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Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Dr. East • Welch MN 55089

February 18, 2003

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10 CFR 50.4

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET NOS. 50-282 AND 50-306
LICENSE NOS. DPR-42 AND DPR-60
PRAIRIE ISLAND EMERGENCY PLAN IMPLEMENTING PROCEDURES

Furnished with this letter are the recent changes to the Prairie Island Nuclear Generating Plant Emergency Plan Implementing Procedures. This submittal includes the following documents:

INDEXES:
Emergency Plan Implementing Procedures Table of Contents

REVISIONS:

F3-7	Activation and Operation of Operational Support Center (OSC)	Rev. 16
F3-17	Core Damage Assessment	Rev. 10
F3-19	Personnel and Equipment Monitoring and Decontamination	Rev. 8

TEMPORARY CHANGE ADDITIONS:

F3-23.1	Emergency Hotcell Procedure	TCN # 2003-0080

DELETIONS:
None

A045

INSTRUCTIONS:
Please post changes in your copies of the Prairie Island Nuclear Generating Plant Emergency Plan Implementing Procedures (F3). Procedures, which have been superseded or deleted, should be destroyed. Please sign and return the acknowledgment of this update to Bruce Loesch, Prairie Island Nuclear Generating Plant, 1717 Wakonade Drive East, Welch, MN 55089.

As per 10 CFR 50.4, two copies have also been provided to the Regional III Office and one to the NRC Resident Inspector. The Nuclear Management Company has not made new or revised existing Nuclear Regulatory Commission commitments in this letter or the attachments. If you have any questions, please contact Mel Agen at 651-388-1121 Extension 4240.

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Steve Orth, USNRC, Region III (2 copies)
NRC Resident Inspector-Prairie Island Nuclear Generating Plant
(w/o attachment)

Mfst Num: 2003 - 0074
FROM : Bruce Loesch/Mary Gadient
TO : UNDERWOOD, BETTY J
Copy Num: 515
SUBJECT : Revisions to CONTROLLED DOCUMENTS

Date : 01/30/03
Loc : Prairie Island
Holder : US NRC DOC CONTROL DESK

Procedure #	Rev	Title
Revisions:		
F3-17	10	CORE DAMAGE ASSESSMENT
F3-7	16	ACTIVATION & OPERATION OF OPERATIONAL SUP CENTER (OSC)
F3-19	8	PERSONNEL & EQUIPMENT MONITORING & DECONT
Temporary Change Additions:		
2003 0080 F3-23.1		EMERGENCY HOTCELL PROCEDURE

UPDATING INSTRUCTIONS

Place this material in your Prairie Island Controlled Manual or File. Remove revised or cancelled material and recycle it. Sign and date this letter in the space provided below within ten working days and return to Bruce Loesch or Mary Gadient, Prairie Island Nuclear Plant, 1717 Wakonade Drive E., Welch, MN 55089.
Contact Bruce Loesch (ext 4664) or Mary Gadient (ext 4478) if you have any questions.

Received the material stated above and complied with the updating instructions

_______________________________ Date _______________

PRAIRIE ISLAND NUCLEAR GENERATING PLANT	Title: Emergency Plan Implementing Procedures TOC Effective Date : 01/30/03
Approved By: Joyce Chitty /mb BPS Supt	

F3	ACTIVATION AND OPERATION OF OPERATIONAL SUPPORT CENTER	NUMBER: F3-7
		REV: 16

REFERENCE USE

- *Procedure segments may be performed from memory.*
- *Use the procedure to verify segments are complete.*
- *Mark off steps within segment before continuing.*
- *Procedure should be available at the work location.*

O.C. REVIEW DATE:	OWNER:	EFFECTIVE DATE
12/26/02 SC	M. Werner	1-30-03

F3	ACTIVATION AND OPERATION OF OPERATIONAL SUPPORT CENTER	NUMBER: F3-7
		REV: 16

1.0 PURPOSE

This procedure provides instruction for the activation and monitoring requirements of the Operational Support Center (OSC).

2.0 APPLICABILITY

This instruction **SHALL** apply to the OSC Coordinator and all other plant personnel who may report to the OSC.

3.0 PRECAUTIONS

3.1 Only those personnel designated by this instruction or as requested by plant supervisors, should assemble in the Operational Support Center. All other personnel in Records Room should evacuate when the OSC is activated.

3.2 All personnel assigned to the OSC should remain in the OSC unless directed to report elsewhere. DO NOT congregate in the Control Room.

3.3 The OSC is provided with makeup and exhaust ventilation from the Unit 1 turbine deck. This ventilation is equipped with fire dampers and motor dampers that fail shut on fan shutdown. The fans are controlled by local switches on the west wall of the OSC. These fans should be operated together and may be shutdown to mitigate introduction of smoke or airborne contamination from the turbine deck into the OSC.

4.0 RESPONSIBILITIES

4.1 The OSC Coordinator is responsible to implement the actions directed in this procedure.

4.2 The Radiation Protection Specialists are responsible to provide oversight of radiation monitoring of personnel and facilities in the OSC and Control Room.

4.3 Operators, I&C Staff, Electricians, Mechanical Maintenance Staff and Nuclear Plant Service Attendants are responsible to assist in the repair and operation of plant equipment as directed by their supervisors or OSC Coordinator.

5.0 DISCUSSION

The Operational Support Center (OSC) is located in the plant Operating Records Room adjacent to the Control Room.

The OSC provides a central location to assemble the necessary Operations Staff, Radiation Survey Teams, I&C Supervisors and technicians, Electrical & Mechanical Supervisors and staff, and Nuclear Plant Service Attendants. These teams support the operations of the plant during emergency conditions. If more space is needed for OSC personnel waiting for emergency work, the Operators Lounge area may be used.

The OSC **SHALL** be activated whenever an Alert, Site Area Emergency, or General Emergency is declared.

Monitoring of the OSC for direct radiation and airborne radioactive materials (particulates and iodine) **SHALL** be performed to ensure habitability of the OSC.

6.0 PREREQUISITES

An Alert, Site Area, or General Emergency has been declared.

7.0 PROCEDURE

7.1 Activation and Operation of the OSC

7.1.1 The Operational Support Center **SHALL** be activated whenever an Alert, Site Area or General Emergency is declared.

A. During normal work hours, the following personnel should immediately report to the Operational Support Center, whenever an Alert, Site Area or General Emergency is announced over the Public Address System.

1. Operations personnel onsite, but not assigned to the on-shift crew.

2. Mechanical & Electrical Maintenance Supervisors.

3. Lead Electricians, Lead Machinists, Lead Riggers.

4. I&C Supervisors and designated I&C Specialists.

5. Radiation Survey Teams (unless directed otherwise by the Emergency Director or Radiological Emergency Coordinator).

6. Nuclear Plant Service Attendants.

7. Anyone as requested by their supervisor or the Emergency Director.

B. If activation occurs during off-normal work hours, the Emergency Director **SHALL** direct the Shift Emergency Communicator (SEC) to notify and activate the emergency organization in accordance with F3-5. The following personnel should report to the OSC to establish an initial compliment of support personnel to assist in the emergency:

1. Mechanical & Electrical Maintenance Supervisors
2. I&C Supervisors and designated I&C Specialists
3. Designated Electricians
4. Radiation Survey Teams
5. Nuclear Plant Service Attendants
6. Designated Purchasing & Inventory Control personnel
7. Designated operations personnel and extra operators considered necessary by the Shift Supervisor.
8. Anyone called in by their supervisor or Emergency Director.

7.1.2 All nonessential personnel should evacuate the Records Room when the OSC is activated.

7.1.3 Additional personnel may augment the Operational Support Center staff as deemed necessary.

7.1.4 The Operational Support Center should remain activated until the emergency has been terminated or as otherwise directed by the Emergency Director.

7.1.5 The Operational Support Center Coordinator should ensure proper activation and operation of the OSC by completing the duties listed on PINGP 574, OSC Coordinator checklist.

7.2 Radiological Monitoring

Described below are those actions taken to ensure radiological monitoring in the OSC and Control Room.

7.2.1 Verify operation of R-65, OSC Area Monitor.

7.2.2 If R-65 fails, or is not working, set up the AM-2 for monitoring:

A. Obtain the AM-2 from the OSC Emergency Locker.

B. Plug the AM-2 in and verify the green power light is on.

C. Source check the AM-2 with the button source in the OSC Locker and verify an upscale reading.

D. If the AM-2 fails (power loss, incorrect reading, etc.) contact the Radiation Protection Group for additional radiation monitors.

7.2.3 At about 15 mR/hr, consider evacuating all nonessential personnel to a low dose rate areas, such as the Operators Lounge area or first floor TSC.

7.2.4 Establish operation of the OSC/Control Room CAM.

A. The CAM is in a hot standby condition with the electronics energized and the blower, chart, and filter paper off. Perform the following:

1. Turn the blower switch (located next to the recorder) to the ON position to start the blower, strip chart recorder, and the filter paper.

2. Adjust the blower flow rate to 3 SCFM using the toggle switch located on the right side of the CAM. Then adjust suction flow so that 50% is from OSC and 50% is from Control Room.

3. Verify the CAM is in operation (i.e., verify the blower, filter, and strip charts are operating, meters are on scale, etc.).

4. If the CAM fails to operate properly contact the Radiation Protection Group for additional sampling.

B. Periodically monitor the CAM for airborne particulate and iodine activity.

C. Recommend the following Protective Actions, based on readings from the CAM.

1. CAM - Particulate

$1x10^{-9}$ µCi/cc - no protective action necessary.

> $1x10^{-9}$ - establish program of regular portable air samples by the Radiation Protection Group.

If portable air sample results > 1 DAC - consider evacuation of unnecessary personnel and limit exposures to less than 40 DAC - hours/week if possible.

If OSC portable air sample results > 10 DAC - consider evacuation of OSC personnel to the Control Room. Surplus OSC personnel may go to first floor TSC.

If Control Room air sample results > 10 DAC - respiratory protection for Control Room personnel is required.

2. CAM - Iodine

If the CAM alarms for iodine (5 x 10^{-9} µCi/cc) - establish program of regular portable air samples by the Radiation Protection Group.

If portable air sample results > 1 DAC - consider evacuation of unnecessary personnel and limit exposures to less than 40 DAC - hours/week if possible.

If OSC portable air sample results > 10 DAC - consider evacuation of OSC personnel to the Control Room. Surplus OSC personnel may go to first floor TSC.

If Control Room air sample results > 10 DAC - respiratory protection for Control Room personnel is required.

NOTE:	**The Radiological Emergency Coordinator (REC) should recommend the use of potassium iodide pills (thyroid blocking agent) if the projected thyroid exposure approaches 25 REM CDE. See F3-18, Thyroid Iodine blocking Agent (Potassium Iodide), for determining projected thyroid exposures.**

7.3 Dosimetry Issue

7.3.1 If the event is a radiological event, an RPS should issue dosimetry to each individual in the OSC and Control Room and log initial dosimeter readings on PINGP 652, Emergency Center Activation Exposure Records.

7.3.2 Refer to F3-12, Emergency Exposure Control, for the Administrative Control and Documentation of Exposure, in regards to work teams dispatched from OSC.

7.4 Operations Lounge Area

The Operations Lounge Area may be used by OSC personnel, as necessary, for some of the following reasons:

7.4.1 Need for more space for large number of OSC personnel on standby for emergency work.

7.4.2 Dose rates in the Operations Lounge Area are lower than dose rates in the primary OSC.

7.5 Considerations during loss of AC power in the OSC

7.5.1 Equipment:

A. Phones will continue to operate.

B. Emergency lights will operate.

C. Flashlights are stored in emergency locker.

D. Dosimetry and radiation meters will continue to operate.

E. The OSC access door will continue to operate.

7.5.2 Re-evaluate the number of craft workers needed in OSC and Operations Lounge.

7.5.3 Consider sending extra personnel back to work, to Maintenance Lunchroom, Old Admin Lunchroom, NAB Lunchroom, or PITC depending on radiological conditions.

F3	CORE DAMAGE ASSESSMENT	NUMBER: F3-17
		REV: 10

REFERENCE USE
• *Procedure segments may be performed from memory.*
• *Use the procedure to verify segments are complete.*
• *Mark off steps within segment before continuing.*
• *Procedure should be available at the work location.*

O.C. REVIEW DATE:	OWNER:	EFFECTIVE DATE
012903 SC	M. Werner	1-30-03

F3	CORE DAMAGE ASSESSMENT	NUMBER: F3-17
		REV: 10

1.0 PURPOSE

The purpose of this procedure is to provide a means to best estimate the degree of reactor core damage from the measured fission product concentrations in water and gas samples taken for the primary system and containment under accident conditions.

2.0 APPLICABILITY

This procedure **SHALL** apply to the Nuclear Engineering Staff.

3.0 PRECAUTIONS

3.1 The numbers obtained using this procedure are at best, estimates only.

3.2 When making core damage calculations as per this procedure, considerations should be given to other plant indicators, for example:

3.2.1 Incore Thermocouples.

3.2.2 Reactor Coolant Loop Radiation Monitors (R70/71).

3.2.3 Containment Radiation Monitors (R48/49).

3.2.4 Hydrogen Concentration in the Containment Atmosphere

3.3 Spiking may occur after a shutdown or significant power change, usually during the 2 to 6 hour period following the power change. Iodine spiking is a characteristic of the condition where an increase in the normal primary coolant activity is noted, but no damage to the cladding has occurred.

4.0 RESPONSIBILITIES

The Nuclear Engineering Group is responsible to estimate the degree of reactor core damage according to the guidance provided in this procedure.

5.0 DISCUSSION

The approach utilized in this methodology of core damage assessment is measurement of fission product concentrations in the primary coolant system, and containment, when applicable, utilizing the post accident sampling system.

Certain nuclides have been selected to be associated with each particular core damage state, i.e., clad damage, fuel overheat and fuel melt. These nuclides reach equilibrium quickly within the fuel cycle. Once equilibrium condition are reached, a fixed inventory of the nuclides is assumed to exist within the fuel pellet. For these nuclides which reach equilibrium, their relative ratios within the fuel pellet can also be considered to be constant. During operation, certain volatile fission products collect in the gap. The relative ratios in the gap can also be considered to be constant, however, the distribution of the nuclides in the gap is not in the same proportion as the fuel pellet inventory since the migration of each nuclide into the gap is dependent on its particular diffusion rate. The relative ratios of the nuclides analyzed during an accident may be compared to the predicted relative ratios existing in the gap and fuel pellet to determine the source of the fission product release, i.e., gap release or fuel pellet.

Clad damage is characterized by the release of these fission products, i.e., isotopes of the noble gases, iodine, and cesium which have accumulated in the gap and during the operation of the plant. When the cladding ruptures, it is assumed that the fission product gap inventory of the damaged fuel rods is instantaneously released to the primary system. For this methodology it is assumed that the noble gases will escape through the break of the primary system boundary to the containment atmosphere and the iodines will stay in solution and travel with primary system water during the accident.

