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PRC|EPIP-TOC-COMMON|000	19	A	1	H	127610
PRC|HC.EP-EP.ZZ-0205|000	3	A	1	H	127785
PRC|SC.EP-EP.ZZ-0205|000	2	A	1	H	127827
PRC|NC.EP-EP.ZZ-0202|000	3	A	1	H	127699
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PSEG NUCLEAR
ONSITE IMPLEMENTING PROCEDURES
February 6, 2002

CHANGE PAGES FOR
REVISION #19

The Table of Contents forms a general guide to the current revision of each section of the Onsite EPEPs. The changes that are made in this TOC Revision #19 are shown below. Please check that your revision packet is complete and remove the outdated material listed below:

ADD			REMOVE		
Page	Description	Rev.	Page	Description	Rev.
ALL	TOC	19	ALL	TOC	18
All	NC.EP-EP.ZZ-0201	04	All	NC.EP-EP.ZZ-0201	03
All	NC.EP-EP.ZZ-0202	03	All	NC.EP-EP.ZZ-0202	02
All	NC.EP-EP.ZZ-0203	02	All	NC.EP-EP.ZZ-0203	01
All	HC.EP-EP.ZZ-0205	03	All	HC.EP-EP.ZZ-0205	02
All	SC.EP-EP.ZZ-0205	02	All	SC.EP-EP.ZZ-0205	01

Note: **H**C.EP-EP.ZZ-0205 goes behind the BLUE 205 tab

SC.EP-EP.ZZ-0205 goes behind the GREEN 205 tab

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PSEG NUCLEAR LLC
EMERGENCY PLAN ONSITE IMPLEMENTING PROCEDURES
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STATION PROCEDURES

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NC.EP-EP.ZZ-0201 (Q) Rev. 04

TSC – INTEGRATED ENGINEERING RESPONSE

USE CATEGORY: II

REVISION SUMMARY: Biennial Review Yes _X_ No __

Reference to Nuclear Fuels Engineer has been deleted.

IMPLEMENTATION REQUIREMENTS

2/6/02

APPROVED: ______________________ 1-29-02
EP Manager Date

APPROVED: ______________________ 1-29-02
Vice President - Operations Date

TSC – INTEGRATED ENGINEERING RESPONSE

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides guidance to emergency response personnel for administration of TSC Integrated Engineering Response during an emergency at Hope Creek or Salem Nuclear Generating Stations.

2.0 PREREQUISITES

2.1. Prerequisites To Be Followed Prior To Implementing This Procedure

2.1.1 Implement this procedure at:

- The discretion of Technical Support Supervisor (TSS), Technical Support Team Leader (TSS), or Technical Support Team Member (TSTM).
- Upon staffing of your Emergency Response Facility.

3.0 PRECAUTIONS AND LIMITATIONS

3.1. Precautions and Limitations To Be Followed Prior To Implementing This Procedure:

3.1.1 It is recommended that initials be used in the place keeping sign-offs, instead of checkmarks.

3.1.2 Personnel who implement this procedure shall be trained and qualified IAW the Emergency Plan.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 PROCEDURE

> **NOTE:**
> The TSS or TSS may require Engineering support prior to TSS activation. Precautionary MANNING of the TSS with key engineering support personnel does NOT require ACTIVATION of the TSS.

Initials

5.1. The Technical Support Supervisor (TSS) Should Perform The Following:

5.1.1 DIRECT the TSTL to Implement Attachment 9, TSTL Checklist. ______

5.1.2 IMPLEMENT Attachment 1, TSS Checklist. ______

6.0 RECORDS

Forward all completed procedures, forms and attachments to EP Manager.

7.0 REFERENCES

7.1. References

7.1.1 PSEG Nuclear Emergency Plan

7.1.2 EP 96-02 post-OBE spent fuel pool rack gap evaluation

7.1.3 EP 99-01 Monitor the spent fuel pool temperature

7.1.4 EP 96-01 Monitoring Salem diesel generator fuel oil

7.2. Cross-References

None

ATTACHMENT 1
Page 1 of 3

TECHNICAL SUPPORT SUPERVISOR CHECKLIST

The Technical Support Supervisor (TSS) shall:

NOTE
Should the EDO/ERM be unable to fulfill the duties of Emergency Coordinator (EC) for any reason (e.g., sudden illness, accident, etc.) the Technical Support Supervisor (TSS) or Site Support Manager (SSM) may assume the duties and responsibilities of EC until another qualified EDO/ERM arrives at the facility. The TSS or SSM position must be filled by another individual.

NOTE
SALEM – Refer to page 3 of this attachment for additional actions required at Salem Station

	Initials
1. ASSUME Command and Control of the TSC facility until relieved by the EDO.	________ TSS
2. ASSUME Supervision of the Technical Support Team.	________ TSS
3. INITIATE and maintain a chronological log of your activities.	________ TSS
4. ESTABLISH communications with the Control Room(Ops Advisor) and obtain as a minimum the following information on a routine basis: A. Overall plant status to include: 1. Fission Product Boundary Integrity 2. Major equipment availability 3. Electrical system line up and power sources 4. Radiation Monitoring system parameters (SPDS, RM-11) 5. Implementation of Emergency, Abnormal, or Integrated Operating Procedures	________ TSS

ATTACHMENT 1
Page 2 of 3

TECHNICAL SUPPORT SUPERVISOR CHECKLIST

	Initials
5. NOTIFY the EDO when staffing is adequate to perform the technical support functions. Request the Admin. Support Supervisor to callout additional personnel as required.	________ TSS
6. PROVIDE assistance to the Emergency Duty Officer (EDO) in plant status and direction, Event Classification, Protective Action Recommendations (PARs) and any other activities as required.	________ TSS
7. COMPLETE the Task Assignments using Attachment 4, TSC Technical Support Tracking Form, and provide to the TSTL for implementation.	________ TSS
8. REVIEW all completed Attachment 4, TSC Technical Support Tracking Forms. Forward to the EDO for review.	________ TSS
9. REVIEW as appropriate, procedures being implemented by the Control Room and periodically brief the EDO and the TSTL on: • Support that may be required by the OS. • Any projected problems or areas of concern. • Overall direction in which the Control Room is proceeding. • Status of the Control Room implementation of EOP, AOP, and IOPs. • Status of engineering recommendations provided to the OS.	________ TSS
10. IF a radiological release is imminent or in progress, THEN place the TSC Emergency Ventilation System in service in the pressurized mode:	________ TSS

HOPE CREEK	SALEM
Place the TSC Mode Control Switch (HS-9764) to PRESS (local Panel 10N211 at Central Alarm Station).	Direct a TSTM to place the TSC Ventilation System in Emergency Operation, IAW SC.OP-SO.TSC-0051 Section 5.3, Operation During High Radiation Conditions.

ATTACHMENT 1
Page 3 of 3
TECHNICAL SUPPORT SUPERVISOR CHECKLIST

Initials

11. IF smoke or toxic gases are detected in the TSC air supply, THEN place the TSC Emergency Ventilation System in service in the recirculation mode: ________ TSS

HOPE CREEK	**SALEM**
Place the TSC Mode Control Switch (HS-9764) to RECIRC (local Panel 10N211 at Central Alarm Station).	Direct a TSTM to place the TSC Ventilation System in Emergency Operations IAW SC.OP-SO.TSC-0051 Section 5.4, Operation During Chemical Release.

12. WHEN the TSC Emergency Filtration System is no longer required, THEN request operations to return the TSC ventilation to normal alignment IAW: ________ TSS

HOPE CREEK	**SALEM**
HC.OP-SO.GR-0001(Z)	SC.OP-SO.TSC-0051

13. ESTABLISH communications with the Site Support Manager (SSM) at the EOF and provide periodic updates on the plant status. ________ TSS
14. IF vendor assistance is required, THEN direct the TSTL to contact the Technical Support Manager at the EOF and request they provide the necessary support. ________ TSS
15. Implement Severe Accident Management Guidelines (SAMG/SAG) and Supplemental Severe Accident Management Guidelines (SSAMG) as required based on degrading plant conditions. ________ TSS
16. COMPLETE Attachment 2, Turnover Log - TSS Checklist when being relieved. ________ TSS
17. AT the conclusion of the event, THEN insure that the TSC area is returned to ready status and all paperwork related to the event is collected and forwarded to the Emergency Preparedness Manager. ________ TSS

SALEM ONLY

1. IF R44A or R44B Dose Rate is $\geq$ 1.0E+04 R/Hr, THEN Implement Adverse Containment Monitoring in accordance with Attachment 6 of this procedure. ________ TSS
2. WHEN the plant is placed in the "Recirculation Mode"(i.e., RHR suction is aligned to the containment sump) THEN direct the TSTL to implement Attachment 5 of this procedure, Post Accident Low Pressure Injection Monitoring. ________ TSS

ATTACHMENT 2
Page 1 of 1

TURNOVER LOG - TECHNICAL SUPPORT SUPERVISOR (TSS)

Date: ______/____

1. [UE] [A] [SAE] [GE] was declared @ ____________ hrs. on _/_/_

   Due to: ____
   ____

2. The present classification, [A] [SAE] [GE] was declared @ was declared ______________ hrs. on ______/_____/______

3. The Emergency Coordinator (EC) is ________________________________
   in the [EOF] [TSC] name

4. The Oncoming and Offgoing TSS should:

	Initials On / Off
A. DISCUSS current conditions. Include any problems encountered or anticipated, and any ongoing, or expected actions.	_____/_____
B. REVIEW all applicable documentation including procedures, logs, etc., ensuring they are completed, correct and signed.	_____/_____
C. DISCUSS the TSC's priorities, personnel requirements and any support or material needs.	_____/_____
D. DISCUSS any Radiological, Safety, or Environmental concerns.	_____/_____
E. INSURE that technical support assignments are completed or reassigned as necessary.	_____/_____
F. NOTIFY the TSTL and the EDO of the change in command of the TSS. Update plant status and priorities as applicable.	_____/_____
G. As soon as possible, the oncoming TSS shall hold a briefing with the oncoming TSTL and TSTM's to insure a smooth transition between the oncoming and off going TSC personnel.	_____/_____

____________________	__________	____________________
Oncoming TSS signature	Time	Offgoing TSS signature

ATTACHMENT 3
Page 1 of 1

TSC ENGINNERING TASK ASSIGNMENT LOG

DATE: ___________ PAGE ______OF ________

TASK #	TASK ASSIGNMENT TITLE (BRIEF DESCRIPTION)	ASSIGNED TO (NAME)	COMMENTS / REMARKS/ PROBLEMS	CHECK APPROPRATE STATUS
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE
				▯ HOLD ▯ FORWARD ▯ IN PROGRESS ▯ COMPLETE

NOTE: TSTL – maintain this rooster as a formal document of engineering Assignments and keeping others informed of activities in progress.

ATTACHMENT 4
Page 1 of 1
TSC TECHNICAL SUPPORT TRACKING FORM

TASK ASSIGNMENT: (DESCRIPTION, BRIEF)		TASK #
REQUESTED BY: (CIRCLE ONE) TSTL/SSM	ASSIGNED TO:	TIME/DATE:
ENGINEERING RESPONSE/RECOMMENDATION:		

REVIEW: TIME: INITIALS:	TSTL*	TSS	EDO	OS**	DISPOSITION ☐ Implement ☐ Hold ☐ Reject

* RETAIN A COPY OF THIS DOCUMENT FOR FUTURE REFERENCE
**ALL CORRECTIVE ACTIONS IMPLEMENTED IN THE PLANT MUST BE APPROVED BY THE OS.

POST ACCIDENT LOW PRESSURE INJECTION MONITORING (SALEM ONLY)

1. Action Level

RHR lined up for suction from the Containment Sump in a post accident situation.

2. Action Statements

THE TECHNICAL SUPPORT TEAM LEADER SHALL:

		Initials
2.1	Request the OS to have the ECCS Pump Performance Analysis Log (Attachment 5, page 2), completed hourly and provided (FAXED) to the TSTL for review.	____ TSTL
2.2	Evaluate data (see pump curves in Attachment 5) and inform TSS of any unsatisfactory pump performance and any recommended corrective actions.	____ TSTL

SALEM UNIT # ___________

ECCS PUMP PERFORMANCE ANALYSIS LOG

NOTE: CIRCLE ALL UNSATISFACTORY INDICATIONS AND NOTIFY THE TSS/EDO.

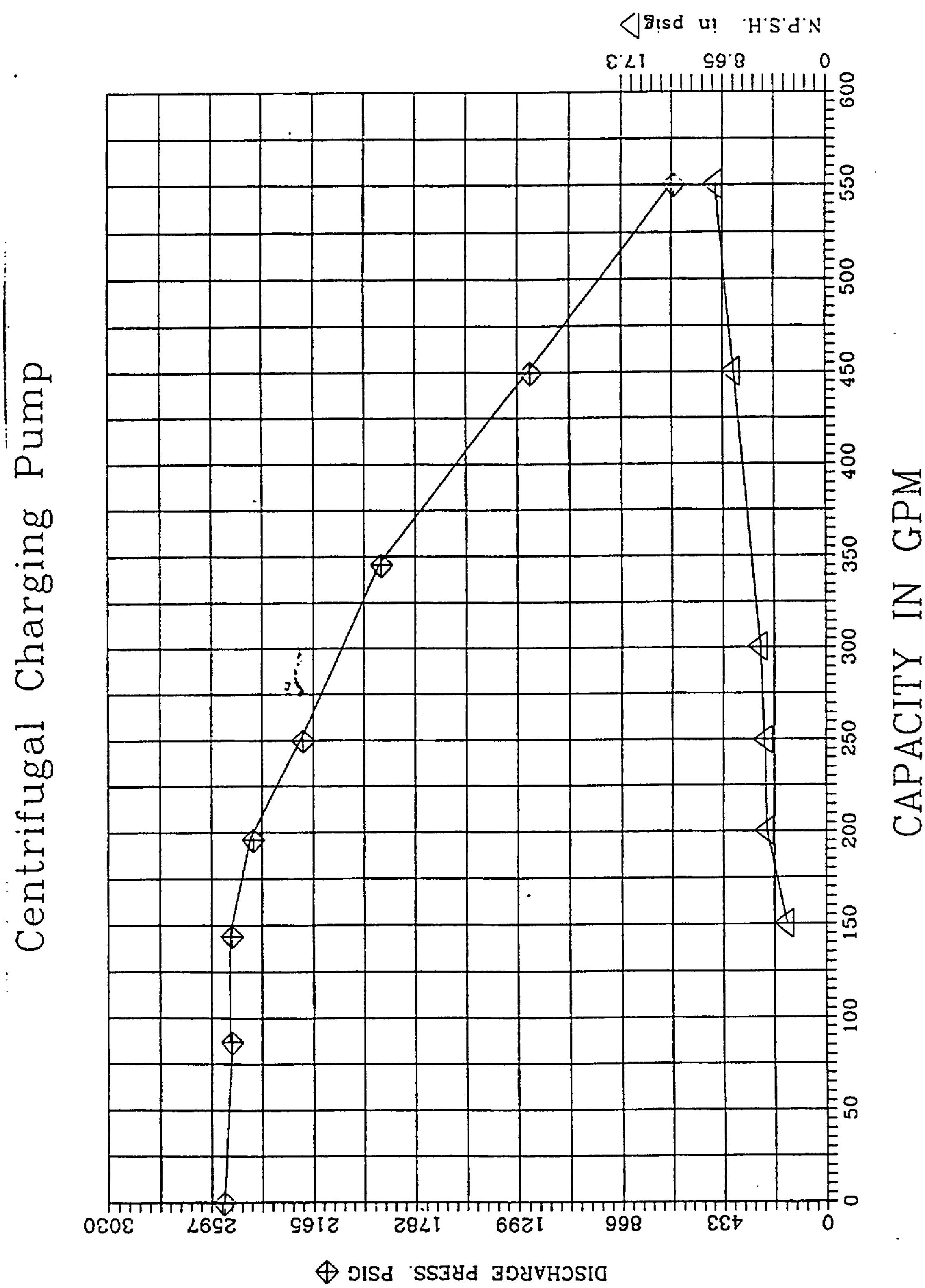
DATE: _________

INSTRUMENT NO.	INDICATIONS (HOURLY)			TIME	TIME	TIME	TIME	TIME	TIME	TIME
	PARAMETER	MIN	MAX							
LA 2445	CONT. SUMP LEVEL (%)	41	86							
PA 5511	CONT. PRESSURE (PSIG)	-3	+47							
PI 942	BIT PRESSURE (PSIG)	440 (1 PUMP)	2500 (2 PUMPS)							
IA 5310	#1 CH. PUMP AMPS	49	84							
IA 5311	#2 CH. PUMP AMPS	49	84							
FI 917	CH. PUMP FLOW (TOTAL) (GPM)	0 (1 PUMP)	780 – 830 (2 PUMPS)							
FI 128b	SEAL INJECTION FLOW (GPM)	0	78							
PI 923	#1 SI PUMP DISH PRESS	770	1500							
PI 919	#2 SI PUMP DISH PRESS	770	1500							
IA 5432	#1 SI PUMP AMPS	28	56							
IA 5433	#2 SI PUMP AMPS	28	56							
FI 922	#1 SI PUMP FLOW (GPM) (COLD LEG)	0 (CL)	650 (CL)							
FI 918	#2 SI PUMP FLOW (GPM) (COLD LEG)	0 (CL)	650 (CL)							
PI 635	#1 RHR PUMP DISH PRESS	130	190							
PI 647	#2 RHR PUMP DISH PRESS	130	190							
IA 5001	#1 RHR PUMP AMPS	28	55							
IA 5002	#2 RHR PUMP AMPS	28	55							
FI 946	#1 RHR PUMP FLOW (GPM)	0	5000							
FI 947	#2 RHR PUMP FLOW (GPM)	0	5000							

NOTES:
1. REFERE TO ATTACHED PUMP CURVES FOR CAPACITY VS TOTAL HEAD.
2. VALIDATE ABNORMAL INDICATIONS BY CORRELATING ALL AVAILABLE DATA.
3. X-OUT – INDICATIONS FOR NON–RUNNING PUMPS.
4. <u>CONTROL ROOM STAFF SHOULD TRANSMIT THIS DATA HOURLY TO THE TSS IN THE TSC.</u>

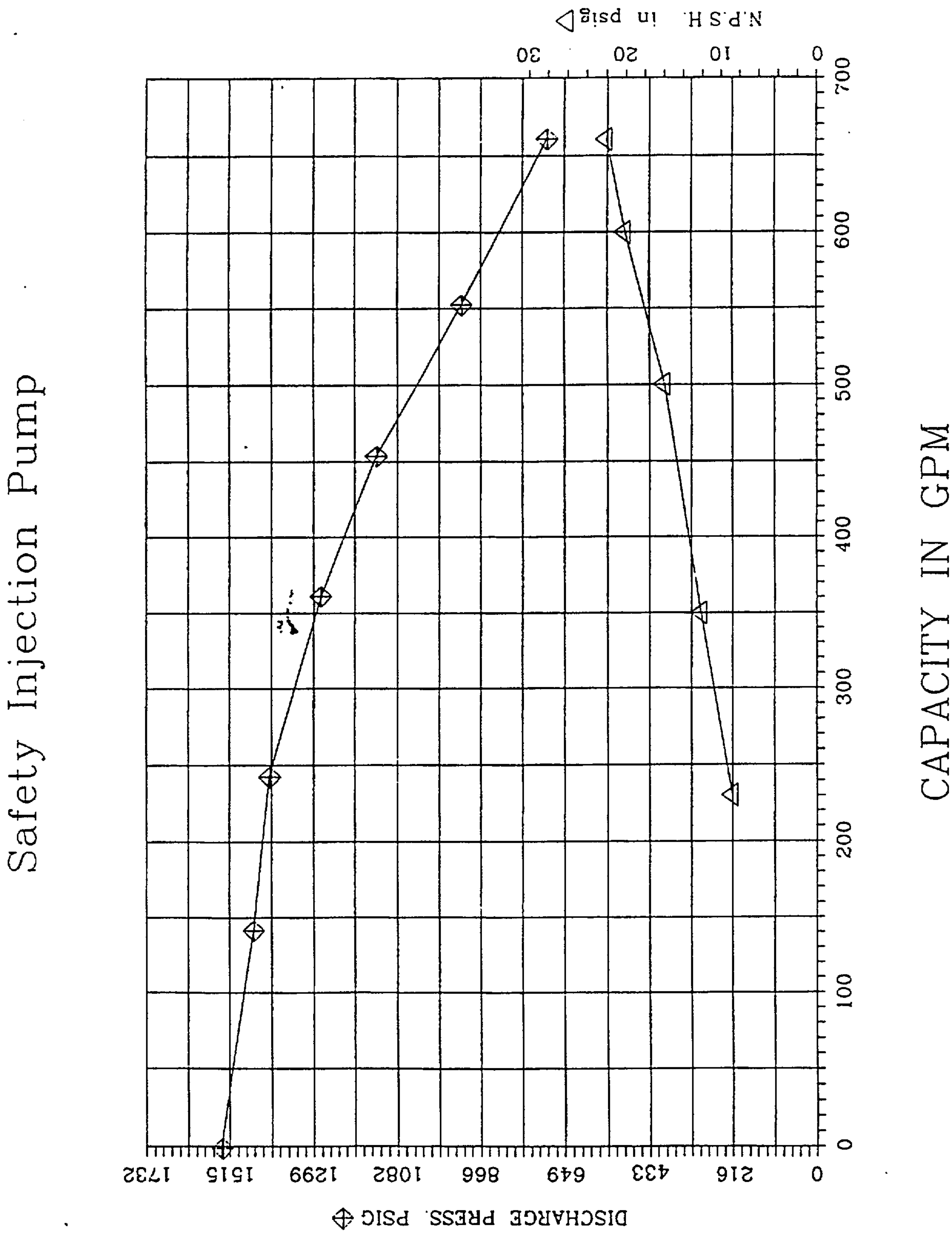
ATTACHMENT 5 (cont.)
Page 3 of 5

POST ACCIDENT LOW PRESSURE INJECTION MONITORING (SALEM ONLY)

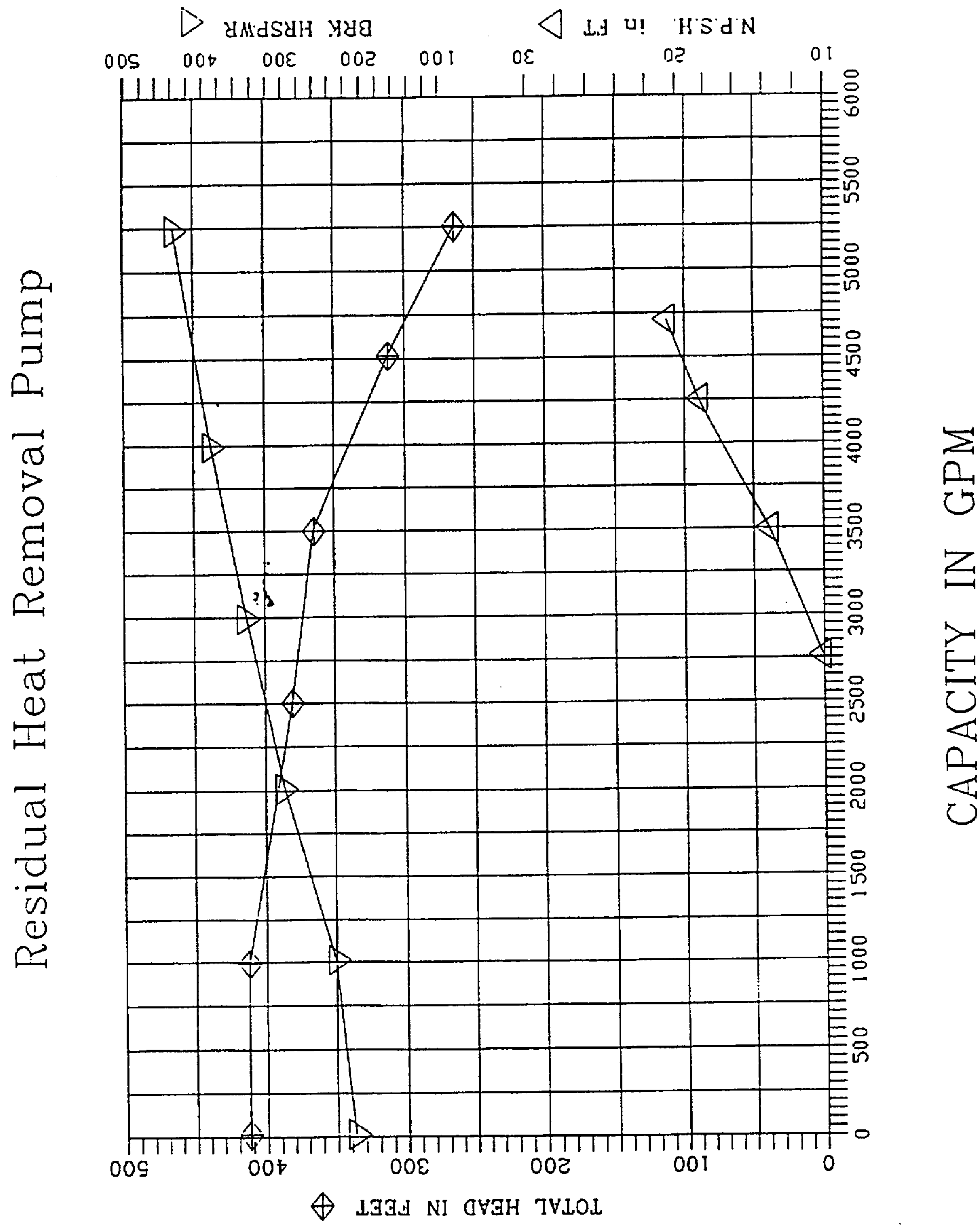


ATTACHMENT 5 (cont.)
Page 4 of 5

POST ACCIDENT LOW PRESSURE INJECTION MONITORING (SALEM ONLY)



POST ACCIDENT LOW PRESSURE INJECTION MONITORING (SALEM ONLY)



ADVERSE CONTAINMENT MONITORING (SALEM ONLY)

TECHNICAL SUPPORT SUPERVISOR (TSS) SHALL:

1. Monitor (at least once every 30 minutes) the Integrated Dose and Dose Rate in the containment as indicated on Radiation Monitor Locations Display on SPDS and perform the following:

Initial/Time

a. IF the Dose Rate, as indicated by R44A or R44B, is greater than or equal to 1E5 R/HR (>100,000 R/HR) AND the Integrated Dose as indicated by R44A or R44B is less than 1E6 R, (<1,000,000 R),

THEN
Contact the STA with the above information and inform that Adverse Containment Condition due to high radiation DOES NOT exist and he/she should refer to EOP-CFST-1 for actions required. ____/____ TSS Time

b. IF the Integrated Dose as indicated by R44A or R44B is greater than or equal to 1E6 R (>1,000,000 R),

THEN
Contact STA and notify of the Integrated Dose value and that Adverse Containment Conditions due to high radiation DO exist and he/she should refer to EOP-CFST-1 for action required. ____/____ TSS Time

ATTACHMENT 7
Page 1 of 1

DIESEL GENERATOR LOAD MONITORING (SALEM ONLY)

1. EVALUATED loading of any and all running Emergency Diesel Generators against the "Excessive Load Chart" below:

EXCESSIVE LOAD CHART

TIME LIMIT	*MAX KW RATING*	*ASSUMED PF*	*MAX KVAR RATING*	*MAX KVA RATING*
1/2 HOUR	3100	.8	2325	3875
2 HOURS	2860	.8	2145	3575
2000 HOURS	2750	.8	2063	3438
CONTINUOUS	2600	.8	1950	3250

2. IF the MAXIMUM LOAD RATING IN KW, KVAR, or KVA is exceeded, recommend to the TSS, using a TSC Technical Support Tracking Form (Attachment 4) of this procedure, that unnecessary loads be secured until the load is less then maximum rating.

3. EVALUATE the cumulative loading effects by performing the following calculation:

   N1/8000 + N2/2000 + N3/730 + N4/.5 = X(effect factor)

   Where: (run time in hours)

   N1 = Run time $\geq$ the continuous rating
   N2 = Run time $\geq$ 2000 hour rating
   N3 = Run time $\geq$ 2 hour rating
   N4 = Run Time $\geq$ 1/2 hour rating

4. IF the effect factor (X above) is approaching or above 1.0, THEN recommend to the TSS, using TSC Technical Support Tracking Form (Attachment 4) of this procedure, that unnecessary loads be secured. Emergency Diesel Inspection or Overhaul should be considered per manufacture guidance.

5. DISCONTINUE this attachment if no Emergency diesel generators are powering a vital bus.

MAINTENANCE OF FUEL OIL TO SALEM EDG(SALEM ONLY)

EP96-001

Initials

1. REVIEW available MAJOR EQUIPMENT AND ELECTRICAL STATUS sheets ELECTRICAL STATUS section and determine current Emergency Diesel Generator (EDG) use. ________ TSTL

2. EVALUATE plant conditions and determine if EDG operations will be required for > 4 hours. ________ TSTL

3. IF EDG operations ARE NOT required for > 4 hours,
THEN notify the TSTL that EDG fuel oil maintenance is not required at this time. Continuously monitor plant conditions for changes that may require re-evaluation of step 2 above. ________ TSTL

4. IF EDG operations ARE required for > 4 hours,
THEN provide the TSS with an "TSC Technical Support Tracking Form" requesting the OS to direct the OSC Coordinator to perform the following actions: ________ TSTL

   A. DETERMINE the status of the 20,000 Barrel Fuel Oil Storage Tank (FOST) and associated transfer piping used to fill the Diesel Fuel Oil Storage Tanks (DFOSTs).

   B. IF plant conditions preclude Operations Dept. routine checks of DFOST levels at the prescribed frequency (e.g. radiological or higher priority concerns),
   THEN assemble and dispatch a team to evaluate DFOST inventory.

   C. IF normal fuel oil transfer capability DOES exists,
   THEN maintain DFOSTs level greater than Tech Spec limits IAW Operations Dept. Normal Operating Procedures.

   D. IF normal fuel oil transfer capability DOES NOT exist,
   THEN request the TSM (NETS 5007) to develop an action plan for alternative EDG fueling methods per TSC Technical Support Tracking Form (Attachment 4) and inform the OS that you have requested EOF Technical Support.

ATTACHMENT 8 (cont.)
Page 2of 2

MAINTENANCE OF FUEL OIL TO SALEM EDG(SALEM ONLY)

> **NOTE**
>
> Maplewood Lab should be contacted to provide onsite support for fuel oil sampling.

Initials

5. WHEN requested to develop an action plan for alternative EDG fueling methods to the DFOSTs,
THEN implement the following actions: ________ TSTL

   B. COORDINATE with the EOF Purchasing Support to procure and deliver fuel oil, see note below, from an offsite source and fill the DFOST from the Emergency fill Connection.

> **NOTE**
>
> The following diesel fuel oil suppliers and transport companies have current enforceable contracts with PSE&G for emergency fuel oil supply:

Fuel suppliers	Fuel transport companies if supplier cannot transport
Amerada Hess Corp.	Dana transport Inc.
Coastal Oil NY	Marshall Service Inc.
Ross Fogg Oil Corp	S. J. Transportation Inc.

   C. IF needed as a backup to action "A" above, THEN assemble temporary hoses and pumps to transfer fuel oil from the FOST to the DFOST Emergency Fill Connection.

6. COMPLETE and forward the selected action plan for alternative EDG fueling methods on a "TSC Technical Support Tracking Form." ________ TSTL

ATTACHMENT 9
Page 1 of 3

TECHNICAL SUPPORT TEAM LEADER CHECKLIST

NOTE

SALEM – Refer to page 3 of this attachment for additional actions required at Salem Station

	Initials
1. REPORT to the TSS and obtain a briefing.	______ TSTL
2. INITIATE and maintain a chronological log of your activities.	______ TSTL
3. BRIEF team members on the plant and emergency status.	______ TSTL
4. DIRECT Core Thermal Hydraulics Engineer to implement Attachment 11.	______ TSTL
5. DIRECT TSTM(s) to verify SPDS and CRIDS availability.	______ TSTL
6. DIRECT a TSTM to monitor plant activities on the OSC Radio Monitor.	______ TSTL
7. WHEN staffing is adequate to perform the technical support functions, THEN inform the TSS. Request the Administrative Support Supervisor to callout additional personnel as required.	______ TSTL
8. ASSIGN TSTM(s) specific tasks using TSC Technical Support Tracking Form (Attachment 4) of this procedure.	______ TSTL
9. DOCUMENT all assigned tasks using Attachment 3, TSC Engineering Task Assignment Log of this procedure.	______ TSTL

ATTACHMENT 9
Page 2 of 3

TECHNICAL SUPPORT TEAM LEADER CHECKLIST

Initials

10. ENSURE the engineer's OPERATIONAL STATUS BOARD is updated every 15 minutes: ______ TSTL

HOPE CREEK

a. Obtain the operational information by requesting the TSC Communicators to provide the information from VAX LA 120 printer, Menu Option #2 or from the Control Room Communicators.

OR

b. Obtain the information from CRIDS PAGE DISPLAY #232.

SALEM

a. Obtaining the information from SPDS UNIT MASTER MENU OPTION #9 (Press Shift #9 on the Unit Master Menu)

OR

b. Request the TSC Communicators to obtain the information from the Control Room Communicators

11. ENSURE the engineer's MAJOR EQUIPMENT & ELECTRICAL STATUS BOARD is updated once per event or upon any significant change in plant status. ______ TSTL

12. ESTABLISH communications with the Technical Support Manager (TSM) at the EOF. Provide an update on engineering activities and request engineering support, if required. ______ TSTL

13. REVIEW and forward all Attachment 4, TSC Technical Support Tracking Forms to the TSS. (Retain a copy for formal documentation). ______ TSTL

14. COMPLETE Attachment 10, Turnover Log - Technical Support Team Leader, when being relieved. ______ TSTL

15. AT the conclusion of the event, THEN insure that the team's area is returned to ready status and that all paperwork related to the event is turned over to the TSS. ______ TSTL

TECHNICAL SUPPORT TEAM LEADER CHECKLIST

SALEM ONLY

1. IF any Vital Bus is powered from an emergency diesel generator, THEN direct staff member to implement Attachment 7, "DG Load Monitoring", of this procedure.

