TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

August 1, 1985

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Enclosed are changes to the Watts Bar Nuclear Plant (WBN), units 1 and 2 Final Safety Analysis Report (FSAR) which are necessary as a result of TVA's verification that the as-built plant is in conformance with the description in the FSAR as amended. These changes will be included in the next amendment (56) to the WBN FSAR.

If there any questions, please get in touch with K. P. Parr at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R.H. Shell

R. H. Shell
Nuclear Engineer

Sworn to and subscribed before me
this 1st day of August 1985
Paulette H. White
Notary Public
My Commission Expires 8-24-88

Enclosure
cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attention: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

B001
1/1

TABLE 1.3-3 (Continued)

System	Reference Section	Changes
Acrolein System		The Acrolein System has been deleted in favor of some as yet undetermined clamascide
Auxiliary control air	9.3.1	Credit is now taken for auxiliary air system as a safety feature.
Instrumentation and control	9.3.5	A gross failed fuel detector has been added
Heating, ventilating and air conditioning	9.4.1	Ventilation, heating and air-conditioning provided for the reactor auxiliary board rooms
	9.4.2	Shutdown Board Room air-cond. system - outside air is taken from intake on roof of Auxiliary Building and filtered thru HEPA filters only. In the event of an accident or high radiation signal, operator will close isolation dampers from main control room.
	9.4.3	Auxiliary Building Ventilation System is assisted by operation of the General Cooling System by providing chilled water to the building air intake coils and various stratezically located air handling equipment.
	9.4.7	An annulus vacuum control subsystem was included in the emergency gas treatment system to continuously maintain the shield building annulus space at a negative pressure during plant operation.
Compressed air system	9.3.1	Several portable breathing air stations have been provided.

Delete

(a) ~~Non-commercial traffic - Tennessee Wildlife Resources Agency (TWRA),~~ 52

(b) ~~Commercial traffic - U.S. Coast Guard (USCG).~~

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particular emphasis on the stator end turns. It is then ducted to the external air/water heat exchangers. Each motor has two such coolers, mounted diametrically opposite to each other. In passing through the coolers the air is cooled to below ~~122~~ 123°F so that minimum heat is rejected to the containment from the motors.

5.5.1.4 Tests and Inspections

The reactor coolant pumps can be inspected in accordance with ASME Section XI, Code for Inservice Inspection of Nuclear Reactor Coolant Systems. Any full penetration welds in the pressure boundary are prepared with a smooth surface transition between weld metal and parent metal for radiographic inspection.

Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for usual access to the internal surface of the pump casing.

The reactor coolant pump quality assurance program is given in Table 5.5-2.

5.5.2 Steam Generators

5.5.2.1 Design Bases

Steam generator design data are given in Table 5.5-3. The design sustains transient conditions given in Section 5.2.1. Although the required secondary side ASME classification is Class II, the current design philosophy is to use Class I requirements for all pressure retaining parts of the steam generator. Assurance of adequate fracture toughness of all pressure boundary materials is, therefore, as described in Section 5.2.4 and complies with Article NB-2300 of Section III of the ASME Code. Estimates of radioactivity levels anticipated in the secondary side of the steam generators during normal operation, and the bases for the estimates are given in Chapter 11. Rupture of a steam generator tube is discussed in Chapter 15.

The internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.25 percent by weight under the following conditions:

