B&W FUEL COMPANY

BW

An American Company with Worldwide Resources

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November 9, 1993

Robert C. Pierson, Chief
Licensing Branch
Division of Fuel Cycle Safety and Safeguards, NMSS
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Pierson:

REFERENCE: Docket No. 70-1201, SNM-1168

B&W Fuel Company (BWFC) has modified it's ventilation system at it's Commercial Nuclear Fuel Plant (CNFP). The improvements were initiated to support the requirements of new part 20. In addition, the new system incorporates a new stack with a height of 21 meters versus the former height of 10 meters. The release of potentially uranium contaminated air from the stack has also been reduced. These design changes have significantly lowered offsite dose projections for the release of radioactive materials from the CNFP. The new system shall be operational by January 1, 1994.

In accordance with the provisions of 10 CFR 70.22(i)(1)(i), BWFC has performed an evaluation of the new ventilation system using USNRC Regulatory Guides 3.34 and 3.35 to demonstrate that the maximum dose to a member of the public offsite due to a releases of radioactive materials from the CNFP does not exceed 1 rem effective dose equivalent. With this, we request that the requirement for an Emergency Plan for NRC approval be exempt. We shall continue to maintain an emergency plan and implementing procedures for internal use to include an emergency response organization.

Attachment I contains the evaluation and its results. Chapter eight of our license, has also been revised to reflect this request. As required, six copies of this request have been provided. If you should require any additional information concerning this matter, please feel free to contact me at (804) 386-5202.

NF12 1

Sincerely,

B&W FUEL COMPANY
Commercial Nuclear Fuel Plant

Kathryn S. Knapp

Kathryn S. Knapp, Manager,
Safety and Licensing

cc: NRC Region II
101 Marietta St. N. W.
Atlanta, GA 30323

ATTACHMENT I

References

(1) NRC Regulatory Guide 3.34 (Rev. 1) 7/79.
(2) NRC Regulatory Guide 3.35 (Rev. 1) 7/79.
(3) NRC 10 CFR 70.22 (i).
(4) DOE/TIC 11223, "Handbook on Atmospheric Diffusion", 1982.
(5) B&W Environmental Report, BAW-1412, 1974.
(6) EPA-520/1-88-020," Federal Guidance Report No. 11.
(7) NRC License SNM-1168, Docket 70-1201, B&W Fuel Co., CNFP, Rev. 7, 6/16/93.

Facility Description

The B&W Fuel Company (BWFC), Commercial Nuclear Fuel Plant (CNFP) is located on a 76 acre site in Campbell County, Virginia approximately 4 miles from Lynchburg City limits. The CNFP fabricates nuclear fuel assemblies for commercial nuclear reactors utilizing relatively low enriched uranium oxide pellets. Authorization to possess more than 350 grams of U_{235} in powder, UF_6, or in solution form is not granted.

The CNFP site is adjacent to the Babcock and Wilcox NNFD and NES plant sites. The site is bounded on three sides by the James River and on the fourth side by Virginia State Route 726. State 726 connects with US Highway 460, which is a major link between Roanoke and Richmond. The site boundary is 250 meters from the plant. The nearest resident is about 800 meters east north east of the plant.

The principal steps in the manufacturing operations are as follows:

1.	Pellet receipt.	The UO2 pellets are packaged in cardboard containers and shipped in NRC approved packagings. The containers are stored on shelves in cubicles denoted as the vault in the pellet loading room.
2.	Cladding receipt.	The zircaloy cladding upon receipt is inspected and then one end cap is welded in place prior to fuel pellet loading.
3.	Pellet Loading.	The pellets are loaded and the second end cap is welded in place. After the weld is ultrasonically inspected, the fuel rods are pressurized with helium.
4.	Machine Shop.	Machining of components used to fabricate fuel assemblies.
5.	Parts Cleaning.	All machined parts and the fuel rods are chemically cleaned in the parts cleaning room.
6.	Fuel Assembly.	Acceptable fuel rods are inserted into the rod holding cage and the fittings of the fuel bundle assembly.

7. Fuel Shipment. The completed fuel bundle assemblies are cleaned, inspected, packaged, and stored or shipped to a commercial reactor site.

Typical quantities of SNM involved in process areas of the plant are provided in below:

LOCATION OF FUEL	MATERIAL FORM AND QUANTITY	AMOUNT OF URANIUM (KG)
Pellet Loading Room Vault	Pellets - 330 boxes	7,260
Manufacturing area	Pellets contained in rods - 600 rods.	1,320
	Fuel rods in channels - 25 channels.	11,350
Fuel Bundle Assembly Room	Assemblies on fabrication tables - 4 assemblies	1,816
Inspection Area	Fuel assemblies - seven	3,178
Storage Area	Fuel assemblies - sixty	27,240
Shipping Area	Fuel assemblies - twelve	5,448
TOTAL	**NA**	**57,612**

The CNFP also has two buildings that support Field Service activities. The equipment associated with these operations are contaminated with by-product material which is under by our SNM license. Our SNM license authorizes possession of 10 curies of by-product material but our typical by-product inventory onsite is less than 5 curies.

The main plant stack exhausts air from the pellet loading room and SERF-1. It is located on the south east end of the plant and is 21 meters from the ground. The discharge is 11,000 cfm and the air goes through a pre-filter and a HEPA filter prior to being discharged. 2,000 cfm originates from the pellet loading room (PLR) and the remaining 9,000 cfm enters from the Service Equipment Refurbishment Facility (SERF-1).

