Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

February 3, 2000
LIC-00-0009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Subject: Transmittal of Changes to Fort Calhoun Radiological Emergency Response Plan (RERP), Emergency Plan Implementing Procedures (EPIP), and Emergency Planning Forms (EPF) Manuals

In accordance with 10 CFR 50 Appendix A Part V and 10 CFR 50.4(b)(5)(iii), please find RERP, EPIP, and EPF change packages enclosed for the Document Control Center (holder of Copy 165), and the NRC Emergency Response Coordinator (holder of Copies 154, 155, and 156).

The document update instructions and summary of changes are included on the Confirmation of Transmittal (Form EP-1) forms attached to each controlled copy change package. Please return the Confirmation of Transmittal forms by March 6, 2000.

The revised documents included in the enclosed packages are:

RERP Index, Page 1, issued 01/06/00
RERP Section J, R16, issued 01/06/00

EPIP Index, Pages 1 & 2, issued 01/13/00
EPIP-EOF-10, R10, issued 01/13/00
EPIP-TSC-8, R12, issued 01/13/00

EPIP Index, Page 1, issued 01/19/00
EPIP-TSC-8, R13, issued 01/19/00
FC-EPF Index, Pages 1 & 2, issued 01/20/00
FC-EPF-38, R5, issued 01/20/00

RERP Index, Pages 1 & 2, issued 01/27/00
RERP, R14, issued 01/27/00
RERP-Section L, R11, issued 01/27/00

Please contact me if you have any questions regarding the enclosed changes.

Sincerely,

S. K. Gambhir
Division Manager
Nuclear Operations

SKG/jmh

Enclosures

c: T. H. Andrews, Emergency Response Coordinator (3 sets)
L. R. Wharton, NRC Project Manager, (w/o enclosures)
W. C. Walker, NRC Senior Resident Inspector (w/o enclosures)
Winston & Strawn (w/o enclosures)

FORM EP-1
Rev. 6 (04/29/99)

OMAHA PUBLIC POWER DISTRICT

Confirmation of Transmittal for Emergency Planning Documents/Information

- [x] Radiological Emergency Response Plan (RERP)
- [ ] Emergency Plan Implementing Procedures (EPIP)
- [ ] Emergency Planning Forms (EPF)
- [ ] Emergency Planning Department Manual (EPDM)
- [ ] Other Emergency Planning Document(s)/ Information

Transmitted to:

Date: 2-3-00

Name:	Copy No:
Document Control Desk	165
Tom Andrews	154
Tom Andrews	155
Tom Andrews	156

The following document(s) / information is forwarded for your manual:

REMOVE SECTION
RERP Index page 1 of 2 dated 11/30/99
RERP Section J R15 issued 04/22/97

INSERT SECTION
RERP Index page 1 of 2 dated 01/06/00
RERP Section J R16 issued 01/06/00

Summary of Changes:
Revise population numbers per new Evacuation Time Estimate, realigned EPZ, insert new maps/charts and New Sub Areas.

Supervisor - Emergency Planning

I hereby acknowledge receipt of the above documents/information and have included them in my assigned manuals.

Signature: ____________________ **Date:** ____________

Please sign above and return by **03/06/00** to:

Karma Boone
Fort Calhoun Station, FC-2-1
Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

NOTE: If the document(s)/information contained in this transmittal is no longer requested or needed by the recipient, or has been transferred to another individuals, please fill out the information below.

- [ ] Document(s)/Information No Longer Requested/Needed
- [ ] Document(s)/Information Transferred to:

Name: ____________________ Mailing Address: ____________________

RADIOLOGICAL EMERGENCY RESPONSE PLAN INDEX
RERP

PROCEDURE NUMBER	TITLE	REVISION/DATE
RERP	Definitions and Abbreviations	R13 02-25-98
RERP-SECTION A	Assignment of Organizational Responsibility (Organizational Control)	R11 02-27-97a
RERP-SECTION B	Organizational Control of Emergencies	R23 09-30-97
RERP-SECTION C	Emergency Response Support and Resources	R9 09-30-98
RERP-SECTION D	Emergency Classification System	R9 04-29-97a
RERP-SECTION E	Notification Methods and Procedures	R22 10-20-98
RERP-SECTION F	Emergency Communications	R14 09-09-99
RERP-SECTION G	Public Education and Information	R10 03-11-97a
RERP-SECTION H	Emergency Facilities and Equipment	R27 10-10-97
RERP-SECTION I	Accident Assessment	R11 09-02-99
RERP-SECTION J	Protective Response	R16 01-06-00
RERP-SECTION K	Radiological Exposure Control	R8 12-03-97
RERP-SECTION L	Medical and Public Health Support	R10 09-18-97
RERP-SECTION M	Recovery and ReEntry Planning and Post Accident Operations	R14 03-11-97a
RERP-SECTION N	Exercises and Drills	R12 10-28-99
RERP-SECTION O	Radiological Emergency Response Training	R13 09-23-97a
RERP-SECTION P	Responsibility for the Planning Effort: Development, Periodic Review and Distribution	R10 10-23-97

Distribution Authorized

This procedure does not contain any proprietary information, or such information has been censored. This issue may be released to the public document room. Proprietary information includes personnel names, company phone numbers, and any information which could impede emergency response.

Fort Calhoun Station
Unit No. 1

RERP SECTION J

RADIOLOGICAL EMERGENCY RESPONSE PLAN

Title: PROTECTIVE RESPONSE

FC-68 Number: DCR 10973

Reason for Change: Revise population numbers per new Evacuation Time Estimate and realigned EPZ. Insert new maps and charts, New Sub Areas. Minor editorial changes.

Initiator: Mark Reller

Preparer: Mark Reller

RADIOLOGICAL EMERGENCY RESPONSE PLAN
RERP-J

PROTECTIVE RESPONSE

1. PROTECTIVE RESPONSE FOR ONSITE PERSONNEL

1.1 Notification

Onsite personnel are notified of a nuclear emergency via the emergency alarm. This alarm is identified by an intermittent howl and is distinguished from the fire alarm which is a continuous howl. Once the emergency alarm is sounded, the command and control position will give the emergency classification, with other pertinent information, using the intra-plant communication system (Gaitronics). If the owner-controlled area is to be evacuated, personnel will be notified by: 1) Gaitronics System, 2) Administration and Training Building paging systems, 3) Security Personnel, and/or 4) Alert Notification System, if used.

1.2 Evacuation

If the emergency requires Protected Area evacuation, personnel not assigned specific emergency duties will normally proceed to the Administration Building. In the event the site must be evacuated, all non-essential personnel, will be given instructions by the command and control position. One of the two routes shown in Figures J-1 and J-2 will be used to proceed to the North Omaha Power Station.

Approximately 600 persons might be evacuated during normal work hours and operation; approximately 900 persons might be evacuated during a major outage. During normal operating off-shift hours, no evacuation of onsite individuals is expected. Both OPPD and personal vehicles are used for site evacuation transportation. Agreements with the State of Nebraska and specifically the State Patrol guarantee professional handling and control of traffic. Normal travel time to North Omaha Station is 25 minutes at an average speed of 40 mph. RP personnel reporting to the EOF will coordinate personnel/vehicle monitoring and decontamination activities, if required.

Security and RP personnel inspect the owner controlled area after a site evacuation has taken place. If any persons other than emergency workers are in the owner controlled area during or after site evacuation, they will be given specific directions and/or escorted off-site.

1.3 Security and Accountability

1.3.1 Security

The security program is designed to deter, delay and detect an intruder. The Security Area of the plant site is enclosed by an eight foot security fence topped by three strands of barbed wire. All gates to the fence are normally kept locked. An inner perimeter consists of personnel doors, roof hatches, and overhead doors equipped with magnetic alarm switches.

Personnel assigned by the Site Director to enter the plant must pass through the main gate which is guarded. It is extremely unlikely that any unauthorized person would be able to enter the site undetected even during an emergency condition.

1.3.2 Accountability

If accountability of onsite personnel is necessary, the onsite command and control position will notify personnel onsite by announcements on the Gaitronics System, and by sounding the Emergency Alarm (if required). At the completion of the notification(s), the accountability process begins, to be completed within 30 minutes.

Accountability is a process taking place in several areas:

A. Accountability of personnel reporting to the Control Room, TSC, or OSC for emergency response is performed by those personnel placing accountability cards in the designated container. A security officer will retrieve the cards and the cards will be compared with a list of personnel that the security computer indicates as still being inside the protected area.

B. Accountability of personnel not able to evacuate for whatever reason, will be performed by placing accountability cards in designated containers as explained above.

C. Accountability of security force personnel will be accomplished using established security procedures.

D. Once initial accountability is complete, the command and control position, will be notified of the results.

1.3.2 E. Accountability is maintained by the use of rosters at the Control Room, OSC and TSC. Persons must sign in and out as they enter and leave. These rosters will be compared to a list of personnel who accessed the protected area whenever necessary. Continuous accountability of security personnel is accomplished using established security procedures.

1.4 Protective Measures

1.4.1 It is the policy of OPPD to keep personnel radiation exposure within the NRC and State regulations, and, beyond that, to keep it As Low As Reasonably Achievable (ALARA). Every effort will be made to keep their exposures within the limits of 10 CFR 20.

1.4.2 Personnel monitoring devices are required for all personnel meeting the conditions specified in 10 CFR 20 Section 20.1502, Technical Specifications Section 5.11 and in Radiation Protection Procedures. During emergency conditions, implementing procedure EPIP-EOF-11 will be utilized.

Dosimeters and TLD's are typically located in each of the emergency lockers in the Control Room, EOF, OSC and the TSC. Additional dosimeters and TLD's may be obtained from the dosimetry group.

1.4.3 Clothing

Protective clothing is a normal use item utilizing both washable and disposables. For entry into affected areas, the OSC has approximately 50 complete sets of protective clothing available. The Control Room has approximately 12 complete sets available. Additional sets are available at the Radiation Control Point. Approximately 2000 sets are maintained in the laundering cycle, and a large supply of washable and disposable coveralls is maintained in the warehouse and RP storage areas. Water-proof protective clothing is also a standard stock item.

1.4.4 Respiratory Protection

Respiratory protective devices may be required in any situation arising from plant operations where an airborne radioactivity condition is potential or existent. In such cases, the air will be monitored and the necessary protective devices specified according to the concentration and type of airborne contaminants present. Monitoring and issue of respiratory protection equipment will be conducted in accordance with Radiation Protection Manual Procedures. Precautions will be taken to keep airborne contamination to a minimum through the use of proper engineering controls and decontamination.

Limits for inhalation of radionuclides are established in Appendix B, Table 1 of 10CFR20. The Radiation Protection Manual establishes the station's administrative limits for inhalation which will be adhered to in emergencies if possible.

Types and recommended use for each type of respirator is specified in the Radiation Protection Manual.

Approximately 35 self contained breathing apparatus are maintained onsite. Of these, a portion are maintained for fire brigade use, or normal use, and the remainder for emergency response . Spare bottles are also stored in some locations. The site has the capability to refill bottles with a compressor/air bank unit, with a cascade tank unit as a backup. Full-face respirators are maintained in some emergency gear lockers. Respirators are staged for use in plant radiation areas. The onsite Stores warehouse stocks approximately 150 full-face respirators for reserve supply.

1.4.5 Radioprotective Drugs

The need for issuance of radioprotective drugs, specifically potassium-iodide, is determined using appropriate procedures.

Radioprotective drugs in the form of potassium iodide tablets are available in the Control Room, Technical Support Center, Operations Support Center, Emergency Operations Facility and the Field Team equipment lockers. Each bottle contains dosage supply for 14 days. Emergency workers are instructed on the advantages and disadvantages of taking the tablets to provide thyroid blockage. The final decision for use of the potassium iodide is made by the emergency worker.

2. PROTECTIVE RESPONSE FOR RESIDENTS WITHIN THE PLUME EXPOSURE PATHWAY

2.1 Protective Action Recommendations

2.1.1 OPPD Guidelines

Fort Calhoun Station is designed and equipped with a series of safety systems engineered to meet all of 10 CFR 100 criteria for reactor safety. OPPD recognizes that in any accident situation, it would be prudent and logical to make every effort to further reduce and minimize exposure to the public. OPPD management will recommend to appropriate State and local authorities that protective actions be initiated if any person is expected to receive an emergency exposure in excess of Environmental Protection Agency (EPA) guidelines.

Tables J-1[1] through J-4[1] provide some information and guidance on formulating Protective Action Recommendations (PAR's). Table J-1[1] summarizes the considerations for selecting the evacuation Protective Action Guides (PAG's). Table J-2[1] outlines the early (plume) phase PAG's due to exposure of airborne and deposited radioactivity. Table J-3[1] summarizes the considerations for selecting relocation PAG's. Tabie J-4[1] outlines the immediate (relocation) phase PAG's due to exposure to deposited radioactivity.

During the early (plume) phase of a radiological emergency, professional judgement will be required in the application of PAG's, due to varying characteristics, such as; plant conditions, evacuation time estimates, environmental conditions, affected population groups, etc. In all cases, the PAR's transmitted by OPPD to the states of Iowa and/or Nebraska are strictly recommendations. The respective government agencies in each state have the ultimate responsibility for implementing necessary protective actions for the general public.

Tables J-5 and J-6 provide information pertaining to emergency worker exposure limits and health risks associated with exposure to higher dose levels.

1 Taken from "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." EPA-400-R-92-001, Revised May, 1992.

2.1.2 Initiation of Recommendations

Recommendations will originate from an Emergency Response Facility based upon data derived from implementing procedure, EPIP-EOF-6. This procedure establishes a method for determining projected doses to the population-at-risk. Protective action recommendations based on radiological parameters or plant conditions are determined using EPIP-EOF-7. Total population exposure can be estimated using projected or known dose values and population densities.

2.2 Notification

In the event public notification is required, both transient and resident population within the plume exposure pathway will be initially notified through the Alert Notification System (reference Section E) and local sheriffs' departments. Information will be provided for transient and resident population as well as the general public outside the EPZ through the Emergency Alert System. Radio station KFAB-AM (1110 KHz) is the Local Primary One (LP1) information station for local emergencies.

Information brochures describing notification, protective actions and general radiological education are provided to residents by mail and by public service posting to transients within the EPZ. The States of Iowa and Nebraska will issue messages describing the incident and recommended public protective actions.

2.3 Evacuation

2.3.1 Evacuation Time Estimate Study

Studies estimating the time required to evacuate the residents in the plume exposure pathway from the emergency planning zone were conducted in accordance with NUREG-0654, Rev. 1, Appendix 4 criteria. These studies are supporting documents to this Plan. Summaries of the Nebraska and Iowa evacuation time estimate studies are outlined in Figures J-3 and J-4, and Tables J-9 and J-10.

During normal weather conditions the maximum time to evacuate the approximately 4,365 residents of Iowa and the 13,138 Nebraska residents of the EPZ, is estimated at 170 minutes. In adverse conditions these times would increase to 175 minutes for Iowa and 185 Minutes for Nebraska.

2.3.2 Evacuation of Areas within the EPZ

The Governor (or Governor's Authorized Representative) of Nebraska can authorize the Nebraska State Patrol and Emergency Management Agency, based on recommendations of the State Health Department, to evacuate Nebraska residents to reception centers in Fremont and Bellevue, Nebraska.

The Governor (or Governor's Authorized Representative) of Iowa can authorize the Iowa State Patrol and the Emergency Management Division to evacuate Iowa residents to Denison, Iowa, based upon recommendations of the Iowa Department of Public Health.

Normal evacuation routes within the EPZ are, on the Nebraska side of the river, people in Sub Area 1, Blair, North of Blair and North of U.S. Highway 30 (Sub Area 2) should go to the Fremont Reception Center. People south of Blair between the Missouri River and U.S. Highway 30 (Sub Areas 3, 4 and 5) should go to the Bellevue Reception Center. People in the Iowa portion of the EPZ should proceed to the Denison Reception Center. If alternate evacuation is necessary, directions will be given through the designated local primary one (LP1) Emergency Alert System radio station. Figure J-5 shows the boundaries and highways leading to the Reception Centers.

The relocation centers for the host areas are as follows:

A. Bellevue
First Baptist Church Activities Building
Hancock Street and 23rd Avenue

B. Fremont
Fremont Senior High School
1750 N. Lincoln

C. Denison
Denison Community High School, North 16th

2.3.2 The ingestion planning zone (IPZ) encompasses a 50 mile radius as illustrated in Figure J-6. Population for the IPZ is presented in Figure J-7 by sectors.

The plume exposure EPZ encompasses an approximate 10 mile radius as illustrated in Figure J-5. The EPZ includes portions of Harrison and Pottawattamie Counties in Iowa, Washington and Douglas Counties in Nebraska. The States of Iowa and Nebraska are separated by the Missouri River. Figure J-8 shows the total population within the EPZ, and population totals for each sector by ring miles (0-2, 2-5 and 5-10 miles) and total miles (0-2, 0-5 and 0-10 miles). Figure J-4 shows Sub Area population, including Estimated Transient population.

2.4 Protective Methods (Other than Evacuation)

2.4.1 Sheltering

Remaining indoors during the passage of a radioactive cloud affords the dweller a reduction in the quantity of radionuclides inhaled, as well as providing shielding. Figure J-9[1] shows the ratio of the inhaled dose inside a shelter to that outside the shelter as a function of the ventilation rate. A ventilation rate survey showed a rate variance of 0.07 to 3.0 per hour. The ventilation rate is affected by temperature differential, wind speed and direction, quality of construction and topographical setting.

Walls of buildings absorb and scatter gamma rays, thus providing a lower dose to the occupants. The shielding factor of a building is the ratio of the interior dose to the exterior dose. Shielding factor estimates applicable to residential housing units were made using the shielding technology by Z. G. Burson and A. E. Profio (1975). Table J-7[2] summarizes shielding factors for designated structures/locations from a gamma cloud source.

1,2,3 2.4.1 Table J-8[3] summarizes the shielding factors for designated structures/locations from surface deposition of radioactive material. Burson and Profio proved that the fallout shielding technology developed via nuclear weapons tests could be directly applied to radioactivity deposited on surfaces after a reactor accident. The shielding factors listed in Table J-8 assumes uniform distribution of the radioactive fallout.

In each of the cases discussed, inhalation and shielding factors from a gamma cloud source and shielding factors from surface deposition of radioactive material, it is noted that the shielding factors using sheltering as a method of protection ranges from 0.6 to 0.005. Although the best protection seems to be the basement of large multi-structured buildings, the basement of any house has been proven to provide significant shelter from airborne and surface deposited radioactive material.

2.5 Radiological Environmental Monitoring

In the event of an emergency, the permanent air particulate stations are first utilized for immediate data, concerning airborne releases. Background radiation stations provide short term exposure data and are periodically replaced. TLD use can be increased during the longer term as the District maintains a TLD services contract with a off-site vendor. The Environmental laboratory personnel perform accelerated collection and analysis of samples as their primary responsibility after an emergency occurs. Sampling requirements will be determined by the environmental laboratory personnel.

Sample analysis will be performed by the station and at offsite facilities as deemed necessary.

1 Taken from WASH-1400(NUREG-75/014), October 1975, Figure VI.11-4.
2 Taken from WASH-1400(NUREG-75/014), October 1975, Figure VI-11-7.
3 Taken from WASH-1400(NUREG-75/014), October 1975, Figure VI-11-8.

Table J-1 - Summary of Considerations for Selecting the Evacuation PAG's[1]

DOSE Rem (mrem)	Consideration(s)
50 Rem (50000 mrem)	Assumed threshold for acute health effects in adults.
10 Rem (10000 mrem)	Assumed threshold for acute health effects in the fetus.
5 Rem (5000 mrem)	Maximum acceptable dose for normal occupational exposure for adults.
5 Rem (5000 mrem)	Maximum dose justified to average members of the population, based on the cost of evacuation.
0.5 Rem (500 mrem)	Maximum acceptable dose to the general population from all sources from nonrecurring, non-accidental exposure.
0.5 Rem (500 mrem)	Minimum dose justified to average members of the population, based on the cost of evacuation.
0.5 Rem (500 mrem)	Maximum acceptable dose[2] to the to the fetus from occupational exposure of the mother.
0.1 Rem (100 mrem)	Maximum acceptable dose to the general population from all sources from routine (chronic) non-accidental exposure.
0.03 Rem (30 mrem)	Dose that carries a risk assumed to be equal to or less than that from evacuation.

[1] Taken, in part, from Table C-8, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-400-R-92-001, May, 1992.

[2] This is also the dose to the 8- to 15-week-old fetus at which the risk of mental retardation is assumed to be equal to the risk of fatal cancer to adults from a dose of 5 rem.

Table J-2 - PAG's and PAR's Based on Radiological Data[1]

Early Phase (PLUME Phase):

Types of Exposure	PROTECTIVE ACTION GUIDES (Projected or Actual Dose)	PROTECTIVE ACTION RECOMMENDATIONS (PARs)
TEDE or CDE (Thyroid)	< 1 REM TEDE < 5 REM CDE (Thyroid)	*** * NO PAR REQUIRED * *** Continue to monitor environmental radiation levels. However, at >10% of limit: (>100 mrem TEDE or >500 mrem CDE (Thyroid)) [2]Consider sheltering and/or early evacuation of children and pregnant women.
TEDE or CDE (Thyroid)	≥ 1 REM TEDE ≥ 5 REM CDE (Thyroid)	*** * EVACUATE * *** [2]Shelter, if it will provide protection equal to or greater than evacuation, up to 10 REM.
CDE (Thyroid)	≥ 25 REM CDE (Thyroid)	*** * EVACUATE * *** Recommend administration of stable iodine (KI) to OPPD emergency workers.
SDE (Skin)	≥ 50 REM SDE (Skin)	*** * EVACUATE * ***

[1] Based upon guidance in chapter 2, EPA-400-R-92-001, "Manual of Protective Action Guides And Protective Actions For Nuclear Incidents," May, 1992.

[2] Sheltering may be preferable to evacuation as a protective action in some situations. Because of the higher risk associated with evacuation of some special groups in the population (e.g. those who are not readily mobile), sheltering may be the preferred alternative for such groups as a protective action at projected doses up to 5 rem. In addition, under unusually hazardous environmental conditions, use of sheltering at projected doses up to 5 rem to the general population (and up to 10 rem to special groups) may become justified.

Table J-3 - Summary of Considerations for Selecting PAG's for Relocation[1]

DOSE Rem (mrem)	Consideration(s)
50 Rem (50000 mrem)	Assumed threshold for acute health effects in adults.
10 Rem (10000 mrem)	Assumed threshold for acute health effects in the fetus.
6 Rem (6000 mrem)	Maximum projected dose in first year to meet 0.5 Rem in the second year[2].
5 Rem (5000 mrem)	Maximum acceptable dose for normal occupational exposure for adults.
5 Rem (5000 mrem)	Minimum dose that must be avoided by one year relocation based on cost.
3 Rem (3000 mrem)	Minimum projected first-year dose corresponding to 5 Rem in 50 years[2].
3 Rem (3000 mrem)	Minimum projected first-year dose corresponding to 0.5 Rem in the second year[2].
2 Rem (2000 mrem)	Maximum dose in first year corresponding to 5 Rem in 50 years from a reactor incident, based on radioactive decay and weathering only.
1.25 Rem (1250 mrem)	Minimum dose in first year corresponding to 5 Rem in 50 years from a reactor incident, based on radioactive decay and weathering only.
0.5 Rem (500 mrem)	Maximum acceptable single-year dose to the general population from all sources from non-recurring, non-incident exposure.
0.5 Rem (500 mrem)	Maximum acceptable dose to the fetus from occupational exposure of the mother.
0.1 Rem (100 mrem)	Maximum acceptable annual dose to the general population from all sources due to routine (chronic), non-incident, exposure.
0.03 Rem (30 mrem)	Dose that carries a risk assumed to be equal to or less than that from relocation.

[1] Taken, in part, from Table E-5, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-400-R-92-001, May, 1992.

[2] Assumes the source term is from a reactor incident and that simple dose reduction methods are applied during the first month after the incident to reduce the dose to persons not relocated from contaminated areas.

Table J-4 - Protective Action Guides for Exposure to Deposited Radioactivity [1]

Protective Action	PAG (projected dose in first year)[2]	Comments
Relocate the general population.[3]	≥ 2 Rem (≥ 2000 mrem)	Beta dose to skin may be up to 50 times higher.
Apply simple dose reduction techniques.[4]	< 2 Rem (< 2000 mrem)	These protective actions should be taken to reduce doses to as low as practicable levels.

[1] Taken, in part, from the "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-400-R-92-001, May, 1992.

[2] The projected sum of total effective dose equivalent (TEDE) from external gamma radiation and committed effective dose equivalent (CEDE) from inhalation of resuspended materials, from exposure or intake during the first year. Projected dose refers to the dose that would be received in the absence of shielding from structures or the application of dose reduction techniques. These PAG's may not provide adequate protection from some long-lived radionuclides.

[3] Persons previously evacuated from areas outside the relocation zone defined by this PAG may return to occupy their residences. Cases involving relocation of persons at high risk from such action (e.g., patients under intensive care) should be evaluated individually.

[4] Simple dose reduction techniques include scrubbing and/or flushing hard surfaces, soaking or plowing soil, minor removal of soil from spots where radioactive materials have concentrated, and spending more time than usual indoors or in other low exposure rate areas.

Table J-5 - Emergency Worker Exposure Limits

DOSE LIMIT	ACTIVITY	CONDITION(S)
≤500 mrem TEDE	All Activities	Pregnant Emergency Workers
≤5 Rem TEDE	All Activities	Non-Pregnant Emergency Workers
≤10 Rem TEDE	Protecting Valuable Property	A lower dose is not practicable
≤25 Rem TEDE	Life Saving or Protection of Large Populations	A lower dose is not practicable
>25 Rem TEDE	Life Saving or Protection of Large Populations	Only on a voluntary basis to persons fully aware of the risks involved. (See Table J-6)

NOTE: Fort Calhoun Station has established separate administrative limits for use during normal operating conditions. These limits are outlined in various Radiation Protection procedures and policies.

Table J-6 - Summary of Risks Involved with Higher Dose Limits
(taken form EPA 400_R-92-001, May, 1992)

Health Effects Associated with Whole-Body Absorbed Dosed Received Within a Few Hours[a]

Whole Body Absorbed Dose (rad)	Early Fatalities[b] (percent)	Whole Body Absorbed Dose (rad)	Prodromal Effects[c] (percent affected)
140	5	50	2
200	15	100	15
300	50	150	50
400	85	200	85
460	95	250	98

[a] Risks will be lower for protracted exposure periods.

[b] Supportive medical treatment may increase the dose at which these frequencies occur by approximately 50 percent.

[c] Forewarning symptoms of more serious health effects associated with large doses of radiation.

Approximate Cancer Risk to Average Individuals from 25 Rem Effective Dose Equivalent Delivered Promptly

Age at Exposure (years)	Approximate Risk of Premature Death (deaths per 1,000 persons exposed)	Average Years of Life Lost in Premature Death Occurs (years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15
50 to 60	3.5	11

Table J-7 - Representative Shielding Factors from Gamma Cloud Source

Structure or Location	Shielding Factor [a]	Representative Range
Outside	1.0	----
Vehicles	1.0	----
Wood - frame [b] (no basement)	0.9	----
Basement of wood house	0.6	0.1 to 0.7 [c]
Masonry house (no basement)	0.6	0.4 to 0.7 [c]
Basement of masonry house	0.4	0.1 to 0.5 [c]
Large office or industrial building	0.2	0.1 to 0.3 [c,d]

[a] The ratio of the interior dose to the exterior dose.

[b] A wood frame house with brick or stone veneer is approximately equivalent to a masonry house for shielding purposes.

[c] This range is mainly due to different wall materials and different geometries.

[d] The reduction factor depends on where the personnel are located within the building (e.g., the basement or an inside room).

NOTE: Consideration is limited to gamma radiation since beta and alpha particles cannot penetrate the walls of structures.

* Taken from WASH-1400 (NUREG-75/104), October 1975.

Table J-8 - Representative Shielding Factors for Surface Deposition

Structure or Location	Representative [a] Shielding Factor	Representative Range
1 m above an infinite smooth surface	1.00	----
1 m above ordinary ground	0.70	0.47 - 0.85
1 m above center of 50-ft roadways, half contaminated	0.55	0.4 - 0.6
Cars on 50-ft road:		
Road fully contaminated	0.5	0.4 - 0.7
Road 50% decontaminated	0.5	0.4 - 0.6
Road fully decontaminated	0.25	0.2 - 0.5
Trains	0.40	0.3 - 0.5
One and two-story wood-frame house (no basement)	0.4 [b]	0.2 - 0.5
One and two-story block and brick house (no basement)	0.2 [b]	0.04 - 0.40
House basement, one or two walls fully exposed:	0.1 [b]	0.03 - 0.15
One story, less than 2 ft of basement, walls exposed	0.3 [b]	0.03 - 0.07
Two stories, less than 2 ft of basement, walls exposed	0.3 [b]	0.02 - 0.05
Three or four-story structures, 5,000 to 10,000 ft^2 per floor:		
First and second floor	0.05 [b]	0.01 - 0.08
Basement	0.01 [b]	0.001 - 0.07
Multistory structures, >10,000 ft^2 per floor		
Upper floors	0.01 [b]	0.001 - 0.02
Basement	0.005 [b]	0.001 - 0.015

[a] The ratio of the interior dose to the exterior dose.

[b] Away from doors and windows.