1. Steady state operation up to 100 percent of full load steam flow, with water at the normal operating level.

new figure 7/85 - to replace
existing Fig 6.2.3-17

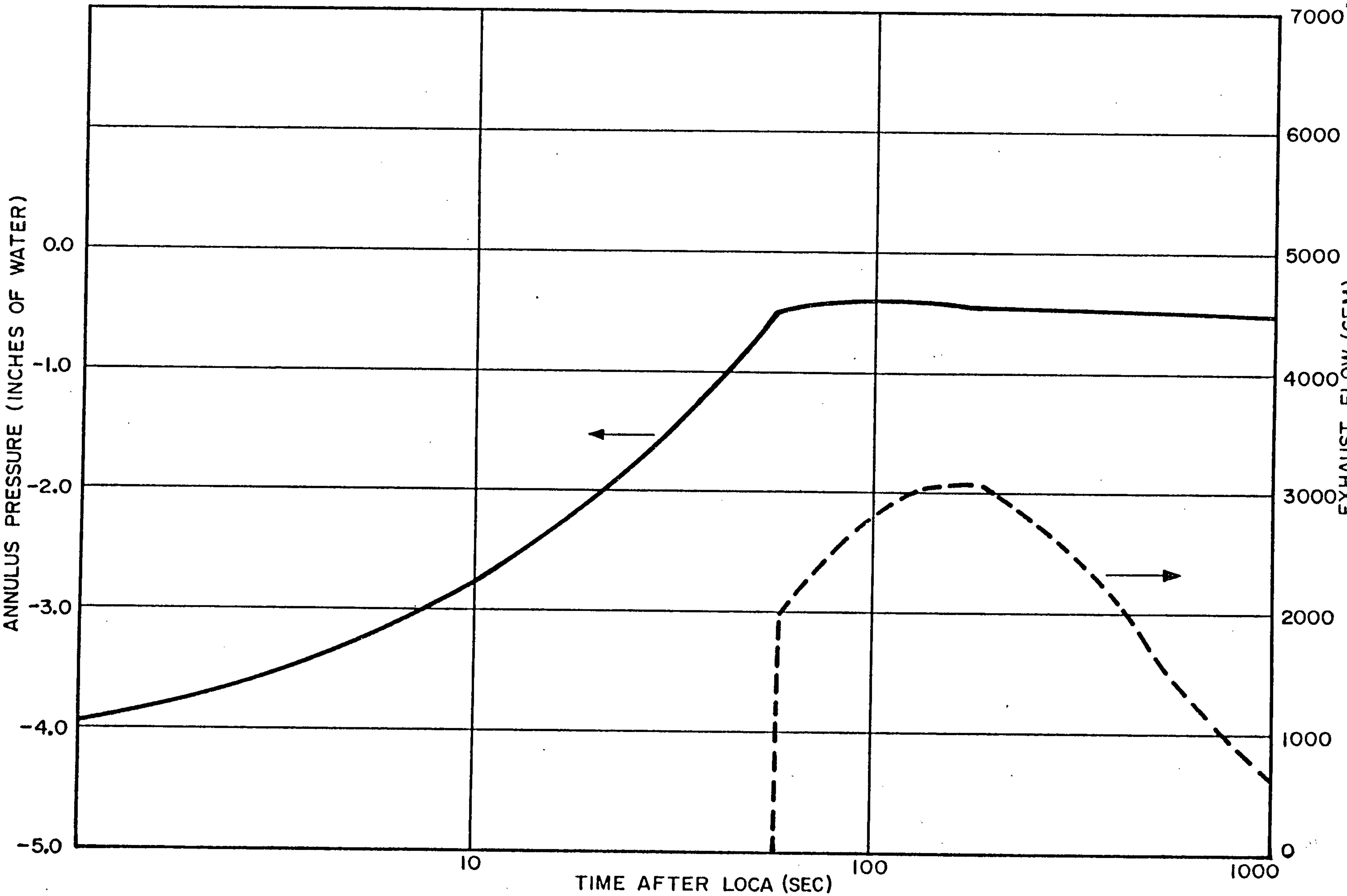

Figure 6.2.3-17 POST ACCIDENT ANNULUS PRESSURE AND REACTOR UNIT VENT FLOW RATE TRANSIENTS

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TABLE 6.3-10

ECCS ACTIVE VALVES

TVA Valve No #	W Valve No #	Normal Position	Cold Leg Injection (After "S" Signal)	Recirculation Cold Leg	Recirculation Hot Leg	Normal Shutdown	RHR Containment Spray
FCV-68-307	8025	Closed	Closed	Closed	Closed	Closed	Closed
FCV-81-12	8028	Closed	Closed	Closed	Closed	Closed	Closed
FCV-62-72	8149A	Closed(1)	Closed	Closed	Closed	Open	Closed
FCV-62-73	8149B	Open(1)	Closed	Closed	Closed	Open	Closed
FCV-62-74	8149C	Closed(1)	Closed	Closed	Closed	Open	Closed
FCV-74-16	HCV-606	Open	Open	Open	Open	Open	Open
FCV-74-28	HCV-607	Open	Open	Open	Open	Open	Open
FCV-63-118, 98 80, 67	MOV 8808A B, C, D	Open	Open	Closed	Closed	Closed	Open
FCV-63-71	8871	Closed	Closed	Closed	Closed	Closed	Closed
FCV-63-64	8880	Closed	Closed	Closed	Closed	Closed	Closed
FCV-63-84	8964	Closed	Closed	Closed	Closed	Closed	Closed
FCV-63-47, 48	MOV 8923A, B	Open	Open	Open	Open	Open	Open
FCV-63-23	8888	Closed	Closed	Closed	Closed	Closed	Closed
FCV-74-3, 21	MOV 8700A, B	Open	Open	Closed	Closed	Open	Closed
FCV-74-33	MOV 8716A	Open	Open	Closed	Open	Open	Closed
FCV-74-35	MOV 8716B	Open	Open	Closed	Open	Open	Closed
FCV-63-8, 11	MOV 8804A, B	Closed	Closed	Open	Open	Closed	Open
FCV-63-5	MOV 8806	Open	Open	Closed	Closed	Open	Closed
FCV-63-72, 73	MOV 8811A, B	Closed	Closed	Open	Open	Closed	Open
FCV-63-1	MOV 8812	Open	Open	Cloed	Closed	Closed	Closed
FCV-63-3	MOV 8813	Open	Open	Closed	Closed	Open	Closed
FCV-63-4	MOV 8814	Open	Open	Closed	Closed	Open	Closed
FCV-63-175	MOV 8920	Open	Open	Closed	Closed	Open	Closed
FCV-63-7, 6	MOV 8807A, B	Closed	Closed	Open	Open	Closed	Open
FCV-62-135, 136	MOV LCV-112D, E	Closed	Open	Closed	Closed	Closed	Closed
FCV-68-308	8026	Open	Closed	Closed	Closed	Open	Closed
FCV-68-305	8033	Open	Closed	Closed	Closed	Open	Closed
FCV-62-132, 133	MOV LCV-112B, C	Open	Closed	Closed	Closed	Open	Closed
FCV-62-63	MOV 8100	Open	Closed	Closed	Closed	Open	Closed