In the event of a criticality, only the air from the PLR would be effected by the event. The air from the SERF-1 which is essentially uncontaminated air would dilute the contaminated air and the radioactive release from the stack would be minor.

Introduction

An analysis of the CNFP facility and processes was conducted to identify and quantify the potential environmental impact of postulated conceivable accidents. Since CNFP operations involve UO_2 in pellet form only, the consequences of an accident at CNFP that has an environmental impact is in the case of a criticality. In the event of a criticality, the major environmental impact would result from the release of fission products. In order for the fission products to be dispersed to affect the general public, it would have to be assumed that the event took place in the pellet loading room and the ventilation system remained operational.

The objective of this analysis and evaluation is to assess the risk to public safety and health resulting from a postulated criticality accident for the first 30 minutes. The guides used in this report for the analysis and evaluation of the postulated criticality accident are adopted from Reg. Guide 3.34 (ref. 1) and Reg. Guide 3.35 (ref. 2). These guides are used by the NRC staff in the analysis of such accidents and are described in the USNRC Regulatory Guides 3.34 and 3.35 (reference 1 and 2). Furthermore, it is stated in Reg. Guide 3.34 that these methods resulted from review and action on a number of specific cases and reflect the latest general NRC-approved approach to the problem.

The assumptions (listed in Reg. Guide 3.34) used in the NRC-approved approach to evaluate an estimate of the radiological consequences of various postulated accidents have been developed from reviewing applications for licenses to operate uranium fuel fabrication plants. These assumptions are considered by the NRC to be "appropriately conservative," and they are based on previous accident experience, engineering judgement, and the analysis of applicable experimental results from safety research programs.

For the purpose of estimating the radiological consequences of the CNFP fuel fabrication facility, a description of a credible criticality accident is not relevant and will not be included in this report. It will be assumed that a criticality event takes place and the characteristics of this event will be assumed as those described in Reg. Guide 3.34 and 3.35.

This report provides the analysis of the effects of a criticality event with characteristics based on the assumptions adopted from Reg. Guide 3.34 on individuals at the site boundary the nearest residence, and the highest exposed individual for the first 30 minutes of the event.

Characteristics of criticality event

The characteristics of a criticality event for use in estimating the consequences of a criticality accident in process systems are outlined in Reg. Guide 3.34 and summarized next. A review of 34 occasions prior to 1966 in which the power level of a fissile system increased without control as a result of unplanned or unexpected change in its reactivity resulted in estimates for the total number of fissions per incident. The estimated total number of fissions ranged from 1E+15 to 1E+20 with a median of about 2E+17. The shutdown for these events was caused by the action of an automatic control device or as a consequence of the fission energy release which can cause thermal expansion, density reduction due to the formation of bubbles, loss of water moderator by boiling, or expulsion of parts of the mass. These incidents occurred in a number of different fissile systems including inhomogeneous water-moderated systems.

The criticality events were observed to have an initial burst followed by a number of bursts of lesser and declining fission rate. The suggested estimates of possible fission yields from excursions in various type of systems are spike yields of 1E+17 for criticality accidents occurring in solution systems of 100 gallons or less and 3E+19 fissions for solution systems with more than 100 gallons. The mechanical damage predicted at these levels will be little or none.

The development of methods for estimating the number of fissions in the initial burst and the total number of fissions can be found in Reg. Guide 3.34 (reference 1). It is also stated in Reg. Guide 3.34 that the fission yields for criticality accidents occurring in solutions and some heterogeneous systems (aqueous-fixed geometry) can be estimated with reasonable accuracy using "existing" methods. Therefore, for simplicity and conservatism, the magnitude of the criticality event should be taken "equal to, or some multiple of, the historical maximum for all criticality accidents outside reactors without using any scenario clearly defined by the specific operation being evaluated."[Ref. 1]

Furthermore, because of the nature of the storage facility, the form of the fissile material and the way it is stored, it can be argued that no possible criticality accident occurring as a result of simultaneous breakdown of at least two independent controls in the facility will exceed in severity the criticality accident postulated in Reg, Guide 3.34, section C (reference 1). Therefore, the magnitude of the criticality event used for estimating the radiological consequences of a criticality accident will be assumed to consist of an initial burst of 1E+18 fissions in 0.5 seconds followed successively by bursts of 1.9E+17 fissions at 10-minute intervals for a total of 1E+19 fissions in 8 hours. Per Reg. Guide 3.34 and 3.35 these values bound accident scenarios at fuel processing plants.

Assumptions used to estimate the release of radioactive material

The following is a list of the assumptions which should be made to comply with the requirements outlined in Reg. Guide 3.34 & 3.35 (ref. 1 & 2):

1 - It will be assumed that all of the noble gas fission products and 25% of the iodine radionuclides resulting from the criticality accident are released directly to a ventilated room atmosphere.

2- Aerosol should be assumed to be released directly to the room atmosphere comprising 0.05% of the "salt content of the solution that is evaporated." But, since fissile material is present at the CNFP facility only in the form of UO_2 pellets, which is a insoluble ceramic material, the "salt content" of the inhomogeneous mixture will be assumed to be negligible.

3- The effects of room volume and air ventilation rate and retention time should be considered on an individual basis.

4- The effects of radiological decay during transit within the plant should be evaluated on an individual basis.

5- A reduction in the amount of radioactive materials available for release to the plant environment through filtration systems in the plant exhaust system(s) may be taken into account on an individual basis.

6- Table 1 in Reg. Guide 3.34 (reference 1) lists the radioactivities of significant radionuclides released, but it does not include the iodine depletion allowance (25% of the iodine radionuclides released directly to the room atmosphere).

Assumptions used for atmospheric diffusion

Since the CNFP plant employs a ventilation system which exhausts the gaseous effluents through a stack, certain assumptions, which are listed in Reg. Guide 3.35 (reference 2), have to be made for a conservative estimate of the atmospheric diffusion calculations. The following is a list of the assumptions made in this analysis in accordance with the Reg. Guide 3.35 recommendations:

1- The elevated releases are considered to be at the actual stack height. Although, there is an "elevated topography" (Mt. Athos) to the southeast of the plant, the effects are ignored in this analysis due to the prevailed wind direction (from the southwest) as described in reference 5 (BAW-1412).

2- Since the plant has a stack, the atmospheric diffusion model used is the one outlined in Reg. Guide 3.35 section 4.b. (reference 2).

For conservatism and according to Reg. Guides 3.34 and 3.35, the stack discharges are assumed to be transported by wind having speed of 1 meter per second and atmospheric stability Class F to and beyond the site boundary where individuals may breathe the air for at least 30 minutes. This assumption is clearly conservative since the average wind speed recorded year around is over 3.5 meters per second and the atmospheric dispersion (X/Q) is inversely proportional to the wind speed (ref. 5).

The plume rise due to the momentum rise and buoyant rise is ignored for more conservatism. The plume rise increases the effective stack height which decreases the atmospheric dispersion factor X/Q and reduces the concentration of the radioactive nuclides in the air at the point of maximum X/Q.

Selection of Area for analysis

The area selection where the radioactive materials are assumed to be released is based on the worst case scenario. The facility consists of two major areas. The first is the Pellet Loading Room and Vault (PLR) area, which has a volume of 491 m^3. This area is ventilated and has boxes of UO_2 pellets stored on shelves and UO_2 pellets on trays on the down draft table where they await loading into fuel pins. The second major area is the rest of the plant. This area is not ventilated and has a volume many times larger than the Pellet Loading area. Furthermore, this area does not have any UO_2 pellets outside fuel pins or fuel assemblies.

A release of radioactive materials due to a criticality accident in the pellet loading room area has more impact on the outside of the facility. The concentration of the radioactive nuclides in air released to the atmosphere through the stack from the PLR area will be much higher. The air from the PLR is released directly into the outside atmosphere after being diluted with air from the SERF-1 bay. Moreover, the air from the rest of the plant is not discharged to the outside atmosphere, and the effect of a release of radioactive materials in this area on the outside is believed to be much lower than the previous case.

For these reasons the Pellet Loading Room and Vault area will be selected for this analysis. A criticality event will be assumed to take place in the Pellet Loading Room (PLR) and vault area. As required by Reg. Guide 3.34 and as a conservative assumption, this area's ventilation system is assumed to continue operation during and after the accident for at least 30 minutes. The air discharged from the PLR area is diluted with air from the SERF-1 bay. The diluted air is then released to the atmosphere through the plant stack.

Other assumptions

Prompt neutron and gamma doses have been calculated at the site boundary (250 m), the nearest residence (800 m), and the maximum X/Q receptor point (1000 m) (ref. 5). Eight inch thick concrete walls are present around the PLR area and assumed as attenuation media. Other walls do not present an attenuation medium in the path of the prompt neutrons and gamma rays and are ignored for conservatism.

The prompt neutron and gamma doses are added to the inhalation dose of the principal nuclides. Considerations have been made for short as well as longer lived nuclides. Appropriate inhalation dose conversion factors, which are recommended by Reg. Guide 3.35, have also been applied as part of the inhalation dose assessment. These conversion factors can be found in reference 6.

It is assumed that the volume of the vault is the measured value of 491 m^3. The number of boxes normally stored on the shelves in the pellet loading room vault was counted to be 330 boxes containing 7,260 Kg. Each area has 110 boxes (2,420 Kg uranium) and 55 boxes (1,210 Kg uranium) on each set of shelves.

The total stack discharge rate is assumed to be 11,000 cfm with 2000 cfm discharge rate from the PLR area.

Calculations

The following is a list of the calculations performed in this report:

1. The prompt neutron and gamma doses are calculated at 250 m, 800 m and 1000 m using formulas from RG 3.34 (Rev.1) 7/79.

2. Airborne dose calculations at site boundary, nearest residence and maximum receptor point. To include: event room volume, discharge rate, and room air concentration vs. time.

3. Stack air concentration vs. time. To include: stack release concentrations attributed to each pulse of the first 30 minutes of the criticality event

4. Calculate diffusion or dispersion factors for the site boundary (250 meters), the nearest residence (800 meters), and the point of maximum X/Q at ground level location (1000 meters).

5. Inhaled dose.

6. Sum the doses and compare against the criteria of 10CFR 70.22(i).

Prompt Gamma & Neutron Dose Calculation at Site Boundary (250 m), Nearest Residence (800 m) and 1000 m

The prompt gamma and neutron doses were calculated using the following equations from Reference RG-3.34 (Rev. 1) 7/79.

Prompt gamma dose $= 2.1E\text{-}20\ N\ d^{-2} \exp(-3.4d)$

Prompt neutron dose $= 7.0E\text{-}20\ N\ d^{-2} \exp(-5.2d)$

where

N = number of fissions
d = distance from source (km)

For 8 inch concrete walls, reduce by:

gamma reduction = 1/2.5
neutron reduction = 1/2.3

The prompt dose for 30 min is equal to the prompt dose from 10^{18} fissions (1st pulse) and the doses from three pulses of 1.9E17 fissions each .