Table J-9

STATE OF NEBRASKA EVACUATION TIME ANALYSIS					
	SUB AREA 1	SUB AREA 2	SUB AREA 3	SUB AREA 4	SUB AREA 5
Approximate distance from FCS, in miles	0-2	2-10	2-9	2-9	9-10
Permanent population	488	8972	2064	2767	353
Permanent population vehicles	197	3589	826	1107	142
Transient population	1110	1873	0	1678	0
Transient population vehicles	1110	1736	0	678	0
Persons without vehicles *	0	3105	0	610	0
Special vehicles	0	78	0	16	0
Notification Time	00:15	00:15	00:15	00:15	00:15
Preparation time, permanent population	01:35	01:35	01:35	01:35	01:35
Preparation time, transient population	00:45	00:45	00:45	00:45	00:45
Preparation time, special population *	NA	2:00	NA	2:00	2:00
Evacuation time, permanent population, normal conditions	2:40	2:55	2:55	2:50	2:40
Evacuation time, permanent population, adverse conditions	3:20	3:55	3:55	3:20	2:40
Evacuation time, transient population, normal conditions	2:40	2:55	2:55	2:50	2:40
Evacuation time, transient population, adverse conditions	3:20	3:55	3:55	3:20	2:40
Evacuation time, special population, normal conditions	NA	2:55	2:55	2:50	NA
Evacuation time, special population, adverse conditions	NA	3:55	3:55	3:20	NA
Confirmation time					
* includes schools, nursing homes and transportation dependent					

Table J-10

STATE OF IOWA EVACUATION TIME ANALYSIS

	SUB AREA 10	SUB AREA 11	SUB AREA 12	SUB AREA 13	SUB AREA 14
Approximate distance from FCS, in miles	0-2	2-9	8-11	8-11	2-11
Permanent population	29	162	423	3023	142
Permanent population vehicles	12	65	170	1210	57
Transient population	5000	0	0	536	500
Transient population vehicles	1667	0	0	361	167
Persons without vehicles *	0	0	0	1062	0
Special vehicles	0	0	0	29	0
Notification time	00:15	00:15	00:15	00:15	00:15
Preparation time, permanent population	02:30	02:30	02:30	02:30	02:30
Preparation time, transient population	00:60	00:60	00:60	00:60	00:60
Preparation time, special population *	NA	NA	NA	2:15	NA
Evacuation time, permanent population, normal conditions	2:35	2:35	2:55	2:45	2:45
Evacuation time, permanent population, adverse conditions	2:40	2:40	3:05	2:50	2:50
Evacuation time, transient population, normal conditions	2:35	NA	2:55	2:45	2:45
Evacuation time, transient population, adverse conditions	2:40	NA	NA	2:50	2:50
Evacuation time, special population, normal conditions	NA	NA	NA	2:50	NA
Evacuation time, special population, adverse conditions	NA	NA	NA	2:55	NA
Confirmation time					

* includes schools, nursing homes and transportation dependent

Figure J-1 - Normal Evacuation Routes

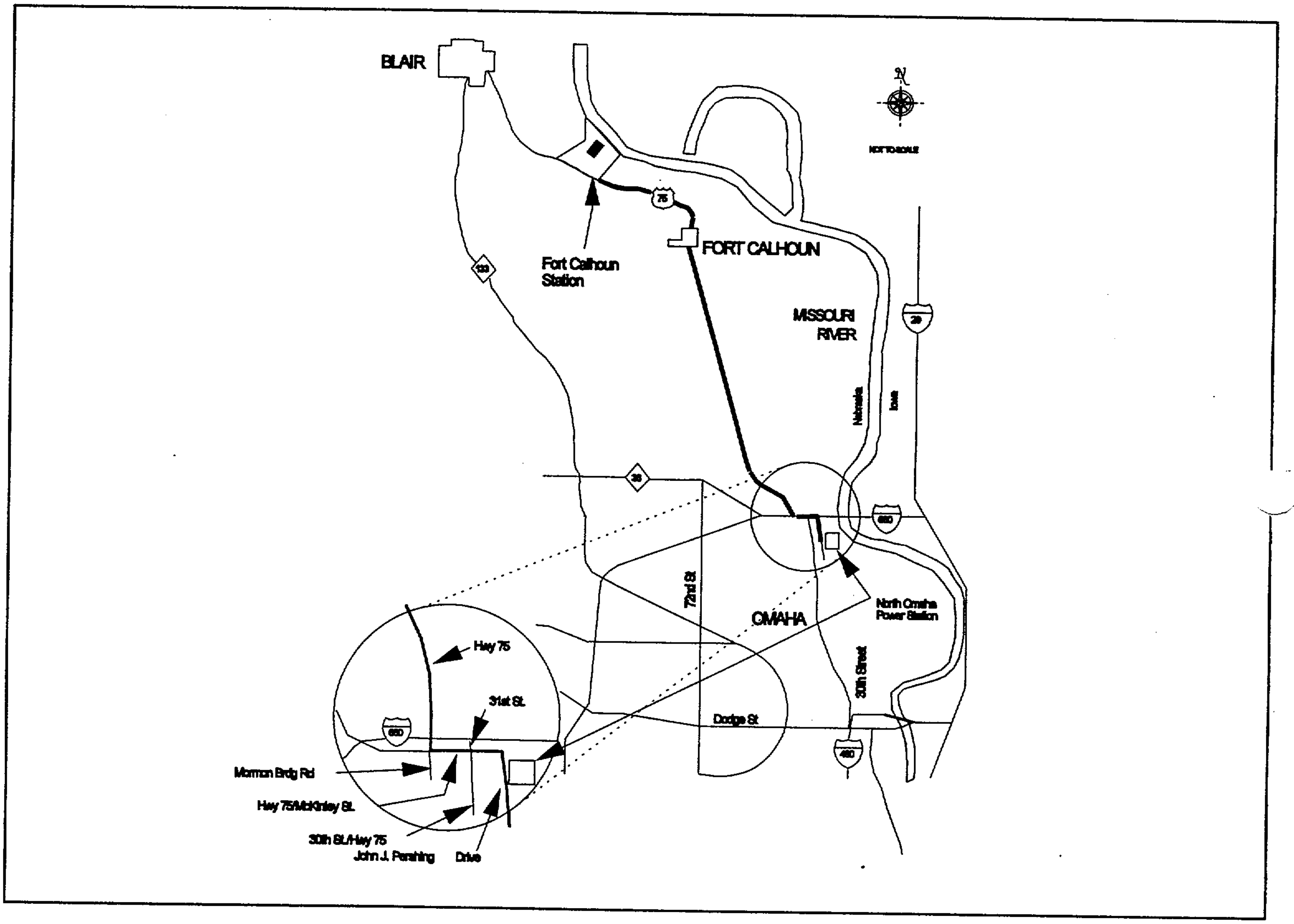


Figure J-2 - Alternate Evacuation Route

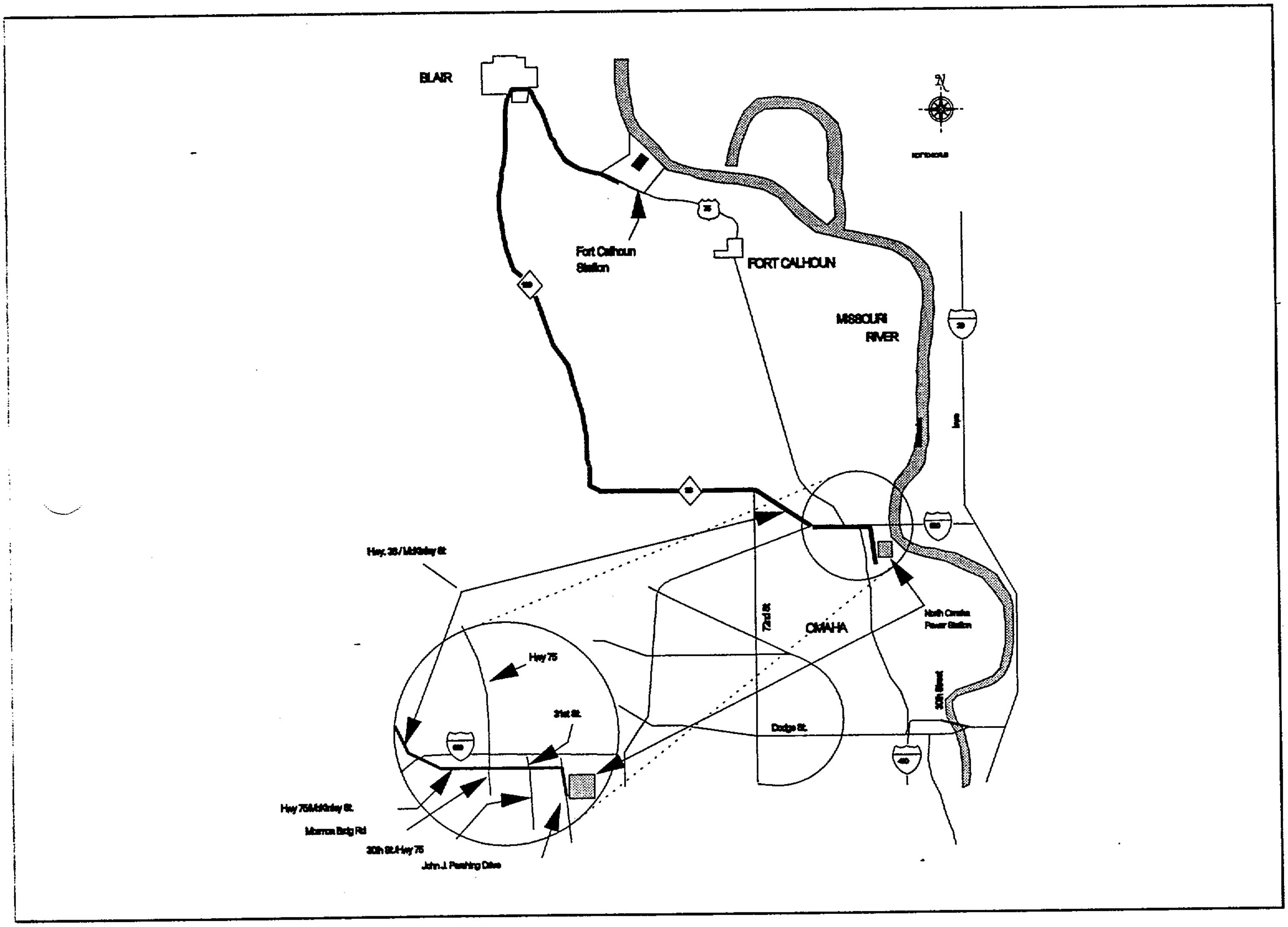


Figure J-3 - Evacuation Time Estimate Study Sub-area Designations

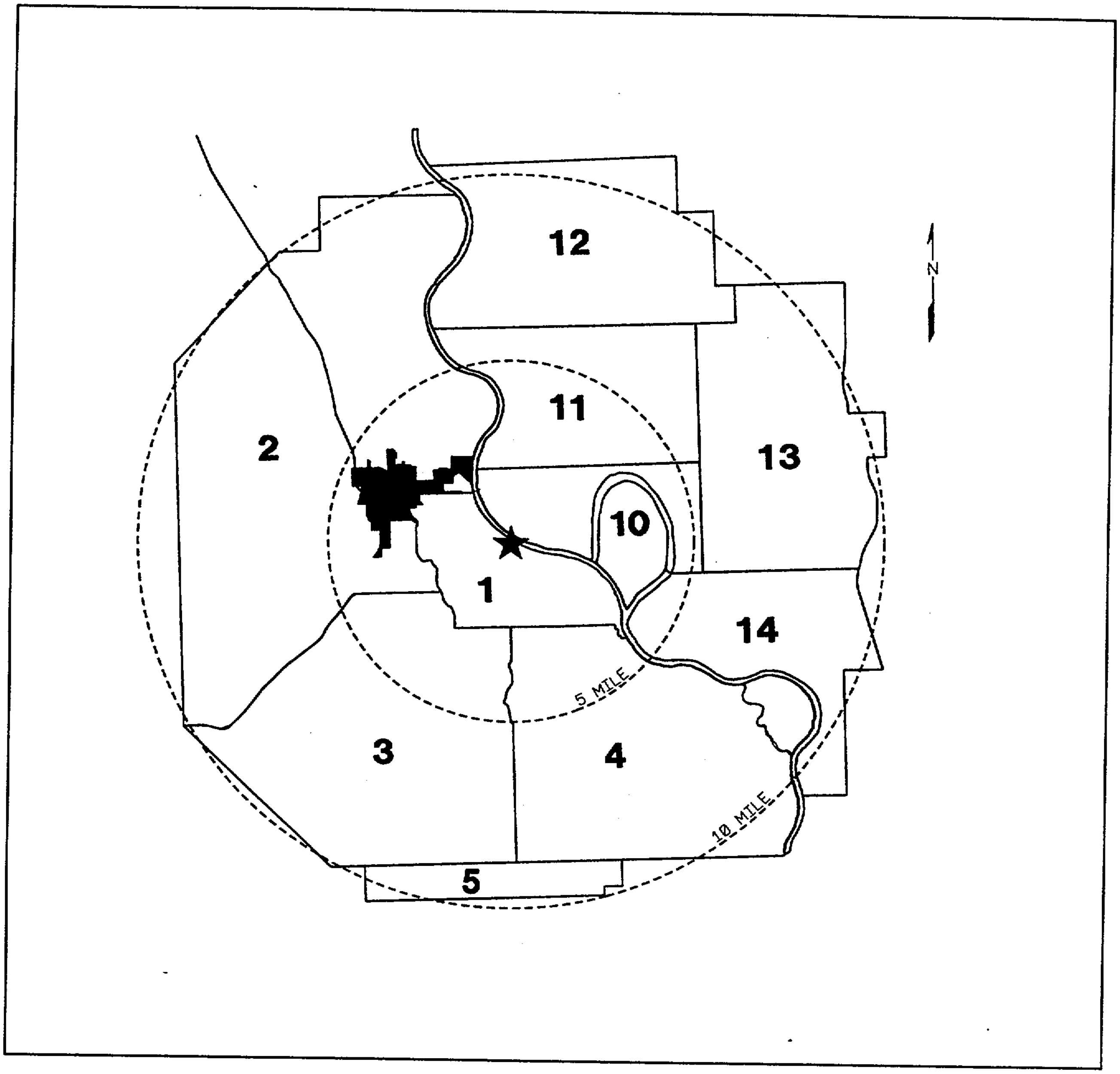
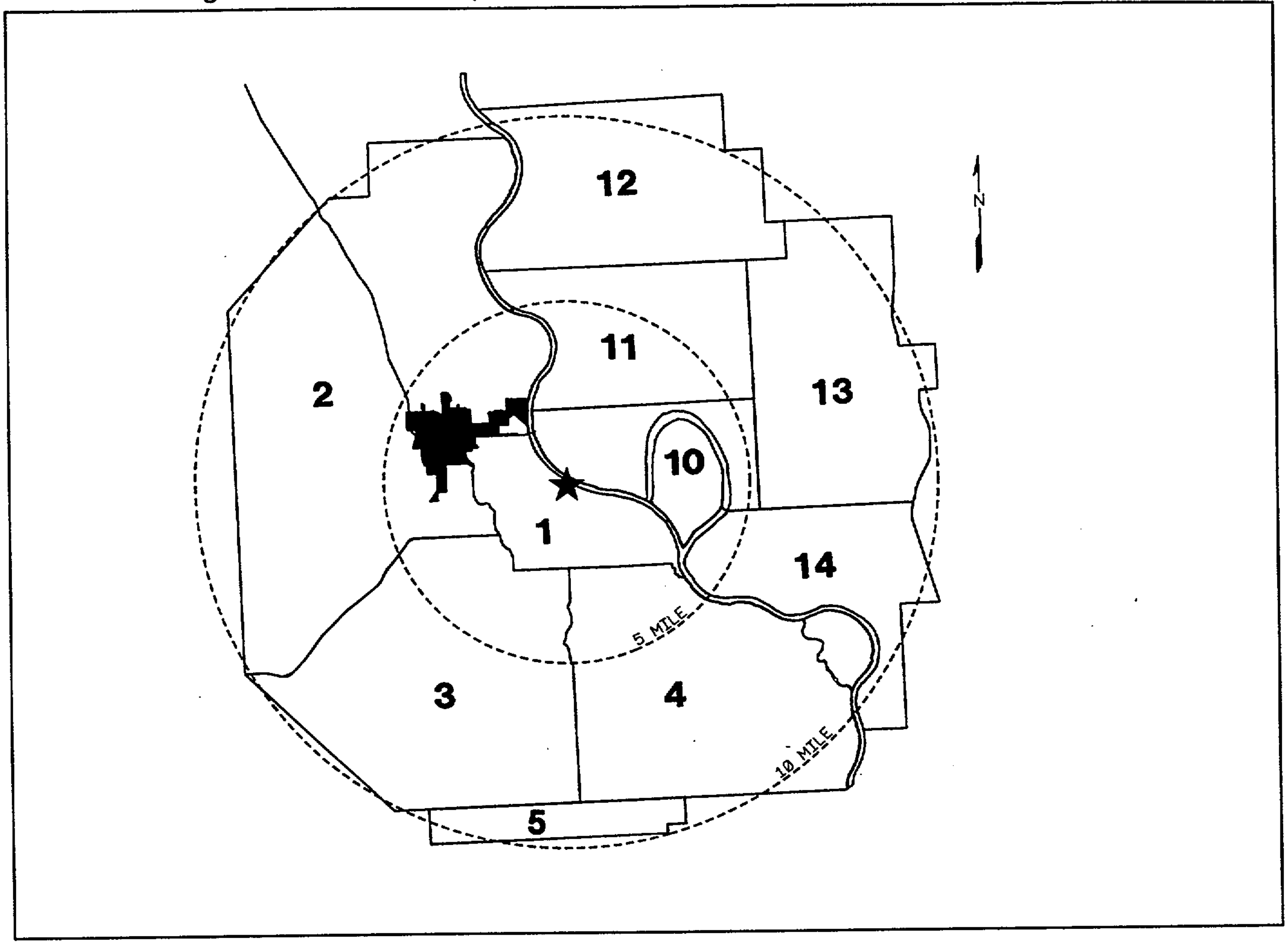


Figure J-4 - Plume Exposure EPZ Population and Evacuation Times



Nebraska					Iowa				
Sub Area	Population		Evacuation Time		Sub Area	Population		Evacuation Time	
	Minimum	Maximum	Normal	Adverse		Minimum	Maximum	Normal	Adverse
1	488	1598	2:40	3:10	10	29	5044	2:35	2:40
2	8972	13950	2:55	3:55	11	162	162	2:35	2:40
3	2064	2064	2:55	3:55	12	423	423	2:55	3:05
4	2767	3705	2:50	3:10	13	3023	4784	2:45	2:50
5	353	353	2:40	2:40	14	142	642	2:50	2:50
Total	14644	21670	2:50	3:10	Total	3779	11055	2:55	3:05

Figure J-5 - Routes to Relocation Centers

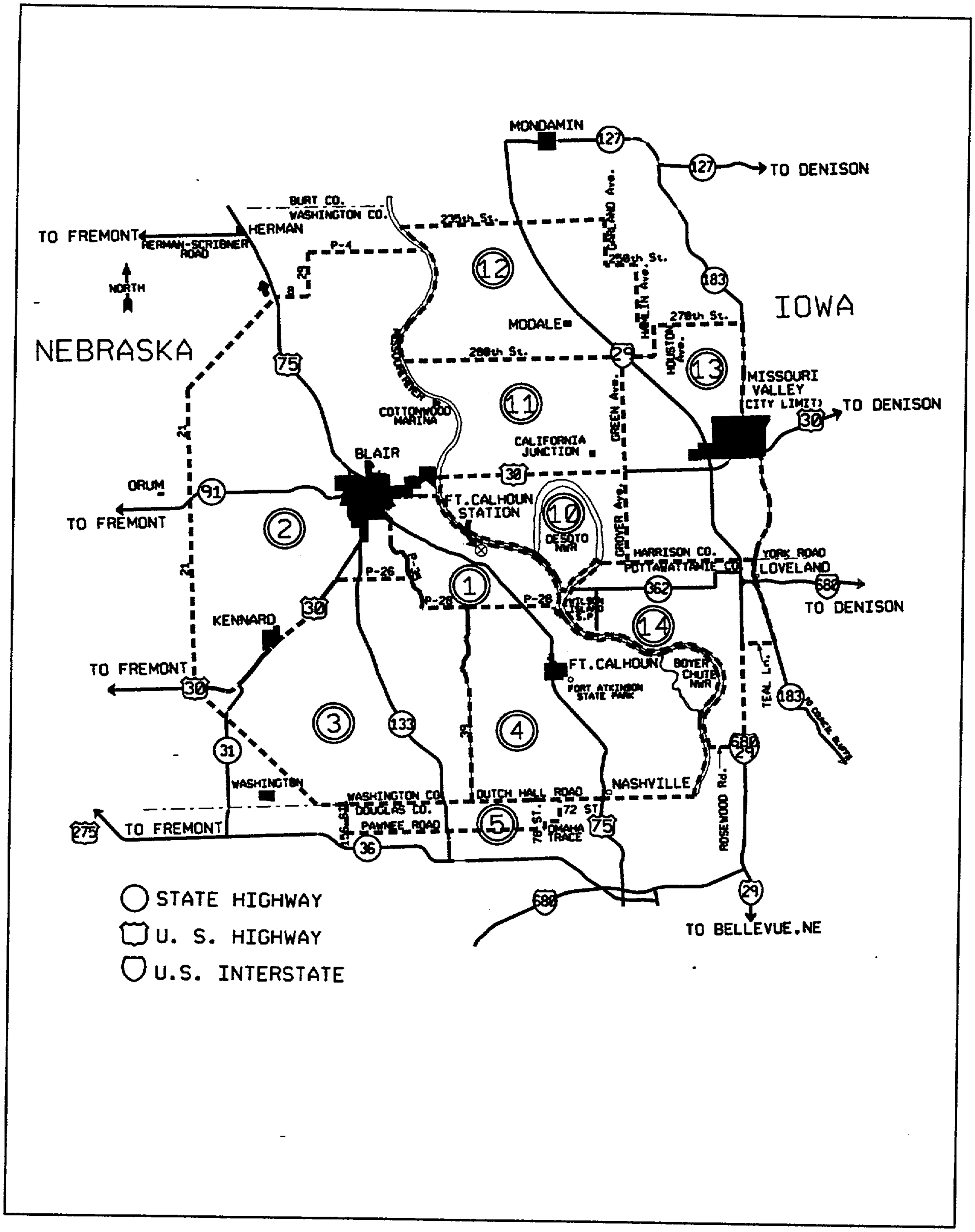


Figure J-6 - Ingestion Pathway EPZ

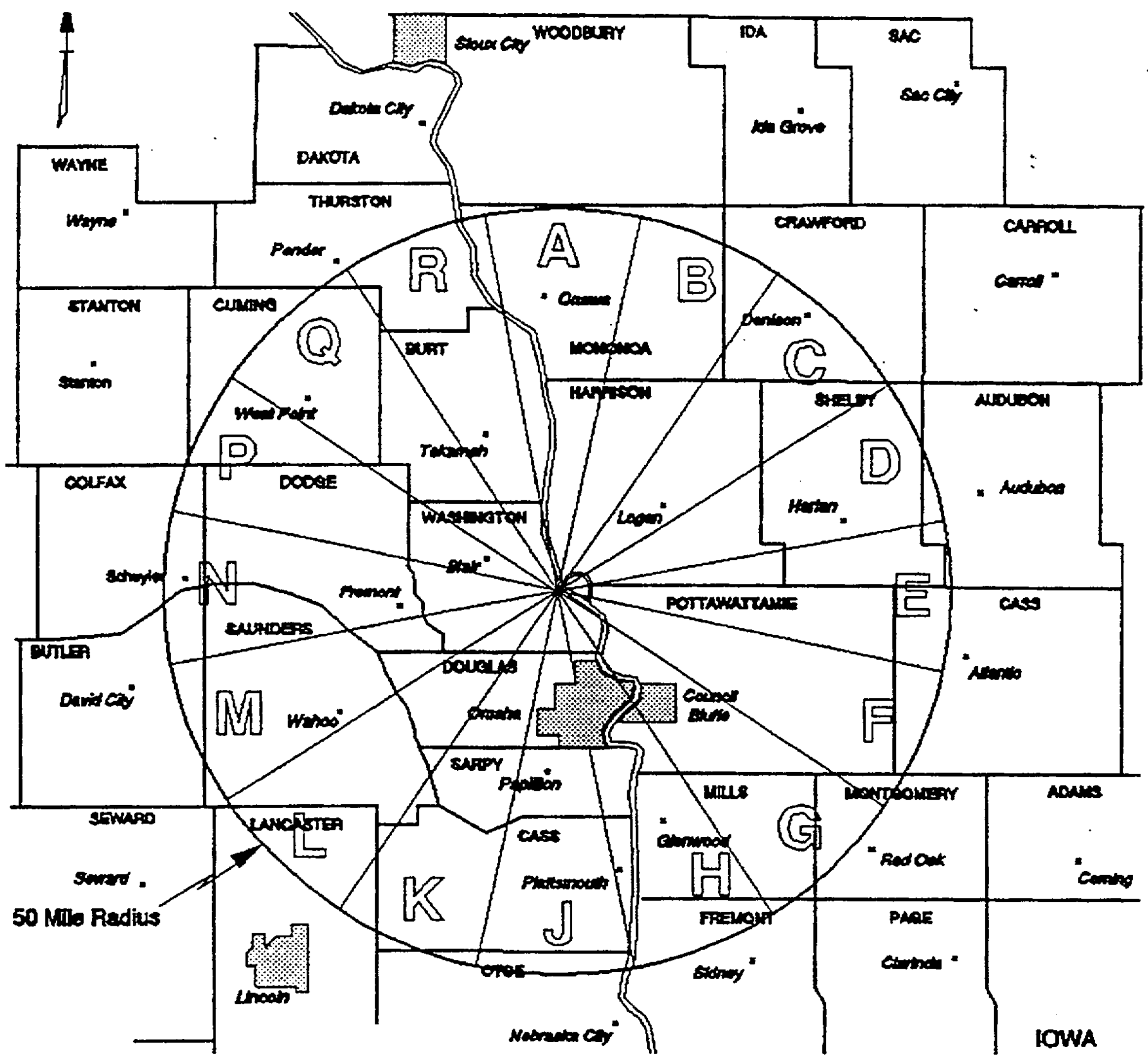


Figure J-7 - Population Estimates Ingestion Pathway SPZ

POPULATION DISTRIBUTION AS OF 1990						
DISTANCE FROM FCS IN MILES OF SECTOR SEGMENT						
SECTOR	0 TO 10	10 TO 20	20 TO 30	30 TO 40	40 TO 50	TOTALS
A	49	710	948	3720	1871	7298
B	393	499	1067	1578	3351	6888
C	114	703	2299	2369	1151	6636
D	2077	5092	1009	1903	7748	17829
E	102	696	2820	2615	2319	8552
F	120	893	1795	3699	2307	8814
G	879	5432	12296	3151	3012	24730
H	1961	107123	151221	13637	3801	277743
J	807	163724	82464	6740	4745	258480
K	1405	18269	23318	4452	6335	53779
L	420	2996	4534	5111	3176	16237
M	881	2263	22334	1672	1984	29134
N	516	1184	4030	2731	1693	10154
P	6361	1085	2060	5494	2838	17838
Q	1273	1271	740	3437	2492	9213
R	146	1085	2364	1356	3087	8038
TOTALS	17504	313025	315299	63665	51910	761403
CUMULATIVE	17504	330529	645828	709493	761403	761403