   ______ TSTL

2. IF Salem has experienced an Operating Basis Earthquake (OBE), THEN coordinate implementation of the spent fuel rack inspection IAW SC.DE-TS.ZZ-4406(Q), Evaluation of Post-OBE Rack-to-Rack and Rack-to-Wall Gaps. (EP96-002)

   ______ TSTL

3. MONITOR the Spent Fuel Pool (SPF) Temperature (local indication). (EP99-001) when SPF temperature increases past 125^0 F

   a. IF the temperature reaches 149^0 F., THEN COORDINATE through the EDO to have Operations restore one SFP Cooling Pump and one Heat Exchanger to cool down the SFP IAW S1(2).OP-SO.SF-0002(Q), SPENT FUEL COOLING SYSTEM OPERATION.
   b. ASSURE this action is taken in sufficient time to prevent exceeding the pool design temperature of 180^0 F. (It should take approximately 18 hours to reach 180^0 F.)
   c. MONITOR the SFP temperature until it returns to < 125^0 F.

   ______ TSTL

4. MONITOR the status of Emergency Diesel Generator (EDG) fuel oil availability by implementing Attachment 8, Maintenance of Fuel Oil to Salem Emergency Diesel Generators, as appropriate. (EP96-001)

   ______ TSTL

5. IF both SI pumps have been running for > 24 hours, Then monitor SI pump room temperatures (P250 and/or local monitoring) at least every 12 hours. If room temperature exceeds 120°F, then complete a TSC Technical Support Tracking Form (Attachment 4) which recommends the following to the OS:

   a. Stop one of the two running SI pumps.
   b. Stop the SI pump room cooler fan, if running.
   c. If room temperature continues to increase above 120°F, then request Site Protection in the OSC to put together temporary ventilation to the SI pump room if the area is accessible.

   ______ TSTL

6. REQUEST the SPDS Engineer to down load the SPDS memory every 12 hours. The SPDS Engineer's phone numbers are listed in NC.EP-WB.ZZ-0001(Z), EP Phone # Directory.

   ______ TSTL

TURNOVER LOG - TECHNICAL SUPPORT TEAM LEADER

Date: ____/____

1. [UE] [A] [SAE] [GE] was declared @ ____________ hrs. on __/__/__

   Due to: ____
   ____

2. The present classification, [A] [SAE] [GE] was declared @ was declared ____________ hrs. on ____/____/____

   Due to: ____
   ____

3. The Emergency Coordinator (EC) is ______________________________
   in the [EOF] [TSC] name

4. The Oncoming and Offgoing TSTL shall:

	Initials On / Off
A. DISCUSS current conditions. Include any problems encountered or anticipated, and any ongoing, or pending technical support assignments.	____/____
B. REVIEW all applicable documentation including procedures, logs, etc., ensuring they are completed, correct and signed.	____/____
C. DISCUSS the team's priorities, personnel requirements and any support or material needs.	____/____
D. DISCUSS any Radiological, Safety, or Environmental concerns.	____/____
E. INSURE that technical support assignments are completed or reassigned prior to TSTM(s) being relieved or dismissed.	____/____
F. NOTIFY the TSS and the EDO of the change in command of the TSTL. Update plant status and priorities as applicable.	____/____

______________________ ______ ______________________
Oncoming TSTL signature time Offgoing TSTL signature

CORE THERMAL HYDRAULICS ENGINEER CHECKLIST

The Core Thermal Hydraulics Engineer (CTHE) shall:

	Initials
1. REPORT to TSTL upon arrival and receive assignment.	________ CTHE
2. ANALYZE core thermal hydraulic parameters to determine current conditions of the core.	________ CTHE
3. DEVELOP recommendations concerning plant operations to maintain safe core conditions.	________ CTHE
4. PROVIDE support to the Control Room staff on core reactivity conditions such as shutdown margin, boration requirements, control rod movements or patterns, etc.	________ CTHE
5. EVALUATE fuel damage based on core thermal conditions, rad monitoring, and/or specific chemistry sample results:	________ CTHE
HOPE CREEK IAW HC.EP-EP.ZZ-0205, "TSC – Post Accident Core Damage Assessment". **SALEM** IAW SC.EP-EP ZZ-0205, "TSC – Post Accident Core Damage Assessment".	
6. COORDINATE fuel damage assessments with the TSC Chemistry Supervisor and inform TSTL of results.	________ CTHE
7. FORWARD all procedures, forms, etc., to the TSTL when the emergency is terminated.	________ CTHE

PSE&G CONTROL COPY # EP1P059

OPERATIONS SUPPORT CENTER (OSC) ACTIVATION AND OPERATIONS

USE CATEGORY: II

REVISION SUMMARY:

Biennial Review Performed Yes X No___

Reference to the Storekeeper has been removed to agree with changes to the Emergency Plan. The Storekeeper's duties have been combined with the Planner's duties.

Attachment 8 was formatted to match the order of the briefing checklist.

IMPLEMENTATION REQUIREMENTS

2/6/02

APPROVED: ____________________ 1-29-02
EP Manager Date

APPROVED: ____________________ 1-29-02
Vice President - Operations Date

OPERATIONS SUPPORT CENTER (OSC) ACTIVATION AND OPERATIONS

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides guidance to emergency response personnel for administration of Operations Support Center (OSC) activation and operations during an emergency at Hope Creek or Salem Nuclear Generating Stations.

2.0 PREREQUISITES

2.1 Prerequisites To Be Followed Prior To Implementing This Procedure

2.1.1 Implement this procedure:

- At the request of the Operations Superintendent (OS) or the Emergency Duty Officer (EDO).
- Upon staffing of your Emergency Response Facility.

2.1.2 Steps within a section may be completed in any order.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions and Limitations To Be Followed Prior To Implementing This Procedure:

3.1.1 It is recommended that initials be used in the place keeping sign-offs, instead of checkmarks if more than one person may implement this procedure.

3.1.2 Personnel who implement this procedure shall be trained and qualified IAW the Emergency Plan.

3.1.3 Attachment 6 page 2 of 2 should only be completed if the card readers for the accountability system fail.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 PROCEDURE

NOTE		
OSC Coordinator responsibility is filled by		
_______________ **Name**	_______ **Date**	_______ **Time**

5.1 Preparation for Activation

5.1.1 NOTIFY the following OSC support personnel to report to the OSC:

a. Shift Electrician ________

b. Shift I&C Technician ________

c. Radwaste Operator ________

d. Equipment Operators ________

e. CALL the Fire Protection Supervisor (ext. 2800) or use the plant page, and direct the Fire Brigade to the OSC. ________

NOTE
The Fire Brigade normally brings equipment to the station that has declared the emergency. Advise the Fire Protection Supervisor of conditions that would not permit leaving equipment in any location

5.1.2 DESIGNATE at least two individuals to set up the OSC IAW Attachment 7, OSC SETUP ________

IF Hope Creek
THEN implement Attachment 7a, OSC Setup, Hope Creek.

IF Salem
THEN implement Attachment 7b, OSC Setup, Salem

5.1.3 REPORT to the Control Room (CR) and obtain the following information from the Operations Superintendent (OS): ________

a. Update of emergency status. ________

b. Plant conditions including any known radiological problems. ________

c. Status of personnel dispatched into the plant by the CR staff. ________

d. **(Salem Only)** Discuss Auxiliary Building Ventilation System Status and ensure that One Supply and Two Exhaust Fans are operating properly. If less than the normal compliment of fans is in service, then request the OS prioritize actions IAW Attachment 2 of the Operating Procedure. Ensure design based auxiliary building temperatures are maintained. The aux. building ventilation system needs to be restored within 2 hours following an accident initiation. ________

5.1.4 ENSURE OSC clock time is approximately the same as the CR/SPDS time. ________

5.1.5 IMPLEMENT the following steps in any order:

- IMPLEMENT Attachment 2 - Accountability Instructions, when Accountability is called. ________
- DIRECT one individual to ensure all OSC personnel sign in on the Attachment 6, OSC Sign In Roster. DO NOT have control room personnel sign in at this time on Attachment 6 ________
- ASSEMBLE available OSC personnel and brief them on the status of the emergency. ________
- DIRECT non-supervisory OSC personnel who are not assisting in set up of the OSC to assemble in the OSC Ready Room. ________
- DIRECT the Radiation Protection Supervisor-Exposure Control (RPS-EXP) or Shift Radiation Protection Technician to implement NC.EP-EP.ZZ-0304(Q), OSC – Radiation Protection Response. ________

5.1.6 ACTIVATE the OSC as follows: ________

- INFORM the OS and OSC personnel that the OSC is activated, at (time) __________ on (date)____/____/____ ________
- MAKE the following page announcement (Announce Twice): ________

"The OSC is now activated. All Plant Activities should be coordinated thru the OSC."

NOTE

Each individual's annual limit will automatically be raised to 4500 mRem upon the declaration of an Alert or higher emergency classification provided a completed NRC FORM-4 is on record.

NOTE

The following steps should be completed simultaneously

5.2 **Post Activation of the OSC** DIRECT an OSC Supervisor or OSC staff member to periodically brief the OSC READY ROOM on plant conditions and update the EMERGENCY INFORMATION OPS READY ROOM status board in the OSC Ready Room.

5.2.1 WHEN Accountability is called, THEN:

a. IMPLEMENT Attachment 2, ACCOUNTABILITY INSTRUCTIONS. ________

b. **(SALEM ONLY)** IF requested THEN IMPLEMENT Attachment 9, Assembly/Accountability Instructions for High Noise Areas – OSC (Salem Only). ________

5.2.2 IF directed to perform bomb search activities, THEN IMPLEMENT Attachment 1, BOMB SEARCH Operations. ________

5.2.3 IF loss of habitability occurs for: ________

- The OSC Ready Room, THEN direct personnel to the Technical Support Center (TSC) or other location as directed by the OS/EDO.
- The OSC, THEN direct OSC personnel to the TSC or other location as directed by the OS/EDO. ________

5.2.4 DIRECT Ops, Mechanical, I&C, Fire Protection, and Electrical Supervisors to implement Attachment 5, OPS/MECH/ELECT/I&C/FIRE PROTECTION DUTIES. ________

5.2.5 DIRECT the Planner to implement Attachment 4, PLANNER DUTIES. ______

5.2.6 DIRECT the OSC Clerk to: ______

a. MAINTAIN the OSC Coordinator's Log. ______

b. PROVIDE the log to the OSC Coordinator for periodic review. ______

c. MAINTAIN the OSC Team Status Board and Priority Status Board. ______

5.2.7 ASSEMBLE available OSC personnel and brief on the emergency. ______

5.2.8 DIRECT the OSC Clerk report to the TSC, if conditions degrade where respirators are needed for OSC personnel.

5.2.9 ENSURE the OSC priorities and activities are periodically discussed with the OS. ______

5.2.10 BRIEF the oncoming OSC Coordinator and conduct the turnover IAW Attachment 3, OSCC TURNOVER LOG. ______

5.2.11 IF additional personnel are needed, (e.g. Duty Storekeeper) contact the ADMSS in the TSC. ______

5.3 Restoration

ENSURE the OSC is restored to a ready status. ______

6.0 RECORDS

Return completed procedure, attachments, and or forms to the EP Manager.

7.0 REFERENCES

7.1 References

None

7.2 Cross References

PSEG Nuclear Emergency Plan

ATTACHMENT 1
Page 1 of 3
BOMB SEARCH OPERATIONS

1.0 BOMB SEARCH OPERATIONS

1.1 Perform the following for bomb search operations, as directed by the OS/EDO:

1.1.1 The OSC Coordinator should:

CAUTION
Portable radios and portable phones shall not be used during a bomb search. Radio transmissions may cause an explosive device to detonate.

A. MAKE the following page announcement TWICE:

"Attention all personnel, terminate use of all portable radios phones until further notice."

B. CALL the Salem OS and request they make the proceeding page announcement.

C. CALL PSEG Security Supervisor (ext. 2222) and request that a Security Team Leader or Security Technical Analyst reports to the OSC.

D. OBTAIN a briefing from the OS concerning the requested bomb search to include, as a minimum:
- Nature of the bomb threat
- Specific areas mentioned in the threat
- Current bomb search activities
- Restrictions due to radiological concerns
- Copy of a completed Bomb Threat Checklist and Report Form if available.

E. DIRECT the Security Team Leader or Security Technical Analyst, assigned to the OSC, to recommend bomb search areas priorities, in accordance with Security Contingency Procedures.

ATTACHMENT 1
Page 2 of 3

F. MODIFY bomb search priorities recommended by the Security Team Leader or Security Technical Analyst as required due to plant operational and/or radiological conditions.

G. Assemble bomb search teams that should consist of a Plant Operator as the Team Leader and a Security Force Member (SFM).
- ♦ Fire Protection Operators may be utilized if Plant Operators are not available
- ♦ Radiation Protection Technicians (RPT) should be assigned to bomb search teams in the RCA.

H. BRIEF the bomb search teams, in conjunction with the Security Team Leader or Security Technical Analyst and the SRPT/RPS – EXP.

I. OFFER the team leader a copy of Attachment 8, TEAM LEADER BRIEFING SHEET.

J. BRIEF each bomb search team to include discussion of the following:
- ♦ Information known from the bomb threat.
- ♦ Assigned search area(s) for each team.
- ♦ Operational and Radiological concerns such as exposure limits, dosimetry, protective clothing, etc.
- ♦ Abort conditions.
- ♦ OSC phone call back numbers for team communications.
- ♦ Instruct each team member to look for physical evidence of suspected sabotage in addition to the location of destructive device(s).
    1. **Mechanical** (valve alignment, piping, loose ducts, structures, normal running equipment checks).
    2. **Electrical** (checks for any breakers in an off-normal position, open cabinets, open conduit, etc.).
    3. **Physical Barriers** (blocked open doors, misplaced contamination/high radiation areas, etc.).
    4. **Suspicious looking articles** (lunch boxes, tool boxes, packing crates, etc.).
    5. **Abnormal log readings/indications for the area.**

K. Using the plant page or hard wire phone, inform the OSC Coordinator immediately upon locating a potential bomb or sabotaged area.

ATTACHMENT 1
Page 3 of 3

L. IF a bomb is discovered, THEN:

a. ENSURE that all personnel in the vicinity of the bomb are relocated to a safe area and direct the bomb search team to secure access to the area.
b. CONTACT the OS, the OSC coordinator, the OSC Security Force Supervisor and EDO to take appropriate actions.

M. RECALL, debrief and deactivate the Bomb Search Team when:

a. Completion of search with negative results is reported.
b. Termination of the bomb threat.

ATTACHMENT 2
Page 1 of 3
ACCOUNTABILITY INSTRUCTIONS - OSC

1.0 ACCOUNTABILITY INSTRUCTIONS

1.1 Upon Hearing the Page Announcement to Implement Accountability, the OSC Coordinator Shall:

Initials

1.1.1 ENSURE OSC and Control Room personnel are listed on the Attachment 6, OSC SIGN IN ROSTER (only obtain control room sign-in if back up Accountability method is being used.) ______ OSCC

> **NOTE**
>
> Personnel who have carded into the Control Room Security Area are not required to pass their photobadges through the OSC accountability station cardreader. This is because Control Room personnel are accounted for automatically by the access door cardreaders into the Control Room.

1.1.2 Upon hearing the page announcement, **"All ACCOUNTABILITY STATIONS IMPLEMENT ACCOUNTABILITY":**

A. DIRECT all personnel who are not within the Control Room Security Area to pass their photobadges through the special accountability cardreader located just outside the OSC. ______ OSCC

B. DIRECT all OSC teams in the field to verbally accounted for by using an ACCOUNTABILITY EXEMPTION FORM, page 3 of this Attachment. ______ OSCC

1.1.3 MAINTAIN current status of all OSC personnel. This should include name, assignment, time dispatched. ______ OSCC

ATTACHMENT-2
Page 2 of 3

1.1.4 Upon hearing the page announcement, **"All ACCOUNTABILITY STATIONS COMPLETE YOUR 3O MINUTE ACCOUNTABILITY,"**

A. ENSURE that any personnel arriving at the OSC since the initial call for accountability also have passed their photobadges through the OSC accountability cardreader. ______ OSCC

B. ENSURE the badge numbers of personnel listed on page 3 of this attachment (Accountability Exception Form) are called to security at extension 2222. If unable to contact security on 2222, contact the security liaison at NETS 5214 (Hope Creek) or NETS 5217 (Salem) ______ OSCC

1.1.5 IF personnel arrive at the OSC after 30 minutes have elapsed, THEN:

A. NOTIFY security of the badge numbers of the personnel that have reported to the OSC using ext. 2222. ______ OSCC

B. DIRECT those personnel to:

1. PASS their photobadge through the accountability cardreader. ______ OSCC

2. SIGN Attachment 6, OSC SIGN IN ROSTER if not already signed in ______ OSCC

1.1.6 IF directed,
THEN form search teams for unaccounted for personnel. ______ OSCC

1.1.7 MAINTAIN continuous accountability until the emergency is terminated. ______ OSCC

2.0 <u>AUTOMATED ACCOUNTABILITY SYSTEM MALFUNCTION</u>:

2.1 <u>Perform the Following Should the Automated Accountability System Malfunction</u>:

2.1.1 PROVIDE a copy of Attachment 6, OSC SIGN IN ROSTER Sheet(s), to the Security Force Member dispatched to the OSC. ______ OSCC

ATTACHMENT 2
Page 3 of 3

ACCOUNTABILITY EXEMPTION FORM
OSC ________________ STATION

NAME (print)		BADGE NUMBER
LAST	FIRST	

NOTE

Call Security at extension 2222 and provide the badge numbers of the personnel listed above. The personnel listed above have been accounted for verbally. These personnel are involved in actions to mitigate emergency events. If unable to contact security at extension 2222, contact the TSC security liaison at NETS 5214 (Hope Creek) or NETS 5217 (Salem).

______________________________ ________/________

Signature (OSC Coordinator or OS) Time Date

ATTACHMENT 3
Page 1 of 1
OSCC TURNOVER LOG

Date: ______/___

1. [UE] [A] [SAE] [GE] was declared @ ____________ hrs. on __/__/__

   Due to: ____
   ____

2. The present classification, [A] [SAE] [GE] was declared @ ____________ hrs. on ____/____/____

   Due to: ____
   ____

3. The Emergency Coordinator (EC) is ________________________________ in the [EOF] [TSC] [CR] name

4. The Oncoming and Offgoing OSCC should:

	Initials On / Off
A. DISCUSS current conditions. Include any problems encountered or anticipated, and any ongoing, or expected actions.	_____/_____
B. REVIEW all applicable documentation including procedures, logs, etc., ensuring they are completed, correct and signed.	_____/_____
C. DISCUSS the OSC's priorities, personnel requirements and any equipment or material needs.	_____/_____
D. DISCUSS any Radiological, Safety, or Environmental concerns.	_____/_____
E. ENSURE that OSC teams in the field are recalled or relieved on the job as necessary.	_____/_____
F. NOTIFY the OS of the change in command in the OSC. Update plant status and priorities as applicable.	__________ (ONCOMING)
G. HOLD a briefing with the OSC staff to ensure a smooth transition between the oncoming and off-going OSC personnel.	__________ (ONCOMING)

ATTACHMENT 4
Page 1 of 1
PLANNER DUTIES

NOTE:

When the OSCC is activated, the Work Control Office becomes an extension of the OSC.

1.0 **Planner Duties**

1.1 **Perform the Following, as Appropriate:**

1.1.1 LOG on to SAP in Work Control on any available computer with LAN capabilities.

1.1.2 SELECT a Personnel Computer in Work Control to run the parts program, if necessary

1.1.3 OBTAIN an inventory list of available parts for PSEG Nuclear.

1.1.4 TRACK any items issued and document for post-emergency review

1.1.5 PROVIDE current status of repairs in progress.

1.1.6 INITIATE Work Order/Notification packages as necessary for jobs originating in the OSC.

- **IF** time is not available,
  AND
  at the discretion of the OSCC,
  THEN use a paper work request and follow up with a Work Order/Notification as soon as possible.

1.1.7 PROVIDE guidance as to procedures, tools, and time that past repairs required.

1.1.8 As requested, obtain spare parts, materials, tools, etc.

1.1.9 ASSIST the other OSC staff members as directed by the OSCC including answering phones, page announcements and OSC ready room updates.

1.1.10 COLLECT all work orders and paper work for review and documentation after the emergency has been terminated.

ATTACHMENT 5
Page 1 of 4
OPS/MECH/ELECT/I&C/FIRE PROTECTION DUTIES

1.0 **OSC DUTIES**

1.1 **Perform the Following Duties, as Appropriate:**

1.1.1 PROVIDE corrective action recommendations based on your expertise to the OSCC for accident mitigating activities.

1.1.2 MONITOR Oxygen content in the OSC and Control Room every hour.

1.1.3 MONITOR Fire Protection System Status and ensure compensatory measures (fire watch) are maintained in accordance with plant conditions.

NOTE

1. An individual, in lieu of a team, may be dispatched by the OSCC. The individual should be in contact with the OSC or the Control Point, via some type of audio communications (page, radio, or telephone), and should check in every 15 minutes with the OSC or the Control Point.

   A. An individual shall not be used under the following circumstances:

      - An individual's exposure could be expected to exceed 1000 mRem.
      - The task would require entry into a "Harsh Environment Area" (i.e., a steam atmosphere, a heat stress area, unknown, etc.)
      - Acts of sabotage or suspected sabotage.

2. Prompt team dispatch is of the utmost importance. If necessary, a radiological briefing may be conducted on the way to the job.

1.1.4 ASSEMBLE teams and provide team briefings and debriefings in your areas of expertise.

- ENSURE that all OSC teams are made up of at least two people unless a task meets the criteria from the note above.
- ENSURE that teams dispatched from the Control Point are tracked using the OSC Team Status Board.
- ASSEMBLE the Team and designate a Team Leader.
- IF adverse radiological conditions exist, THEN a Radiation Protection Technician (RPT) should be included on the team.
- IF a Personnel Injury/Medical Response incident occurs, THEN a Fire Brigade Member should be included on the team.

- OFFER Attachment 8, Team Leader Briefing Sheet, to each team leader.
- ENSURE all Teams obtain a briefing IAW, Team Briefing Guidelines of this Attachment.

CAUTION

Portable radios are not to be keyed in the vicinity of the Control Room or Relay Room areas.

Any portable radio and cellular phone transmissions are prohibited during bomb search operations.

- IF a Team is dispatched, THEN ENSURE the OSC Team Status Board is updated.
- MONITOR and support dispatched teams (Point of Contact).
- RECALL the Team IF:
  1. The mission or objective is accomplished.
  2. Directed by OS (higher priority, etc.).
  3. Plant conditions degrade to where an individual's exposure to unanticipated changes in radiological conditions:
     - May exceed authorized exposure limit.
     - Life threatening environmental conditions are encountered.
- WHEN the team returns, CONDUCT a general debriefing.
- COLLECT all procedures, attachments, etc. and retain to forward to the Manager CA/EP/IT.
- RELEASE team to RPT/RPS – EXP for a radiological debriefing.
- COORDINATE with the RAC for the transportation of injured person or personnel receiving Emergency Exposures of $\geq$ 5 REM to the appropriate medical facilities, if necessary.
- Deactivate the team and ensure the OSC Team Status Board is updated.

ATTACHMENT 5
Page 3 of 4

1.1.5 INTERFACE with the TSC Engineers for their support when needed by:

- REQUEST the OSCC to contact the OS for technical support, as needed.
- WHEN TSC Technical support is requested, THEN OBTAIN a copy of the Technical Support Tracking Form from the OSCC for implementing mitigating actions recommended by the engineering staff, which are not addressed in a procedure.
- ENSURE the Technical Support Tracking Form is approved for implementation by the OS.

NOTE

The Work Control Office becomes an extension of the OSC when the OSC is activated

1.1.6 ASSIST the OSC Planner in Work Request/Notification generation IAW the appropriate NAPs.

- Initiate tagging requests in Work Control.
- Request the Operations Supervisor assist in tag request generation.

1.1.7 If acts of sabotage are expected, team safety must be the number one priority. Contact security in the TSC to determine what areas are safe for personnel dispatch. If any area is considered unsafe by security, discuss delaying team dispatch until the area is cleared by security.

ATTACHMENT 5
Page 4 of 4
TEAM BRIEFING GUIDELINES

A. Mission Details	1. Designate a team leader 2. Explain purpose of mission 3. Define work to be performed 4. State expected results 5. Team identifier (team #)
B. Safety	1. Ensure required safety gear is used 2. Equipment C/T for work 3. Review Environmental Concerns: ■ Heat stress ■ Oxygen deficient atmosphere ■ Toxic/caustic chemical concerns 4. Fire hazards
C. Communications	1. Establish preferred method ■ Portable radios ■ Phones (Dimension) ■ Other (sound pwrd phones, etc.) 2. Point of contact 3. Check in time cycle 4. Estimated job completion time
D. Equipment	1. Required tools 2. Test equipment 3. Procedures (current copy[s] as required) 4. Availability/location of spare parts 5. Authorization for "cannibalization" of other equipment for parts
E. Radiological	1. Discuss expected dose rates 2. Establish stay times 3. Establish abort conditions criteria 4. Identify desired routes 5. Review rwp with the team including dress out criteria 6. Review exposure limits 7. Obtain high rad. key if required 8. Ensure the control point is notified.
F. Debrief	1. Team is expected to debrief the assigning supervisor when the job is complete or the team is recalled.

OSC SIGN IN ROSTER/ACCOUNTABILITY FORM

(PRINT NAME)

POSITION	NAME	BADGE #	POSITION	NAME	BADGE #
OSC COORD			RAD PRO SUPV.		
OSC CLERK			R.P. SUPPORT		
			R. P. SUPPORT		
OPS SUPV			FIRE PROT. SUPV.		
OPERATOR			FIRE BRIG. MBR.		
OPERATOR			FIRE BRIG. MBR		
OPERATOR			FIRE BRIG. MBR		
OPERATOR			FIRE BRIG. MBR		
OPERATOR			FIRE BRIG. MBR		
OPERATOR			FIRE BRIG. MBR		
OPERATOR					
OPERATOR			PLANNER		
MAINT.SUPV. ( )			**ADDITIONAL OSC SUPPORT PERSONNEL**		
MAINT.SUPV. ( )			**POSTION**	**NAME**	**BADGE #**
SHIFT ELECT.					
SHIFT I&C TECH					
MAINT.SUPPORT ( )					
MAINT.SUPPORT ( )					
MAINT.SUPPORT ( )					
MAINT.SUPPORT ( )					

( ) INDICATESJOB CLASSIFICATION (i.e. WELD, MACH, BLR, REP, ETC.)

OSC SIGN IN ROSTER/ACCOUNTABILITY FORM

(PRINT NAME)

POSITION	NAME	BADGE #	POSITION	NAME	BADGE #
OS					
CRS					
STA					
SRO 1					
SRO 2					
NCO					
NCO					
NCO					
NCO					
COMMUNICATOR 1					
COMMUNICATOR 2					
OPS ADVISOR					

ADDITIONAL CONTROL ROOM/OSC PERSONNEL

OSC Coordinator Signature ______________________

COMMENTS

1.0 **HOPE CREEK OSC SETUP**

1.1 **Perform The Following to Setup the OSC:**

1.1.1 PROCEED to the OSC storage closet located in the North West corner of the OSC.

- BREAK the glass on the key box located on the wall next to the storage closet and obtain the keys for the closet, if necessary.

1.1.2 OBTAIN equipment and supplies stored in the OSC locker:

- REMOVE cordless phones from chargers.
- DISTRIBUTE OSC cordless phones IAW OSC Setup Diagram found in this Attachment. The setup is for guidance only. Personnel may be relocated around the table as needed or at the discretion of the OSC coordinator.
- PERFORM the following in case of cordless phone failure:
  A. PLACE the phone strip under the OSC table.
  B. IMMEDIATELY SETUP NETS and DID phones IAW OSC Setup Diagram found in this Attachment.

1.1.3 PLACE the OSC base radio station in operation as follows:

- TURN the OSC radio's power on. (The **On/Off Switch** is on the front of the unit).
- VERIFY the Control Room radio **TAKEOVER** yellow indicator is illuminated. If **TAKEOVER** green indicator is illuminated depress the **TAKEOVER** push button.
- POSITION the **FREQUENCY BUTTON** to the UP position for frequency #3.
- POSITION the HEADSET TOGGLE SWITCH to the DOWN position for headset use or in the UP position for speaker use.
- POSITION **VOLUME CONTROL** to a comfortable level. **(VOLUME CONTROL SWITCH** is on the front of the radio).
- PRESS the **TRANSMIT BAR** on the microphone to transmit a message when the headset **OFF**.
- PRESS the hand held transmit button on the headset cord to transmit, if the headset is **ON.**

ATTACHMENT 7a
Page 2 of 3

- ENSURE OSC hand held radios are selected to the same frequency as the base station radio.
- TEST the base station radio to a hand held radio.

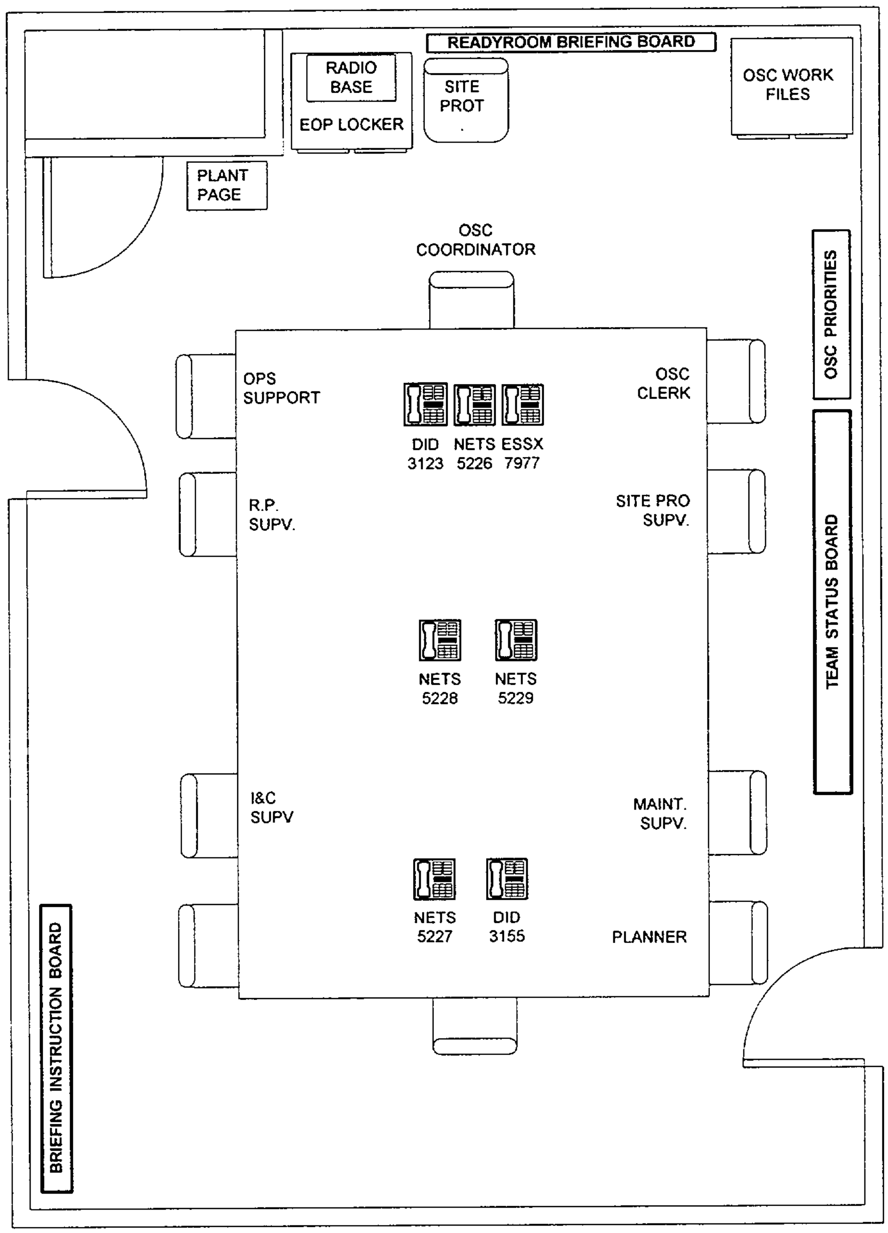
1.1.4 INCREASE or decrease the OSC page volume control to a comfortable level.

1.1.5 POST a clean copy of the OSC TEAM STATUS BOARD in the OSC.

1.1.6 SETUP the OSC Priority Status Board in the Control Room. This board is found in the storage closet in the CRS office area in the Control Room Complex.

ATTACHMENT 7a
Page 3 of 3

HOPE CREEK GENERATING STATION
OSC FLOOR PLAN



OSC

1.0 **SALEM OSC SETUP**

1.1 **Perform The Following to Setup the OSC:**

1.1.1 OBTAIN the keys for the OSC cabinets from the Red Key Box on the wall in Work Control. Break the glass if necessary.

1.1.2 OBTAIN equipment and supplies stored in the OSC locker and cabinets.

1.1.3 SETUP the OSC phones IAW figure found in this Attachment 7b. Phone connections are located on the underside of the table. The setup is for guidance only. Personnel may be relocated around the table as needed or determined by the OSC coordinator.

NOTE
The OSC base radio station is permanently set up in the OSC.

1.1.4 PLACE OSC base radio station in operation as follows:

A. TURN the unit power on. (The **On/Off Switch** is on the front of the unit.)

B. VERIFY the Control Room radio **TAKEOVER** yellow indicator is illuminated in the affected unit's Control Room. (IF **TAKEOVER** green indicator is illuminated, THEN depress the **TAKEOVER** push button).