FCV-62-61	MOV 8112	Open	Closed	Closed	Closed	Open	Closed
FCV-62-90	MOV 8105	Open	Closed	Closed	Closed	Open	Closed
FCV-62-91	MOV 8106	Open	Closed	Closed	Closed	Open	Closed
FCV-62-98	MOV 8110	Open	Open	Open	Open	Open	Open
FCV-62-99	MOV 8111	Open	Open	Open	Open	Open	Open
FCV-62-77	8152	Open	Closed	Closed	Closed	Open	Closed
FCV-63-26, 25	MOV 8801A, B	Closed	Open	Open	Open	Closed	Open
* FCV-63-39, 40	MOV 8803A, B	Closed	Open	Open	Open	Closed	Open
** FCV-63-42, 41	8870A, B	Open	Closed	Closed	Closed	Open	Closed
** FCV-63-38	8883	Open	Closed	Closed	Closed	Open	Closed

(1) May be open or closed during normal operation.

Revised by Amendment 55

*Unit 1 valves are open during all modes of operation (due to Boron Injection Tank (BIT) function removal.

** Unit 1 valves are closed during all modes of operation (due to BIT function removal).

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T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.2°F).
S	=	As defined for overtemperature ΔT trip.
$f_2(\Delta I)$	=	0 for all ΔI.

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The source of temperature and flux information is identical to that of the overtemperature ΔT trip and the resultant ΔT setpoint is compared to the same ΔT. Figure 7.2-1, Sheet 3, shows the logic for this trip function. The detailed functional description of the process equipment associated with this function is contained in reference [1].

3. Reactor Coolant System Pressurizer Pressure and Water Level Trips

The specific trip functions generated are as follows:

a. Pressurizer low pressure trip

The purpose of this trip is to protect against low pressure which could lead to DNB. The parameter being sensed is reactor coolant pressure as measured in the pressurizer. Above P-7 the reactor is tripped when the pressurizer pressure measurements (compensated for rate of change) fall below preset limits. This trip is

(Insert attached Insert 7.2-7A

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blocked below P-7 to permit startup. The trip logic and interlocks are given in Table 7.2-1.

The trip logic is shown on Figure 7.2-1, Sheet 6. A detailed functional description of the process equipment associated with the function is contained in reference [1].