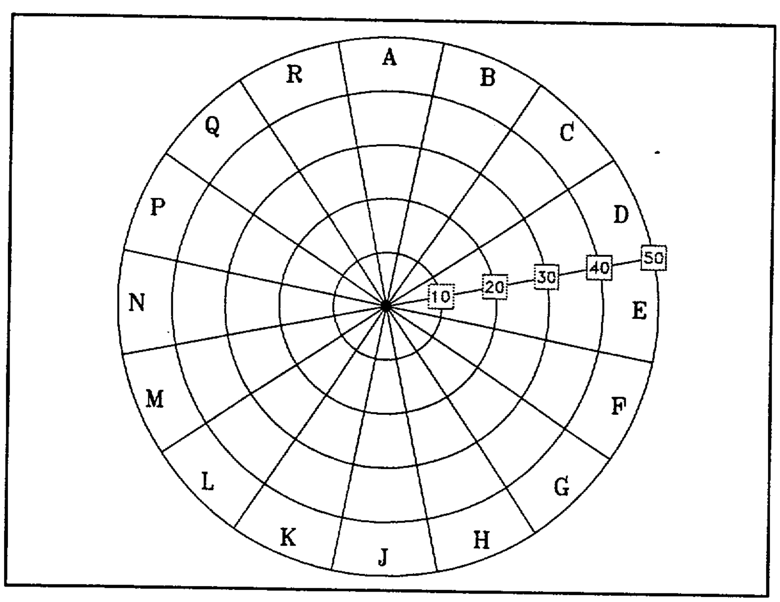
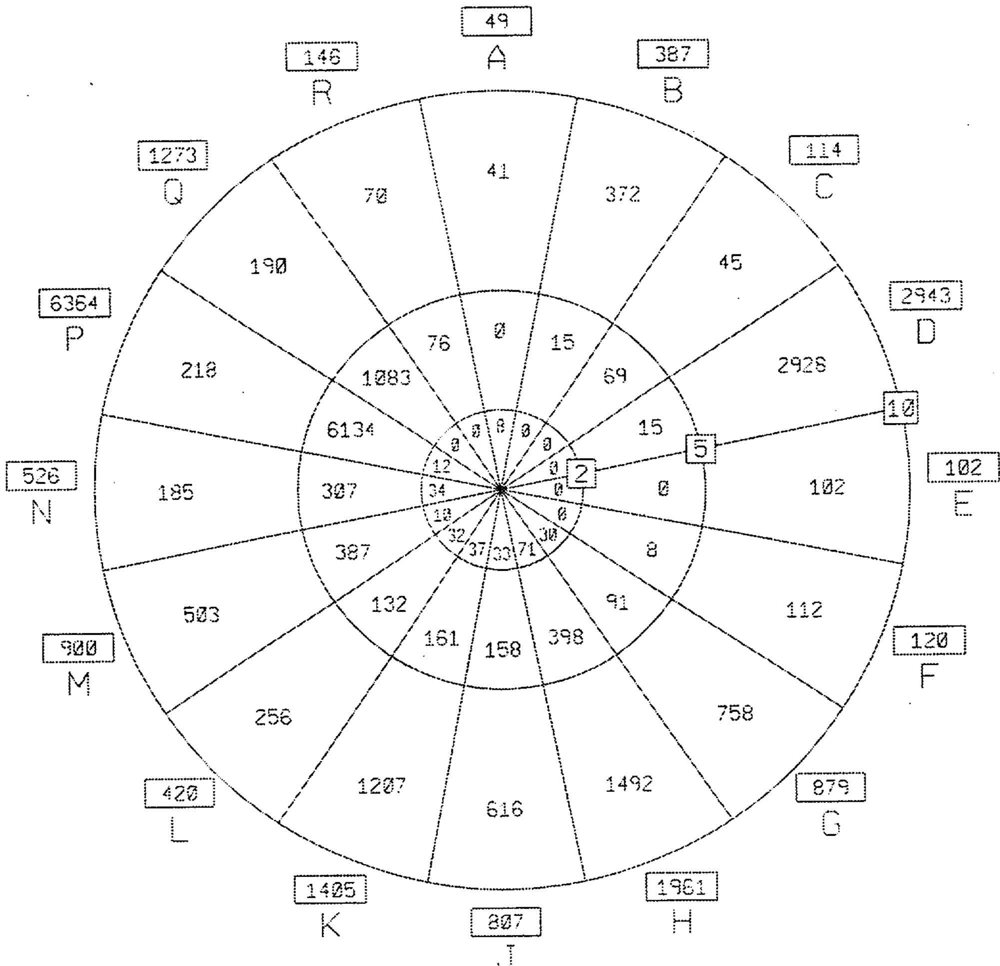
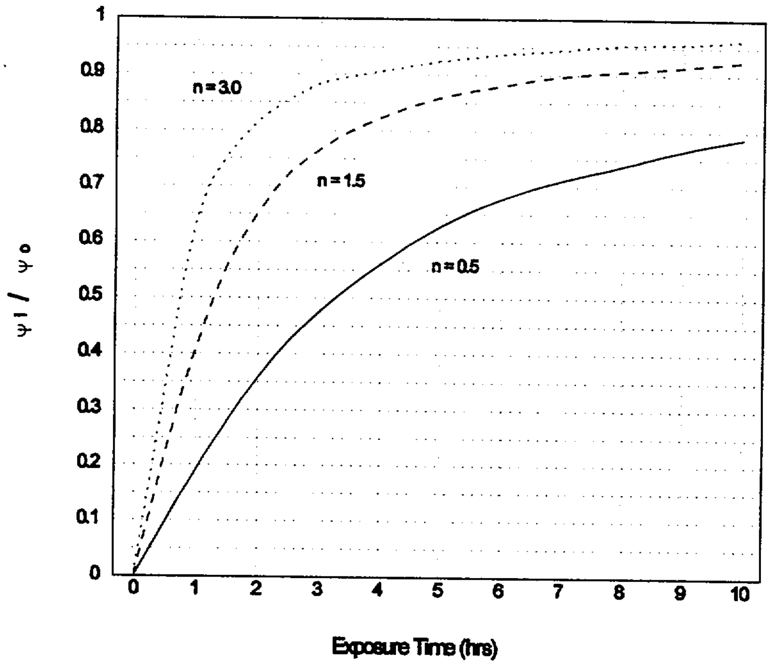


Figure J-8 - Population Estimates Plume Exposures EPZ



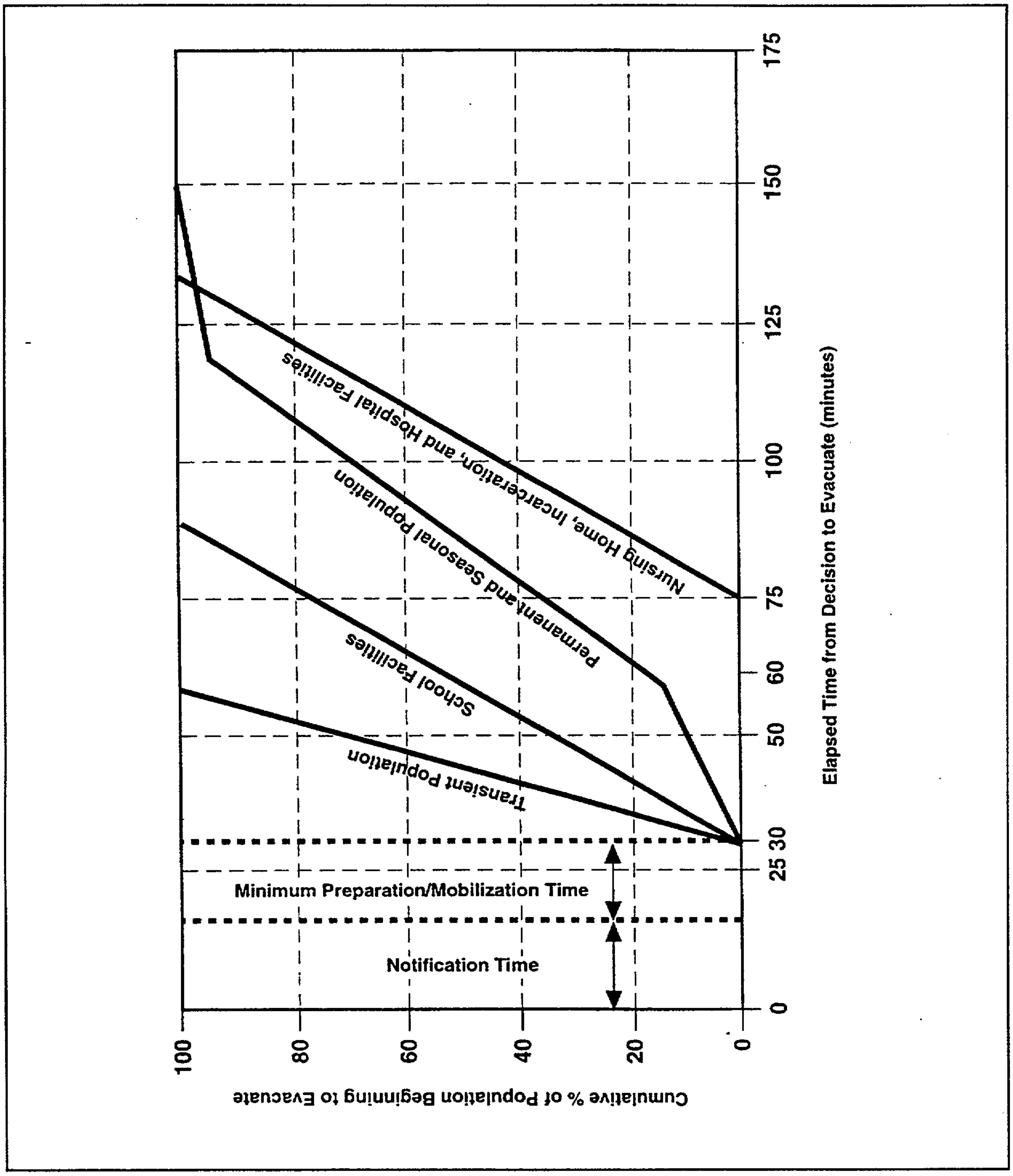
Population Totals			
Ring Miles	Ring Population	Total Miles	Cumulative Population
0-2	267	0-2	267
2-5	9034	0-5	9301
5-10	9095	0-10	18396

Figure J-9 - Ratio of Inhaled Dose Inside a Shelter to That Outside the Shelter as a Function of Ventilation Rates



KEY:	
n—Ventilation Rate	
ψ I	- Dosage inside of building
ψ o	- Dosage outside of building

Figure J-10 - Notification/Preparation/Mobilization Time Distributions



FORM EP-1
Rev. 6 (04/29/99)

OMAHA PUBLIC POWER DISTRICT

Confirmation of Transmittal for Emergency Planning Documents/Information

☐ Radiological Emergency Response Plan (RERP)
■ Emergency Plan Implementing Procedures (EPIP)
☐ Emergency Planning Forms (EPF)
☐ Emergency Planning Department Manual (EPDM)
☐ Other Emergency Planning Document(s)/ Information

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Date: 2-3-00

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REMOVE SECTION
EPIP Index page 1 of 2 Issued 12/22/99
EPIP-EOF-10 R9 Issued 11/01/90
EPIP-TSC-8 R11 Issued 02/25/97

INSERT SECTION
EPIP Index page 1 of 2 issued 01/13/00
EPIP-EOF-10 R10 Issued 01/13/00
EPIP-TSC-8 R12 Issued 01/13/00

Summary of Changes:
Both procedures were reformatted per Writers Guide. Information on the PRG PASS and Excel spreadsheets were deleted from EPIP-TSC-8 and the Equilibrium Source Inventory was updated.

For Ronald L. Moray
Supervisor - Emergency Planning

I hereby acknowledge receipt of the above documents/information and have included them in my assigned manuals.

Signature: ______________________ **Date:** ____________

Please sign above and return by **03/12/00** to:

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EMERGENCY PLAN IMPLEMENTING PROCEDURE INDEX

PROCEDURE NUMBER	TITLE	REVISION/DATE
EPIP-OSC-1	Emergency Classification	R32 07-29-99
EPIP-OSC-2	Command and Control Position Actions/Notifications	R34 10-07-98a
EPIP-OSC-9	Emergency Team Briefings	R7 12-09-99
EPIP-OSC-15	Communicator Actions	R19 12-14-99
EPIP-OSC-20	Site Population Exposure Estimates	R6 11-10-95
EPIP-OSC-21	Activation of the Operations Support Center	R8 09-30-97
EPIP-TSC-1	Activation of the Technical Support Center	R20 10-08-99
EPIP-TSC-2	Catastrophic Flooding Preparations **REINSTATED**	(R0 03-22-95) DELETED 05-09-95 R2 02-06-96
EPIP-TSC-8	Core Damage Assessment	R12 01-13-00
EPIP-EOF-1	Activation of the Emergency Operations Facility	R11 09-23-99b
EPIP-EOF-3	Offsite Monitoring	R16 10-26-99
EPIP-EOF-6	Dose Assessment	R27 03-11-97a
EPIP-EOF-7	Protective Action Guidelines	R12 09-01-94
EPIP-EOF-10	Warehouse Personnel Decontamination Station Operation	R10 01-13-00
EPIP-EOF-11	Dosimetry Records, Exposure Extensions and Habitability	R18 09-18-97b
EPIP-EOF-19	Recovery Actions	R7 09-30-98
EPIP-EOF-21	Potassium Iodide Issuance	R3 09-18-97
EPIP-EOF-23	Emergency Response Message System	R5 10-12-99

Distribution Authorized

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Fort Calhoun Station
Unit No. 1

EPIP-EOF-10

EMERGENCY PLAN IMPLEMENTING PROCEDURE

Title: WAREHOUSE PERSONNEL DECONTAMINATION STATION OPERATION

FC-68 Number: DCR 10955

Reason for Change: Reformat per Writers Guide.

Initiator: Doug Levine

Preparer: Mark Reller

WAREHOUSE PERSONNEL DECONTAMINATION
STATION OPERATION

NON-SAFETY RELATED

1. PURPOSE

1.1 The purpose of this procedure is to provide instructions for operating the Warehouse Personnel Decontamination Station.

2. REFERENCES/COMMITMENT DOCUMENTS

2.1 RPI-1, Personnel Monitoring and Decontamination

2.2 FC-RP-207-1, Personnel/Clothing Contamination Report

3. DEFINITIONS

NONE

4. PREREQUISITES

NONE

5. PROCEDURE

5.1 Use the Warehouse Decontamination Station Operation Checklist, Attachment 6.1, as an aid to completing required actions.

5.2 Retain all documentation (logs, calculation sheets, notes, etc.) generated or used during the emergency. At the termination, deliver all documentation to the Administrative Logistics position in the TSC.

6. ATTACHMENTS

6.1 Warehouse Decontamination Station Operation Checklist

6.2 Warehouse Personnel Decontamination Station Layout

Attachment 6.1 - Warehouse Decontamination Station Operation Checklist

INITIALS

1. Set up a frisking station near the entrance to the Decontamination Showers (see Attachment 6.2). ____________

2. Set up a frisking station at the exit of the Women's Decontamination Shower. ____________

NOTE: Set up the Men's Decontamination Shower only if there is a large number of personnel to be decontaminated.

3. Set up a frisking station at the exit of the Men's Decontamination Shower. ____________

4. Verify that the Holding Tank Alert Alarm System Box is plugged into the wall receptacle below the box. (ALARM BOX on Attachment 6.2) ____________

5. Test the Holding Tank Alert Alarm System by momentarily depressing the Alarm test button. ____________

6. Verify the Submersible Pump SD-13 controller is in the "OFF" position. (CONTROLLER on Attachment 6.2) ____________

7. Verify that Breaker No. 13 in MPP-41 (Section 2) is in the "OFF" position. ____________

8. Verify that the Submersible Discharge Valve SD-106 operator, located outside the warehouse and a few feet to the west of the Holding Tank Manway, is "LOCKED SHUT". ____________

9. Survey personnel reporting to the Decontamination area in accordance with RP procedures. ____________

NOTE: Periodically survey area to confirm contamination control.

9.1 If contaminated:

9.1.1 Direct the person into the undressing area

9.1.2 Follow the direction of Radiation Protection Procedures to decontaminate the individual.

Attachment 6.1 (continued)

9.2 If not contaminated, direct individual to continue Site Evacuation.

10. If/When the High Level Alarm in the Holding Tank sounds:

NOTE: If personnel decontamination is still required, the Radiation Protection Coordinator and Site Director should consider using the decontamination shower in the Auxiliary Building or weigh the risk of delaying personnel decontamination against pumping the Holding Tank.

10.1 Secure the decontamination showers and sinks. ______________

NOTE: The Holding Tank contents can be pumped out only with an approved release permit. The Radiation Protection Coordinator should request a portable tank and pumping equipment if the contents cannot be released.

10.2 Contact the Radiation Protection Coordinator in the OSC: ______________

10.2.1 Report the alarm.

10.2.2 Report the status of personnel decontamination.

10.2.3 Request that the Chemistry Coordinator be contacted to have the tank sampled and a release permit prepared.

11. Following use, survey and decontaminate the Decontamination Station in accordance with Radiation Protection Procedures. ______________

12. If the showers or sinks were used, post signs indicating that the Holding Tank and associated piping is potentially contaminated. ______________

Attachment 6.2 - Warehouse Personnel Decontamination Station Layout

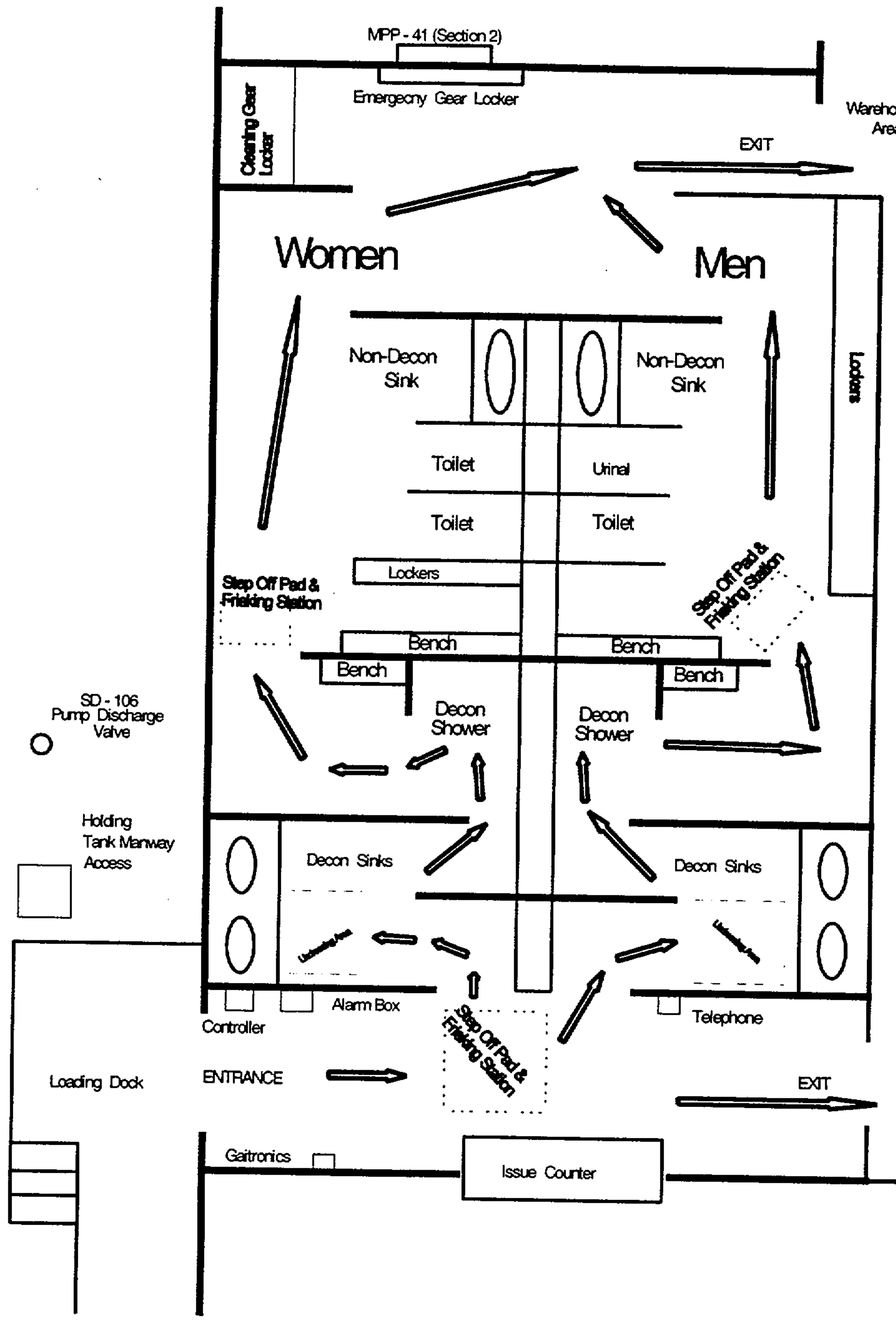

Figure 1 {EOF-10.WPG}

Fort Calhoun Station
Unit No. 1

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EPIP-TSC-8

EMERGENCY PLAN IMPLEMENTING PROCEDURE

Title: CORE DAMAGE ASSESSMENT

FC-68 Number: DCR 10311

Reason for Change: Reformat delete PRG PASS, delete Excel Spread Sheets and update Equilibrium Source Inventory.

Initiator: Ron Meng

Preparer: Ron Meng

CORE UNCOVERY PREDICTIONS / CORE DAMAGE ASSESSMENTS

NON-SAFETY RELATED

1. PURPOSE

1.1 This procedure provides methods for trending Reactor Coolant System (RCS) volume above the active core during Loss of Coolant Accidents (LOCA), methods for estimating the time to core uncovery, and performing core damage assessments.

2. REFERENCES/COMMITMENTS DOCUMENTS

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, Task 467, July 1983

2.2 FC-0204-98, Source Terms Used in Emergency Plan Core Damage Assessment Spreadsheet (TSC-8)

2.3 EA-FC-90-94, Fuel Handling Accident and Bounding Source Term

2.4 NUREG 1465, Accident Source Terms for Light - Water Nuclear Power Plants

2.5 Calculation Number FC06727, Estimated Containment 994' Elevation Volume vs. Level for EPIP-TSC-8

3. DEFINITIONS

3.1 Time to Core Uncovery - A time calculated based on a snapshot of plant parameters that estimates the time remaining before the start of core uncovery (Reactor Vessel Level at the top of the active core).

3.2 Core Damage Assessment - The categorization of a core damage into four major types as follows: no fuel damage, fuel cladding failures, fuel pellet overheating and fuel pellet melting. The three later categories are delineated as initial, intermediate, and major. Each of the ten categories of core damage can be characterized by the type of fuel damage, the corresponding temperature range and the mechanism of fission product release.

3.3 Category of no fuel damage - that which is characterized by the release of fission products through the release mechanisms of Halogen Spiking and tramp uranium fission. The release source is the Gas Gap and the characteristic isotopes are I-131, Cs-137, and Rb-88.

3.4 Category of Fuel Cladding Failure - that which is characterized by the release of fission products through the release mechanisms of burst and Gas Gap diffusion. The release source is the Gas Gap. The characteristic fission products are noble gases and halogens (Xe-131m, Xe-133, I-131 and I-133) because they are volatile and can migrate quickly through the fuel pellet and gas gap for release following cladding rupture. These isotopes are volatile in the Temperature range of 1300-1800° F.

3.5 Category of Fuel Pellet Overheat - that which is characterized by the release of radioactivity through gain boundary diffusion and by diffusion from within the UO_2 grains. Grain boundary diffusion begins above 2450° F. Moderately volatile isotopes of cesium, rubidium and tellurium (Cs-134, Rb-88, Te-129, and Te-132) are characteristic of this type of damage.

3.6 Category of Fuel Pellet Melt - occurs at greater temperatures (2550-3450°F), reactions begin between the solid UO_2 and the solid metallic zircaloy, melting of the control rod materials, and melting of zirconium. At these temperatures greater amounts of tellurium are released. Alkali metals, such as barium, volatize as well as rare earths and actinides such as lanthanum and protactinium. These include the isotopes of Sr-89, Ba-140, La-140, La-142, and Pr-144.

3.7 Activity Ratios - theoretical calculations employed to determine typical ratios for isotopes of a fission product in the Gas Gap or the Fuel Pellet. Comparison of the ratios obtained from sample data with these calculated values determines the source of the fission product release. These ratios are made by comparing the noble gas/iodine isotopes in the sample to Xe-133/I-131 in the sample.

4. PREREQUISITES

4.1 This procedure is intended to be used during Loss of Coolant Accidents and other plant transients that may lead to core damage.

NOTE: Steps in this procedure may be done out of order or be completed concurrently.

NOTE: HJTCs may be deenergized per OI-RC-2A RCS Fill and Drain Operations causing RVLMS to indicate 100% while in reduced RCS inventory conditions. While the HJTCs are deenergized, use LI-119 and LI-197 and Attachment 6.1 -Figure #1, RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level for determining RCS volumes. Energizing HJTCs is acceptable for short periods to determine if a void exists in the Reactor Vessel per the precautions in OI-RC-2A.

5. PROCEDURE

NOTE: Trending the RCS Inventory above the active Core should be done if a significant decrease is observed in the RCS Inventory.

5.1 Trend the RCS inventory volume above the core using Attachment 6.1 - Trending RCS Inventory above the Core.

NOTE: The time to core uncovery is determined using two methods. The first (normal) method, Attachment 6.2, determines a Net Loss/Gain in ft^3/min and uses this rate as the basis to determine the time remaining for Reactor Vessel Level to drop to the top of the active core. The second method, Attachment 6.3, uses Steaming Rate and RCS density to determine the boil-off rate in ft^3/min.

NOTE: The second method (Attachment 6.3) is used when the Reactor Vessel level is below that of any suspected leaks, the inventory is being lost is due to RCS boil-off, and it is determined that no make up flow is reaching the reactor vessel.

5.2 During Loss of Coolant Accidents, estimate the time to core uncovery by one of the following methods:

5.2.1 First (normal) method, complete Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery.

5.2.2 Second (Steaming Rate) method, complete Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery.

5.3 Determine the power history for the last 30 days from the time of reactor trip/shutdown by completing Attachment 6.4 - Prior 30-Day Power History.

NOTE: Core Damage Assessments should be done when the Fuel Cladding Fission Product Barrier Criteria has been exceeded per EPIP-OSC-1, Attachment 6.3, Three Fission Product Barrier Criteria or at the discretion of the Command and Control position.

5.4 Complete Attachment 6.5 - Assessment of Core Damage Using Containment Radiation Dose Rates.

5.5 For slow transients only, complete Attachment 6.6 - Assessment of Core Damage using CETs.

NOTE: The total quantity of fission products available at different locations in the Containment may be changing due to transient plant conditions. Samples of the Reactor Coolant System / Discharge of the Low Pressure Safety Injection Pump and the Containment Atmosphere used for core damage assessments should be obtained within a minimum time and under stabilized plant conditions when possible.

5.6 Prepare to do a Core Damage assessment by Radiological Analysis of samples as follows:

5.6.1 Verify that plant conditions are stabilized as much as possible that plant conditions can support sampling operations for a Reactor Coolant System or Discharge of the Low Pressure Safety Injection Pump sample and/or a Containment Atmosphere sample.

5.6.2 Before the sampling team is dispatched to collect the required samples, hold a briefing with the sampling team and coordinate the collection of data that needs to be obtained when samples are isolated per Attachment 6.7, Worksheet #1 and #2.

5.7 Complete Attachment 6.7 - Manual Core Damage Assessment Calculations.

5.8 Using the completed Attachment 6.7, make a conclusion on the extent of core damage using following three parameters:

5.8.1 Identification of the fission product isotopes that characterize the Reactor Coolant System / LPSI Pump and Containment Atmosphere Samples.

5.8.2 Identification of the source of the fission products (the Gas Gap or the Fuel Pellet) from Attachment 6.7 - Worksheets #4 and #5.

5.8.3 The quantity of the fission products available for release to the environment expressed as a percent of source inventory on Attachment 6.7 - Worksheet #7.

NOTE: Knowledgeable judgement is used to compare the three parameters with the definitions of the 10 NRC categories of fuel damage found on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage. Core Damage is not anticipated to take place uniformly. Therefore, when evaluating the three parameters listed above, the procedure is anticipated to yield a combination of one or more of the 10 categories defined on Attachment 6.8. The categories will exist simultaneously. The type of core damage is described in the 10 NRC categories listed on Attachment 6.8. The degree of core damage is described as the percent of the fission products in the source inventory that is now available in the Containment for release to the environment.

5.9 Based on the conclusions above, circle the categories of fuel damage on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage.

6. ATTACHMENTS

6.1 Trending RCS Volume above the Core

6.2 Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

6.3 Using Steaming Rate to Estimate the Time to Core Uncovery

6.4 Prior 30 Day Power History

6.5 Assessment of Core Damage Using Containment Radiation Dose Rates

6.6 Assessment of Core Damage using CETs

6.7 Manual Core Damage Assessment Calculations

6.8 Clad Damage Characteristics of NRC Categories of Fuel Damage

Attachment 6.1 - Trending RCS Inventory above the Core (Page 1 of 4)

1. On Worksheet 1 - RCS Inventory Data Sheet, record the Time, Pressurizer Level and RVLMS.

2. Record the RCS Volume above the Core on Worksheet 1 using the information from Figure 1, RCS Inventory above the Core vs. Pressurizer/RVLMS Level.

3. Plot the RCS Volume above the Core on Worksheet 2, Plot of RCS Volume above the Core vs. Time.

Attachment 6.1 - Trending RCS Inventory above the Core (Page 2 of 4)

Worksheet 1, RCS Inventory Data Sheet

Time	Pressurizer Level %	RVLMS %	Volume above the Core (ft^3)

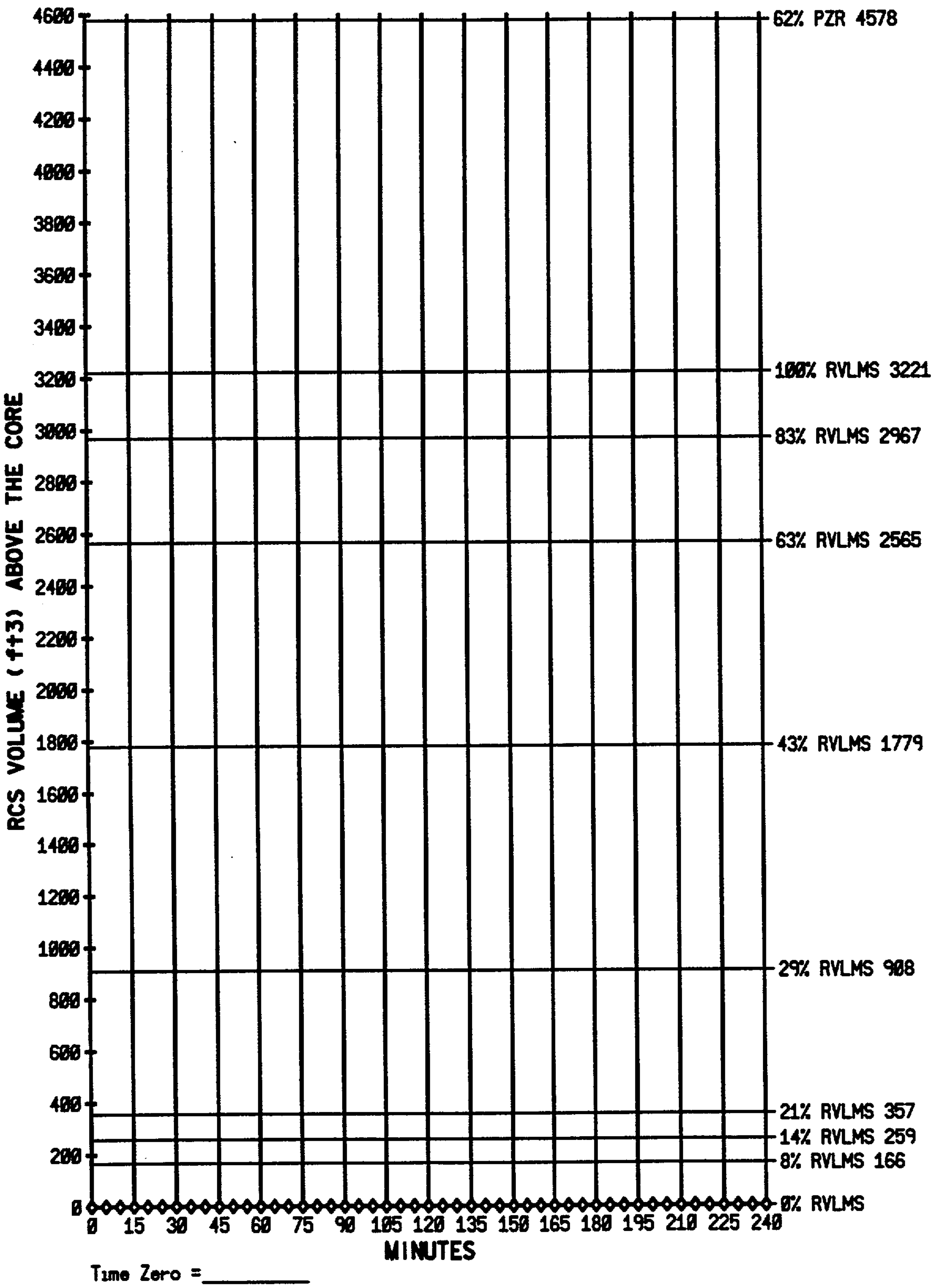
Attachment 6.1 - Trending RCS Inventory above the Core (Page 3 of 4)

Figure 1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level

PZR/RVLMS (% Level)	RCS INV. ABOVE TOP OF CORE (ft^3)	LI-119, LI-197 (Feet)	NOTES
100 PZR	4962		
90 PZR	4821		
80 PZR	4760		
70 PZR	4665		
65 PZR	4608		
60 PZR	4533		
55 PZR	4445		
50 PZR	4356		
45 PZR	4267		
40 PZR	4179		
35 PZR	4090		
30 PZR	4001		
25 PZR	3913		
20 PZR	3824		
15 PZR	3735		
10 PZR	3646		
5 PZR	3558		
0 PZR	3469		
100 RVLMS	3221	1018.31	Top of Upper Head
83 RVLMS	2967	1017.174	
63 RVLMS	2565	1014.018	
43 RVLMS	1779	1010.867	
29 RVLMS	908	1007.708	Top of Hot Leg
21 RVLMS	357	1006.375	Center of Hot Leg
14 RVLMS	259	1005.042	Bottom of Hot Leg
8 RVLMS	166	N/A	
0 RVLMS	0	N/A	Top of Active Core

Attachment 6.1 - Trending RCS Inventory above the Core (Page 4 of 4)

Worksheet 2 - Plot of RCS Volume above the Core vs. Time



Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

NOTE: When a bubble is in the Reactor Vessel use RVLMS to determine the RCS volume (i.e., Pressurizer Level at 40% and RVLMS at 83%, a volume of 2967 ft^3 based on 83% RVLMS should be used).