C. POSITION the **FREQUENCY BUTTON** to the UP position for frequency #1 (Unit #1) or DOWN for frequency #2 (Unit #2).

D. POSITION the HEADSET TOGGLE SWITCH to the DOWN position for headset use or in the UP position for speaker use.

E. POSITION **VOLUME CONTROL** to a comfortable level. **(VOLUME CONTROL SWITCH** is on the front of the radio.)

F. DEPRESS the **TRANSMIT BAR** on the microphone to transmit a message when the headset **OFF.**

G. DEPRESS the hand held transmit button on the headset cord to transmit, if the headset is **ON.**

ATTACHMENT 7b
Page 2 of 3

H. ENSURE OSC hand held radios are selected to the same frequency as the base station radio.

I. TEST the base station radio to a hand held radio outside of the Control Room/OSC Complex.

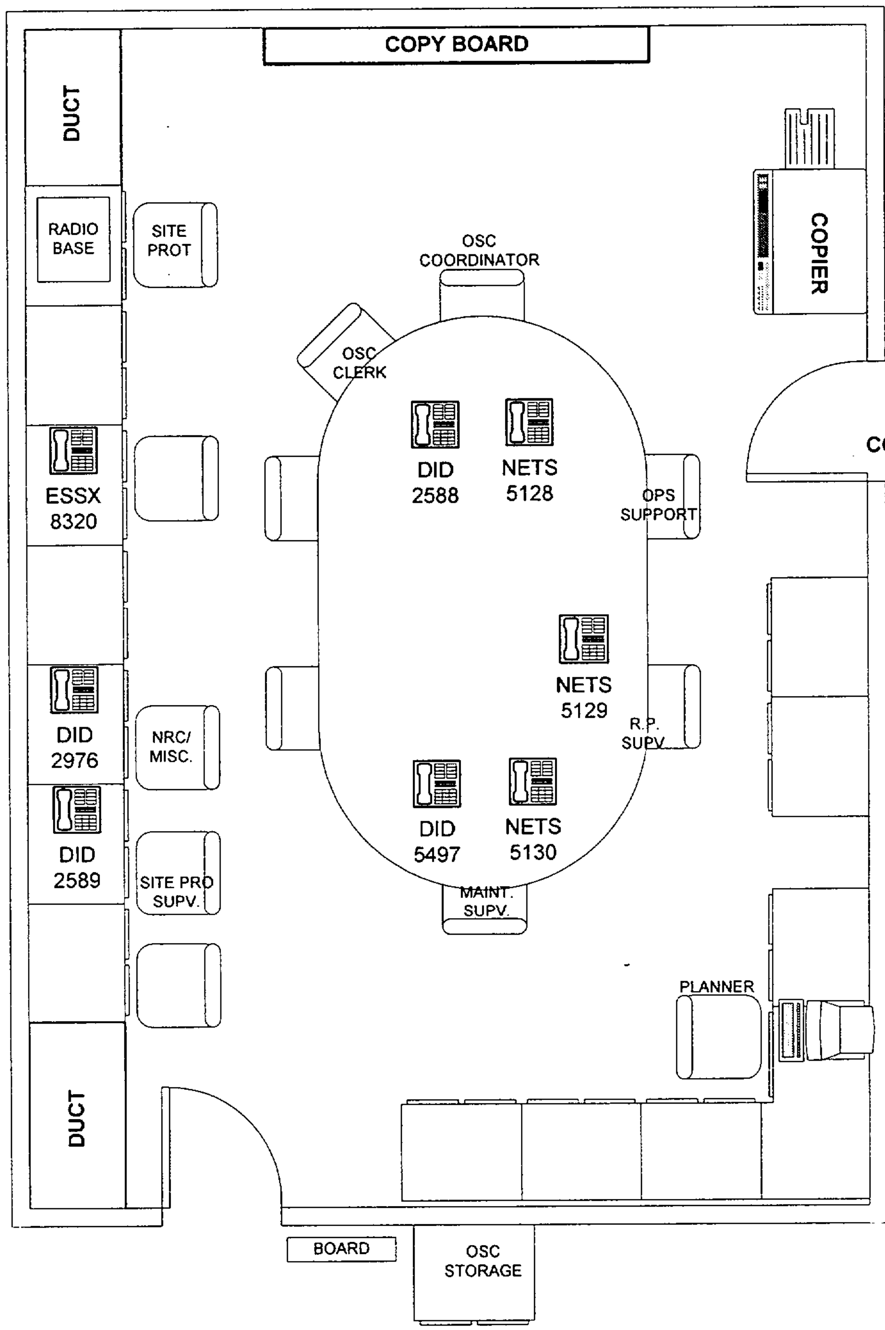
1.1.5 DIRECT the OSC Page turned up to a comfortable level.

1.1.6 SETUP the OSC TEAM STATUS BOARD.

1.1.7 PROVIDE the Control Room with one of the OSC PRIORITY STATUS BOARDS.

1.1.8 SETUP the other OSC PRIORITY STATUS BOARD in the OSC near the OSCC.

SALEM GENERATING STATION
OSC FLOOR PLAN



TEAM BRIEFING AREA

ATTACHMENT 8

TEAM LEADER BRIEFING SHEET

TEAM LEADER ____________ TIME DISPATCHED ____________

TEAM MEMBERS

____

TASK

____

PURPOSE/EXPECTED RESULTS

____

TEAM NUMBER _____

SAFETY EQUIPMENT NEEDED/ENVIRONMENTAL CONCERNS

____

BY RADIO, PLANT PAGE OR DID PHONE # _______ CONTACT _______________
EVERY _________ MIN.

SPECIAL INSTRUCTIONS/EQUIPMENT

____

____

____

BACK OUT DOSE ____________________REM

EXPECTED RADIOLOGICAL CONDITIONS
INDICATE DOSE RECEIVED NEXT TO NAME ABOVE AND INFORM RP SUPERVISOR

____

____

DEBRIEF INFORMATION

INDICATE WHAT WAS OBSERVED IN THE FIELD/TEAM RESULTS. DEBRIEF THE ASSIGNING SUPERVISOR ON FINDINGS/RESULTS

____

____

ATTACHMENT 9
Page 1 of 1
ASSEMBLY/ACCOUNTABILITY INSTRUCTIONS FOR HIGH NOISE AREAS - OSC (SALEM ONLY)

> **NOTE**
>
> This ATTACHMENT shall be implemented upon request by Security Force Supervision to support implementation of **ASSEMBLY** or **ACCOUNTABILITY** for high noise areas.

1.0 HIGH NOISE AREA INSTRUCTIONS FOR ACCOUNTABILITY/ASSEMBLY

1.1 THE OSC COORDINATOR SHALL:

> **NOTE**
>
> Acceptable methods that should be used to check on high noise areas are:
> - Physical Observation.
> - Verification of Key Control.
> - Card Reader Record of Entry (requested from Security).

1.1.1 IF requested by the Security Force Supervision, THEN form teams to check the areas listed below. **(EP97-001)**

- All Trailers within the Protected Area
- Salem Containment (priority during outages)
- Outer Penetrations (Salem U1/U2)
- Inner Penetrations (Salem U1/U2)
- Check all rooms on Aux Building 84' elevation (Salem U1/U2)
- Check all rooms on 45' elevation (Salem U1/U2).
- Salem Service Water Intake Pump Bays
- Salem Circulating Water Intake Structure
- Salem Turbine Bldg. (U1/U2)(el. 88' and 100')

1.1.2 IF any personnel are observed still onsite in any of these high noise areas, THEN INSTRUCT them to either leave site or report to their Accountability Station.

ADMINISTRATIVE SUPPORT/COMMUNICATION TEAM RESPONSE - TSC

USE CATEGORY: II

REVISION SUMMARY: **Biennial Review Yes X No ___**

- Reference to the Security Force Member, OSC Welder, Nuclear Fuels Engineer, one Admin Support and Duty Storekeeper positions have been removed to coincide with the changes to the Emergency Plan.
- Added Operations Advisor to shift manning list.
- Added step to set clocks to SPDS time
- Corrected reference from Manager - EP & IT to Emergency Preparedness Manager
- Added reference to food storage bins.
- Added reference to DTG for facility restoration

IMPLEMENTATION REQUIREMENTS

Effective Date 2/6/02

APPROVED: ______________________ 1-29-02
Emergency Preparedness Manager — Date

APPROVED: ______________________ 1-29-02
Vice President - Operations — Date

ADMINISTRATIVE SUPPORT/COMMUNICATION TEAM RESPONSE - TSC

TABLE OF CONTENTS

1.0 PURPOSE

To provide direction for the emergency actions of the Administrative Support Supervisor (ADMSS), Administrative Support Team, Emergency Preparedness Advisor (EPA), and Communications Team in the Technical Support Center (TSC).

2.0 PREREQUISITES

2.1 Prerequisites To Be Followed Prior To Implementing This Procedure

2.1.1 An emergency has been declared at Salem or Hope Creek Nuclear Generating Station and the TSC is required to be staffed or activated.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Emergency Staffing/Relief:

3.1.1 It is expected that the two duty TSC Communicators will implement this procedure until the arrival of the ADMSS or EPA. Upon arrival, the ADMSS and EPA are expected to continue with implementation of this procedure.

3.1.2 The Administrative Support Supervisor (ADMSS) in the Technical Support Center (TSC) should coordinate the assessment and completion of minimum staffing of the Emergency Response Facilities (ERFs) with the Administrative Support Manager (ASM) in the Emergency Operations Facility (EOF).

3.1.3 After completion of initial staffing, the ADMSS and ASM should begin to assess and arrange relief staffing in order to continue 24-hour minimum staffing of ERFs.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 PROCEDURE

NOTE
The two duty TSC Communicators are expected to implement this procedure until the arrival of the ADMSS or EPA. Upon arrival, the ADMSS and/or EPA are expected to continue with implementation of this procedure.

5.1 **Administrative Support Supervisor and Emergency Preparedness Advisor (TSC1/TSC2 until arrival of the ADMSS/EPA) Duties**

5.1.1 INITIATE and MAINTAIN a chronological log of activities and events. _______ ADMSS/EPA

5.1.2 INFORM Emergency Duty Officer of arrival and when prepared to assume functional duties. _______ ADMSS/EPA

5.1.3 DIRECT staff to verify operation of the following:

A. Telephone lines and Telecopiers _______ ADMSS/EPA

B. Set clocks to SPDS time _______ ADMSS

C. Reprographic equipment _______ ADMSS

5.1.4 COORDINATE mobilization of additional support personnel with the ASM. _______ ADMSS

5.1.5 IMPLEMENT Attachment 1. _______ ADMSS

5.1.6 IMPLEMENT Attachment 2 of this procedure. _______ EPA

6.0 RECORDS

6.1 **The Administrative Support Supervisor Should Ensure The Following**

6.1.1 All written communications and documentation produced during an emergency are important for recording actions taken and reconstruction of events and should not be discarded.

6.1.2 The ADMSS should ensure that TSC Administrative Support Team captures and controls all material received and generated throughout the emergency including all logbooks and original data forms. Attachment 3, Emergency Response Team - Telecopy Log, should be used for all data sent or received over the emergency telecopier network.

6.1.3 All attachments, forms, appendices, and logs are to be retained in accordance with NAP-1 and sent to the Emergency Preparedness Manager, who will ensure all materials are retained.

7.0 REFERENCES

7.1 References

7.1.1 PSEG Nuclear Emergency Plan

7.2 Cross-References

7.2.1 Emergency Response Callout/Personnel Recall EPIP 204H(C)

7.2.2 Emergency Response Callout/Personnel Recall EPIP 204S(C)

ATTACHMENT 1
Page 1 of 2
ADMINISTRATIVE SUPPORT SUPERVISOR (ADMSS) CHECKLIST

1.0 **Prior to Activation of the Facility:**

ENSURE files and emergency response lockers are unlocked; (keys are maintained in the Red Emergency Key Box mounted on wall). ________ ADMSS

ASSIST Emergency Response Team members with distribution of workstation supplies. ________ ADMSS

RUN test copy through copy machine, then, MAKE copies of any data sheets found in Telecopier tray for distribution. ________ ADMSS

ENSURE data sheets are properly date-stamped and initialed prior to distribution. ________ ADMSS

RUN telecopier test to insure proper operation of equipment at the direction of Communications Team. ________ ADMSS

DESIGNATE Administrative Support Team Member to report to the Operations Support Center (OSC) to be the OSC Administrative Clerk if requested by the OSC Coordinator. ________ ADMSS

DESIGNATE Administrative Support Team member to perform telecopier and copy machine duties (i.e. – receive and stamp incoming forms and make sufficient copies for distribution). Utilize Attachment 3, Emergency Response Team - Telecopy Log, to keep track of all forms received and sent. The sequential log number shall be recorded on Attachment 3 and placed in the upper left corner of each telecopied form. ________ ADMSS

DESIGNATE Administrative Support Team member to distribute copies to staff and to properly log completed deliveries. ________ ADMSS

DESIGNATE an Administrative Support Team member to initiate and maintain Emergency Duty Officer's log book and Attachment 5, EDO clerk guidelines. ________ ADMSS

DESIGNATE another Administrative Support Team member to assist in updating key status boards. ________ ADMSS

ATTACHMENT 1
Page 2 of 2

2.0 **Post Activation of Facility**

1. When requested by the EDO to arrange for shift relief, **CONTACT** the Admin Support Manager (ASM) in the EOF and request implementation of "Shift Relief Callout" as per instructions in EPEP 0701. **PROVIDE** the following information:
   - Time of shift relief (12 hour rotation)
   - Any special access instructions

   ______ ADMSS

2. IF automated callout system fails or will not be used, AND EOF requests assistance, **UTILIZE** Attachment 4, Onsite Shift Relief Schedule Manning Chart, as needed.

   ______ ADMSS

3. **DISTRIBUTE** copies of forms as appropriate to each area. **PLACE** 1 copy in each basket. The originals should be maintained in sequential order.

   ______ ADMSS

4. **MAINTAIN** 5 file copies of each procedure and attachment for use within the facility.

   ______ ADMSS

5. **DETERMINE** the needs for food, lodging, equipment, and transportation. **COORDINATE** these needs with the Administrative Support Manager in the EOF when it is activated.
   IF catering services are not available during emergencies/severe weather conditions
   THEN **DISTRIBUTE** as needed, emergency food packets which are stored in the food bin lockers located in

Salem: Food storage bins are located in the TSC kitchen area	**Hope Creek**: Food storage bins are located inside the TSC entry door

   ______ ADMSS

6. **OBTAIN** additional administrative supplies as needed.

   ______ ADMSS

7. **COORDINATE** with the Materials Supervisor in the EOF for accessing vendor lists and for providing any needed support for the station emergency response team. The EP Emergency Phone Directory, commercial phone books, and internet are all good information sources.

   ______ ADMSS

EVENT TERMINATION/CLOSEOUT

1. **RESTORE** the Facility to its original state in. Refer to NC.EP-DG.ZZ-0002(Z) Maintenance of Emergency Response Facilities. ____ ADMSS
2. **FORWARD** all completed forms to the Emergency Preparedness Manager . **ATTACH** any referenced and completed EPEPs and appendices. ____ ADMSS

ATTACHMENT 2
Page 1 of 2
EMERGENCY PREPAREDNESS ADVISOR (EPA) CHECKLIST

Prior to Activation of Facility:

Initials

1. **VERIFY** <u>or</u> have TSC Communicators **VERIFY** communication capabilities with other Emergency Response Facilities using NETS, ESSX, and DID systems.

   ______ EPA

2. **VERIFY** <u>or</u> have TSC Communicators **VERIFY** operability of telecopiers (may **COORDINATE** with Admin Support Team).

   ______ EPA

3. **OBTAIN** anticipated ECG Communicator Attachments located in work file drawer.

   ______ EPA

4. **OBTAIN** a turnover briefing from Control Room Communicators to include the following information:

   - Current ECG attachment being implemented.
   - Status of notifications being made.
   - Due time for next station Status Checklist (SSCL). SSCLs are due every 30 minutes.
   - Request copy of Initial Contact Message Form (ICMF), if necessary.
   - Note any special communication problems, number changes, or contacts.
   - Further note any relevant operational status.
   - Give Communicator your name, phone extension and advise that you will notify him when your facility is activated to assume Communicator duties.
   - Status of the NRC Data Sheet.
   - Status of NRC notification.

   ______ EPA

5. **DESIGNATE** Communicator 1 for voice notification duties and Communicator 2 for form preparation/ coordination and status board maintenance. <u>IF</u> a third Communicator is available, **ASSIGN** to NRC-ENS telephone duty.

   ______ EPA

ATTACHMENT 2
Page 2 of 2
EMERGENCY PREPAREDNESS ADVISOR (EPA) CHECKLIST

Post Activation of Facility:

Initials

1.	**ASSUME** all emergency communications functions <u>**when**</u> Facility is declared activated.	_______ EPA
2.	**IMPLEMENT** Event classification Guide Attachments <u>**only**</u> as directed by the Emergency Duty Officer (EDO).	_______ EPA
3.	**MONITOR** all Telecopier activity including transmissions and malfunctions. Ensure all data forms are initialed or signed off.	_______ EPA
4.	**ESTABLISH** timetable for telecopier transmissions of critical data forms ensuring deadlines are met.	_______ EPA
5.	**ENSURE** all notifications and procedural requirements are completed accurately and on time.	_______ EPA
6.	**MONITOR** status boards ensuring accuracy and timely updates.	_______ EPA
7.	**BRIEF** the EDO on communication status and carry out EDO requests.	_______ EPA
8.	**REVIEW** Communicator ECG attachments ensuring completeness and accuracy.	_______ EPA

Event Termination/Closeout:

1.	**ASSIST** in restoration of facility to its original state.	_______ EPA
2.	**FORWARD** all completed EPEPs, forms, and attachments to the EP Manager.	_______ EPA

ATTACHMENT 3
Page 1 of 1
EMERGENCY RESPONSE TEAM – TELECOPY LOG

STATION/UNIT: ________________ PAGE ____ OF ____

DATE: ________________________

No.*	Time	Rec'd (R) Sent (S)	Subject (Refer to Legend)	Initials

❖ Assign a sequential number to all documents except for test transmittals.

LEGEND:

ARR = Activity Report Roster
ERFR = Integrated ERF Roster
MEES = Major Equip. & Elec Status
OPS = Operational Status Board
RADS = Radiological Assessment Data Sheet
SRS = Shift Relief Schedule (2 pgs.)
DADS = Dose Assessment Data Sheet
ICMF = Initial Contact Message Form
NRCD = NRC Data Sheet (2 pgs.)
PDL = Plant Display Locations (2 pgs.)
RE = Reduction in Event
SSCL = Station Status Checklist (2 pgs.)

ATTACHMENT 4
Page 1 of 4
ONSITE SHIFT RELIEF SCHEDULE MANNING CHART

Date: ________________

Technical Support Center (TSC)

POSITION	TITLE	SHIFT #1 (__ : __ to __ : __) (Current shift)	SHIFT #2 (__ : __ to __ : __) (Relief shift)
A-03	**EDO**		
E-01	**RAC**		
E-02	RP SUPERVISOR OFFSITE		
E-03	RP TECH		
F-01	**TSS**		
F-02	TSTL		
F-05	CONTROLS ENG		
F-06A	CORE THERMAL ENGR.		
F-03	ELECTRICAL ENGR		
F-04	MECHANICAL ENGR		
F-07	**EPA**		
F-08	CM1		
F-08	CM2		
F08B	OPERATIONS ADVISOR		
I-01	SECURITY LIAISON		
J-03	**ADMIN SUPPORT SUPV**		
J-04	ADMIN STAFF		
J-04	ADMIN STAFF		
J-04	ADMIN STAFF		
J-04a	ADMIN STAFF (TDR CLERK)		

ATTACHMENT 4
Page 2 of 4
ONSITE SHIFT RELIEF SCHEDULE MANNING CHART

Date: ________________

Operations Support Center (OSC)

POSITION	TITLE	SHIFT #1 (__ : __ to __ : __) (Current shift)	SHIFT #2 (__ : __ to __ : __) (Relief shift)
C-01	**OSC COORDINATOR**	________________	________________
C-10	OSC CLERK	________________	________________
C-04A	**SHIFT SUP SUPERVISOR**	________________	________________
C-05A	RAD WASTE OPERATOR	________________	________________
B-05	EO	________________	________________
B-05	EO	________________	________________
B-05	EO	________________	________________
C-04B	**MECH SUPV**	________________	________________
C-04C	**CONTROLS SUPV**	________________	________________
C-05E	I&C TECH	________________	________________
C-05E	I&C TECH	________________	________________
C-05D	ELEC TECH	________________	________________
C-05D	ELEC TECH	________________	________________
C-05B	MACHINIST	________________	________________
C-08	PLANNER	________________	________________
E-02	**RP SUPV EXP CONT**	________________	________________

ATTACHMENT 4
Page 3 of 4
ONSITE SHIFT RELIEF SCHEDULE MANNING CHART

Date: ________________

Operations Support Center (OSC) - Continued

POSITION	TITLE	SHIFT #1 (_ : _ to _ : _) (Current shift)	SHIFT #2 (_ : _ to _ : _) (Relief shift)
C-06	**SITE PROT SUPERVISOR**	________________	________________
C-06A	FIRE BRIGADE	________________	________________
C-06A	FIRE BRIGADE	________________	________________
C-06A	FIRE BRIGADE	________________	________________
C-06A	FIRE BRIGADE	________________	________________
C-06A	FIRE BRIGADE	________________	________________

Date: ________________

Control Room Staff

POSITION	TITLE	SHIFT #1 (_ : _ to _ : _) (Current shift)	SHIFT #2 (_ : _ to _ : _) (Relief shift)
A-04	**OS**	________________	________________
B-04	CR CM1	________________	________________
B-04	CR CM2	________________	________________
B-02	**CRS**	________________	________________
B-01	**NSTA**	________________	________________
B-03	NCO	________________	________________
B-03	NCO	________________	________________

ATTACHMENT 4
Page 4 of 4
ONSITE SHIFT RELIEF SCHEDULE MANNING CHART

Date: ________________

Control Point Staff (CP)

POSITION	TITLE	SHIFT #1 (__ : __ to __ : __) (Current shift)	SHIFT #2 (__ : __ to __ : __) (Relief shift)
E-02	**RP SUPERVISOR IN PLANT**	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-03/04	RP TECH	________	________
E-05	**CHEM SUPERVISOR**	________	________
E-06	CHEM TECH	________	________
E-06	CHEM TECH	________	________

EXTRAS

Position	Title	Shift #1	Shift #2
______	________________	________	________
______	________________	________	________
______	________________	________	________
______	________________	________	________
______	________________	________	________

ATTACHMENT 5
Page 1 of 1
EDO CLERK GUIDELINES

EDO CLERK - CHECKLIST

The following guidelines provide suggested tasks that the EDO Clerk should perform if agreed upon with the EDO.

GUIDELINES:

- **MAINTAIN** EDO logbook. **ENTER** key information as announced or as directed by the EDO. Ensure the EDO reviews the log periodically.
- **TRACK** EDO facility briefings and **PROVIDE** the EDO with a 5-minute warning before the next scheduled briefing (usually every 30 - 45 minutes).
- **MAINTAIN** the facility activation section of the Emergency Status Information Board.
- **ANSWER** unattended phones of the EDO, TSS, or RAC. **TAKE** message or **INTERRUPT** personnel if the caller says the information is urgent.
- **UPDATE** OSC priority board every 30 minutes by **CONTACTING** the OSC Clerk and **ENSURING** that the TSC board reflects the OSC priority board. **INFORM** EDO when priority board is updated.
- **OBTAIN** procedures for the EDO/TSS/RAC as requested from the working file cabinet.
- **ENSURE** EDO/TSS/RAC have administrative supplies.
- **ADJUST** volume of plant page desk speaker (Salem only) to maximum.

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

USE CATEGORY: II

REVISION SUMMARY Biennial Review Performed: Yes X No ____

- Removed all references to Nuclear Fuels Engineer (NFE) and ensured all NFE duties were being performed by the Core Thermal – Hydraulic Engineer.
- Revised Manager – EP to EP Manager.

IMPLEMENTATION REQUIREMENTS

Effective Date: 2/6/02

APPROVED: ______________________ 1-29-02
EP Manager Date

APPROVED: ______________________ 1-29-02
Vice President - Operations Date

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

TABLE OF CONTENTS

ATTACHMENTS

1.0 PURPOSE

This procedure provides guidance for core damage assessment after an ALERT or higher level of emergency has been declared with the reactor shut down.

2.0 PREREQUISITES

2.1 Prerequisites To Be Followed Prior To Implementing This Procedure

Implement this procedure at:

- The discretion of Core Thermal Engineer (CTE)
- Upon staffing of your Emergency Response Facility.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 It is recommended that initials be used in the place keeping sign-offs, instead of checkmarks, if more than one person may implement this procedure.

3.1.2 Personnel who implement this procedure shall be trained and qualified IAW the Emergency Plan.

3.1.3 If additional support is needed for performing Fuel Damage Assessment, contact the Nuclear Fuels Manager.

3.2 Limitations

3.2.1 The core damage assessment methodology does not account for fission product spiking.

3.2.2 The core damage assessment methodology assumes reactor coolant cleanup systems are isolated.

3.2.3 Measurement of Cs-137 and Kr-85 activities may not be possible until shorter lived isotopes have decayed.

3.2.4 Clad damage of less than 1% is not considered to be a loss of the fuel cladding boundary.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 PROCEDURE

NOTE:

Due to the multiple and, at times, unpredictable failure mechanism associated with core damage this procedure has been developed to provide GUIDANCE for Core Damage Assessment. The sequence and extent of procedure performance should be based on the knowledge and experience of the Core Thermal Engineer.

5.1 Core Thermal Engineer Should Perform the Following to Initiate Core Damage Assessment(CDA) and CDA Sample Results:

5.1.1 PERFORM HCGS plant-specific calculations and estimations of the types and extent of reactor fuel damage utilizing the guidance of this procedure. [CD-385Y] [CD-548X] ______

5.1.2 PROVIDE recommendations to the Radiological Assessment Coordinator (RAC) to initiate post accident radionuclide samples and review all requests for post accident radionuclide samples for the purposes of core damage assessment. [CD-443D] ______

5.1.3 OBTAIN the post accident radionuclide samples with consideration as to how representative the sample will be of the core condition. ______

5.1.4 DETERMINE need and frequency for post accident radio nuclide samples shall be determined with consideration of the application of the assessment – Accident Classification, Accident Mitigation, Source Term Assessment, Core Damage Assessment Only. ______

5.1.5 Recommend to the RAC post accident sampling system sample points based upon reactor condition or event type. ______

A. SELECTION of Liquid Sample Point

NOTE

Residual Heat Removal (RHR) samples: If RHR is in the Low Pressure Coolant Injection (LPCI) or Suppression Pool Cooling modes, it should be operating an estimated 30 minutes minimum prior to sampling to ensure a representative sample. [CD-384Y]

PASS Reactor Jet Pump samples: At power levels < 1% of rated power, a representative sample may be obtained by raising reactor water level 18 inches to fully flood the moisture separators thus providing a thermally induced recirculation flow path for mixing.

LIQUID SAMPLE

LOCATION	TYPE	MODE
Reactor (Jet Pump)	PASS	Reactor Pressurized
Reactor (RHR-LPCI/Shutdown Cooling)	PASS	Reactor Depressurized
Suppression Pool (RHR) *	PASS	RHR in Suppression Pool Cooling *

* [CD-384Y]

GAS SAMPLE

EVENT	SAMPLE LOCATION
Non – Break	Torus Atmosphere
Small Break	Prior to Depressurization: Drywell Atmosphere After Depressurization: Torus Atmosphere
Large Break (in Drywell)	Drywell Atmosphere
Large Break (outside Drywell)	Torus Atmosphere

B. RECORD on Attachment 1 the current time, selected sample point, the desired frequency of sampling and the basis for the selection and frequency. ______

C. PROVIDE a copy of Attachment 1 to the RAC and the Technical Support Team Leader (TSTL). ______

5.1.6 ESTIMATE the type and extent of core damage based on the Drywell Atmosphere Post Accident (DAPA) Radiation Monitor Reading ______

5.1.7 IF the Drywell Atmosphere Post Accident Monitor has been declared inoperable by Operations, THEN GO TO step 5.2. ______

5.1.8 OBTAIN and record on Attachment 2, DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) MONITOR A AND B READING (R/HR), the time of the reading and the time of reactor shutdown. ______

NOTE

DAPA monitor A and B provide indication for two different locations in the Drywell.

If adverse conditions exist in the Drywell (average Drywell air temperature greater than or equal to 245°F) validate with the Radiological Assessment Coordinator that EPIP 302H, Attachment 5, DAPA CORRECTION CALCULATIONS has been utilized.

5.1.9 DETERMINE the percent of fuel inventory airborne in using Attachment 2. Record the result on Attachment 17. ______

5.2 Estimating the Type and Extent of Core Damage Based on the Drywell Atmosphere Post Accident (DAPA) Equivalent Calculation

5.2.1 GO TO step 5.1.5 if the DAPA monitor is operable.

5.2.2 INFORM the TSTL of the need to determine drywell atmosphere radiation levels without the DAPA monitor. ______

5.2.3 REQUEST from the Radiological Assessment Coordinator a "Contact Dose Rate" at the Drywell Personnel Airlock and a Particulate, Iodine, and Noble Gas air Sample of the 120' El. Rx. Bldg. For the purposes of determining a DAPA EQUIVALENT READING. Calculate a "DAPA EQUIVALENT" value and document the value on Attachment 2 as a DAPA EQUIVALENT in the following manner: ______

EQUIV = 100 x (CDR – (20 x (NUGC))

WHERE: EQUIV = DAPA Equivalent (R/HR) for use in Attachment 2 (if CDR = normal bkg then CDR = 0)
CDR = Contact Dose Rate (R/HR)
NGC = Nobel Gas Concentration (μCi/cc) (if NGC is < 1E-04 (μCi/cc then NGC = 0)

5.2.4 DETERMINE the percent of fuel inventory airborne by using Attachment 2. Record the result on Attachment 17. ______

5.3 **Determining the Percent of Metal-Water Reaction from the Hydrogen Concentration in the Primary Containment Free Volume.**

5.3.1 OBTAIN the hydrogen concentration in the primary containment from the Hydrogen-Oxygen Analyzer System or from the post accident sampling system and record it on Attachment 3. ______

5.3.2 RECORD the time of the reading or sample and the sample point on Attachment 3. ______

5.3.3 RECORD on Attachment 3 any drywell venting or hydrogen recombiner operation. ______

5.3.4 DETERMINE the percent metal-water reaction by using Attachment 3. Record the result on Attachment 17. ______

5.4 **Estimating the Type and Percent of Core Damage From Fission Product Concentrations.**

5.4.1 IF a liquid sample has been selected as identified on Attachment 1, obtain from the Chemistry Supervisor in the TSC the concentration of I-131, I-132, I-133, I-134, I-135, Cs-134, Cs-137, sample point, sampling time, sample analysis time, type of decay correction performed and the time of final reactor shutdown. Record the information on Attachment 4. ______

5.4.2 IF a gas sample has been selected as identified on Attachment 1, obtain from the Chemistry Supervisor in the TSC the concentration of Kr-85m, Kr-85, Kr-87, Kr-88, Xe-133, Xe-135, sample point, sampling time, sample analysis time, type of decay correction performed, and the time of final reactor shutdown. Record the information on Attachment 4. ______

5.4.3 CALCULATE the pressure/temperature corrected fission product concentrations for gas sample radioisotopes as per Attachment 4. ______

NOTE

Pressure/temperature corrections will not be necessary if the corrections have been performed by the Chemistry Department.

5.4.4 CALCULATE the decay corrected fission product concentrations as per Attachment 4 and record the results on Attachment 4. ______

NOTE

Decay corrections will not be necessary if performed by the Chemistry Department.

5.4.5 CALCULATE the fission product inventory correction factors (F_i) as per Attachment 5. ______

5.4.6 CALCULATE the normalized concentrations of the fission products (C_{wn}) as per Attachment 6. ______

5.5 **Utilizing the Normalized Concentrations**

5.5.1 Following the instructions on Attachment 6 and Attachments 7 through 15 estimate the percent cladding failure and percent fuel melting. ______

5.5.2 Record the results on Attachment 17. ______

NOTE

The lines on the graphs are set up in the following manner:

Upper Dashed Line – maximum fission product release for a given fuel condition.
Lower Dashed Line – minimum fission product release for a given fuel condition.
Center Solid Line – nominal fission product release for a given fuel condition.

5.6 **Estimating Release Source (Gap or Fuel Pellet) From the Isotopic Ratios.**

5.6.1 CALCULATE the isotopic ratios as per Attachment 16. ______

5.6.2 COMPARE the calculated isotopic ratios to the values listed in the table on Attachment 16 to estimate the release source. Record the results on Attachment 17. ______

5.7 **Determine If Less Volatile Fission Products are Present in the Reactor Coolant.**

5.7.1 IF the less volatile fission products, such as Sr, Ba, La, or Ru (either soluble or insoluble), are found to have unusually high concentrations in the reactor coolant some degree of fuel melting may be inferred. _______

5.7.2 RECORD observations of less volatile fission products on Attachment 17. _______

5.8 **Estimating If an Interruption of Adequate Core Cooling Has Occurred.**

5.8.1 OBTAIN a history of the reactor vessel water level from the initiation of the accident from SPDS or the VAX LA120. _______

5.8.2 DETERMINE if the top of active fuel (TAF) has been uncovered. _______

5.8.3 RECORD the level history, duration of level below the TAF and an estimate of cooling adequacy on Attachment 17. _______

NOTE

Significant or core-wide damage is not expected unless the TAF has been uncovered. Core-wide clad damage can occur within 30 minutes of uncovering the fuel. However, unless level is below the bottom of the active fuel, boiling heat transfer will provide cooling and significantly extend the duration that a partial uncovering can be withstood without significant core damage.

5.9 **Performing an Assessment of the Type and Extent of Core Damage Based Upon All Available Indicators**

5.9.1 CLASSIFY the type and extent of core damage relative to the following matrix. _______

Degree of Core Damage	Minor (<10%)	Intermediate (10% - 50%)	Major (>50%)
None (<1% clad)	1	1	1
Clad Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

NOTE

The primary indicator for the damage assessment is the estimate of damage based on the post accident radionuclide sample data as utilized in Section 5.5.

5.9.2 EVALUATE the other indicators or parameters to corroborate and further refine the assessment as determined in section 5.5. _______

5.9.3 REQUEST that the TSTL INITIATE appropriate confirmation of accuracy if conflicting indications are identified. _______

5.9.4 RECORD the assessment and bases on Attachment 17. _______

5.10 **Reporting the Results of the Assessment and Recommending Further Actions.**

5.10.1 REPORT the results to the TSTL for dissemination to the TSS and the RAC. _______

5.10.2 REVIEW the current accident status in order to make recommendations for further actions to refine or continue the assessment. _______

5.10.3 RE-ENTER the procedure as appropriate. _______

6.0 RECORDS

Return completed procedure, original copies to the EP Manager.

7.0 REFERENCES

7.1 References

7.1.1 General Electric Document, NEDO-22215 82NEDO90, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions, August 1982.

7.1.2 General Electric Document, C&RE Transmittal, RPE 81CL01, November 1981

7.1.3 PSEG Nuclear Radiation Protection/Chemistry Services File NRP-88-0048, Preplanned Alternate Monitoring Methods for the DAPA Monitoring System, March 3, 1988.

7.1.4 Hope Creek UFSAR 1.14.1.49.2

7.2 Cross References

7.2.1 EPIP NC.EP-EP.ZZ-0302(Q), Radiation Assessment Coordinator Response.

7.2.2 PSEG Nuclear Emergency Plan

7.3 **Closing Documents**

7.3.1 Hope Creek CD-443D

7.3.2 Hope Creek CD-384Y

7.3.3 Hope Creek CD-385Y

7.3.4 Hope Creek CD-548X.

ATTACHMENT 1
Page 1 of 1
POST ACCIDENT RADIONUCLIDE SAMPLE REQUEST

Sample Request No. ______________________________

Time of Request ______________________________

Sample Point ______________________________

Frequency ______________________________

Bases ______________________________

____

____

Comments ______________________________

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____

____

Sample Request No. ______________________________

Time of Request ______________________________

Sample Point ______________________________

Frequency ______________________________

Bases ______________________________

____

____

Comments ______________________________

____

____

____

ATTACHMENT 2
Page 1 of 2
DAPA MONITOR DOSE RATE TO FUEL INVENTORY AIRBORNE

1. Time of Reactor Shutdown ______________________ Date ____________

2. Time of DAPA reading: A ______________________ Date ____________

   B ______________________ Date ____________

3. Time after Shutdown, Hrs ______________________________________

4. Complete the following Table for each reading.

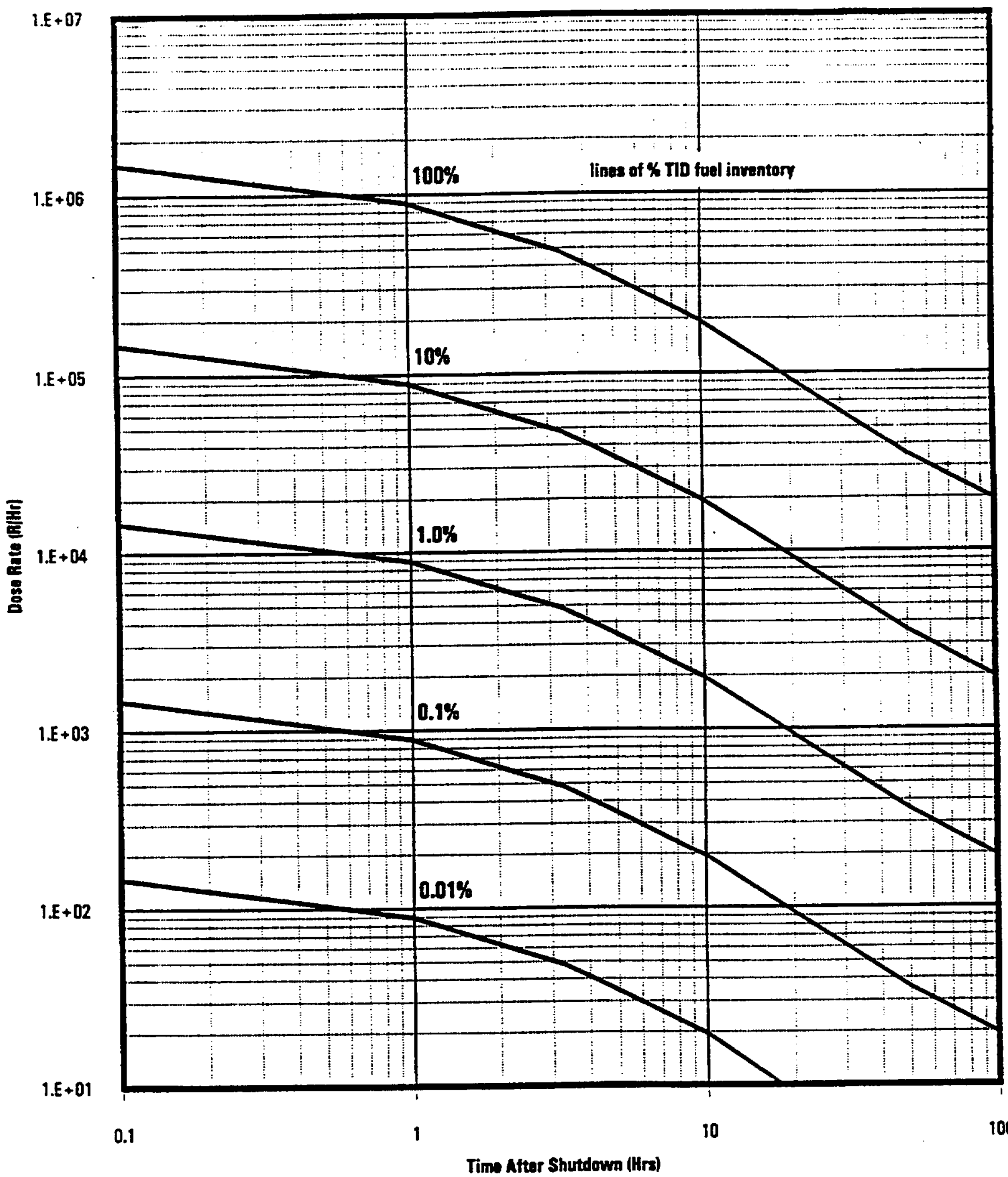
	DAPA EQUIV (Y/N)	Monitor (A/B)	Dose Rate (R/Hr)	Time after Shutdown (Hrs)	Fuel Inventory From Graph, Iinv (%)
1					
2					
3					
4					
5					
6					

5. Use the following as guidance for assessing the amount and type of core damage based on the DAPA readings.

Iinv	Approximate Source and Damage Estimate	
100	100%	TID*, 100% fuel damage, potential core melt
50	50%	TID noble gases, TMI source
10	10%	TID, 100% NRC gap activity, total clad failure/partial core uncovery
3	3%	TID, 100% WASH-1400 gap activity, major clad failure
1	1%	TID, 10% NRC gap, Max 10% clad failure
0.1	0.1%	TID, 1% NRC gap, 1% clad failure, local heating of 5-10 assemblies
0.01	0.01%	TID, 0.1% NRC gap, clad failure of ¾ fuel assembly

* TID = Technical Information Document #14844 source term
100% Noble Gas, 25% Halogens and 1% Solids

ATTACHMENT 2
Page 2 of 2

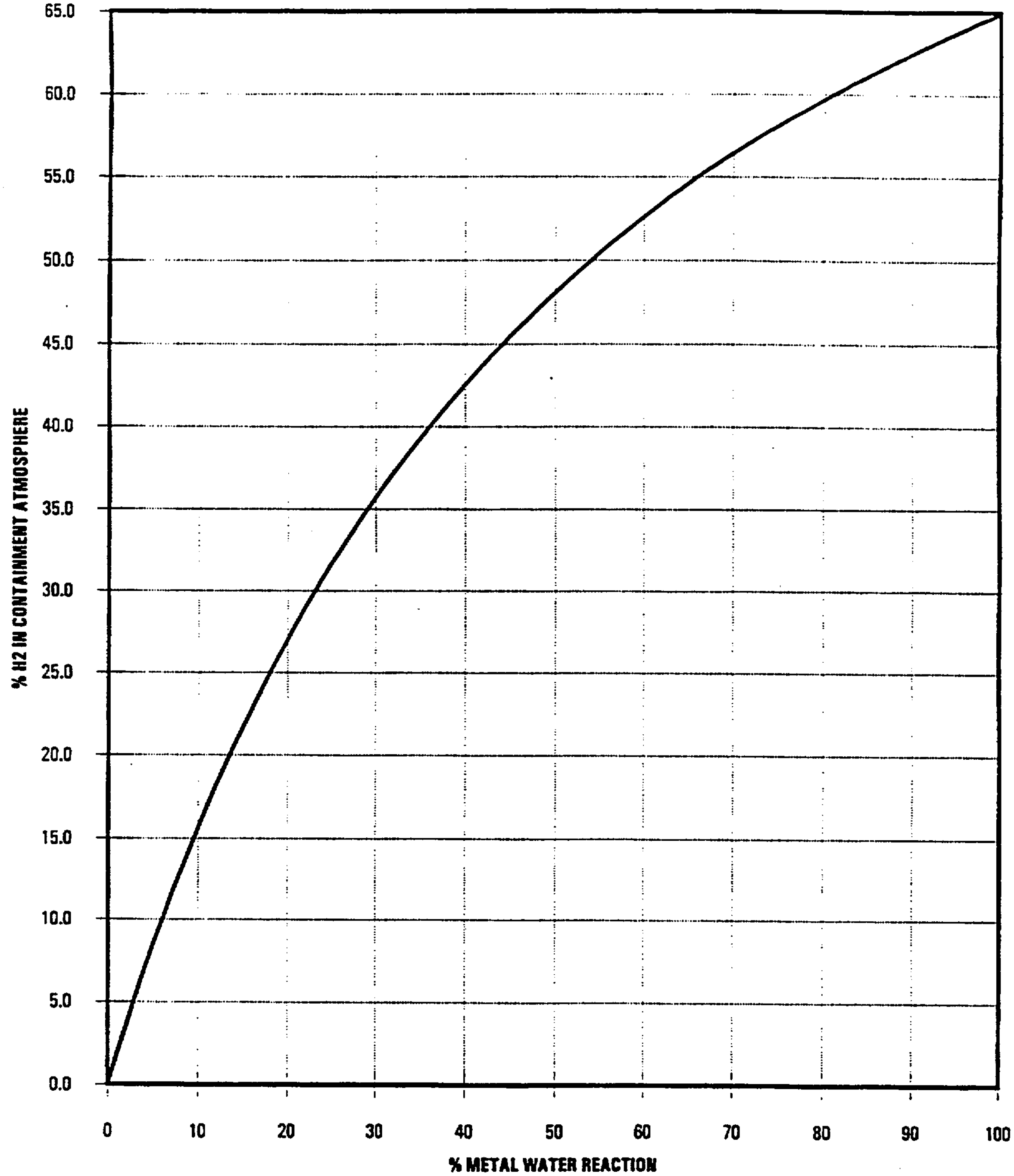


ATTACHMENT 3
Page 1 of 3
PRIMARY CONTAINMENT HYDROGEN CONCENTRATION TO % METAL – WATER REACTION

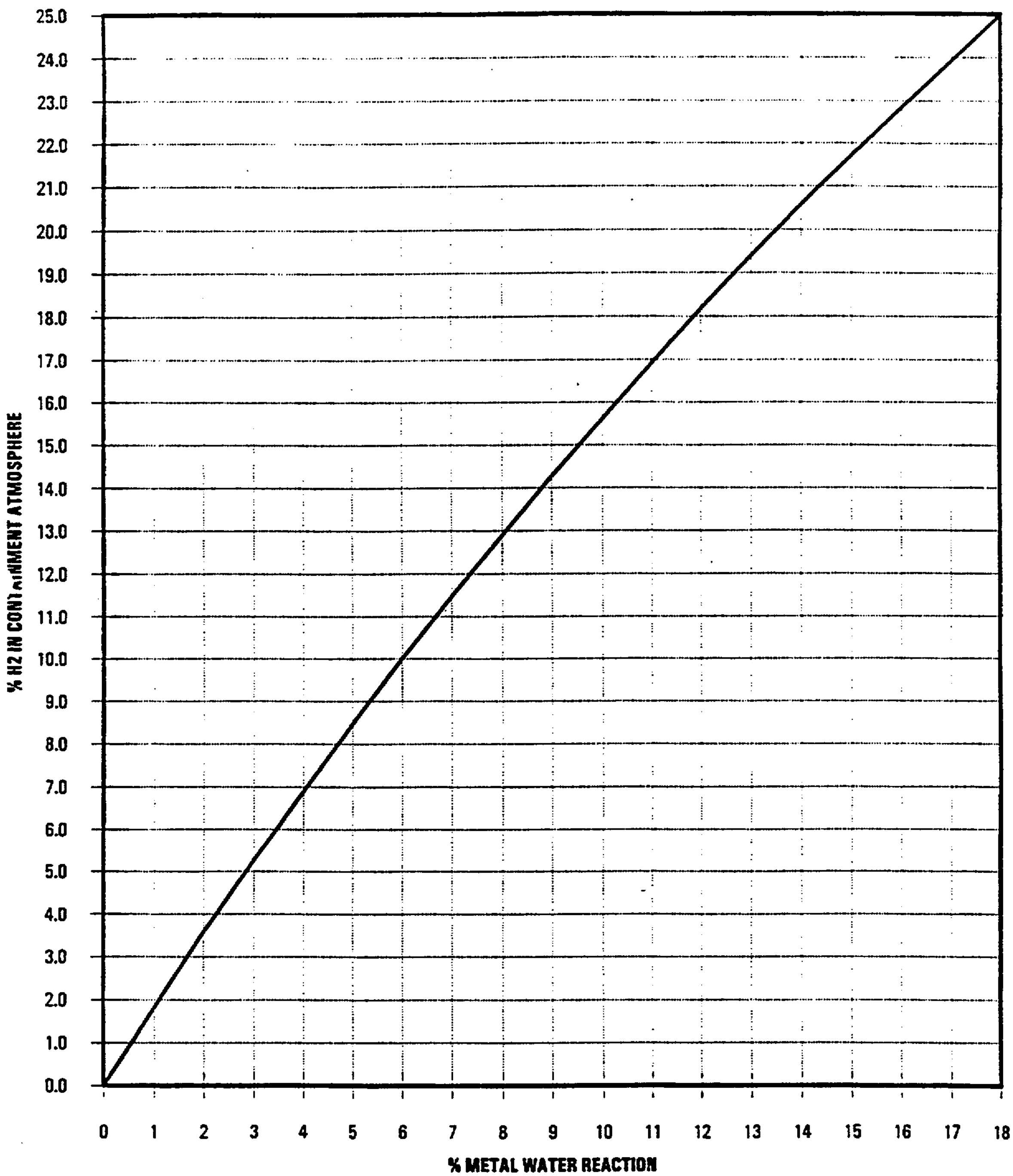
	Date	Time	System and Sample Point	H2 (%)	Metal-Water Reaction (%)
1					
2					
3					
4					
5					
6					

	Comments and Drywell Venting/Recombiner Operation Note
1	
2	
3	
4	
5	
6	

ATTACHMENT 3
Page 2 of 3



ATTACHMENT 3
Page 3 of 3



ATTACHMENT 4
Page 1 of 2
FISSION PRODUCT CONCENTRATIONS

Time of Reactor Trip or Shutdown ______________________

Sample No. ____________ Sample Time ______________

Sample Analysis Time ______________________________

Sample Type ___________ Sample Point ______________

Pressure/Temperature Correction ____

Sample Vial P1 ___________ psi T1 ____________ OK

Sample Point Environment P2 ___________ psi T2 ____________ OK

PTMULT = P2 T1/ P1 T2 = ______________ (PTMULT = 1.0 for liquid)

Decay Correction ____

DMULT = $e^{\lambda t}$

λ = decay constant of the isotope of interest (1/days)
t = time of decay (days)

NOTE

The time of decay must represent the elapsed time from reactor trip or shutdown to the sample analysis time.

NOTE

The decay correction must account for the activity decrease during the time period from reactor trip or shutdown to the sample analysis time.

ATTACHMENT 4
Page 2 of 2

	λ (1/day)	Sample Result (μCI/g)	P/T Corrected (μCi/g)	Decay Corrected (μCi/g)
I-131	8.621E-02			
I-132	7.23E+00			
I-133	7.998E-01			
I-134	I.898E+01			
I-135	2.517E+00			
Cs-134	9.219E-04			
Cs-137	6.294E-05			
Kr-85m	3.713E+00			
Kr-85	1.771E-04			
Kr-87	1.308E+01			
Kr-88	5.858E+00			
Xe-133	1.320E-01			
Xe-135	1.826E+00			

ATTACHMENT 5
Page 1 of 3
FISSION PRODUCT INVENTORY CORRECTION FACTORS

1. Calculate the inventory correction factor (Fi) for each fission product listed in steps 3 and 4 of Attachment 5 using the following:

   1.1 Bases:

   $$F_i = \frac{\text{reference inventory of isotope i in HCGS}}{\text{actual inventory of isotope i in HCGS}}$$

   1.2 If the total operating time for all batches is greater than or equal to the power correction time:

   $$F_i = \frac{3293\,(1-e^{-1095\lambda_i})}{\Sigma_j\,[P_j(1-e^{-\lambda_i T_j})e^{-\lambda_i T_0}]}$$

Where:			
	λ_i	=	decay constant of isotope i (1/days)
	Σ_j	=	sum of the batches
	P_j^*	=	steady reactor power (total core power) operated in period j (MWt)
	T_j^*	=	duration of operating period j (days)
	T_0	=	time between the end of operating period j and time of the final reactor shutdown (days)

* For each time period, T_j^* the variation of steady reactor power, P_j^*, should be limited to $\pm$ 20%.

1.3 If the total operating time for any batch is less than the power correction time:

$$F_1 = \frac{3293(1-e^{-1095\lambda i})}{\Sigma_k \Sigma_j [BP_j(1-e^{-\lambda iTj})e^{-\lambda iT0}]}$$

Where: λ_i = decay constant of isotope i (1/days)

BP_j^* = steady reactor power (total core power) multiplied by 1/3 to approximate batch power operated in period j (MWt)

T_j^* = duration of operating period j (days)

T_0 = time between the end of operating period j and time of the final reactor shutdown (days)

Σ_j = sum of the batches

Σ_k = the operation of calculating the denominator of the inventory correction for each batch and then summing the batch results prior to division

* For each time period, T_j^*, the variation of steady reactor power, P_j^*, should be limited to $\pm$ 20%.

2. Each fission product must be corrected for either 6 half-lives or 3 fuel cycles whichever is shorter. The times are delineated in steps 3 and 4 as the "Power Correction Time".

3. Liquid Sample

Fission Product	Power Correction Time	λ (1/day)	3293 * $(1-e^{-1095\lambda i})$	F_i
I-131	49 days	8.621E-2	3.293E3	
I-133	6 days	7.998E-1	3.293E3	
I-135	2 days	2.517E+0	3.293E3	
Cs-134	3 fuel cycles	9.219E-4	2.089E3	
Cs-137	3 fuel cycles	6.294E-5	2.192E2	

ATTACHMENT 5
Page 3 of 3

4. Gas Sample

Fission Product	Power Correction Time	λ (1/day)	3293 * $(1-e^{-1095\lambda i})$	F_i
Kr-85m	2 days	3.713E+0	3.293E3	
Kr-85	3 fuel cycles	1.771E-4	5.800E2	
Xe-133	35 days	1.320E-1	3.293E3	
Xe-135	3 days	1.826E+0	3.293E3	

ATTACHMENT 6
Page 1 of 3
NORMALIZED CONCENTRATION OF FISSION PRODUCTS

1. For each fission product in steps 2 and 3 of Attachment 6 perform the following calculation using the applicable data from Attachment 4 and Attachment 5.

   $C_w = C_t * F_i$

   Where: C_w = the normalized concentration of the fission product (uCi/g for liquids and uCi/cc for gases)

   C_t = the decay and pressure/temperature corrected fission product concentration from Attachment 4.

   F_i = the inventory correction factor from Attachment 5.

2. Liquid sample - Activity concentrations dispersed equally through reactor water and torus water

Fission Product	C_t	F_i	C_w
I-131			
I-133			
I-135			
Cs-134			
Cs-137			

3. Gas Sample - Activity concentrations dispersed equally through drywell and torus free volumes

Fission Product	C_t	F_i	C_w
Kr-85m			
Kr-85			
Xe-133			
Xe-135			

ATTACHMENT 6
Page 2 of 3

4. Additional normalizations may be required if plant parameters indicate that the specific activity from a liquid sample represent a sample environment different than the reference environment. Concentration or dilution corrections should be performed and documented in step 7. Reference and typical constants required for the corrections are delineated in step 6.

   Reference mass = the total mass of the reactor water and torus water

   If the actual mass of liquid water does not equal the reference mass a correction factor should be applied.

   $F_{d/c}$ = actual mass (g) / reference mass (g)

5. Additional normalizations may be required if plant parameters indicate that the specific activity from a gas sample represents a sample environment different than the reference environment. Concentrations of dilution corrections should be performed and documented in step 7. Reference and some typical constants required for the corrections are delineated in step 6.

   Reference volume = drywell plus torus free volume

   If the actual volume of gas does not equal the reference volume a correction factor should be applied.

   $F_{d/c}$ = actual mass (cc) / reference mass (cc)

6. Dilution/Concentration Data

Reference liquid mass	3.633E9 g (8.01E6 lbs)
Reactor liquid mass	
At Power	2.93E8 g (6.46E5 lbs)
Hot Standby	3.03E8 g (6.68E5 lbs)
Cold Shutdown	4.09E8 g (9.02E5 lbs)
Torus liquid mass	3.34E9 g (7.36E6 lbs)
Reference gas volume	8.57E9 cc (3.03E5 ft^3)
Torus free volume	3.78E9 cc (1.33E5 ft^3)
Drywell free volume	4.79E9 cc (1.69E5 ft^3)

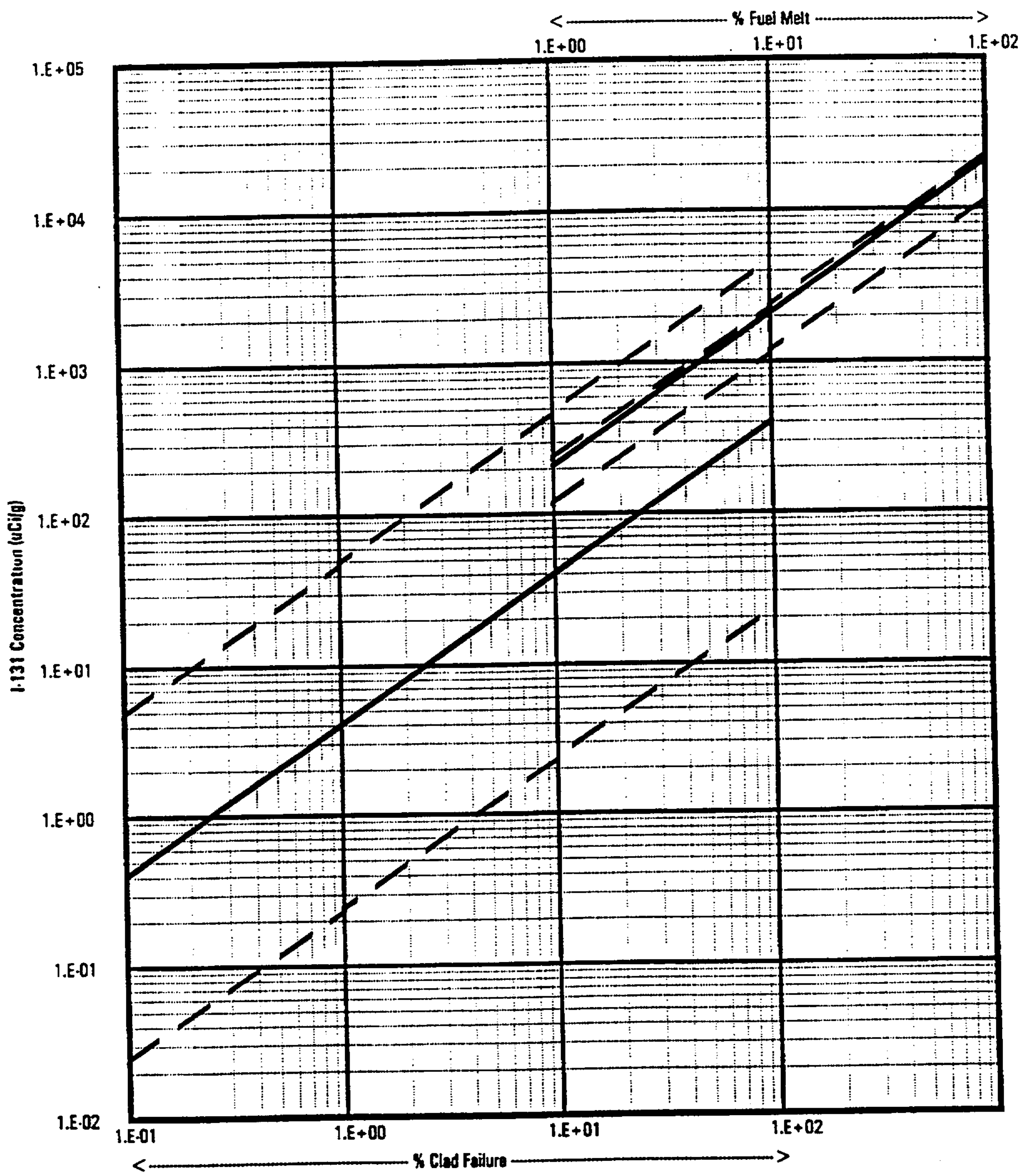
ATTACHMENT 6
Page 3 of 3

7. Additional Normalizations

Fission Product	C_w times	$F_{d/c}$ equals	C_{wn}
I-131			
I-133			
I-135			
Cs-134			
Cs-137			
Kr-85m			
Kr-85			
Xe-133			
Xe-135			

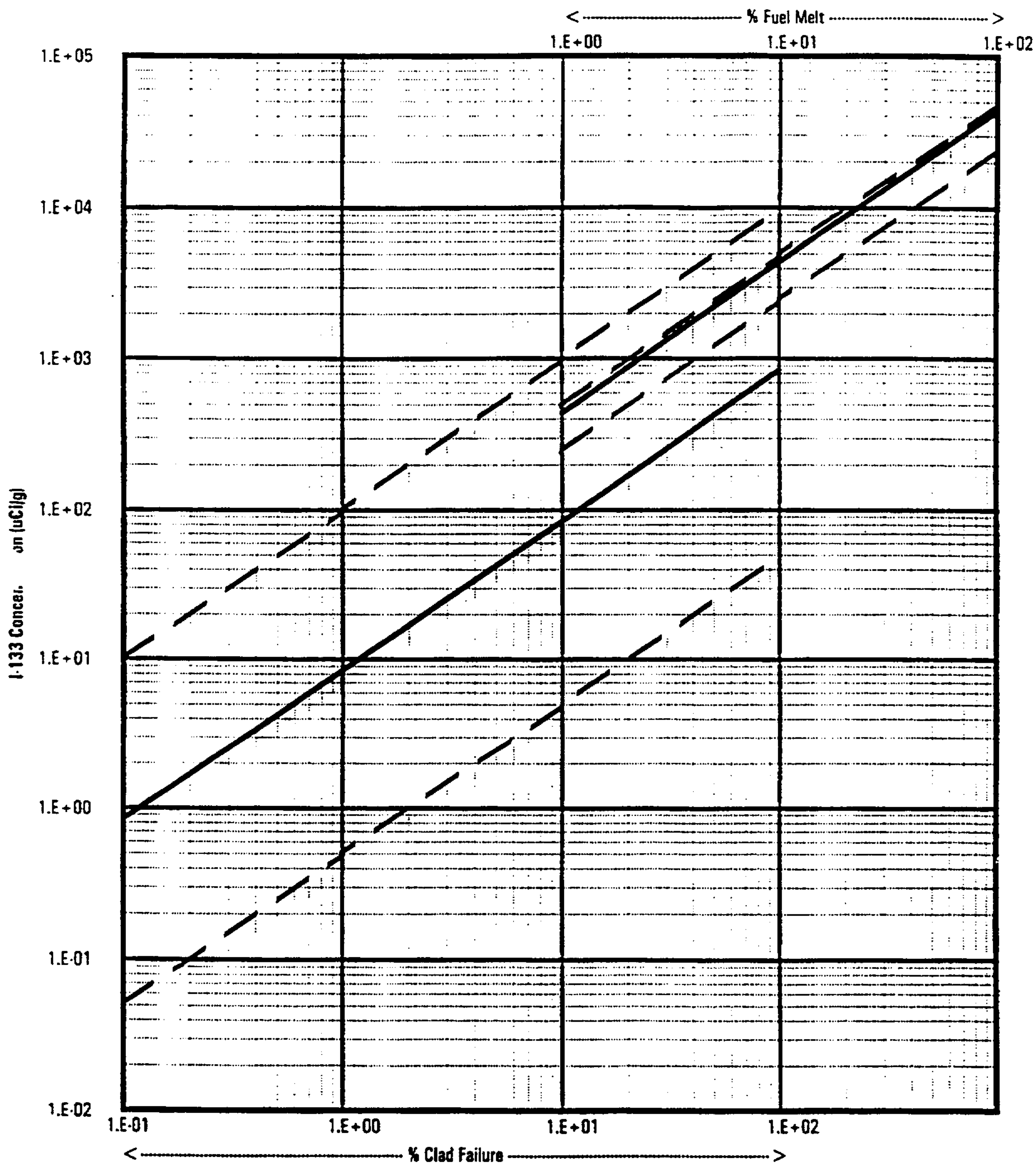
ATTACHMENT 7
Page 1 of 1
I-131 CONCENTRATION VS. INDICATION OF CORE DAMAGE

I-131 Concentration vs. Indication of Core Damage

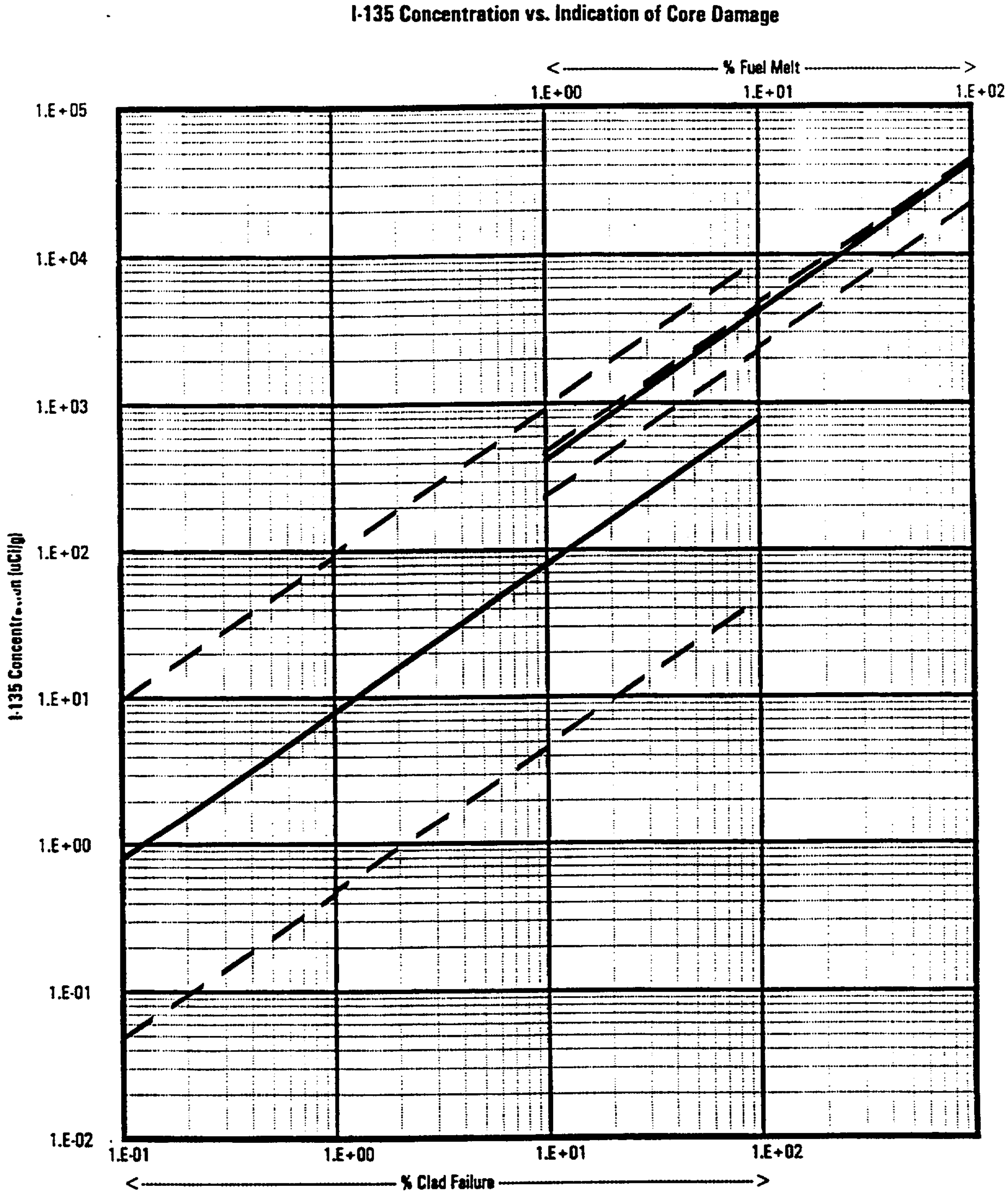


ATTACHMENT 8
Page 1 of 1
I-133 CONCENTRATION VS. INDICATION OF CORE DAMAGE

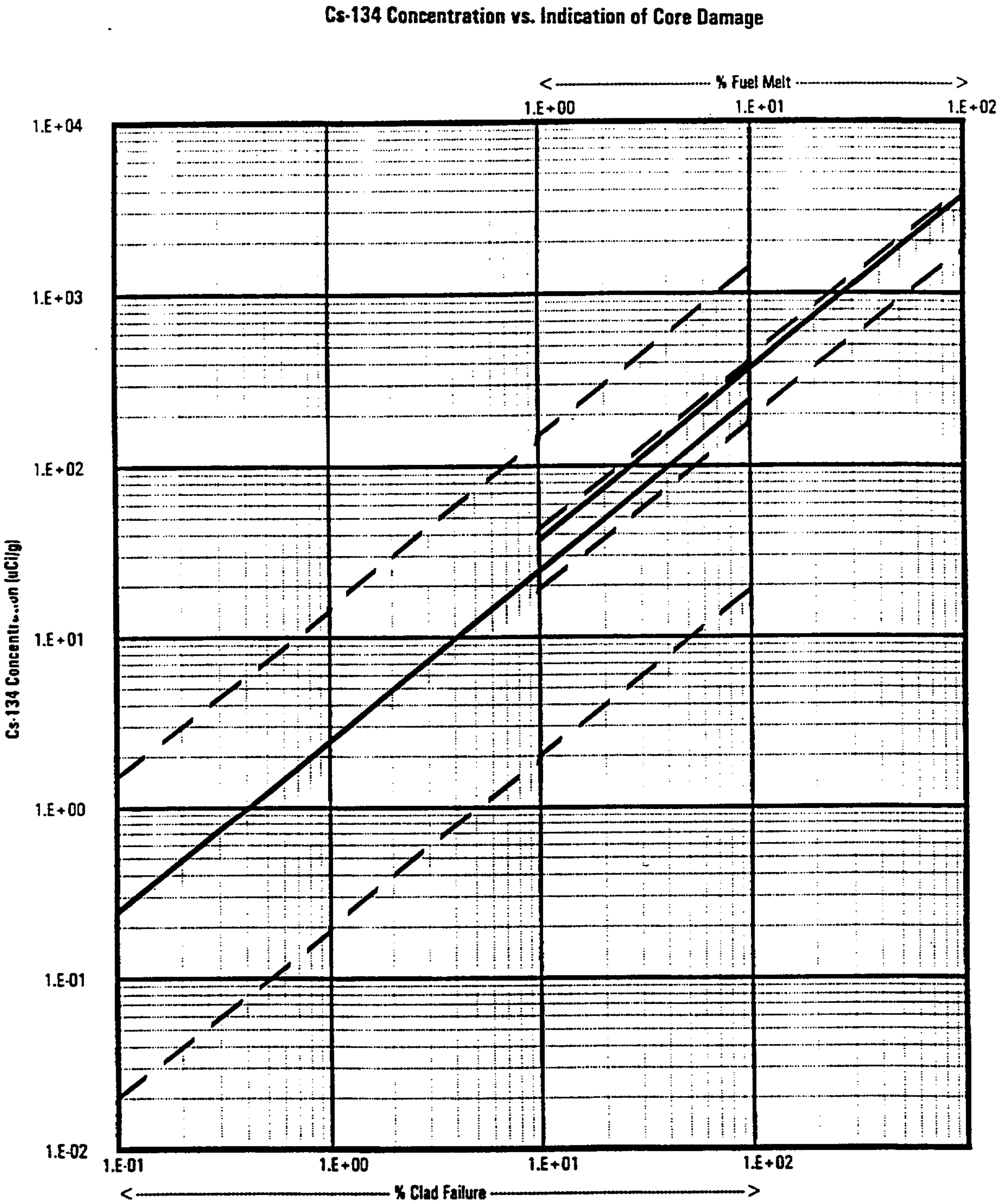
I-133 Concentration vs. Indication of Core Damage



ATTACHMENT 9
Page 1 of 1
I-135 CONCENTRATION VS. INDICATION OF CORE DAMAGE

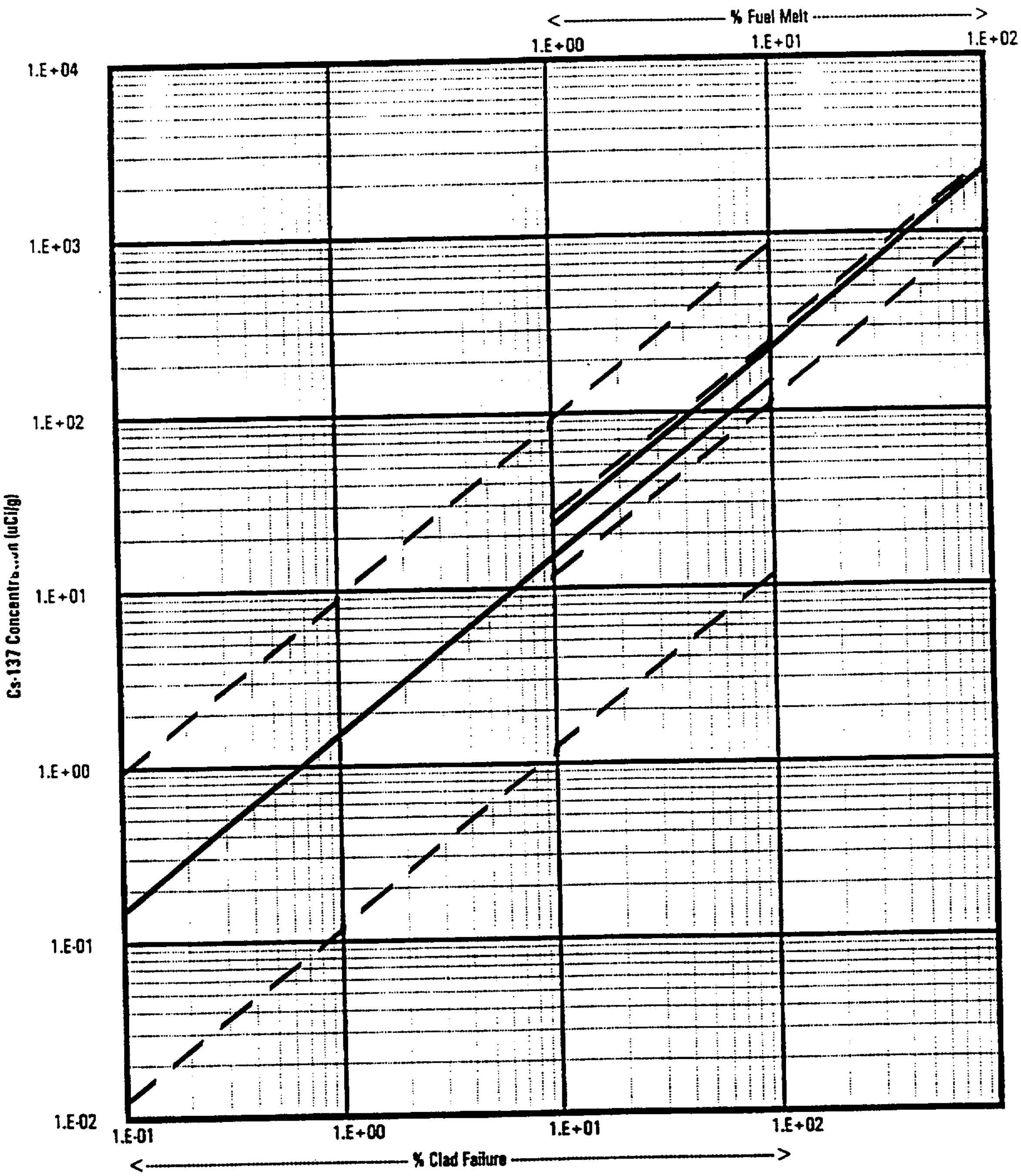


ATTACHMENT 10
Page 1 of 1
Cs-134 CONCENTRATION VS. INDICATION OF CORE DAMAGE



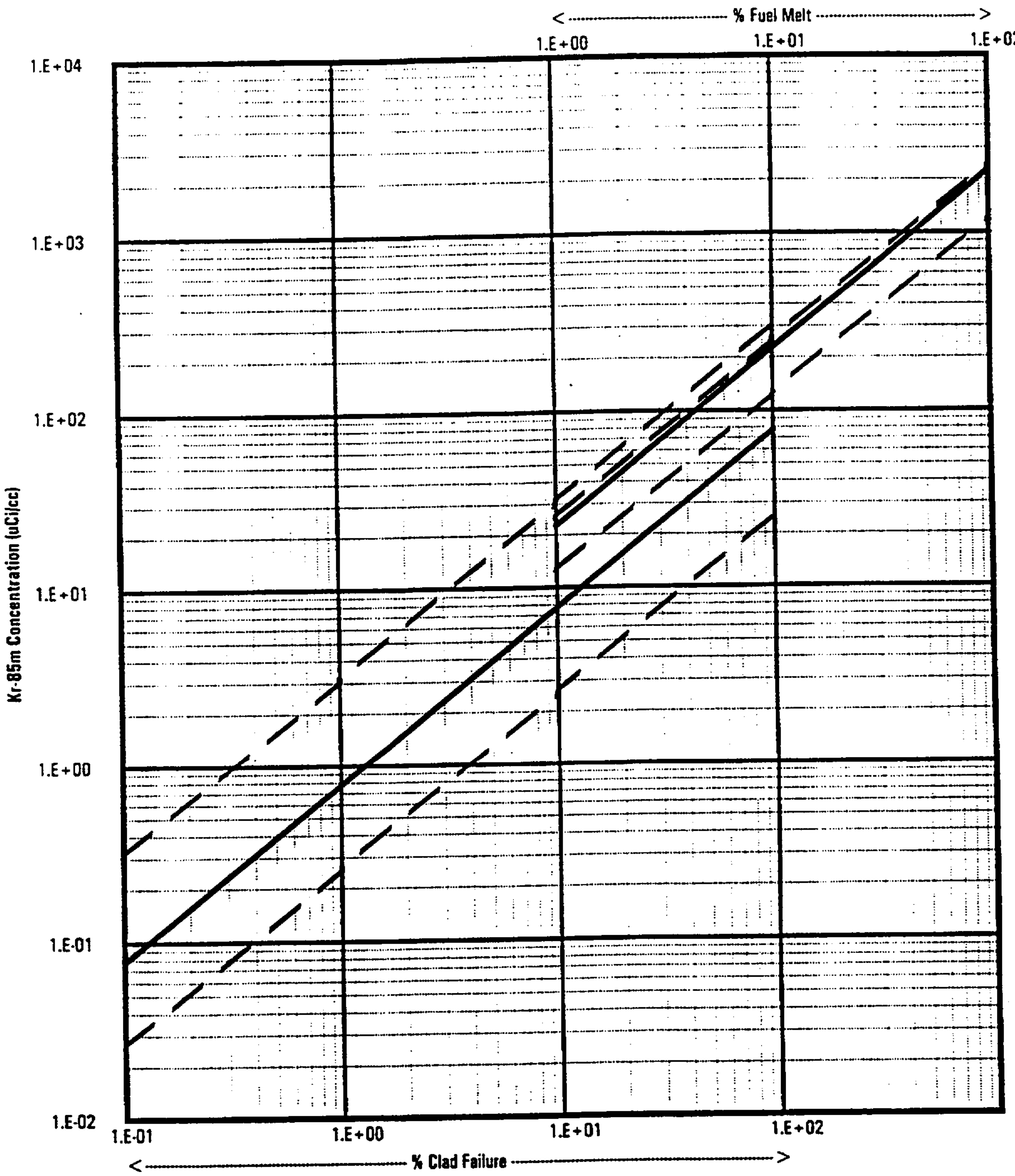
ATTACHMENT 11
Page 1 of 1
Cs-137 CONCENTRATION VS. INDICATION OF CORE DAMAGE

Cs-137 Concentration vs. Indication of Core Damage



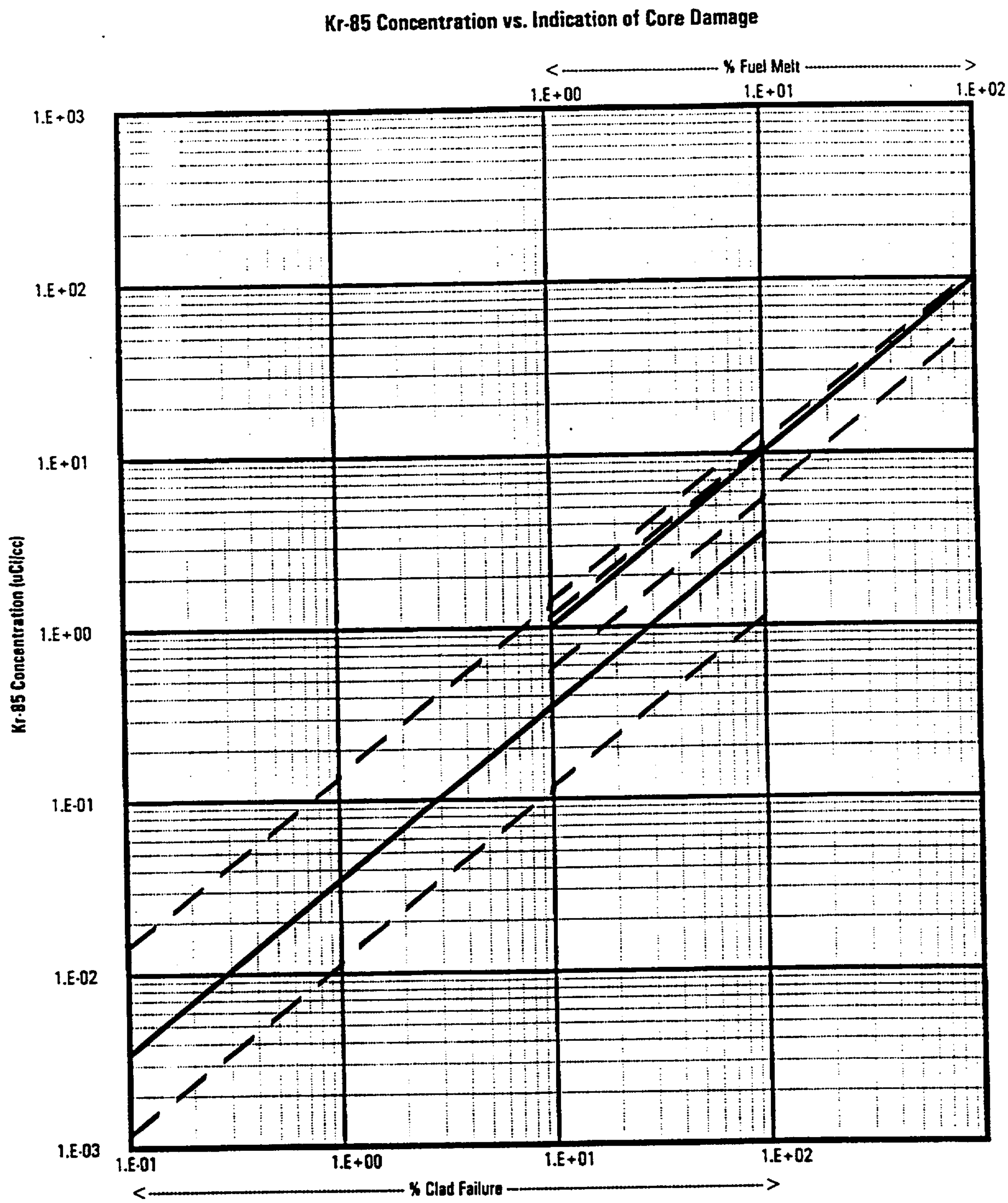
ATTACHMENT 12
Page 1 of 1
Kr-85m CONCENTRATION VS. INDICATION OF CORE DAMAGE

Kr-85m Concentration vs. Indication of Core Damage

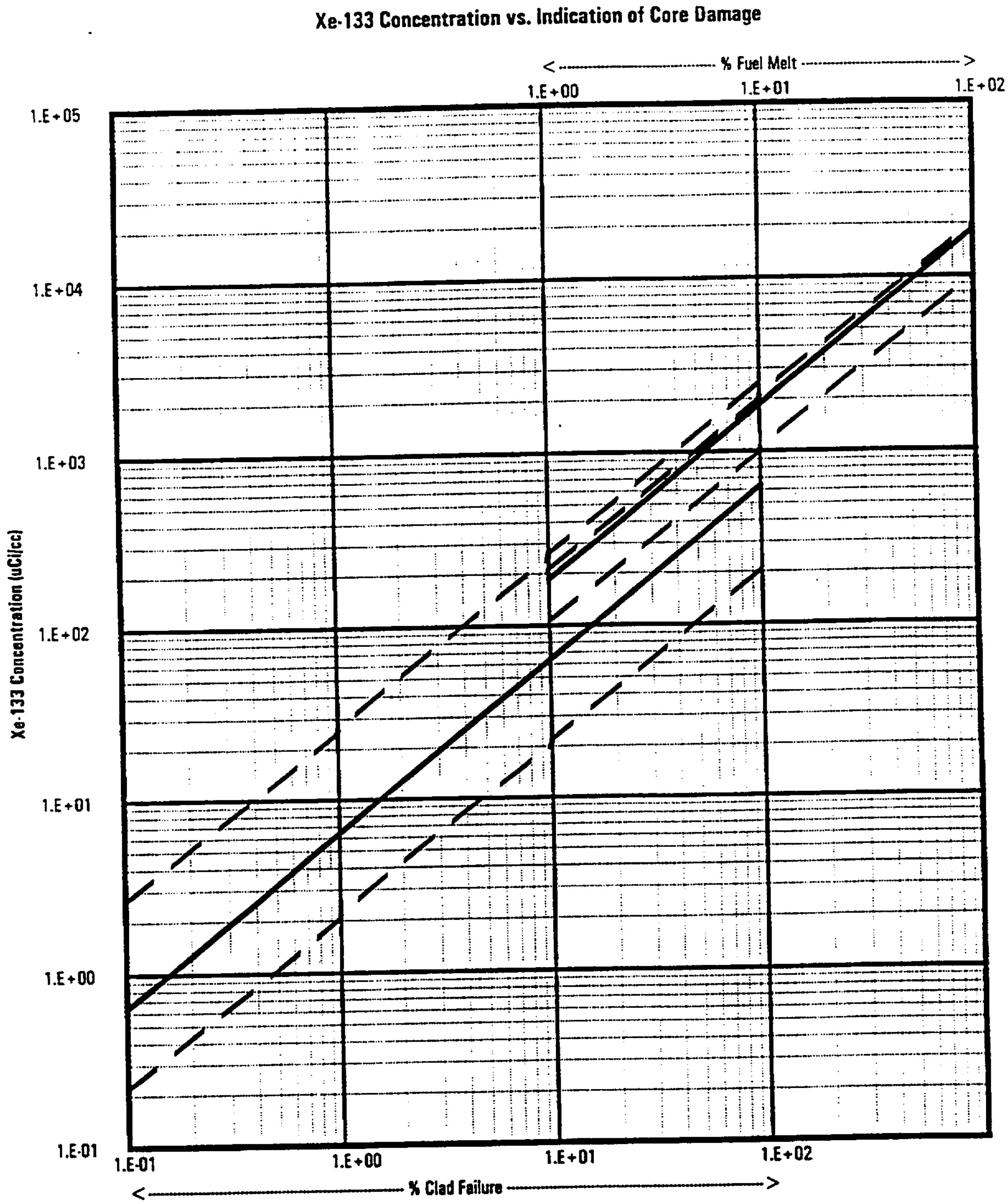


ATTACHMENT 13
Page 1 of 1

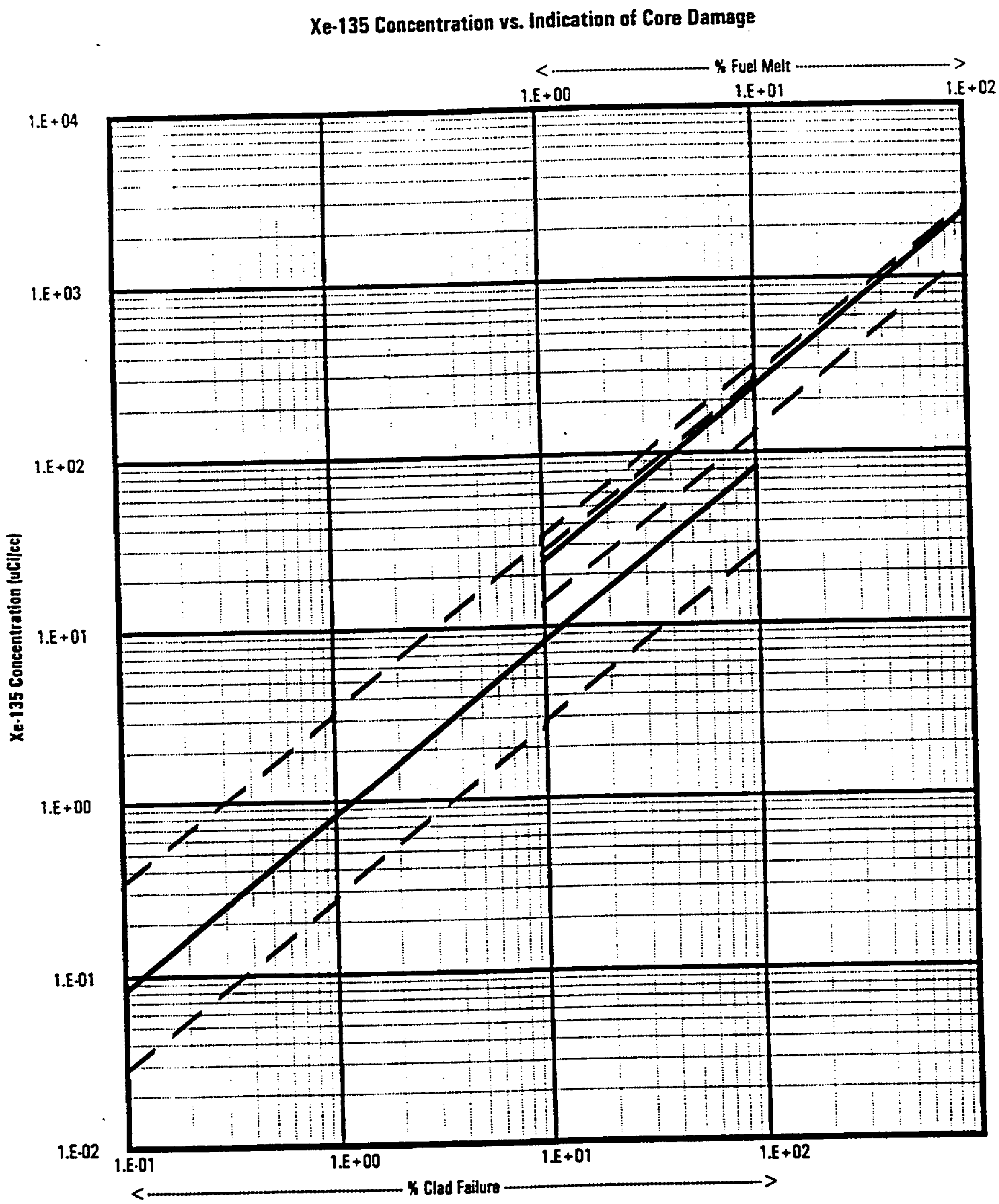
Kr-85 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 14
Page 1 of 1
Xe-133 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 15
Page 1 of 1
Xe-135 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 16
Page 1 of 2
ISOTOPIC RATIO INDICATION OF RELEASE SOURCE

1. Obtain the decay corrected fission products from Attachment 4 and calculate the ratios as described in step 2.

2. $\frac{\text{Kr-85m}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{Kr-87}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{Kr-88}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{I-132}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-133}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-134}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-135}}{\text{I-131}}$ = ______________________

ATTACHMENT 16
Page 2 of 2

Fission Product Ratio	Ratio in Pellet (indicates fuel melt)	Ratio in Pellet/Clad Gap (indicates clad damage)
Kr-85m / Xe-133	0.122	0.023
Kr-87 / Xe-133	0.233	0.0234
Kr-88 / Xe-133	0.33	0.0495
I-132 / I-131	1.46	0.127
I-133 / I-131	2.09	0.685
I-134 / I-131	2.3	0.155
I-135 / I-131	1.97	0.364

ATTACHMENT 17
Page 1 of 3
CORE DAMAGE ASSESSMENT SUMMARY, DETERMINATION AND RECOMMENDATIONS

Date: ________________ Time: ________________ Summary No.: ___________

1. Assessment of amount and type of core damage based on DAPA readings.

____

____

____

____

2. Assessment of the % metal-water reaction and corresponding clad failure (determine in conjunction with assessment of adequacy of core cooling if possible).

____

____

____

____

3. Assessment of the adequacy of core cooling.

____

____

____

____

4. Assessment of release source based on isotopic ratios.

____

____

____

ATTACHMENT 17
Page 2 of 3

5. Observations of less volatile fission products.

____

____

____

____

6. Core damage estimates based on fission product concentrations from post accident samples as determined utilizing Attachments 7-15.

Fission Product	% Clad Failure	% Fuel Melt
	Liquid Sample	
I-131		
I-133		
I-135		
Cs-134		
Cs-137		
	Gas Sample	
Kr-85m		
Kr-85		
Xe-133		
Xe-135		

NOTES

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ATTACHMENT 17
Page 3 of 3

7. Summary, Determinations and Recommendations

____

____

____

____

8. Final Core Damage Estimate

____

____

____

____

Core Thermal Engineer	Date	Ti

PSE&G CONTROL COPY # EPIP059

SALEM GENERATING STATION

SC.EP-EP.ZZ-0205(Q) Rev. 02

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

USE CATEGORY: II

REVISION SUMMARY Biennial Review Performed: Yes X No ___

- Removed all references to Nuclear Fuels Engineer (NFE) and ensured all NFE duties were being performed by the Core Thermal – Hydraulic Engineer.
- Revised reference made to an attachment in NC.EP-EP.ZZ-0201(Q) in Section 1 of this procedure to match the latest revision of NC.EP-EP.ZZ-0201(Q).
- Replaced all references to EPIP202S/NC.EP-EP.0201(Q) with NC.EP-EP.ZZ-0201(Q).
- Revised Manager – EP to EP Manager.

IMPLEMENTATION REQUIREMENTS:

Implementation Date: 2/6/02

APPROVED: [signature] 1-29-02
EP Manager Date

APPROVED: [signature] 1-29-02
Vice President – Operations Date

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides information for the assessment of the degree of core damage during an accident, as required per NC.EP-EP.ZZ-0201(Q), Attachment 11. Specifically, the information contained in this procedure relates to:

- Determination of the degree of damage to the fuel rod cladding that results in the release of the fission product inventory in the fuel gap space.
- Determination of the degree of core overheating that results in the release of fission product inventory in the fuel pellets.
- **Section 5.1** of this procedure may be utilized by the Core Thermal-Hydraulics Engineer (CTE) to perform a core damage assessment using current plant parameters and Dose Equivalent Iodine (DEI) readings. Section 5.1 can be utilized immediately after the identification of a plant transient condition.
- **Section 5.2** of this procedure may be utilized by the Core Thermal Hydraulics Engineer to perform a core damage assessment using isotopic data obtained from a P.A.S.S. sample. Section 5.2 can be utilized after isotopic data is made available, which normally occurs several hours after the identification of a plant transient condition.

2.0 PREREQUISITES

2.1 Prerequisites To Be Followed Prior To Implementing This Procedure

2.1.1 Implement this procedure:

- At the discretion of the Core Thermal-Hydraulic Engineer (CTE)
- Upon staffing of the Emergency Response Facility

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 Personnel who implement this procedure shall be trained and qualified IAW the Emergency Plan.

3.1.2 This procedure should be used whenever the following indications are present:

- There are indicated core temperatures that trigger the use of Functional Restoration Guidelines, or
- There are indicated containment radiation levels that trigger an alarm.
- There has been an initiating event which may have resulted in fuel damage.

3.1.3 If additional support is needed for performing Fuel Damage Assessment, contact the Nuclear Fuels Manager.

3.2 Limitations

3.2.1 Only the core damage assessment methodology based on Dose Equivalent Iodine (DEI) can be used at any reactor power level. All other assessment methodologies should only be used after reactor trip or shutdown.

3.2.2 Clad damage of less than 1% is not considered to be a loss of the fuel cladding boundary.

3.2.3 The core damage assessment methodology does not account for fission product spiking.

3.2.4 The methodology used in Section 5.1 defines three types of core damage: no damage, cladding failure, and fuel over-heating; while the methodology utilizing isotopic data in Section 5.2 defines fuel melting as a fourth type of core damage. Fuel melt is not considered a separate core damage category in Section 5.1 because the diagnosis of core melting is not required to evaluate and implement in-plant recovery strategies. As a result, the diagnosis of a core over-temperature condition is adequate.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 PROCEDURE

> **NOTE:**
>
> Due to the multiple and, at times, unpredictable failure mechanism associated with core damage, this procedure has been developed to provide GUIDANCE for Core Damage Assessment. The sequence and extent of procedure performance should be based on the knowledge and experience of the Core Thermal-Hydraulics Engineer.

5.1 Perform Core Damage Assessment Using Current Plant Status and Dose Equivalent Iodine (DEI) [*CTE*]

(The methodology used in this section is found in Reference 1. The Salem-specific setpoints used in this methodology are discussed in References 4 and 7.)

5.1.1 RECORD in Attachment 1, "Plant Parameter Trending," the Unit, Date, and Time of Reactor Trip or Shutdown, if trip/shutdown has occurred.

5.1.2 OBTAIN the current status of each plant parameter specified in Attachment 1 and write it in Table 1, "Plant Parameter Trend," along with the time at which the plant parameters were recorded.

> Note: A core damage estimate is not used explicitly in the Salem Emergency Classification Guidelines (SECG) to determine the appropriate Emergency Action Level (EAL) in an accident. However, it is important to recognize that several plant parameters which indicate potential core damage are also used to determine EALs., most notably those found in **SECG Section 3.0 – Fission Product Barriers**. These parameters are as follows and can be used to perform a qualitative assessment of damage to the fission product barriers.

EAL #1.1.1.a	Reactor Coolant > 1 μCi/gm DEI for > 48 hours	Unusual Event
EAL #3.1.2	Reactor Coolant > 300 μCi/gm DEI	Alert or Higher
EAL #3.1.3.a	5 or More CETs > 700 °F	Alert or Higher
EAL #3.1.3.b	5 or More CETs > 1200 °F	Alert or Higher
EAL #3.1.4	RVLIS Full Range < 39% (Modes 1-4)	Alert or Higher
EAL #8.1.3.b	RVLIS Full Range < 57% (Modes 5,6)	Site Area Emerg.
EAL #3.1.5	Containment Radiation (R44A or R44B) > 300 R/hr	Alert or Higher
EAL #3.2.4	Containment Radiation (R44A or R44B) > 10 R/hr	Alert or Higher
EAL #3.3.5	Containment Radiation (R44A or R44B) > 2000 R/hr	Site Area Emerg.

* Adapted from SECG – Salem Event Classification Guidelines.

5.1.3 IDENTIFY the possible status of the reactor core by using Attachment 2, "High Level Core Damage Assessment."

Note: *A clean copy of Attachment 2, "High Level Core Damage Assessment" should be used each time this step is performed.*

5.1.4 ASSESS fuel rod cladding damage.

5.1.4.1 COMPLETE Attachment 3, "Clad Damage Estimate," to assess fuel rod cladding damage.

Note: *A clean copy of Attachment 3, "Clad Damage Estimate," should be used each time this step is performed.*

5.1.5 ASSESS fuel pellet over-temperature damage.

5.1.5.1 COMPLETE Attachment 4, "Fuel-Pellet Over-Temperature Damage Estimate," to assess fuel pellet over-temperature damage.

Note: *A clean copy of Attachment 4, "Fuel-Pellet Over-Temperature Damage Estimate," should be used each time this step is performed.*

RECOMMENDATION:

Because steps 5.1.3, 5.1.4, and 5.1.5 may be performed at numerous times throughout the duration of an event, it is recommended that several copies of Attachments 2, 3, and 4 be made at the time this procedure is implemented. Note that the core damage assessments performed in Attachments 3 and 4 are collected and trended in Attachment 6. Also, copies of Attachments 1 and 6 may be made as needed.

5.2 Perform Core Damage Assessment By Analyzing An Isotopic P.A.S.S. Sample

(See Reference 2 for a more detailed discussion of the methodology used in this section.)

5.2.1 PROVIDE recommendations to the Radiological Assessment Coordinator (RAC) to initiate post accident radionuclide samples according to the "Suggested Sampling Locations" in the table below.

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

"**Suggested Sampling Locations**" taken from Reference 2, page 48.

* Assume operating at that level for some appreciable time.

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

5.2.2 OBTAIN the P.A.S.S. sample isotopic data from the Chemistry Department.

- The majority of P.A.S.S. samples will be taken from the following media: Reactor Coolant System (RCS), containment sump, and/or containment atmosphere.

- More than one sample may be provided, each from different media.

5.2.3 If it is possible to analyze the isotopic data via the "Core Damage Assessment" Excel spreadsheet, GO TO Step 5.2.4.

If the Excel spreadsheet can NOT be utilized, GO TO Step 5.2.5 to perform a manual analysis of the data.

5.2.4 P.A.S.S. Sample – Automatic Calculation of Isotopic Data via the "Core Damage Assessment" Excel Spreadsheet

5.2.4.1 OPEN the "Core Damage Assessment" Excel Spreadsheet, located on the laptop, as "Read Only."

The spreadsheet is installed on the Laptop Computer, kept in Drawer C2, along the wall in the "Admin Support" area of the TSC.

- The Excel spreadsheet is found under D:\Core Damage Assessment\core_dam.xls. A shortcut to this directory exists on the desktop.

- The spreadsheet is password protected with the word "coredamage".

5.2.4.2 SAVE the "Read Only" spreadsheet by clicking **File** and then **Save As**. When prompted, rename your copy of the "Core Damage Assessment" spreadsheet with today's date in MMDDYYYY.xls format.

5.2.4.3 INPUT the following information on the spreadsheet.

Note: Only input data into the green boxes on the spreadsheet.

√	Input Parameter	Excel Spreadsheet Box
	Your Name	F10
	Plant	F11
	Date & Time of Reactor Shutdown / Trip	F4, G4
	Date & Time of RCS Sample	F5, G5
	Date & Time of Sump Sample	F6, G6
	Date & Time of Containment Atmosphere Sample	F7, G7

5.2.4.4 CHECK Box C14 if the isotopic samples taken from the RCS or sump are in units of µCi/gm.

5.2.4.5 INPUT the pressure and temperature of the containment atmosphere, sump, and RCS in Boxes D19 – D30.

5.2.4.6 INPUT the pressure, temperature, and water density ratio (ρ/ρ_{STP}), if needed, of any samples taken from the containment atmosphere, sump, and RCS in Boxes G19 – G31. See Attachment 7 to obtain the correct water density ratio.

NOTE

If isotopic data is not obtained from all three media (RCS, sump, and containment atmosphere), the pressure, temperature, and ρ/ρ_{STP} boxes for the unused media should contain dummy values. This practice will not impact the calculated results and will keep "Division By Zero" errors from appearing throughout the spreadsheet.

5.2.4.7 INPUT the isotopic data obtained from chemistry in the appropriate location of the spreadsheet.

Sample	Box Numbers
RCS	H42 – H53 B57 – B64
Sump	H74 – H85 B89 – B96
Containment Atmosphere	H104 – H115 B119 – B126

> **Note:**
> **For sample media and/or specific isotopes for which data was not obtained, fill in "0.00E+00" in the appropriate box.**

5.2.4.8 CHECK the appropriate box if the isotopic data has been previously decay corrected by Chemistry.

Sample	Decay Corrected Data Check Box
RCS	E44
Sump	E77
Containment Atmosphere	E107

> **Note:**
> **It is standard practice that all isotopic data received from the Chemistry Department will have been decay-corrected back to the time of reactor shutdown by the Chemistry Department.**

5.2.4.9 DETERMINE all liquid volumes that contribute inventory to the RCS and/or sump.

5.2.4.9.1 CHECK the appropriate boxes under the "RCS" or "Sump" headings in boxes L12 – L20 or M12 – M20, respectively.

5.2.4.9.2 INPUT the correct liquid volumes (Boxes N12 – N19) and RWST water levels (Q20 and S20) if these are different from the default values. The default values are the liquid volume at 100% capacity. **(Table 1, "Liquid Volumes Available to the RCS and Sump" of Section 5.2.5.3 contains the default values).**

5.2.4.10 INPUT the power history data in boxes M32 –M35, P32 – P35, S32 – S35.

- List power history information in chronological order from the most recent reactor start-up to shutdown.
- If isotopic data for Cs-134 exists, then use Attachment 8 to input the power correction factor in Box U56.

5.2.4.11 CHECK the applicable fuel burnup conditions.

Beginning of Cycle	Box AA3
Middle of Cycle	Box AA4
End of Cycle	Box AA5

5.2.4.12 ASSESS the amount of core damage by examining the information found in Column AH, in the Noble Gas Ratio and Iodine Ratio tables of Columns AL and AO, and the Dose Equivalent Iodine (DEI) table of Columns AL and AN.

5.2.4.13 SUMMARIZE a final core damage assessment in Attachment 23.

5.2.4.14 REPORT findings.

5.2.4.14.1 Report core damage estimate to appropriate emergency response personnel as identified in EPIP 201S/NC.EP-EP.0201(Q).

5.2.4.15 GO TO Step 5.1 to perform a core damage assessment based on current plant status, if needed.

5.2.5 P.A.S.S. Sample – Manual Calculation of Isotopic Data

5.2.5.1 DETERMINE if the sampling results need to be decay-corrected. **See Section 2.4.5.3 "Decay Correction" (Reference 2) for a more detailed discussion of this methodology.**

5.2.5.1.1 If the results have NOT been decay-corrected by the Chemistry Department. GO TO Attachment 9 and decay-correct the isotopic data.

5.2.5.1.2 If the results have already been decay-corrected by the Chemistry Department. GO TO Step 5.2.5.2.

Note:
It is standard practice that all isotopic data received from the Chemistry Department will have been decay-corrected back to the time of reactor shutdown by the Chemistry Department.

5.2.5.2 DETERMINE if the sampling results need to be pressure and temperature-corrected. **See Section 2.4.5.2 "Pressure and Temperature Adjustment" (Reference 2) for a more detailed discussion of this methodology.**

5.2.5.2.1 The isotopic data does NOT need to be pressure and temperature-corrected if the following is **true**. GO TO Step 5.2.5.3.

- The RCS sample or sump sample is in units of μCi/gm.
- The RCS sample or sump sample is in units of μCi/cc and the temperature of the RCS or sump is less than 200 °F.
- The pressure and temperature of the containment atmosphere sample at the time of analysis is the same as the conditions of the containment atmosphere.

5.2.5.2.2 The isotopic data DOES need to be pressure and temperature-corrected if the following is **true**. GO TO Attachment 10 and pressure/temperature-correct the isotopic data.

- The RCS sample or sump sample is in units of μCi/cc and the temperature of the RCS or sump is greater than 200 °F.
- The pressure and temperature of the containment atmosphere sample at the time of analysis is different than the conditions of the containment atmosphere.

5.2.5.3 CALCULATE the total volume of the coolant inventory available to the RCS (if RCS sample) or sump (if sump sample).

5.2.5.3.1 ADD each of the volumes in Table 1 (on the next page) that contributes to the coolant inventory in either the RCS or sump.

5.2.5.3.2 WRITE the total volume of coolant inventory of the RCS or sump in Table 2.

Table 1: *Liquid Volumes Available to the RCS and Sump

Coolant Volumes	Liquid Volume (ft^3)
Reactor Vessel (To Top of Vessel)	4945
Reactor Vessel (To Hot/Cold Leg)	3300
Reactor Vessel (To Top of Fuel)	2500
All 4 Accumulators (1 = 850 ft^3)	3400
All 4 Steam Generators **(Salem Unit 1)** (1 SG= 966.1 ft^3)	3864
All 4 Steam Generators **(Salem Unit 2)** (1 SG= 1080 ft^3)	4320
All 4 Hot Legs (1 = 108 ft^3)	432
All 4 Cold Legs (1 = 302 ft^3)	1208
Pressurizer (nominal)	1080
Pressurizer (solid)	1800
Pressurizer Surge Line	45

***The references for each value are found in Appendix 1**

Table 2: Calculated Volumes of Coolant Inventory in RCS and Sump

Volume of Coolant Inventory	**(ft^3)**
Sump	
RCS	

5.2.5.4 CALCULATE the total mass of the coolant inventory (for any RCS/sump samples) and total volume of containment (for any containment atmosphere samples).

5.2.5.4.1 USE the following equation for an RCS or sump sample.

$$M = V \times \frac{\rho}{\rho_{STP}} \times 1.0\ g/cc \times 28.3E3\ cc/ft^3$$

Where,

M = mass of coolant inventory (gm)

$\frac{\rho}{\rho_{STP}}$ = water density ratio obtained from Attachment 7

V = volume of coolant inventory (ft^3) from Table 2

5.2.5.4.2 USE the following equation for a containment atmosphere sample.

$$CV = (2.6E6\ ft^3 - SV) \times 28.3E3\ cc/ft^3 \times \frac{14.7\ psia}{P_2} \times \left(\frac{T_2 + 460}{32^\circ F + 460}\right)$$

Where,
SV = volume of coolant inventory in sump (ft^3) in Table 2
T_2, P_2 = temperature (°F) and pressure (psia) of the containment atmosphere at time of sample
CV = corrected containment atmosphere volume (cc)

5.2.5.4.3 WRITE the calculated mass and/or volume in Table 3 below.

Table 3: Calculated RCS and Sump Masses and Containment Atmosphere Volume

RCS Mass (g)	
Sump Mass (g)	
Containment Atmosphere Volume (cc)	

5.2.5.5 CALCULATE the total activity released for each isotope from each sample medium.

5.2.5.5.1 RECORD the corrected specific activities from each sample on the appropriate page in Attachment 11. Make sure to transfer the corrected specific activities for each sample medium from the most recently used attachment (either Attachment 9 or 10). If neither Attachment 9 nor 10 was utilized, then transfer the isotopic activities from the Chemistry Data.

5.2.5.5.2 TRANSFER the coolant inventory total mass (for an RCS or sump sample) or volume (for a containment atmosphere sample) from Table 3 onto the appropriate page in Attachment 11.

5.2.5.5.3 COMPLETE the multiplication for each isotope as indicated in Attachment 11 for each sample analyzed.

5.2.5.6 CALCULATE the total amount of activity released.

- If there is only one sample, GO TO Step 5.2.5.7.
- If there are several samples from different media, PERFORM the indicated additions to calculate the total amount of activity released for each isotope from all of the media in Attachment 12.

5.2.5.7 CALCULATE the noble gas ratios for the following isotopes.

Xe-131m	Kr-85m
Xe-133m	Kr-87
Xe-135	Kr-88

5.2.5.7.1 USING the total isotopic activities found in Attachment 12 (for several samples) or in Attachment 11 (for one sample), the noble gas ratios are found by dividing the above activities by the activity for Xe-133.

$$\text{Noble Gas Ratio} = \frac{\text{Total Activity of Noble Gas Isotope}}{\text{Total Activity of Xe - 133}}$$

5.2.5.7.2 RECORD the calculated noble gas ratios in Table 4 below.

Table 4: Noble Gas Ratios

Noble Gas Isotope	Noble Gas Ratio (/Xe-133)
Xe-131m	
Xe-133m	
Xe-135	
Kr-85m	
Kr-87	
Kr-88	

5.2.5.8 CALCULATE the iodine ratios for the following isotopes.

I-132 I-133 I-135

5.2.5.8.1 USING the total isotopic activities found in Attachment 12 (for several samples) or in Attachment 11 (for one sample), the iodine ratios are found by dividing the above activities by the activity for I-131.

$$\text{Iodine Ratio} = \frac{\text{Total Activity of Iodine Isotope}}{\text{Total Activity of I - 131}}$$

5.2.5.8.2 RECORD the calculated iodine ratios in Table 5.

Table 5: Iodine Ratios

Iodine Isotope	Iodine Ratio (/I-131)
I-132	
I-133	
I-135	

5.2.5.9 CALCULATE the Dose Equivalent Iodine (DEI), if not already provided by Chemistry.

5.2.5.9.1 USE the equation below to find the DEI in units of $\mu Ci/cc$ or $\mu Ci/gram$. (The constants used in the equation below are from Reference 5.)

$$DEI = (1.00 \times \textit{I-131}) + (0.0361 \times \textit{I-132}) + (0.270 \times \textit{I-133}) + (0.0169 \times \textit{I-134}) + (0.0838 \times \textit{I-135})$$

5.2.5.10 RECORD the DEI in Table 6.

Table 6: Dose Equivalent Iodine (DEI)

DEI	

5.2.5.11 OBTAIN the power history prior to shutdown.

5.2.5.12 CALCULATE the Power Correction Factor (PCF) for each isotope that is available for release utilizing the appropriate Attachment. **See Section 2.3.1, "Power Correction Factor" (Reference 2) for a discussion of this methodology.**

Attachment 13	Attachment 14	Attachment 15	Attachment 16
Kr-85m	Xe-131m	Sr-90	Sr-89
Kr-87	Xe-133	Cs-134	
Kr-88	Xe-133m	Cs-137	
Xe-135	I-131		
I-132	Te-132		
I-133	Ba-140		
I-135			
Te-129			
Pr-144			

5.2.5.13 CALCULATE the inventory available from each isotope that is available for release in Attachment 17.

5.2.5.14 CALCULATE Percent Inventory Released in Attachment 18.

5.2.5.15 EXECUTE an assessment of the core damage based on the percent inventories released in Attachment 19, using the figures found in Attachment 20.

5.2.5.16 EXECUTE an assessment of the core damage based on the Noble Gas and Iodine ratios calculated in Tables 4 and 5, using the table in Attachment 21.

5.2.6.17 EXECUTE an assessment of the core damage based on the Dose Equivalent Iodine (DEI), either provided from Chemistry or calculated in Table 6, using Attachment 22.

5.2.5.18 SUMMARIZE a final assessment of core damage based on the percent inventories released and the noble gas and iodine ratios in Attachment 23.

5.2.5.19 REPORT findings.

5.2.5.19.1 Report core damage estimate to appropriate emergency response personnel as identified in NC.EP-EP.ZZ-0201(Q).

5.2.5.20 GO TO Step 5.1 to perform a core damage assessment based on current plant status, if needed.

6.0 REFERENCES

6.1 WCAP-14696-A, Westinghouse Owners Group, Core Damage Assessment Guidance, Rev. 1, November 1999.

6.2 Westinghouse Owners Group Document, Post Accident Core Damage Assessment Methodology, Rev. 2, November 1984.

6.3 NC.EP-EP.ZZ-0201(Q), TSC-Integrated Engineering Response, Rev. 2.

6.4 NFS Calc File DS1.6-0244, "Basis For Salem-Specific Values Used In SC.EP-EP.ZZ-0205(Q) – *TSC – Post Accident Core Damage Assessment*," 07/26/1999.

6.5 EPRI Document TR-108779, "Fuel Integrity Monitoring & Failure Evaluation Handbook," Appendix B, August 1998.

6.6 NFS Calc File DS1.6-0098, "Verification of Emergency Action Levels for Event Classification Guide," 02/10/1995.

6.7 NFS Calc File DS1.6-0316, "Updated Basis For Salem-Specific Values Used In SC.EP-EP.ZZ-0205(Q) – *TSC – Post Accident Core Damage Assessment*," 12/19/2000.

APPENDIX 1
Page 1 of 2

REFERENCES FOR TABLE 1
"Liquid Volumes Available to the RCS and Sump"

Coolant Volumes

1. **Reactor Vessel (To Top of Vessel) = 4945 ft^3**
Per *Salem UFSAR, Table 5.2-3, "Reactor Vessel Design Data," Rev 16, 01/31/1998,* the total reactor vessel water volume with core and internals in place is 4945 ft^3.

2. **Reactor Vessel (To Hot/Cold Leg) = 3300 ft^3**
Per *Salem UFSAR, Figure 5.1-1, "Reactor Vessel Schematic," Rev 6, 02/15/1987,* the nozzles are at a height roughly two-thirds above the bottom of the vessel. Thus,
$\frac{2}{3} \times 4945\ \text{ft}^3 \approx 3300\ \text{ft}^3$.

3. **Reactor Vessel (To Top of Fuel) = 2500 ft^3**
Per PSBP #117627, *"Reactor General Assembly Internals," Rev 4, 07/05/1989* the top of the fuel (approximately at the upper core plate) is at a height roughly one half above the bottom of the vessel. Thus, $\frac{1}{2} \times 4945\ \text{ft}^3 \approx 2500\ \text{ft}^3$.

4. **All 4 Accumulators = 3400 ft^3**
Per *Salem UFSAR, Table 6.2-3, "Accumulator Design Parameters," Rev 6, 02/15/1987,* the minimum water volume for 1 accumulator at operating conditions is 850 ft^3. Thus, the volume of all 4 accumulators is 4 x 850 ft^3 = 3400 ft^3.

5. **All 4 Steam Generators (Salem Unit 1) = 3864 ft^3**
All 4 Steam Generators (Salem Unit 2) = 4320 ft^3
Per *Salem UFSAR, Table 5.2-5a, "Steam Generator Design Data (Model F)," Rev 18, 04/26/2000,* the reactor coolant water volume in one steam generator is 966.1 ft^3. Thus, the volume of all 4 steam generators for Salem Unit 1 is 4 x 966.1 ft^3 = 3864 ft^3.
Per *Salem UFSAR, Table 5.2-5, "Steam Generator Design Data (Model 51)," Rev 18, 04/26/2000,* the reactor coolant water volume in one steam generator is 1080 ft^3. Thus, the volume of all 4 steam generators for Salem Unit 2 is 4 x 1080 ft^3 = 4320 ft^3.

6. **All 4 Hot Legs = 432 ft^3**
Per *NFS Calc File T01.6-093, "Consolidated Documentation for the RETRAN02 Mod03 Model of Salem Units 1 and 2, 02/29/1991,* the volume of one hot leg is 109 ft^3. Thus, the volume of all 4 hot legs is 4 x 108 ft^3 = 432 ft^3.

7. **All 4 Cold Legs and RCS Pumps = 1208 ft^3**
Per *NFS Calc File T01.6-093, "Consolidated Documentation for the RETRAN02 Mod03 Model of Salem Units 1 and 2, 02/29/1991,* the volume of one pump suction volume is 143.76 ft^3, the volume of one pump is 56.0 ft^3, and the volume of one cold leg is 102.33 ft^3. The sum of these volumes is 143.76 ft^3 + 56.0 ft^3 + 102.33 ft3 = 302 ft^3. Thus, the volume of all 4 cold legs, pumps, and pump suction volumes is 4 x 302 ft^3 = 1208 ft^3.

APPENDIX 1 (con't)
Page 2 of 2

REFERENCES FOR TABLE 1
"Liquid Volumes Available to the RCS and Sump"

<u>Coolant Volumes</u>

8. **Pressurizer (nominal)** = **1080 ft^3**
Per *Salem UFSAR, Table 5.2-4, "Pressurizer and Pressurizer Relief Tank Design Data," Rev 16, 01/31/1998,* the water volume of the pressurizer at full power is 1080 ft^3.

9. **Pressurizer (solid)** = **1800 ft^3**
Per *Salem UFSAR, Table 5.2-4, "Pressurizer and Pressurizer Relief Tank Design Data," Rev 16, 01/31/1998,* the water volume of the pressurizer at full power is 1080 ft^3 and the steam volume at full power is 720 ft^3. If the pressurizer was water solid, the amount of liquid would be 1080 ft^3 + 720 ft^3 = 1800 ft^3.

10. **Pressurizer Surge Line** = **45 ft^3**
Per *NFS Calc File T01.6-093, "Consolidated Documentation for the RETRAN02 Mod03 Model of Salem Units 1 and 2, 02/29/1991,* the volume of the pressurizer surge line is 45 ft^3.

11. **Refueling Water Storage Tank (RWST)** = **calculated by:**
= (Initial RWST Level – Final RWST Level) x 1115 ft^2
Per *Salem UFSAR, Table 6.3-4, "Refueling Water Storage Tank Design Parameters," Rev 16, 01/31/1998,* the tank capacity is 400,000 gallons, while the minimum volume is 364,500 gallons. The straight side height is given as 48 ft.

400,000 gallons = 53,472 ft^3
364,500 gallons = 48,726 ft^3

The "100%" value is assumed to be the minimum allowable RWST volume of 48,726 ft^3. This occurs at a RWST level of $\left(\frac{48{,}726\ ft^3}{53{,}472\ ft^3}\right) \times 48\ ft = 43.7\ ft$. The total volume of water entering the Reactor Coolant System (RCS) from the RWST is found below.

Total RWST Volume Entering RCS = (Initial RWST Level – Final RWST Level) x $\left(\frac{48{,}726\ ft^3}{43.7\ ft}\right)$

Total RWST Volume Entering RCS = (Initial RWST Level – Final RWST Level) x 1115 ft^2

ATTACHMENT 1
Page 1 of 1

PLANT PARAMETER TRENDING

Unit:_________ Date:_________

Time of Reactor Trip or Shutdown:_____________

Table 1: Plant Parameter Trend

#	Plant Parameter	Reading at Time =	Reading at Time =	Reading at Time =	Reading at Time =	Reading at Time =
1.	DEI (if known)	µCi/gr or µCu/cc	µCi/gr or µCu/cc	µCi/gr or µCu/cc	µCi/gr or µCu/cc	µCi/gr or µCu/cc
2.	Hottest CET Temperature	°F	°F	°F	°F	°F
3.	Number of CETs > 1200°F					
4.	Number of CETs > 1400°F					
5.	Number of CETs > 2000°F					
6.	Number of Operable CETs (Max. is 58)					
7.	R44 A Reading	Rad/hr	Rad/hr	Rad/hr	Rad/hr	Rad/hr
	R44 B Reading	Rad/hr	Rad/hr	Rad/hr	Rad/hr	Rad/hr
8.	RCS Pressure	psig	psig	psig	psig	psig
9.	T_{sat} at the above RCS Pressure [#8.]	°F	°F	°F	°F	°F
10.	Containment H_2 Level	volume %	volume %	volume %	volume %	volume %
11.	RVLIS Reading	%	%	%	%	%
12.	Hot Leg RTD Temperature	°F	°F	°F	°F	°F
13.	SRM Reading	cts/sec	cts/sec	cts/sec	cts/sec	cts/sec

CET Core Exit Thermocouple
DEI Dose Equivalent Iodine
H_2 Hydrogen
RCS Reactor Coolant System
RTD Resistance Temperature Detector
RVLIS Reactor Vessel Level Indication System
SRM Source Range Monitor
T_{sat} Saturated Temperature

ATTACHMENT 2
Page 1 of 2

HIGH LEVEL CORE DAMAGE ASSESSMENT

High Level Core Damage Assessment

STEP 1: DETERMINE the time elapsed between Reactor Shutdown and the time at which the plant parameters in Table 1 of Attachment 1, "Plant Parameter Trending," were obtained and RECORD in Table 2.

> **Note: If a Reactor Trip or Shutdown has not occurred, WRITE "Not Applicable," and assess only the DEI, if known, in Table 3.**

STEP 2: DETERMINE the correct values of <CRM1> by using Figure 1, "Containment Radiation Level Vs. Time For RCS Release," in Attachment 5, "Figures For Section 5.1," and <CRM2> by using Figure 2, "Containment Radiation Level Vs. Time For 1% Fuel Pellet Over-Temperature Release," in Attachment 5, "Figures For Section 5.1," and RECORD in Tables 2 and 3.

Table 2: Values of Containment Radiation Level Setpoints <CRM1> and <CRM2>

#14. Time After Rx Shutdown (hrs)	#15. <CRM1>	#16. <CRM2>
hrs	R/hr	R/hr

> **NOTE:**
> The value of <CRM1> is the postulated containment radiation level corresponding to normal reactor coolant system activity with an iodine spike, while <CRM2> is the postulated containment radiation level corresponding to a 1% fuel pellet over-temperature fission product release at specific times after reactor shutdown. The actual containment radiation monitor (R-44) reading is compared to <CRM3> and <CRM4> for the purpose of discriminating between fuel rod gap activity from cladding damage and the onset of fission product releases from the fuel pellet due to high fuel pellet temperatures.

ATTACHMENT 2 (Con't.)
Page 2 of 2

HIGH LEVEL CORE DAMAGE ASSESSMENT

High Level Core Damage Assessment (Continued)

STEP 3: IDENTIFY the possible status of the reactor core by using Table 3, "Fuel Rod Fission Product Status."

Table 3: Fuel Rod Fission Product Status

Plant Status	Fuel Rod Fission Product Status
Core Exit Thermocouples LESS THAN 700 °F **AND** Containment Radiation LESS THAN <CRM1> #15 Rad/hr **AND** Dose Equivalent Iodine (DEI) [if known] LESS THAN 1 µCi/gm or 1 µCi/cc	No Core Damage Continue to Monitor Plant Parameters Go To Step 5.1.2
Core Exit Thermocouples LESS THAN 2000 °F **AND** Containment Radiation LESS THAN <CRM2> #16 Rad/hr **OR** Dose Equivalent Iodine (DEI) [if known] GREATER THAN 1 µCI/gm or 1 µCI/cc	Possible Fuel Rod Clad Damage Go To Step 5.1.4 and Attachment 3
Core Exit Thermocouples GREATER THAN 2000 °F **OR** Containment Radiation GREATER THAN <CRM2> #16 Rad/hr	Possible Fuel Pellet Over-temperature Damage Go To Step 5.1.5 and Attachment 4

Values for #15 <CRM1> and #16 <CRM2> are obtained from Table 2 on Page 22.

ATTACHMENT 3
Page 1 of 5

CLAD DAMAGE ESTIMATE (STEP 5.1.4)

A. Estimate Fuel Rod Clad Damage Based on <u>Containment Radiation (CR) Levels</u>

STEP 1: DETERMINE the correct value of <CRM3> by using Figure 3, "Containment Radiation Level Vs. Time For 100% Clad Damage Release," in Attachment 5, "Figures For Section 5.1," and RECORD in Table 4.

Table 4: Values of Containment Radiation Level Setpoint <CRM3>

#14. Time After Rx Shutdown	#17. <CRM3>
hrs	R/hr

STEP 2: ESTIMATE clad damage by using the following equation.

$$\text{Clad Damage}_{CR}\ (\%) = \frac{\text{Current Containment Radiation Level}}{\text{Predicted Containment Radiation Level at 100\% Clad Damage}} \times 100$$

$$\text{Clad Damage}_{CR}\ (\%) = \frac{\boxed{\quad}_{\#7}}{\boxed{\text{<CRM3>}}_{\#17}} \times 100 = \boxed{\quad \%}_{\#18}$$

ATTACHMENT 3 (Con't)
Page 2 of 5

CLAD DAMAGE ESTIMATE (STEP 5.1.4)

B. Estimate Fuel Rod Clad Damage Based on Core Exit Thermocouple (CET) Readings

STEP 3: ESTIMATE clad damage by using one of the following two (2) equations.

- ***For RCS Pressure*** [#8] ***> 1600 psig,***

$$\text{Clad Damage}_{CET}\ (\%) = \frac{\text{Number of CETs} > 1400\,°\text{F}}{\text{Total Number of Operable CETs}} \times 100$$

$$\text{Clad Damage}_{CET}\ (\%) = \frac{[\ \#4\ ]}{[\ \#6\ ]} \times 100 = [\ \ \%\ \#19\ ]$$

OR

- ***For RCS Pressure*** [#8] ***< 1600 psig,***

$$\text{Clad Damage}_{CET}\ (\%) = \frac{\text{Number of CETs} > 1200\,°\text{F}}{\text{Total Number of Operable CETs}} \times 100$$

$$\text{Clad Damage}_{CET}\ (\%) = \frac{[\ \#3\ ]}{[\ \#6\ ]} \times 100 = [\ \ \%\ \#19\ ]$$

CLAD DAMAGE ESTIMATE (STEP 5.1.4)

C. Estimate Fuel Rod Clad Damage Based on <u>Dose Equivalent Iodine (DEI) Readings</u>

STEP 4: ESTIMATE clad damage by using Figure 5, "Clad Damage Vs. Dose Equivalent Iodine, " in Attachment 5, "Figures For Section 5.1."

DEI is: [#1]

Clad $Damage_{DEI}$ (%) = [% #20]

Note: Figure 5, "Clad Damage Vs. Dose Equivalent Iodine," was created from data found in Reference 6. Due to the very rough correlation between DEI and Clad Damage, the relatively small difference between the DEI in units of μCi/cc and units of μCi/gm does not substantially impact the estimated clad damage.

CLAD DAMAGE ESTIMATE (STEP 5.1.4)

D. Estimate Reasonableness of Clad Damage Estimates.

STEP 5: COMPARE current plant parameters to expected response by completing Table 5, "Current Vs. Expected Value Comparison Table," below.

Table 5: Current Vs. Expected Value Comparison Table

Expected Response	YES / NO ?
Containment H_2 Level Is [#10] < 0.5 volume %?	
RVLIS Reading Is 44% < [#11] < 57%	
Hot Leg RTD Temperature Is [T_{sat} #9] < [#12] < 650°F	
Source Range Monitor (SRM) Reading Is [#13] > 20,000 cps ?	
Clad Damage Estimates Is $\left\lvert \dfrac{[\ \%\ \#18\ ] - [\ \%\ \#19\ ]}{[\ \%\ \#18\ ]} \right\rvert$ x 100 $\leq$ 50 % *"Is the absolute value of [(% Clad Damage$_{CR}$ – % Clad Damage$_{CET}$) / % Clad Damage$_{CR}$] x 100 $\leq$ 50% ?"*	

CLAD DAMAGE ESTIMATE (STEP 5.1.4)

D. Estimate Reasonableness of Clad Damage Estimates (Continued)

STEP 6: For any "NO" responses in Table 5, DETERMINE if the deviation can be explained from the accident progression.

	✓
Injection of water to the RCS	
Bleed paths from the RCS	
Direct radiation to the containment radiation monitors	
Fuel burnup	
Fission product retention in the RCS	
Fission product removal from containment	
Conservatisms in the predictive model	
Other:	

E. Trend and Report Findings

STEP 7: RECORD estimated fuel pellet over-temperature damage in ATTACHMENT 6, "Core Damage Assessment Trending."

STEP 8: REPORT findings using guidance below.

1. If clad damage estimates have increased by more than 1% in the past 30 minutes or if estimates exceed 2% clad damage, report possible degradation of fission product barrier to appropriate emergency response personnel as identified in NC.EP-EP.ZZ-0201(Q).

2. Report core damage estimate to appropriate emergency response personnel as identified in NC.EP-EP.ZZ-0201(Q)

STEP 9: RETURN to 5.1.2 or 5.2 and PERFORM another core damage assessment, as needed.

ATTACHMENT 4
Page 1 of 6

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

A. Estimate Fuel Pellet Over-Temperature Damage Based on Containment Radiation (CR) Levels

STEP 1: DETERMINE the correct value of <CRM4> by using Figure 4, "Containment Radiation Level Vs. Time For 100% Fuel Over-Temperature Release," in Attachment 5, "Figures For Section 5.1," and RECORD in Table 6 below.

Table 6: Values of Containment Radiation Level Setpoint <CRM4>

#14. Time After Rx Shutdown	#21. <CRM4>
hrs	R/hr

STEP 2: ESTIMATE fuel pellet over-temperature damage by using the following equation.

$$\text{Over - Temp. Damage}_{CR}\ (\%) = \frac{\text{Current Containment Radiation Level}}{\text{Predicted Containment Radiation Level at 100\% Over - temp Damage}} \times 100$$

$$\text{Fuel Pellet Over-temperature Damage}_{CR}\ (\%) = \frac{\boxed{\#7}}{\boxed{\text{<CRM4>}\ \#21}} = \boxed{\%\ \#22}$$

ATTACHMENT 4 (Con't)
Page 2 of 6

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

B. Estimate Fuel Pellet Over-Temperature Damage Based on Core Exit Thermocouple (CET) Readings

STEP 3: ESTIMATE fuel pellet over-temperature damage by using the following equation.

$$\text{Over - Temp. Damage}_{CET}\ (\%) = \frac{\text{Number of CETs} > 2000\,^{\circ}\text{F}}{\text{Total Number of Operable CETs}} \times 100$$

$$\text{Over-Temperature Damage}_{CET}\ (\%) = \frac{\boxed{\quad \#5}}{\boxed{\quad \#6}} = \boxed{\quad \% \quad \#23}$$

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

C. Estimate Fuel Pellet Over-Temperature Damage Based on Containment Hydrogen Level

STEP: 4 DETERMINE the expected Containment H_2 Concentration at 100% Fuel Pellet Over-Temperature by using Table 7 below.

Table 7: Expected Containment Hydrogen Concentration At Various Conditions

RCS Pressure [#8] psig	Water Injection to the RCS	Predicted Containment Hydrogen Concentration For 100% Fuel Pellet OT Damage
RCS Pressure < 1050 psig	Yes	50 volume %
	No	25 volume %
RCS Pressure > 1050 psig	Yes	75 volume %
	No	25 volume %

The Expected Containment Hydrogen Concentration is [#24] %

STEP 5: ESTIMATE fuel pellet over-temperature damage by using the following equation.

$$\text{Over-Temperature Damage}_{\text{HYDROGEN}}\ (\%) = \frac{[\#10]}{[\#24]} \times 100 = [\ \ \ \%\ \#25]$$

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

D. Estimate Reasonableness of Fuel Pellet Over-Temperature Damage Estimates.

STEP 6: COMPARE current plant parameters to expected response by completing Table 8 below.

Table 8: Current Vs. Expected Value Comparison Table

Expected Response	YES / NO ?
RVLIS Reading Is [#11] < 44%	
Hot Leg RTD Temperature Is [#12] > 650°F	
Source Range Monitor (SRM) Reading Is [#13] > 20,000 cps ?	
Fuel Pellet Over-Temperature Damage Estimates Is $\left\| \frac{[\ \%\ \#22] - [\ \%\ \#23]}{[\ \%\ \#22]} \right\| \times 100 \leq 50\ \%$ *"Is the absolute value of [(% Over-Temp. Damage$_{CR}$ – % Over-Temp. Damage$_{CET}$) / % Over-Temp. Damage$_{CR}$] x 100 ≤ 50% ?"*	

(Table 8 is continued on the next page)

ATTACHMENT 4 (Con't)
Page 5 of 6

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

D. Estimate Reasonableness of Fuel Pellet Over-Temperature Damage Estimates (Continued).

Table 8 (Con't): Current Vs. Expected Value Comparison Table

Expected Response	YES / NO ?
Fuel Pellet Over-Temperature Damage Estimates Is | [% #22] — [% #25] | ≤ 25 % *"Is the absolute value of [(% Over-Temp. Damage$_{CR}$ – % Over-Temp. Damage$_{HYDROGEN}$) ≤ 25% ?"*	
Fuel Pellet Over-Temperature Damage Estimates Is | [% #23] — [% #25] | ≤ 25 % *"Is the absolute value of [(% Over-Temp. Damage$_{CET}$ – % Over-Temp. Damage$_{HYDROGEN}$) ≤ 25% ?"*	

FUEL PELLET OVER-TEMPERATURE DAMAGE ESTIMATE (STEP 5.1.5)

D. Estimate Reasonableness of Clad Damage Estimates (Continued)

STEP 7: For any "NO" responses in Table 8, DETERMINE if the deviation can be explained from the accident progression.

	✓
Injection of water to the RCS	
Bleed paths from the RCS	
Direct radiation to the containment radiation monitors	
Hydrogen burn in containment or operation of hydrogen igniters	
Fuel burnup	
Fission product retention in the RCS	
Fission product removal from containment	
Conservatisms in the predictive model	
Other:	

E. Trend and Report Findings

STEP 8: RECORD estimated fuel pellet over-temperature damage in ATTACHMENT 6, "Core Damage Assessment Trending."

STEP 9: REPORT findings using guidance below.

1. If clad damage estimates have increased by more than 1% in the past 30 minutes or if estimates exceed 2% clad damage, report possible degradation of fission product barrier to appropriate emergency response personnel as identified in NC.EP-EP.ZZ-0201(Q).

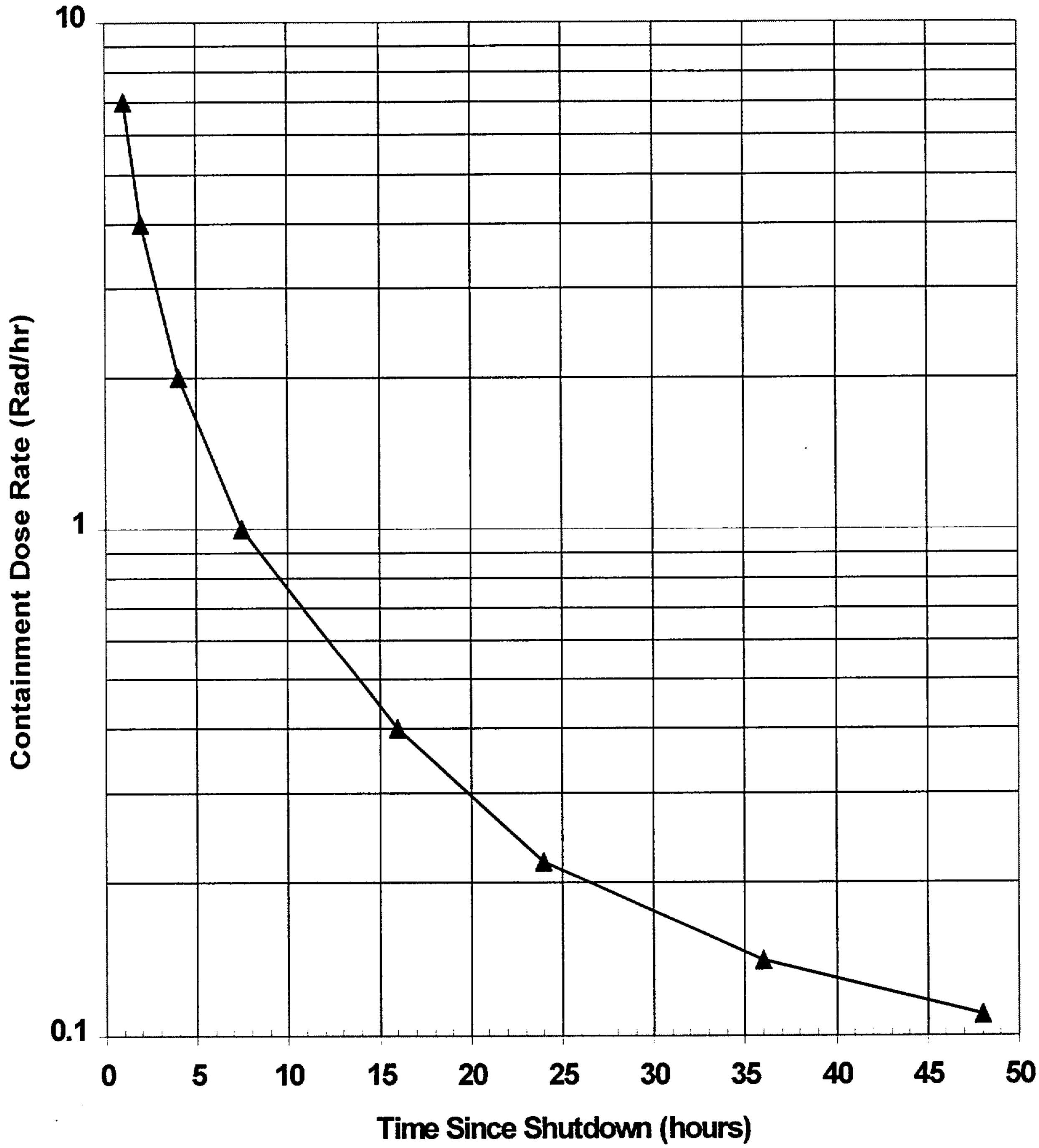
2. Report core damage estimate to appropriate emergency response personnel as identified in NC.EP-EP.ZZ-0201(Q)

STEP 10: RETURN to 5.1.2 or 5.2 and PERFORM another core damage assessment, as needed.

ATTACHMENT 5
Page 1 of 5

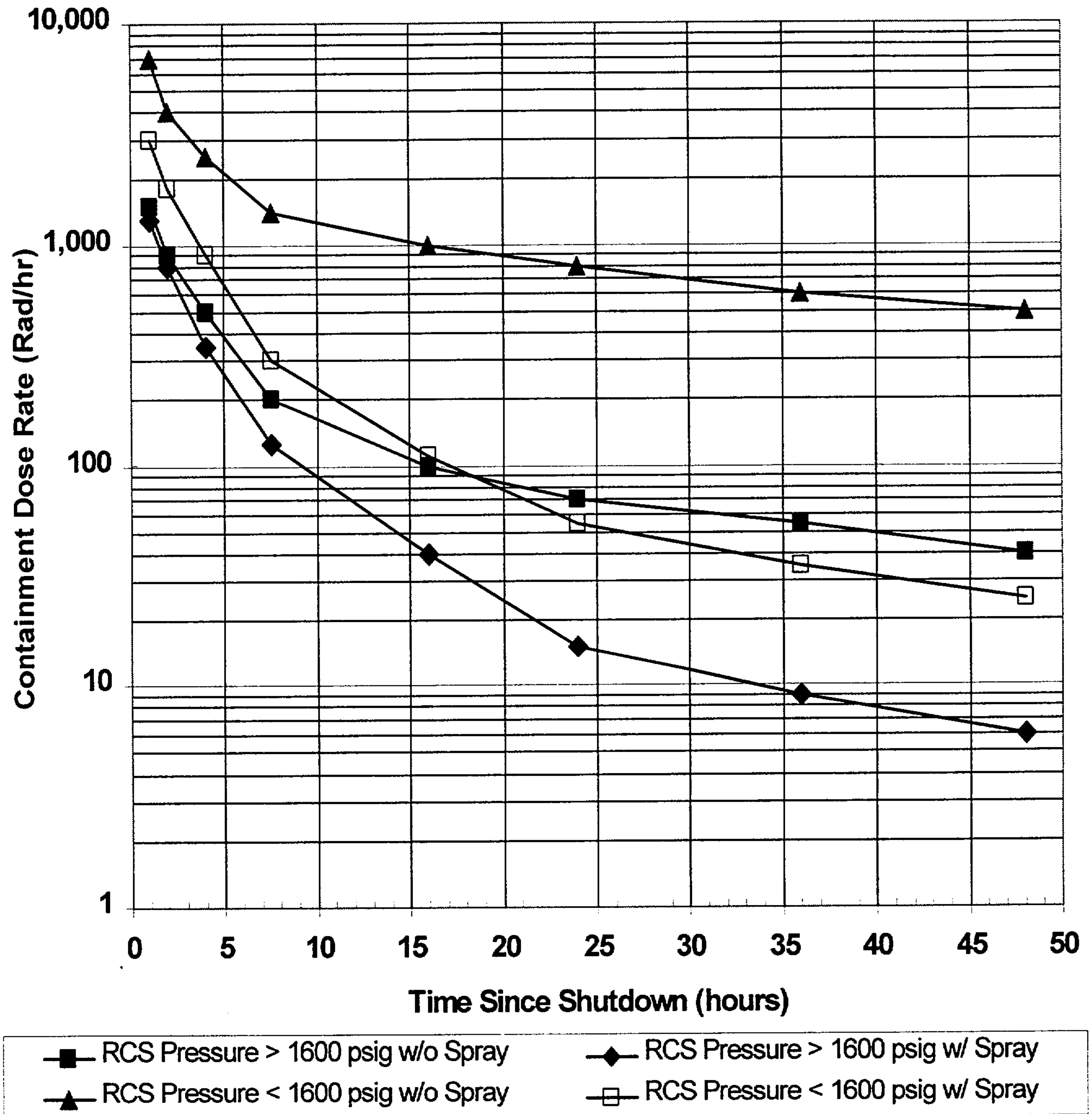
FIGURES FOR SECTION 5.1

FIGURE 1: <CRM1>
"CONTAINMENT RADIATION LEVEL VS. TIME FOR RCS RELEASE"
(Adapted from Figure 1 of Reference 1)



FIGURES FOR SECTION 5.1

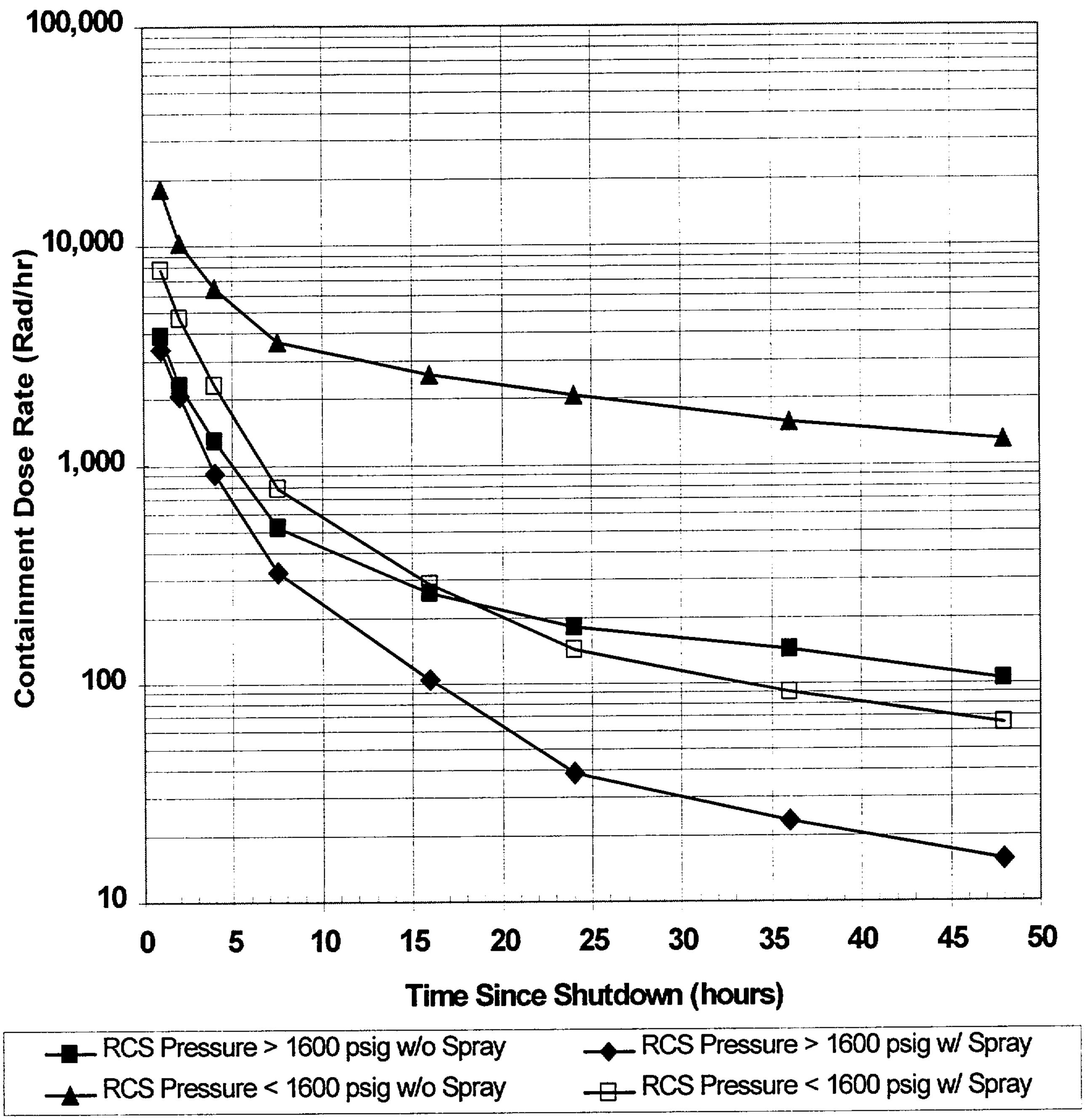
FIGURE 2: <CRM2>
"CONTAINMENT RADIATION LEVEL VS. TIME FOR 1% FUEL PELLET OVER-TEMPERATURE RELEASE"
(Adapted from Figure 2 of Reference 1)



ATTACHMENT 5 (Con't)
Page 3 of 5

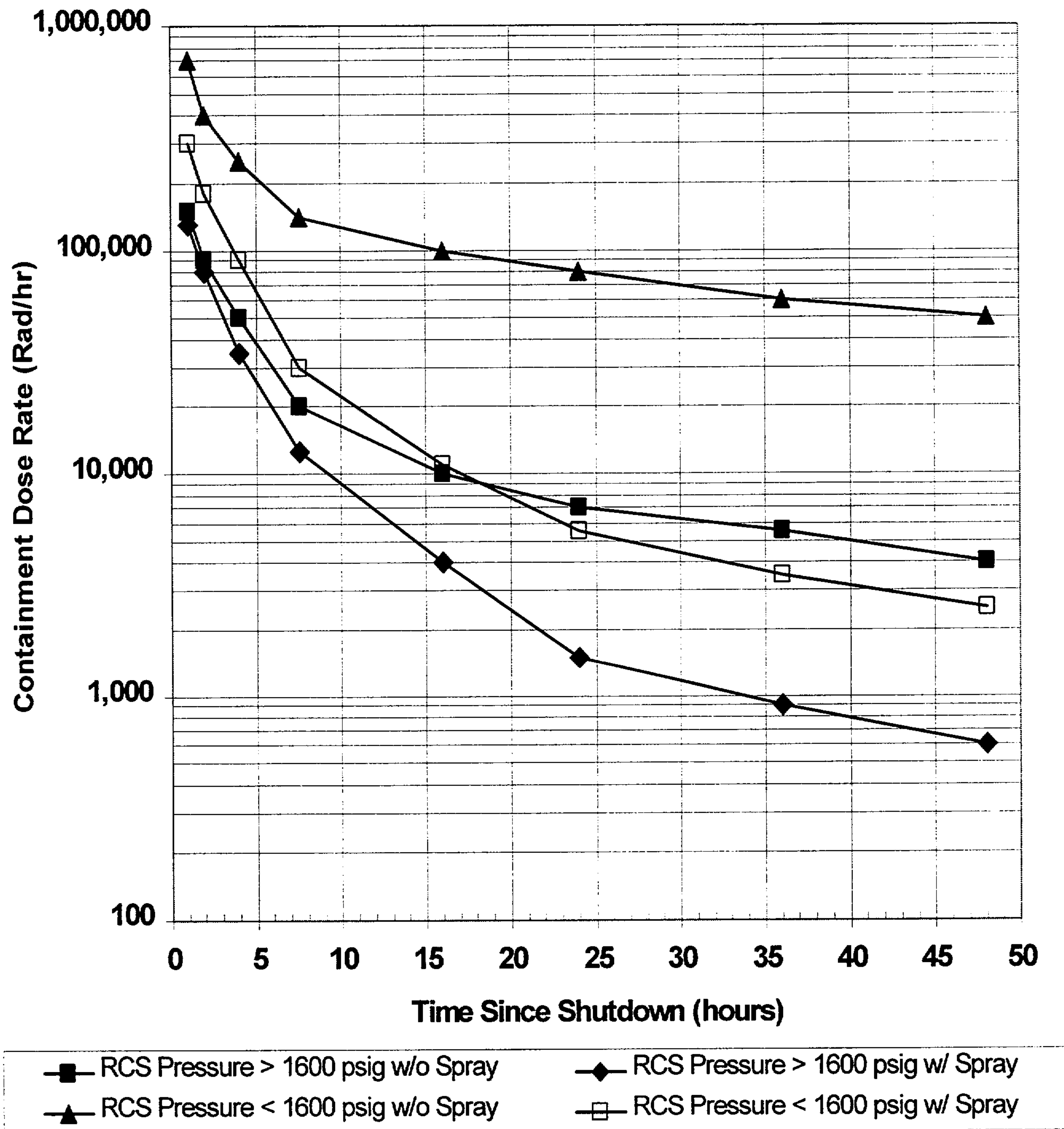
FIGURES FOR SECTION 5.1

FIGURE 3: <CRM3>
"CONTAINMENT RADIATION LEVEL VS. TIME FOR 100% CLAD DAMAGE RELEASE"
(Adapted from Figure 3 of Reference 1)



FIGURES FOR SECTION 5.1

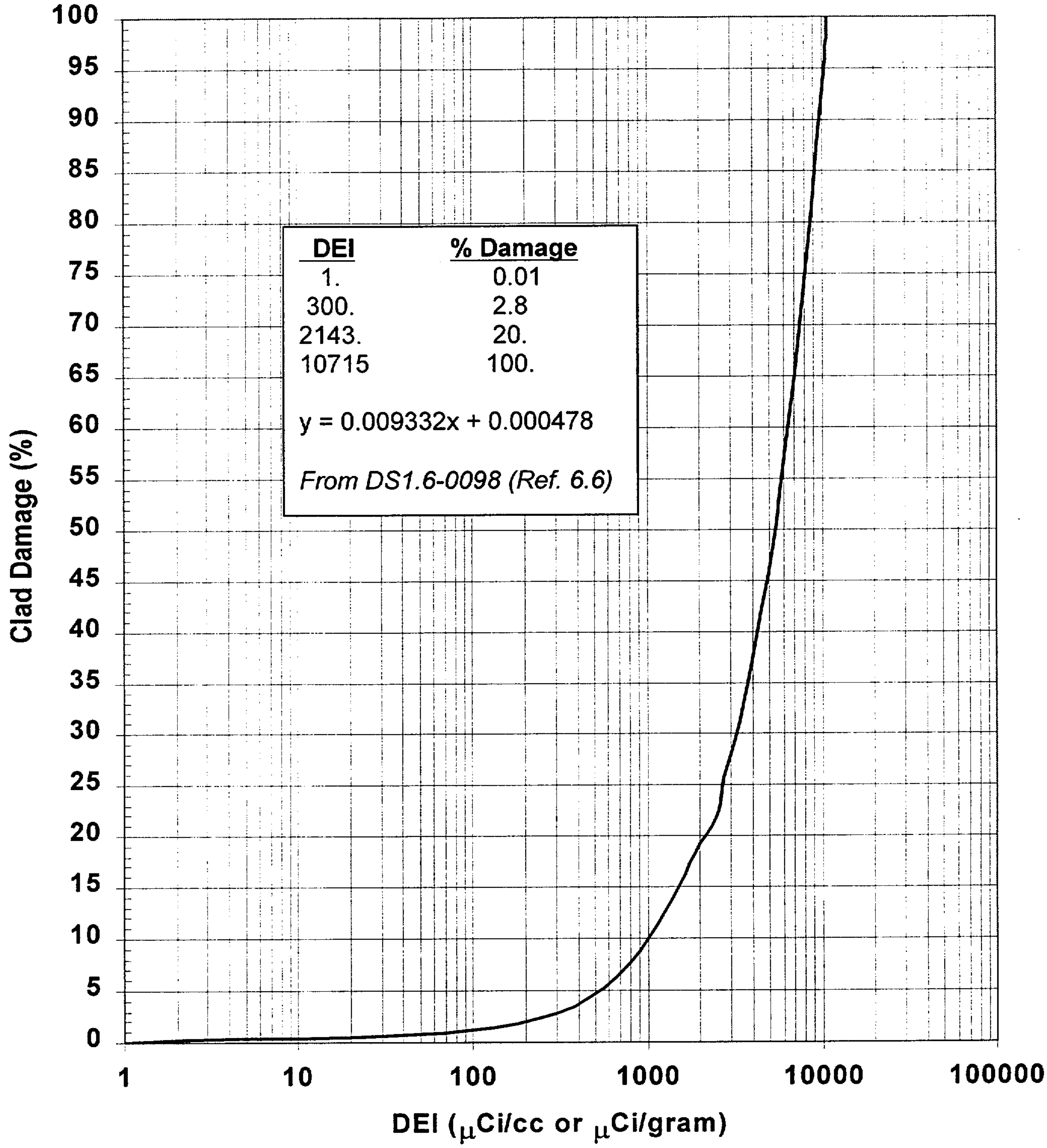
FIGURE 4: <CRM4>
"CONTAINMENT RADIATION LEVEL VS. TIME FOR 100% FUEL OVER-TEMPERATURE RELEASE"
(Adapted from Figure 4 of Reference 1)



ATTACHMENT 5 (Con't)
Page 5 of 5

FIGURES FOR SECTION 5.1

FIGURE 5: "CLAD DAMAGE VS. DOSE EQUIVALENT IODINE (DEI)"



ATTACHMENT 6
Page 1 of 1

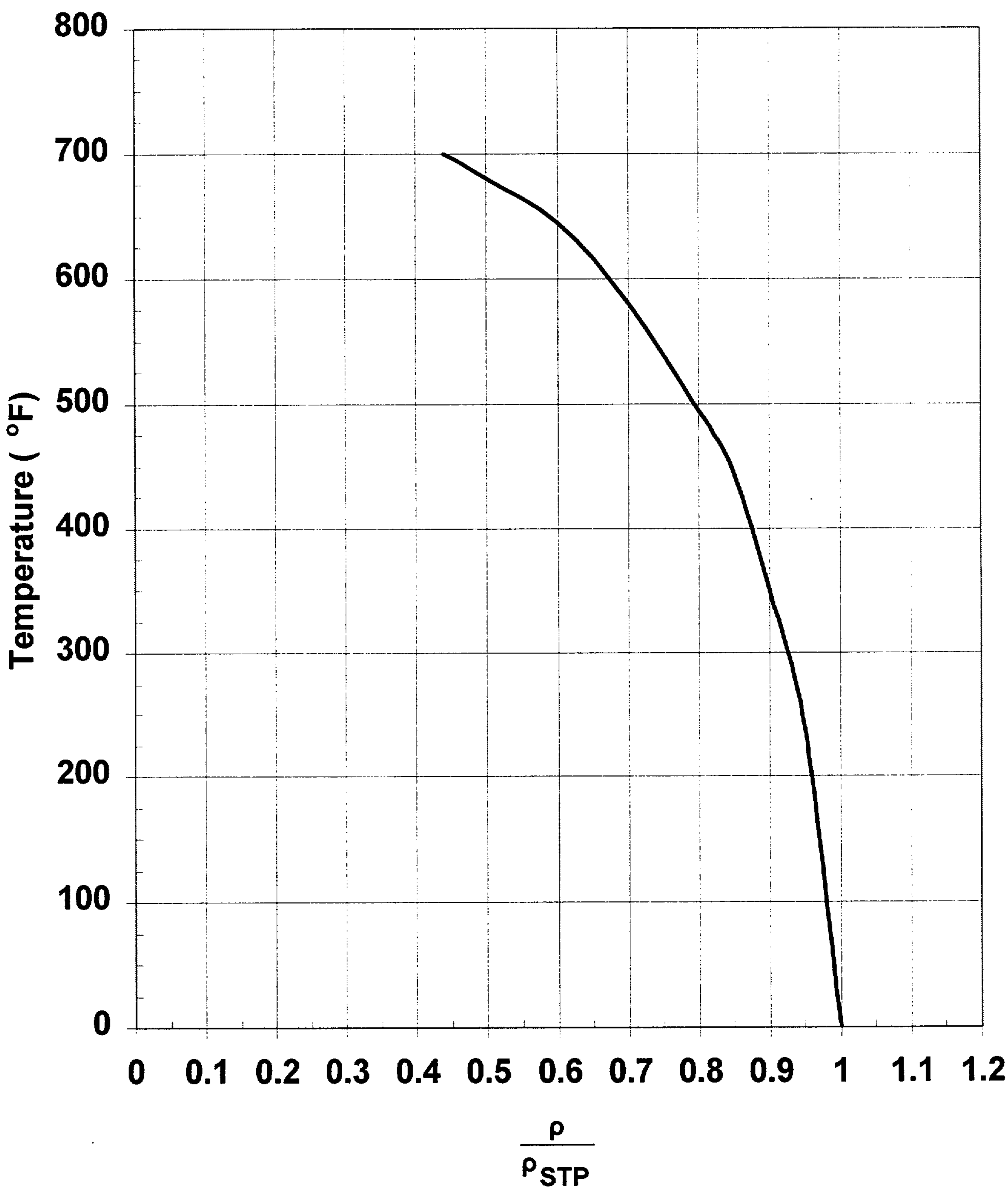
CORE DAMAGE ASSESSMENT TRENDING

Time	Estimated Clad Damage	Estimated Fuel Over-Temperature Damage

ATTACHMENT 7
Page 1 of 1

WATER DENSITY RATIO (TEMPERATURE VS. $\frac{\rho}{\rho_{STP}}$)

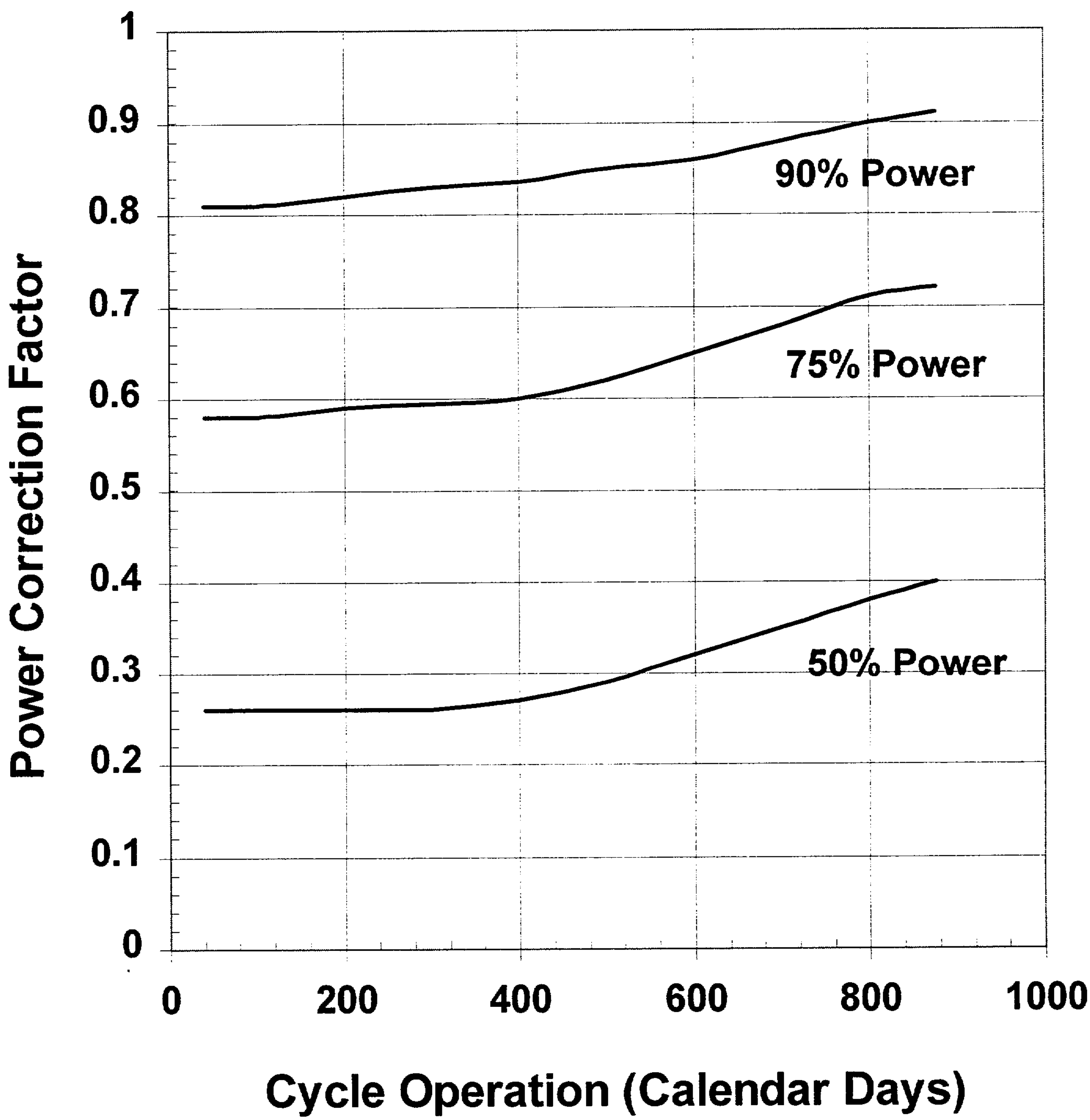
(Figure 2-10 of Reference 2)



ATTACHMENT 8

Page 1 of 1

POWER CORRECTION FACTOR FOR CS-134 BASED ON AVERAGE POWER DURING OPERATION
(Figure 2-1 of Reference 2)



DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

As previously stated in Step 5.2.5.1, the Chemistry Department normally delivers the isotopic data decay-corrected as standard practice. If the isotopic data is not decay-corrected, two methods are used to accomplish this. **See Section 2.4.5.3 "Decay Correction" (Reference 2) for a more detailed discussion of this methodology.**

Method 1

For the following isotopes, parent-daughter relationships need not be accounted for.

I-131	Kr-85m	Cs-134	Sr-89
I-133	Kr-87	Cs-137	Sr-90
I-135	Kr-88	Te-132	Ba-140

Use the following equation and table to calculate the decay-corrected specific activities for each sample and list them in the appropriate "Decay-Corrected Activities" page at the end of this attachment.

Note: 1 cc = 1 ml

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

where:

A = measured specific activity, (μCi/gm) or (μCi/cc)
λ_i = decay constant of isotope i, (sec^{-1})
t = time elapsed from reactor shutdown to time of sampling (sec)
A_0 = specific activity at shutdown, (μCi/gm) or (μCi/cc)

Constants Required for Method 1

ISOTOPE	$T_{1/2}$	λ (sec^{-1})
I-131	8.05 d	9.966E-7
I-133	20.3 h	9.485E-6
I-135	6.68 h	2.882E-5
Kr-85m	4.4 h	4.376E-5
Kr-87	76 m	1.520E-4
Kr-88	2.8 h	6.876E-5
Cs-134	2 y	1.098E-8
Cs-137	30 y	7.322E-10
Te-132	77.7 h	2.478E-6
Sr-89	52.7 d	1.522E-7
Sr-90	28 y	7.845E-10
Ba-140	12.8 d	6.268E-7

(Adapted from Table 2-1 of Reference 2)

DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Method 2

For the following isotopes, parent-daughter relationships do need to be taken into consideration.

I-132	Xe-133	Pr-144
Xe-131m	Xe-135	
Xe-133m	Te-129	

Use the following equations and tables to calculate the decay-corrected specific activities for each sample and list them in the appropriate "Decay-Corrected Activities" page at the end of this attachment.

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

$$Q_B(t) = k\frac{\lambda_B}{\lambda_B - \lambda_A} \cdot Q_A^0 (e^{-\lambda_A t} - e^{-\lambda_B t}) + Q_B^0 e^{-\lambda_B t}$$

Where,
$Q_A{}^0$ = 100% source inventory (Ci) of parent (Ref. 2, Table 2-2 or 2-8)
$Q_B{}^0$ = 100% source inventory (Ci) of daughter (Ref. 2, Table 2-2 or 2-8)
$Q_B(t)$ = hypothetical daughter activity (Ci) at sample time
k = if parent has 2 daughters, k is the branching factor (Ref. 2, Table 2-7)
λ_A = parent decay constant (sec^{-1})
λ_B = daughter decay constant (sec^{-1})
t = time period from shutdown to time of sample (sec)

2. Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time.

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

Where:
Fr = fraction of daughter activity at sample time due to the decay of only the daughter initial activity

DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

3. Calculate the amount of the measured sample specific activity associated with the decay of the daughter that was released.

$$M_B = M \times Fr$$

Where:
M = measured specific activity (μCi/gm) or (μCi/cc)
M_B = parent-daughter compensated activity (μCi/gm) or (μCi/cc)

4. Decay correct the specific activity to reactor shutdown and list it in the appropriate "Decay-Corrected Activities" page at the end of this attachment.

$$M_B^0 = \frac{M_B}{e^{-\lambda_B t}}$$

Where,
M_B^0 = decay corrected specific activity

Constants Required for Method 2:

Daughter	$T_{1/2}$	λ (sec^{-1})	Parent	$T_{1/2}$ (sec)	λ (sec^{-1})
I-132	2.26 h	8.520E-05	Te-132	77.7 h	2.478E-06
Xe-131m	11.8 d	6.799E-07	I-131	8.05 d	9.966E-07
Xe-133m	2.26 d	3.550E-06	I-133	20.3 h	9.485E-06
Xe-133	5.27 d	1.522E-06	I-133	20.3 h	9.485E-06
Xe-135	9.14 h	2.107E-05	I-135	6.68 h	2.882E-05
Te-129	68.7 m	1.682E-04	Sb-129	4.3 h	4.478E-05
Pr-144	17.27 m	6.689E-04	Ce-144	284 d	2.825E-08

(Adapted from Table 2-7 of Reference 2)

Daughter	Parent	100% Daughter Inventory Q_B^0 (Ci)	100% Parent Inventory Q_A^0 (Ci)	K
I-132	Te-132	1.4E8	1.4E8	1.0
Xe-131m	I-131	6.3E5	9.8E7	0.008
Xe-133m	I-133	2.8E7	2.0E8	0.024
Xe-133	I-133	2.0E8	2.0E8	0.976
Xe-135	I-135	3.7E7	1.8E8	0.700
Te-129	Sb-129	3.3E7	3.2E7	0.827
Pr-144	Ce-144	1.2E8	1.1E8	1.0

(Adapted from Tables 2-2 , 2-7, and 2-8 of Reference 2)

ATTACHMENT 9 (Con't.)
Page 4 of 6

DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Decay Corrected Activities – RCS Sample

I-131 = ____________________

I-132 = ____________________

I-133 = ____________________

I-135 = ____________________

Kr-85m = ____________________

Kr-87 = ____________________

Kr-88 = ____________________

Xe-131m = ____________________

Xe-133m = ____________________

Xe-133 = ____________________

Xe-135 = ____________________

Cs-134 = ____________________

Cs-137 = ____________________

Te-129 = ____________________

Te-132 = ____________________

Sr-89 = ____________________

Sr-90 = ____________________

Ba-140 = ____________________

Pr-144 = ____________________

Additional Isotopes For Core Damage Assessment

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

ATTACHMENT 9 (Con't.)
Page 5 of 6

DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Decay Corrected Activities – Sump Sample

I-131 = ______________________
I-132 = ______________________
I-133 = ______________________
I-135 = ______________________
Kr-85m = ______________________
Kr-87 = ______________________
Kr-88 = ______________________
Xe-131m = ______________________
Xe-133m = ______________________
Xe-133 = ______________________
Xe-135 = ______________________
Cs-134 = ______________________
Cs-137 = ______________________
Te-129 = ______________________
Te-132 = ______________________
Sr-89 = ______________________
Sr-90 = ______________________
Ba-140 = ______________________
Pr-144 = ______________________

Additional Isotopes For Core Damage Assessment

= ______________________
= ______________________
= ______________________
= ______________________
= ______________________
= ______________________
= ______________________

DECAY-CORRECTION OF ISOTOPIC DATA FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Decay Corrected Activities – Con't Atmosphere Sample

I-131 = ______________________

I-132 = ______________________

I-133 = ______________________

I-135 = ______________________

Kr-85m = ______________________

Kr-87 = ______________________

Kr-88 = ______________________

Xe-131m = ______________________

Xe-133m = ______________________

Xe-133 = ______________________

Xe-135 = ______________________

Cs-134 = ______________________

Cs-137 = ______________________

Te-129 = ______________________

Te-132 = ______________________

Sr-89 = ______________________

Sr-90 = ______________________

Ba-140 = ______________________

Pr-144 = ______________________

Additional Isotopes For Core Damage Assessment

= ______________________

= ______________________

= ______________________

= ______________________

= ______________________

= ______________________

ATTACHMENT 10

Page 1 of 4

PRESSURE AND TEMPERATURE-CORRECTION FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

RCS and Sump Samples

Use the following equation to pressure- and temperature-correct the activities of the RCS or sump sample and list them in the appropriate "Pressure/Temperature/Decay-Corrected Activities" page at the end of this attachment. **See Section 2.4.5.2 "Pressure and Temperature Adjustment" (Reference 2) for a more detailed discussion of this methodology.**

$$A_{PTC} = A_0 \times \frac{\rho}{\rho_{STP}}$$

Where,

A_0 = decay-corrected specific activity

A_{PTC} = decay-pressure-temperature corrected specific activity

$\frac{\rho}{\rho_{STP}}$ = water density ratio at RCS or sump temperature taken from Attachment 6.