Fission product release associated with overtemperature fuel conditions arises initially from the portion of the noble gas, cesium and iodine inventories that was previously accumulated in grain boundaries. In addition, small amounts of the more refractory elements, barium-lanthanum, and strontium are also released.

Fuel pellet melting leads to rapid release of many noble gases, halides, and cesiums remaining in the fuel after overheat conditions. Significant release of the strontium, barium-lanthanum chemical groups is perhaps the most distinguishing feature of melt release conditions.

Auxiliary indicators such as core exit thermocouples, reactor vessel water level, reactor coolant loop radiation monitors, containment radiation monitors, and the containment hydrogen concentration are available for estimating core damage. These indications should confirm the core damage estimates which in turn are based on the radionuclide analysis.

6.0 PREREQUISITES

An emergency of an Alert, Site Alert, or General Emergency has been declared.

7.0 PROCEDURE

NOTE:	The program B80DAMASS may be used whenever core damage estimates are desired.

7.1 Request the Radiation Protection Group to obtain the applicable samples to enable an adequate assessment of core damage. See Table 1 for suggested sampling locations.

7.2 Obtain the following plant data at the approximate sample time:

7.2.1 Incore Thermocouple Map

7.2.2 Containment Pressure

7.2.3 Containment Temperature

7.2.4 Containment Hydrogen Concentration

7.2.5 Containment Radiation Level

7.2.6 Containment Sump Level

7.2.7 RVLIS Level

7.3 Perform B80DAMASS according to the instructions in SWI-NE-5 (23) to obtain core damage estimates. Continue with Step 7.15 of this procedure when the B80DAMASS run is complete.

NOTE:	If the computer is not available, perform the following manual calculations to obtain core damage estimates.

7.4 Decay correct the specific activities determined by the sample analysis, back to the time of reactor shutdown, as follows:

NOTE:	The decay correction may have been accomplished by the computer during the spectrum analysis. Therefore, this step may not need to be completed.

$$A_0 = \frac{A}{e^{-\lambda_i t}}$$

Where:

A = measured specific activity, µCi/gm or µCi/cc

λ_i = decay constant of isotope i, sec^{-1}

t = time elapsed from reactor shutdown to time of sampling, sec.

A_0 = decay corrected specific activity µCi/gm or µCi/cc

7.5 If a parent-daughter relationship exists for a specific isotope, the following steps should be followed to calculate the fraction of the measured activity due to the decay of the daughter that was released and then to calculate the activity of the daughter released at shutdown.

7.5.1 Calculate the hypothetical daughter concentration (Q_B) at the time of the sample analysis assuming 100 percent release of the parent and daughter source inventory:

$$Q_B(t) = K_i \frac{\lambda_B}{\lambda_B - \lambda_{Ai}} Q^0_{Ai}\left(e^{-\lambda_{Ai} t} - e^{-\lambda_B t}\right) + Q^0_B\, e^{-\lambda_B t}$$

Where:

Q^0_{Ai} = 100% source inventory (Ci) of parent i, Table 2 or Table 4.

Q^0_B = 100% source inventory (Ci) of daughter, Table 2 or Table 4.

$Q_B(t)$ = hypothetical daughter activity (Ci) at sample time.

K_i = if parent has 2 daughters, K_i is the branching factor, Table 3.

λ_{Ai} = decay constant of parent i, sec^{-1}

λ_B = daughter decay constant, sec^{-1}

t = time period from shutdown to time sample, sec.

7.5.2 Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time:

$$Fr = \frac{Q^0_B e^{-\lambda_B t}}{Q_B(t)}$$

7.5.3 Calculate the amount of decay corrected sample specific activity associated with just the daughter that was released.

$$M^0_B = Fr \times A_0$$

Where: A_0 = decay corrected specific activity (µCi/gm or µCi/cc) as determined by the analysis.

7.6 Determine the total volume or mass of the medium which was sampled.

7.6.1 Containment Volume:

V = containment free volume (cc's)

= 3.74×10^{10} cc's

7.6.2 Liquid Mass:

A. Liquid temperature < 200°F

$$\text{Mass (grns)} = \text{volume (ft3)} \times \rho_{STP} \times \frac{28.3 \times 10^3 cc}{ft^3}$$

Where: ρ_{STP} = water density at STP = 1.0 gm/cc

B. Liquid temperature > 200°F

$$\text{Mass (gms)} = \text{volume } (ft^3) \times \frac{\rho}{\rho_{STP}} (2) \times \rho_{STP} \times \frac{28.3 \times 10^3 cc}{ft^3}$$

Where: $\frac{\rho}{\rho_{STP}}(2)$ = water density ratio at medium temperature, from Figure 1

ρ_{STP} = water density at STP = 1.0 gm/cc

7.7 Determine the total activity of each isotope in each medium.

7.7.1 Containment Atmosphere:

$$\text{Total containment Activity (curies)} = A_0\ (\mu Ci/cc) \times V\ (cc's) \times \frac{Curie}{1 \times 10^6 \mu Ci}$$

Where: A_0 = Specific activity of containment atmosphere (μCi/cc), decay corrected to time of reactor shutdown and temperature/pressure corrected.

V = containment free volume (cc's)

= 3.74×10^{10} cc's

7.7.2 Liquid Sample:

$$\text{Total Liquid Activity (Curies)} = \text{Liquid MASS (gms)} \times A_0\ (\mu Ci/cc) \times \frac{Curie}{1 \times 10^6 \mu Ci}$$

Where: A_0 = Specific activity of liquid sample (μCi/gm), decay corrected to time of reactor shutdown.

7.8 The approximate total activity of each isotope in the liquid samples can now be calculated.

Total Water Activity = RCS Activity + Sump Activity + Activity Leaked to Secondary System.

7.9 Now the total activity of each isotope released at the time of the accident can be determined:

Total Activity Released = Total Water Activity + Containment Atmosphere Activity

7.10 Utilizing the total activity of each isotope released, calculate the activity ratios of the released fission products.

7.10.1 Noble Gas Ratio $= \dfrac{\text{Noble Gas Activity}}{\text{Xe}-133\text{ Activity}}$

7.10.2 Iodine Ratio $= \dfrac{\text{Iodine Activity}}{\text{I}-131\text{ Activity}}$

NOTE: **Steady state power conditions may be assumed where power does not vary by more than ± 10% of rated power level from time averaged value.**

7.11 Determine the power history prior to reactor shutdown.

7.12 Using the power history, determine a power correction factor for each isotope, in accordance with the following guidelines:

NOTE: **Steady state power condition is assumed where the power does not vary by more than ± 10% of rated power level from time averaged value.**

7.12.1 Steady State power prior to shutdown.

A. Half-life of nuclide < 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 4 Days}}{\text{Rated Power Level (Mwt)}}$$

B. Half-life of nuclide > 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 30 Days}}{\text{Rated Power Level (Mwt)}}$$

C. Half-life of nuclide ~ 1 year

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 1 year}}{\text{Rated Power Level (Mwt)}}$$

7.12.2 Transient power history in which the power has not remained constant prior to reactor shutdown.

NOTE: **For the majority of the selected nuclides, the 30-day power history prior to shutdown is sufficient to calculate a power correction factor.**

A. Power Correction Factor $= \dfrac{\Sigma_j \; P_j \left(1 - e^{-\lambda_i t_j}\right) e^{-\lambda_i t^0_j}}{RP}$

P_j = average power level (Mwt) during operating period t_j

RP = rated power level of the core (Mwt)

t_j = operating period in days at power P_j where power does not vary more than ±10 percent power of rated power level from time averaged value (P_j).

λ_i = decay constant of nuclide i in inverse days.

t^0_j = time between end of period j and time of reactor shutdown in days.

B. For the few nuclides with half-lives around one year or longer, a power correction factor which ratios effective full power days to total calendar days of cycle operation is applied.

$$\text{Power Correction Factor} = \frac{\text{Actual Operating EFPD of equilibrium cycle}}{\text{Total expected EFPD of equilibrium cycle operation}}$$

Where: Equilibrium Cycle = three (3) cycles of core operation (approximately 1050 EFPD)

7.12.3 For Cs-134, Figure 2 is used to determine the power correction factor. To use Figure 2, the average power during the entire operating period is required.

7.13 The total inventory of fission products available for release at reactor shutdown are calculated by applying the power correction factors to the equilibrium, end-of-life core inventories.

$$\text{Corrected Inventory} = \begin{array}{c}\text{Equilibrium Inventory at} \\ \text{end-of-life (Ci)} \\ \text{(Table 2)}\end{array} \times \begin{array}{c}\text{Power} \\ \text{Correction} \\ \text{Factor}\end{array}$$

7.14 Determine the percentage of inventory released, for each isotope.

$$\text{Release Percentage (\%)} = \frac{\text{Total Activity Released (Ci)}}{\text{Corrected Inventory (Ci)}} \times 100$$

7.15 The results of radionuclide analysis may now be used to determine an estimate of the extent of core damage.

7.15.1 From Figure 3 thru 15, estimate the extent of core damage by categorizing the percentage of clad damage, fuel over-temperature, and fuel melt.

7.15.2 Compare the calculated activity ratios with those listed in Table 5. Measured relative ratios greater than the gap activity ratios listed in Table 5 are indicative of more severe failures, e.g., fuel overheat.

7.16 To verify the conclusion of the radionuclide analysis, other indicators should now be used to provide verification of the estimate of core damage.

7.16.1 Containment Hydrogen Concentration:

A. Obtain the containment hydrogen concentration (%).

NOTE:	**Within the accuracy of this methodology, it is assumed that recombiners will have an insignificant effect on the hydrogen concentration when it is indicated that extensive zirconium-steam reaction could have occurred.**

B. From Figure 16, determine the percentage (%) zirconium water reaction.

C. Table 6 can be used to validate the extent of core damage estimate.

7.16.2 Core Exit thermocouple Readings:

A. Obtain as many core exit thermocouple readings as possible for evaluation of core temperature conditions.

NOTE:	If a thermocouple reads greater than 1650°F or is reading considerably different than neighboring thermocouples, thermocouple failure should be considered.

B. Compare the thermocouple readings with those in Table 7 to confirm the core damage estimate.

NOTE:	Radiation Monitors in containment may experience errors during first 4 hours after a DBA LOCA due to thermally induced errors. See Attachment 1 for more information.

7.16.3 Containment Radiation Monitor:

A. Obtain the containment dome monitor readings, R/Hr, from R-48 and/ or R-49.

B. From Figure 17, verify core damage estimate. The exposure rate in Figure 17 is based on the release of only noble gases to the containment. Halogens and other fission products were not considered to be significant contributors to the containment monitor reading.

7.16.4 Reactor Coolant Loop Radiation Monitor:

A. Obtain the reactor coolant loop radiation monitor readings, R/Hr, from R-70 and/or R-71.

B. From Figure 18, determine estimated core damage.

7.17 All indicators should confirm any core damage estimates. If radio-nuclide analysis and auxiliary indicators do not agree on core damage estimates, then recheck of indications may be performed, or certain indicators may be discounted, based on engineering judgment.

Table 1 Suggested Sampling Locations

Scenario	Principal Sampling Locations	Other Sampling Locations
Small Break LOCA Reactor Power > 1%*	RCS Hot Leg, Containment Atmosphere	
Reactor Power < 1%*	RCS Hot Leg**	
Large Break LOCA Reactor Power > 1%*	Containment Sump, Containment Atmosphere, RCS Hot Leg	
Reactor Power < 1%*	Containment Sump, Containment Atmosphere	
Steam Line Break	RCS Hot Leg,	Containment Atmosphere
Steam Generator Tube Rupture	RCS Hot Leg, Secondary System	
Indication of Significant Containment Sump Inventory	Containment Sump, Containment Atmosphere	
Containment Building Radiation Monitor Alarm	Containment Atmosphere, Containment Sump	
Safety Injection Actuated	RCS Hot Leg	
Indication of High Radiation Level in RCS	RCS Hot Leg	

* Assume operating at that level for some appreciable time.

** If a RCS hot leg sample is unavailable and the RHR system is operating, obtain a RHR system sample. However, for a RHR system sample to be a good representation of the RCS, the primary water should be circulating through the system.

Table 2 Fuel Pellet Inventory

Fuel Pellet Inventory*

Nuclide	Half Life	Inventory Curies**
Kr 85m	4.4	1.0×10^7
Kr 87	76 m	1.85×10^7
Kr 88	2.8 h	2.69×10^7
Xe 131m	11.8 d	2.94×10^5
Xe 133	5.27 d	9.26×10^7
Xe 133m	2.26 d	1.35×10^7
Xe 135	9.14 h	1.77×10^7
I 131	8.05 d	4.54×10^7
I 132	2.26 h	6.65×10^7
I 133	20.3 h	9.26×10^7
I 135	6.68 h	8.33×10^7
Rb 88	17.8 m	2.69×10^7
Cs 134	2 yr	1.09×10^7
Cs 137	30 yr	4.96×10^6
Te 129	68.7 m	1.51×10^7
Te 132	77.7 h	6.65×10^7
Sr 89	52.7 d	3.70×10^7
Sr 90	28 yr	3.36×10^6
Ba 140	12.8 d	7.91×10^7
La 140	40.22 h	8.33×10^7
La 142	92.5 m	7.07×10^7
Pr 144	17.27 m	5.81×10^7

* Inventory based on ORIGEN run for equilibrium, end-of-life core.
** Westinghouse, 2-Loop, 1650 Mwt Plant

Table 3 Parent-Daughter Relationships

Parent	Parent Half-Life*	Daughter	Daughter Half-Life*	K**
Kr-88	2.8 h	Rb-88	17.8 m	1.00
I-131	8.05 d	Xe-131m	11.8 d	.008
I-133	20.3 h	Xe-133m	2.26 d	.024
I-133	20.3 h	Xe-133	5.27 d	.976
Xe-133m	2.26 d	Xe-133	5.27 d	1.00
I-135	6.68 h	Xe-135	9.14 h	.70
Xe-135m	15.6 m	Xe-135	9.14 h	1.00
I-135	6.68 h	Xe-135m	15.6 m	.30
Te-132	77.7 h	I-132	2.26 h	1.00
Sb-129	4.3 h	Te-129	68.7 m	.827
Te-129m	34.1 d	Te-129	68.7 m	.680
Sb-129	4.3 h	Te-129m	34.1 d	.173
Ba-140	12.8 d	La-140	40.22 h	1.00
Ba-142	11 m	La-142	92.5 m	1.00
Ce-144	284 d	Pr-144	17.27 m	1.00

* Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition

** Branching decay factor

Table 4 Source Inventory of Related Parent Nuclides

Nuclide	Half-Life	Inventory, Curies
Xe-135m	15.6 m	1.97×10^7
Sb-129	4.3 h	1.49×10^7
Te-129m	34.1 d	3.74×10^6
Ba-142	11 m	7.65×10^7
Ce-144	284 d	4.83×10^7

Table 5 Isotopic Activity Ratios of Fuel Pellet and Gap

Isotopic Activity Ratios of Fuel Pellet and Gap*

Nuclide	Fuel Pellet Activity Ratio	Gap Activity Ratio
Kr-85m	0.11	0.022
Kr-87	0.22	0.022
Kr-88	0.29	0.045
Xe-131m	0.004	0.004
Xe-133	1.0	1.0
Xe-133m	0.14	0.096
Xe-135	0.19	0.051
I-131	1.0	1.0
I-132	1.5	0.17
I-133	2.1	0.71
I-135	1.9	0.39

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Inventory}}{\text{Xe-133 Inventory}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Isotope Inventory}}{\text{I-131 Inventory}}$$

* The measured ratios of various nuclides found in reactor coolant during normal operation is a function of the amount of "tramp" uranium on fuel rod cladding, the number and size of "defects" (i.e., "pin holes"), and the location of the fuel rods containing the defects in the core. The ratios derived in this report are based on calculated values of relative concentrations in the fuel or in the gap. The use of these present ratios for post accident damage assessment is restricted to an attempt to differentiate between fuel overtemperature conditions and fuel cladding failure conditions. Thus the ratios derived here are not related to fuel defect levels incurred during normal operation.