Dose from 10^{18} fissions

At 250 m is 0.1436 rem_{γ} + 0.3052 rem_{n} = 0.4488 rem
At 800 m is 0.0022 rem_{γ} + 0.0017 rem_{n} = 0.0039 rem
At 1000 m is 0.0007 rem_{γ} + 0.0004 rem_{n} = 0.0011 rem

Dose from 1.9×10^{17} fissions

At 250 m is 0.0273 rem_{γ} + 0.0580 rem_{n} = 0.0853 rem
At 800 m is 0.0004 rem_{γ} + 0.0003 rem_{n} = 0.0007 rem
At 1000 m is 0.0001 rem_{γ} + 0.0001 rem_{n} = 0.0002 rem

Total dose for 30 min (without attenuation factor)

At 250 m is 0.4488 rem + 3(0.0853 rem) = 0.7047 rem
At 800 m is 0.0039 rem + 3(0.0007 rem) = 0.0060 rem
At 1000 m is 0.0011 rem + 3(0.0002 rem) = 0.0017 rem

The results of the calculated prompt gamma and neutron doses are presented in Table I. The attenuated doses at the site boundary and nearest residence are also shown in the table.

TABLE I
PROMPT GAMMA & NEUTRON DOSES

Number of fissions	1.0E+18	with 8" concrete	1.0E+18	with 8" concrete	1.9E+17	with 8" concrete	1.9E+17	with 8" concrete
Dist. from source (m)	250	250	800	800	250	250	800	800
Prompt Gamma dose (rem)	0.1436	0.0574	0.0022	0.0009	0.0273	0.0109	0.0004	0.0002
Prompt Neutron dose (rem)	0.3052	0.1327	0 .0017	0.0007	0.0580	0.0252	0.0003	0.0001
Total Dose (rem)	0.4488	0.1902	0.0039	0.0016	0.0853	0.0361	0.0007	0.0003

Airborne Dose Calculations at Site Boundary and Nearest Residence

The amount of UO_2 normally present in the PLR area is carefully packaged and placed so as to insure that under optimum moderation conditions the maximum k_{eff} is less than 0.95. The UO_2 pellets are stored on trays, 20 rows per tray, and placed in boxes, 10 trays per box and stored in the vault on six sets of shelves, two per area. The amount of UO_2 material that can be included in an accident scenario is not very large if any. For a "non-catastrophic" type accident, the structure of the PLR is assumed to hold and the ventilation system is assumed to operate for the first 30 minutes as recommended by Reg. Guide 3.34.

The assumptions and characteristics of the criticality event are adopted from Reg. Guide 3.34 and 3.35. This criticality event will have one pulse of 1E+18 fissions followed by pulses of 1.9E+17 fissions ten minutes apart for eight hours, a total of 47 pulses. The radioactive materials released from each of the pulses are assumed to rapidly mix in the air present in the surrounding open volume. The air is then discharged through the ventilation system and diluted with SERF-1 air before being discharged out of the plant stack into the atmosphere.

The stack discharges are assumed to be transported by wind having speed of 1 meter per second and atmospheric stability Class F to and beyond the site boundary where individuals may breathe the air for at least 30 minutes. The atmospheric dispersion is estimated using the model recommended in Reg. Guide 3.34. This development is presented in the next section. Furthermore, the average breathing rate is assumed to be 3.5 E-4 m^3/sec, according to Reg. Guide 3.35, which will result in 0.63 m^3 of air breathed in 30 minutes period.

Table II shows the calculated room volume and discharge rates. The dilution factor for the PLR area is shown in Table III. The concentrations of the nuclides released from the first pulse of 10^{18} fissions and pulses of 1.9×10^{17} fissions are presented in Table IV. Table V shows the dose conversion factors for inhaled radionuclides.

TABLE II

ROOM VOLUMES AND DISCHARGE RATES

	Room Volume (m^3)	Room Discharge Rate (m^3/sec)	Room Discharge m^3/min
Pellet Loading Room & vault area	491 m^3	0.9433	56.63

TABLE III
STACK DILUTION FACTORS

Total stack discharge rate = 11000 cfm (311.485 m^3/min)
Pellet Loading Room Area's = 2000 cfm (56.63 m^3/min)

Tne PLR area's discharge is diluted to **0.1818** of the room's concentration of discharge rates.

TABLE IV

ROOM CONCENTRATIONS FROM CRITICALITY

Nuclide	T1/2	Ci (1E+18)	Ci/m^3 (1E+18)	Ci (1E+17)	Ci/m^3 (1E+17)
Kr 83m	1.8 h	14.08	2.87E-02	2.64	5.37E-03
85m	4.5 h	13.44	2.74E-02	2.52	5.13E-03
85	10.7 y	1.4 E-4	2.87E-07	2.62 E-5	5.37E-08
87	76.3 m	89.6	1.82E-01	16.80	3.42E-02
88	2.8 h	58.25	1.19E-01	10.92	2.22E-02
89	3.2 m	3.78 E+3	7.68E+00	7.08 E+2	1.44E+00
Xe 131m	11.9 d	7.04 E-3	1.43E-05	1.32 E-3	2.69E-06
133m	2.0 d	1.6 E-1	3.26E-04	3.0 E-3	6.11E-05
133	5.2 d	2.43	4.95E-03	0.456	9.28E-04
135m	15.6 m	1.98 E+2	4.04E-01	37.2	7.57E-02
135	9.1 h	32.0	6.51E-02	6.0	1.22E-02
137	3.8 m	4.42 E+3	8.99E+00	8.28 E+2	1.69E+00
138	14.2 m	1.15 E+3	2.34E+00	2.16 E+2	4.40E-01
I 131	8.0 d	0.192	3.91E-04	0.036	7.33E-05
132	2.3 h	24.0	4.88E-02	4.5	9.16E-03
133	20.8 h	3.52	7.16E-03	0.66	1.34E-03
134	52.6 m	100.80	2.05E-01	18.9	3.85E-02
135	6.6 h	10.56	2.15E-02	1.98	4.03E-03

Values in were taken from Table 1 in Reg. Guide 3.35 on page 3.35-10. The activities for the nuclides in Table 1 in Reg. Guide 3.35 are the cumulative yield for the first 0.5 Hours (1.57E+18 Fissions). The values of the nuclides activities in this table are multiplied by (1.9E+17/1.57E+18).