1. Record Time, Pressurizer Level, RVLMS Level and the RCS Volume above the Core from Attachment 6.1, Figure 1 - RCS Inventory above the Core vs. Pressurizer/RVLMS Level.

   T_1 (Time)____________ Pressurizer Level (%)____________RVLMS Level (%)____________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) ____________

2. Wait for drop in Pressurizer/RVLMS Level and record the Time, Pressurizer Level, RVLMS Level, and RCS Volume below:

   T_2 (Time)____________ Pressurizer Level (%)____________ RVLMS Level (%)____________

   V_2 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) ____________

3. Determine the Time in Minutes (T_M) to Core Uncovery using the equation below.

$$T_M = \frac{V_2\_\_\_\_\_\_\ ft^3}{(V_1\_\_\_\_\_\_\ ft^3 - V_2\_\_\_\_\_\_\ ft^3) / (T_3\_\_\_\_\_\_\ \text{minutes})} = \_\_\_\_\_\_\_\_\ \text{Minutes}$$

   Where:

   T_3 = elapsed time in minutes ($T_2 - T_1$)

4. Add the time T_M to the time T_2 to determine the Projected Time of Core Uncovery.

   Projected Time of Core Uncovery = T_2__________ + T_M__________ = ____________(hh:mm)

5. Remarks:

   ____

   ____

   ____

Done by:______________________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 1 of 3)

1. Record Time, Pressurizer Level, RVLMS Level, RCS Volume from Attachment 6.1, Figure 1, Time of Reactor Trip/Shutdown, Steaming Rate from Attachment 6.3, Figure 1, Highest CET Temperature below and RCS Density from Attachment 6.3, Figure 2.

   T_1 (Time)__________ Pressurizer Level______________RVLMS Level_______________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) ______________

   Time of Reactor Trip/Shutdown ______________hh:mm

   SR (Steaming Rate from Figure 1) ____________lb/sec

   Highest CET Temperature __________________°F

   D (RCS Density from Figure 2) _______________lb/ft^3

2. Determine the T_M (Minutes to Core Uncovery) using the equation below:

$$\text{Minutes to Core Uncovery} = \frac{(V_1\_\_\_\_\_\_\_ ft^3) * (D\_\_\_\_\_\_\_ lb/ft^3)}{(SR\_\_\_\_\_\_\_ lb/sec) * (60)} = \_\_\_\_\_\_\_\_\_\_ \text{Minutes}$$

3. Determine the Time of Core Uncovery using the equation below:

   Time of Core Uncovery = (T_1__________) + (T_M____________) = ___________(hh:mm)

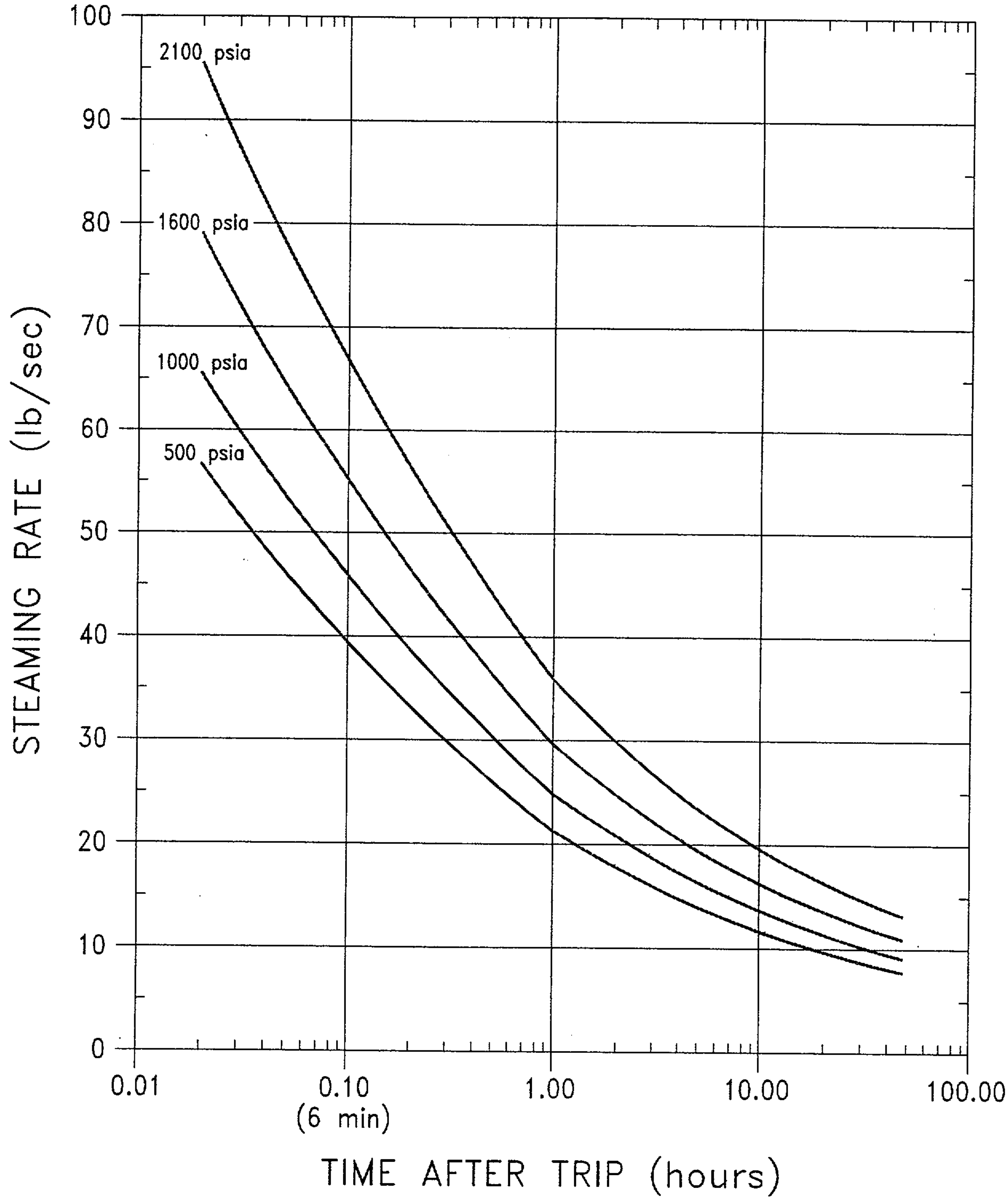
4. Remarks:

   ____
   ____
   ____
   ____

5. Done by:______________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 2 of 3)

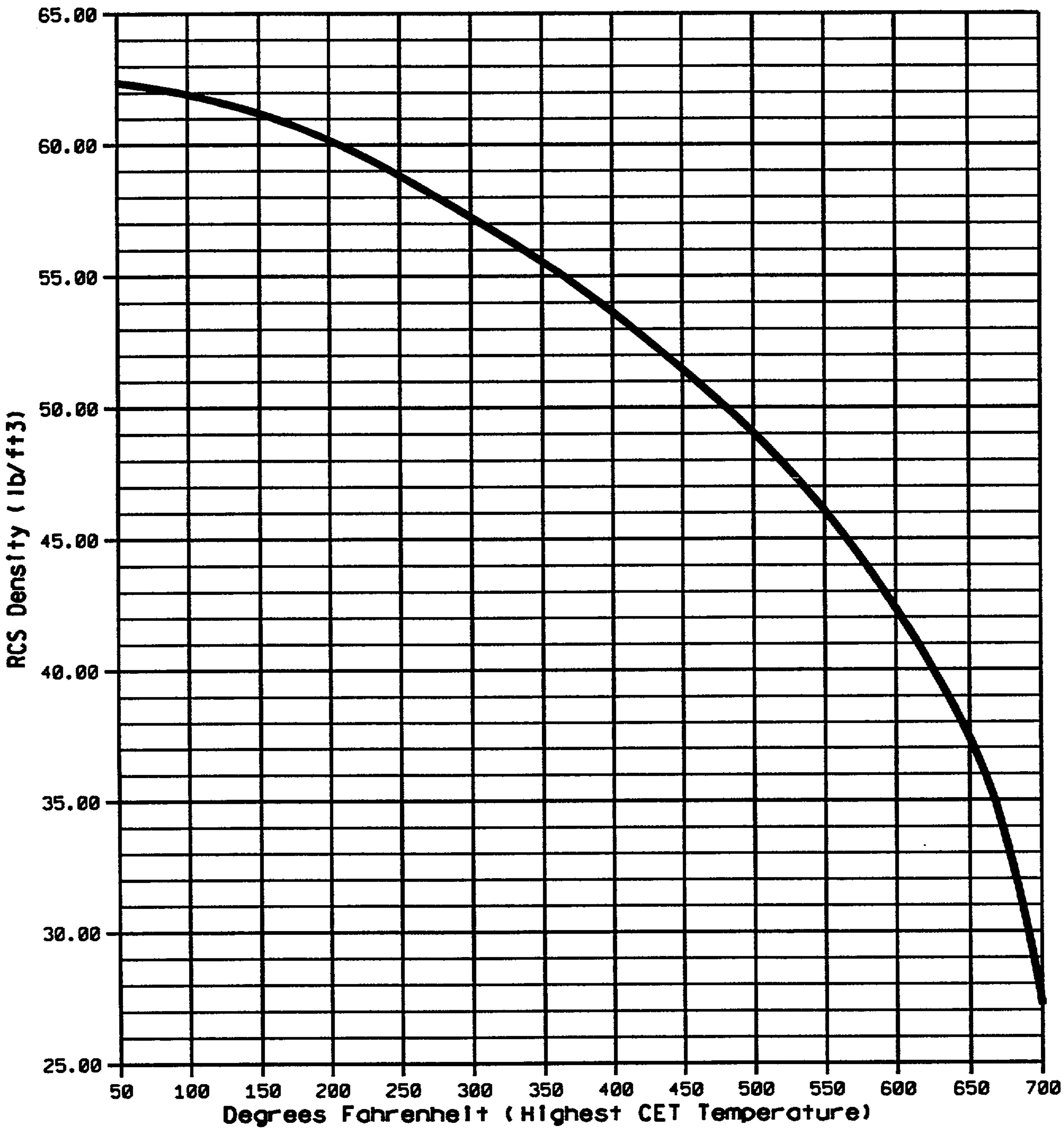
Figure 1, RCS Steaming Rate vs. Time After Trip



NOTE 1: If pressure is >2100 psia use the 2100 psia curve.
NOTE 2: If pressure =<2100 psia =>500 psia interpolate.
NOTE 3: If pressure <500 psia use the 500 psia curve.

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 3 of 3)

Figure 2 - RCS Density vs. CET Temperature



Attachment 6.4 - Prior 30 Day Power History

Date of Reactor Shutdown:	Time of Reactor Shutdown:

If reactor power has been steady state (+/- 10%) in the four days befoe the Reactor Shutdown, record the steady state power level:

If reactor power has been steady state (+/- 10%) in the 30 days before the Reactor Shutdown, record the steady state power level:

If the plant's power history has not been steady state for the the last 30 days, enter data below for each steady state power period during the last 30 days (record up to eight power periods where power has change more than +/- 10% power):

Power Period (j)	Steady State Power for Period (P_j)	Duration of Power Period (d)		Time From the End of Operating Period (P_j) To Reactor Shutdown (t)	
Number	Percent	Days	Seconds Days X 8.64E4	Days	Seconds Days X 8.64E4
1					
2					
3					
4					
5					
6					
7					
8					

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 1 of 3)

1. Record Date and Time of Reactor Shutdown: Date:__________Time:__________(T_1)

2. Date and Time of RM-091A and RM-091B Reading: Date:__________Time:__________(T_2)

3. Record: RM-091A Reading: ______________ RM-091B Reading:__________________

4. Determine the Time Post Accident, Hours by subtracting the Date and Time of RM-091A and RM-091B Reading from the Date and Time of Reactor Shutdown.

   Time Post Accident, Hours = (T_2 - T_1) = ____________________________________ = Hours

NOTE: Use the following information to estimate the 30-Day Average Power Level:

- The average power during the 30-day time period is not necessarily the most representative value for correction to equilibrium conditions.
- The last power levels at which the reactor operated should weigh more heavily in the judgement than earlier levels.
- Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.
- In the case in which the reactor has produced power for less than 30 days the procedure may be employed. However, the estimate of core damage obtained under this condition may be an under estimate of the actual condition.

5. Estimate the 30-Day Average Power Level, using the plants power history recorded on Attachment 6.4, engineering judgement, and the information in the note above.

6. Record the 30 Day Average Power Level based on engineering judgement = __________%.

7. Determine the Equilibrium Dose Rate as follows:

   $$\text{Equilibrium Dose Rate} = (\text{Higher reading of RM-091 A or B}) \times \frac{100}{(\text{30 Day Average Power})}$$

   $$\text{Equilibrium Dose Rate} = (\_\_\_\_\_\_\ \text{R/hr}) \times \frac{100}{(\_\_\_\_\_\_\ \text{30 Day Average Power})} = \_\_\_\_\_\_\_\_\ \text{R/hr}$$

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 2 of 3)

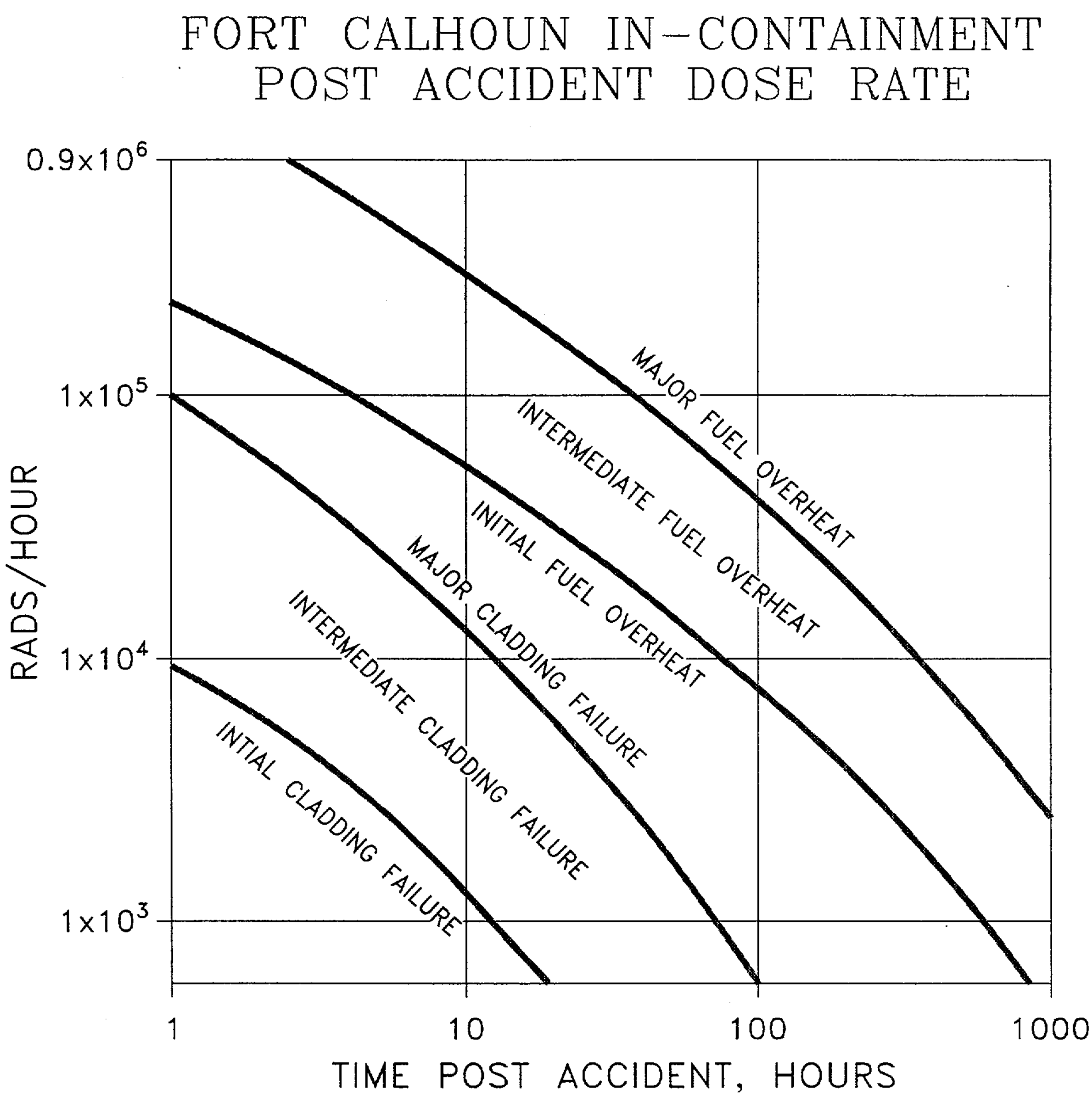
NOTE: Use the following information to consider which category of core damage is most representative of the particular value:

- Dose rate measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.
- This attachment may not be employed to estimate the degree of Fuel Pellet Melting.
- Dose rates significantly above the lower bound for the category of Major Fuel Over Heat may indicate concurrent Fuel Pellet Melting.
- This attachment may not be used to distinguish the relative contributions of the two categories (Cladding Failure and Fuel Pellet Overheat) to the total dose rate. This procedure does give the estimate of the highest category of damage.
- Dose rates within any category of fuel overheating may be anticipated to include concurrent Fuel Cladding Failure.
- Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Figure 1 - Categories of Core Damage Based on Containment Post Accident Dose Rates. The evaluation of other parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

8. On Figure 1, Category of Core Damage Based on Containment Post Accident Dose Rates, **plot** the Equilibrium Dose Rate as a function of the Time Post Accident, Hours.

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 3 of 3)

Figure 1- Category of Core Damage Based on Containment Post Accident Dose Rates



Attachment 6.6, Assessing Core Damage Using CETs (Page 1 of 3)

1. Record the following:

   1.1 Maximum CET Temperature: ______________________________F.

   1.2 Pressurizer Pressure at the time of Max CET Temperature: __________psia

   1.3 Time of the maximum CET Temperature: ______________________

2. Select the curve on Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure that represents a pressure approximately equal to or greater than the Pressurizer Pressure at the time of the Maximum CET Temperature. Read across the curve and record the Percent of Fuel Rods With Ruptured Clad Below:

   Percent of Fuel Rods With Ruptured Clad _________________________%

Attachment 6.6 - Assessing Core Damage Using CETs (Page 2 of 3)

NOTE: The Percent of Fuel Rods With Ruptured Clad obtained above is probably a lower limit of the estimate of damage. Some judgement on the bias is available as follows:

- This procedure applies most directly for slow core uncovery with a maximum temperature below the rapid oxidation temperature of 1800° F. A smooth rise in CET temperature and an uncovery duration of 20 minutes or longer are indicators for good prediction of clad ruptures.
- If pressure dropped to less than 100 psia within two minutes of the accident initiation, a large break is suggested. This causes undetected core heat up followed by flashing during a refill. Depending on the rate of refill, the CET temperature may rise rapidly then quench when the core is recovered. This procedure would then yield a very low estimate for the percent of fuel rods ruptured.
- If pressure was above 1650 psia, it could exceed the fuel rod internal gas pressure depending on rod burn up, causing clad collapse onto the fuel pellet instead of outward clad ballooning. The clad rupture criteria are less well defined for such conditions, but at temperatures above 1800° F where the highest pressure curve applies on Figure 1, clad failure sufficient to release fission gas is likely and this procedure may be used to obtain estimates of damage.

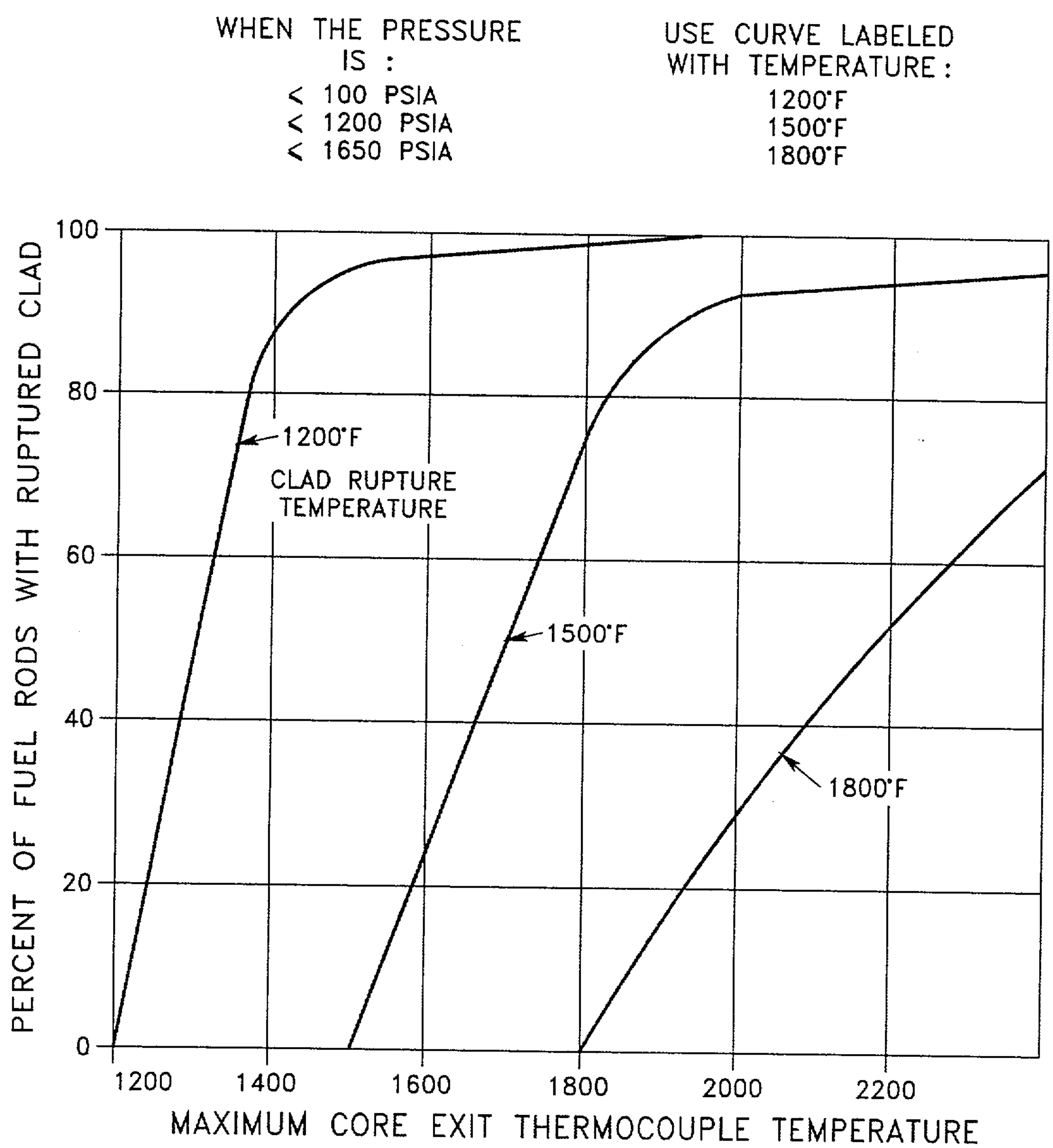
3. Comment on the probable bias in the result of Percent of Fuel Rods With Rupture Clad:

____
____
____
____
____
____
____

4. On Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage, determine and underline the NRC Category of Fuel Damage and its Characteristics.

Attachment 6.6 - Assessing Core Damage Using CETs (Page 3 of 3)

Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure



Attachment 6.7 - Manual Core Damage Assessment (Page 1 of 19)

1. On Worksheet #1, Section 1 complete the following:

   1.1 At the time of the Reactor Coolant Sample / LPSI Pump sample isolation, record the Date/Time, RCS Pressure, RCS Temperature, Pressurizer Level, RVLMS, and Containment Sump Level.

   1.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Reactor Coolant Sample / LPSI Pump sample isolation.

   1.3 Record the Sample Specific Activities from the Reactor Coolant / Discharge of the LPSI Pump sample.

   1.4 Record the Temperature Correction Factor as follows:

      1.4.1 For RCS samples obtain the temperature correction factor from Figure #1- Ratio of H_20 Density to H_20 Density at STP vs. Temperature.

      1.4.2 For the Discharge of the LPSI Pump samples record a temperature correction factor of 1.0.

   1.5 Multiply the Sample Specific Activity by the Temperature Correction Factor and record the Temperature Corrected Specific Activity (A_T).

   1.6 Determine and record the Decay Corrected Specific Activity using the following equation:

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

Where:

A_0 = the Decay Corrected Specific Activity in μCi/cc.

A_T = the Temperature Corrected Specific Activity in μCi/cc.

λ = the radioactive decay constant, 1/sec.

t = the time from a reactor shutdown to sample isolation in seconds.

Attachment 6.7 - Manual Core Damage Assessment (Page 2 of 19)

2. On Worksheet #2, complete the following:

   2.1 At the time of the Containment Atmosphere sample isolation, record the Date/Time, Containment Pressure and the Containment Temperature.

   2.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Containment Atmosphere sample.

   2.3 Determine the Temperature/Pressure Correction Factor for the Containment using the equation below and record on Worksheet #3, Section 1:

$$\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492}$$

Where:

P_1 = Containment pressure (psig) at time of Containment Atmosphere sample isolation

T_1 = Containment Temperature at the time of the Containment Atmosphere sample isolation

   2.4 Correct the Sample Specific Activities to standard Temperature and Pressure by multiplying the Sample Specific Activity by the Temperature/Pressure Correction Factor and record in the Column labeled Temperature/Pressure Corrected Specific Activity.

   2.5 Correct and record the Decay Corrected Specific Activities for decay back to the time of reactor shutdown using the following equation:

$$A_0 = \frac{A_{PT}}{e^{-\lambda t}}$$

Where:

A_0 = the Decay Corrected Specific Activities in µCi/cc.
A_{PT} = the Temperature Pressure Corrected Specific Activity in µCi/cc.
λ = the radioactive decay constant, 1/sec.
t = the time from a reactor shutdown to sample isolation in seconds.

Attachment 6.7 - Manual Core Damage Assessment (Page 3 of 19)

3. On Worksheet #3, Identify the Fission Product Release Source (Gas Gap or the Fuel Pellet) as follows:

3.1 Copy the Decay Corrected Specific Activities from Worksheet #1 for KR-87, Xe-131m, Xe-133, I-131, I-132, I-133, and I-135.

3.2 Copy the Decay Corrected Specific Activity from Worksheet #2 for KR-87, Xe-131m, Xe-133, I-131, I-132, I-133, and I-135.

3.3 Calculate the noble gas and iodine ratios using the equations below:

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Decay Corrected Specific Activity}}{\text{Xe-133 Decay Corrected Specific activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Decay Corrected Specific Activity}}{\text{I-131 Decay Corrected Specific Activity}}$$

NOTE: Select as the source of the release that ratio that is closest to the Sample Isotope Ratio. An accurate comparison is not anticipated.

3.4 Determine the Identified Source (Fuel Pellet or Gas Gap) of the release by comparing the Sample Isotope Ratio results with the predicted isotope ratios for the Fuel Pellet and the Gas Gap Inventories. Record the source in the column labeled Identified Source Fuel Pellet or Gas Gap.

4. On Worksheet #4, determine the Volumes of the Reactor Coolant System, Containment Sumps and Containment Atmosphere as follows:

4.1 Determine the volume (ft^3) of the RCS by using the Pressurizer Level and RVLMS level recorded on Worksheet #1 and Attachment 6.1, Figure #1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level, record the volumes in Section 1.

4.2 Determine and record the Density Correction in Section 1 as follows:

4.2.1 If an RCS Sample was obtained, obtain density correction from Figure #1 - Ratio of H_20 Density to H_20 Density at STP vs. Temperature by using the RCS Temperature recorded on Worksheet #1, at the time of the RCS sample.