Insert 7.2-7A

b. Pressurizer high pressure trip

The purpose of this trip is to protect the Reactor Coolant System against system overpressure.

The same sensors and transmitters used for the pressurizer low pressure trip are used for the high pressure trip except that separate bistables are used for trip. These bistables trip when uncompensated pressurizer pressure signals exceed preset limits on coincidence as listed in Table 7.2-1. There are no interlocks or permissives associated with this trip function.

The logic for this trip is shown on Figure 7.2-1, Sheet 2. The detailed functional description of the process equipment associated with this trip is provided in reference [1].

c. Pressurizer high water level trip

This trip is provided as a backup to the high pressurizer pressure trip and serves to prevent water relief through the pressurizer safety valves. This trip is blocked below P-7 to permit startup. The coincidence logic and interlocks of pressurizer high water level signals are given in Table 7.2-1.

The trip logic for this function is shown on Figure 7.2-1, Sheet 2. A detailed description of the process equipment associated with this function is contained in reference [1].

4. Reactor Coolant System Low Flow Trips

These trips protect the core from DNB in the event of a loss of coolant flow situation. The means of sensing the loss of coolant flow are as follows:

a. Low reactor coolant flow

The parameter sensed is reactor coolant flow. One elbow tap in each coolant loop is used as a flow device that indicates the status of reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow has occurred. An output signal from two out of the three bistables in a loop would indicate a low flow in that loop.

The coincidence logic and interlocks are given in Table 7.2-1.

The detailed functional description of the process equipment associated with the trip function is contained in reference [1].

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7. Garber, I., 'Topical Report, Test Report on Isolation Amplifier', WCAP-7689, June 15, 1971. 52

8. Lipchak, J. B. and Bartholomew, R. R., 'Test Report Nuclear Instrumentation System Isolation Amplifier,' WCAP-7506-P-A, April 1975 (Proprietary) and WCAP-7819-Revision 1-A, April 1975 (Non-Proprietary).

9. The Institute of Electrical and Electronic Engineers, Inc., 'IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations,' IEEE Standard 279-1971.

10. The Institute of Electrical and Electronic Engineers, Inc., 'IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems,' IEEE Standard 338-1971.

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4. Containment Pressure (Narrow Range)

a. Channels provided:

Four channels on separate power supplies. One channel is recorded.

b. Range: ~~1 lb/in^2~~ vacuum to ~~100%~~ 113% of containment design pressure (-1 to ~~15~~ 17 lb/in^2 d)

c. Purpose:

Determine margin to design pressure.

Monitor containment conditions following a break inside containment.

Verify if accident is properly controlled.

Determine actual containment breach.

d. Minimum accuracy: ~~±0.74~~ ±0.83 lb/in^2

5. Steamline Pressure

   a. Channels provided:

      Two channels per steam line on separate power supplies with one channel per steam line recorded.

   b. Range: 0 to 1300 lb/in^2g

   c. Purpose:

      Determine if high energy secondary line rupture has occurred.

      Maintain an adequate reactor heat sink.

      Verify AFW to steam generator associated with pipe rupture is isolated.

      Verify proper operation of pressure control steam dump system.

      Ensure proper cooldown rate.

      Diverse to T_{cold} for natural circulation determination.

      Determination that faulted SG is isolated.

   d. Minimum accuracy: ~~+173~~ lb/in^2

      ±134

Table 7.5-3

2. Containment Atmosph[illegible]e Hydrogen Concentration

a. Channels provided:

Two channels on separate power supplies.

b. Range: 0 to 10% hydrogen concentration (by volume)

c. Purpose:

To ~~assess~~ ASSESS the possibility of a hydrogen explosion inside containment during an accident.

d. Minimum accuracy: ~~±1%~~ ±1.4% hydrogen concentration (by volume)

Sheet 2
Added by Amendment 55

Vital instrument cables for the generating station protection system (GSPS) which includes the RPS and ESF may be routed in the same conduits, wireways, or cable trays provided the circuits have the same characteristics such as power supply and channel identity (I, II, III, or IV).

Automatic actuation and power circuits for the generating station protection system which includes the RPS and ESF may be routed in the same conduits, wireways, or cable trays provided the circuits have the same characteristics such as power supply and train identity (train A or train B).

Unit 1 analog circuits and Unit 2 analog circuits may be routed in the same conduits, cable trays, or wireways provided the circuits have the same characteristics such as power supply and channel identity (I, II, III, or IV). In like manner, Unit 1 train A cables may be routed in the same conduits, cable trays, and wireways as Unit 2 train A cables. Unit 1 train B cables may be routed in the same conduits, cable trays, or wireways as Unit 2 train B cables.

Cables installed prior to October 18, 1984

Cables for non-safety-related functions are not run in conduit used for essential circuits except at terminal equipment where only one conduit entrance is available. The non-safety-related cable is separated from the safety-related cable as near the device as possible. Generally, the non-safety-related cables are for annunicator functions. Exposed conduits containing redundant cables are separated by a minimum 1-inch air gap or by a minimum 1/2-inch thickness of Marinite (or its equivalent) fire-resistant barrier between conduits. Although there is no established minimum separation requirement between open top nondivisional cable trays and conduit containing redundant cables, it is concluded based on the following, that safety-related cables are not degraded. ~~Except as noted below, the exposed surfaces of cables~~ are coated with a fire resistant material. This coating significantly reduces the ignitibility and combustibility of cable insulation. TVA conducted fire tests, externally initiated by a propane burner, on a full scale-mockup of trays loaded with cables coated with a fire-resistant material. No self-sustaining fire could be established until the coating was fractured and cables separated. The cable coating also protects against development of a fire from electrical faults since it restricts availability of oxygen needed for combustion. Therefore, TVA takes credit for the ~~cable~~ coating, together with adequate circuit protective device(s) (as described below) as meeting the intent of Regulatory Guide 1.75 requirements to achieve independence between Class IE and non-Class IE cables routed in cable trays or conduits. Effective October 18, 1984, the use of coating on IEEE 383-1974 cables is not required except when the coating is used as part of electrical penetration fire stops as discussed in 8.3.1.4.4. 48 55

on cables not qualified to IEEE 383 flame test or equivalent

Cables for non-safety-related circuits may be run on cable trays with those for essential circuits with the following restrictions. When a non-safety-related cable is routed in a tray with essential (GSPS) cables, that cable or any cable in the

In all cable coating applications, up to 10 cables not qualified to the IEEE 383 flame test or equivalent may remain uncoated on cable trays

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opening through floors are covered with a fire barrier board that is cut to fit around the cables and cable tray configuration. All cables installed prior to March 4, 1985 ~~In addition, the exposed surfaces of cables~~ are coated from the fire barrier board for a minimum distance of 5 feet or to the nearest electrical panel or enclosure with an ablative material that is approved by Factory Mutual Research Corporation. 44

Effective March 4, 1985, this 5-foot cable coating requirement is discontinued if the cables have been qualified to IEEE 383 flame test or equivalent.

Typical electrical penetration fire stops and pressure seals through walls and floors that are designated fire barriers and pressure barriers are shown in Figure 6.2.3-3.

Insert Attachment A (8.3-49a)

Attachment A
(8.3-49a)

Conduit penetrations, containing cables, through designated fire barriers utilize Dow Corning 3-6548 silicone RTV foam (components A&B) as the sealant material. This material is installed around the cables in either the end of the conduit termination over a portion of the length of the penetration or in the nearest available conduit box on each side of the barrier. Inorganic fiber is used, and RTV silicone rubber may be used if needed as the damming material on each side of the sealant material. Spare conduits are plugged or capped until used.

There are no penetration fire stops installed at intervals in vertical and horizontal cable trays. Penetration fire stops are installed through walls and floors as described above.

The sealant material used in cable tray penetration fire stops is Dow Corning's 3-6548 silicone RTV foam (components A and B). This material in its cured foam state is noncorrosive and fire resistant. A sample of this material has been tested by an independent laboratory according to ASTM E84, standard method of testing of 'Surface Burning Characteristics of Building Materials.' The result of the test was that the material has a flame spread rating of 20. In addition, Dow Corning has conducted a flammability (vertical burn) test on samples of the cured foam in accordance with a corporate test method which is comparable to Underwriters Laboratories, UL 94.

The following results were obtained:

Flammability (Vertical Burn)

Time in Flame	Average Time to Flame- & Glow-Out	Average Percent % Weight Loss
15 seconds	7.2 seconds	1.3
60 seconds	15.6 seconds	13.5

The results from the vertical burn test show that the material self-extinguishes after flame source is removed.