Table 6 Characteristics of Categories of Fuel Damage

Core Damage Category	Percent and Type of Fission Products Released	Fission Product Ratio***	Containment Radiogas Monitor R/hr 10 hrs after shutdown**	Core Exit Thermocouples Readings (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)
No clad damage	KR-87 $< 1 \times 10^{-3}$ Xe-133 $< 1 \times 10^{-3}$ I-131 $< 1 \times 10^{-3}$ I-133 $< 1 \times 10^{-3}$	Not Applicable	--	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 10^{-3} - 0.01 Xe-133 10^{-3} - 0.1 I-131 10^{-3} - 0.3 I-133 10^{-3} - 0.1	Kr-87 = 0.022 I-133 = 0.71	0 - 50	750 - 1300	Core uncovery	0 - 6
50-100% clad damage	Kr-87 0.01 - 0.02 Xe-133 0.1 - 0.2 I-131 0.3 - 0.5 I-133 0.1 - 0.2	Kr-87 = 0.022 I-133 = 0.71	50 to 100	1300 - 1650	Core uncovery	6 - 13
0-50% fuel pellet overtemperature	Xe-Kr, Cs, I 1 - 20 Sr-Ba 0 - 0.1	Kr-87 = 0.22 I-133 = 2.1	100 to 1.15E4	> 1650	Core uncovery	6 - 13
50-100% fuel pellet overtemperature	Xe-Kr, Cs, I 20 - 40 Sr-Ba 0.1 - 0.2	Kr-87 = 0.22 I-133 = 2.1	1.15E4 to 2.3E4	> 1650	Core uncovery	6 - 13
0-50% fuel melt	Xe, Kr, Cs, I 40-70 Sr-Ba 0.2 - 0.8 Pr 0.1 - 0.8	Kr-87 = 0.22 I-133 = 2.1	2.3E4 to 2.7E4	> 1650	Core uncovery	6 - 13
50-100% fuel melt	Xe, Xr, Cs, I, Te > 70 Sr, Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 2.7E4	> 1650	Core uncovery	6 - 13

Characteristics of Categories of Fuel Damage*

* This table is intended to supplement the methodology outlined in this report and should not be used without referring to this report and without considerable engineering judgement.

** These values are from Figure 17 and should be revised for times other than 10 hours.

*** $\frac{Kr\text{-}87}{Xe\text{-}133}$, $\frac{I\text{-}133}{I\text{-}131}$

Table 7 Expected Fuel Damage Correlation With Fuel Rod Temperature

Fuel Damage	Temperature °F*
No Damage	< 1300
Clad Damage	1300 - 2000
Ballooning of zircaloy cladding	> 1300
Burst of zircaloy cladding	1300 - 2000
Oxidation of cladding and hydrogen generation	> 1600
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Fuel Melt	> 3450
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
Melting of remaining UO_2	5100

* These temperatures are material property characteristics and are non-specific with respect to locations within the fuel and/or fuel cladding.

Figure 1 Water Density Ratio (Temperature vs. STP)

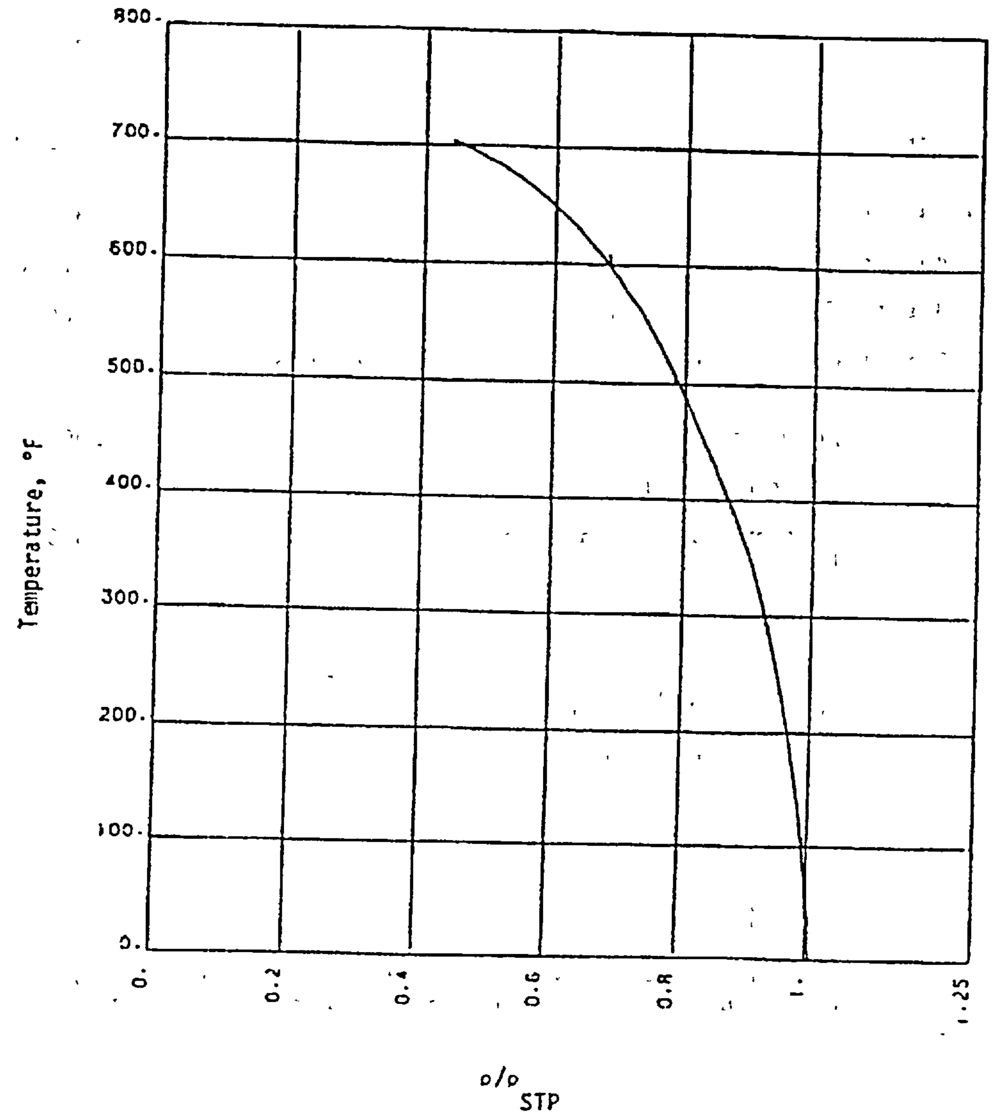

Figure 2 Power Correction Factor For CS-134 Based on Average Power During Operation

