 2.

15.4.1.2.2 Sequence of Events and Systems Operation

The withdrawal of CEAs from low power conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase with corresponding increases in reactor coolant temperatures and Reactor Coolant System (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters result in the approach to specified fuel design limits and to RCS and secondary system pressure limits, thereby requiring the protective action of the Reactor Protection System (RPS).

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at lower power conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. Depending on the initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a variable overpower trip, high pressurizer pressure trip, a low departure from nucleate boiling ratio (DNBR) trip or a high local power density trip. The secondary system pressure increases following reactor trip and is limited by the steam generator safety valves.

Table 15.4-3 gives the sequence of events for the limiting CEA withdrawal transient at low power (10^{-4} percent power) discussed in Subsection 15.4.1.2.3.

15.4.1.2.3 Core and System Performance

a) Mathematical Model

The nuclear steam supply system (NSSS) response to a CEA withdrawal from low power conditions was simulated the CESEC computer program described in Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program with the CE-1 CHF correlation described in Chapter 4.

b) Input Parameters and Initial Conditions

The input parameters and initial conditions used to analyze the NSSS response to a CEA withdrawal from low power conditions are discussed in Section 15.0. In particular, those parameters which were unique to the analysis of fuel performance discussed below are listed in Table 15.4-4.

The initial conditions and NSSS characteristics assumed in this radiological release analysis have been identified as a limiting set of conditions allowed by the limiting conditions for operation (LCOS) in terms of providing the nearest and most rapid approach to the fuel design limits. These initial conditions are as follows: