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Public Service Company of Colorado

July 13, 1981
Fort St. Vrain
Unit No. 1
P-81184

RECEIVED
AUG 03 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. Karl V. Seyfrit, Director
U. S. Nuclear Regulatory Commission
gion IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

SUBJECT: Fort St. Vrain Unit No. 1
Emergency Planning Early
Warning Alert System

REFERENCE: NRC Letter Seyfrit to
Lee Dated July 1, 1981

Dear Mr. Seyfrit:

In response to your letter of July 1, 1981, we recognize the position setforth by 10CFR50 Appendix E with regard to an early warning alert system. As you are no doubt aware we have resisted and in many cases have taken specific exception to criteria that was developed for water reactors without any justification or technical basis for applying these same criteria to a gas cooled reactor and in particular Fort St. Vrain. The early warning alert system has been one of these areas.

As early as March, 1980, in our letter P-80083, we outlined our position on the early warning alert system, and in the submittal of our emergency response plan and the State's response plan our position on the early warning alert system was setforth.

This was followed by several other submittals in response to various letters and NUREG's all of which setforth our position to various requirements and guidance that had been published by the Nuclear Regulatory Commission concerning TMI-2 actions and associated emergency response. We did not receive any response to these submittals and we finally requested a meeting with the emergency preparedness group and the special projects staff which was held on December 10, 1980, in Bethesda.

This meeting did not result in any resolution of our concerns although we were promised comments on our emergency response plan and the various issues that had been raised. We are still waiting for those comments.

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Recognizing the futil'ty of some of our efforts we appealed dirctly to the Commission on Ja. iary 9, 1981 (P-81009) concerning the early warning alert system wherein technical justification for relief from the early warning alert system was provided. To date we have received no comments on our position.

It is our understanding that our various submittals have been turned over to Oak Ridge for evaluation, but as of this date we have received no response.

We believe we have been responsive to all Nuclear Regulatory Commission action items, and obviously because of our reactor concept have provided technical justification for various exceptions including the early warning alert system.

We are enclosing herewith a copy of P-81009 which contains several attachments which have been submitted to the Nuclear Regulatory Commission which setsforth our notification system. The State RERP expands on the various methods which the State intends to utilize for notification, radiological exposure control, and evacuation if necessary. Our public information brochures have been distributed to the population in the 5 mile EPZ, and we have held two (2) meetings for the general public in the EPZ to educate the people on the particulars of our emergency response plan.

We believe our plan is more than adequate given the accident scenarios for Fort St. Vrain, and we are not proceeding with any further activities concerning the early warning alert system until we receive a response to our various submittals.

Very truly yours

Don W Warembourg

Don W. Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclea
Generating Station

DWW/alk

cc: Brian Grimes
Jim Miller
George Kuzmycz

Public Service Company of Colorado

January 9, 1981
Fort St. Vrain
Unit No. 1
P-81009

Dr. John F. Ahearne, Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: Fort St. Vrain Unit No. 1
Emergency Planning Early
Warning Alert System

Dear Dr. Ahearne:

In our letter of April 1, 1980, to Mr. Brian Grimes (P-80066) which is included as Attachment I, we set forth our evaluation, and justification for reducing the EPZ and Ingestion Pathway distances from 10 and 50 miles respectively, to 5 and 30 miles. Our justification was accepted and the EPZ and Ingestion Pathway was subsequently reduced along with those of LWR's 250 MW and smaller.

In our transmittal of our emergency response plan on March 18, 1980, we took exception to the early warning alert system requirement on the basis that Attachment I clearly indicates that for a LOFC accident (DBA #1) at Fort St. Vrain we do not exceed the Protective Action Guidelines (PAG's) at the exclusion area boundary for at least twelve (12) hours and do not exceed the PAG's at one (1) mile for at least twenty (20) hours. These studies were made utilizing a persistent plume trajectory with Pasquill Category F meteorological conditions, and conservative source term estimates. (See excerpt of our response from P-80083 in Attachment II.)

Following the initial submittal of our emergency response plan, we submitted our revised emergency response plan on August 23, 1980, and again we referenced the technical justification for eliminating the early warning alert system. (See excerpt of response from P-80288 in Attachment III.)

On December 10, 1980, we met with Mr. Brian Grimes along with other members of the emergency preparedness group and NRR staff in an attempt to resolve some of the major issues of our emergency response plan as well as attempt to justify the major differences in HTGR Technology versus LWR technology. We were unable to resolve any of the major issues at the December 10, 1980, meeting, primarily due to the fact that our transmittal letter (P-80288) was apparently misplaced and never received a staff review. From all indications, however, it was apparent that Brian Grime's group was not in a position to provide relief from the early warning alert system.

Recognizing that our initial evaluations (Attachment I) were based on extremely conservative source term estimates, we performed an

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additional evaluation (see Attachment IV). While we removed some of the extreme conservatism of our initial EPZ study we have still maintained a considerable factor of conservatism in our alternate scenario. Based on this alternate scenario, the following conclusion can be summarized:

1. The whole body gamma dose cumulative over a thirty (30) day period for an LOFC accident at Fort St. Vrain never exceeds the PAG for shelter protection and never approaches the PAG for mandatory evacuation.

2. The maximum whole body gamma dose for a 30 day period at the exclusion area is 800 millirem with the conservative meteorological plume trajectory. (See Figure 2 of Attachment IV.)

3. The most restrictive PAG, Thyroid Inhalation, is not exceeded at the exclusion area boundary for fifty (50) hours, (see Figure 3 of Attachment IV), and the PAG for mandatory evacuation is never exceeded.

4. None of the PAG's for the closest population center, Platteville, is exceeded or even approached.

On the basis of the conservative estimates represented by either Attachment I or Attachment IV, more than adequate time exists to warn the general public and institute protective action. Most certainly a 15 minute time limit cannot be justified, in light of the potential doses to the general public.

Based on the characteristics of Fort St. Vrain and the comparison of time for accidents to develop, we cannot see the justification for imposing requirements developed for LWR Technology upon Fort St. Vrain. We have been unable, to date, to obtain any relief based on the technical justification provided, and apparently the emergency preparedness group is not in a position to judge the requirements based on technical merit.

Since we have been unable to obtain any definitive guidance from the review groups, we are appealing to the Commission for consideration of relief from the early warning alert system requirements, and maintain that our position set forth in Attachment III provides adequate protection for the public in the EPZ.

Because of the significant delays we have already experienced, in obtaining some definitive guidance, prompt consideration of this appeal will be greatly appreciated. Should you require any additional information, please contact Mr. Don Warembourg, Manager, Nuclear Production, (303) 571-7436, Extension 200.

Very truly yours,

C. K. Millen

C. K. Millen
Senior Vice President

CKM/dkm

ATTACHMENT I

P-80066 APRIL 1, 1980

FORT SAINT VRAIN EPZ STUDY

Public Service Company of Colorado

16805 ROAD 19½
PLATTEVILLE, COLORADO 80651

April 1, 1980
Fort St. Vrain
Unit No. 1
P-80066

Mr. Brian K. Grimes
Director, Emergency Preparedness Task Group
Office of Nuclear Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Fort St. Vrain Unit No. 1
Emergency Planning

REFERENCES: 1) P-79205
2) P-80011

Dear Mr. Grimes:

In Reference (1), P-79205, we expressed our concern with the extension of the EPZ to a ten (10) mile radius and the ingestion pathway to a fifty (50) mile radius. We did not receive any acknowledgement of this letter, however, on January 8, 1980, we met with Messrs. Roe, Kuzmycz, and Williams in Bethesda to discuss our concerns. As a result of this meeting we agreed to evaluate the EPZ for Fort St. Vrain with reference to the guidelines of NUREG-0396 to provide further justification for smaller EPZ and ingestion pathway radii for Fort St. Vrain.

We have completed our evaluation, and we are transmitting herewith five (5) copies of this evaluation for your review. The attached evaluation is self explanatory and fully supports our original contention that the EPZ for Fort St. Vrain can be considerably smaller than that for a 1000 MWe LWR which was utilized as basis for developing the guidelines of NUREG 0396.

As a brief summary our evaluation indicates that the comparable EPZ for Fort St. Vrain would be about five (5) miles and a comparable ingestion pathway would be about thirty (30) miles. The evaluation is based on conservative estimates of source term releases, worst case meteorological conditions and extremely conservative PCRV (Prestressed Concrete Reactor Vessel) leakage rates.

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On the basis of our evaluation it is requested that the Fort St. Vrain EPZ and ingestion pathway be established as five (5) and thirty (30) miles, respectively.

It should be noted further that we have been through our bases with State officials, and we have complete agreement with the State concerning the inherent safety of Fort St. Vrain versus a LWR. We have also discussed our position with the local FEMA representatives, and again have concurrence that there is no justification for penalizing Fort St. Vrain with water reactor criteria.

It is requested that you give this matter your immediate attention. As you are aware we are in the process of finalizing our Emergency Response Plan with the State as well as the emergency planning for station. Our planning is presently based on the smaller EPZ and ingestion pathway exposure radii and any changes in this criteria could have a significant impact on final acceptance of our plans, both for the station and the State.

Very truly yours,

Don Warembourg

Don Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

DW/alk

EVALUATION OF EMERGENCY PLANNING ZONE DISTANCES
APPLICABLE TO THE FORT ST. VRAIN NUCLEAR GENERATING STATION

1. Introduction

In an NRC letter to all reactor licensees dated November 29, 1979, a request was made for information regarding estimates for evacuation of various areas around nuclear power reactors. NUREG-0396, EPA 520/1-78-016 specified, based upon studies of a 1000 MW(e) LWR, that the emergency planning zone (EPZ) for plume exposure pathway should be about 10 mile radius about the plant and about 50 mile radius for the ingestion exposure pathway. Both design basis accidents with various active engineered safety features, and the accident release categories of the Reactor Safety Study (Wash 1400) were considered in determination of the specified distances. A factor in establishing the about 50 mile radius ingestion exposure pathway is that distance has been evaluated and discussed in detail in Chapter 2 of Facilities Safety Analysis Reports and in Environmental Reports.

The purpose of the evaluation presented herein is to determine the appropriate EPZ distances for Fort St. Vrain which provide the general public equivalent protection that would be afforded by the 10 mile and 50 mile EPZs for a similar sited 1000 MW(e) LWR. At the onset of the investigation it was believed the principle differences would be a smaller source term due to size, 330 MW(e) at 39% efficiency versus 1000 MW(e) LWR, and time to perform emergency actions due to the relatively slow heatup rate provided by the large graphite core of an HTGR. It was confirmed by the detailed study that the FSV smaller source term coupled with unique slow and gradual core heatup characteristic of the plant, did in fact justify sharply reduced EPZ distances for Fort St. Vrain and significantly longer times to perform offsite emergency actions.

2. Conduct of Study

This section describes the methodology, parameter assignments and assumptions used in the conduct of this EPZ study. Section 2.1 describes the accident scenario used as the basis for FSV EPZ determination as well as the description of what was assumed to be a comparable accident scenario for the LWR. Section 2.2 describes the conservatisms applied to the accident scenario. Section 3 describes the conservative meteorological conditions assumed to prevail during the duration of the EPZ determining event. Dose computation methodology and parameter assignments and assumptions appear in Sections 4 and 5 respectively. The appropriate EPZ distances for FSV are presented in Section 6. Conclusions of this study are presented in Section 7.

2.1 Accident Scenario Used to Establish the EPZ for FSV

The FSV Design Basis Accident #1 (DBA #1) which was the only accident identified and analyzed in the FSAR that results in extensive core damage, was selected as the representative scenario used to establish the plant's EPZ. This event is postulated to be initiated by a "non-mechanistic" permanent loss of forced circulation while operating at full power. The reactor is scrammed by the plant protection system and all attempts to restore forced circulation using the multiple heat sinks, circulators, and motive power for the circulators fail. When it becomes apparent to the plant operators that the loss of forced circulation is permanent, e.g. when restoration of cooling would cause steam generator damage, the primary coolant system would be depressurized to storage in a controlled manner through the helium purification system. The reserve shutdown system would be operated after this initial period to assure an adequate shutdown margin.

Because of the large heat sink provided by the graphite core, considerable time is available to initiate primary coolant depressurization and to restore forced circulation. The FSV FSAR specifies the time available to initiate depressurization to be 5 hours, which was later amended by PSC letter P-77250 dated December 22, 1977 to be 2 hours. The reduction in time was due to the re-evaluation of the capability of the helium purification system to process primary coolant during the planned depressurization and release of the clean primary coolant to the reactor building ventilation stack. Thus, the depressurization of the PCRV is now initiated after 2 hours and completed 7 hours later (or 9 hours from the onset of the accident).

The fuel is slow to heat up due to the large heat sink provided by the core graphite. A peak average active core temperature of 5400°F is reached about 80 hours after the onset of the accident. At this temperature, the core structural integrity and geometry are not compromised since the vaporization temperature of graphite is 6900°F. Peak activity released to the primary coolant, considering decay, is reached about 24 hours into the accident.

PCRV containment integrity is maintained through heat removal by the liner cooling system in the "redistribution mode" which maximizes cooling in the top head of the PCRV.

Leakage of primary coolant from the PCRV is assumed to occur at a conservatively high leakage rate of 0.2% of the primary coolant inventory per day.

The reactor building ventilation system maintains continuous venting and processing of the reactor building environment at 1.5 building volumes/hr during the entire period of the accident.

2.2 Conservatisms Applied to DBA #1 Assumptions

2.2.1 Primary Coolant Leakage Rate During DBA #1

The FSV FSAR DBA #1 (appendix D, page D.1-56) assumed an arbitrarily conservative and non-mechanistic estimate of PCRV leakage after the intentional depressurization by assuming that the liner has failed completely (or does not exist) and only concrete permeability controls the leakage. An internal 5 psi pressure differential was assumed which purportedly gave a PCRV leak rate of 8.33×10^{-5} fraction per hr (0.2%/day). Reference was made to Question IX.7 of Amendment No. 9 of the FSV FSAR for the calculation of the permeation rate for the FSV PCRV concrete under these conditions.

Examination of Question D.2 revealed simply the conclusion that a 5 psi positive differential pressure led to 0.2%/day and 2 psi positive differential pressure led to 0.08%/day. Question IX.7 also did not provide details of the calculation of the 0.2%/day rate. However, considerable detail and a derivation was provided for the analysis of leakage rate tests at high pressure. The following equation was provided (eqn. 14 on page IX.7-8):

$$W \text{ (lb/day)} = 1.13 \times 10^{-5} \frac{\Delta P}{\Delta Po} \frac{A}{X} \ln\left(\frac{P_1}{P_2}\right) + 2.2 \times 10^{-6} \frac{\Delta P}{\Delta Po} \frac{A}{X} \left(P_1^{\,2} - P_2^{\,2}\right) \qquad \text{(eqn. 14)}$$

Where ΔP = PCRV inside pressure in psig

ΔPo = PCRV inside pressure in psig for which the net compressive stress in concrete = 0

A = Face area of concrete, ft^2

X = Concrete thickness, ft

P_1 = Permeation or high side pressure, psia

P_2 = Ambient or low side pressure, psia

Numerical values were inserted for P_1 = 845 psig with the assumption that ΔPo was approximately equal to P_1 in the following equation (eqn. 15 on next page):

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10} \ln \frac{857.5}{12.5} + 9.1 \times 10^{-7} \frac{9000}{10} (857.5^2 - 12.5^2)$$

$= 0.43 + 602 = 600$ lb/day (eqn. 15)

The first item to note is that the coefficient for the second (laminar flow) term is in error which is most likely a single error in transcribing from equation 14 to 15 since equation 13 has the 9.1×10^{-7} coefficient. Equation 15 should read:

$$W = 1.13 \times 10^{-5} \times \frac{9000}{10} \ln \frac{857.5}{12.5} + 2.2 \times 10^{-6} \frac{9000}{10} (857.5^2 - 12.5^2)$$

$= 0.043 + 1445 = 1450$ lb/day (eqn. 15 revised)

The second item is that the $\Delta P/\Delta Po$ term has been dropped in going from eqn. 14 to eqn. 15, which is significant if it is assumed that these equations are appropriate for evaluating the leak rate at $P_1 = 5$ psig. Including this factor, we find:

LEAK RATE

Pressure P_1 (psig)	lb/day Eqn 14	lb/day 15	lb/day 15 Revised	%/day 14	%/day 15	%/day 15 Revised	%/day Given
							App D; Amend 9 Question D.2
5	.0019	.13	.30	$\boxed{.001}$	.07	$\boxed{.17}$	$\boxed{.20}$
							Amend 9 Question D.2
2	.0003	.046	.107	.0001	.025	.059	.08

Since equation 14 is the appropriate equation, the 0.2%/day leak rate is conservative by a factor of 200.

For purposes of this evaluation, the historic 0.2%/day is assumed to exist as an upper limit of all potential contaminated primary coolant leakage including permeability through the PCRV concrete. This is judged to be conservative since the primary coolant with any significant activity is contained within the PCRV or helium purification components contained in wells within the PCRV.

For this study, the PCRV leak rate of 0.2%/day is conservatively assumed to apply for the total accident duration period beginning at t = 0 hours. In actuality, this arbitrarily conservative estimate of PCRV leakage would not apply during the period of controlled PCRV depressurization or in the subsequent post depressurized period prior to an assumed liner failure. However, for conservatism this PCRV leak rate will be applied uniformly for all time.

2.2.2 Radionuclide Source Terms for DBA-1:

As previously stated, the fuel within the graphite core is slow to heatup during DBA#1. After about 7.5 hours, some portions of the slowly heating core will have reached the FSAR fuel particle coating failure temperature of 1725°C (3137°F). Once reaching the FSAR fuel particle coating failure temperature, the fission products are assumed, for purposes of this evaluation, to be released in a time dependent fashion normalized per the TID-14844 core release assumptions. For release to the primary coolant within the PCRV, the TID-14844 conditions assume a release of 100% of noble gases, 50% of the iodines and 1% others.

Consistent with TID-14844 release assumptions, 50% of the iodines plateout within the primary coolant system resulting in a depletion of the iodine to 25% of core inventory in the resultant PCRV leakage to the reactor building atmosphere.

These TID-14844 releases and plateout assumptions are very conservative for an HTGR but never the less have been utilized in this evaluation. This conservatism in release and plateout assumptions is particularly evident when the halogen release fractions estimated in the FSV FSAR (Appendix D) are compared to TID-14844 values. The FSAR estimates a core halogen available leakage fraction of 5.5% which is a factor of 4.5 below the TID-14844 effective core release fraction of 25%. The FSAR fuel particle coating failure temperature of 1725°C is also conservative compared to test data simulating a core heatup accident. Figure 2-1 illustrates the margin from the test data to the 1725°C failure temperature used in this evaluation. Also shown is a NRC model (NUREG-0111) for TRISO UC_2 particle which is similar to the FSV FSAR model.

3. Meteorology

3.1 Site Specific Meteorology

The Fort St. Vrain site meteorology was described in detail in the FSV FSAR Section 2.8. The most likely condition on an annual basis (Table XVII of FSAR) are winds from the N sector of the wind rose which occur 17% of the time. The most likely stability category is D (Table XXIV of FSAR), and the average wind speed at 200 ft. elevation averaged over all directions is 6.51 mph (2.9 m/s) as given in Table II of the FSAR. Figure 17 in Section 2.8.3 of the FSAR gives the probability that the dilution factor will exceed a given value. Using the dashed curve shown in Fig. 17 (FSAR Sec. 2.8.3) which includes the effect of using corrected anemometer wind velocity indicates that a dilution factor of about 3.5×10^{-3} s/m^3 at 590m will be exceeded only 5% of the time (the 5% cut-off). This corrected cut-off corresponds to stability category F with a wind speed of about 2.4 m/s. Thus the dilution factor for stability category F with 1 m/s wind speed will be exceeded considerably less than 5% of the time and represents a "worst case" type condition.

3.2 NRC Staff Meteorology from SER

The Fort St. Vrain Safety Evaluation Report (SER) (Ref. 4) prepared by the then AEC staff includes the following meteorology and breathing rate assumptions for accident condition calculations at the LPZ (16,000m)

Time Period	Dilution Factor (s/m^3)	Breathing Rate ($\frac{m^3}{s}$)
Most Unfavorable 8 hrs.	7.8×10^{-6}	3.47×10^{-4}
Next Most Unfavorable 16 hours	3.9×10^{-6}	1.75×10^{-4}
1 to 4 days	2.4×10^{-6}	2.32×10^{-4}
4 to 30 days	7.8×10^{-7}	2.32×10^{-4}
30 to 180 days	4.6×10^{-7}	2.32×10^{-4}

Plotting these dilution factor values against the time period over which they represent an average value (Fig. 3-1), it appears that there is approximately a $t^{-½}$ time dependence, as would be expected, when allowing for wind meander and change of direction. This can be also be seen from the equations for averaging the dilution factor for wind meander over a 22.5° sector ($\phi = \frac{2.032}{\sigma_z x \bar{u}}$ for ground level release) and for an annual average, (Eqn. 6, Sec. 2.8.3 of FSAR) both of which depend inversely on σ_z (the vertical diffusion parameter). Since at long times the diffusion process is dominated by random walk effects, σ_z^2 is expected to be proportional to t, and so ϕ would be proportional to $t^{-½}$.

3.3 Extension of NRC Staff Meteorology to Other Distances

To derive the time dependence of the dilution factor at other distances, the time dependence of the NRC staff meteorolcgy was estimated from a curve drawn through the 16,000m points from the SER as shown in Fig. 3-1. This gave a $t^{-.493}$ dependence. The reference point used for the other distances was the 8 hr. value, which was calculated from $\phi(8hr) = \frac{0.5}{\Pi \sigma_y \sigma_z \bar{u}}$ with $\bar{u}$ = 1 m/s and $\sigma_y \sigma_z$ for all distances taken for stability category F. This is equivalent to

taking the short term dilution factor for ground level release,

$$\phi = \frac{1}{\Pi \sigma_y \sigma_z \bar{u}}$$

as the 2 hour value, and then extrapolating to 8 hours by the factor $\sqrt{\frac{2}{8}} = 0.5$.
Table 3-1 gives the values at other times obtained from $\phi(t) = \phi(8) \left(\frac{8}{t}\right)^{0.493}$ where t is in hours. The breathing rates used were taken from the SER and are given above in 3.2.

3.4 Conservative Meteorology Assumptions

Listed below are the conservatisms used in obtaining the dilution factor in Table 3-1.

a) Worst case type meteorological conditions (Category F, $\bar{u}$ = 1 m/s) used are likely to occur much less than 5% of the time.
b) Downwash conditions are assumed to persist over the entire time.
c) The effective stack height is taken to be zero (i.e., ground level release).
d) Bouyancy effects of He relative to air are neglected.
e) Dilution by the building wake is neglected.

4. Dose Computation Methodology

Plume exposure doses were calculated using the computer program TDAC (Ref. 1). TDAC is a computer program which utilizes analytical methods to calculate the time-dependent radiological impact due to routine or accidental release of radionuclides from nuclear power plants or other potential sources of radioactivity release.

The basic model used by TDAC is comprised of eight volumes which are interrelated by a network of flow paths. The program calculates the amount of activity in each volume at specified times and determines radiological doses are specified distances based on activity entering volume 8 and specified meteorology. The calculated doses are external whole body gamma, external whole body beta, and the internal inhalation commitment doses to the skeletal system, thyroid, lung, gastro-intestinal tract and total body. The required input data consists of

three types: (1) flow model data such as flow rates and filter efficiencies, (2) nuclear parameters such as decay constants, br ching ratios, and disintegration energies, and (3) radiological dose input such as windspeed, atmospheric dilution factors, distances, and br athing rates.

The calculation of activity in each volume is based on the assumption of instantaneous homogeneous mixing. The calculation of radiological doses is based on the semi-infinite cloud approximation. All calculations are based on the analytical solution of the coupled linear differential equations governing the activity in the different volumes.

The TDAC code, with appropriate parameter assignments was also used to evaluate the plume exposure doses for the Reference LWR of this study.

For this study, it was observed that the dose categories of whole body gamma external plume exposure and thyroid inhalation and ingestion dose control and for this reason other organ dose categories are omitted from this presentation.

5. <u>Parameter Assignments and Assumptions</u>

Table 5-1 provides a listing and summary of the parameter values and analysis assumptions of this study. The characteristics of the FSV siting event, the DBA #1, and the reference LWR event, DBA/LOCA are described. These events are felt to bound the envelope of emergency preparedness planning.

6. Applicable EPZ Distances for Fort St. Vrain

6.1 Evaluation of Evacuation EPZ for FSV

The EPZ (Emergency Planning Zone) determination for Fort St. Vrain is based on the estimated accident doses in the environment and their comparison to the recommended ~~Protective Action Guides~~ (PAGs) contained in Ref. 3, Table 5.2. These PAGs for the general population consist of dose ranges in the event of an emergency condition, along with recommended actions to be taken by the responsible local authorities.

Figures 6-1 and 6-2 show the comparison between FSV and a reference 1000 MW(e) LWR of the cumulative dose vs. distance. Also shown on Figure 6-1 are the 1 and 5 Rem whole body gamma PAG dose levels. Below 1 Rem, no protective action is required according to the PAG, while the state may issue an advisory, and environmental radiation levels are monitored. For doses of 1 to less than 5 Rem, the general population should seek shelter and await further instructions, evacuation of particularly susceptible elements of the general population should be considered, access should be controlled, and environmental radiation levels monitored. For 5 Rem or higher whole body gamma doses, mandatory evacuation of the population in predetermined areas is recommended. Figure 6-2 is similar to Figure 6-1 except that it is for inhalation thyroid dose and the PAG levels are 5 and 25 Rem. Clearly, thyroid dose is the controlling factor since the distance to which protective action is required is larger. From Fig. 6-2, the 0 to 30 day inhalation thyroid dose for FSV of 25 Rem occurs at about 4100m (2.5 mi.). The above data is also summarized in Table 6-1.

For a 10 mi. LWR evacuation radius (16,000m), the dose for the reference LWR is about 11 Rem inhalation thyroid. The distance for FSV for the same dose as the referenc LWR at 10 mi. is about 7500m (4.7 mi.). Therefore an evacuation radius of about 5 mi. for FSV would afford a comparable degree of public protection at

the 10 mi. LWR distance.

There is also a considerable difference in the time of the activity released from FSV compared to the reference LWR as shown in Fig. 6-3 for the inhalation thyroid. This figure shows that for the LWR high thyroid doses (100 rem) have already been received at 1 mi. in the first hour, while 12 hours should be available even at 590m for FSV, before the lower PAG (5 rem thyroid) is reached allowing an orderly and organized evacuation to be done if necessary.

6.2 Ingestion Pathway EPZ Evaluation

In addition to direct plume exposure, the population surrounding the nuclear power plant can be subjected to pathway doses from contamination of the surrounding crops, milk, and food stuffs subsequent to the atmospheric release of radioactivity in the accident. The protective action guides for exposure from food stuffs or water (Ref. 3) define the guidance to be exercised by the utility in its assessment and control of the post accident ingestion pathways.

The most severely restrictive ingestion pathway for FSV and the reference LWR accident examined was found to be the iodine-grass-cow-milk-infant thyroid pathway. The predicted thyroid inhalation dose vs. distance for both the reference LWR and FSV is displayed in Fig. 6-2. The iodine-grass-cow-milk-infant pathway may be related to I-131 plume exposure dose by using an approximate factor of 300 (Ref. 2) and recognizing the fact that the thyroid dose reported in Fig. 6-2 is nearly all due to I-131 exposure.*

The plume thyroid inhalation dose, which is proportional to ingestion pathway dose, for the reference LWR is 0.35 Rem.**The distance at which this dose level is reached for FSV is about 30 miles and is consistent with FSAR Chpater 2 discussions of demographic studies at distances up to 30 miles from the site.

*I-131, at 50 miles, contributes 89% of the reference LWR plume inhalation dose commitment and correspondingly 97% for FSV.

** Ref. 2, Table I-2 extended to 50 mile radius.

7. Conclusions

It is concluded from this EPZ study that a significantly smaller evacuation radius can be justified for the Fort St. Vrain plant than for a reference 1000 MW(e) LWR. A further conclusion from this study is that the unique characteristics of the FSV plant provide a slow gradual heatup of the core allowing considerable time (on the order of tens of hours) for orderly notification, monitoring and evacuation to proceed. Also, a smaller radius for the controlling ingestion pathway, the grass-cow-milk-infant thyroid, can be derived which offers the public the same protection as a similarly sited LWR. In both cases, thyroid dose is the limiting constraint. These conclusions are based on siting event analyses using generally conservative core release fractions and meteorology. Table 7-1 summarizes the applicable emergency planning zones for the plume exposure pathway and the ingestion exposure pathway for Fort St. Vrain providing equivalent protection to the general public.

References

1. "TDAC - An Analytical Computer Program to Calculate the Time Dependent Radiological Effects of Radionuclide Release", D. W. Buckley, General Atomic Co. Report GA-D13476, 1976.

2. NUREG-0396, " Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Plants".

3. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", Environmental Protection Agency Report, EPA-520/1-75-001, 1975.

4. FSV Safety Evaluation Report, Division of Reactor Licensing, USAEC, Jan. 20, 1972.

TABLE 3-1

ATMOSPHERIC DILUTION FACTORS[a] FOR FSV DBA #1 AND REFERENCE LWR DBA/LOCA BASED ON NRC STAFF VALUES USED IN THE FSV SER

	DISTANCE (M)									
TIME (HR)	590	1600	4800	8000	16,000	24,140	32,200	48,300	64,400	80,500
8	7.2-4	1.43-4	2.76-5	1.65-5	7.89-6	5.09-6	3.77-6	2.5-6	1.9-6	1.5-6
16	5.1-4	1.02-4	1.96-5	1.17-5	5.61-6	3.62-6	2.68-6	1.8-6	1.3-6	1.0-6
24	4.1-4	8.35-5	1.61-5	9.60-6	4.59-6	2.96-6	2.19-6	1.4-6	1.1-6	8.6-7
34	3.5-4	7.03-5	1.35-5	8.09-6	3.87-6	2.50-6	1.85-6	1.2-6	9.1-7	7.2-7
40	3.2-4	6.49-5	1.25-5	7.46-6	3.57-6	2.30-6	1.70-6	1.1-6	8.4-7	6.7-7
52	2.8-4	5.7-5	1.1-5	6.56-6	3.14-6	2.02-6	1.50-6	9.8-7	7.4-7	5.8-7
58	2.65-4	5.4-5	1.04-5	6.21-6	2.97-6	1.92-6	1.42-6	9.3-7	7.0-7	5.5-7
100	2.05-4	4.13-5	7.95-6	4.75-6	2.27-6	1.47-6	1.08-6	7.1-7	5.4-7	4.2-7
400	1.03-4	2.09-5	4.01-6	2.4-6	1.15-6	7.4-7	5.47-7	3.6-7	2.7-7	2.1-7
720	7.7-5	1.56-5	3.0-6	1.8-6	8.59-7	5.54-7	4.1-7	2.7-7	2.0-7	1.6-7
1500	5.3-5	1.09-5	2.09-6	1.25-6	5.98-7	3.86-7	2.85-7	1.9-7	1.4-7	1.1-7
4320	3.2-5	6.45-6	1.24-6	7.42-7	3.55-7	2.29-7	1.69-7	1.1-7	8.4-8	6.6-8

[a]Values in units of S/m^3 are listed in exponential form such that, e.g., 7.2-4 means 7.2×10^{-4} .

Table 5-1

EPZ Study - Assumptions and Parameter Selections

Parameter	Description
FSV	
Source Term	
Power level, MW(t) design	879 MW(t)
Accident scenario assumed	DBA #1 (FSV FSAR Appendix D, page D.1-56)
Release of fission products from the FSAR fuel particle	Time dependent release* over a 475 hour period
Release fraction from fuel to primary coolant system	TID-14844 fractions employed
Noble gases	100%
Iodines	50%
Others	1%
Plateout reduction factor (Iodine only)	TID-14844 reduction of 50% applied to Iodine transported from the primary coolant system to the reactor containment building.
PCRV leak rate (total duration of accident)	0.2% volume/day
Reactor building ventilation system characteristics	
Purge rate	1.5 volumes/hour (on for total duration of the accident)
Filter efficiencies	
Noble gases	0%
Halogens	90%
Particulates	95%
Additional plateout and gravitational settling of condensible nuclides in the confinement building	Ignored

*Time-dependent source term identified by designation FSVTID*FUELREL. and discussed in engineering file C-70-001, F. S. Dombek.

Table5-1(Cont.)

Parameter	Description
Reference LWR*	
Source term	
Power level, MW(t)	2958.
Accident scenario assumed	DBA/LOCA
Release of fission products from the core matrix to the reactor building	Near instantaneous release of TID-14844 core fractions
Core inventory, available for release from containment	TID-14844 assumptions (including iodine reduction due to plateout)
Noble gas	100%
Iodine	25%

Isotopes	Available for leakage from the reactor containment immediately after LOCA (Ci)
I-131	1.9+07
I-132	2.7+07
I-133	4.2+07
I-134	4.8+07
I-135	3.8+07
Kr-83M	1.3+07
Kr-85M	3.2+07
Kr-85	7.3+05
Kr-87	6.3+07
Kr-88	8.8+07
Kr-89	1.1+08
Xe-131M	6.5+04
Xe-133M	4.0+06
Xe-133	1.7+08
Xe-135M	4.5+07
Xe-135	2.8+07
Xe-137	1.5+08
Xe-138	1.5+08

*While not specifically identified as the Sun Desert Nuclear Power Plant, the reference LWR of this study has many characteristics similar to this current PWR design.

Table 5-1 (Cont.)

Parameter	Description
Iodine composition	
Elemental	91%
Particulate	5%
Organic	4%
Containment building leak rate	
0-24 hours	0.2%/day
>24 hour	0.1%/day
Containment spray characteristics	
Initiation time	150 sec
Sprayed region	70%
Unsprayed region	30%
Mixing rate sprayed to unsprayed region	2 hr^{-1}
Elemental Iodine removal rate	10 hr^{-1}
Particulate Iodine removal rate	0.45 hr^{-1}
Organic Iodine removal rate	0
Iodine DF (elemental)	100
Resultant single volume LWR spray removal constant used this study	3.0 hr^{-1}
Post LOCA containment purge characteristics (to reduce combustible gas concentration)	
Purge interval	336 to 720 hours
Purge rate	50 SCFM
Purge filter efficiency	
Elemental & Methyl Iodine	70%
Radionuclide leakage from ESF area	Ignored

Table 5-1 (Cont.)