The iodine activities in include the iodine reduction factor (25%) allowed in Reg. Guide 3.35 Section C.2.a.

The value in the last column are the nuclide activities per m^3 for a room volume of 491 m^3.

Table V

The Dose Conversion Factors for Inhaled Radionuclides

Iodine	**mrem/μCi**
131	3.3 E+1
132	3.8 E-1
133	5.8 E-0
134	1.3 E-1
135	1.2 E-0

Values in this table are derived from the Federal Guidance Report No. 11 (EPA-520/1-88-020).

Stack Air Concentration VS. Time

The air concentration for long lived nuclides and gas components is given by the following equation.

$$C(t) = C_o \ \mathrm{Exp}(- Wt/V)$$

where

W	=	room discharge rate m^3/minute
V	=	room volume m^3
t	=	time (minutes)
C_o	=	initial concentration

The stack discharge activity as a function of time for the nuclides with relatively shorter half-lives is given by the following equation.

$$\begin{aligned} C_i(t) = \ & C_{io} e^{-\lambda t} \mathrm{df}\, W e^{-Wt/V} \quad + \\ & C_{i10} e^{-\lambda(t-10)} \mathrm{df}\, W e^{-W(t-10)/V} \quad + \\ & C_{i20} e^{-\lambda(t-20)} \mathrm{df}\, W e^{-W(t-20)/V} \quad + \\ & C_{i30} e^{-\lambda(t-30)} \mathrm{df}\, W e^{-W(t-30)/V} \end{aligned}$$

When the (t-n) exponent is less than 0, set the entire exponent to zero.

C_{io} is the initial concentration of nuclide i due to the first pulse of 1E+18 fissions. Also note that C_{i10} ,C_{i20} ,and C_{i30} are the initial concentrations of nuclide i due to the second, third, and fourth pulses and are equal.

The following three figures show the concentrations of the released radionuclides in air at the stack discharge as a function of time for the first 35 minutes due to the assumed pulses (ten minutes apart).