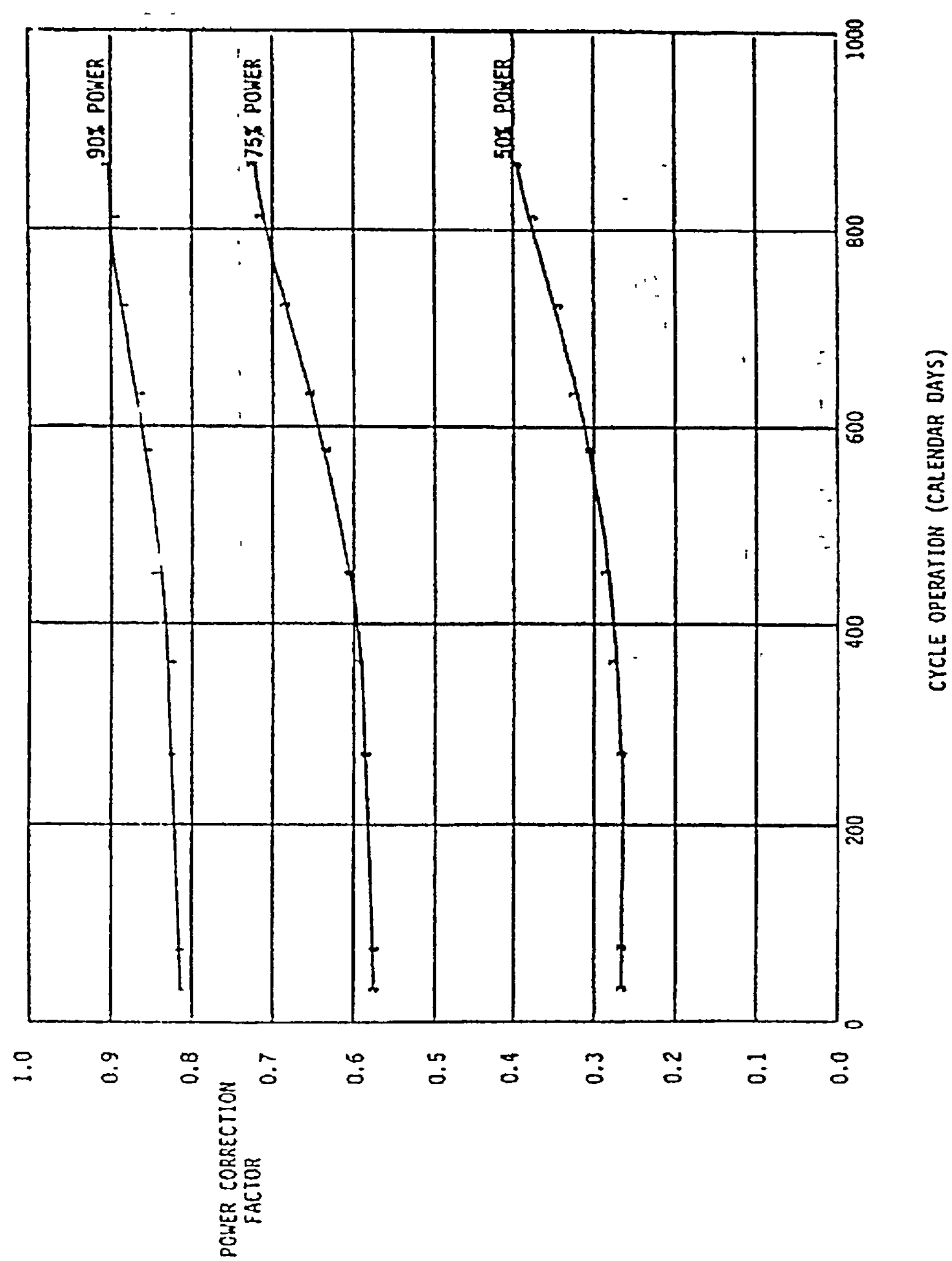

Figure 3 Relationship of % Clad Damage With % Core Inventory Released of XE-133

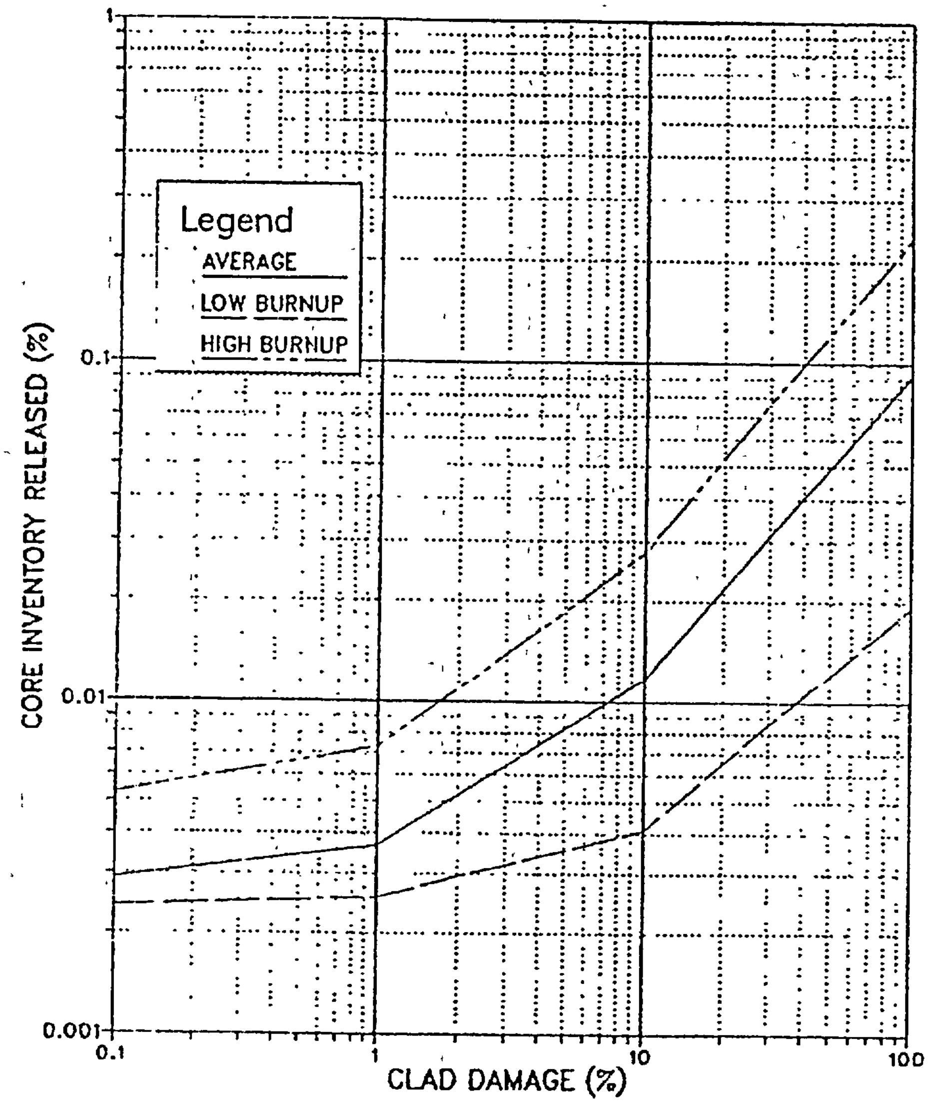

Figure 4 Relationship of % Clad Damage With % Core Inventory Released of I-131

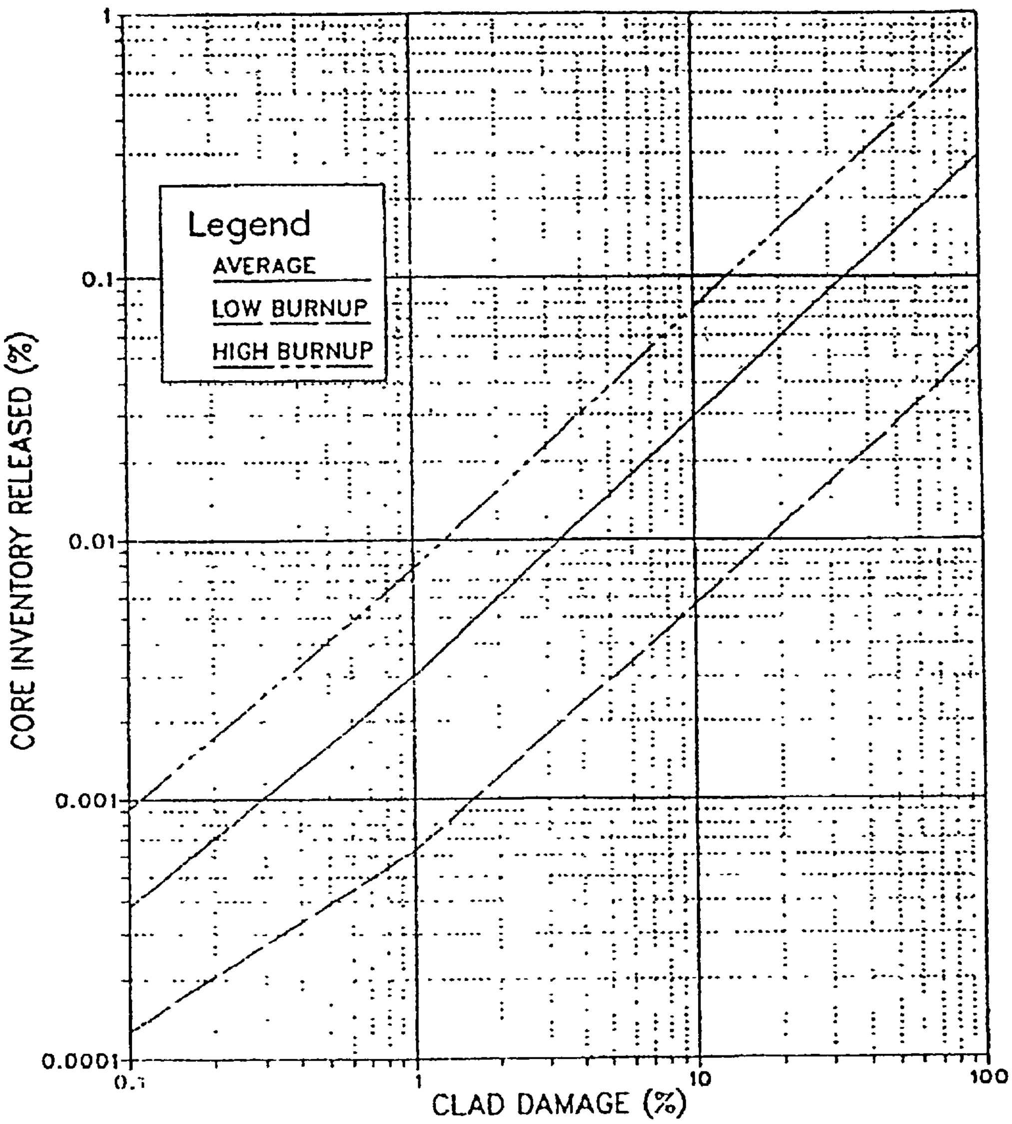

Figure 5 Relationship of % Clad Damage With % Core Inventory Released of I-131 W/Spiking