Parameter	Description
Specific Site and Dose Parameters (Used for both FSV & Reference LWR dose estimates)	
Atmospheric dilution factors employed	See Table 3-1 and Figure 3-1
Breathing rates	
Reference LWR	
0-8 hour	3.47×10^{-4} m³/sec
8-24 hour	1.75×10^{-4} m³/sec
24-720 hour	2.32×10^{-4} m³/sec
FSV*	
0-40 hour	2.32×10^{-4} m³/sec
40-52 hour	3.47×10^{-4} m³/sec
52-58 hour	1.75×10^{-4} m³/sec
58-720 hour	2.32×10^{-4} m³/sec
Average wind speed	1 m/sec
Dose code utilized	TDAC - GA-D13476
Decay & buildup estimate	Considered
Fallout deposition enroute to dose receptor	Considered for Iodine only (see Reg. Guide 1.111 Figure 6)
Conversion factor for infant-milk thyroid Iodine ingestion pathway (ratio of thyroid dose commitment factor for milk pathway to the inhalation plume exposure pathway)	300 (Ref. NUREG-0396 Appendix I, page I-19)

*Note: FSV breathing rates and atmospheric dilution factors have been permuted to maximize the off-site dose, i.e., the times of highest breathing rate and poorest atmospheric dilution correspond to the peak in time dependent release of fission products from the fuel to the primary coolant.

Table 6-1

Summary of Derived EPZ Evacuation Distances for FSV and a Reference 1000 MW(e) LWR

Whole Body Gamma - Direct Plume Exposure

No Pr[illegible]ive Action (PAG <1 rem whole body)

	EPZ Distance	
	FSV	LWR
0-8 Hour	< EAB (1)	2.2 Mi.
0-24 Hour	< EAB	2.5 Mi.
0-30 Day	0.6 Mi.	3.0 Mi.

Evacuation Required (PAG ≥ 5 rem whole body)

	EPZ Distance	
	FSV	LWR
0-8 Hour	< EAB	0.9 Mi.
0-24 Hour	< EAB	1.0 Mi.
0-30 Day	< EAB	1.2 Mi.

Thyroid Inhalation - Direct Plume Exposure

No Protective Action (PAG <5 rem Thyroid)

	EPZ Distance	
	FSV	LWR
0-8 Hour	< EAB	7.5 Mi.
0-24 Hour	1.2 Mi.	9.0 Mi.
0-30 Day	8.1 Mi.	16.0 Mi.

Evacuation Required (PAG ≥ 25 rem Thyroid)

	EPZ Distance	
	FSV	LWR
0-8 Hour	<EAB	3.1 Mi.
0-24 Hour	0.5 Mi.	3.5 Mi.
0-30 Day	2.5 Mi.	5.3 Mi.

(1) FSV EAB (Exclusion Area Boundary Distance) 590 meters.

Table 7-1

FSV EPZ Distance Compared to Generic LWR EPZ Distance of NUREG-0396 (1)

Accident Phase	Generic LWR Emergency Planning Zone Distance NUREG-0396	Comparable FSV Emergency Planning Zone Distance (2)
Plume Exposure Pathway	About 10 miles radius	About 5 miles (3)
Whole body (external) Thyroid (inhalation) Other organs (inhalation)		
Ingestion Pathway	About 50 mile radius	About 30 miles (4)
Thyroid, whole body, bone marrow (ingestion)		

(1) NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Power Plants", Joint Report USEPA and USNRC. Collins, H. E. et. al., Dec. 1978.

(2) FSV EPZ distances were obtained by (1) assuming that the Reference LWR of this study meets the intent of NUREG-0396 for design basis accidents, (2) estimating the Reference LWR dose at 10 miles (plume) and 50 miles (ingestion) for the DBA/LOCA of this study and then, (3) determining the corresponding equi-dose distance for the FSV-DBA #1 event.

(3) Thyroid inhalation controls with about a 5 mile EPZ. EPZ distance for whole body is approximately 2 miles and other organ doses are negligible.

(4) The grass-cow-milk-infant thyroid exposure pathway dominates for the FSV environmentally released radionuclide source term.

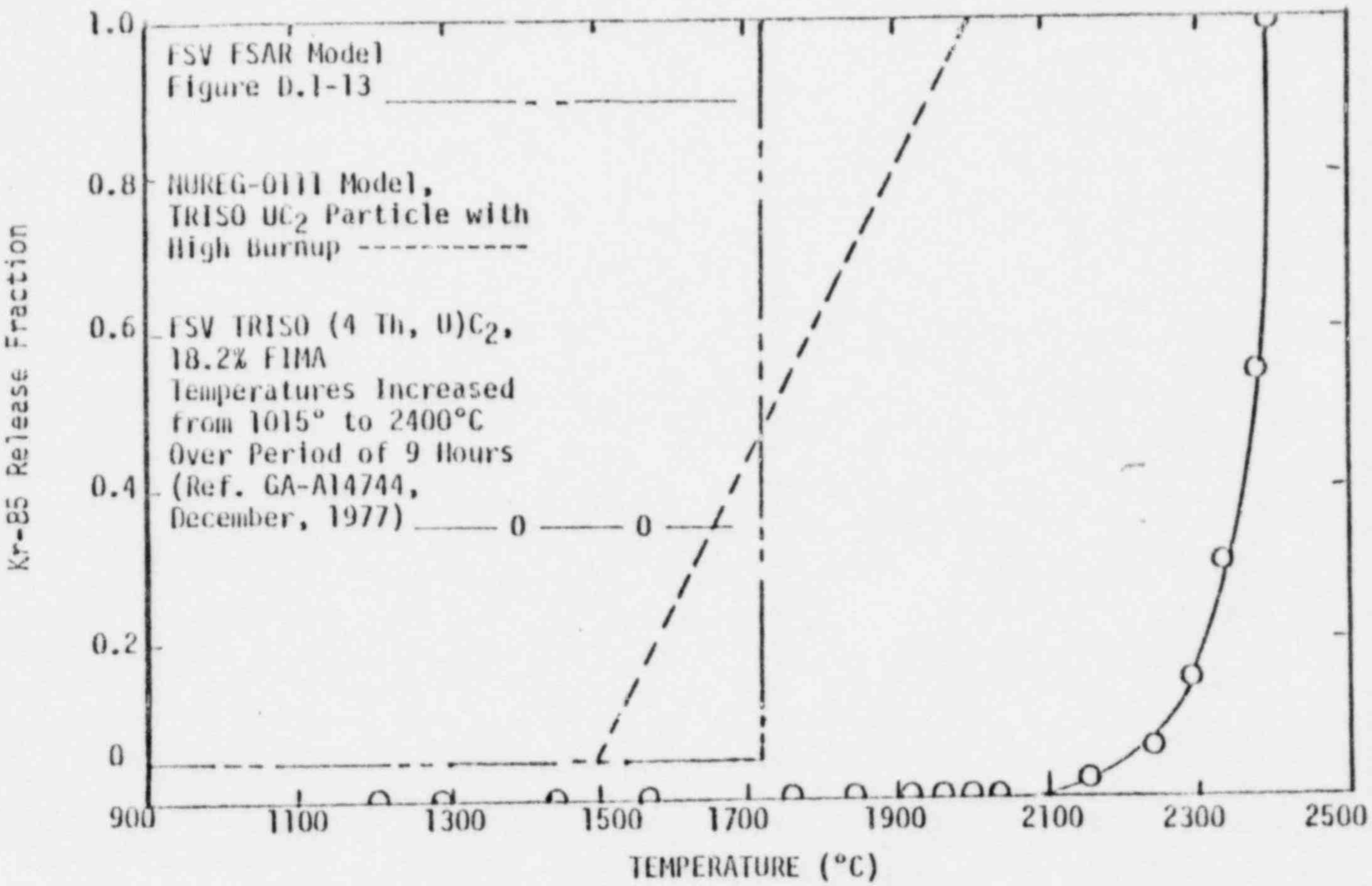


Figure 2-1 Comparison of Models for Fuel Failure as a Function of Temperature versus Observed Release Fraction on Fissile A FSV Production Fuel

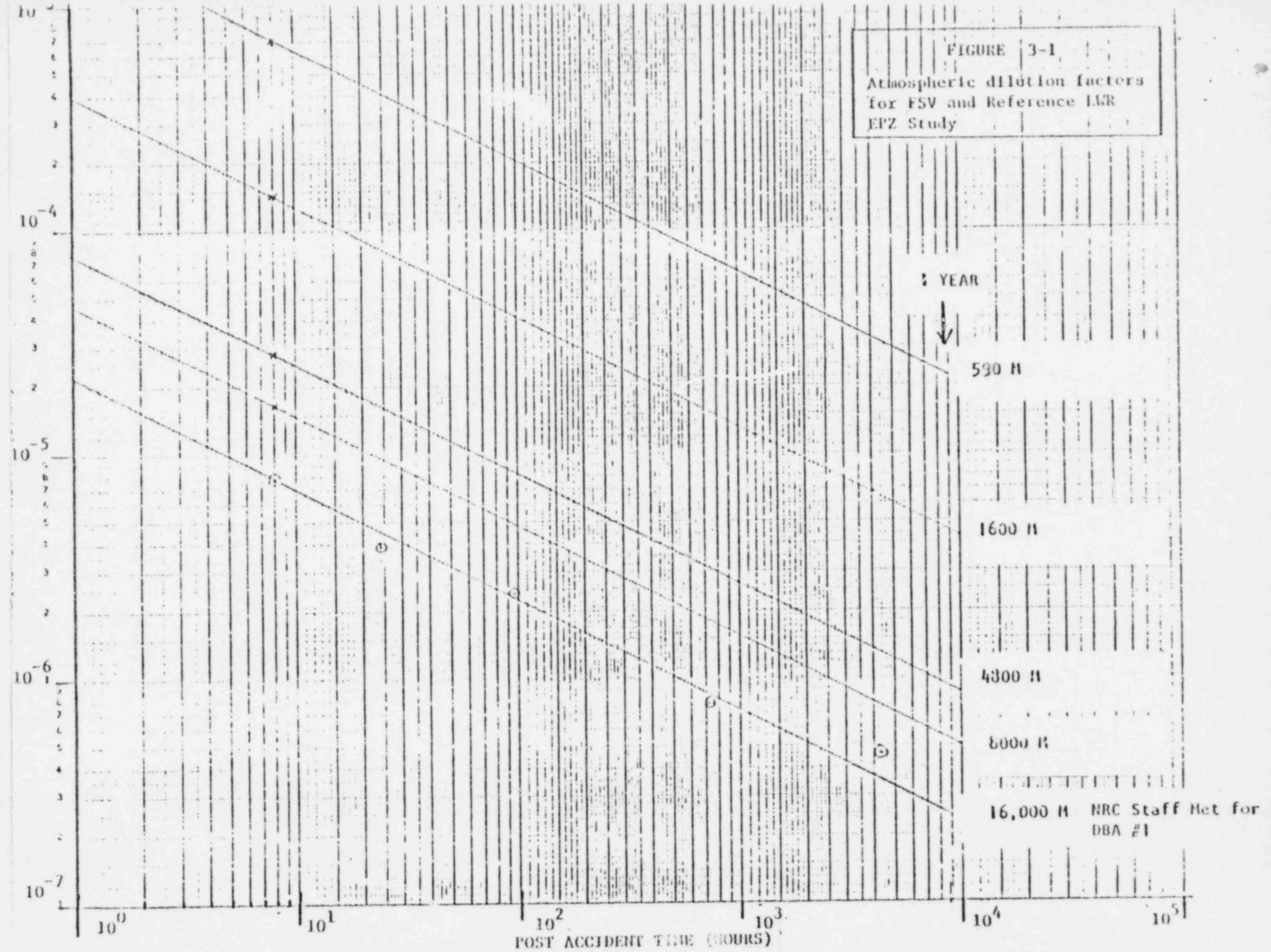


FIGURE 3-1
Atmospheric dilution facters for FSV and Reference LWR EPZ Study

Figure 6-1

Cumulative plume exposure, whole body gamma dose vs. distance. Integration intervals of 8 hr, 1 day, and 30 days. Comparison of FSV-DBA-1 Event and 1000 MW(e) LWR-DBA/LOCA siting event. NRC meteorology (from SER) for both.

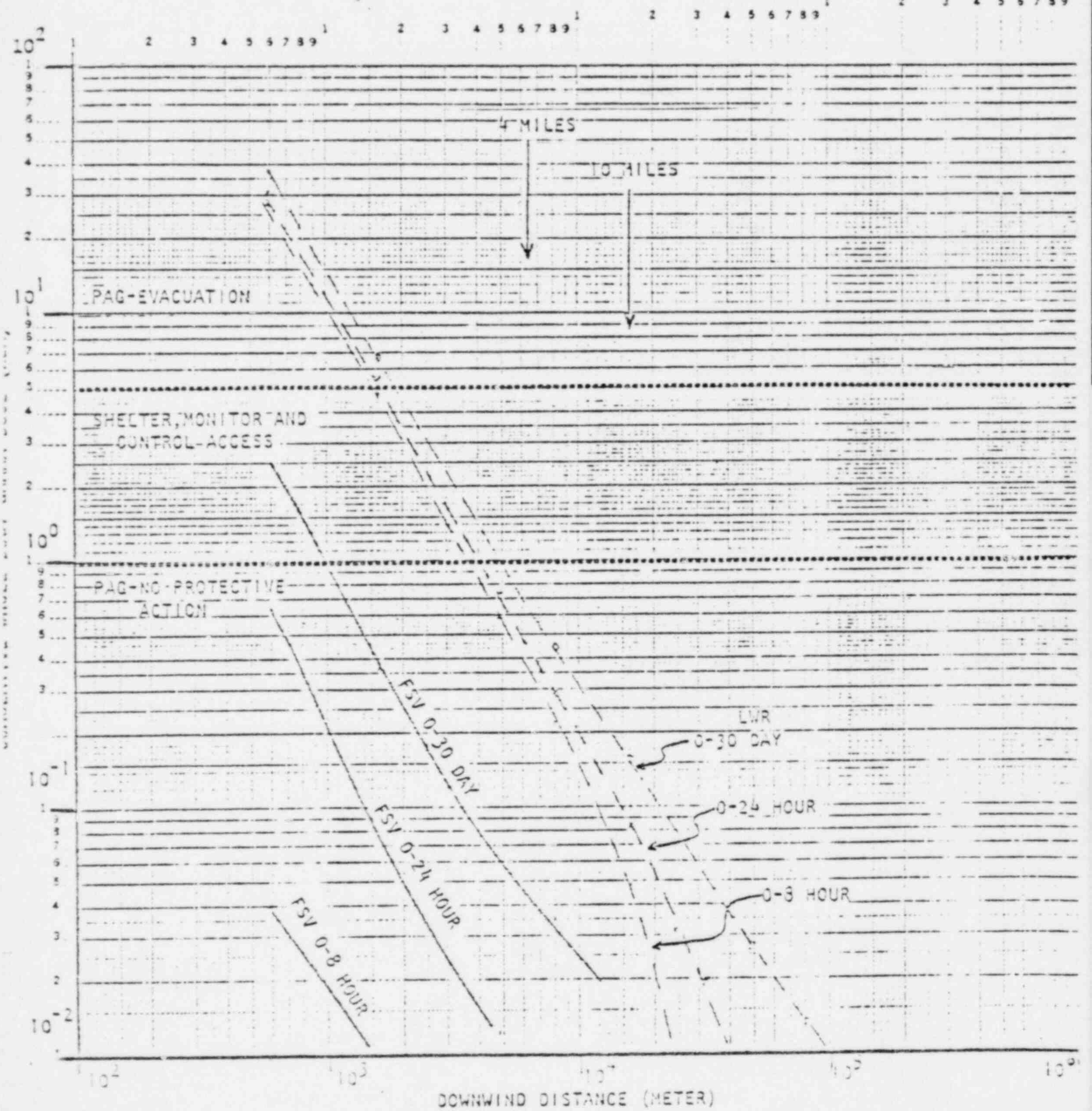


FIGURE 6-2

Cumulative plume exposure inhalation thyroid dose vs. downwind distance. Integration intervals of 8 hr, 1 day, and 30 days. Comparison of FSV-DBA 1 Event and 1000 MW(e) LWR DBA/LOCA siting Event. NRC meteorology (from SER) for both.

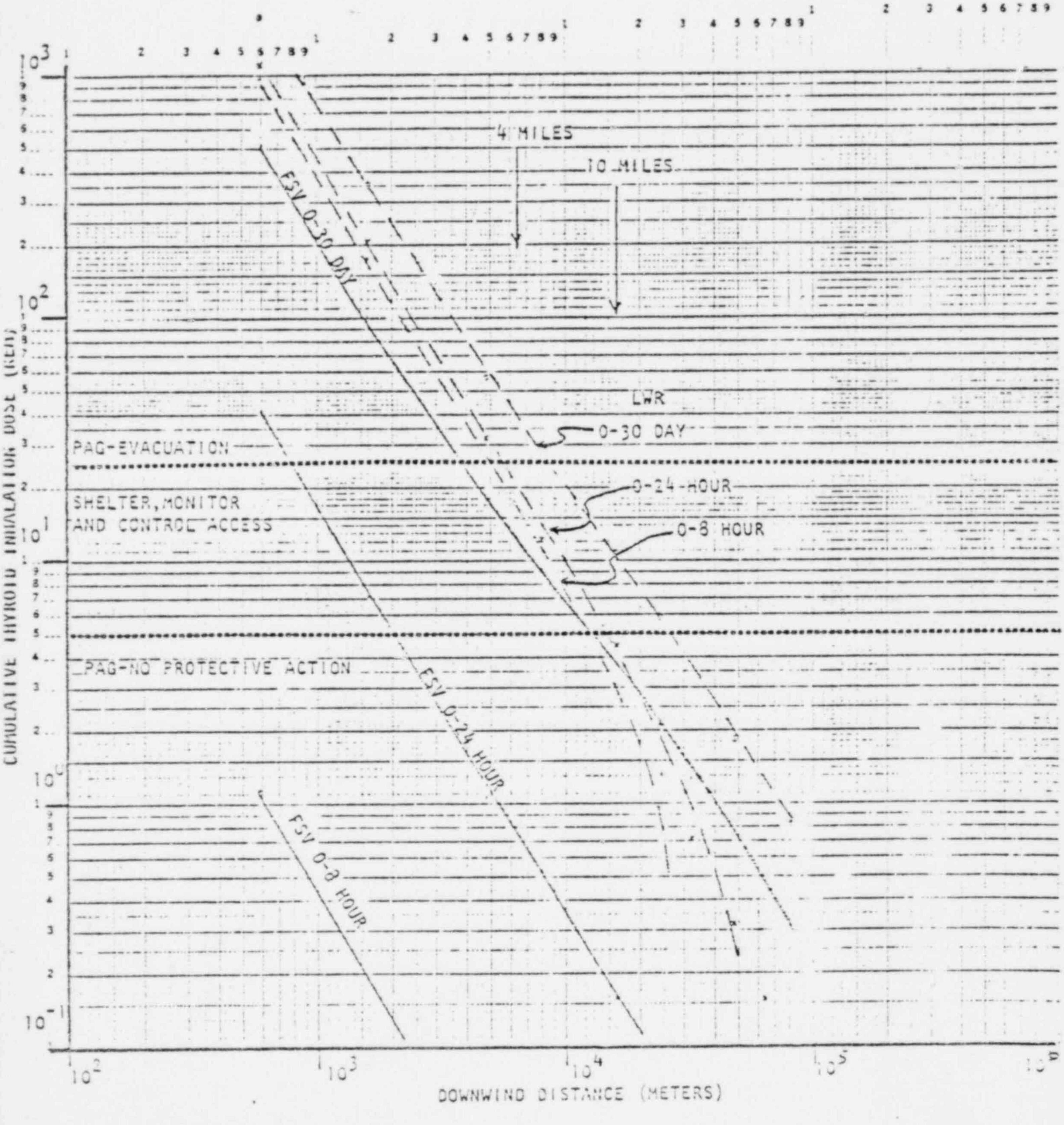
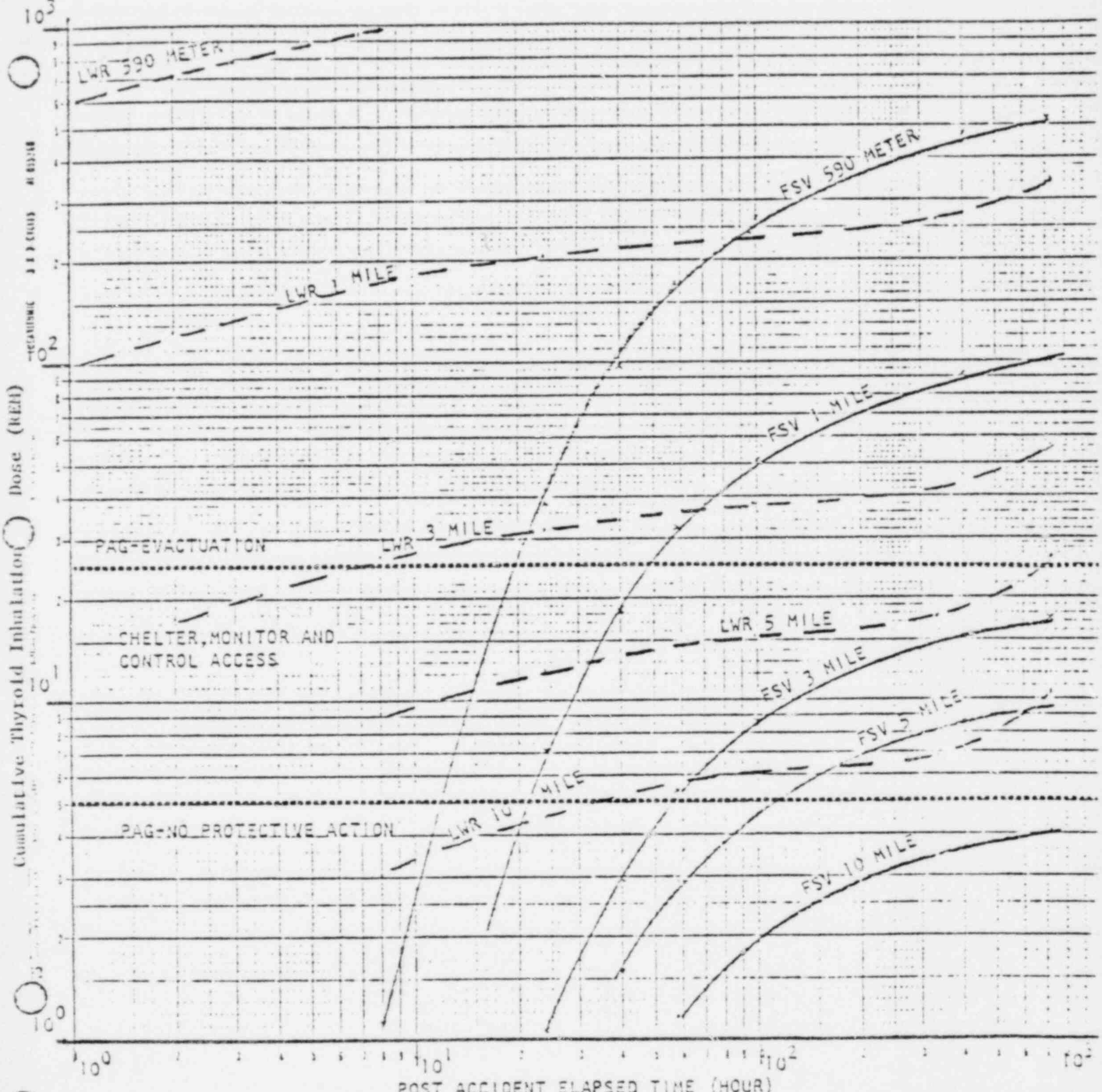


FIGURE 6-3

Cumulative plume exposure inhalation thyroid dose vs. time. Comparison of FSV-DBA 1 Event and 1000 MW(e) LWR-DBA/LOCA Siting Event. NRC Meteorology (from SER) for both.



ATTACHMENT II

EXCERPT P-80083, MARCH 18, 1980

FIRST TRANSMITTAL
FORT SAINT VRAIN EMERGENCY RESPONSE PLAN

Public Service Company of Colorado
16805 ROAD 19½
PLATTEVILLE, COLORADO 80651

March 18, 1980
Fort St. Vrain
Unit No. 1
P-80083

Mr. Brian K. Grimes
Director Emergency Preparedness Task Group
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Fort St. Vrain Unit No. 1
Radiological Emergency
Response Plan

Dear Mr. Grimes:

We are transmitting herewith two (2) copies of Radiological Emergency Response Plan in draft form. There are some detailed areas of the plan which have not been completed as yet, but the basic plan should be complete enough for you to begin your review.

The plan has been developed on the basis of the acceptance criteria published as a part of NUREG 0654, and generally meets the intent of NUREG 0654 with the following exceptions.

1. Near Site Emergency Operations Center

Our near site emergency operations center (Forward Command Post) has been designated as the Fort Lupton Municipal Building which is located approximately ten (10) miles south-southeast of the plant. As indicated in our letter P-79205 our Forward Command Post location was developed with the State/Local agencies and has been equipped with necessary communication systems, supplies, etc. This facility was utilized during our combined drill with the State on February 28, 1980, and proved to be more than satisfactory in all respects in accomplishing the coordination and direction of emergency activities.

We can see absolutely no justification for establishing an emergency operations center within one (1) mile of the site as required by NUREG 0654. In fact such a center within the EPZ presents problems of access, personnel control, possibility of evacuation and other problems which are not experienced at the Fort Lupton location. We clearly demonstrated during the

DUPE 8004220147 (4pp.)

1. Near Site Emergency Operations Center (continued)

February 28, 1980, drill that an emergency can be handled at the Fort Lupton facility with a minimum of problems, and in our opinion with far less problems than emergency center within one (1) mile of the site. The services available in Fort Lupton would not be available at a near site center, and we feel moving the center from Fort Lupton would only serve as a detriment to our overall plan.

Based on the results of our drill and the lack of any justification for being closer to the site we feel that adequacy of our Forward Command Post location has been justified.

2. Plant Staffing

NUREG 0654 set forth certain minimal requirements for plant staffing in Table B-1, which in effect requires ten (10) people on shift and the capability of having an additional twenty-six (26) people on site within thirty (30) minutes. As indicated in our previous responses to the TMI-2 Lessons Learned requirements and as confirmed by Nuclear Regulatory Commission acceptance of these requirements (undated letter from T. Speis to J. Fuller, March, 1980), emergency situations at Fort St. Vrain develop very slowly as compared to a water reactor. As a result, operator and emergency response times are considerably longer. Our emergency plan has been developed on this basis, and therefore, the requirements of Table B-1, NUREG 0654, are considered inapplicable to Fort St. Vrain.

3. Early Warning Alert System

As indicated in our recently transmitted study (see P-80066) concerning the EPZ for Fort St. Vrain we do not exceed any of the PAG's for cumulative whole body gamma dose for the first twenty-four (24) hours following a LOFC accident, and never reach the PAG's requiring evacuation. For thyroid inhalation exposures we do not exceed the PAG for "no protective action required" for up to twelve hours at the exclusion area boundary (22 hours at one mile) and we do not exceed the "shelter, monitor and control access" PAG for up to twenty (20) hours at the exclusion area boundary (50 hours at one mile). Based on these times more than adequate time exists for notification of the public and for initiating evacuation of the public without the need for an early warning alert system. It should be noted that these exposures are based on extremely conservative releases of Iodine and extremely conservative reactor vessel leak rates. More realistic release rates of Iodine would be a factor of ten (10) below those assumed for the EPZ study.

4. Meteorological Systems

Our present plan does not meet the requirements of NUREG 0654 concerning backup met[illegible]ological instrumentation nor the requirement for remote i[illegible]terrogation capabilities. We are presently investigating other alternatives.

5. Iso-Dose Curves

The Iso-Dose curves have not been included in the draft plan as we have been unable to determine from the Nuclear Regulatory Commission just what is required in this area.

6. Letters of Agreement

Letters of agreement with various off site support organizations have not been included at this time as we are still in the process of developing these agreements. In general, however, our State plan is designed as a general agreement for all participating agencies, and those agencies that are signatory to the State Plan will not be called upon for separate letters of agreement.

7. NUREG 0610

The guidance provided by NUREG 0610 has been modified to best fit HTGR technology versus water reactor technology. The requirements for "Unusual Event Notification" are included in an agreement between Public Service Company and the State which was originated prior to NUREG 0654. Since this agreement meets the intent of NUREG 0610 and has the concurrence of both Public Service Company and the State we feel it is adequate to fulfill the notification requirements for unusual events.

Other than minor differences we believe the plan meets the intent of NUREG 0654 acceptance criteria.

If you should have any problems or questions we would appreciate hearing from you prior to the May 21, 1980, meeting date.

Very truly yours,

Don Warembourg

Don W. Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

DWW/alk

cc: Dave Przbindowski
OSB Pacific Northwest Lab
Batelle Boulevard
Richland, Washington 99352

Jim Miller
G. Kuzmycz
G. McPeak

ATTACHMENT III

EXCERPT P-80288 AUGUST 28, 1980

SUBMITTAL OF REVISED EMERGENCY
RESPONSE PLAN

Public Service Company of Colorado

August 28, 1980
Fort St. Vrain
Unit No. 1
P-80268

Mr. Robert L. Tedesco, Assistant Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Fort St. Vrain Unit No. 1
Emergency Response Plan

REFERENCE: NRC Letter July 23, 1980

Dear Mr. Tedesco:

We are transmitting herewith three (3) copies of our revised emergency response plan. This revised plan includes changes as a result of the May 21, 1980, plant site review meeting as well as certain changes that resulted from comments contained in your July 23, 1980, letter. In addition to the revised plan, we are providing our response as Attachment A to this letter to address your July 23, 1980, letter.

As we indicated in the May 21, 1980, meeting as well as in various correspondence submitted as a part of the TMI-2 Lessons Learned Tasks, we believe Fort St. Vrain is a completely different reactor concept. This reactor concept coupled with the size of the reactor negates many of the requirements setforth by NUREG's 0654 and 0610 which were developed primarily on the basis of 1,000 MW(e) light water reactor technology. It is imperative, therefore, that our Emergency Plan be evaluated on the basis of our reactor design and size, and that generic requirements be evaluated on the basis of specific technical, safety, and environmental differences.

We have had to essentially develop our own criteria for Fort St. Vrain utilizing water reactor criteria setforth by various Nuclear Regulatory Commission documents. On this basis our criteria is necessarily different from that published and we have taken justifiable exception to the NUREG's. These exceptions were supported in various correspondence (see reference list attached) and are further supported by Attachment A of this letter.

DUPE 8101130200

In the May 21, 1980, meeting we were informed by the Nuclear Regulatory Commission review team that many of the items were a matter of policy, and that the review team did not have the authority to make exceptions on policy matters regardless of the technical justification. We cannot accept this position, and we request that as soon as you have had the opportunity to review our revised emergency plan and our response that we be given the opportunity to meet with you and other personnel who do have the authority to evaluate and/or accept our positions on the basis of the technical justification provided.

In the interest of time it is requested that such a meeting be established at the earliest possible date so that we may finalize our plans to meet the various commitment dates setforth. We will be available to meet with you at your convenience and are looking forward to hearing from you shortly.

Very truly yours,

Don W Warembourg

Don W. Warembourg
Manager, Nuclear Production
Fort St. Vrain Nuclear
Generating Station

DWW/alk

Attachments

PSC CORRESPONDENCE LISTING
LETTERS TO NRC INVOLVING
TMI-2/EMERGENCY PLANNING/EMERGENCY RESPONSE PLANS

Correspondence Number		Date	Subject
1.	P-79130	June 15, 1979	Gaseous Effluent Monitors
2.	P-79205	September 10, 1979	Emergency Planning, Fort St. Vrain
3.	P-79239	October 17, 1979	Followup Action TMI-2
4.	P-79249	October 29, 1979	Followup Action Resulting From NRC Reviews Regarding the TMI-2 Accident
5.	P-79290	November 30, 1979	NUREG-0610
6.	P-79298	December 12, 1979	Fort St. Vrain, Unit No. 1, TMI Lessons Learned
7.	P-79299	December 12, 1979	Revised Followup Actions Resulting From NRC Reviews Regarding TMI-2 Accident
8.	P-79305	December 18, 1979	Supplementary Response, Item 2.2.1.b, Lessons Learned Task Force, TMI-2
9.	P-79312	December 28, 1979	Additional Information Regarding June 1, 1980, Action Items Resulting from TMI-2
10.	P-80011	January 29, 1980	Request for Evacuation Times
11.	P-80028	February 20, 1980	Additional Information Resulting from TMI-2 NRC Review Team Site Visit, January 21-22, 1980
12.	P-80041	March 5, 1980	Request for Evacuation Times
13.	P-80083	March 18, 1980	Fort St. Vrain, Unit No. 1, Radiological Emergency Response Plan
14.	P-80066	April 1, 1980	Fort St. Vrain, Unit No. 1, Emergency Planning

ATTACHMENT A

PSC RESPONSE TO NRC COMMENTS
FSV EMERGENCY RESPONSE PLAN

1. NRC Question/Comment

Plan must be revised to establish a principal and an alternate EOF. Both facilities should meet the requirements of Darrell G. Eisenhut's letter of April 25, 1980, subject "Clarification of NRC Requirements for Emergency Response Facilities at Each Site."

PSC Response

We cannot address the principle and alternate EOF as we have never received the April 25, 1980, letter which you reference. It is our understanding that new criteria will be published as a part of NUREG-0696. Upon receipt and evaluation of this document we will modify our emergency plan. In the interim we intend to continue with our plans to utilize the Fort Lupton Municipal Building for the EOF as stated in our letter P-80083. As we understand the new criteria being developed under NUREG-0696, a distance of approximately 10 miles from the reactor would be acceptable for the EOF. Depending on the criteria specified for the EOF and an alternate EOF we will re-evaluate our position at the time NUREG-0696 is published.

2. NRC Question/Comment

Plan must be revised to take into consideration the plant staffing in Table B-1 of NUREG-0654. There must be some augmentation of on-site personnel within 30 minutes. Must identify position that will not be filled and provide rationale for not having 10 personnel on shift at all times.

PSC Response

Consideration was given to the plant staffing in our April draft of the RERP. Figures 5.1-1 through 5.2-6 of the RERP depict both the normal and the emergency staffing for the plant. Figure 5.1-2 provides the normal operating staff (9 personnel plus a Lead Security Officer) for the plant and fulfills the on-shift requirements of Table B-1, NUREG-0654, with the exception that we do not have a Rad/Chem Technician on shift. The on-shift Health Physics Technician has sufficient training to perform the necessary initial surveys and radiological assessments to protect in-plant personnel. The operating staff has sufficient training and procedures to evaluate the off-site effects. We can see no immediate requirement for the Rad/Chem Technician especially since our accidents develop at a much slower rate than comparable water reactor accidents (see NRC letter Themis Speis to J. Fuller, March, 1980, Acceptance of Category A TMI-2 Requirements).

With reference to the augmented staff called for in Table B-1, NUREG-0654, we have justified delaying the response time of the Shift Technical Advisor (STA) (based again on the rate in which our accidents develop) from 10 minutes to one (1) hour (see PSC letters P-79249, October 29, 1979; P-79299, December 12, 1979; P-79305, December 18, 1979; P-79312, December 28, 1979). The accident time frames and the associated response times were accepted by the Nuclear Regulatory Commission by the above referenced letter (Themis Speis to J. Fuller, March, 1980) in the overall acceptance of the Category A TMI-2 requirements. Since the Nuclear Regulatory Commission found the response time of the STA to be acceptable we maintain that the 30 minute augmented staff time called for in NUREG-0654, Table B-1, is not applicable to Fort St. Vrain. On the basis of the slow time in which accidents develop and the one (1) hour resonse time of the Technical Advisor we committed in our RERP (Section 5.2) to have the emergency organization activated within 90 minutes which would include an augmented staff equivalent to Table B-1, NUREG-0654. This staff augmentation is consistent with Technical Advisor response time and is certainly consistent with accident analysis and the accident development time frames.

3. <u>NRC Question/Comment</u>

The plan must (in addition to other NUREG-0610 notification requirements) specify that when a "general" emergency is declared that the off-site authorities <u>responsible</u> for implementation of protective measures will be notified by the "Plant Emergency Director" and advised of recommended protective actions within 15 minutes of the direction of the emergency condition. The plan must specify the content of this initial message to include:

a. Class of emergency
b. Whether a release is taking place
c. Affected areas
d. Protective measures

NOTE: The protective measures recommended in the initial message off-site may be - "go inside - turn on radio" (30 minutes) provided a followup message indicating more detail protective measures based on dose projections.

<u>PSC Response</u>

Per your request the notification time of fifteen (15) minutes after determination that a "general" emergency exists has been added to Table 4.1-4 of the RERP.

Sample notification messages as well as followup messages have been included in Section 6 of the RERP (see Figures 6.1-1 through 6.1-3).

4. NRC Question/Comment

Your plan must describe the public notification system to include:

a. The initial off-site contact who will be responsible for notifying the affected population (either the specific organization or individual).

b. The capability for 24-hour-per-day notification (to off-site authorities).

c. The physical alerting system to be used: sirens. NOAA weather of emergency alert, telephone automatic dialers, aircraft with loudspeakers (which will be used to alert public).

(10 db above average daytime ambient background is a target level for design of an adequate siren system.)

Distance	% Notified in 15 Minutes
5 miles	100%

d. The basis for any exceptions (e.g., for extended water areas with transient boats or remote hiking trails) must be documented.

e. Every year you must take a statistical sample of the residents of all areas within the 5 mile EPZ to assess the public's awareness of the prompt notification system and the availability of information on what to do in an emergency.

Plan must also include a provision for corrective measures to provide reasonable assurance that coverage approaching the design objectives is maintained.

f. The provisions for use of a public media system (radio, TV) to provide clear instructions to the public.

a. Twenty-four hour station - total plume coverage.

b. Include in the plan the messages to be transmitted to the public (cover a range of protective actions).

It is the operator's responsibility to ensure that the means exist for notifying and providing prompt instructions to the public.

It is the responsibility of the State and local governments to activate the system.

PSC Response

a. The County and the State have the responsibility for notifying the affected population. Additional clarification has been added to Section 6.1 of the RERP.

b. The capability of 24 hour per day notification to off-site authorities is included in Section 7.2 of the RERP.

c. and d. The physical alerting systems proposed both by the Fort St. Vrain RERP and the State RERP is the use of the emergency broadcast system (radio) and the dispatch of County personnel in vehicles and loud speakers to warn the general public. As indicated in our letter P-80083, March 18, 1980, we have more than adequate time to effect necessary protective actions prior to exceeding any of the Protective Action Guidelines. During the February 28, 1980, drill we tested the State Emergency Response Plan and determined that the general public in the affected zone could be notified within 90 minutes. As indicated in our letter P-80066 we have up to twelve (12) hours to notify the public prior to exceeding the most restrictive Protective Action Guideline (iodine inhalation) at the exclusion area boundary. The plan for public notification includes the use of several emergency broadcast bands and the use of television augmented, of course, with dispatched personnel. On the basis of the time frames associated with our accident analyses and the use of radio and television media we can see no justification for an early warning alert system.

e. As a part of the public information program a statistical sample of the residents within the five (5) mile EPZ will be taken to assess the public's awareness of what to do in an emergency. The details of the public information program are still being worked out between PSC and the State. Once this program is defined Section 8.1.1.d of the RERP will be developed to define the program.

f. Use of the media systems to provide instructions to the public is the responsibility of State, County, and local authorities. The State RERP which is approved by PSC has provisions for the use of media systems that meet the requirements of NUREG-0654. Section 6.4.1.b of the RERP addresses this subject.

5. NRC Question/Comment

Under PSC actions, Table 4.1-1, page 1, the first item, the 2-hour time limit for notification of an occurrence of an Unusual Event should be changed to "as soon as discovered". Similarly, the fourth item under PSC action should include a requirement for a written summary within 24 hours.

ATTACHMENT IV

ALTERNATE SCENARIO, FSV EMERGENCY PLANNING ZONES

REVISED SCENARIO OF ATTACHMENT I

ATTACHMENT IV

ESTIMATE OF FSV EMERGENCY PLANNING
ZONES - ALTERNATE SCENARIO

Summary

The FSV emergency planning zone(EPZ) distances have been re-evaluated utilizing a set of accident scenario assumptions felt to be responsive to the inherent features of the HTGR while still maintaining a sufficiently conservative analysis approach. The alternate EPZ calculation described herein, may supplant previous conservative treatments of the EPZ question (Ref. 1, Attachment I) with a more prudent selection of parameter assignments for the DBA-1 scenario.

The results of the alternate scenario analysis indicate that the direct plume exposure protective action guide (PAG) limits for population evacuation are not exceeded outside the plant exclusion boundary (see Figure 1-3). However, the simplified approach of comparing the resulting FSV dose versus distance curve for the DBA-1 scenario with that of a "reference" 1000 MW(e) LWR undergoing a traditional (TID) siting event at the same site, an alternate scenario EPZ of about one mile for plume exposure protection can be derived. Additionally, utilizing this same comparative technique, an EPZ radious for ingestion pathway considerations of about five miles is derived.

Discussion

The Reference 1 (Attachment I) analysis was re-examined in light of more prudent selections of scenario parameter assignements and assumptions. For the current alternate scenario analysis, the assumptions are summarized as follows:

1. A stack filter efficiency of 95% for iodine removal was used in the current analysis. A value of 90% was used previously (Ref. 1, Attachment I)

Initial and periodic testing of the iodine filters at FSV ensure that the filter units shall show >90% radioactive methyl iodide removal and bypass leakage (DOP test) of <1% (Ref. 2). Additionally, elemental iodine adsorption efficiencies for the activated carbon filters are expected to be on the order of 99.9% (Ref 3 and 4). Since elemental and particulate iodine constitute the largest fraction of iodine likely to be present in the PCRV to containment release,* an effective total iodine (i.e., methyl and elemental forms) of approximately 99% can be justified. However, a conservative assignment of 95% was employed in the present alternate scenario analysis. It should be pointed out that this assignement of filter efficiency agrees well with the Regulatory Guide 1.52 (Ref. 5) assigned activated carbon decontamination efficiency of 95% for a 2-in. bed with relative humidity ** ≤70%.

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* For LWR LOCA analysis, U. S. Reg. Guide 1.3 & 1.4 specifies the elemental and particulate iodine fraction as 96% of the total and organic (methyl) forms as 4%.

** The DBA-1 scenario is not anticipated to produce large water-to-containment releases and hence the relative humidity in the containment atmosphere throughout the course of the accident should be low.

2. The King Furnace fuel-to-core release fractions were employed for iodine release in the alternate scenario, i.e., 5.5% of the core iodine is available for leakage from the PCRV circuit versus 25% for the TID-14844 release previously employed.

3. A PCRV leak rate of 0.1%/day was employed, consistent with large HTGR containment building leak rates.

4. Natural deposition of condensible fission products in the reactor building was allowed. A reduction of about 40% for iodines was employed for the alternate scenario.

   In the previous Ref. 1 (Attachment I) analysis, TID-14844 assumptions on iodine reduction were employed, i.e., 50% of the fuel-borne iodine is assumed to be released from the fuel to the PCRV circuit and of this 50%, a further reduction of 50% was employed within the PCRV to account for plateout, thus resulting in an effective source of 25% of the fuel iodine inventory available for leakage from the PCRV. No credit for further plateout of the iodine once released to the reactor building was employed in the Ref. 1 (Attachment I) analysis (the current alternate scenario employs about a 40% reduction due to reactor building plateout phenomenon).

5. All other conditions were identical to PSC EPZ assumptions as given in Ref. 1 (Attachment I).

Table 1 sumarizes the assumptions employed in the current alternate scenario analysis, as well as provides the reader with a statement of some of the additional conservatisms felt to be present in the Ref. 1 (Attachment I) analysis, but not modified at this time.

Results

The analysis methods employed in this current study are those same methods employed in the Ref. 1 (Attachment I) work. Figures 1 and 2 summarize the cumulative (to 30 days) thyroid inhalation dose and whole body gamma exposure, respectively, for the DBA-1 alternate scenario of this study. Additionally, for comparison purposes, the previous FSV EPZ study results as well as 1000 MW(e) LWR results are also displayed.

It is observed that beyond the FSV EAB of 590 meters, neither whole body gamma nor thyroid inhalation dose exceeds the PAG for mandatory evacuation on the basis of DBA-1 predicted doses using the alternate scenario.

Applying the Ref. 1 (Attachment I) approach of comparing the FSV dose versus distance curves for the DBA-1 scenario with a "reference" 1000 MW(e) LWR, to the alternate scenario dose curves of Figs.1 and 2 results in an EPZ of about one mile for the plume submersion pathway. Using this same method, five miles was derived for the ingestion exposure pathway. These results are an order of magnitude lower than the "reference" LWR EPZ's which are 10 miles for plume exposure and 50 miles for the ingestion food pathway. It is noted that the NRC subsequently has ruled that the plume exposure and ingestion pathway distances are respectively 5 and 30 miles for Fort St. Vrain and LWR's of 250 MW(e) or less in size.

Fig. 3 presents the time history of the inhalation thyroid dose at the EAB (590 meters) and at the Platteville boundary distance of three miles. This figure shows graphically the slow buildup of dose at the dose receptor point, characteristic of the HTGR.