4.2.2 If a sample was obtained from the discharge of the LPSI Pump, enter a Density Correction of 1.0 in Section 1.

4.3 Complete the RCS Volume Calculation in Section 1.

4.4 Determine and record the volume in the Containment Sump by using the Containment Sump Level recorded on Worksheet #1 and Figure #2 - Containment Sump/Basement Volume vs. L-387 or L-388 in Section 2. Record the volume in gallons in Section 2 of Worksheet #4.

4.5 Complete the calculation for the Containment Sump Volume in Section 2 on Worksheet #4.

4.6 Correct the Containment Atmosphere Volume to STP in Section 3 using the Containment Pressure and Temperature on Worksheet #2 and the following equation:

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \left(\frac{14.2 + P_1}{14.2}\right) \times \left(\frac{492}{T_1 + 460}\right)$$

Where:

2.97E+10 = the free Volume of the Containment in cc

P_1 = Containment Pressure (psig)

T_1 = Containment Temperature (° F)

NOTE: The Specific activities in the Reactor Coolant System and the Containment Sump are assumed to be equal. The results of the RCS/Discharge of the LPSI pump sample may be used for both the Containment Sump and Reactor Coolant System calculations below.

5. On Worksheet #5, calculate the Total Quantity (curies) in Containment as follows:

   5.1 Using the equation below calculate the Quantity (Curies) in the Containment Sump for each isotope and record in Column #2.

   $$\text{Curies in the Containment Sump} = \frac{SA}{1.0E+6} \times \text{CS Vol}$$

   Where:

   SA = the Decay Corrected Specific Activity on Worksheet #1

   CS Vol = the Containment Sump Volume From Worksheet #4

   1.0E+6 = Constant to convert μCurie to Curies

   5.2 Using the equation below calculate the Quantity (Curies) in the Reactor Coolant System for each isotope and record in Column #3:

   $$\text{Curies in the Reactor Coolant System} = \frac{SA}{1.0E+6} \times \text{RCS Vol}$$

   Where:

   SA = the Decay Corrected Specific Activity on Worksheet #1

   RCS Vol = the Reactor Coolant Volume From Worksheet #4

   1.0E+6 = Constant to convert μCuries to Curies

5.3 Using the equation below calculate the Quantity (Curies) in the Containment Atmosphere for each isotope and record in Column #3:

$$\text{Curies in the Containment Atmosphere} = \frac{SA}{1.0E{+}6} \times \text{CA Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #2

CA Vol = the Containment Atmosphere Volume From Worksheet #4

1.0E+6 = Constant to convert µCurie to Curies

5.4 Add Columns 2, 3, and 4. Record the sum (Total Quantity in Containment (Curies)) in Column 5.

6. On Worksheet #6, determine the Power Correction Factor and correct the Equilibrium Source Inventory for power history as follows:

6.1 Using the power history data recorded on Attachment 6.4, determine the Power Correction Factor as follows:

6.1.1 If the reactor power has been steady state (+/- 10%) for the four (4) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (2) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 4 Days}}{100}$$

6.1.2 If the reactor power has been steady state (+/- 10%) for the thirty (30) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (1) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 30 Days}}{100}$$

NOTE: The following step is used to figure out the Power Correction Faction for those isotopes in the Fuel History Groups not determined in steps 6.1.1 and 6.1.2 above.

6.2 If the reactor power has not been steady state (+/- 10%), determine the Power Correction Factor for each of the applicable power periods listed on Attachment 6.4, and record on Pages 2 and 3 of Worksheet #6 using the equation below:

$$PCF = \frac{P_j\,(1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to a reactor shutdown

6.3 For nonsteady state power histories, Sum up the power correction factors for each isotope on Pages 2 and 3 of Worksheet #6 and record on Page 1, Column #2 of Worksheet #6.

6.4 Correct the Equilibrium Source Inventory for power history, by multiplying the Power Correction Factor by the Equilibrium Source Inventory and record in Column 4 of Worksheet #6.

7. Determine and record the Percent of Source Inventory in Column 6 of Worksheet #5 using the equation below:

$$PSI(\%) = \frac{TQC}{CSI} \times 100$$

Where:

PSI(%) = Percent of Source Inventory in %

TQC = Total Quantity in Containment in Column 5 on Worksheet #7

CSI = Corrected Source Inventory in Column 4 of Worksheet #8

100 = Constant used to change a result to a percent

Attachment 6.7 - Manual Core Damage Assessment (Page 9 of 19)

Worksheet #1, Measured Specific Activity (μCi/cc) and Sample Temperature Correction for RCS/LPSI Sample

At the time RCS / LPSI Pump sample isolation record the following:

Date/Time__________ RCS Pressure __________ RCS Temperature __________

Pressurizer Level __________ RVLMS __________ Containment Sump Level __________

Time in Seconds between reactor shutdown and RCS / LPSI Pump sample isolation: __________ (seconds)

Temperature Correction Factor from Step 1.4: __________

	Sample Specific Activity	Temperature Corrected Specific Activity (A_T)	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_T}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Attachment 6.7 - Manual Core Damage Assessment (Page 10 of 19)

Worksheet #2, Measured Specific Activity (μCi/cc) and Sample Temperature Pressure Correction for Containment Atmosphere Sample

At the time of Containment Atmosphere sample isolation record the following: Date/Time__________ Containment Pressure ________ Containment Temperature _______
Time in seconds between reactor shutdown and Containment Sample isolation (t): ________
Determine Temperature / Pressure Correction Factor: $\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492} = \frac{14.2}{(\quad\quad + 14.2)} \times \frac{(\quad\quad + 460)}{492} = \_\_\_\_\_\_$

Isotope	Containment Atmosphere Sample Specific Act.	Temperature Pressure Corrected Spec. Act. (A_{PT})	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_{PT}}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Attachment 6.7 - Manual Core Damage Assessment (Page 11 of 19)

Worksheet #3 - Fission Product Release Source Identification

Fission Product Release Source Identification for RCS/LPSI Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #1	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Fission Product Release Source Identification for Containment Atmosphere Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #3	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Attachment 6.7, Manual Core Damage Assessment (Page 12 of 19)

Worksheet #4 - RCS, Containment Sump and Containment Atmosphere Volume Calculations

1. RCS Volume Calculation:

RCS Volume above the Core (V) = ________ ft^3 Density Correction (D) = ___________

RCS Volume = (V + 1823 ft^3) X (D) X (28317 cc/ft^3)

Where:

V = RCS Volume above the Core

1823 ft^3 = RCS Volume Below the Top of the Active Core

D = the Density Correction

RCS Volume = (__________+ 1823) X (__________) X (28317) = ____________________cc

2. Volume in the Containment Sump Calculation:

Volume in the Containment Sump = ____________________Gals X (3780 cc/gal)

= _____________________cc

3. Correct the Containment Atmosphere volume to STP using the equation below:

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \frac{14.2 + P_1}{14.2} \times \frac{492}{(T_1 + 460)}$$

Where:

2.97E+10 = the free Volume of the Containment in cc

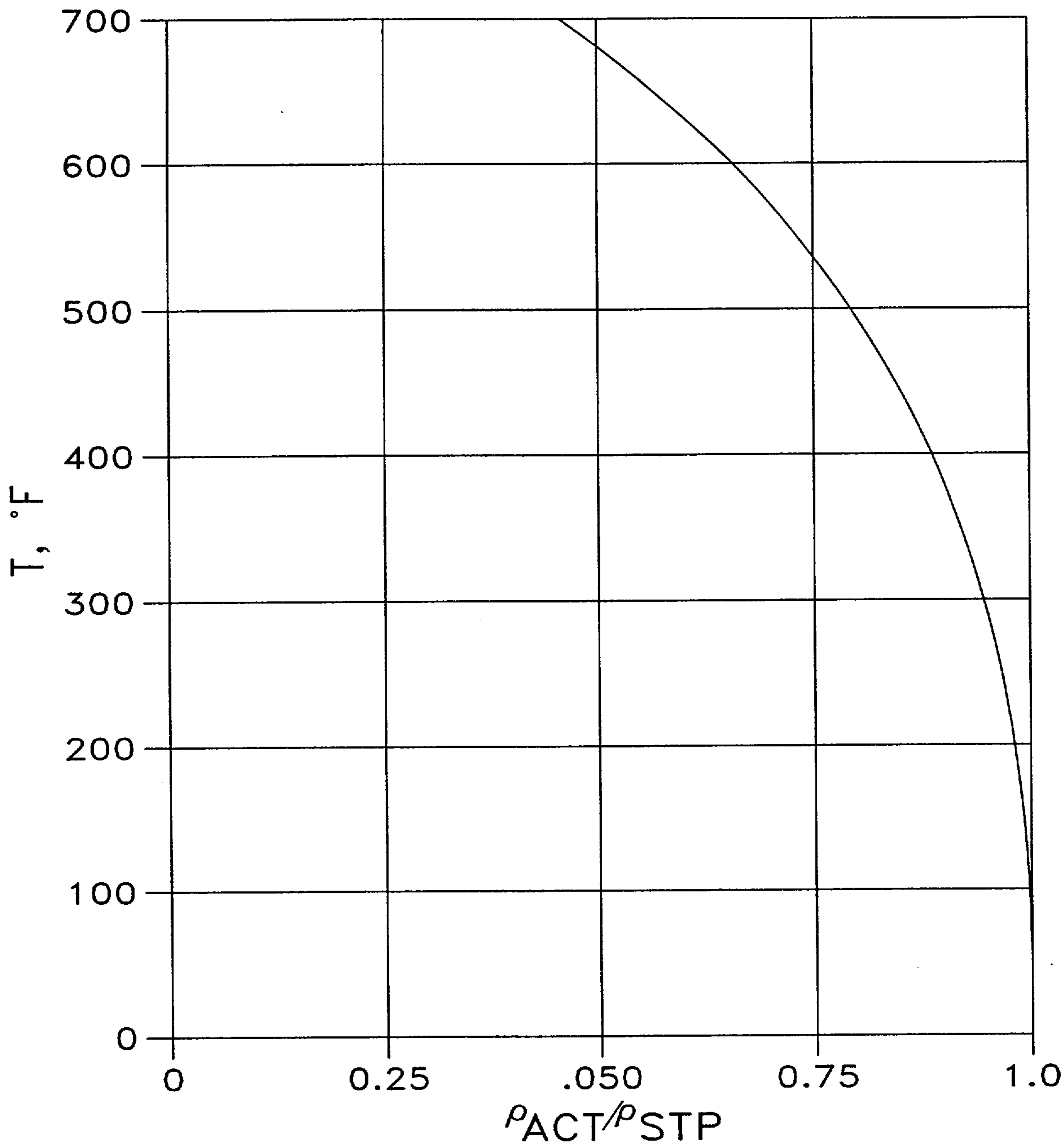
P_1 = Containment Pressure (psig) from Worksheet #2

T_1 = Containment Temperature (F) from Worksheet #2

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \frac{14.2 + \_\_\_\_\_\_\_\_}{14.2} \times \frac{492}{(\_\_\_\_\_\_\_\_ + 460)}$$

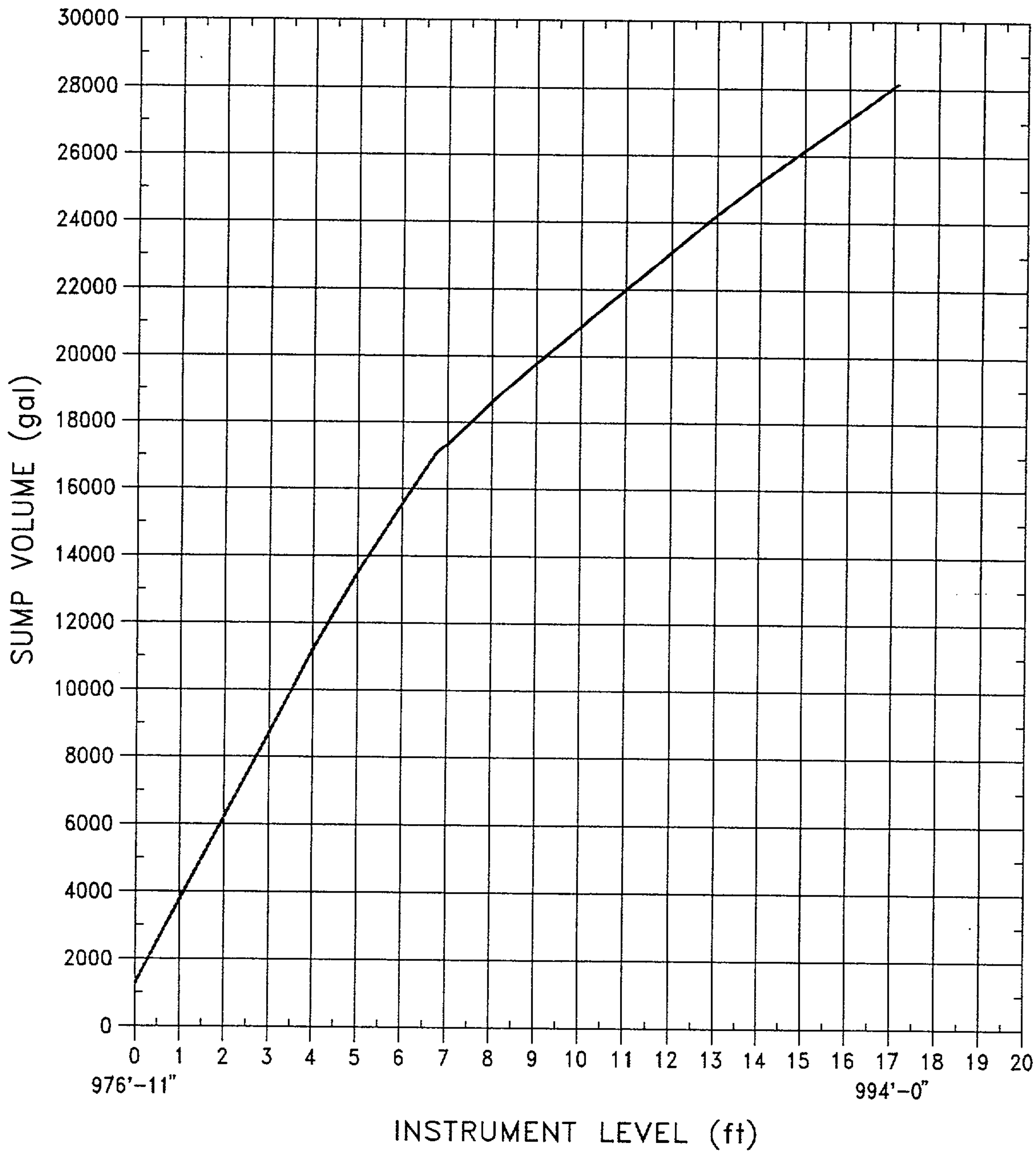
= ________________________cc

Worksheet #4, Figure #1, Ratio of H_20 Density to H2O Density at STP vs. Temperature



<u>Attachment 6.7 - Manual Core Damage Assessment</u> (Page 14 of 19)

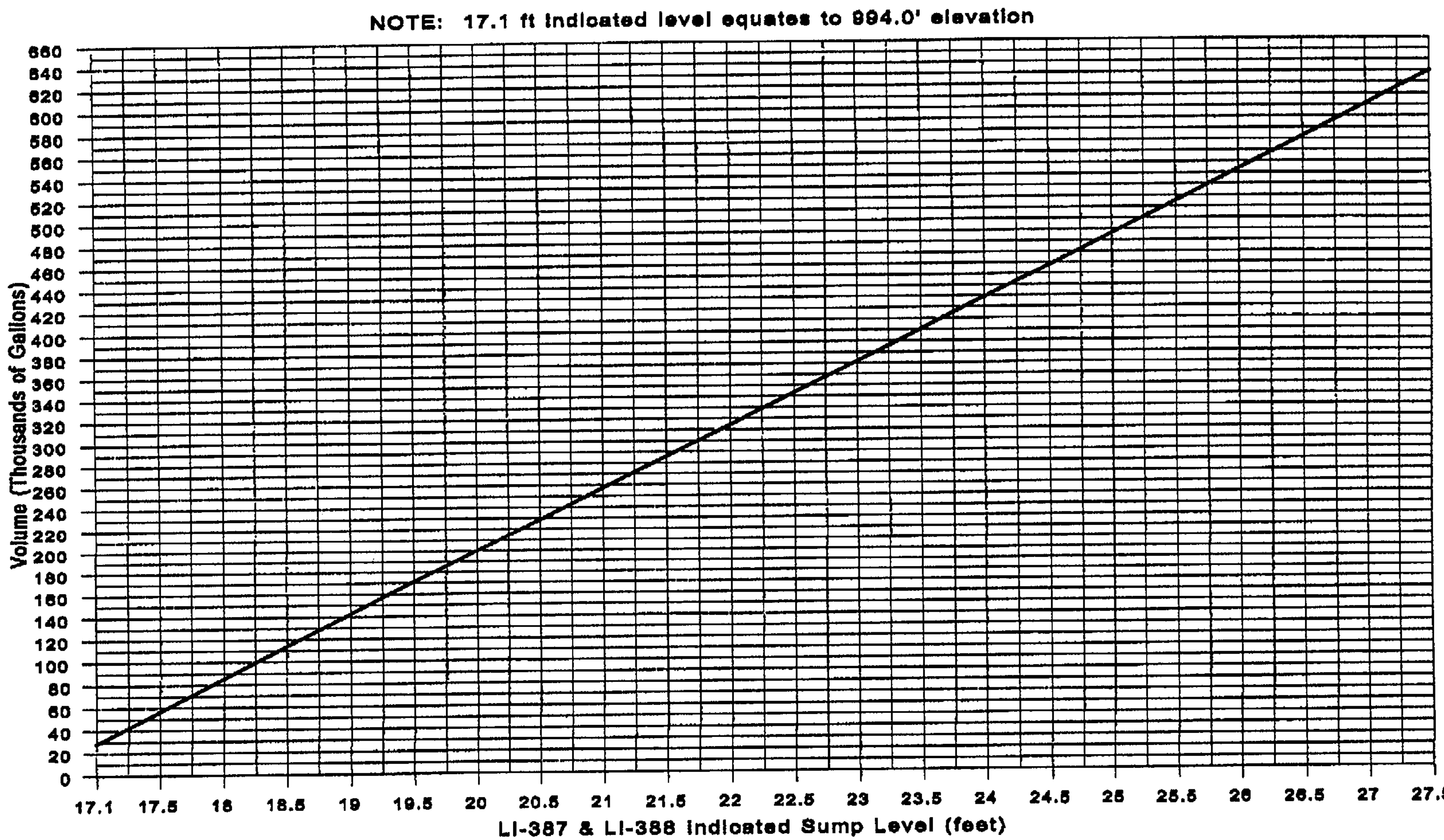
Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 1 of 2)



Attachment 6.7 - Manual Core Damage Assessment

(Page 15 of 19)

Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 2 of 2)



Attachment 6.7 - Manual Core Damage Assessment (Page 16 of 19)

Worksheet #5, Record of Release Quantity in the Containment

Column 1	Column 2	Column 3	Column 4	Column 5	Column 6
Isotope	Quantity in the Containment Sump (Curies)	Quantity in the RCS (Curies)	Quantity in the Containment Atmosphere (Curies)	Total Quantity in Containment (Curies)	Percent of Source Inventory
Gas Gap Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Fuel Pellet Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Cs-134					
Rb-88					
Te-129					
Te-132					
Sr-89					
Ba-140					
La-140					
La-142					
Pr-144					

Attachment 6.7, Manual Core Damage Assessment (Page 17 of 19)
Worksheet #6, Power Correction Factor (Page 1 of 3)

Column 1		Column 2	Column 3	Column 4
Isotopes (Fuel History Grouping)		Power Correction Factor	Equilibrium Source Inventory (Curies)[1]	Corrected Source Inventory
Gas Gap Inventory				
Kr-87	(2)		9.64E+05	
Xe-131m	(1)		2.17E+04	
Xe-133	(1)		3.84E+06	
I-131	(1)		1.93E+06	
I-132	(2)		2.79E+06	
I-133	(2)		3.95E+06	
i-135	(2)		3.70E+06	
Fuel Pellet Inventory				
Kr-87	(2)		1.83E+07	
Xe-131m	(1)		4.12E+05	
Xe-133	(1)		7.30E+07	
I-131	(1)		3.67E+07	
I-132	(2)		5.30E+07	
I-133	(2)		7.50E+07	
I-135	(2)		7.02E+07	
Cs-134	(1)		8.59E+07	
Rb-88	(2)		2.61E+07	
Te-129	(2)		1.16E+07	
Te-132	(1)		5.22E+07	
Sr-89	(1)		3.47E+07	
Ba-140	(1)		6.45E+07	
La-140	(1)		6.92E+07	
La-142	(2)		5.89E+07	
Pr-144	(2)		4.86E+07	

NOTE[1] Source Inventory based on 80% of values in EA-FC-90-111. Gas Gap Inventory is 5% of the source inventory (NUREG 1465). Fuel Pellet inventory is 0.95% of the Source Inventory.

Worksheet #6, Power Correction Factor (Page 2 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 1	PCF for Period 2	PCF for Period 3	PCF for Period 4
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j\,(1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j

d = the duration of the period j in seconds

λ = the decay constant (1/sec)

t = the time in seconds from the end of Power Period j to reactor shutdown

Worksheet #6, Power Correction Factor (Page 3 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 5	PCF for Period 6	PCF for Period 7	PCF for Period 8
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j\,(1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to reactor shutdown

Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage

NRC Category of Fuel Damage	Damage Mechanism	Release Mechanism	Release Source	Characteristic Isotopes	Characteristic Measurement	Measurement Range	Percentage Released From Source
① None	None	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137, Rb-88			<1
② Initial Cladding Failure	Rupture due to Gas Gap	Clad Burst and Gas Gap	Gas Gap	Xe-131m Xe-133 I-131 I-133	Maximum Core Exit	<1500°F*	<10
③ Intermediate Cladding Failure	Over Pressurization		Gas Gap		Thermocouple Temperature	<1700°F*	10 to 50
④ Major Cladding Failure			Gas Gap			≈< 2300°F ≈ <2% Oxidation	> 50
⑤ Initial Fuel Pellet Overheating	Loss of Structural Integrity Due to Fuel Cladding Oxidation	Grain Boundary Difusion	Fuel Pellet	Cs-134 Rb-88 Te-129 Te-132	Amount of H_2 Gas Produced (Equivalent to % of Core Oxidation)	Equivalent Core Oxidation < 3%	< 10
⑥ Intermediate Fuel Pellet Overheating			Fuel Pellet			Equivalent Core Oxidation < 18%	10 to 50
⑦ Major Fuel Pellet Overheating		Diffusional Release From UO_2 Grains	Fuel Pellet			Equivalent Core Oxidation < 65%	>50
⑧ Fuel Pellet Melt	Reactions between UO_2 and solid metallic zircaloy melting of control Rod materials, and zirconium	Escape From Molten Fuel	Fuel Pellet	Sr-89 Ba-140 La-140 La-142 Pr-144			<10
⑨ Intermediate Fuel Pellet Melt			Fuel Pellet				10 to 50
⑩ Major Fuel Pellet Melt			Fuel Pellet				> 50

* Depends on Reactor Pressure and Fuel Burn up. Values given for Pressure ≤1200 psia and Burn up ≥0.

OMAHA PUBLIC POWER DISTRICT

Confirmation of Transmittal for Emergency Planning Documents/Information

☐ Radiological Emergency Response Plan (RERP) ■ Emergency Plan Implementing Procedures (EPIP) ■ Emergency Planning Forms (EPF)

☐ Emergency Planning Department Manual (EPDM) ☐ Other Emergency Planning Document(s)/ Information

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The following document(s) / information is forwarded for your manual:

REMOVE SECTION	INSERT SECTION
EPIP Index page 1 of 2 issued 01/13/00	EPIP Index Page 1 of 2 issued 01/19/00
EPIP-TSC-8 R12 issued 01/13/00	EPIP-TSC-8 R13 Issued 01/19/00
FC-EPF Index Pages 1 issued 11/06/99 & 2 issued 09/30/99	FC-EPF Index Pages 1 & 2 issued 01/20/00
FC-EPF-38 R4 issued 09/30/99	FC-EPF-38 R5 issued 01/20/00

Summary of Changes:
EPIP-TSC-8 was revised to add three notes to permit the use of an Excel spreadsheet for trending RCS inventory, core damage assessment, and estimating the time to core uncovery. Minor editorial corrections.
On FC-EPF-38 in Section 7, Direction and Wind Speed were switched around to coincide with the order the information is provided by the National Weather Service and in Section 8, a note was added to recommend listening to a local radio station for emergency announcements.

Supervisor - Emergency Planning

I hereby acknowledge receipt of the above documents/information and have included them in my assigned manuals.

Signature: ______________________ **Date:** ______________

Please sign above and return by **03/19/00** to:

Karma Boone
Fort Calhoun Station, FC-2-1
Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

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EMERGENCY PLAN IMPLEMENTING PROCEDURE INDEX

PROCEDURE NUMBER	TITLE	REVISION/DATE
EPIP-OSC-1	Emergency Classification	R32 07-29-99
EPIP-OSC-2	Command and Control Position Actions/Notifications	R34 10-07-98a
EPIP-OSC-9	Emergency Team Briefings	R7 12-09-99
EPIP-OSC-15	Communicator Actions	R19 12-14-99
EPIP-OSC-20	Site Population Exposure Estimates	R6 11-10-95
EPIP-OSC-21	Activation of the Operations Support Center	R8 09-30-97
EPIP-TSC-1	Activation of the Technical Support Center	R20 10-08-99
EPIP-TSC-2	Catastrophic Flooding Preparations	(R0 03-22-95) DELETED 05-09-95
	REINSTATED	R2 02-06-96
EPIP-TSC-8	Core Damage Assessment	R13 01-19-00
EPIP-EOF-1	Activation of the Emergency Operations Facility	R11 09-23-99b
EPIP-EOF-3	Offsite Monitoring	R16 10-26-99
EPIP-EOF-6	Dose Assessment	R27 03-11-97a
EPIP-EOF-7	Protective Action Guidelines	R12 09-01-94
EPIP-EOF-10	Warehouse Personnel Decontamination Station Operation	R10 01-13-00
EPIP-EOF-11	Dosimetry Records, Exposure Extensions and Habitability	R18 09-18-97b
EPIP-EOF-19	Recovery Actions	R7 09-30-98
EPIP-EOF-21	Potassium Iodide Issuance	R3 09-18-97
EPIP-EOF-23	Emergency Response Message System	R5 10-12-99

Fort Calhoun Station
Unit No. 1

EPIP-TSC-8

EMERGENCY PLAN IMPLEMENTING PROCEDURE

Title: CORE DAMAGE ASSESSMENT

FC-68 Number: DCR 11544

Reason for Change: Add three notes to permit the use of an EXCEL spreadsheet for trending RCS inventory core damage assessment and estimating the time to core uncovery plus minor editorial corrections.

Initiator: Ron Meng

Preparer: Ron Meng

CORE UNCOVERY PREDICTIONS / CORE DAMAGE ASSESSMENTS

NON-SAFETY RELATED

1. PURPOSE

1.1 This procedure provides methods for trending Reactor Coolant System (RCS) volume above the active core during Loss of Coolant Accidents (LOCA), methods for estimating the time to core uncovery, and performing core damage assessments.

2. REFERENCES/COMMITMENTS DOCUMENTS

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, Task 467, July 1983

2.2 FC-0204-98, Source Terms Used in Emergency Plan Core Damage Assessment Spreadsheet (TSC-8)

2.3 EA-FC-90-94, Fuel Handling Accident and Bounding Source Term

2.4 NUREG 1465, Accident Source Terms for Light - Water Nuclear Power Plants

2.5 Calculation Number FC06727, Estimated Containment 994' Elevation Volume vs. Level for EPIP-TSC-8

3. DEFINITIONS

3.1 Time to Core Uncovery - A time calculated based on a snapshot of plant parameters that estimates the time remaining before the start of core uncovery (Reactor Vessel Level at the top of the active core).

3.2 Core Damage Assessment - The categorization of a core damage into four major types as follows: no fuel damage, fuel cladding failures, fuel pellet overheating and fuel pellet melting. The three later categories are delineated as initial, intermediate, and major. Each of the ten categories of core damage can be characterized by the type of fuel damage, the corresponding temperature range and the mechanism of fission product release.

3.3 Category of no fuel damage - that which is characterized by the release of fission products through the release mechanisms of Halogen Spiking and tramp uranium fission. The release source is the Gas Gap and the characteristic isotopes are I-131, Cs-137, and Rb-88.

3.4 Category of Fuel Cladding Failure - that which is characterized by the release of fission products through the release mechanisms of burst and Gas Gap diffusion. The release source is the Gas Gap. The characteristic fission products are noble gases and halogens (Xe-131m, Xe-133, I-131 and I-133) because they are volatile and can migrate quickly through the fuel pellet and gas gap for release following cladding rupture. These isotopes are volatile in the Temperature range of 1300-1800° F.

3.5 Category of Fuel Pellet Overheat - that which is characterized by the release of radioactivity through gain boundary diffusion and by diffusion from within the UO_2 grains. Grain boundary diffusion begins above 2450° F. Moderately volatile isotopes of cesium, rubidium and tellurium (Cs-134, Rb-88, Te-129, and Te-132) are characteristic of this type of damage.

3.6 Category of Fuel Pellet Melt - occurs at greater temperatures (2550-3450°F), reactions begin between the solid UO_2 and the solid metallic zircaloy, melting of the control rod materials, and melting of zirconium. At these temperatures greater amounts of tellurium are released. Alkali metals, such as barium, volatize as well as rare earths and actinides such as lanthanum and protactinium. These include the isotopes of Sr-89, Ba-140, La-140, La-142, and Pr-144.

3.7 Activity Ratios - theoretical calculations employed to determine typical ratios for isotopes of a fission product in the Gas Gap or the Fuel Pellet. Comparison of the ratios obtained from sample data with these calculated values determines the source of the fission product release. These ratios are made by comparing the noble gas/iodine isotopes in the sample to Xe-133/I-131 in the sample.