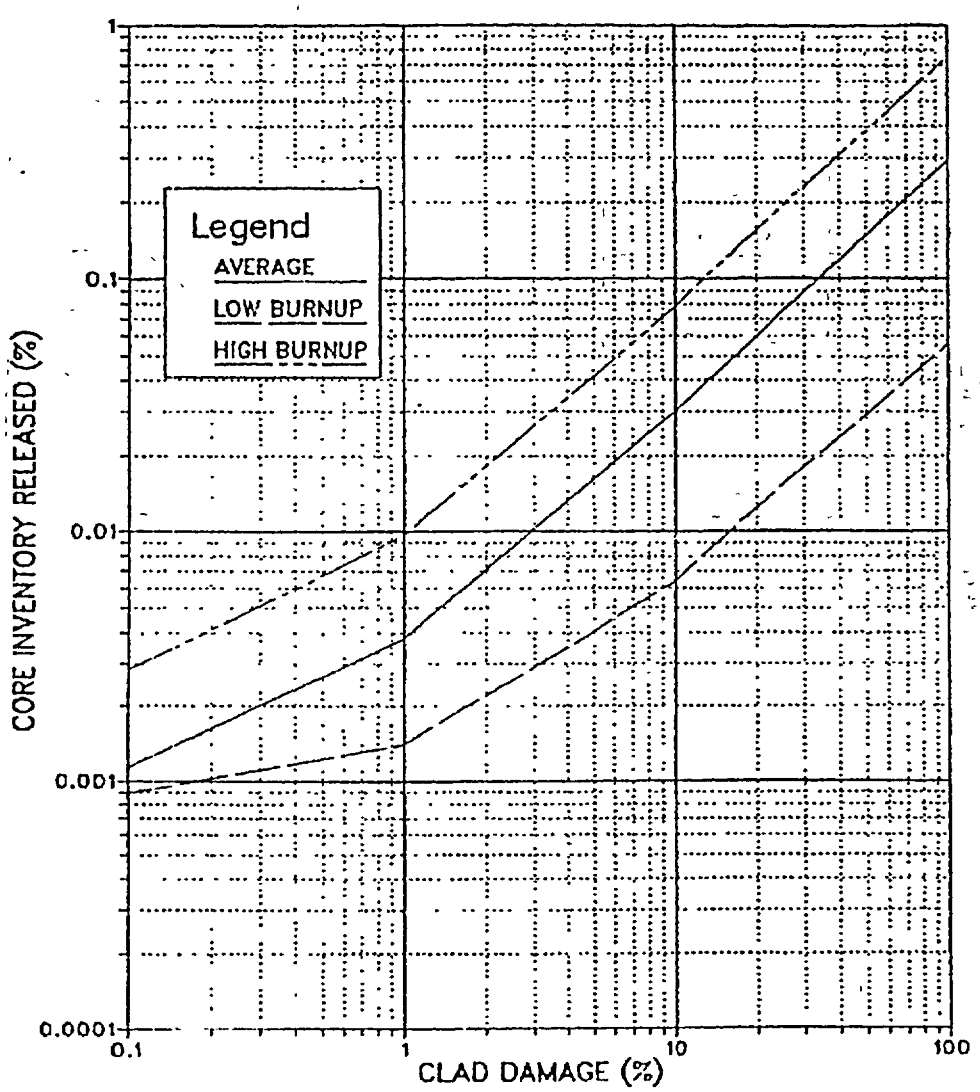

Figure 6 Relationship of % Clad Damage With % Core Inventory Released of KR-87

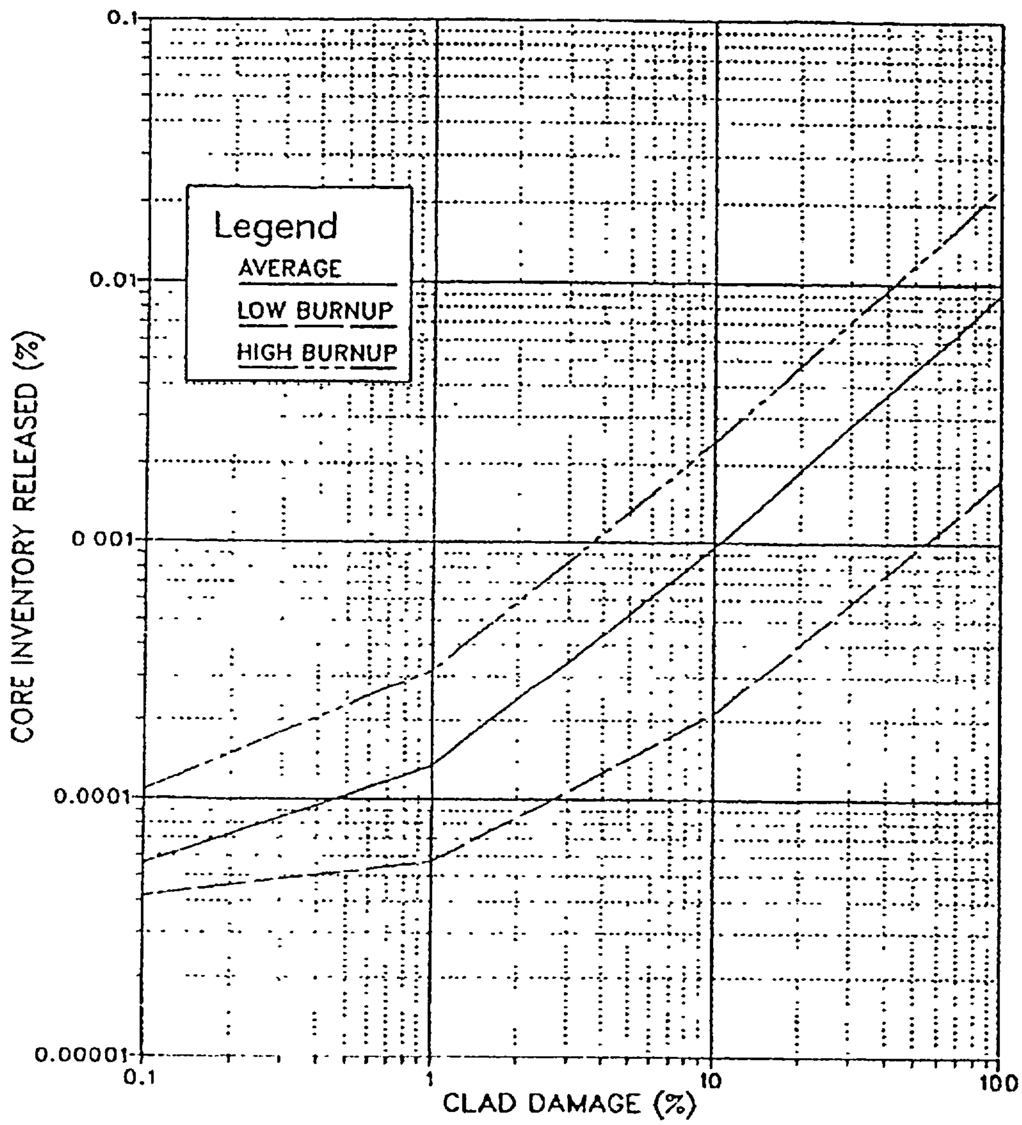

Figure 7 Relationship of % Clad Damage With % Core Inventory Released of XE-131M

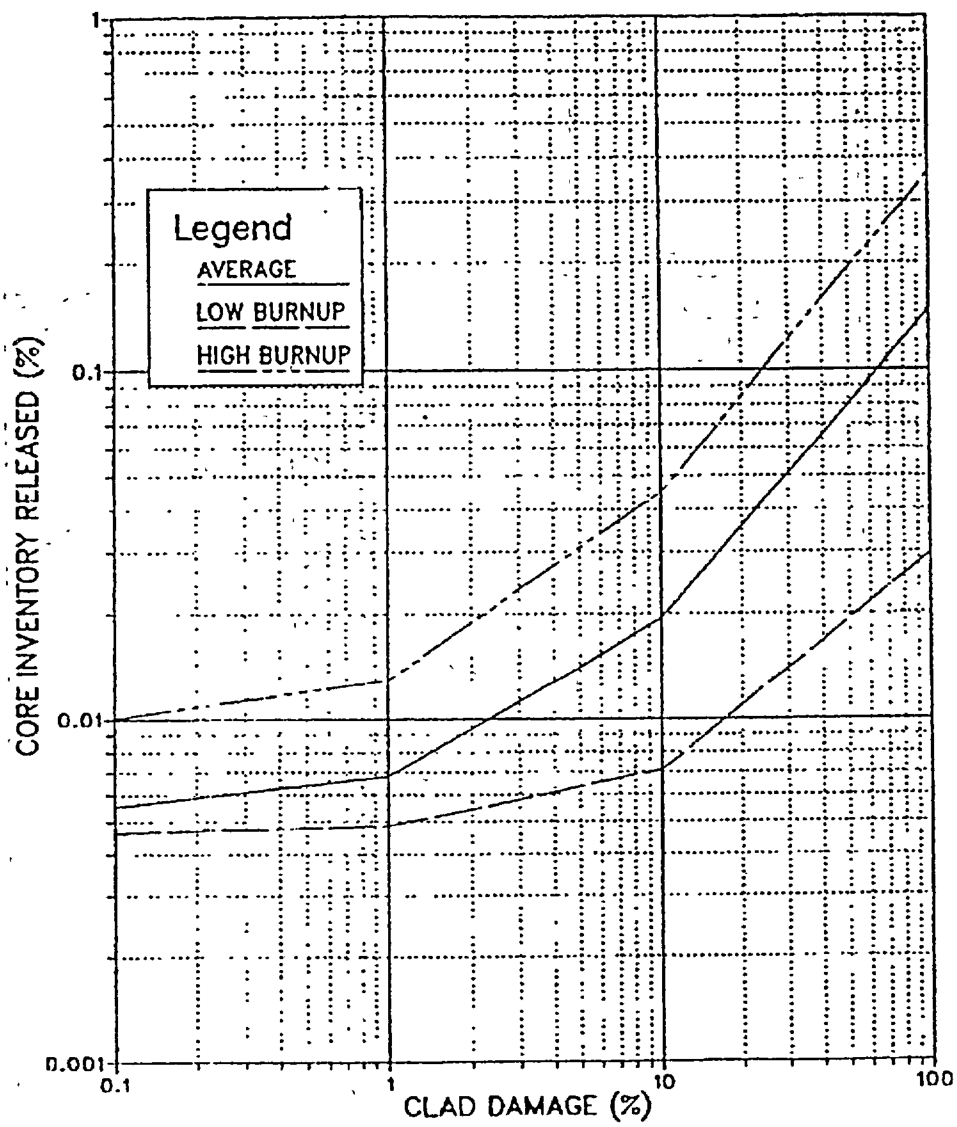

Figure 8 Relationship of % Clad Damage With % Core Inventory Released of I-132

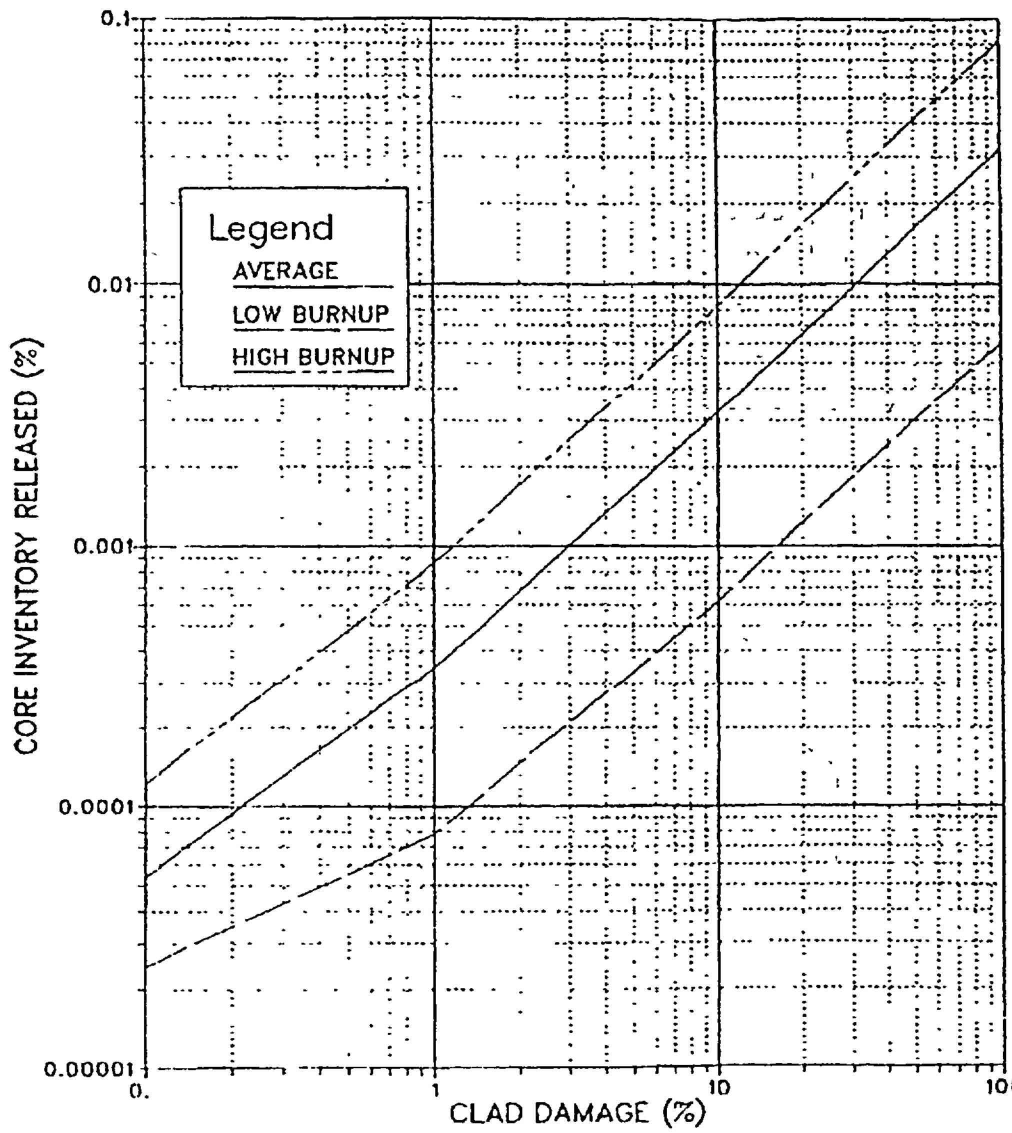

Figure 9 Relationship of % Clad Damage With % Core Inventory Released of I-133

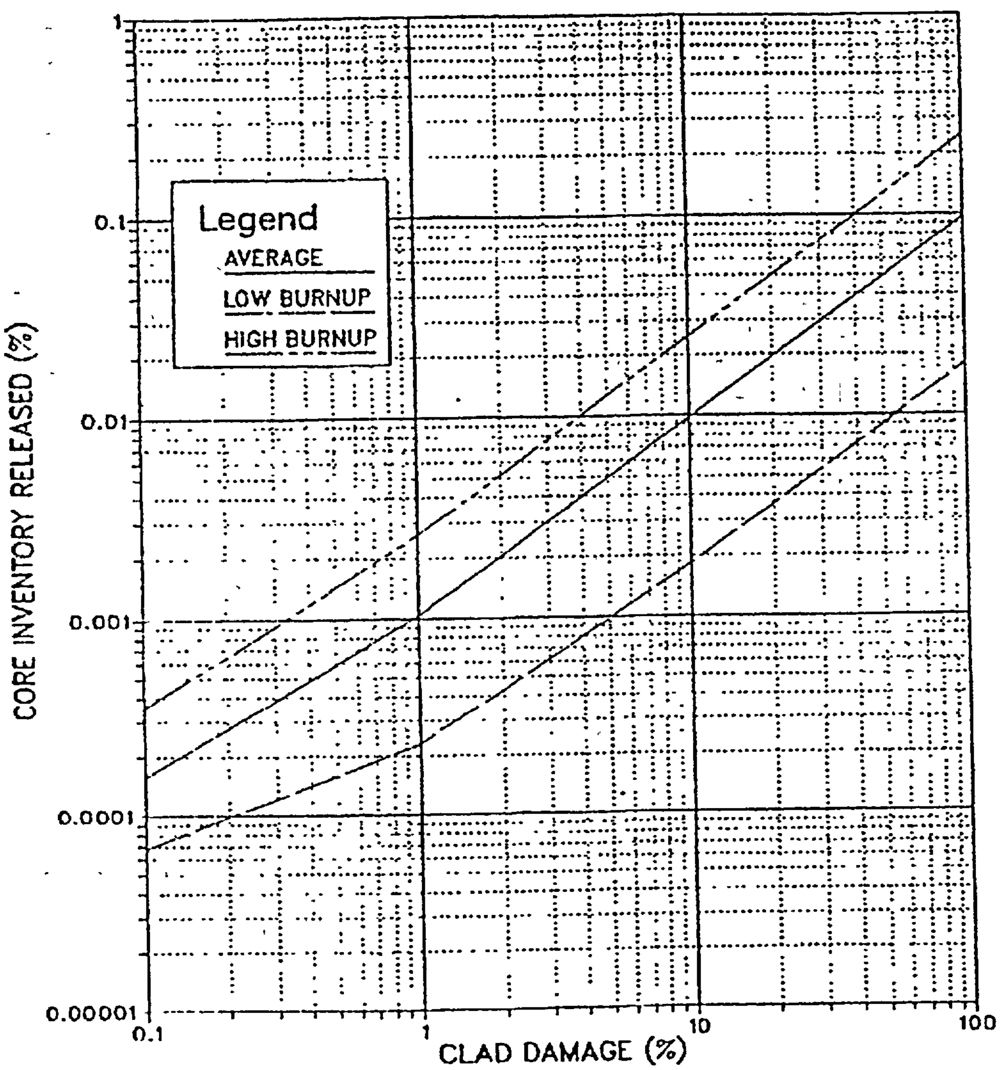

Figure 10 . Relationship of % Clad Damage With % Core Inventory Released of I-135

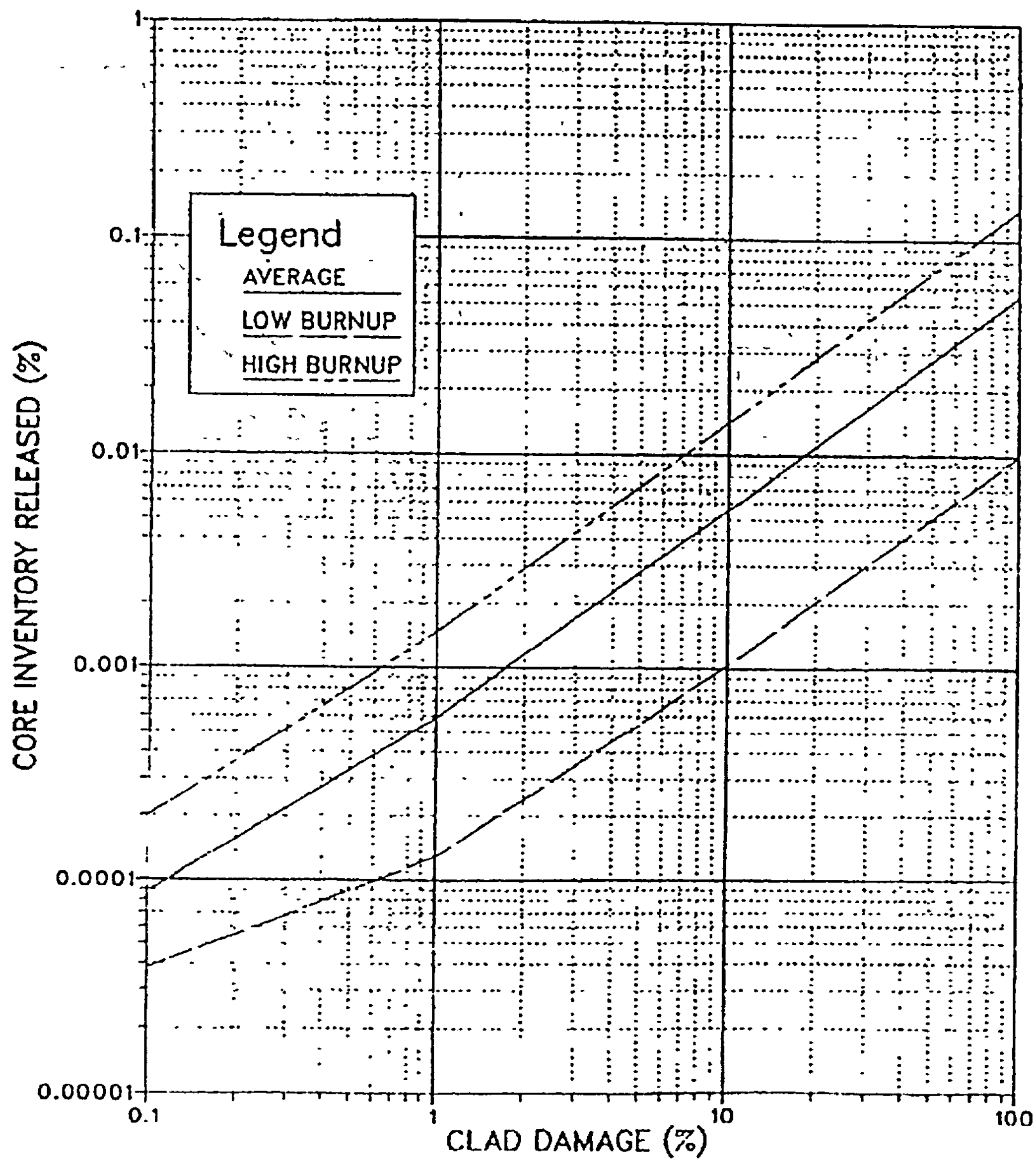

Figure 11 Relationship of % Fuel Over Temperature With % Core Inventory Released of XE, KR, I, or CS

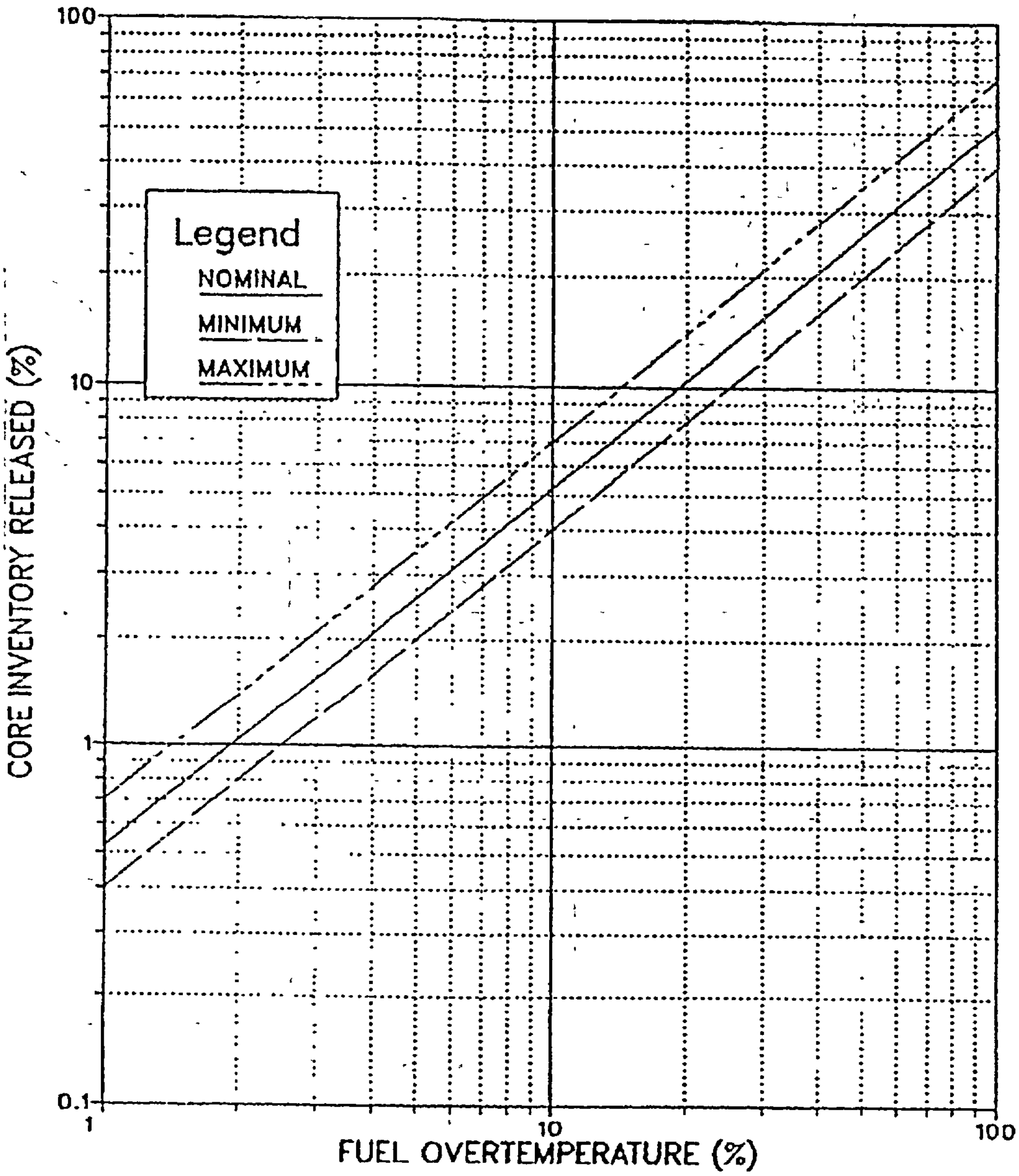

Figure 12 Relationship of % Fuel Over Temperature With % Core Inventory Released of BA or SR