The fire barrier materials used in the design and installation of the penetration fire stops employ a combination of inorganic fiber and fiber board. These materials are made from

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TABLE 8.3-28

Major Non-Safety Related Electrical Equipment That Could Become Submerged Following a LOCA

Equipment	Evaluation
Motors for the fans of the control rod drive mechanism coolers	These coolers are used to maintain the ambient temperature in the area of the control rod drives within an acceptable range during normal operation. Their function is not required for LOCA mitigation (Ref. section 9.4.7)
~~Motors for the fans of the lower compartment coolers~~	~~These coolers are used to maintain the ambient temperature in the lower containment within an acceptable range during normal operation. Their function is not required for LOCA mitigation (Ref. section 9.4.7)*~~
Reactor Coolant Drain Tank Pumps	These pumps remove from inside containment the normal leakage of the reactor coolant system that has been collected in the reactor coolant drain tank. This is not a safety function. The discharge path of the pumps is automatically isolated in a LOCA. (Ref. section 9.3.3.3)
Floor and Equipment Drain Sump Pumps	These pumps remove from inside containment any leakage inside containment that is not collected in the reactor coolant drain tank. This is not a safety function. The discharge path of the pumps is automatically isolated in a LOCA (Ref. section 9.3.3.3)
Pressurizer Heaters	Automatically deenergized in the event of a LOCA.

~~*If energized when flooded, the motors will short to ground and cause opening of the power supply breakers.~~

Added by Amendment 35

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During normal operation, the sources of radioactive material not normally considered part of the radioactive waste management system are as follows:

1. Containment Purging System

2. Turbine Gland Sealing System

3. Main Condenser Evacuation System

4. Auxiliary Building Ventilation System

5. Turbine Building Ventilation System

6. Steam Generator Blowdown System

Estimates for the release of radioactive materials from sources 1 through 6 (above) are presented in Section 11.3.7. The Turbine Gland Sealing System, the Main Condenser Evacuation System, and the Turbine Building Ventilation System are released from a common vent and are thus considered as a single release. The release paths and transport mechanism for these sources of radioactive material are also presented in Section 11.3.7. 52

The Steam Generator Blowdown System (SGBS) is another source of radioactive material that is not normally considered part of the Waste Management System. The system description, release paths, and flow rates are presented in Section 11.2 and in Section 10.4.8. The release path that is of concern in evaluating the radiological consequences of routing liquid releases consist of the optional path from the flash tank to the river via the cooling tower blowdown line. This route is used primarily during startups, when non-radioactive impurity levels are higher than normal. The normal route for the blowdown liquid is to the condensate demineralizer via the blowdown pump. The discharge from the flash tank to the river will be monitored for radioactivity as specified in Section 11.4. An alarm in the Main Control Room will alert the operator of an increasing radioactivity level in the discharge from the flash tank. If the radiation setpoint is exceeded, the blowdown discharge will be automatically diverted to the condensate demineralizers. The basis for the setpoint is presented in Section 11.4. 52

~~11.1.5 References~~ 42

~~1. American National Standard Source term Specification N237. Draft 'Specific Activity of Radioactive materials in the Principal Fluid Streams of Light-Water-Cooled Nuclear Power Plants'. ANS 18-1 Working Group; May 20, 1974.~~

Repeated on p. 11.1-5

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3. Thyroid doses (inhalation pathway)

4. Thyroid doses (pasture/cow/milk pathway)

The basic assumptions and calculational methods used in computing these doses are described in Subsection ~~11.3.8~~. 11.3.10.1.

Continual review of the data resulting from the offsite monitoring program and reevaluations of the adequacy of the dose models will ensure that the actual doses received by individuals and the population as a whole remain as low as practicable and within the applicable Federal Regulations.

11.6.2.2 Internal Doses from Liquid Effluents

The following doses will be calculated for exposures to radionuclides routinely released in liquid effluents:

1. Internal doses from the ingestion of water

2. Internal doses from the consumption of fish

3. External doses from water sports

A ~~detailed~~ description of the basic assumptions and calculational methods used in calculating the doses is given in Subsection ~~11.2.8.~~ 11.2.9.1 42

The dose models employed will be continually reevaluated in light of the data resulting from the offsite monitoring program to ensure that all significant pathways are included in the calculations and to ensure that the actual doses received by individuals and the population as a whole remain as low as reasonably achievable and within the applicable Federal Regulations.

11.6.3 SAMPLING MEDIA, LOCATIONS, AND ~~FREQUENCY~~

The proposed operational environmental radiological monitoring program is presented in Table ~~16A 3.2-1~~ 3.12-1 of the Technical Specifications. The final program will be outlined in the technical specifications issued with the operating license. The media selected were chosen on two bases: First, those vectors which would readily indicate releases from the plant, and secondly, those vectors which would indicate long-term buildup of radioactivity. Consideration was also given to the pathways which would result in exposure to man, such as milk and food crops. Locations for sampling stations were chosen after considering meteorological factors and population density around the site. Frequencies for sampling the various vectors

This should be transmitted prior to FL. DBE

TABLE 12.3-3
(Sheet 1 of 2)

Ventilation Air Exhaust Flow Rates

Location	Volume (cu ft)	Exhaust Rate (cfm)
REACTOR BUILDING (General Purge)	1,053,800	
Vicinity of Reactor Vessel		9150
Refueling Canal		18250
Instrument Room		600
MAIN CONTROL ROOM (Level of Control Building)	260,000	
AUXILIARY BUILDING - GENERAL	3,480,000	
Pipe Shafts		36600
Elevation 737		
Heat Exchanger Rooms		900
Hot Instrument Shop		1250
General Spaces		44200
Elevation 713		
Sample Room		~~4500~~ 5400
Waste Gas Compressor Rooms		3000
Decontamination Equipment Room		4000
Demineralizer and Filter Rooms		Note 1
Ion Exchange and Filter Rooms		Note 1
Sample Exhaust Hood		200
Waste Gas Analyzer Room		100
Valve Gallery		100
General Spaces		20800
Boric Acid Filter Rooms		Note 1
Demineralizer Ion Exchange and Filter Valve Room		500
Elevation 692		
Gas Stripper Rooms		5400
Hold-Up Tank Rooms		4000
Waste Evaporator Rooms		3000
Gas Decay Tank Rooms		2600
Turbine Driven Aux. Feedwater Pump Rooms		2400
Filter Storage Area		2000
Charging Pump Rooms		900
Valve Gallery		700
Safety Injection System Pump Rooms		600
Refueling Purification Filter Rooms		Note 1
Sample Exhaust Hood		200
General Areas		19050
Concentrates Filter Rooms		Note 1

TABLE 12.3-3 (Continued)
(Sheet 2 of 2)
Ventilation Air Exhaust Flow Rates

Location	Volume (cu ft)	Exhaust Rate (cfm)
Elevation 676		
Gas Stripper Feed Pump Room		3300
Waste Hold-up Tank Room		1950
Floor Drain Collector Tank Room		1450
Waste Evap. Feed Pump Room		1400
Sump Tank and Pump Room		1000
Containment Spray Pump Rooms		600
Residual Heat Removal Pump Rooms		600
General Areas		800
Waste Evaporator Feed Filter Rooms		Note 1
Auxiliary Waste Evaporator Feed Filter Rooms		Note 1
FUEL HANDLING AND RADWASTE PACKAGING AREA	1,012,900	
Fuel Handling Area		16000
Spent Fuel Pool Area		11,160
Waste Package Area		8010
Fuel Transfer Canal		5400
Cask Loading Area		4560
Cask Decontamination Tank Room		2600
Nitrogen Storage Area		2800
EGTS Room		2050
Blowdown Treatment Room		2050
Spent Resin Tank Room		1800
~~Fuel Transfer Valve Areas~~ Post Accident Sampling Facility		~~1720~~ 1450
Cask Decontamination Room		450
Condensate Demineralizer Waste Evaporator Area		1400
TURBINE BUILDING	8,000,000	
Elevation 820 Exhaust		570,000
Elevation 755 Exhaust		410,684
Lube Oil Purification Room		1300
DIESEL GENERATOR BUILDING	292,000	
Diesel Generator Rooms (1 of 2 fans operating)		180,000
Air Intake Rooms		22,000
Electrical Board Rooms		~~14,000~~ 11,400
Battery Exhaust Hoods		4000
Storage Room		1700
Fuel Transfer Room		200

Note 1: Flow is 2100 CFM with a filter room hatch removed.

Revised by Amendment 53