FIGURE 1
Kr CONCENTRATION IN AIR AS RELEASED TO ATMOSPHERE

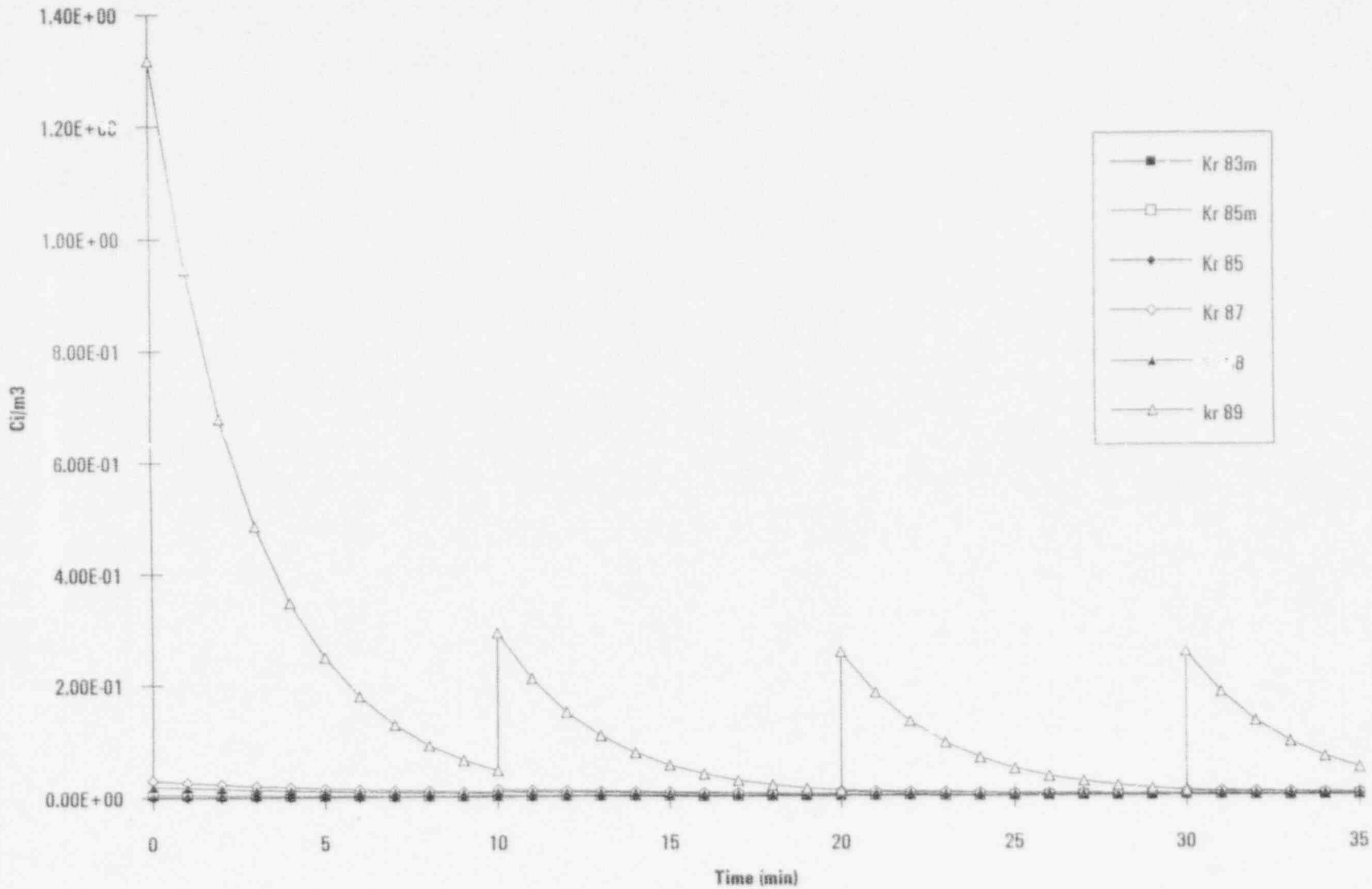


FIGURE 2
Xe CONCENTRATION IN AIR AS RELEASED TO ATMOSPHERE

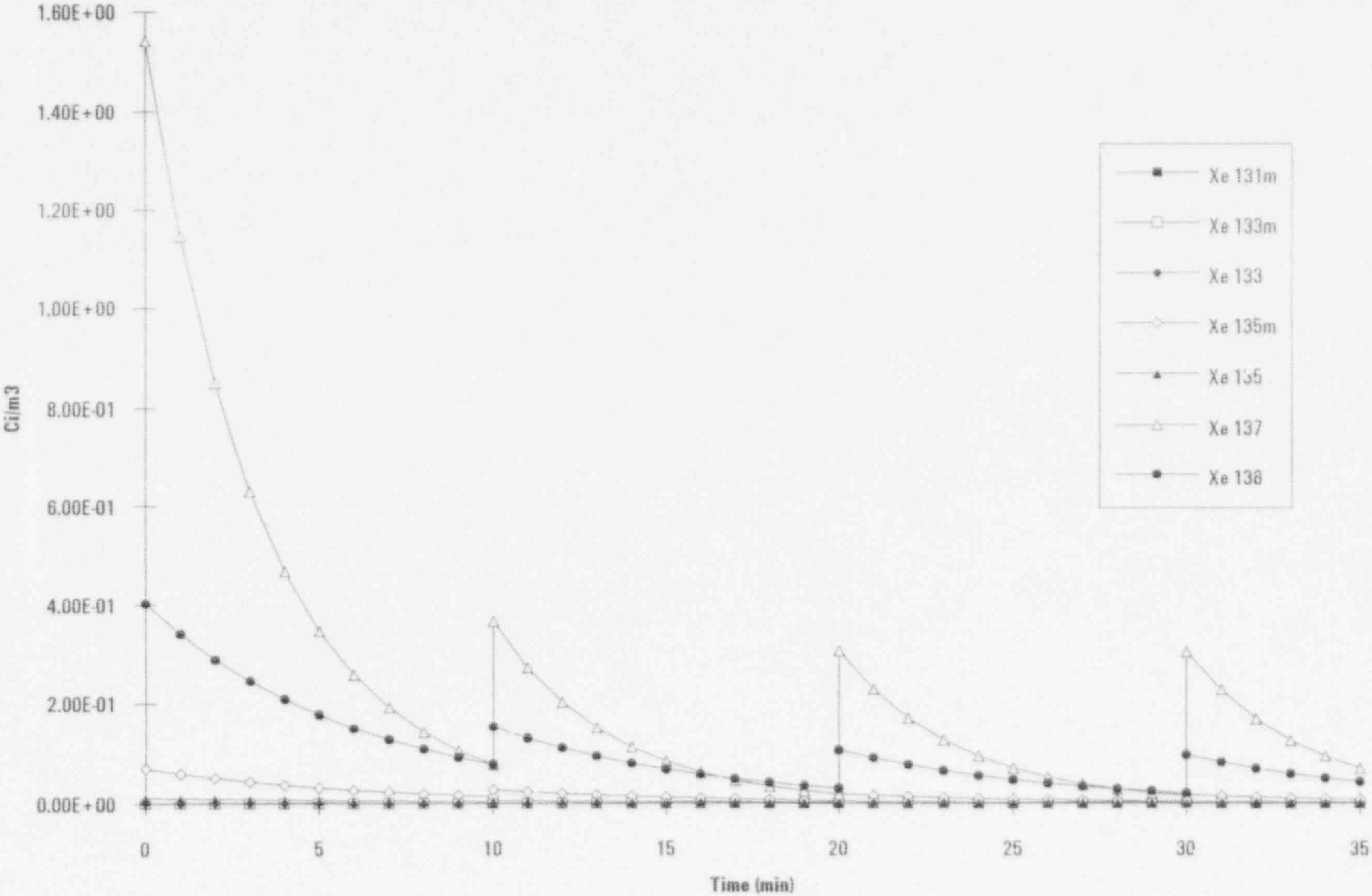
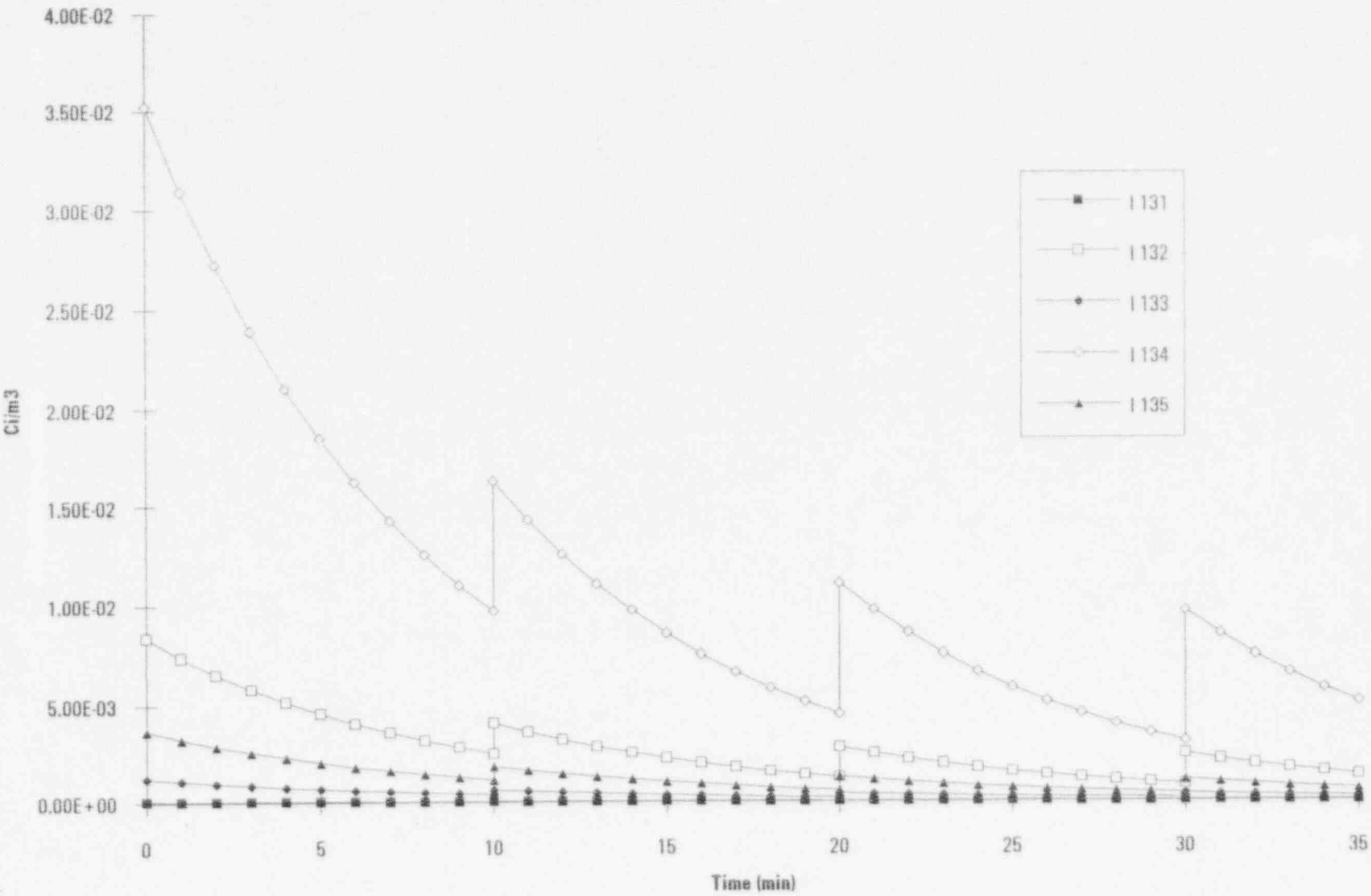