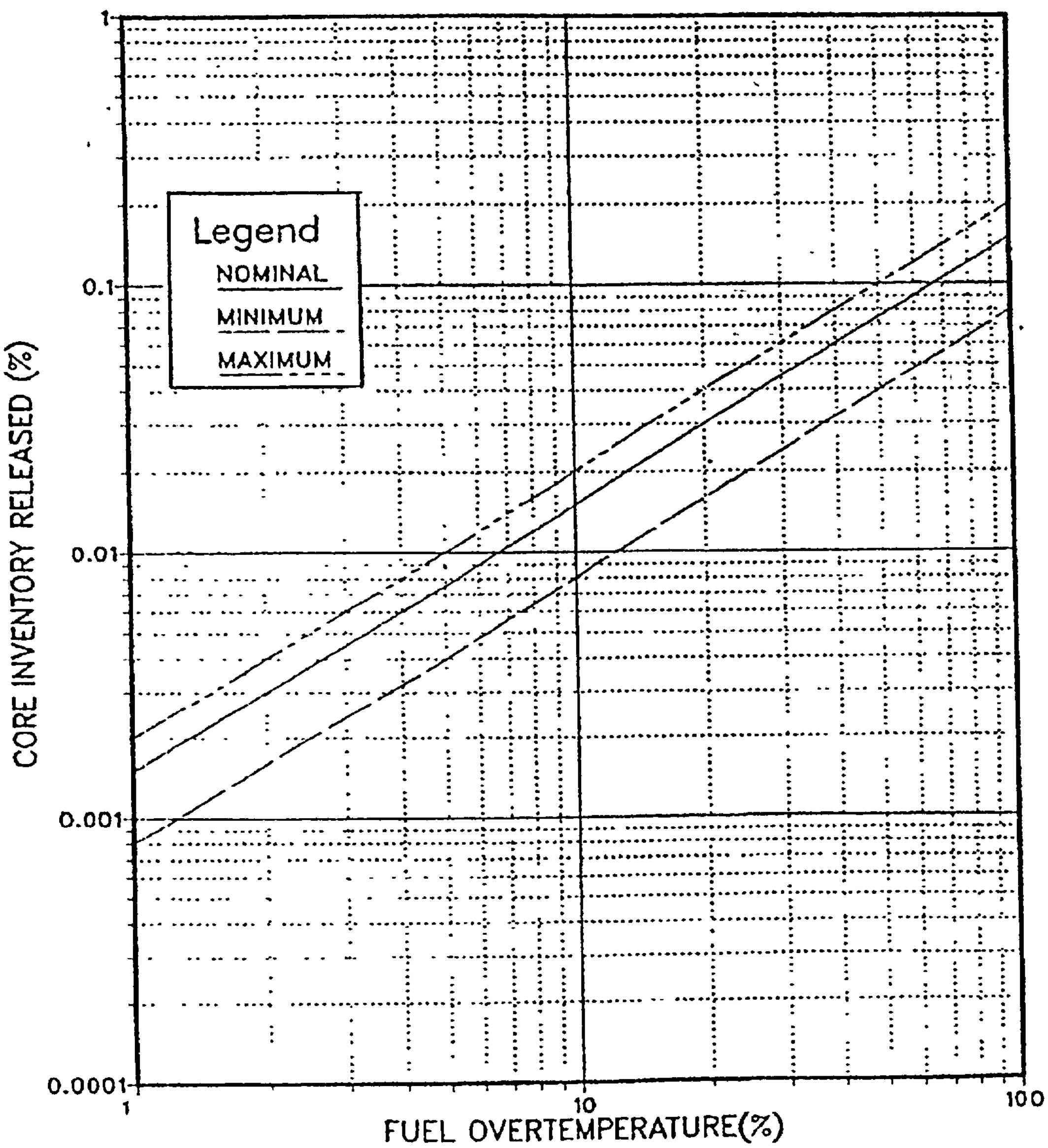

Figure 13 Relationship of % Fuel Melt With % Core Inventory Released of XE, KR, I, CS or TE