4. PREREQUISITES

4.1 This procedure is intended to be used during Loss of Coolant Accidents and other plant transients that may lead to core damage.

NOTE: Steps in this procedure may be done out of order or be completed concurrently.

NOTE: HJTCs may be deenergized per OI-RC-2A RCS Fill and Drain Operations causing RVLMS to indicate 100% while in reduced RCS inventory conditions. While the HJTCs are deenergized, use LI-119 and LI-197 and Attachment 6.1 -Figure #1, RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level for determining RCS volumes. Energizing HJTCs is acceptable for short periods to determine if a void exists in the Reactor Vessel per the precautions in OI-RC-2A.

5. PROCEDURE

NOTE: Trending the RCS Inventory above the active Core should be done if a significant decrease is observed in the RCS Inventory.

NOTE: Trending the RCS Inventory above the active Core may be performed using the Excel Spreadsheet "RCS_INV(R0)". This spreadsheet contains worksheets "RCS Inventory Data" and "RCS Inventory Chart" which are equivalent to Attachment 6.1 - Trending RCS Inventory above the Core.

5.1 Trend the RCS inventory volume above the core using Attachment 6.1 - Trending RCS Inventory above the Core.

NOTE: The time to core uncovery is determined using two methods. The first (normal) method, Attachment 6.2, determines a Net Loss/Gain in ft^3/min and uses this rate as the basis to determine the time remaining for Reactor Vessel Level to drop to the top of the active core. The second method, Attachment 6.3, uses Steaming Rate and RCS density to determine the boil-off rate in ft^3/min.

NOTE: The second method (Attachment 6.3) is used when the Reactor Vessel level is below that of any suspected leaks, the inventory is being lost is due to RCS boil-off, and it is determined that no make up flow is reaching the reactor vessel.

5.2 During Loss of Coolant Accidents, estimate the time to core uncovery by one of the following methods:

NOTE: Estimating the time to core uncovery, using the normal method, may be performed using the EXCEL spreadsheet "RCS_INV(R0)." This spreadsheet contains a worksheet labeled "Uncvy_Vol Chg" which is equivalent to Attachment 6.2 - Using the Change in RCS Volume vs. Time To Estimate the Time to Core Uncovery.

5.2.1 First (normal) method, complete Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery.

NOTE: Estimating the time to core uncovery, using the second method, may be performed using the EXCEL spreadsheet "RCS_INV(R0)". This spreadsheet contains a worksheet labeled "Uncvy_Stm Rate" which is equivalent to Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery.

5.2.2 Second (Steaming Rate) method, complete Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery.

NOTE: The prior power history for the last 30 days may be recorded on the EXCEL spreadsheet "Cor_dam(R0) .xls", worksheet "Prior Power History". The worksheet "Prior Power History" is equivalent to Attachment 6.4 - Prior 30-Day Power History.

5.3 Determine the power history for the last 30 days from the time of reactor trip/shutdown by completing Attachment 6.4 - Prior 30-Day Power History.

NOTE: Core Damage Assessments should be done when the Fuel Cladding Fission Product Barrier Criteria has been exceeded per EPIP-OSC-1, Attachment 6.3, Three Fission Product Barrier Criteria or at the discretion of the Command and Control position.

5.4 Complete Attachment 6.5 - Assessment of Core Damage Using Containment Radiation Dose Rates.

5.5 For slow transients only, complete Attachment 6.6 - Assessment of Core Damage using CETs.

NOTE: The total quantity of fission products available at different locations in the Containment may be changing due to transient plant conditions. Samples of the Reactor Coolant System / Discharge of the Low Pressure Safety Injection Pump and the Containment Atmosphere used for core damage assessments should be obtained within a minimum time and under stabilized plant conditions when possible.

5.6 Prepare to do a Core Damage assessment by Radiological Analysis of samples as follows:

5.6.1 Verify that plant conditions are stabilized as much as possible that plant conditions can support sampling operations for a Reactor Coolant System or Discharge of the Low Pressure Safety Injection Pump sample and/or a Containment Atmosphere sample.

5.6.2 Before the sampling team is dispatched to collect the required samples, hold a briefing with the sampling team and coordinate the collection of data that needs to be obtained when samples are isolated per Attachment 6.7, Worksheet #1 and #2.

NOTE: Manual core damage calculations may be performed using the EXCEL spreadsheet "Cor_dam(R0).xls". This spreadsheet "Cor_dam.xls" and its worksheets are equivalent to Attachment 6.7 - Manual Core Damage Assessment Calculations.

5.7 Complete Attachment 6.7 - Manual Core Damage Assessment Calculations.

5.8 Using the completed Attachment 6.7, make a conclusion on the extent of core damage using following three parameters:

5.8.1 Identification of the fission product isotopes that characterize the Reactor Coolant System / LPSI Pump and Containment Atmosphere Samples.

5.8.2 Identification of the source of the fission products (the Gas Gap or the Fuel Pellet) from Attachment 6.7 - Worksheets #4 and #5.

5.8.3 The quantity of the fission products available for release to the environment expressed as a percent of source inventory on Attachment 6.7 - Worksheet #7.

NOTE: Knowledgeable judgement is used to compare the three parameters with the definitions of the 10 NRC categories of fuel damage found on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage. Core Damage is not anticipated to take place uniformly. Therefore, when evaluating the three parameters listed above, the procedure is anticipated to yield a combination of one or more of the 10 categories defined on Attachment 6.8. The categories will exist simultaneously. The type of core damage is described in the 10 NRC categories listed on Attachment 6.8. The degree of core damage is described as the percent of the fission products in the source inventory that is now available in the Containment for release to the environment.

5.9 Based on the conclusions above, circle the categories of fuel damage on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage.

6. ATTACHMENTS

6.1 Trending RCS Volume above the Core

6.2 Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

6.3 Using Steaming Rate to Estimate the Time to Core Uncovery

6.4 Prior 30 Day Power History

6.5 Assessment of Core Damage Using Containment Radiation Dose Rates

6.6 Assessment of Core Damage using CETs

6.7 Manual Core Damage Assessment Calculations

6.8 Clad Damage Characteristics of NRC Categories of Fuel Damage

Attachment 6.1 - Trending RCS Inventory above the Core (Page 1 of 4)

1. On Worksheet 1 - RCS Inventory Data Sheet, record the Time, Pressurizer Level and RVLMS.
2. Record the RCS Volume above the Core on Worksheet 1 using the information from Figure 1, RCS Inventory above the Core vs. Pressurizer/RVLMS Level.
3. Plot the RCS Volume above the Core on Worksheet 2, Plot of RCS Volume above the Core vs. Time.

Attachment 6.1 - Trending RCS Inventory above the Core

Worksheet 1, RCS Inventory Data Sheet

Time	Pressurizer Level %	RVLMS %	Volume above the Core (ft^3)

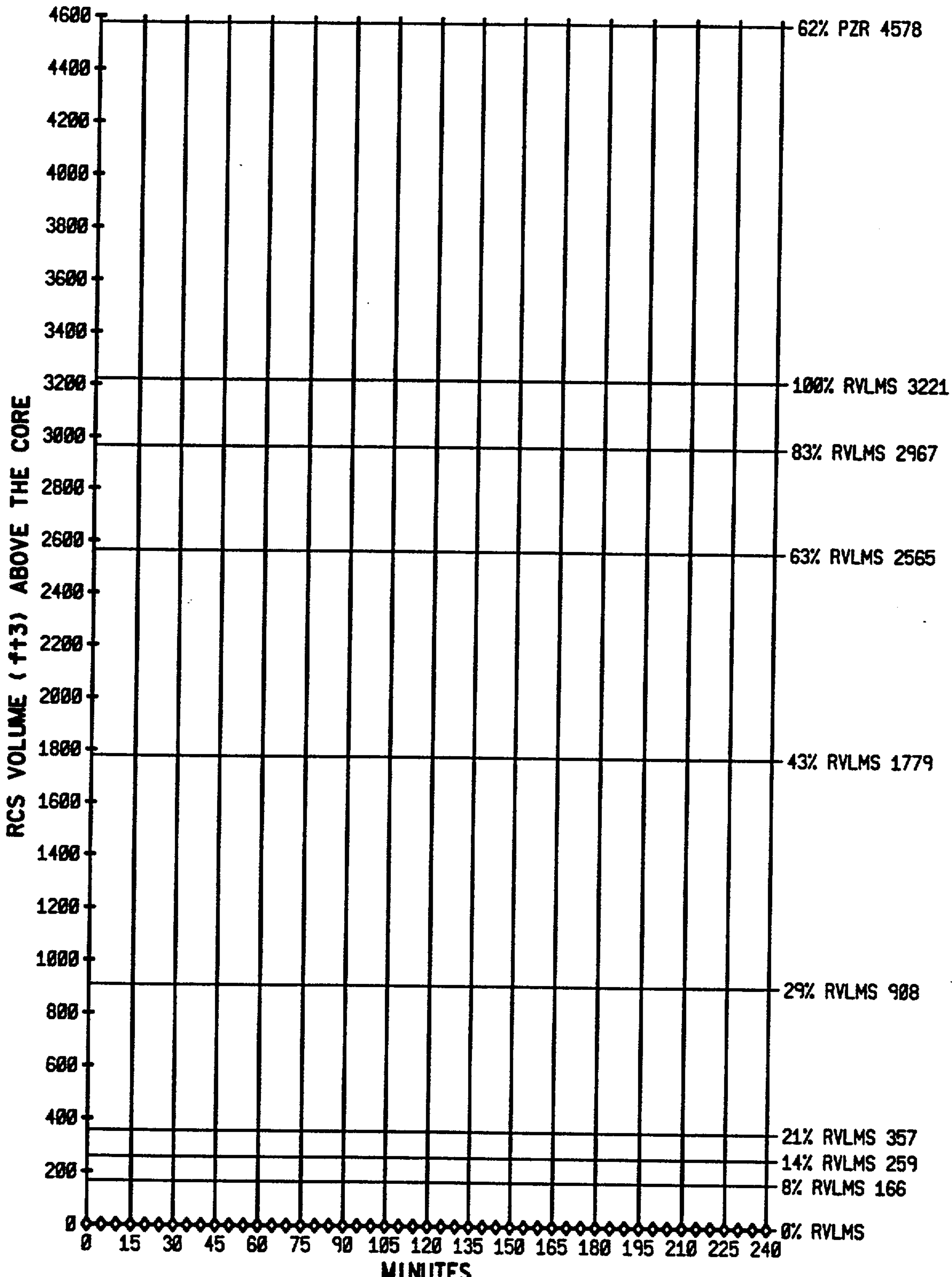
Attachment 6.1 - Trending RCS Inventory above the Core (Page 3 of 4)

Figure 1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level

PZR/RVLMS (% Level)	RCS INV. ABOVE TOP OF CORE (ft^3)	LI-119, LI-197 (Feet)	NOTES
100 PZR	4962		
90 PZR	4821		
80 PZR	4760		
70 PZR	4665		
65 PZR	4608		
60 PZR	4533		
55 PZR	4445		
50 PZR	4356		
45 PZR	4267		
40 PZR	4179		
35 PZR	4090		
30 PZR	4001		
25 PZR	3913		
20 PZR	3824		
15 PZR	3735		
10 PZR	3646		
5 PZR	3558		
0 PZR	3469		
100 RVLMS	3221	1018.31	Top of Upper Head
83 RVLMS	2967	1017.174	
63 RVLMS	2565	1014.018	
43 RVLMS	1779	1010.867	
29 RVLMS	908	1007.708	Top of Hot Leg
21 RVLMS	357	1006.375	Center of Hot Leg
14 RVLMS	259	1005.042	Bottom of Hot Leg
8 RVLMS	166	N/A	
0 RVLMS	0	N/A	Top of Active Core

Attachment 6.1 - Trending RCS Inventory above the Core (Page 4 of 4)

Worksheet 2 - Plot of RCS Volume above the Core vs. Time



Time Zero =__________

Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

NOTE: When a bubble is in the Reactor Vessel use RVLMS to determine the RCS volume (i.e., Pressurizer Level at 40% and RVLMS at 83%, a volume of 2967 ft^3 based on 83% RVLMS should be used).

1. Record Time, Pressurizer Level, RVLMS Level and the RCS Volume above the Core from Attachment 6.1, Figure 1 - RCS Inventory above the Core vs. Pressurizer/RVLMS Level.

   T_1 (Time)__________ Pressurizer Level (%)__________ RVLMS Level (%)__________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) __________

2. Wait for drop in Pressurizer/RVLMS Level and record the Time, Pressurizer Level, RVLMS Level, and RCS Volume below:

   T_2 (Time) Pressurizer Level (%) RVLMS Level (%)__________

   V_2 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) __________

3. Determine the Time in Minutes (T_M) to Core Uncovery using the equation below.

$$T_M = \frac{V_2\underline{\qquad\qquad} ft^3}{(V_1\underline{\qquad\quad} ft^3 - V_2\underline{\qquad\quad} ft^3) / (T_3\underline{\qquad\quad} \text{ minutes})} = \underline{\qquad\qquad}\text{Minutes}$$

   Where:

   T_3 = elapsed time in minutes ($T_2 - T_1$)

4. Add the time T_M to the time T_2 to determine the Projected Time of Core Uncovery.

   Projected Time of Core Uncovery = T_2__________ + T_M__________ = __________(hh:mm)

5. Remarks:

   ____
   ____
   ____

Done by:____________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 1 of 3)

1. Record Time, Pressurizer Level, RVLMS Level, RCS Volume from Attachment 6.1, Figure 1, Time of Reactor Trip/Shutdown, Steaming Rate from Attachment 6.3, Figure 1, Highest CET Temperature below and RCS Density from Attachment 6.3, Figure 2.

   T_1 (Time)__________ Pressurizer Level______________RVLMS Level______________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) ______________

   Time of Reactor Trip/Shutdown ______________hh:mm

   SR (Steaming Rate from Figure 1) _____________lb/sec

   Highest CET Temperature _________________°F

   D (RCS Density from Figure 2) ________________lb/ft^3

2. Determine the T_M (Minutes to Core Uncovery) using the equation below:

$$\text{Minutes to Core Uncovery} = \frac{(V_1\_\_\_\_\_\_ ft^3) * (D\_\_\_\_\_\_ lb/ft^3)}{(SR\_\_\_\_\_\_ lb/sec) * (60)} = \_\_\_\_\_\_\_\_ \text{Minutes}$$

3. Determine the Time of Core Uncovery using the equation below:

   Time of Core Uncovery = (T_1___________) + (T_M____________) = ___________(hh:mm)

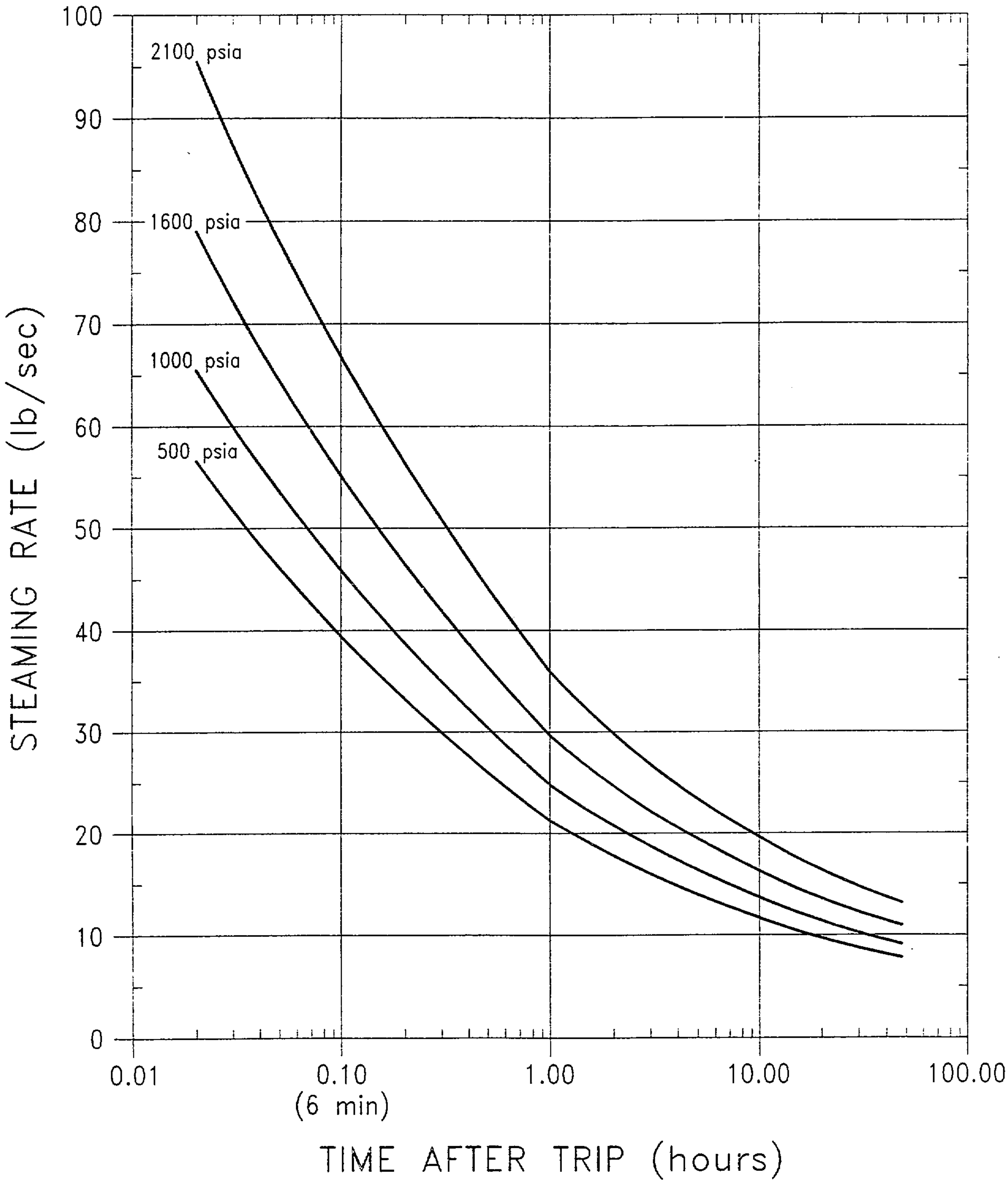
4. Remarks:

____
____
____
____

5. Done by:______________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 2 of 3)

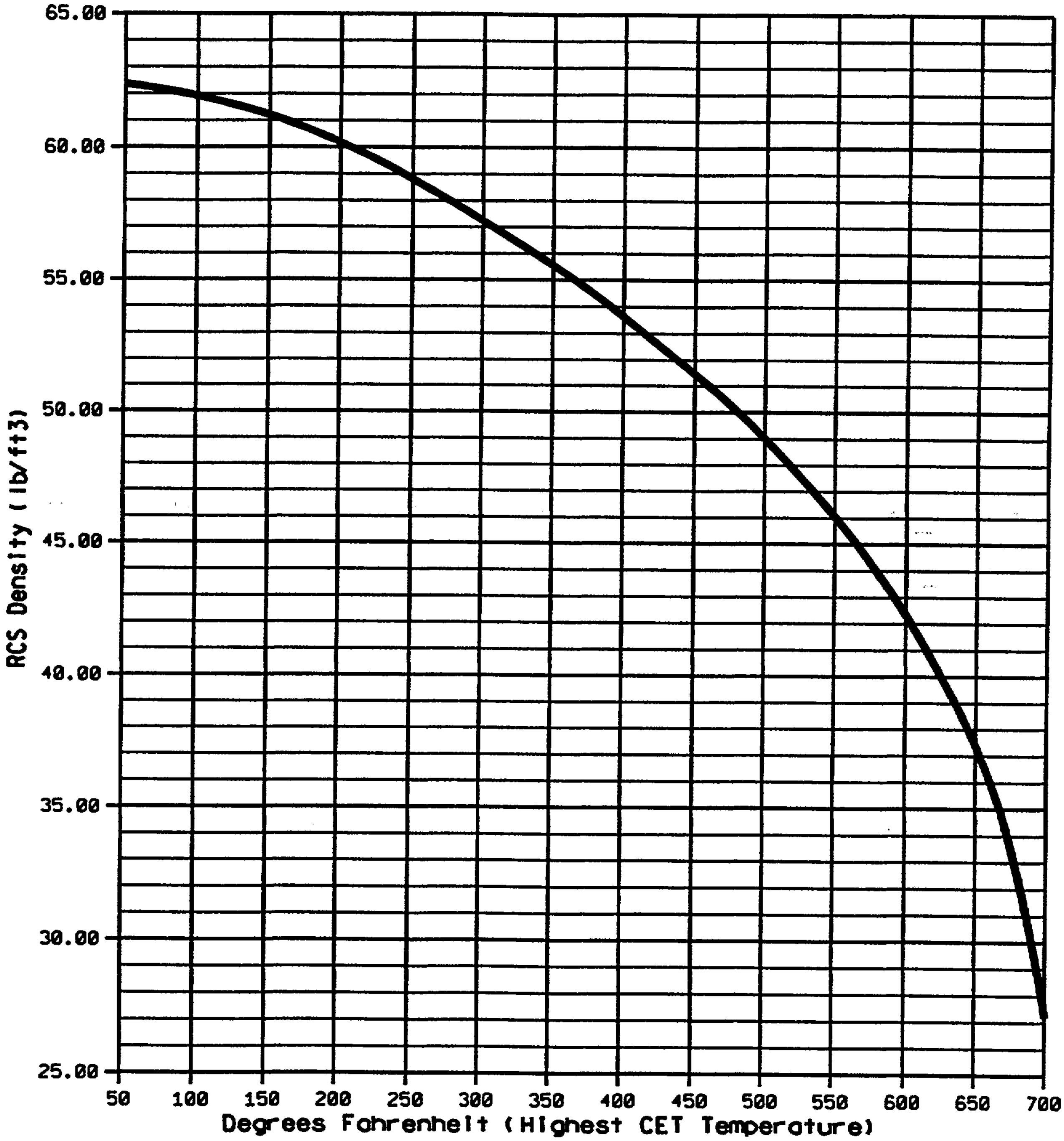
Figure 1, RCS Steaming Rate vs. Time After Trip



NOTE 1: If pressure is >2100 psia use the 2100 psia curve.
NOTE 2: If pressure =<2100 psia =>500 psia interpolate.
NOTE 3: If pressure <500 psia use the 500 psia curve.

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 3 of 3)

Figure 2 - RCS Density vs. CET Temperature



Attachment 6.4 - Prior 30 Day Power History

Date of Reactor Shutdown:	Time of Reactor Shutdown:

If reactor power has been steady state (+/- 10%) in the four days befoe the Reactor Shutdown, record the steady state power level:

If reactor power has been steady state (+/- 10%) in the 30 days before the Reactor Shutdown, record the steady state power level:

If the plant's power history has not been steady state for the the last 30 days, enter data below for each steady state power period during the last 30 days (record up to eight power periods where power has change more than +/- 10% power):

Power Period (j)	Steady State Power for Period (P_j)	Duration of Power Period (d)		Time From the End of Operating Period (P_j) To Reactor Shutdown (t)	
Number	Percent	Days	Seconds Days X 8.64E4	Days	Seconds Days X 8.64E4
1					
2					
3					
4					
5					
6					
7					
8					

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 1 of 3)

1. Record Date and Time of Reactor Shutdown: Date:__________Time:__________(T_1)

2. Date and Time of RM-091A and RM-091B Reading: Date:__________Time:__________(T_2)

3. Record: RM-091A Reading: ______________ RM-091B Reading:________________

4. Determine the Time Post Accident, Hours by subtracting the Date and Time of RM-091A and RM-091B Reading from the Date and Time of Reactor Shutdown.

   Time Post Accident, Hours = (T_2 - T_1) = ______________________________ = Hours

NOTE: Use the following information to estimate the 30-Day Average Power Level:

- The average power during the 30-day time period is not necessarily the most representative value for correction to equilibrium conditions.
- The last power levels at which the reactor operated should weigh more heavily in the judgement than earlier levels.
- Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.
- In the case in which the reactor has produced power for less than 30 days the procedure may be employed. However, the estimate of core damage obtained under this condition may be an under estimate of the actual condition.

5. Estimate the 30-Day Average Power Level, using the plants power history recorded on Attachment 6.4, engineering judgement, and the information in the note above.

6. Record the 30 Day Average Power Level based on engineering judgement = __________%.

7. Determine the Equilibrium Dose Rate as follows:

$$\text{Equilibrium Dose Rate} = \text{(Higher reading of RM-091 A or B)} \times \frac{100}{\text{(30 Day Average Power)}}$$

$$\text{Equilibrium Dose Rate} = (\_\_\_\_\_\_\text{R/hr}) \times \frac{100}{(\_\_\_\_\_\_\text{30 Day Average Power})} = \_\_\_\_\_\_\_\_\text{R/hr}$$

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 2 of 3)

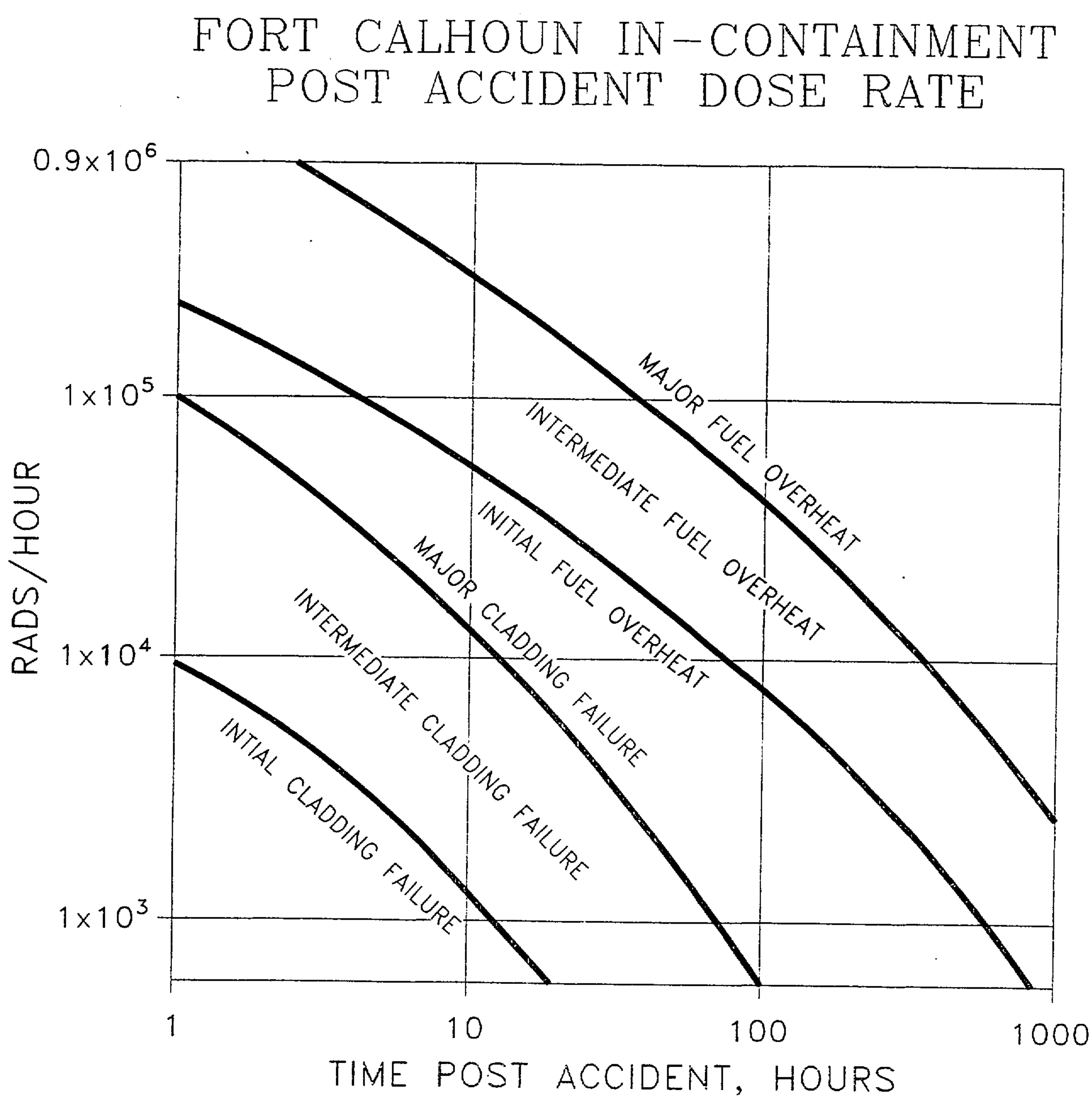
NOTE: Use the following information to consider which category of core damage is most representative of the particular value:

- Dose rate measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.
- This attachment may not be employed to estimate the degree of Fuel Pellet Melting.
- Dose rates significantly above the lower bound for the category of Major Fuel Over Heat may indicate concurrent Fuel Pellet Melting.
- This attachment may not be used to distinguish the relative contributions of the two categories (Cladding Failure and Fuel Pellet Overheat) to the total dose rate. This procedure does give the estimate of the highest category of damage.
- Dose rates within any category of fuel overheating may be anticipated to include concurrent Fuel Cladding Failure.
- Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Figure 1 - Categories of Core Damage Based on Containment Post Accident Dose Rates. The evaluation of other parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

8. On Figure 1, Category of Core Damage Based on Containment Post Accident Dose Rates, **plot** the Equilibrium Dose Rate as a function of the Time Post Accident, Hours.