References

1. "FSV Emergency Planning Zone (EPZ) Assessment," PSC to NRC Letter P-80066.

2. FSV Technical Specifications, "SR 5.5.3 - Reactor Building Exhaust Filters, Surveillance," Amendment No. 18.

3. FSV FSAR, Amendment 27, p. 6-1-8.

4. "Nuclear Power Plant Air Cleaning Units and Components," Table 5-1 Test 5d., elemental iodine retention, ANSI/ASME Standard N509-1976.

5. "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," U. S. Regulatory Guide 1.52, Rev. 2, March 1978.

ALTERNATE SCENARIO - PARAMETER ASSIGNMENTS, FSV EMERGENCY PLANNING ZONE ANALYSIS

Parameter	PSC-EPZ Study Assumption (Ref. 1 Anal.)	Alternate Scenario Assumption (this study)	Comments
	Employed in Analysis		
PCRV leak rate	0.2%/day	0.1%/day	While 0.2%/day was conservatively assigned in the FSAR, a value of 0.024%/day was calculated (Ref. FSAR D.1.3.4.2). The value of 0.1%/day selected for the alternate scenario is consistent with large HTGR containment leakrate assignment.
Stack filter efficiency Halogens	90%	95%	Stack filters are currently tested in >90% for methyliodide removal. However, the iodine specie expected to be present in the containment atmosphere is predominantly elemental iodine with removal efficiencies <99%. (See text).
Iodine release from the fuel to the PCRV circuit (includes plateout removal of 50% within the PCRV for PSC-EPZ study)	25%	5.5%	King furnace fuel release fractions consistent with model described in FSV FSAR.
Natural deposition of condensible radionuclides in the containment bldg. Halogens	0	~40% (a)	Percentage given specifies the fraction of the initially airborne iodine source removed by natural deposition.
	Parameters maintained unchanged in current alternate scenario analysis in which additional conservative margin is present.		
Thermal power	879 MW	Same	100% power is 842 MW(t)
Availability	100%	Same	80% availability is more realistic
Stack filter efficiency for particulate removal	95%	Same	99% removal is more characteristic of HEPA filter operation (Note: 1% bypass should be considered).

TABLE 1 (cont.)

Parameter	PSC-EPZ Study Assumption (Ref. 1 anal.)	Alternate Scenario Assumption (this study)	Comments
Natural deposition of particulate (other than halogens) in the reactor	ignored	same	Reduction of about 12% of the particulate materials present in the reactor building atmosphere is possible (see note (b)).
Particulate release from the fuel to the PCRV circuit (other than halogens and noble gases)	1%	same	King Furnace fractions predict miniscule amounts of most particulates, i.e., <<1% (see Table D-1-10
Meteorology assignment	Pasquill category F like, constant persistance of plume trajectory.	same	A more moderate, Pasquill category D like condition is expected to prevail.

(a) For the large HTGR (representative of FSV) $\lambda_{nat\ \mathrm{Iodine}} = 1\ \mathrm{hr}^{-1}$: $\frac{\alpha c}{\alpha c + \alpha p} = \frac{1.5}{1.5 + 1} = 0.6$; Amount of airborne source removed by natural deposition (plateout) = 100-60 = 40%

(b) For the large HTGR (representative of FSV) $\lambda_{nat\ \mathrm{part.}} = 0.2\ \mathrm{hr}^{-1}$; $\frac{\alpha c}{\alpha c + \alpha p} = \frac{1.5}{1.5 + 0.2} = 0.88$; Amount of airborne source removed by natural deposition (gravitational settling) = 100-88 = 12%

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Figure 1

EPZ Accident Scenario - FSV Cumulative Thyroid Inhalation Dose (to 30 days) vs. Downwind Distance

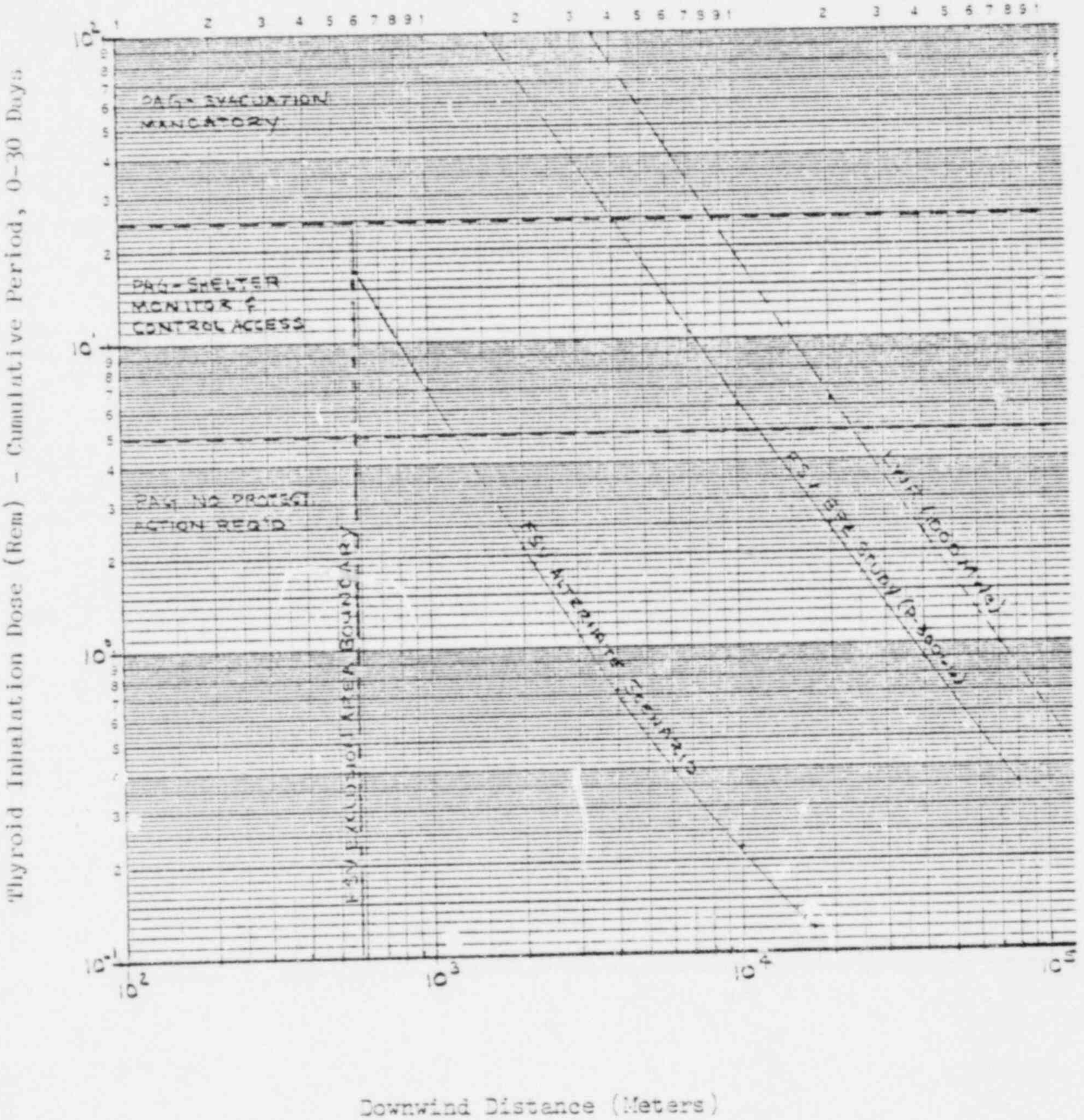

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Figure 2

EPZ Accident Scenario - FSV Cumulative Whole Body Gamma Dose (to 30 Days) vs. Downwind Distance

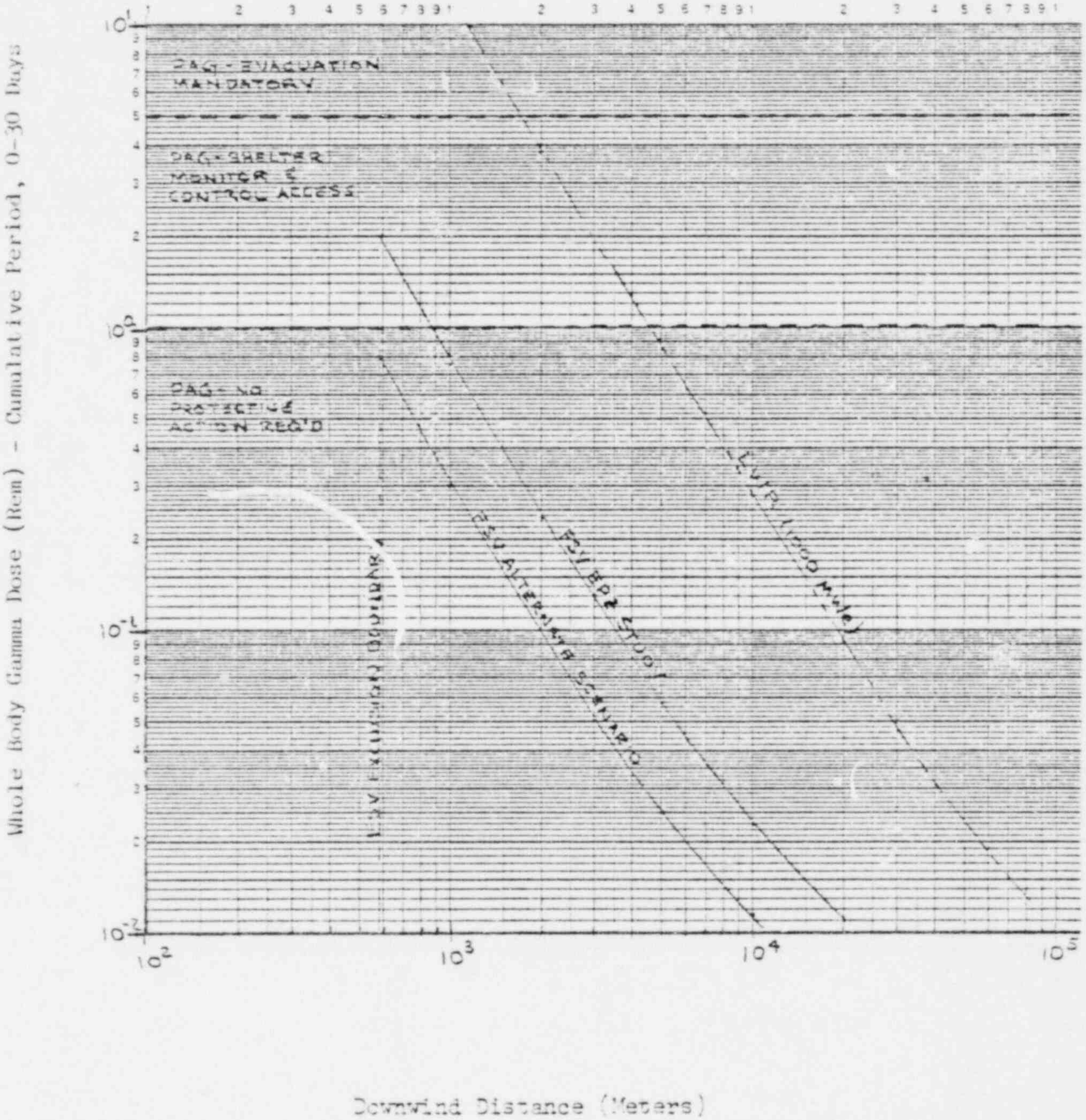

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Figure 3

EPZ Accident Scenario - FSV Cumulative Thyroid Inhalation Dose vs Elapsed Time for Exclusion Boundary (590 Meters) and Platteville Boundary (5 miles)