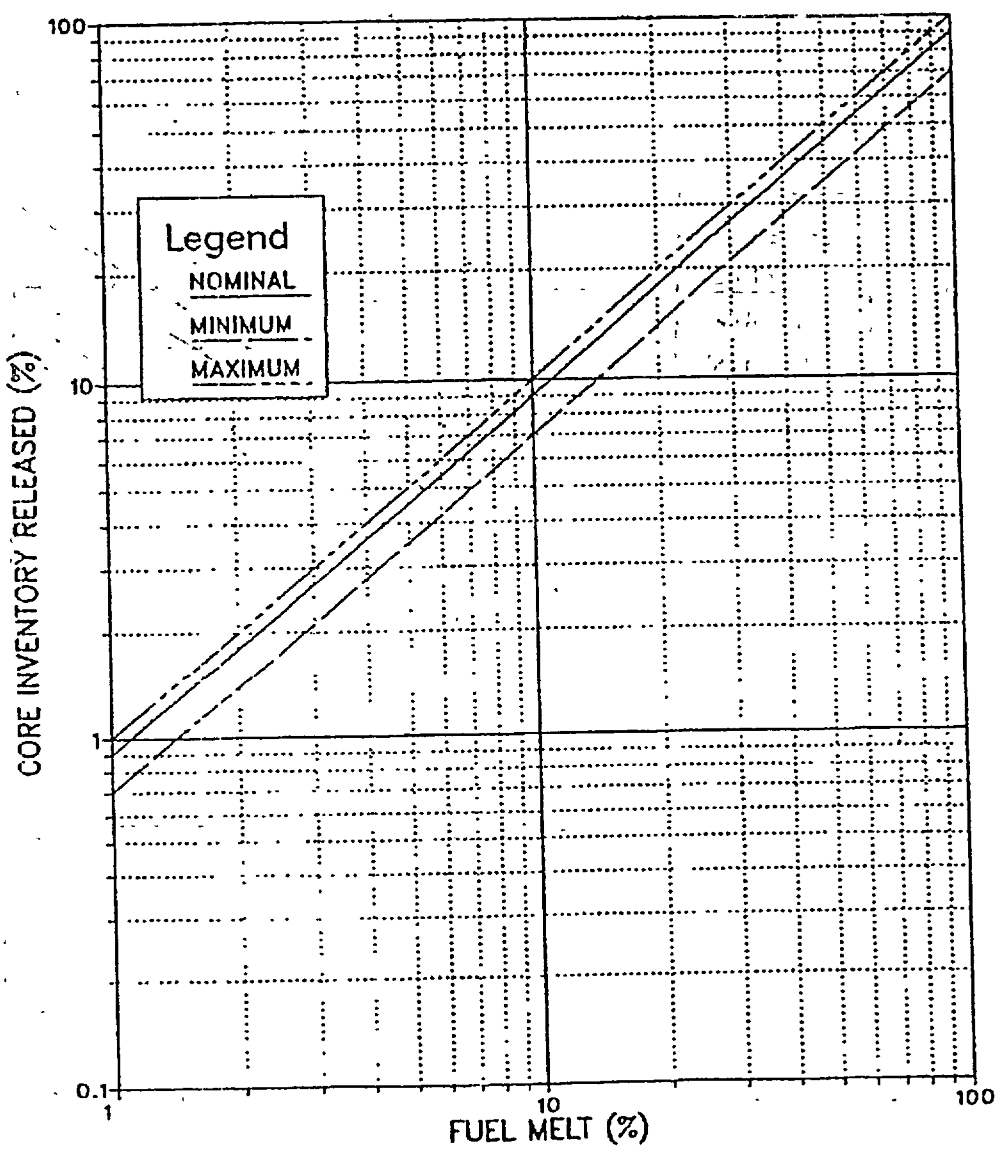

Figure 14 Relationship of % Fuel Melt With % Core Inventory Released of BA or SR

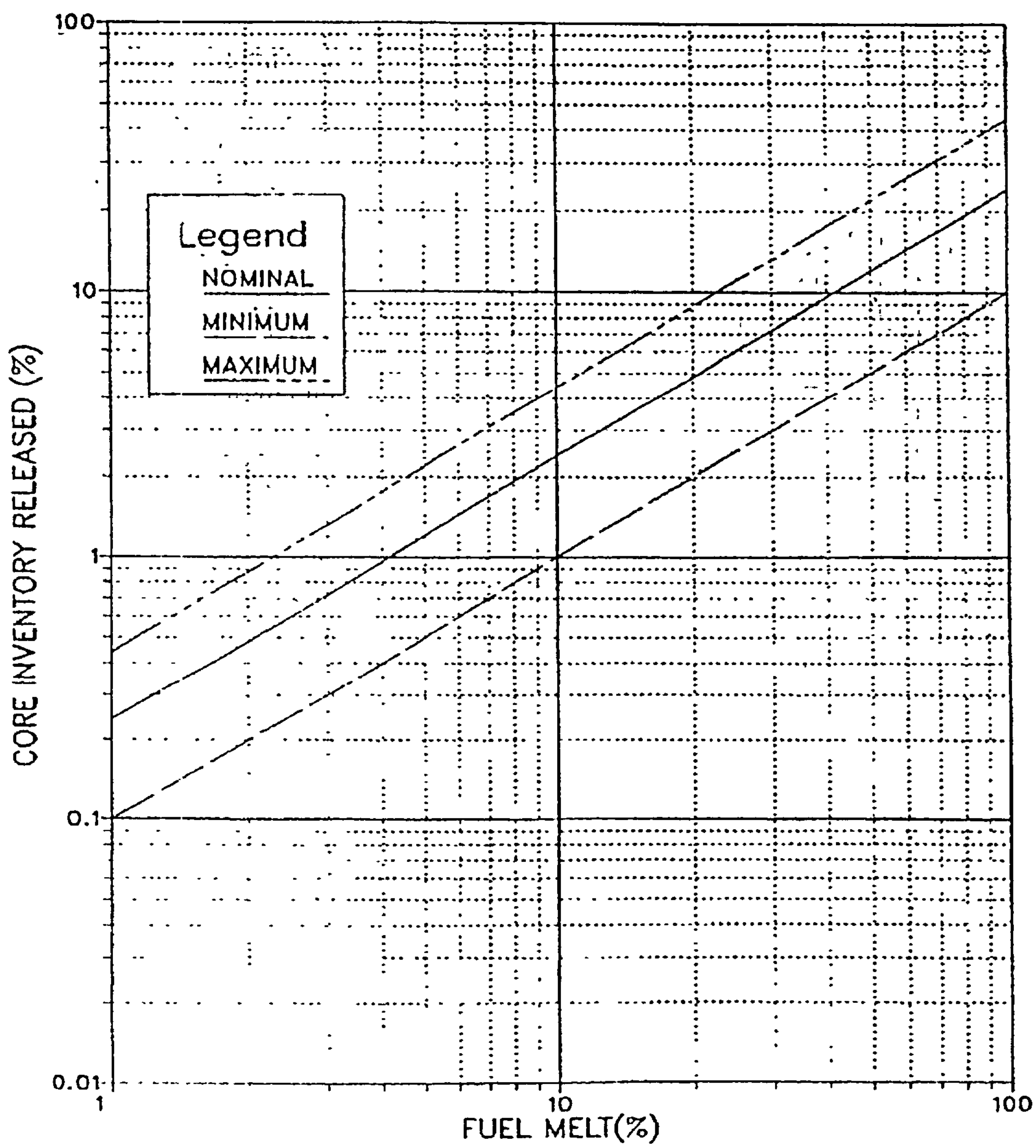

Figure 15 Relationship of % Fuel Melt With % Core Inventory Released of PR

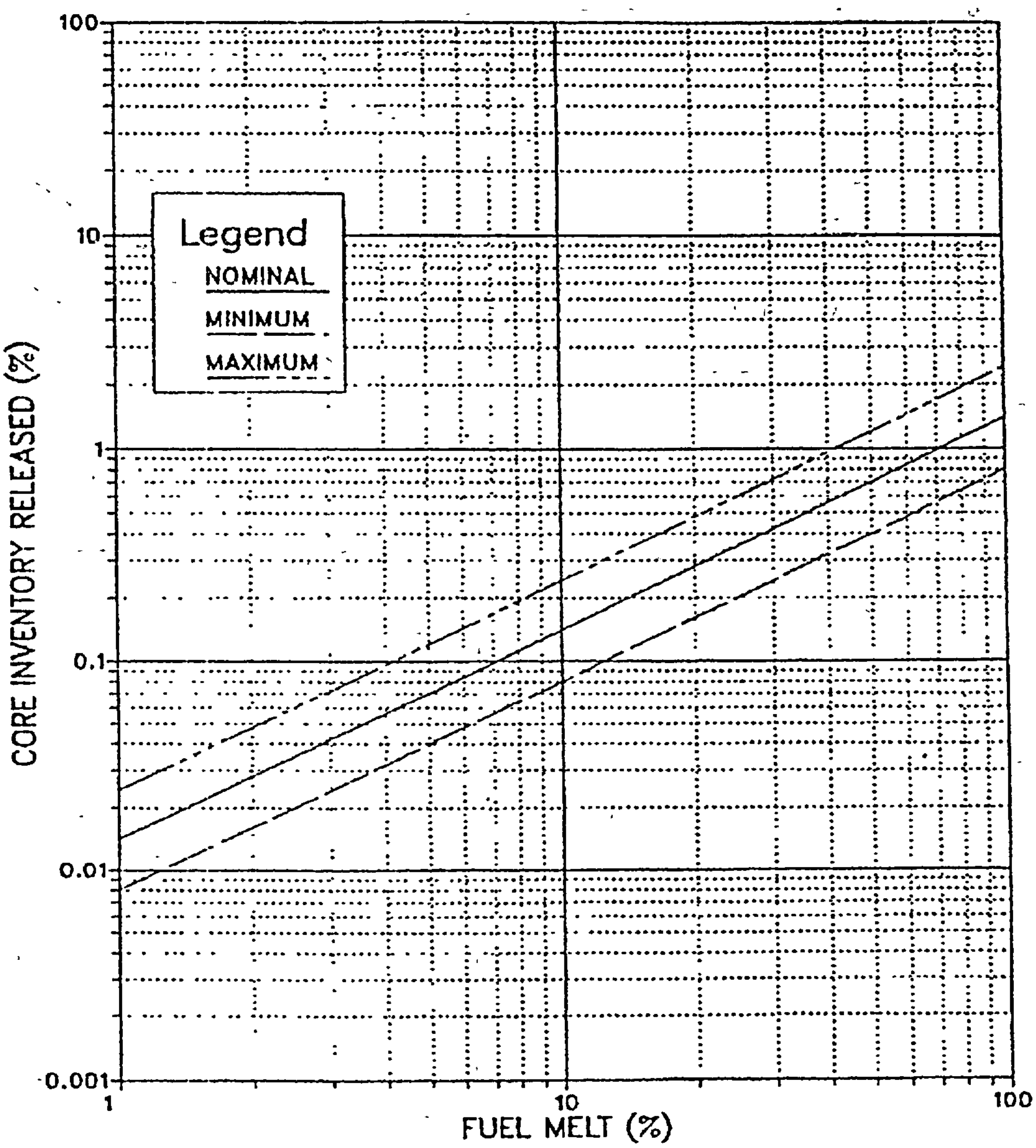

Figure 16 Containment Hydrogen Concentration Based on Zirconium Water Reaction

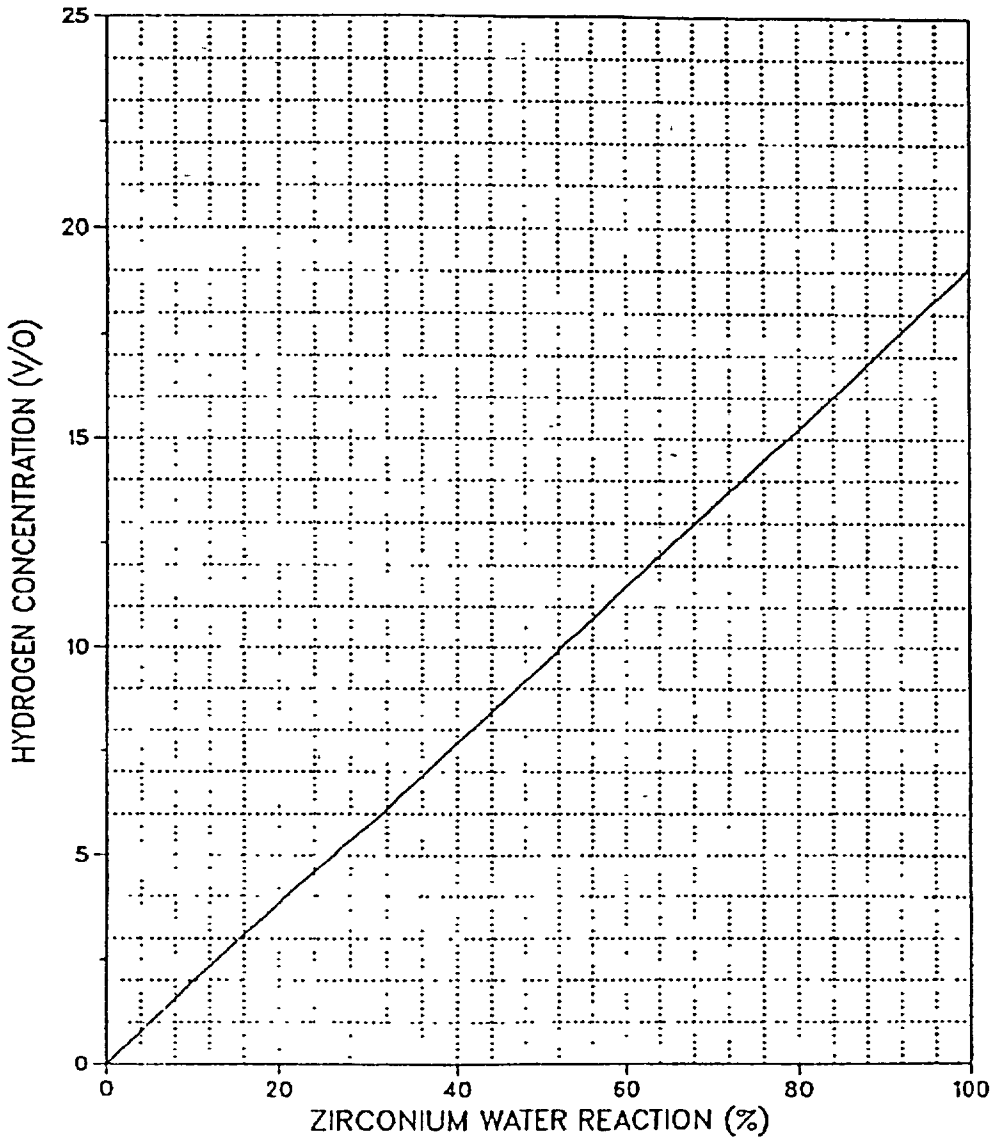

Figure 17 Percent Noble Gases in Containment

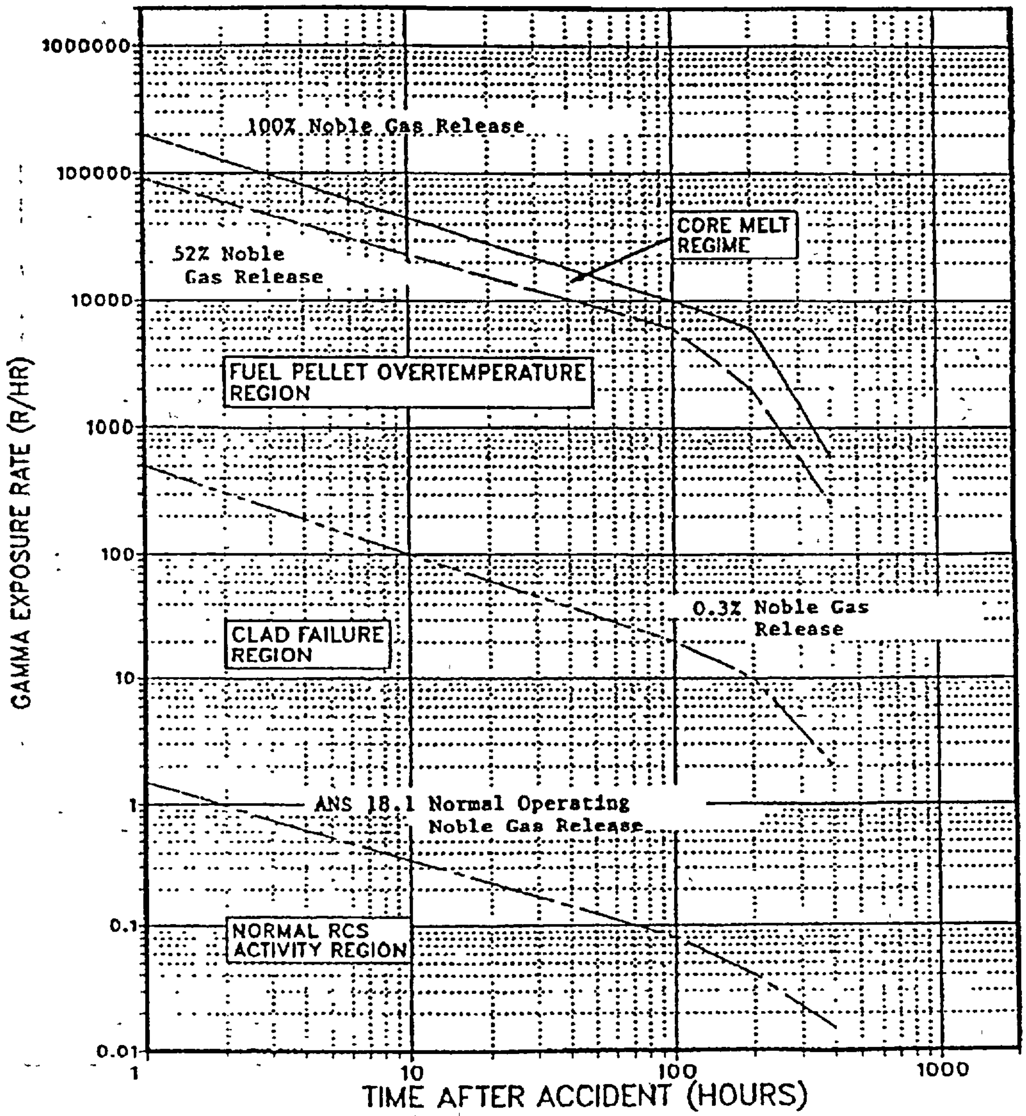

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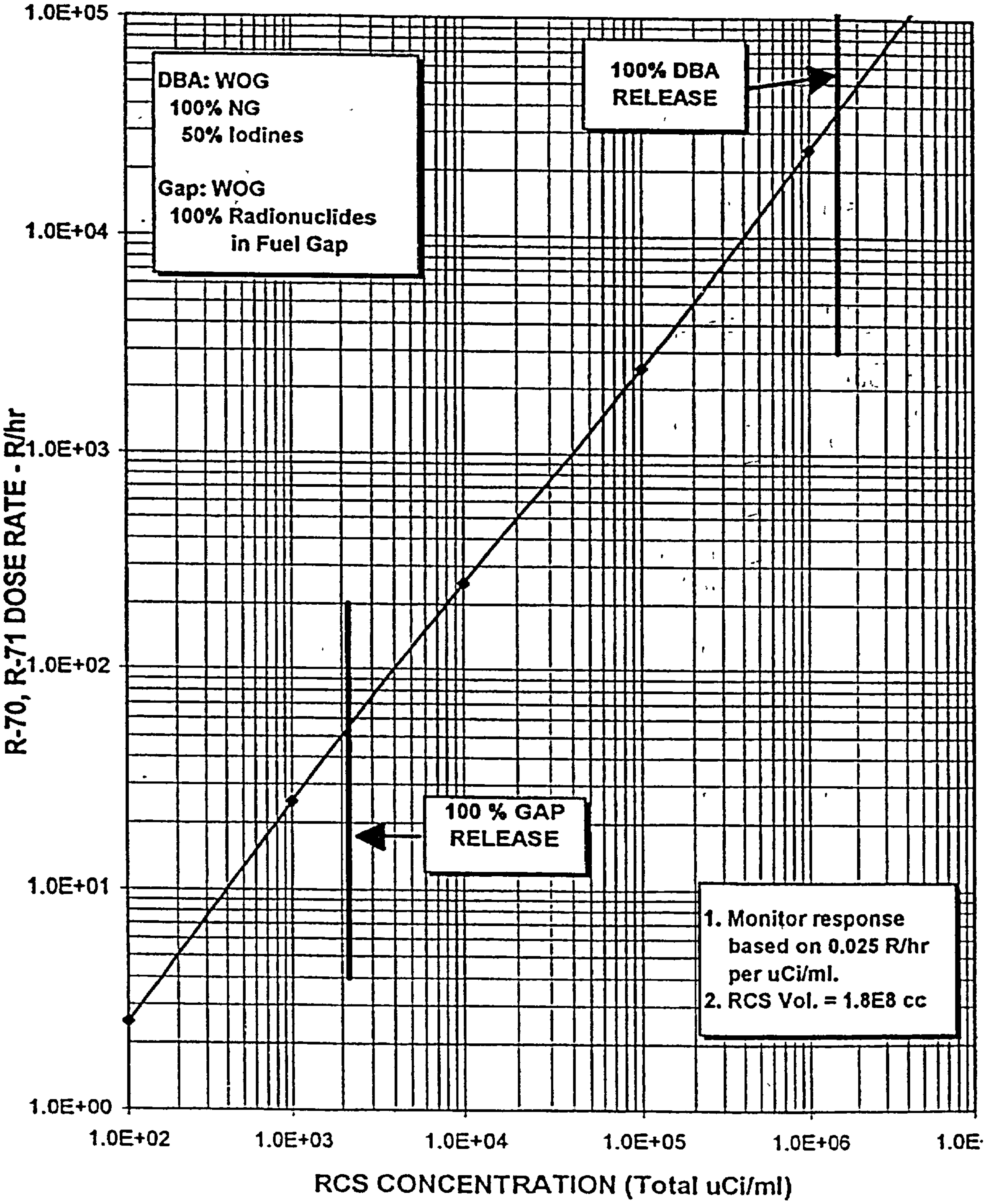
Figure 18 RCS Dose Rate vs. RCS Activity Concentrations 1 Hour After Shutdown



Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

1. R-48/R-49 & R-70/R-71Thermally Induced Errors

R-48/49 or R-70/71 signals may experience errors during the first 4 hours after a DBA LOCA. Industry testing of high range radiation monitor (HRRM) systems has revealed that signal errors or the loss of signal are the result of thermally induced current (TIC) and/or moisture intrusion into the coaxial connectors. Based on the EPRI Plant Support Engineering study, worst case estimated errors are summarized below:

Time After Postulated DBA	Estimated Errors in Readings
~ 1 minute	> 3000 R/hr
~ 2 minutes	~ 100 R/hr
~ 8 minutes	~ 15 R/hr
~ 2 to 4 hours	~ - 9 R/hr
> 4 hours	No Effect from TIC

More background information concerning thermally induced current in high range radiation monitors is described in Section III.

Please note that errors in the range of ±10 R/hr one hour after a postulated DBA has minimal effect on our assessment of fission product release to containment when we are considering magnitudes of > 100 R/hr reading to be confirmation of fission product released to containment.

2. Background on Thermally Induced Current (TIC) on High Radiation Monitors

<u>Background</u>

Excerpts from: PINGP Response to High Range Radiation Monitor Cable Study: Phase II, Report No. TR-112582 November 2000.

Transient signal errors have been observed in industry testing of the high range radiation monitor (HRRM) system. At PINGP, these are plant radiation monitors RE-48 and RE-49. The investigation into this issue revealed that signal errors or the loss of signal are the result of thermally induced currents (TICs) and/or moisture intrusion into the coaxial connectors. Information Notices, IN 97-45 and IN 97-45 Supplement 1, were issued by the NRC to alert licensees to these potential issues.

Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

EPRI Plant Support Engineering (PSE) was tasked to study the significance of this issue, which resulted in the issuance of TR-112582, "High Range Radiation Monitor Cable Study: Phase II". This study was focused on the thermally induced current phenomena since moisture intrusion issues are well understood within the industry and have more generic applications. Phase I of the EPRI study confirmed that TIC existed and was significant under thermal transients. Phase II of the study identified the sources of the TIC and developed a mathematical model for cable responses to thermal transients.

Study Results and Analysis

Using the developed profiles in the Phase II study, the actual amplitude, duration, and sign of HRRM signal errors to be expected could be determined. From this data, PINGP was able to ascertain the approximate expected signal error for the HRRMs during the postulated DBA. The expected radiation readings due to the TIC phenomena, based upon the worst case cable length, are as follows:

50 seconds	-3872 R/hr
100 seconds	88 R/hr
500 seconds	13.2 R/hr
8000-15000 seconds	-8.8 R/hr
>15000 seconds	no effect from TIC

From 8000 seconds until 15000 seconds, the HRRMs could provide a "fail" alarm, based on the required "keep alive" signal current of 1E-11 amps since the current may drop to – 8.8E-11 amps. It should be noted that any significant radiation releases would drive the current back up and the HRRMs would function properly, except for the-8.8 R/hr error that may be present. After 15000 seconds (4.1 hours), there would be no TIC effects on the HRRMs.