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 3 of 3)

Figure 1- Category of Core Damage Based on Containment Post Accident Dose Rates



Attachment 6.6, Assessing Core Damage Using CETs (Page 1 of 3)

1. Record the following:

   1.1 Maximum CET Temperature: ______________________________F.

   1.2 Pressurizer Pressure at the time of Max CET Temperature: __________psia

   1.3 Time of the maximum CET Temperature: ______________________

2. Select the curve on Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure that represents a pressure approximately equal to or greater than the Pressurizer Pressure at the time of the Maximum CET Temperature. Read across the curve and record the Percent of Fuel Rods With Ruptured Clad Below:

   Percent of Fuel Rods With Ruptured Clad ________________________%

Attachment 6.6 - Assessing Core Damage Using CETs (Page 2 of 3)

NOTE: The Percent of Fuel Rods With Ruptured Clad obtained above is probably a lower limit of the estimate of damage. Some judgement on the bias is available as follows:

- This procedure applies most directly for slow core uncovery with a maximum temperature below the rapid oxidation temperature of 1800° F. A smooth rise in CET temperature and an uncovery duration of 20 minutes or longer are indicators for good prediction of clad ruptures.

- If pressure dropped to less than 100 psia within two minutes of the accident initiation, a large break is suggested. This causes undetected core heat up followed by flashing during a refill. Depending on the rate of refill, the CET temperature may rise rapidly then quench when the core is recovered. This procedure would then yield a very low estimate for the percent of fuel rods ruptured.

- If pressure was above 1650 psia, it could exceed the fuel rod internal gas pressure depending on rod burn up, causing clad collapse onto the fuel pellet instead of outward clad ballooning. The clad rupture criteria are less well defined for such conditions, but at temperatures above 1800° F where the highest pressure curve applies on Figure 1, clad failure sufficient to release fission gas is likely and this procedure may be used to obtain estimates of damage.

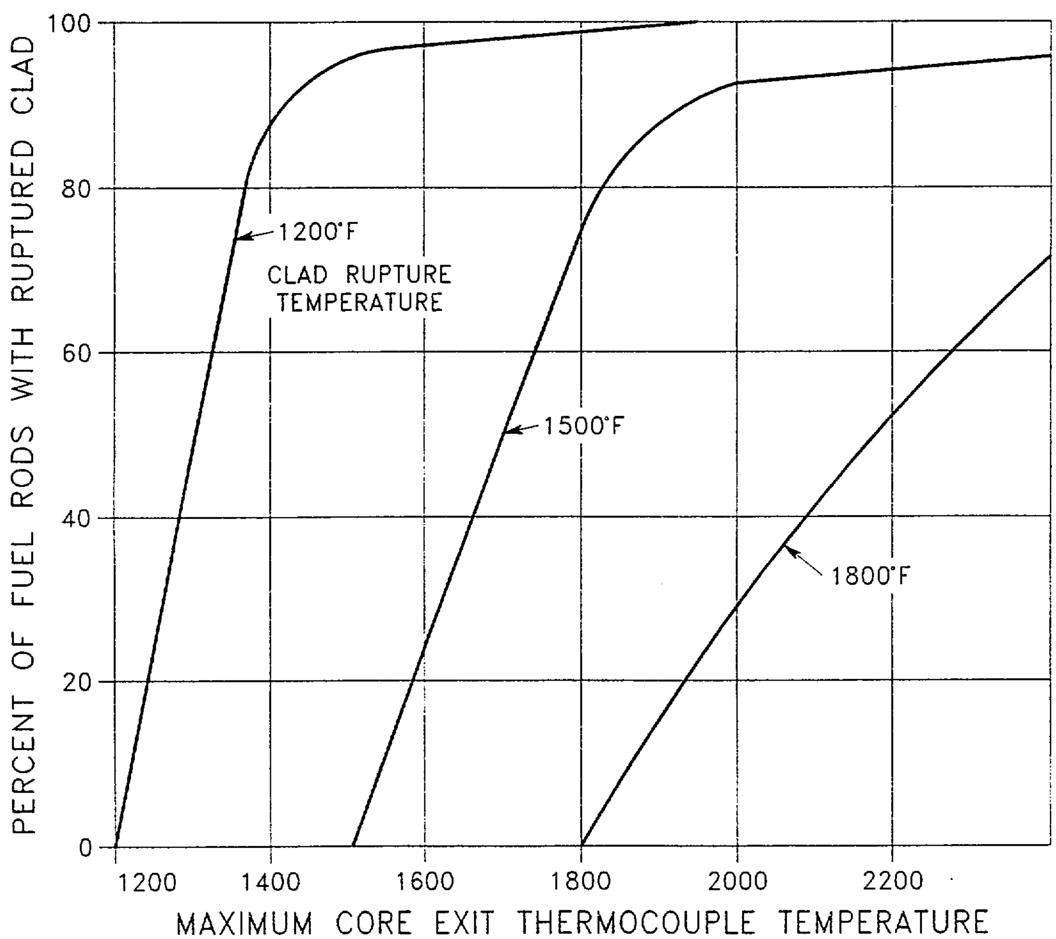
3. Comment on the probable bias in the result of Percent of Fuel Rods With Rupture Clad:
____
____
____
____
____
____
____

4. On Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage, determine and underline the NRC Category of Fuel Damage and its Characteristics.

Attachment 6.6 - Assessing Core Damage Using CETs (Page 3 of 3)

Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure

WHEN THE PRESSURE IS :	USE CURVE LABELED WITH TEMPERATURE :
≤ 100 PSIA	1200°F
≤ 1200 PSIA	1500°F
≤ 1650 PSIA	1800°F



Attachment 6.7 - Manual Core Damage Assessment (Page 1 of 19)

1. On Worksheet #1, Section 1 complete the following:

1.1 At the time of the Reactor Coolant Sample / LPSI Pump sample isolation, record the Date/Time, RCS Pressure, RCS Temperature, Pressurizer Level, RVLMS, and Containment Sump Level.

1.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Reactor Coolant Sample / LPSI Pump sample isolation.

1.3 Record the Sample Specific Activities from the Reactor Coolant / Discharge of the LPSI Pump sample.

1.4 Record the Temperature Correction Factor as follows:

1.4.1 For RCS samples obtain the temperature correction factor from Figure #1- Ratio of H_20 Density to H_20 Density at STP vs. Temperature.

1.4.2 For the Discharge of the LPSI Pump samples record a temperature correction factor of 1.0.

1.5 Multiply the Sample Specific Activity by the Temperature Correction Factor and record the Temperature Corrected Specific Activity (A_T).

1.6 Determine and record the Decay Corrected Specific Activity using the following equation:

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

Where:

A_0 = the Decay Corrected Specific Activity in µCi/cc.

A_T = the Temperature Corrected Specific Activity in µCi/cc.

λ = the radioactive decay constant, 1/sec.

t = the time from a reactor shutdown to sample isolation in seconds.

Attachment 6.7 - Manual Core Damage Assessment (Page 2 of 19)

2. On Worksheet #2, complete the following:

   2.1 At the time of the Containment Atmosphere sample isolation, record the Date/Time, Containment Pressure and the Containment Temperature.

   2.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Containment Atmosphere sample.

   2.3 Determine the Temperature/Pressure Correction Factor for the Containment using the equation below and record on Worksheet #3, Section 1:

$$\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492}$$

Where:

P_1 = Containment pressure (psig) at time of Containment Atmosphere sample isolation

T_1 = Containment Temperature at the time of the Containment Atmosphere sample isolation

   2.4 Correct the Sample Specific Activities to standard Temperature and Pressure by multiplying the Sample Specific Activity by the Temperature/Pressure Correction Factor and record in the Column labeled Temperature/Pressure Corrected Specific Activity.

   2.5 Correct and record the Decay Corrected Specific Activities for decay back to the time of reactor shutdown using the following equation:

$$A_0 = \frac{A_{PT}}{e^{-\lambda t}}$$

Where:

A_0 = the Decay Corrected Specific Activities in µCi/cc.

A_{PT} = the Temperature Pressure Corrected Specific Activity in µCi/cc.

λ = the radioactive decay constant, 1/sec.

t = the time from a reactor shutdown to sample isolation in seconds.

3. On Worksheet #3, Identify the Fission Product Release Source (Gas Gap or the Fuel Pellet) as follows:

3.1 Copy the Decay Corrected Specific Activities from Worksheet #1 for KR-87, Xe-131m, Xe-133, I-131, I-132, I-133, and I-135.

3.2 Copy the Decay Corrected Specific Activity from Worksheet #2 for KR-87, Xe-131m, Xe-133, I-131, I-132, I-133, and I-135.

3.3 Calculate the noble gas and iodine ratios using the equations below:

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Decay Corrected Specific Activity}}{\text{Xe-133 Decay Corrected Specific activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Decay Corrected Specific Activity}}{\text{I-131 Decay Corrected Specific Activity}}$$

NOTE: Select as the source of the release that ratio that is closest to the Sample Isotope Ratio. An accurate comparison is not anticipated.

3.4 Determine the Identified Source (Fuel Pellet or Gas Gap) of the release by comparing the Sample Isotope Ratio results with the predicted isotope ratios for the Fuel Pellet and the Gas Gap Inventories. Record the source in the column labeled Identified Source Fuel Pellet or Gas Gap.

4. On Worksheet #4, determine the Volumes of the Reactor Coolant System, Containment Sumps and Containment Atmosphere as follows:

4.1 Determine the volume (ft^3) of the RCS by using the Pressurizer Level and RVLMS level recorded on Worksheet #1 and Attachment 6.1, Figure #1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level, record the volumes in Section 1.

4.2 Determine and record the Density Correction in Section 1 as follows:

4.2.1 If an RCS Sample was obtained, obtain density correction from Figure #1 - Ratio of H_20 Density to H_20 Density at STP vs. Temperature by using the RCS Temperature recorded on Worksheet #1, at the time of the RCS sample.

4.2.2 If a sample was obtained from the discharge of the LPSI Pump, enter a Density Correction of 1.0 in Section 1.

4.3 Complete the RCS Volume Calculation in Section 1.

4.4 Determine and record the volume in the Containment Sump by using the Containment Sump Level recorded on Worksheet #1 and Figure #2 - Containment Sump/Basement Volume vs. L-387 or L-388 in Section 2. Record the volume in gallons in Section 2 of Worksheet #4.

4.5 Complete the calculation for the Containment Sump Volume in Section 2 on Worksheet #4.

4.6 Correct the Containment Atmosphere Volume to STP in Section 3 using the Containment Pressure and Temperature on Worksheet #2 and the following equation:

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \left(\frac{14.2 + P_1}{14.2}\right) \times \left(\frac{492}{T_1 + 460}\right)$$

Where:

2.97E+10 = the free Volume of the Containment in cc

P_1 = Containment Pressure (psig)

T_1 = Containment Temperature (° F)

NOTE: The Specific activities in the Reactor Coolant System and the Containment Sump are assumed to be equal. The results of the RCS/Discharge of the LPSI pump sample may be used for both the Containment Sump and Reactor Coolant System calculations below.

5. On Worksheet #5, calculate the Total Quantity (curies) in Containment as follows:

5.1 Using the equation below calculate the Quantity (Curies) in the Containment Sump for each isotope and record in Column #2.

$$\text{Curies in the Containment Sump} = \frac{SA}{1.0E+6} \times \text{CS Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #1

CS Vol = the Containment Sump Volume From Worksheet #4

1.0E+6 = Constant to convert μCurie to Curies

5.2 Using the equation below calculate the Quantity (Curies) in the Reactor Coolant System for each isotope and record in Column #3:

$$\text{Curies in the Reactor Coolant System} = \frac{SA}{1.0E+6} \times \text{RCS Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #1

RCS Vol = the Reactor Coolant Volume From Worksheet #4

1.0E+6 = Constant to convert μCuries to Curies

5.3 Using the equation below calculate the Quantity (Curies) in the Containment Atmosphere for each isotope and record in Column #3:

$$\text{Curies in the Containment Atmosphere} = \frac{SA}{1.0E+6} \text{ X CA Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #2

CA Vol = the Containment Atmosphere Volume From Worksheet #4

1.0E+6 = Constant to convert µCurie to Curies

5.4 Add Columns 2, 3, and 4. Record the sum (Total Quantity in Containment (Curies)) in Column 5.

6. On Worksheet #6, determine the Power Correction Factor and correct the Equilibrium Source Inventory for power history as follows:

6.1 Using the power history data recorded on Attachment 6.4, determine the Power Correction Factor as follows:

6.1.1 If the reactor power has been steady state (+/- 10%) for the four (4) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (2) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 4 Days}}{100}$$

6.1.2 If the reactor power has been steady state (+/- 10%) for the thirty (30) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (1) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 30 Days}}{100}$$

NOTE: The following step is used to figure out the Power Correction Faction for those isotopes in the Fuel History Groups not determined in steps 6.1.1 and 6.1.2 above.

6.2 If the reactor power has not been steady state (+/- 10%), determine the Power Correction Factor for each of the applicable power periods listed on Attachment 6.4, and record on Pages 2 and 3 of Worksheet #6 using the equation below:

$$PCF = \frac{P_j\,(1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to a reactor shutdown

6.3 For nonsteady state power histories, Sum up the power correction factors for each isotope on Pages 2 and 3 of Worksheet #6 and record on Page 1, Column #2 of Worksheet #6.

6.4 Correct the Equilibrium Source Inventory for power history, by multiplying the Power Correction Factor by the Equilibrium Source Inventory and record in Column 4 of Worksheet #6.

7. Determine and record the Percent of Source Inventory in Column 6 of Worksheet #5 using the equation below:

$$PSI(\%) = \frac{TQC}{CSI} \times 100$$

Where:

PSI(%) = Percent of Source Inventory in %

TQC = Total Quantity in Containment in Column 5 on Worksheet #5

CSI = Corrected Source Inventory in Column 4 of Worksheet #6

100 = Constant used to change a result to a percent

Attachment 6.7 - Manual Core Damage Assessment

Worksheet #1, Measured Specific Activity (μCi/cc) and Sample Temperature Correction for RCS/LPSI Sample

At the time RCS / LPSI Pump sample isolation record the following:

Date/Time_____________ RCS Pressure _____________ RCS Temperature _____________

Pressurizer Level __________ RVLMS __________ Containment Sump Level _____________

Time in Seconds between reactor shutdown and RCS / LPSI Pump sample isolation: __________(seconds)

Temperature Correction Factor from Step 1.4: _____________

	Sample Specific Activity	Temperature Corrected Specific Activity (A_T)	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_T}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Worksheet #2, Measured Specific Activity (µCi/cc) and Sample Temperature Pressure Correction for Containment Atmosphere Sample

At the time of Containment Atmosphere sample isolation record the following: Date/Time__________ Containment Pressure ________ Containment Temperature _______
Time in seconds between reactor shutdown and Containment Sample isolation (t): _________
Determine Temperature / Pressure Correction Factor: $\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492} = \frac{14.2}{(\quad + 14.2)} \times \frac{(\quad + 460)}{492} = \_\_\_\_\_\_\_\_$

Isotope	Containment Atmosphere Sample Specific Act.	Temperature Pressure Corrected Spec. Act. (A_{PT})	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_{PT}}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Attachment 6.7 - Manual Core Damage Assessment (Page 11 of 19)

Worksheet #3 - Fission Product Release Source Identification

Fission Product Release Source Identification for RCS/LPSI Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #1	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Fission Product Release Source Identification for Containment Atmosphere Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #2	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Attachment 6.7, Manual Core Damage Assessment (Page 12 of 19)

Worksheet #4 - RCS, Containment Sump and Containment Atmosphere Volume Calculations

1. RCS Volume Calculation:

RCS Volume above the Core (V) = ________ ft^3 Density Correction (D) = ____________

RCS Volume = $(V + 1823\ ft^3) \times (D) \times (28317\ cc/ft^3)$

Where:

V = RCS Volume above the Core

$1823\ ft^3$ = RCS Volume Below the Top of the Active Core

D = the Density Correction

RCS Volume = (__________ + 1823) X (__________) X (28317) = __________________ cc

2. Volume in the Containment Sump Calculation:

Volume in the Containment Sump = ____________________ Gals X (3780 cc/gal)

= ____________________ cc

3. Correct the Containment Atmosphere volume to STP using the equation below:

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \left(\frac{14.2 + P_1}{14.2}\right) \times \left(\frac{492}{(T_1 + 460)}\right)$$

Where:

2.97E+10 = the free Volume of the Containment in cc

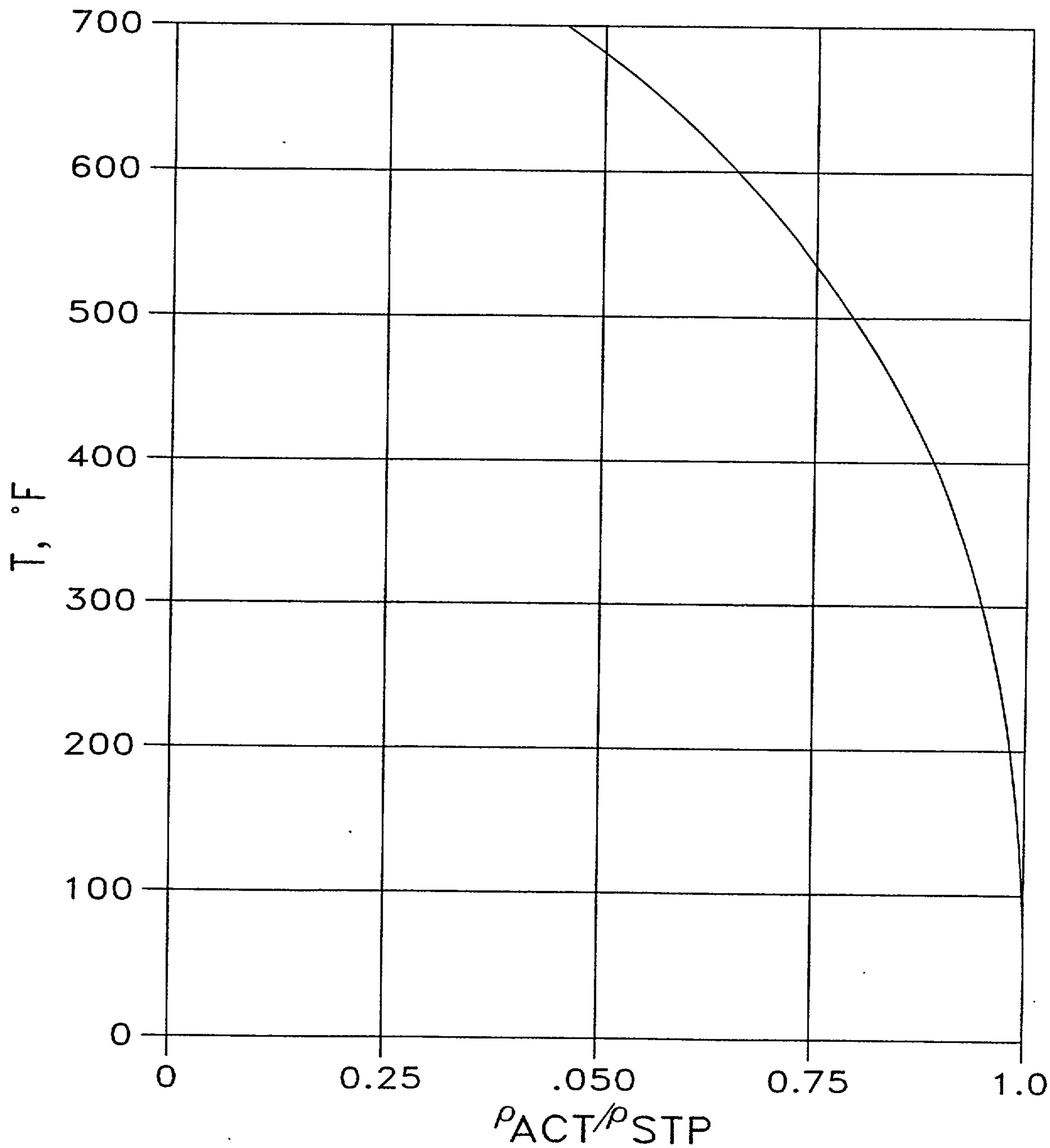
P_1 = Containment Pressure (psig) from Worksheet #2

T_1 = Containment Temperature (F) from Worksheet #2

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \left(\frac{14.2 + \_\_\_\_\_\_\_\_}{14.2}\right) \times \left(\frac{492}{(\_\_\_\_\_\_\_\_ + 460)}\right)$$

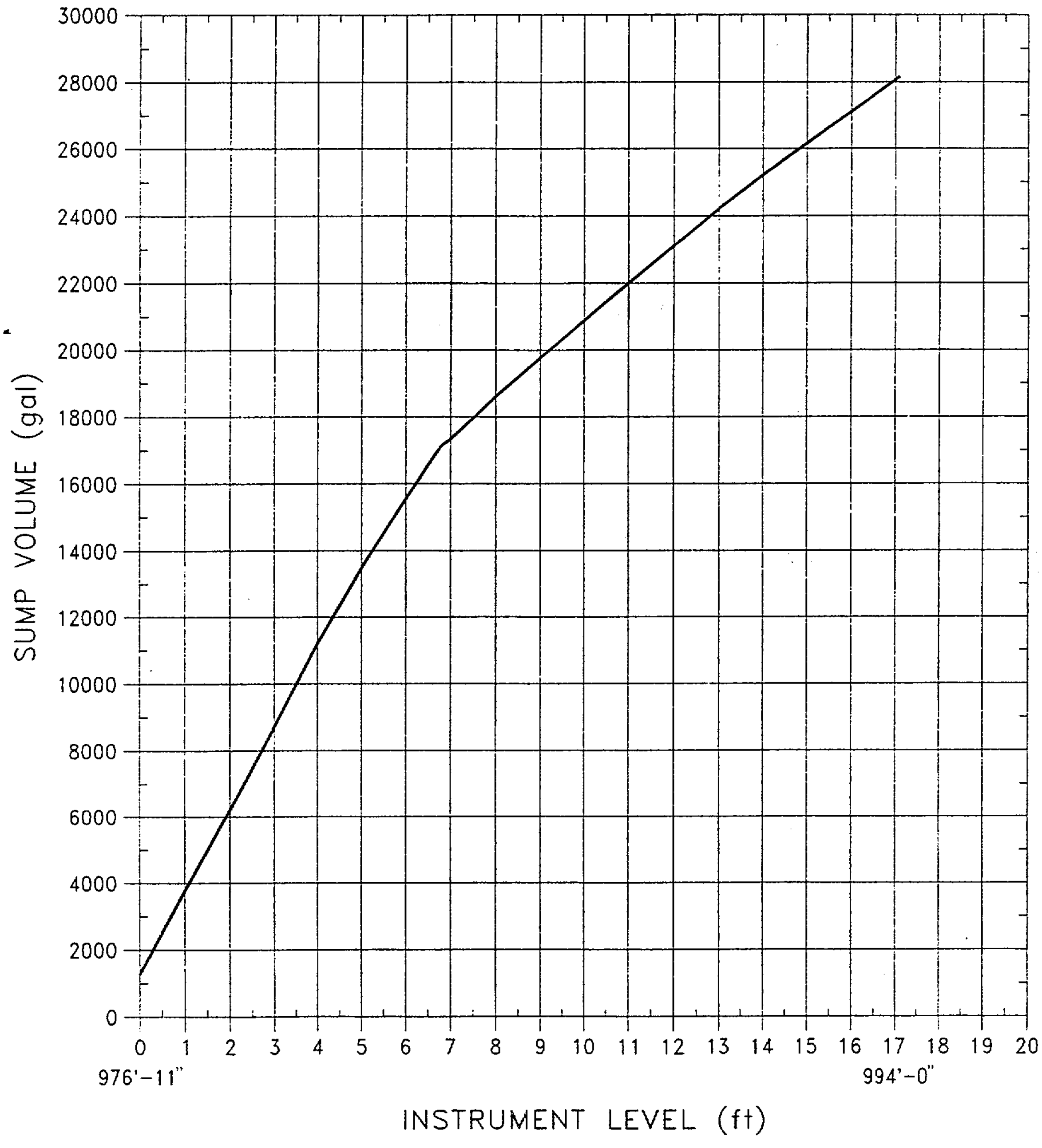
= ______________________ cc

Worksheet #4, Figure #1, Ratio of H_20 Density to H2O Density at STP vs. Temperature



Attachment 6.7 - Manual Core Damage Assessment (Page 14 of 19)

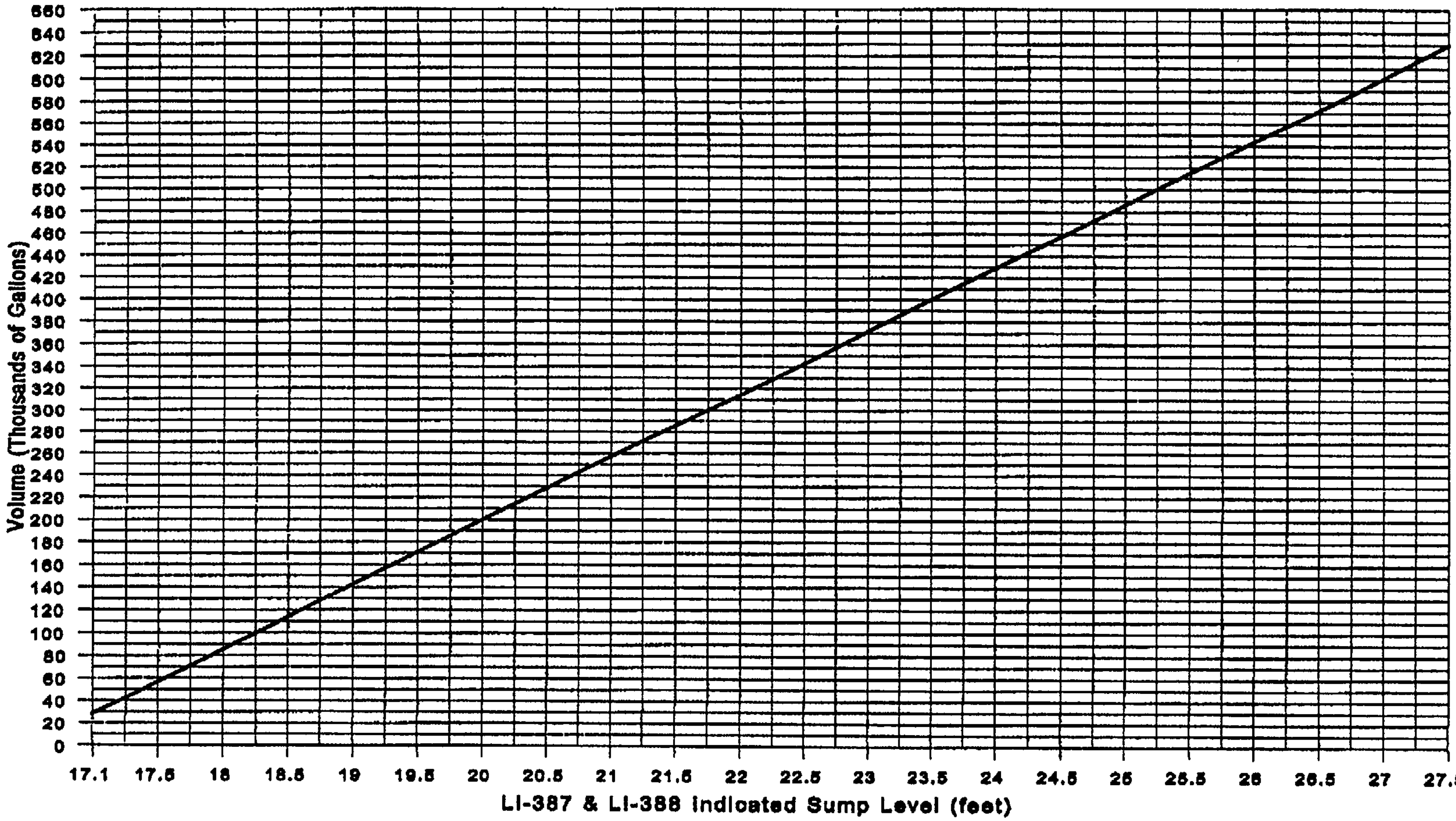
Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 1 of 2)



Attachment 6.7 - Manual Core Damage Assessment

Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 2 of 2)

NOTE: 17.1 ft indicated level equates to 994.0' elevation



Attachment 6.7 - Manual Core Damage Assessment (Page 16 of 19)

Worksheet #5, Record of Release Quantity in the Containment

Column 1	Column 2	Column 3	Column 4	Column 5	Column 6
Isotope	Quantity in the Containment Sump (Curies)	Quantity in the RCS (Curies)	Quantity in the Containment Atmosphere (Curies)	Total Quantity in Containment (Curies)	Percent of Source Inventory
Gas Gap Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Fuel Pellet Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Cs-134					
Rb-88					
Te-129					
Te-132					
Sr-89					
Ba-140					
La-140					
La-142					
Pr-144					

Attachment 6.7, Manual Core Damage Assessment (Page 17 of 19)

Worksheet #6, Power Correction Factor (Page 1 of 3)

Column 1		Column 2	Column 3	Column 4
Isotopes (Fuel History Grouping)		Power Correction Factor	Equilibrium Source Inventory (Curies)[1]	Corrected Source Inventory
Gas Gap Inventory				
Kr-87	(2)		9.64E+05	
Xe-131m	(1)		2.17E+04	
Xe-133	(1)		3.84E+06	
I-131	(1)		1.93E+06	
I-132	(2)		2.79E+06	
I-133	(2)		3.95E+06	
i-135	(2)		3.70E+06	
Fuel Pellet Inventory				
Kr-87	(2)		1.83E+07	
Xe-131m	(1)		4.12E+05	
Xe-133	(1)		7.30E+07	
I-131	(1)		3.67E+07	
I-132	(2)		5.30E+07	
I-133	(2)		7.50E+07	
I-135	(2)		7.02E+07	
Cs-134	(1)		8.59E+07	
Rb-88	(2)		2.61E+07	
Te-129	(2)		1.16E+07	
Te-132	(1)		5.22E+07	
Sr-89	(1)		3.47E+07	
Ba-140	(1)		6.45E+07	
La-140	(1)		6.92E+07	
La-142	(2)		5.89E+07	
Pr-144	(2)		4.86E+07	

NOTE[1] Source Inventory based on 80% of values in EA-FC-90-111. Gas Gap Inventory is 5% of the source inventory (NUREG 1465). Fuel Pellet inventory is 0.95% of the Source Inventory.