FIGURE 3
Iodine CONCENTRATION IN AIR AS
RELEASED TO ATMOSPHERE



Atmospheric Dispersion

The atmospheric dispersion can be estimated by using the following equation from REG 3.34 (ref. 1) to calculate the X/Q at a distance (x) meters from an elevated release.

$$X/Q = \exp(-h_e^2/2\sigma_z^2)/(\pi\mu\sigma_y\sigma_z)$$

where:

μ	=	wind speed (m/sec)
σ_y	=	horizontal standard deviation (m)
σ_z	=	vertical standard deviation (m)
x	=	distance (m)
h_e	=	effective stack height (m)
h_s	=	stack height (m)

where h_e = h_e + plume rise
plume rise = momentum rise + buoyant rise

momentum rise = $1.6\ F_o^{-1/2}\ \mu^{-1}\ x^{2/3}$
buoyant rise = $2.9\ (F/\mu S)^{1/3}$

where S = stability parameter defined as

$$S = (g/T_a)\left\{ \partial T_a/\partial z + \Gamma \right\}$$

where z = vertical distance above stack
and Γ = 0.0098° K/m

A complete description of this atmospheric dispersion model can be found in NRC Regulatory Guide 3.34 (Rev. 1) 7/79 (references 1).

The plume rise will be ignored and the effective stack height h_e = 21 m.

From a dispersion handbook (Ref. 4) at F stability in open country

$\sigma_y = 0.04\ x\ (1 + 0.0001\ x)^{-1/2}$
$\sigma_z = 0.016\ x\ (1 + 0.0003\ x)^{-1}$

The following table is generated by assuming wind speed of 1 m/sec and solving for σ_y, σ_z, and X/Q as a function of distance.