The installed HRRM cable at PINGP is the worst case tested cable, Rockbestos RSS-6-104, and is in greater lengths than were tested, 130 feet tested vs. 290 feet installed (worst case). Other variables that could significantly effect the TIC phenomena are, 1) the tested cable was not installed within conduit whereas the PINGP cable runs are installed entirely within conduit, 2) the temperature differential of the test samples, 100 degc, is greater than the temperature differential from the PINGP accident profile, 68 degc, 3) the EPRI mathematical model was developed based on hypothetical LOCA profiles, which are more severe than the PINGP LOCA profile, and 4) consideration regarding whether the test methodology of immersion of the test samples into a ice bath and then to a boiling water plunge is representative of what the cable would experience during an actual transient.

Attachment 1 Thermally Induced Current Errors in Containment Radiation Monitors

PINGP Response to HRRM Signal Error

During the initial phase of any postulated accident, it would not expected to see indication of actual fuel damage for the first 10-15 minutes. If indeed the alarms would come in for RE-48 and RE 49, Operations would be occupied with accident mitigation and monitoring tasks during this time period and this alarm, even though acknowledged, would be ignored during this period. Other parameters would be available for alarm validation, i.e., core exit temperatures, RVLIS, radiation monitors located in the Auxiliary Building, etc. Due to the nature of the TIC phenomenon, the radiation level readings, even if the alarms have come in, would be decreasing. Again, this is validation of an erroneous signal and not actual core damage.

For emergency plan response and possible SAMG considerations, the TIC phenomenon would no longer be affecting the radiation monitors and/or due to the earlier alarms and decreasing readings that were noted, it would be confirmed that no fuel damage had occurred and these were indeed erroneous readings. A general site emergency alarm would be activated at 1000 R/hr, but as cited previously, this is well after the expected error signal has been significantly reduced. Other variables would be available to verify possible fuel damage and any possible actions required within the emergency plan procedures would not occur until after the TIC phenomena has either passed or has been verified to be erroneous.

F	PERSONNEL AND EQUIPMENT MONITORING AND DECONTAMINATION	NUMBER: F3-19
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REFERENCE USE

- ***Procedure segments may be performed from memory.***
- ***Use the procedure to verify segments are complete.***
- ***Mark off steps within segment before continuing.***
- ***Procedure should be available at the work location.***

O.C. REVIEW DATE:	OWNER:	Effective Date
011303 SC	M. Werner	1-30-03

F	PERSONNEL AND EQUIPMENT MONITORING AND DECONTAMINATION	NUMBER: F3-19
		REV: 8

1.0 PURPOSE

This procedure provides the guidance for contamination monitoring, contamination control, and decontamination procedures for personnel and equipment.

2.0 APPLICABILITY

This instruction **SHALL** apply to all Emergency Directors (ED) and all members of the Radiation Protection Group (RPG).

3.0 PRECAUTIONS

3.1 All personnel decontamination should be supervised by the RPG.

3.2 The safety of personnel **SHALL** take precedence over the monitoring of personnel and vehicles for radiation/contamination control purposes. Monitoring of personnel and/or vehicles **SHALL** be terminated (or not implemented) if such monitoring is known or suspected to be increasing the hazard to personnel during evacuation.

3.3 If any personnel are suspected to have received a biologically significant dose (dose exceeds twice the NRC Annual 10CFR20 Occupational Dose Limits), refer directly to the F3-12, Emergency Exposure Control.

4.0 RESPONSIBILITIES

4.1 The RPG has the responsibility to monitor personnel and equipment to determine if contaminated. When personnel or equipment is found contaminated, the RPG has the responsibility to document contamination levels and to coordinate the decontamination of personnel or equipment.

4.2 The Radiological Emergency Coordinator (REC) has the responsibility to authorize use of elevated contamination levels as listed in Attachment 1 under Emergency Guidelines.

4.3 The ED has the overall responsibility to ensure that radioactive contamination monitoring, control, and decontamination is being conducted throughout the emergency.

5.0 DISCUSSION

During emergency conditions, large areas of elevated surface contamination levels are probable within the plant boundaries. The REC should evaluate the contamination levels and determine if it would be beneficial to raise the contamination limits to the elevated guidelines in Attachment 1. The RPG should then control entry into the plant in accordance with these guidelines and monitor personnel and equipment exiting the plant per these guidelines. Decontamination of personnel and equipment to levels below these guidelines should be performed per applicable decontamination procedures.

6.0 PREREQUISITES

The Prairie Island Nuclear Generating Plant has declared an Emergency classification.

7.0 PROCEDURE

7.1 The RPG is responsible for contamination monitoring, control and decontamination.

7.1.1 All attempts should be made to maintain contamination levels below the normal guidelines, as per Attachment 1.

7.1.2 During emergency conditions, elevated contamination limits may be authorized by the REC, as per Attachment 1.

7.2 Personnel Monitoring and Decontamination

7.2.1 **Monitor** personnel who evacuated directly out of the Radiological Controlled Area first.

7.2.2 **Survey** and **document** results on PINGP 985, Personnel and Vehicle Survey Log.

7.2.3 <u>IF</u> contamination is found, <u>THEN</u> **initiate** PINGP 915, Whole Body Survey Form.

7.2.4 **Segregate** monitored personnel into 3 groups using the following criteria.

Highly Contaminated	≥ 5000 CCPM
Contaminated	≥ 100 CCPM
NOT Contaminated	< 100 CCPM

7.2.5 **Decontaminate** Highly Contaminated personnel ($\geq$ 5000 CCPM) first, followed by the Contaminated personnel ($\geq$ 100 CCPM).

7.2.6 IF personnel contamination is found around the individual's mouth and nose THEN **obtain** a nasal smear.

7.2.7 IF the results are $\geq$ 100 CCPM, THEN **indicate** Bioassay Required on PINGP 915, (See RPIP 1126, Contamination Monitor Alarm Response and Personnel Decontamination).

7.2.8 **Attempt** to reduce any contamination detected on an individual in accordance with RPIP 1126.

A. IF dose rates allow personnel habitability, THEN **use** Decon Showers at Access Control.

B. IF dose rates allow personnel habitability, THEN **use** old Admin Building shower facilities.

C. **Use** EOF Decon Shower at Prairie Island Training Center.

7.2.9 IF the Normal Guidelines are NOT achieved, THEN **refer** to the REC about using the Emergency Guidelines in Attachment 1.

7.2.10 IF contamination is coincident with injury, THEN **follow** procedures outlined in F4, Medical Support and Casualty Care.

7.2.11 **Decontaminate** personal clothing and shoes to Normal Guidelines.

7.2.12 IF Normal Guidelines CANNOT be obtained after reasonable efforts, THEN **dispose** of the items as contaminated waste OR the REC may **authorize** the use of Emergency Contamination Guidelines as specified in Attachment 1.

7.3 **Coordinated** decontamination for Emergency Response personnel remaining onsite, and conducting emergency work activities.

7.4 **Vehicles Monitoring and Decontamination**

NOTE:	Vehicle monitoring and decontamination should be performed as time allows depending on evacuation urgency.

CAUTION:	MAJOR VEHICLE CONTAMINATION MAY POSE A RADIATION HAZARD TO PERSONNEL CONDUCTING SURVEYS AND APPROPRIATE PRECAUTIONS SHOULD BE TAKEN OR SURVEYING SUSPENDED UNTIL LATER.

7.4.1 IF contamination is expected, THEN **survey** vehicles prior to departing from the Assembly Area.

7.4.2 **Survey** each vehicle and **document** on PINGP 985.

A. **See** Attachment 1 for Guidelines.

B. **Survey** exterior and interior surfaces for fix contamination, paying special attention to the air filter, tires, and radiators.

C. IF fuel damage is suspected, THEN **smear** areas where contamination is found with a cloth smear and **save** for alpha counting.

D. **Smear** exterior of the Vehicle using two (2) masslins each covering 5 sq. ft. from, hood, roof, trunk or pick-up bed.

E. IF smears are ≥ 100 CCPM, THEN **log** the vehicle as contaminated.

7.4.3 **Initiate** PINGP 986, Vehicle Survey Form, for vehicles found contaminated.

7.4.4 **Tape** the PINGP 986 to the inside of the windshield with any saved smears.

7.4.5 **Hold** Contaminated vehicles in a designated area for later decontamination.

7.4.6 IF major vehicle contamination exists, THEN **evacuate** personnel as quickly as possible using vehicles that meet the Guidelines of Attachment 1. Outside assistance may be requested as necessary.

7.5 Upon termination of emergency condition, **survey** the exterior and interior surfaces of the vehicles. Paying special attention to the air filter, tires, radiators, etc. Contamination levels **SHALL** be returned to the Normal Guidelines, using approved decontamination procedures as outlined in F2, Radiation Safety, and D-13, Decontamination.

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Attachment 1 Contamination Limits

CONTAMINATION LIMITS		
	NORMAL GUIDELINES	EMERGENCY GUIDELINES
REMOVABLE, LOOSE SURFACE DPM/100 cm^2 Bγ α	 100 DPM/100 cm^2 10 DPM/100 cm^2	 5000 DPM/100 cm^2 500 DPM/100 cm^2
FIXED	100 CPM	500 CPM

Based on Manual of Protective Action Guides and Protective Actions for Nuclear Accidents, EPA 400-R-92-001, May 1992, Table 7-7. Frisker response: 1mR/hr ≈ 5000 CPM Cs 137.

1. Guidelines are based on using pancake probe.
2. Bγ Portable survey instruments are located in all Emergency Centers and at both Assembly Points.
3. α Portable survey instrument is located in the Hotcell Emergency locker.

PINGP 1300, Rev. 5
Retention: 6 years
Document Type: 1.190

TCN # 2003- 0080
Effective Date: 1-28-03
Expiration Date: ———
Page 1 of 3

TEMPORARY CHANGE NOTICE
(5AWI 1.5.10)

I. CHANGE REQUEST

Procedure #: F3-23.1 Rev: 12 Project ID #:

WO #:

Title: Emergency Hotcell procedure

Description of Change: (e.g. page #, step #, summary/reason) Pages 6 & 7, remove references to incorrect procedures

Is this an OC reviewed procedure/critical WO? Yes ☒ No ☐

Is a permanent procedure change needed? Yes ☒ No ☐ If Yes, submit PINGP 436

Originator: Marotz Employee #: MRM05 Date: 1-28-03

II. INITIAL REVIEWS

A. Is this a change in intent? Yes ☐ No ☒ If No, go to II.B

1. OC Review Initial/Date (if procedure is OC reviewed/critical WO):
2. Procedure Approver (Work Supv for WO):

B. 10 CFR 50.59 Review

1. Is this change or procedure exempt from 10CFR 50.59 screening per 5AWI 3.3.5, App. A or B? Yes ☒ No ☐ If Yes, go to II.C
2. Complete 50.59 screening per 5AWI 3.3.5/PINGP 1229. Screening/Evaluation #

C. Does this change affect Special Reviews? Yes ☐ No ☒ If Yes, document Special Review(s) in CHAMPS

D. Does this change affect plant operation or Tech Specs? Yes ☐ No ☒ If Yes and work is in progress, SS notified by ______ (initial)

E. Do controlled copies need to be updated? Yes ☒ No ☐ If Yes, attach marked-up procedure pages to white copy

F. All master/working copies updated? Yes ☐ NA ☒

G. Is training on temporary change needed? Yes ☐ No ☒ If Yes, PINGP 1268 or PINGP 1224 (Ops Only) issued

III. REVIEW AND APPROVAL

Reviewer: [signature] Date: 1-28-03
(Unit Management Staff)

Approver: [signature] Date: 1-28-03
(Unit Management Staff; SRO for OC reviewed procedures/critical WO)

Forward white copy to Procedure Control

IV. POST REVIEWS (required for OC reviewed procedures/critical WO; leave blank if reviewed/approved in II A)

OC Review Initial/Date: Procedure Approver: Date:

V. TEMPORARY CHANGE DELETION (leave blank if controlled copies not updated, refer to II.E)

A. Reviewer: Date:
(Unit Management Staff)

B. Approver: Date:
(Unit Management Staff, SRO for OC reviewed procedures/critical WO)

White Copy - Procedure Control Yellow Copy - Attach to procedure

7.3 Gamma Analysis Preparation

7.3.1 Pipet 10 ml of diluted coolant sample from the 1 L volumetric to a 10 ml vial.

NOTE: Sample should be diluted to give a contact reading of under 1 millirem/hr contact. The diluted sample should NOT exceed 25 millirem/hr contact.

7.3.2 Verify that the indicated dose rate on the 10 ml vial is capable of being counted on extended geometry in EOF Countroom.

7.3.3 Label the vial with the sample point, date, time, and dilution factor to the sample prior to sending to EOF Countroom.

7.3.4 Place the 10 ml vial in the shielded carrier for transport to the EOF Countroom.

7.3.5 WHEN radioactive gas, charcoal, or particulate samples are received, THEN ensure all samples are labeled with date and time of sample, sample point, sample volume and/or correction factor, and flow rate.

7.3.6 Store all samples in the Hotcell Shielded Area until transported to the EOF Countroom.

7.4 Boron Analysis

7.4.1 Using the 1 L sample prepared in Step 7.1, Sample Preparation, analyze in accordance with ~~RPIP 3314, Boron by Ion Exclusion Chromatography~~. the appropriate Boron analysis procedure

7.4.2 Log the results on PINGP 655, Post Accident Chemical Analysis Report.

7.4.3 Dispose of all radioactive waste according to Step 7.6, Post Accident Sample Waste Storage and Disposal.

7.5 Chloride Analysis

NOTE	Chloride analysis SHALL be completed within 4 days of accident.

CAUTION	THE REACTOR COOLANT SAMPLES TAKEN IN AN ACCIDENT CONDITION HAVE THE POTENTIAL TO BE HIGHLY RADIOACTIVE. THIS MAY GIVE RISE TO DOSE RATES FAR IN EXCESS OF WHAT WOULD NORMALLY BE ENCOUNTERED. THE ION EXCHANGE COLUMNS ON THE ION CHROMATOGRAPH COULD HAVE CONTACT READINGS OF UP TO 10 R/HR.

7.5.1 Using the 100 ml sample prepared in Step 7.1, Sample Preparation analyze ~~in accordance with RPIP 3301, Anions by Ion Exchange.~~ for anions IAW the appropriate analysis procedure.

7.5.2 Log the results on PINGP 655, Post Accident Chemical Analysis Report.

7.5.3 Dispose of all radioactive waste according to Step 7.6, Post Accident Sample Waste Storage and Disposal.

7.6 Post Accident Sample Waste Storage and Disposal

NOTE	Ensure samples are labeled. "TO BE SAVED" or "TO BE DUMPED" before storage in shielded area.

7.6.1 Place all capped or covered radioactive sample waste in the Hotcell Shielded Area.

7.6.2 IF additional waste samples are added to the Hotcell Shielded Area, THEN survey the Hotcell general area radiation levels. Add additional shielding, as necessary.

7.6.3 IF making subsequent entries into Auxiliary Building, THEN return the sample waste to the Sample Room for disposal down the affected unit's Sample Hood Drain.