Attachment 6.7 - Manual Core Damage Assessment (Page 18 of 19)

Worksheet #6, Power Correction Factor (Page 2 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 1	PCF for Period 2	PCF for Period 3	PCF for Period 4
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j\,(1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j

d = the duration of the period j in seconds

λ = the decay constant (1/sec)

t = the time in seconds from the end of Power Period j to reactor shutdown

Attachment 6.7 - Manual Core Damage Assessment
Worksheet #6, Power Correction Factor (Page 3 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 5	PCF for Period 6	PCF for Period 7	PCF for Period 8
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j (1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to reactor shutdown

Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage

NRC Category of Fuel Damage	Damage Mechanism	Release Mechanism	Release Source	Characteristic Isotopes	Characteristic Measurement	Measurement Range	Percentage Released From Source
① None	None	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137, Rb-88			<1
② Initial Cladding Failure	Rupture due to Gas Gap	Clad Burst and Gas Gap	Gas Gap	Xe-131m Xe-133 I-131 I-133	Maximum Core Exit	<1500°F*	<10
③ Intermediate Cladding Failure	Over Pressurization		Gas Gap		Thermocouple Temperature	<1700°F*	10 to 50
④ Major Cladding Failure			Gas Gap			≈< 2300°F ≈ <2% Oxidation	> 50
⑤ Initial Fuel Pellet Overheating	Loss of Structural Integrity Due to Fuel Cladding Oxidation	Grain Boundary Difusion	Fuel Pellet	Cs-134 Rb-88 Te-129 Te-132	Amount of H_2 Gas Produced (Equivalent to % of Core Oxidation)	Equivalent Core Oxidation < 3%	< 10
⑥ Intermediate Fuel Pellet Overheating			Fuel Pellet			Equivalent Core Oxidation < 18%	10 to 50
⑦ Major Fuel Pellet Overheating		Diffusional Release From UO_2 Grains	Fuel Pellet			Equivalent Core Oxidation < 65%	>50
⑧ Fuel Pellet Melt	Reactions between UO_2 and solid metallic zircaloy melting of control Rod materials, and zirconium	Escape From Molten Fuel	Fuel Pellet	Sr-89 Ba-140 La-140 La-142 Pr-144			<10
⑨ Intermediate Fuel Pellet Melt			Fuel Pellet				10 to 50
⑩ Major Fuel Pellet Melt			Fuel Pellet				> 50

* Depends on Reactor Pressure and Fuel Burn up. Values given for Pressure ≤1200 psia and Burn up ≥0.

EMERGENCY PLAN FORMS INDEX
FC-EPF

EMERGENCY PLAN FORMS INDEX
FC-EPF

PROCEDURE NUMBER	TITLE	REVISION/DATE
FC-EPF-21	Fort Calhoun Station Sector Map	R2 05-15-97
FC-EPF-27	Onsite/Offsite Dose Comparison Data Record (Using Eagle Program)	R2 11-02-95
FC-EPF-29	Estimation of Unmonitored Release Rates	R1 12-30-93
FC-EPF-31	ΔT - Stability Class - $\chi\mu/Q$	R1 07-25-95
FC-EPF-32	Area Monitor Trending	R0 06-10-93
FC-EPF-33*	Emergency Response Facility Computer System (ERFCS)	R1 07-02-96
FC-EPF-34	MRC Director Checklist	R0 06-23-93
FC-EPF-35	Iowa EOC Route Map (double-sided)	R0 06-21-94
FC-EPF-36	EOF Briefing Checklist	R2 08-04-98
FC-EPF-37	Operations Liaison Out of Service Equipment List	R0 07-11-95
FC-EPF-38	Blair Industrial Park CO-OP	R5 01-20-00
FC-EPF-41	Emergency Planning Simulator Critique	R0 09-30-98
FC-EPF-42	Emergency Action Levels	R0 07-16-99

* Computer Generated

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Blair Industrial Park Co-Op

Event Notification Form

1 Take roll call of responding members (✔ by name of those responding)	☐ Blair Water/Sewer ☐ MACC ☐ Washington County Dispatch	☐ Cargill ☐ Terra Nitrogen	☐ Kelly Ryan ☐ Agro ☐ PURAC
2 Emergency Classification Code	☐ NOUE (Blue) ☐ Alert (Green)	☐ Site Area (Yellow)	☐ General (Red)
3 Classification Date		Time	
Reported by (If other than OPPD)			
4 Person making Report		Title	
Call back #		Time	
Authorized By		Title	Time
5 Incident facts Time of Event		Estimated Duration (hrs)	
6 Event type (✔ all applicable)	☐ Explosion ☐ Radiological ☐ Other; Please list ______	☐ Fire ☐ River Release	☐ Gas Release ☐ Spill
Substance Involved (proper spelling?)		Volume	Units
	DOT ID #	DOT Guide #	
Other Identifying characteristics	☐ Flammable ☐ Toxic	☐ HazMat	☐ Other
7 Weather Direction (from°)		Wind Speed ___ MPH	Precipitation? ☐ Yes ☐ No

8 Comments (At Site Area or General emergency recommend that members tune to KFAB 1110 AM for EAS Message(s))

9 Was an attempt made to contact members that didn't answer Co-Op line ? ☐ Yes ☐ No

List those not contacted

FORM EP-1
Rev. 6 (04/29/99)

OMAHA PUBLIC POWER DISTRICT

Confirmation of Transmittal for
Emergency Planning Documents/Information

■ Radiological Emergency Response Plan (RERP) ☐ Emergency Plan Implementing Procedures (EPIP) ☐ Emergency Planning Forms (EPF)

☐ Emergency Planning Department Manual (EPDM) ☐ Other Emergency Planning Document(s)/ Information

Transmitted to:

Name: Document Control Desk Copy No: 165 Date: 2-3-00
Tom Andrews Copy No: 154
Tom Andrews Copy No: 155
Tom Andrews Copy No: 156

The following document(s) / information is forwarded for your manual:

REMOVE SECTION	INSERT SECTION
RERP Index Pages 1 issued 01/06/00 & 2 issued 09/02/99	RERP Index Pages 1 & 2 issued 01/27/00
RERP R13 issued 02/25/98	RERP R14 issued 01/27/00
RERP-Section L R10 issued 09/18/97	RERP-Section L R11 issued 01/27/00

Summary of Changes:
RERP was revised to delete reference to MIDAS, change UNMC to Nebraska Health System (NHS) University Hospital, and to take credit for NHS's role at the MRC.
RERP-Section L was converted from Word Perfect 6.1 to 8.0 and minor editorial corrections.

Supervisor - Emergency Planning

I hereby acknowledge receipt of the above documents/information and have included them in my assigned manuals.

Signature: ____________________ **Date:** ____________

Please sign above and return by **03/26/00** to:

Karma Boone
Fort Calhoun Station, FC-2-1
Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

NOTE: If the document(s)/information contained in this transmittal is no longer requested or needed by the recipient, or has been transferred to another individuals, please fill out the information below.

☐ Document(s)/Information No Longer Requested/Needed

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Name: ____________________ Mailing Address: ____________________

RADIOLOGICAL EMERGENCY RESPONSE PLAN INDEX
RERP

RADIOLOGICAL EMERGENCY RESPONSE PLAN INDEX
RERP

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Fort Calhoun Station
Unit No. 1

RERP

RADIOLOGICAL EMERGENCY RESPONSE PLAN

Title: DEFINITIONS AND ABBREVIATIONS

FC-68 Number: DCR 11279

Reason for Change: Delete reference to MIDAS, change UNMC to Nebraska Health System (NHS) University Hospital, take credit for NHS's role at the MRC.

Initiator: Mark Reller

Preparer: Mark Reller

DEFINITIONS AND ABBREVIATIONS

1. ABB-CE - ABB, Combustion Engineering, Inc.

2. AIF - Atomic Industrial Forum

3. ALARA - (As Low As Reasonably Achievable) - The level at which OPPD strives to maintain personnel radiation exposure.

4. ALNOR - A brand name of electronic dosimetry utilized at the Fort Calhoun Station.

5. ANI - American Nuclear Insurers

6. AOP - Abnormal Operating Procedures

7. ASSESSMENT ACTIONS - The appropriate actions taken during or following an accident evaluation before implementing the specific corrective and/or protective actions.

8. CE - ABB, Combustion Engineering, Inc.

9. CDE - Committed Dose Equivalent (typically to a specific organ)

10. CEDE - Committed Effective Dose Equivalent

11. CET - Core Exit Thermocouple

12. CFR - Code of Federal Regulations

13. CFS - Cubic Feet per Second

14. CHP - Conference Health Physics Network

15. COP - Conference Operations Network

16. CONTROL ROOM - The onsite location from which the Fort Calhoun Station nuclear power plant is operated.

17. CORRECTIVE ACTIONS - Emergency measures taken to correct or mitigate an emergency condition at its origin in order to prevent an uncontrolled release of radioactive material or to reduce the magnitude of a release.

18. CR - Control Room

19. <u>ΔT</u> - Delta Temperature - The difference in temperature between points 10 meters and 60 meters above the ground in units of centigrade. The value displayed on the ERFCS equates to;

$$100m.\underline{\Delta T} = [(T @ 60m - T @ 10m) \times 2)].$$

20. <u>DDE</u> - Deep Dose Equivalent (typically a dose received from a high penetrating form of radiation).

21. <u>DOE</u> - Department of Energy

22. <u>DRILLS</u> - A drill is a supervised instruction period aimed at testing, developing and maintaining skills in a particular ERO position, function, center, or operation. A drill can be used as a component of a larger exercise.

23. <u>D/Q</u> - Deposition Factor

24. <u>DSO</u> - Director of Site Operation

25. <u>EAL</u> - Emergency Action Levels - Alarms, instrument readings or visual sightings that have exceeded predetermined limits which would categorize the situation into an initiating condition of one of the four Emergency Classifications. (see EPIP-OSC-1 for specifics).

26. <u>EAGLE</u> - Emergency Assessment of Gaseous & Liquid Effluents.

27. <u>EAS</u> - (Emergency Alerting System) - A predesignated network of radio stations whose purpose is to pass emergency information throughout a designated geographical area. Radio station KFAB (1110 KHz) is the primary control station for the Fort Calhoun Station and surrounding communities.

28. <u>ECCS</u> - Emergency Core Cooling System

29. <u>ENS</u> - Emergency Notification System

30. <u>EOC</u> - Emergency Operation Center

31. <u>EOF</u> - Emergency Operation Facility

32. <u>EOP</u> - Emergency Operating Procedures

33. <u>EPA</u> - Environmental Protection Agency

34. <u>EPIP</u> - Emergency Plan Implementing Procedures

35. EPRI - Electrical Power Research Institute

36. EPT - Emergency Planning Test

37. EPZ - Emergency Planning Zone

38. ERDS - Emergency Response Data System, a system which electronically transmits key plant parameters from the station to the NRC during an emergency condition.

39. ERF - Emergency Response Facilities

40. ERFCS - Emergency Response Facilities Computer System

41. ERMS - Emergency Response Message System

42. ERO - Emergency Response Organization

43. EXCLUSION AREA - The area surrounding a nuclear power plant in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from that area. The term is synonymous with "onsite".

44. EXERCISE - An emergency preparedness exercise is an event that tests the integrated capability and major portion of the basic elements existing within the Radiological Emergency Response Plan (RERP), associated Emergency Plan Implementing Procedures (EPIPs) and the various organizations associated with the implementation of the RERP. Typically, an emergency preparedness exercise shall simulate an emergency that results in offsite radiological releases which would require response by offsite authorities.

45. FAA - Federal Aviation Administration

46. FAX - Emergency Assessment Facsimile

47. FCP - Forward or Field Command Post

48. FCS - Fort Calhoun Station

49. FEMA - Federal Emergency Management Agency

50. FOL - Forward Operation Location

51. FSAR - Final Safety Analysis Report, now a historical document.

52. FTS -Federal Telecommunications System (NRC Phone Circuits)

53. GAR - Governor's Authorized Representative

54. GCPM - Gross Counts per Minute

55. HEPA - High Efficiency Particulate Air filter

56. HHS - Health and Human Services

57. HPN - Health Physics Network

58. HVAC - Heating, Ventilation, and Air Conditioning

59. INPO - Institute of Nuclear Power Operations

60. IPZ - Ingestion Pathway Zone

61. KFAB - An A.M. radio station that serves as the Local Primary (LP1) Emergency Alerting System station for emergencies at the Fort Calhoun Station.

62. MUD - Metropolitan Utilities District

63. MRC - Media Release Center

64. NAWAS - NAtional WArning System.

65. NHS - Nebraska Health System University Hospital, located in Omaha, it provides medical aid in Radiological Emergencies and serves as an independent subject matter expert at the MRC, through its Radiation Health Center.

66. NSP - Nebraska State Patrol

67. NRC - Nuclear Regulatory Commission

68. NSSS - Nuclear Steam Supply System

69. OPPD - Omaha Public Power District - Also referred to as the "licensee".

70. OSC - Operations Support Center

71. PAG - (Protective Action Guide) - The Protective Action Guide is a set of radiological dose values for the general population that warrant protective action following an uncontrolled release of radioactive material. The PAG's are established in the "Manual of protective Action Guides and Protective Actions for Nuclear Incidents, EPA 400-R-92-001. May 1992".

72. PAR - (Protective Action Recommendations) - Recommendations regarding protective actions to be taken to safeguard the public.

73. PASS - Post Accident Sampling System

74. PING - Particulate, Iodine, and Noble Gas monitor

75. Protective Actions - Emergency measures taken following a release of radioactive material in order to prevent or minimize the radiological exposures to individuals.

76. PWR - Pressurized Water Reactor

77. QA - Quality Assurance

78. QC - Quality Control

79. QSPDS - Qualified Safety Parameter Display System.

80. RCS - Reactor Coolant System

81. REM - Roentgen Equivalent Man: unit of dose of any ionizing radiation that produces the same biological effect as a unit of absorbed dose of ordinary X-rays.

82. RERP - Radiological Emergency Response Plan

83. RERT - Radiological Emergency Response Team

84. RMS - (Radiation Monitoring System) - Area and process radiological instrumentation within the station.

85. SARC - Safety Audit and Review Committee

86. SB - Site Boundary

87. SCBA - Self Contained Breathing Apparatus

88. SCFM - Standard Cubic Feet per Minute

89. SDE - Shallow Dose Equivalent (typically dose received to the skin)

90. Site Director - The person responsible for all onsite activities and offsite actions until the Emergency Director position is established.

91. Site North - General term used to define the direction within the plant of the walls near the north side of the plant. True north is slightly northeast of Site North.

92. SPDS - Safety Parameter Display System

93. SRD - Self Reading Dosimeter

94. TEDE - Total Effective Dose Equivalent (a measurement of total body burden from receiving a dose externally and internally - replaces "Whole Body" as the key dose limit)

95. TLD - (Thermoluminescent Dosimeter) - A device normally used by plant personnel to measure the amount of radiation received.

96. TSC - Technical Support Center

97. UPS - Uninterruptable Power Supply

98. USAR - (Updated Safety Analysis Report) - An active document evaluating plant systems.

99. WB - Whole Body

AREA DESCRIPTION

1. PLANT LOCATION

Fort Calhoun Station is located midway between Fort Calhoun and Blair, Nebraska, on the west bank of the Missouri River. The site consists of approximately 660.46 acres with an additional exclusion area of 582.18 acres on the northeast bank of the river directly opposite the plant buildings. The distance from the reactor containment to the nearest site boundary is approximately 910 meters; and the distance to the nearest residence is beyond the site boundary. Except for the city of Blair and the villages of Fort Calhoun and Kennard, the area within a ten mile radius is predominantly rural. The land use within the ten mile radius is primarily devoted to general farming. There are no private businesses or public recreational facilities on the plant property. The DeSoto National Wildlife Refuge occupies approximately 7821 acres east of the plant site. This area is open to the public for day use year around. Visitors to the refuge generally use areas from two to five miles from the plant. Estimates by the U.S. Fish and Wildlife Service place annual usage of the facility at approximately 120,000 for the Visitors Center and 400,000 for the refuge. The expected maximum daily usage of the facility has been placed at 2500 visitors for a winter weekday and 5000 on a summer weekend. The Boyer Chute Federal Recreation Area is a day use facility occupying approximately 2000 acres southeast of the plant site. Visitors to the recreation area generally use areas seven to ten miles from the plant. The estimates for annual usage of this facility is approximately 50,000 visitors.

The State of Nebraska operates the Fort Atkinson State Historic Park five and half miles southeast of the plant site. This day use facility is mostly seasonal and estimates place annual usage at 60,000. The State of Iowa maintains Wilson Island State Park with 275 camping spaces south of the DeSoto National Wildlife Refuge and four miles southeast of the plant site. The estimates for usage of this facility range from 500 on a winter weekday to 1000 on a summer weekend.

Two private facilities lie to the north of the plant along the Missouri River. The Cottonwood Marina is located approximately four and a half miles from the plant. Estimates place summer weekend usage at 200 people. Timbers at Rivers Edge is a private campground lying directly south of Cottonwood Marina and ranging from four to four and a half miles from the plant. The campground has approximately 235 campsites and is open from April to October.

2. AREA INDUSTRIES

A listing of various industries located within a ten mile radius of the Fort Calhoun Station, including firm name, product, number of employees, and location from the plant site is contained in the Updated Safety Analysis Report.

3. AREA WATER SUPPLIES

Local public drinking water supplies are not taken from the Missouri River in this area. The first downstream intake is the city of Omaha approximately 19.5 miles downstream. Industrial water use is limited to cooling purposes in the Omaha area. Drinking water near the Fort Calhoun Station is obtained from either well or reservoirs. Since the known public and private water supplies originate at elevations higher than the river, radioactive liquids that might be discharged from the plant into the river should not contaminate these supplies.

There are also many private wells in the region which draw primarily upon ground water rather than on springs or other surface sources. Several marinas are located along the Missouri River, between 3 miles upstream from Blair and Omaha, 18 miles downstream. In the event of a significant waterborne release incident from the Fort Calhoun Station, the Nebraska Department of Environmental Control acting in conjunction with the Nebraska Department of Health, Division of Radiological Health and the U. S. Coast Guard are prepared to notify all downstream users of Missouri River water. Notification is made through OPPD management directly to the Metropolitan Utilities District (MUD) in the event of an inadvertent liquid release to the river. Swimming, boating and other recreational activities involving river water can be controlled by the Coast Guard until adequate surveys have been taken to determine when normal activities may be resumed.

PURPOSE OF THE EMERGENCY PLAN

The purpose of the Fort Calhoun Station "Radiological Emergency Response Plan" (RERP) is to delineate an organization for coping with emergencies, to classify emergencies according to severity, define and assign responsibilities and authorities, and to clearly outline the most effective course of action and protective measures required to mitigate the consequences of an accident and to safeguard the public and station personnel in the event of an incident. The Emergency Plan Implementing Procedures (EPIP's), Radiation Protection procedures, Emergency Operating procedures and other station references are available at the plant to further assist personnel for operating during abnormal occurrences. The various emergency procedures are put into effect whenever a system, component or circuit failure could lead to a personnel hazard or major equipment failure. Emergency Operating Procedures are sufficiently detailed so that the plant is placed, as expeditiously as possible, in a safe condition. The various procedures include such items as radiation hazards, weather conditions and availability of technical and operating personnel.

ACCIDENT CONSIDERATIONS

1. FUEL HANDLING ACCIDENT

The possibility of an incident during fuel handling is unlikely due to the many physical limitations imposed on fuel handling operations and systems. In addition, administrative restrictions placed on fuel handling procedures provide greater control. Nevertheless, the offsite consequences of dropping a spent fuel assembly and breaking 14 fuel rods have been evaluated and are documented in the Fort Calhoun Station, Unit No. 1 USAR. Emergency onsite and offsite monitoring practices would begin immediately following the accident to determine actual consequences, and appropriate emergency actions would be taken. Emergency procedures addressing a Fuel Handling Incident provide emergency actions for this mishap.

2. FIRES

2.1 Internal Plant Fires (within the Protected Area)

Internal Plant fires are normally handled by the station's Fire Brigade, comprised of trained individuals from the Operations, Chemistry, Radiation Protection and Security Departments. All efforts are made to prevent the spread of airborne contamination should the fires occur within the Radiological Controlled Area.

2.2 External Fires (outside the Protected Area)

External fires are controlled by local fire department response. In the event high airborne contamination constitutes a possible hazard to areas outside of the protected area, offsite survey teams/personnel can be dispatched immediately.

3. EXPLOSION

Because of the accumulation of waste gases in the waste gas decay tanks, the possibility and consequences of an explosion have been considered. An explosion could result in an unexpected, uncontrolled release to the atmosphere of radioactive fission gases that were stored in the waste gas system. A failure of any of the waste gas decay tanks or associated piping could also result in a release of gaseous activity. The noble gases stored in the tanks would diffuse and become diluted during their transport to the site boundary. The projected Deep Dose Equivalent (DDE) at the exclusion area boundary would be less than 1.0 Rem. This conservative analysis is based upon 1% fuel cladding defects, and accumulation of all noble gases without release over a full core cycle. Emergency procedures addressing a Waste Gas Incident, would be placed into effect immediately and offsite monitoring teams would be dispatched downwind.

4. TOXIC CHEMICAL RELEASE ACCIDENTS

The primary toxic chemical release accidents which may result in toxic gas concentrations at Fort Calhoun Station are shown below:

TOXIC CHEMICAL	ACCIDENT
Ammonia (NH_3)	Rupture of two 25,000 ton offsite refrigerated tanks.
Ammonia	Rupture of two 30,000 gal. offsite non-refrigerated tanks.
Ammonia	Rupture of a 78 ton railroad tank car.
Ammonia	Rupture of a 2 ton tank truck.

The above accidents will not pose a hazard to control room personnel, due to toxic gas monitors located at the fresh air intake of the control room, which isolates the control room before the gases reach the toxic limit. The stringent odor of ammonia makes station personnel immediately aware of any leakage or toxic gas cloud.

4. (Continued)

The toxic gas monitors sample for NH_3 and continuously monitor the fresh air to the control room during normal plant operations.

At different phases of plant operation, Hydrogen and/or Nitrogen gases blanket the volume control tank and the waste gas system. Considering that the deleterious effect of these gases is the exclusion of oxygen, a release to the atmosphere diminishes the harmful effect and a serious hazard is eliminated.

In the event of an offsite accidental release of chemicals, within a five (5) mile radius of the Fort Calhoun Station, the Blair Fire Department emergency procedures requires notification to the Fort Calhoun Station. The counties of Washington (Nebraska) and Harrison (Iowa) have agreed to notify the Fort Calhoun Station when hazardous chemical accidents occur within five miles of the station. The Blair Industrial Park Co-Op, emergency notification system is an organization of industries, including Fort Calhoun Station that have banded together to form a warning system to notify the member industries and the Washington County Dispatch center of a potential or actual event occurring at a member facility. Appropriate action is taken, especially in the control room, to ensure that air remains breathable. For long duration toxic accidents, six (6) hours of compressed air is available for five (5) control room operators coupled with provisions to obtain additional air within this time period.

5. <u>MAJOR STEAM RELEASE</u>

The offsite consequences of a steam line rupture incident has been evaluated and is documented in the Fort Calhoun Station, Unit NO. 1 USAR. The maximum size steam line rupture is a circumferential double-ended rupture of the 36-inch main steam header. The analysis of this incident at the site boundary is calculated to be 0.9 Rem total whole body exposure. Plant personnel would be protected by normal health physics practices and procedures. Operator action follows the emergency procedures addressing a Steam Line Rupture with Loss of Offsite Power.

6. <u>PERSONNEL INJURY</u>

A fully stocked First Aid Room is available in the Plant. Immediate and temporary care may be given to the injured person using standard First Aid practices. If the injury involves contamination, efforts to decontaminate the injured person to reasonable levels are made prior to transfer to the First Aid Room or to offsite medical facilities. If decontamination is not practical, the injured person is covered in such a manner as to minimize the spread of contamination until either medical aid can be obtained or until the injured person can be transported to the NHS University Hospital Radiation Health Center.

7. NATURAL DISASTERS

A natural disaster may occur which could initiate any of the accidents previously discussed. The reactor may be placed in a shutdown condition, depending upon the anticipated or experienced severity of the disaster.

Fort Calhoun Station
Unit No. 1

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RERP-SECTION L

RADIOLOGICAL EMERGENCY RESPONSE PLAN

Title: MEDICAL AND PUBLIC HEALTH SUPPORT

FC-68 Number: N/A

Reason for Change: Converted from WP6.1 to WP8 with Documentable Error incorporated.

Initiator: Mark Reller

Preparer: Sheila Rasmussen

MEDICAL AND PUBLIC HEALTH SUPPORT

1. Onsite First Aid

There are generally four types of response considered at the Fort Calhoun Station:

1) Minor injury, no contamination
2) Minor injury, contaminated
3) Major injury (requiring offsite treatment), no contamination
4) Major injury, contaminated

The order of medical treatment will be:

1) Care of severe physical injuries
2) Decontamination of personnel
3) First aid to other injuries
4) Monitor for internal contamination
5) Definitive treatment and subsequent therapy as required

All injuries at the station must be immediately reported to the Shift Manager, who will initiate response according to the Fort Calhoun Station Safety Manual.

When personnel are severely injured and contaminated, first aid shall take precedence over decontamination. In cases where internal exposure is suspected, a bioassay program may be performed as directed by the Radiation Protection Manual.

1.1 First Aid Facilities

A First Aid Room is located in the TSC. This room is equipped with an examination table, examination chair, illuminating magnifier and a supply of bandages, splints, etc., to provide emergency first aid to injured personnel.

Other equipment located throughout the plant include first aid kits, Emergency Medical Technician (EMT) kits, personnel carriers, a wheelchair, and contaminated/injured personnel response kit. The Industrial Safety Coordinator inspects and maintains this equipment.

1.2 Medical Response

1.2.1 Minor Injury, No Contamination

The Shift Manager or other evaluators will determine the extent of medical response required. This could include:

A. On the spot treatment by the individual or first aid qualified responders.

1.2.1 B. On the spot treatment by EMT qualified personnel.

C. Movement of the injured party to the first aid room by medical responders for access to additional equipment.

D. Other response determined necessary by responding personnel.

1.2.2 Minor Injury, Contaminated

Personnel that are injured and are potentially contaminated will be treated as explained above, and will also be monitored for contamination by Radiation Protection personnel. Monitoring and decontamination will be performed in accordance with Radiation Protection procedures.

1.2.3 Major Injury, No Contamination

Medical responders will be dispatched to the scene to perform first aid as required and prepare the victim for transport. The Shift Manager or designee will notify offsite authorities to provide victim transport to an available medical facility. Both air and ground transportation are available.

1.2.4 Major Injury, Contaminated

Personnel that are severely injured and are potentially contaminated will be treated as explained above, and will also be monitored for contamination by Radiation Protection personnel. If feasible, monitoring and decontamination will be performed in accordance with Radiation Protection Procedures. If decontamination is successful, the victim may be transported to any available medical facility for treatment.

If decontamination is not successful or not feasible, the victim will be transported to the Nebraska Health System (NHS) University Hospital Regional Radiation Health Center, unless the on-scene medical responders or transportation personnel deem it medically necessary to proceed to a closer facility. If another facility other than NHS is used, additional Radiation Protection personnel should be sent to the facility to assist in monitoring, decontamination and clean up.

2. Medical Transportation

2.1 Blair Fire Department and Rescue Squad

The Blair Fire Department and Rescue Squad Station is located less than four (4) miles from the Fort Calhoun Station. The Rescue squad furnishes transportation for the injured and administers first aid enroute to the hospital.

2.2 Fort Calhoun Fire and Rescue Squad

The Fort Calhoun Fire and Rescue Squad headquarters is located approximately 3-1/2 miles from the Fort Calhoun Station. This rescue squad serves as backup to the Blair Fire Department and Rescue Squad.

2.3 Missouri Valley Fire and Rescue Squad

The Missouri Valley Fire and Rescue Squad is located approximately fifteen (15) miles from the plant.

2.4 Additional support is available to both the Blair Fire Department and Rescue Squad and the Fort Calhoun Fire and Rescue Squad by request through the Tri-Mutual Aid Association (Douglas, Sarpy, and Washington Counties).

2.5 Other Modes of Transportation

If necessary, there are other modes of transportation for delivering injured personnel to appropriate medical facilities.

2.5.1 Medical Ambulance helicopter.

2.5.2 Onsite company vehicles.

2.5.3 Private autos of company personnel.

3.0 Offsite Medical Support

3.1 Non-Contaminated Personnel

The nearest medical facility is the Blair Memorial Community Hospital which is located five (5) miles from the plant. A physician is readily available as a general medical consultant. Other facilities may be used as determined necessary by medical response personnel.

3.2 Contaminated Personnel

Omaha Public Power District maintains an agreement with the NHS, University Hospital Regional Radiation Health Center to supply 24-hour treatment for all injuries involving contamination and/or personnel radiation exposure. The Regional Radiation Health Center is located approximately 25 miles from the plant in Omaha, Nebraska. The facility is part of the NHS complex, and was established specifically for the treatment of injuries occurring from nuclear and radiation related incidents. A entrance is available for the ingress and egress of contaminated victims to a special assessment and decontamination facility. This entrance is from Forty Fourth Street and is depicted in Figure L-1. Patients can also be transported to the facility via medical ambulance helicopters.

The NHS University Hospital Regional Radiation Health Center staff administers medical, decontamination, internal bioassay, and other nuclear medicine capabilities. The staff maintains an appropriate "Standard Operating Procedures Manual" which describes their responsibilities and roles. If additional hospital beds should be required during a major incident, the hospital maintains a mutual agreement with several other Omaha area hospitals to assist with decontaminated patients.

Due to the large, highly qualified staff, the distance from the plant, the specialized capabilities, and the overall size of the NHS complex, the Fort Calhoun Station was granted an exemption from requiring a backup medical facility by the Federal Emergency Management Agency.

Figure L-1 Entrance to Treatment Area
NHS, University Hospital Regional Radiation Health Center