TABLE VI

TABLE OF VALUES VS. DISTANCE

X (m)	σ_y(m)	σ_z(m)	$\pi\mu\sigma_y\sigma_z$	Stack Ht.(m)	$-h_e^2/2\sigma_z^2$	X/Q
250	9.88	3.72	115.46	21	-15.93	1.05E-09
400	15.69	5.71	281.65	21	-6.75	4.15E-06
600	23.31	8.14	595.80	21	-3.33	6.00E-05
800	30.79	10.32	998.56	21	-2.07	1.26E-04
1000	38.14	12.31	1474.65	21	-1.46	1.58E-04
2000	73.03	20.00	4588.59	21	-0.55	1.28E-04
5000	163.30	32.00	16416.64	21	-0.22	4.81E-05
10000	282.84	40.00	35543.06	21	-0.14	2.45E-05
15000	379.47	43.64	52021.12	21	-0.12	1.71E-05
20000	461.88	45.71	66333.23	21	-0.11	1.36E-05
25000	534.52	47.06	79023.62	21	-0.10	1.15E-05
30000	600.00	48.00	90477.87	21	-0.10	1.00E-05
40000	715.54	49.23	110667.85	21	-0.09	8.25E-06
50000	816.50	50.00	128254.98	21	-0.09	7.14E-06

Max X/Q = 1.58E-04 @ 1000 meters

Inhaled Dose Calculation

The inhaled dose was calculated using the following steps:

1- The fission yields for the three pulses are adopted from reference 1.

2- The concentrations of the three nuclide from reference 1 are calculated by dividing the yield of each nuclide by the PLR area of 491 m^3.

3- The nuclides concentrations at time=0 at the stack are calculated by multiplying the value from step 2 by the stack dilution factor. The stack dilution factor is calculated by dividing the PLR discharge rate by the total discharge rate.

4- The nuclide concentrations at the stack vs. time is calculated using the derived equations (on page 20) and plotted as a function of time for each nuclide for the first 30 minutes as shown in figures 1,2, and 3.

5- The total energy inhaled is calculated by finding the area under the curves of step 4 (integrate) for each of the nuclides and multiplying the calculated area by 0.63 and by the average Beta and Gamma energies per disintegration (MeV/dis.) for each of the nuclides. The breathing rate is assumed to be 3.5E-4 m^3/sec. (reference 1). Therefore the air volume breathed is 0.63 m^3 of air for the first 30 minutes.

6- The values calculated in step 5 are multiplied by the dose conversion factors for inhaled radionuclides derived from the Federal Guidance Report No. 11 (EPA-520/1-88-020) for each nuclide.

7- The total inhaled dose rate is the sum of all values from step 6 for all nuclides of appreciable concentrations (iodine isotopes).

For a stack 21 meters high and a dilution factor of 0.1818, the inhalation dose would be negligible at the nearest site boundary (250 meters), 0.50 mrem at the nearest residence, and 0.62 mrem at the receptor point of maximum X/Q (at 1000 meters).

Conclusion

The results of the projected dose calculations performed are illustrated in Table VII.

TABLE VII

PROJECTED DOSES FOR A 30 MINUTE EXPOSURE PERIOD

DOSE (REM) @ DISTANCE	250 METERS	800 METERS	1000 METERS
Prompt Gamma Dose	0.2255	0.0034	0.001
Prompt Neutron Dose	0.4792	0.0026	0.0007
Total Prompt Gamma and Neutron Dose	0.7047	0.0060	0.0017
Inhalation Dose	negligible	0.50	0.62
Total Prompt Gamma, Neutron and Inhalation Dose	0.7047	0.5060	0.6217
Prompt Gamma Dose With 8" Concrete	0.0902	0.0014	0.0004
Prompt Neutron Dose With 8" Concrete	0.2083	0.0011	0.0003
Total Prompt Gamma and Neutron Dose With 8" Concrete	0.2985	0.0025	0.0007
Inhalation Dose	negligible	0.50	0.62
Total Prompt Gamma, Neutron and Inhalation Dose With 8" Concrete	0.2985	0.5025	0.6207

As illustrated in the table, the maximum dose is at the site boundary (0.7047 rem). All of the dose is from prompt gamma and neutron exposure. The dose is below the allowable 1 rem per 10 CFR 70.22 and it is a conservative estimate since the concrete shielding has been ignored. With the shielding, the dose is reduced to 0.2985 rem. The projected total effective dose at the receptor point of maximum X/Q (1000 meters) is 0.6217 rem without shielding and 0.6207 rem with shielding. Due to the large distance, shielding is not a factor.

Essentially, the inhalation dose increases with distance up to 1000 meters and then begins to decline and the prompt gamma and neutron dose decreases with distance. Comparatively, the dose projections not including shielding are fairly close at 250 meters, 800 meters and at 1000 meters. However, when shielding is included, dose projections for the site boundary significantly decrease while the projected dose at 800 and 1000 meters is barely effected. With this, it can be determined that the projected dose, worst case is below the criteria established in 10 CFR 70.22 and therefore an Emergency Plan should not be required.

8.0 RADIOLOGICAL CONTINGENCY PLAN

An evaluation has been performed to demonstrate that the that the maximum dose to a member of the public offsite due to a release of radioactive materials from the CNFP does not exceed 1 rem effective dose equivalent. With the evaluation, in accordance with the provisions of 10 CFR 70.22(i)(1)(i), BWFC is not required to maintain an NRC approved Emergency Plan. An emergency plan and implementing procedures for internal use to include an emergency response organization shall be maintained.

If process changes or modifications to the ventilation system occur that could effect the offsite dose projection, CNFP shall verify that the evaluation submitted to the Commission on November 9, 1993 is still valid and that the dose does not exceed criteria of 10 CFR 70.22.