Containment Atmosphere Sample

Use the following equation to pressure- and temperature-correct the activities of the containment atmosphere sample and list them in the appropriate "Pressure/Temperature Decay-Corrected Activities" page at the end of this attachment.

$$A_{PTC} = A_0 \times \frac{P_2}{P_1} \times \left(\frac{T_1 + 460}{T_2 + 460}\right)$$

Where,

T_1, P_1 = measured sample temperature (°F) and pressure (psia)

T_2, P_2 = containment atmosphere temperature (°F) and pressure (psia)

A_0 = decay-corrected specific activity

A_{PTC} = decay-pressure-temperature corrected specific activity

PRESSURE AND TEMPERATURE-CORRECTION FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Pressure/Temperature/Decay Corrected Activities (RCS Sample)

I-131 = ____________________

I-132 = ____________________

I-133 = ____________________

I-135 = ____________________

Kr-85m = ____________________

Kr-87 = ____________________

Kr-88 = ____________________

Xe-131m = ____________________

Xe-133m = ____________________

Xe-133 = ____________________

Xe-135 = ____________________

Cs-134 = ____________________

Cs-137 = ____________________

Te-129 = ____________________

Te-132 = ____________________

Sr-89 = ____________________

Sr-90 = ____________________

Ba-140 = ____________________

Pr-144 = ____________________

Additional Isotopes For Core Damage Assessment

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

PRESSURE AND TEMPERATURE-CORRECTION FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Pressure/Temperature/Decay Corrected Activities (Sump Sample)

I-131 = ______________________

I-132 = ______________________

I-133 = ______________________

I-135 = ______________________

Kr-85m = ______________________

Kr-87 = ______________________

Kr-88 = ______________________

Xe-131m = ______________________

Xe-133m = ______________________

Xe-133 = ______________________

Xe-135 = ______________________

Cs-134 = ______________________

Cs-137 = ______________________

Te-129 = ______________________

Te-132 = ______________________

Sr-89 = ______________________

Sr-90 = ______________________

Ba-140 = ______________________

Pr-144 = ______________________

Additional Isotopes For Core Damage Assessment

= ______________________

= ______________________

= ______________________

= ______________________

= ______________________

= ______________________

PRESSURE AND TEMPERATURE-CORRECTION FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Pressure/Temperature/Decay Corrected Activities
(Containment Atmosphere Sample)

I-131 = ____________________

I-132 = ____________________

I-133 = ____________________

I-135 = ____________________

Kr-85m = ____________________

Kr-87 = ____________________

Kr-88 = ____________________

Xe-131m = ____________________

Xe-133m = ____________________

Xe-133 = ____________________

Xe-135 = ____________________

Cs-134 = ____________________

Cs-137 = ____________________

Te-129 = ____________________

Te-132 = ____________________

Sr-89 = ____________________

Sr-90 = ____________________

Ba-140 = ____________________

Pr-144 = ____________________

Additional Isotopes For Core Damage Assessment

= ____________________

= ____________________

= ____________________

= ____________________

= ____________________

ATTACHMENT 11
Page 1 of 3

TOTAL ACTIVITY RELEASED FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

RCS Sample

Isotope	Corrected Specific Activity (µCi/g)		RCS Mass (grams)		Total RCS Activity (µCi)
I-131		x		=	
I-132		x		=	
I-133		x		=	
I-135		x		=	
Kr-85m		x		=	
Kr-87		x		=	
Kr-88		x		=	
Xe-131m		x		=	
Xe-133m		x		=	
Xe-133		x		=	
Xe-135		x		=	
Cs-134		x		=	
Cs-137		x		=	
Te-129		x		=	
Te-132		x		=	
Sr-89		x		=	
Sr-90		x		=	
Ba-140		x		=	
Pr-144		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	

ATTACHMENT 11 (Con't.)
Page 2 of 3

TOTAL ACTIVITY RELEASED
FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Sump Sample

Isotope	Corrected Specific Activity (µCi/g)		Sump Mass (grams)		Total Sump Activity (µCi)
I-131		x		=	
I-132		x		=	
I-133		x		=	
I-135		x		=	
Kr-85m		x		=	
Kr-87		x		=	
Kr-88		x		=	
Xe-131m		x		=	
Xe-133m		x		=	
Xe-133		x		=	
Xe-135		x		=	
Cs-134		x		=	
Cs-137		x		=	
Te-129		x		=	
Te-132		x		=	
Sr-89		x		=	
Sr-90		x		=	
Ba-140		x		=	
Pr-144		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	

ATTACHMENT 11 (Con't.)
Page 3 of 3

TOTAL ACTIVITY RELEASED FOR RCS, SUMP, AND CONTAINMENT ATMOSPHERE SAMPLES

Containment Atmosphere Sample

Isotope	Corrected Specific Activity (µCi/cc)		Sample Medium Mass or Volume (cc's)		Total Con't. Atmos. Activity (µCi)
I-131		x		=	
I-132		x		=	
I-133		x		=	
I-135		x		=	
Kr-85m		x		=	
Kr-87		x		=	
Kr-88		x		=	
Xe-131m		x		=	
Xe-133m		x		=	
Xe-133		x		=	
Xe-135		x		=	
Cs-134		x		=	
Cs-137		x		=	
Te-129		x		=	
Te-132		x		=	
Sr-89		x		=	
Sr-90		x		=	
Ba-140		x		=	
Pr-144		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	

ATTACHMENT 12
Page 1 of 1

TOTAL ACTIVITY RELEASED FOR ALL SAMPLES COMBINED

Isotope	Total RCS Activity (µCi)		Total Sump Activity (µCi)		Total Con't Atmos. (µCi)		Total Activity (µCi)
I-131		+		+		=	
I-132		+		+		=	
I-133		+		+		=	
I-135		+		+		=	
Kr-85m		+		+		=	
Kr-87		+		+		=	
Kr-88		+		+		=	
Xe-131m		+		+		=	
Xe-133m		+		+		=	
Xe-133		+		+		=	
Xe-135		+		+		=	
Cs-134		+		+		=	
Cs-137		+		+		=	
Te-129		+		+		=	
Te-132		+		+		=	
Sr-89		+		+		=	
Sr-90		+		+		=	
Ba-140		+		+		=	
Pr-144		+		+		=	
		+		+		=	
		+		+		=	
		+		+		=	
		+		+		=	
		+		+		=	
		+		+		=	
		+		+		=	

ATTACHMENT 13
Page 1 of 2

POWER CORRECTIONS FACTORS FOR:
KR-85M, KR-87, KR-88, XE-135, I-132, I-133, I-135, TE-129, PR-144

A. Has prior 4 days power varied more than ± 10% Rated Thermal Power?

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\text{Average Power For Prior 4 Days (\%)}}{100}$$

where,
PCF = Power Correction Factor

YES: Go to Part B.

B. Is the total operating history greater than 4 times the half life of the isotope of interest? (See table of half-lives and decay constants at end of this attachment).

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^0}}{100\left(1 - e^{-\lambda \sum t_j}\right)}$$

where,

t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)

P_j = Average power (%) during operating period (t_j)

j = The number of times that power has varied more than ± 10% in the previous 4 days

λ = Decay constant ($days^{-1}$)

t_j^0 = Time between end of period "j" and time of reactor shutdown in days

PCF = Power Correction Factor

POWER CORRECTIONS FACTORS FOR:
KR-85M, KR-87, KR-88, XE-135, I-132, I-133, I-135, TE-129, PR-144

<u>YES</u>: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^0}}{100(1.0)}$$

where,

t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)

P_j = Average power (%) during operating period (t_j)

j = The number of times that power has varied more than ± 10% in the previous 4 days

λ = Decay constant ($days^{-1}$)

t_j^0 = Time between end of period "j" and time of reactor shutdown in days

PCF = Power Correction Factor

Half-Lives and Decay Constants for Isotopes

Isotope	Half-Life (Days)	Decay Constant ($Days^{-1}$)
I-131	8.05	8.61E-2
I-132	9.42E-2	7.36
I-133	8.46E-1	8.19E-1
I-135	2.78E-1	2.49
Kr-85m	1.83E-1	3.79
Kr-87	5.28E-2	1.31E1
Kr-88	1.17E-1	5.92
Xe-131m	1.18E1	5.87E-2
Xe-133m	2.26	3.07E-1
Xe-133	5.27	1.32E-1
Xe-135	3.81E-1	1.82
Cs-134	7.30E2	9.50E-4
Cs-137	1.10E4	6.30E-5
Te-129	4.77E-2	1.45E1
Te-132	3.24	2.14E-1
Sr-89	5.27E1	1.32E-2
Sr-90	1.02E4	6.80E-5
Ba-140	1.28E1	5.42E-2
Pr-144	1.20E-2	5.78E1

ATTACHMENT 14
Page 1 of 2

POWER CORRECTIONS FACTORS FOR:
XE-131M, XE-133, XE-133M, I-131, TE-132, BA-140

A. Has prior 30 days power varied more than ± 10% Rated Thermal Power?

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\text{Average Power For Prior 30 Days (\%)}}{100}$$

where,
PCF = Power Correction Factor

YES: Go to Part B.

B. Is the total operating history greater than 4 times the half life of the isotope of interest? (See table of half-lives and decay constants at end of this attachment).

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^0}}{100\left(1 - e^{-\lambda \sum t_j}\right)}$$

where,

t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)

P_j = Average power (%) during operating period (t_j)

j = The number of times that power has varied more than ± 10% in the previous 30 days

λ = Decay constant ($days^{-1}$)

t_j^0 = Time between end of period "j" and time of reactor shutdown in days

PCF = Power Correction Factor

ATTACHMENT 14 (Con't.)
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POWER CORRECTIONS FACTORS FOR: XE-131M, XE-133, XE-133M, I-131, TE-132, BA-140

YES: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^0}}{100(1.0)}$$

where,

t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)

P_j = Average power (%) during operating period (t_j)

j = The number of times that power has varied more than ± 10% in the previous 30 days

λ = Decay constant ($days^{-1}$)

t_j^0 = Time between end of period "j" and time of reactor shutdown in days

PCF = Power Correction Factor

Half-Lives and Decay Constants for Isotopes

Isotope	Half-Life (Days)	Decay Constant ($Days^{-1}$)
I-131	8.05	8.61E-2
I-132	9.42E-2	7.36
I-133	8.46E-1	8.19E-1
I-135	2.78E-1	2.49
Kr-85m	1.83E-1	3.79
Kr-87	5.28E-2	1.31E1
Kr-88	1.17E-1	5.92
Xe-131m	1.18E1	5.87E-2
Xe-133m	2.26	3.07E-1
Xe-133	5.27	1.32E-1
Xe-135	3.81E-1	1.82
Cs-134	7.30E2	9.50E-4
Cs-137	1.10E4	6.30E-5
Te-129	4.77E-2	1.45E1
Te-132	3.24	2.14E-1
Sr-89	5.27E1	1.32E-2
Sr-90	1.02E4	6.80E-5
Ba-140	1.28E1	5.42E-2
Pr-144	1.20E-2	5.78E1

ATTACHMENT 15

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**POWER CORRECTIONS FACTORS FOR:
SR-90, CS-134, CS-137**

A. Is Cs-134 the isotope of interest?

<u>NO</u>: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\text{Actual Number of "Effective Full Power Days" in Cycle To Date}}{\text{Expected Number of "Effective Full Power Days" in Cycle To Date}}$$

where,
PCF = Power Correction Factor

<u>YES</u>: Use Attachment 8 to obtain the Power Correction Factor and record it in Attachment 17.

ATTACHMENT 16
Page 1 of 2

POWER CORRECTIONS FACTORS FOR:
SR-89

A. Has prior 1 year power varied more than ± 10% Rated Thermal Power?

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\text{Average Power For Prior 100 Days (\%)}}{100}$$

where,
PCF = Power Correction Factor

YES: Go to Part B.

B. Is the total operating history greater than 4 times the half life of the isotope of interest? (See table of half-lives and decay constants at end of this attachment).

NO: Calculate the power correction factor using the following equation and record it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j \left(1 - e^{-\lambda t_j}\right) e^{-\lambda t_j^o}}{100\left(1 - e^{-\lambda \sum t_j}\right)}$$

where,
t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)
P_j = Average power (%) during operating period (t_j)
j = The number of times that power has varied more than ± 10% in the previous 1 year
λ = Decay constant (days^{-1})
t_j^o = Time between end of period "j" and time of reactor shutdown in days
PCF = Power Correction Factor

ATTACHMENT 16
Page 2 of 2

POWER CORRECTIONS FACTORS FOR: SR-89

YES: Calculate the power correction factor using the following equation and list it in Attachment 17.

$$PCF = \frac{\sum_{1}^{j} P_j\left(1-e^{-\lambda t_j}\right)e^{-\lambda t_j^0}}{100(1.0)}$$

where,

t_j = Operating period in days at power (P_j) where power does not vary more than ± 10% from the time averaged power (P_j)

P_j = Average power (%) during operating period (t_j)

j = The number of times that power has varied more than ± 10% in the previous 1 year

λ = Decay constant ($days^{-1}$)

t_j^0 = Time between end of period "j" and time of reactor shutdown in days

PCF = Power Correction Factor

Half-Lives and Decay Constants for Isotopes

Isotope	Half-Life (Days)	Decay Constant (Days^{-1})
I-131	8.05	8.61E-2
I-132	9.42E-2	7.36
I-133	8.46E-1	8.19E-1
I-135	2.78E-1	2.49
Kr-85m	1.83E-1	3.79
Kr-87	5.28E-2	1.31E1
Kr-88	1.17E-1	5.92
Xe-131m	1.18E1	5.87E-2
Xe-133m	2.26	3.07E-1
Xe-133	5.27	1.32E-1
Xe-135	3.81E-1	1.82
Cs-134	7.30E2	9.50E-4
Cs-137	1.10E4	6.30E-5
Te-129	4.77E-2	1.45E1
Te-132	3.24	2.14E-1
Sr-89	5.27E1	1.32E-2
Sr-90	1.02E4	6.80E-5
Ba-140	1.28E1	5.42E-2
Pr-144	1.20E-2	5.78E1

ATTACHMENT 17
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ISOTOPIC INVENTORY AVAILABLE FOR RELEASE

Isotope	Source Inventory Table 2-2, Ref. 2 (Ci)		Power Correction Factor		Inventory Available For Release (Ci)
I-131	9.8E7	x		=	
I-132	1.4E8	x		=	
I-133	2.0E8	x		=	
I-135	1.8E8	x		=	
Kr-85m	2.2E7	x		=	
Kr-87	4.0E7	x		=	
Kr-88	5.7E7	x		=	
Xe-131m	6.3E5	x		=	
Xe-133m	2.8E7	x		=	
Xe-133	2.0E8	x		=	
Xe-135	3.7E7	x		=	
Cs-134	2.3E7	x		=	
Cs-137	1.1E7	x		=	
Te-129	3.3E7	x		=	
Te-132	1.4E8	x		=	
Sr-89	7.9E7	x		=	
Sr-90	7.2E6	x		=	
Ba-140	1.7E8	x		=	
Pr-144	1.2E8	x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	

ATTACHMENT 18
Page 1 of 1

PERCENT INVENTORY RELEASED FOR EACH ISOTOPE

A. Copy the "Total Activity" from Attachment 11 (for one sample medium) or Attachment 12 (for several sample media) and the "Inventory Available for Release" from Attachment 17 into the table below.

B. Calculate the "Percent Inventory Released" using the following equation and list it in the table below.

$$\% \text{ Inventory Released} = \frac{\text{Total Activity} \times \left(1\text{Ci} / 1\text{E6}\ \mu\text{Ci} \right)}{\text{Inventory Avaliable For Release}} \times 100$$

Isotope	Total Activity (µCi) (Attachment 11/12)	Inventory Available For Release (Ci) (Attachment 17)	% Inventory Released
I-131			
I-132			
I-133			
I-135			
Kr-85m			
Kr-87			
Kr-88			
Xe-131m			
Xe-133m			
Xe-133			
Xe-135			
Cs-134			
Cs-137			
Te-129			
Te-132			
Sr-89			
Sr-90			
Ba-140			
Pr-144			

ATTACHMENT 19
Page 1 of 2

CORE DAMAGE ASSESSMENT (BASED ON INVENTORY)

Take each percent inventory released from Attachment 18 and compare it to the appropriate figures in Attachment 20. Record the extent and type of damage in the following table. The figures are taken from Reference 2.

Clad Damage

Isotope	Percent Inventory Released	Estimated Clad Damage
I-131		
I-132		
I-133		
I-135		
Xe-131m		
Xe-133		
Kr-87		

Fuel Pellet Over-temperature Estimates

Isotope	Percent Inventory Released	Estimated Clad Damage
I-131		
I-132		
I-133		
I-135		
Xe-131m		
Xe-133		
Xe-133m		
Xe-135		
Kr-85m		
Kr-87		
Kr-88		
Cs-134		
Cs-137		
Te-129		
Te-132		

ATTACHMENT 19
Page 2 of 2

CORE DAMAGE ASSESSMENT (BASED ON INVENTORY)

Fuel Pellet Over-temperature Estimates

Isotope	Percent Inventory Released	Estimated Clad Damage
Ba-140		
Sr-89		
Sr-90		

Fuel Melt Estimates

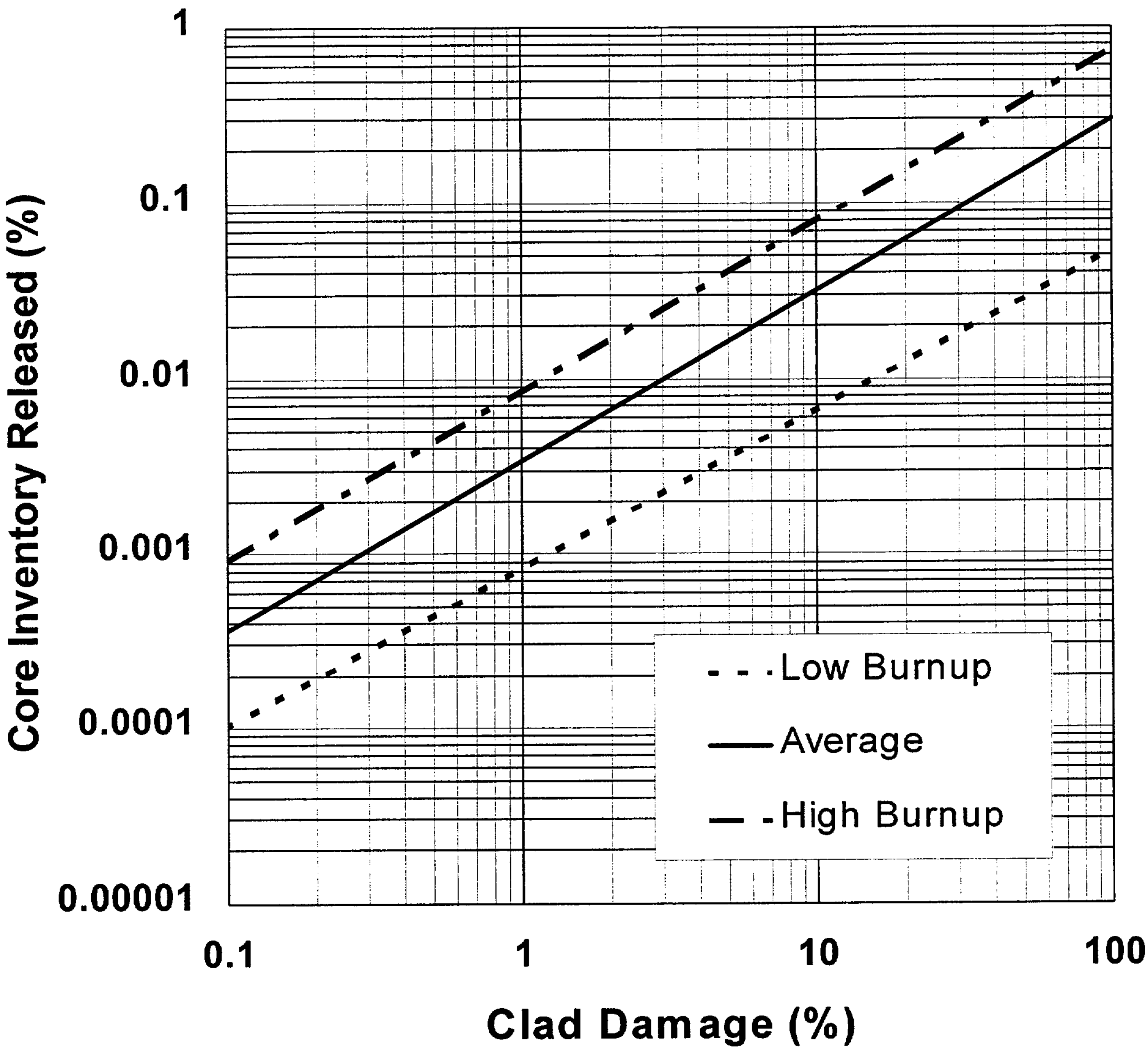
Isotope	Percent Inventory Released	Estimated Clad Damage
I-131		
I-132		
I-133		
I-135		
Xe-131m		
Xe-133		
Xe-133m		
Xe-135		
Kr-85m		
Kr-87		
Kr-88		
Cs-134		
Cs-137		
Te-129		
Te-132		
Ba-140		
Sr-89		
Sr-90		
Pr-144		

ATTACHMENT 20
Page 1 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

I-131

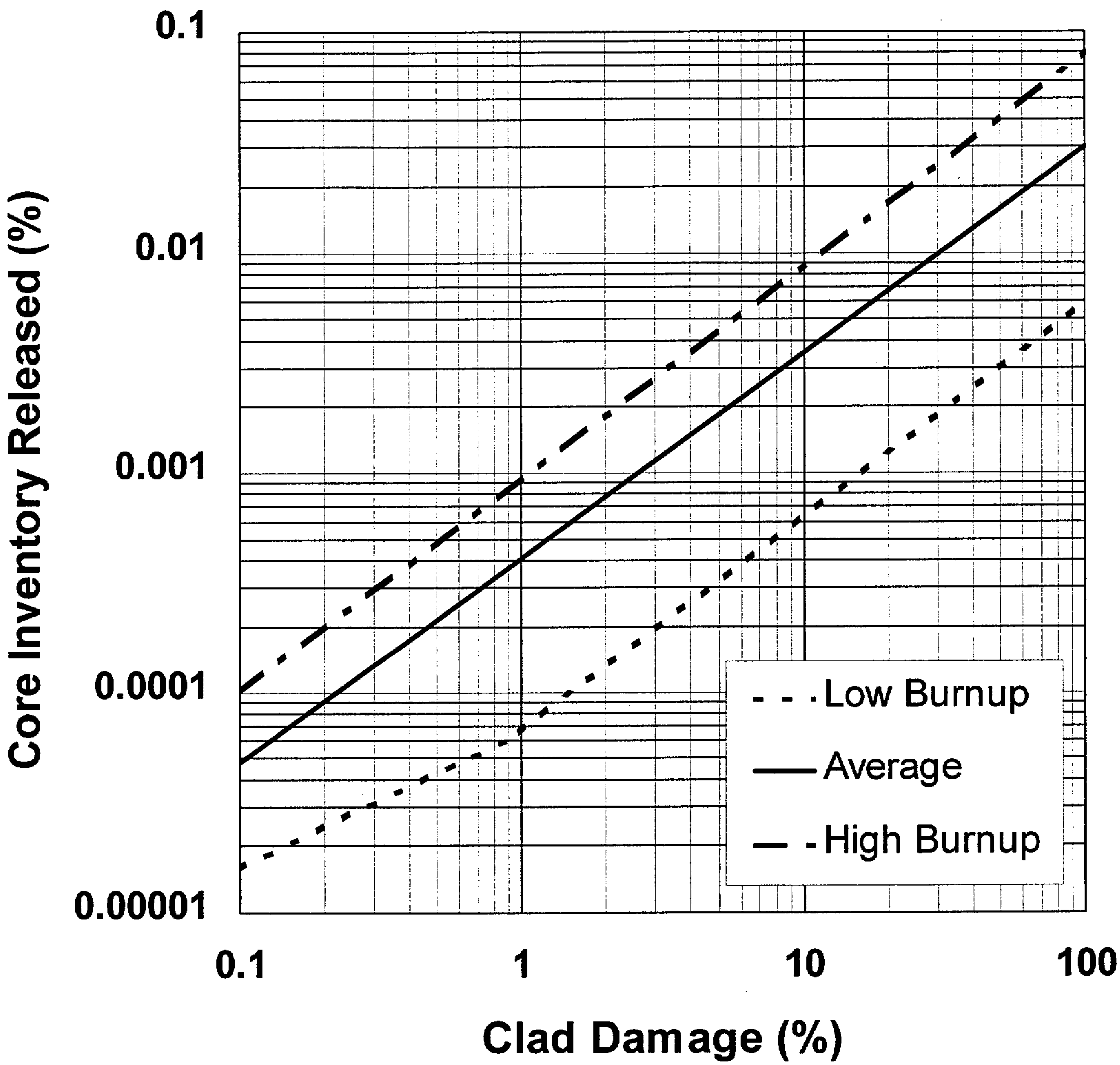


ATTACHMENT 20 (Con't)
Page 2 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

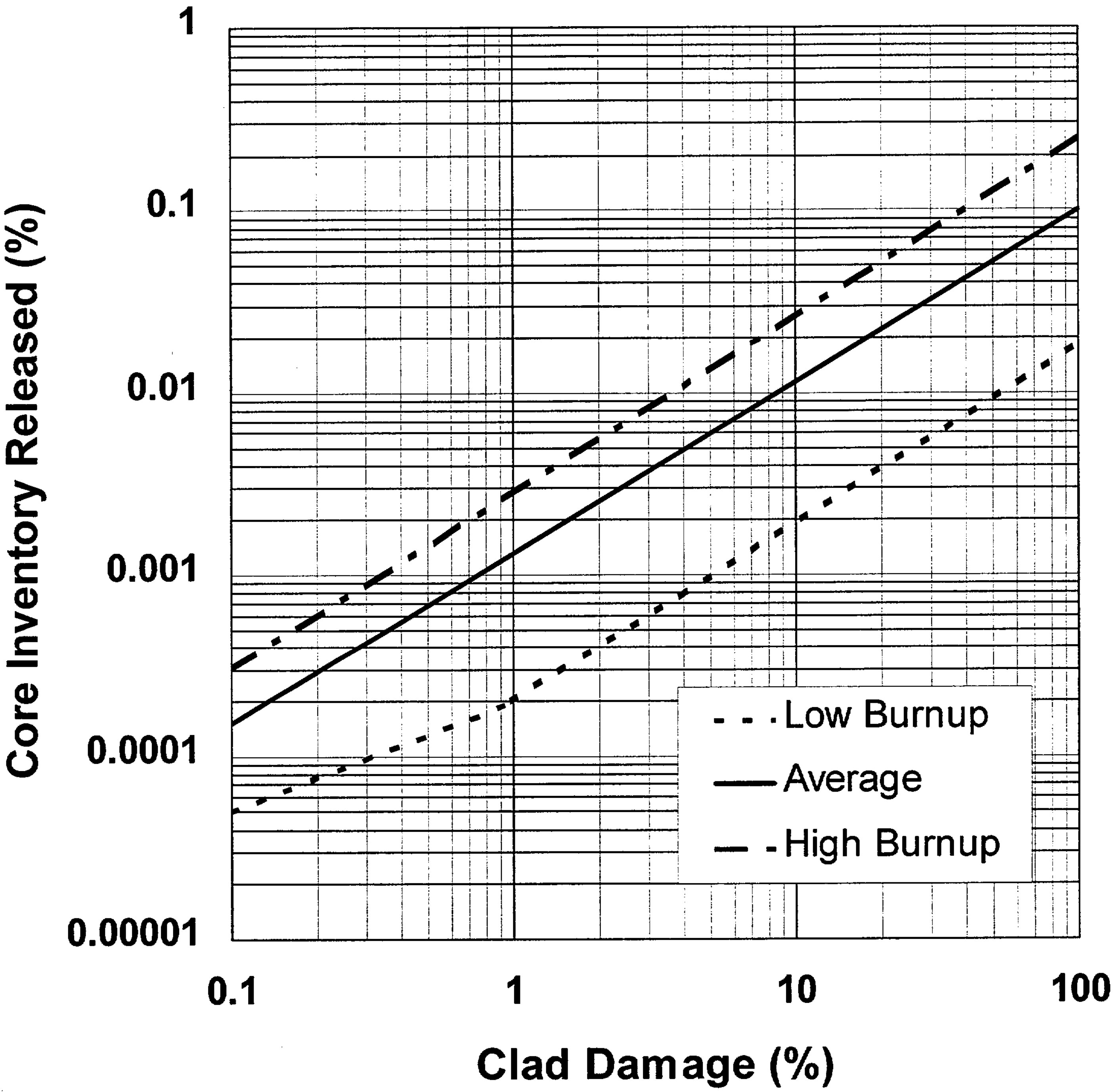
I-132



FIGURES USED IN SECTION 5.2

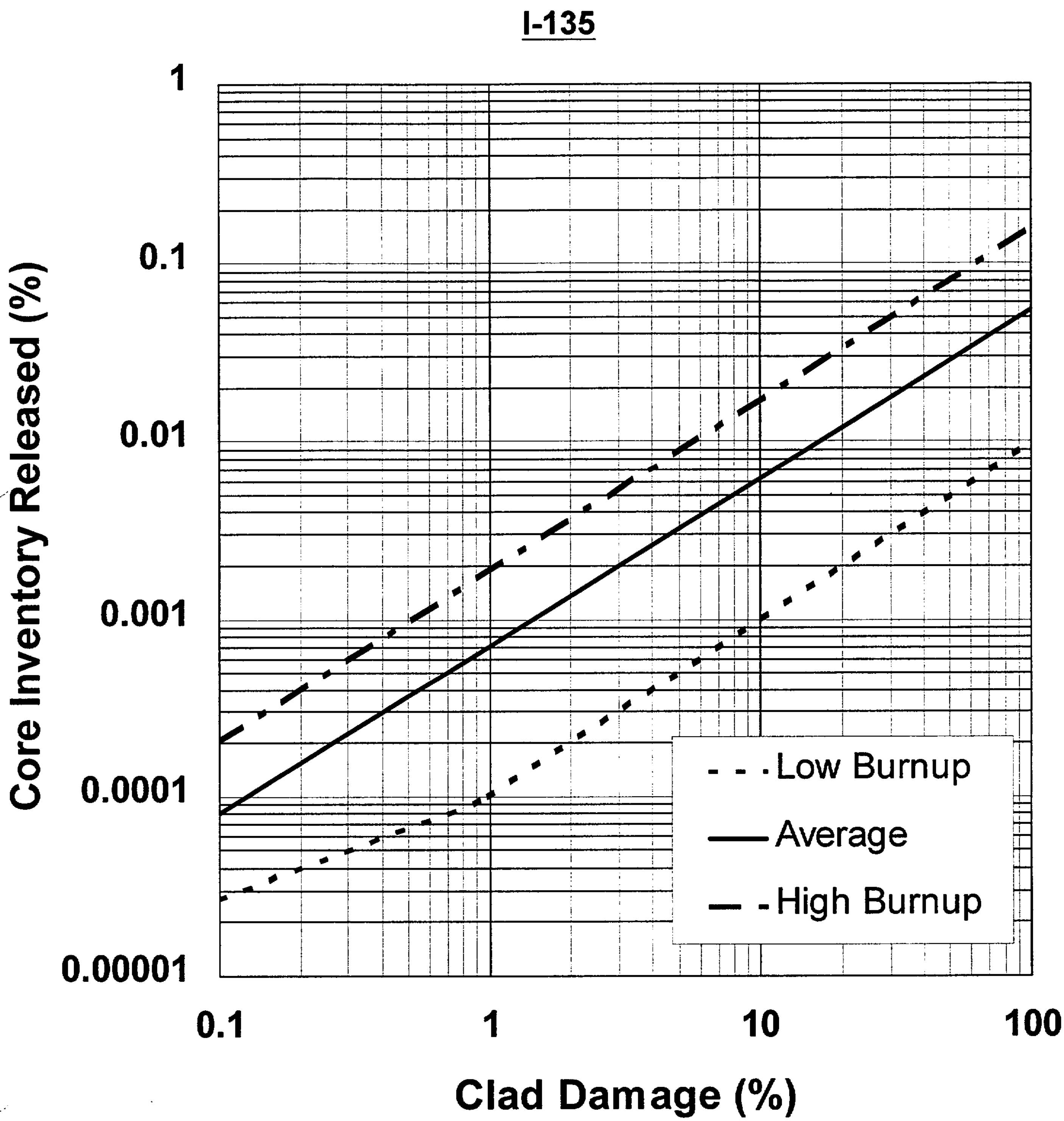
ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

I-133



FIGURES USED IN SECTION 5.2

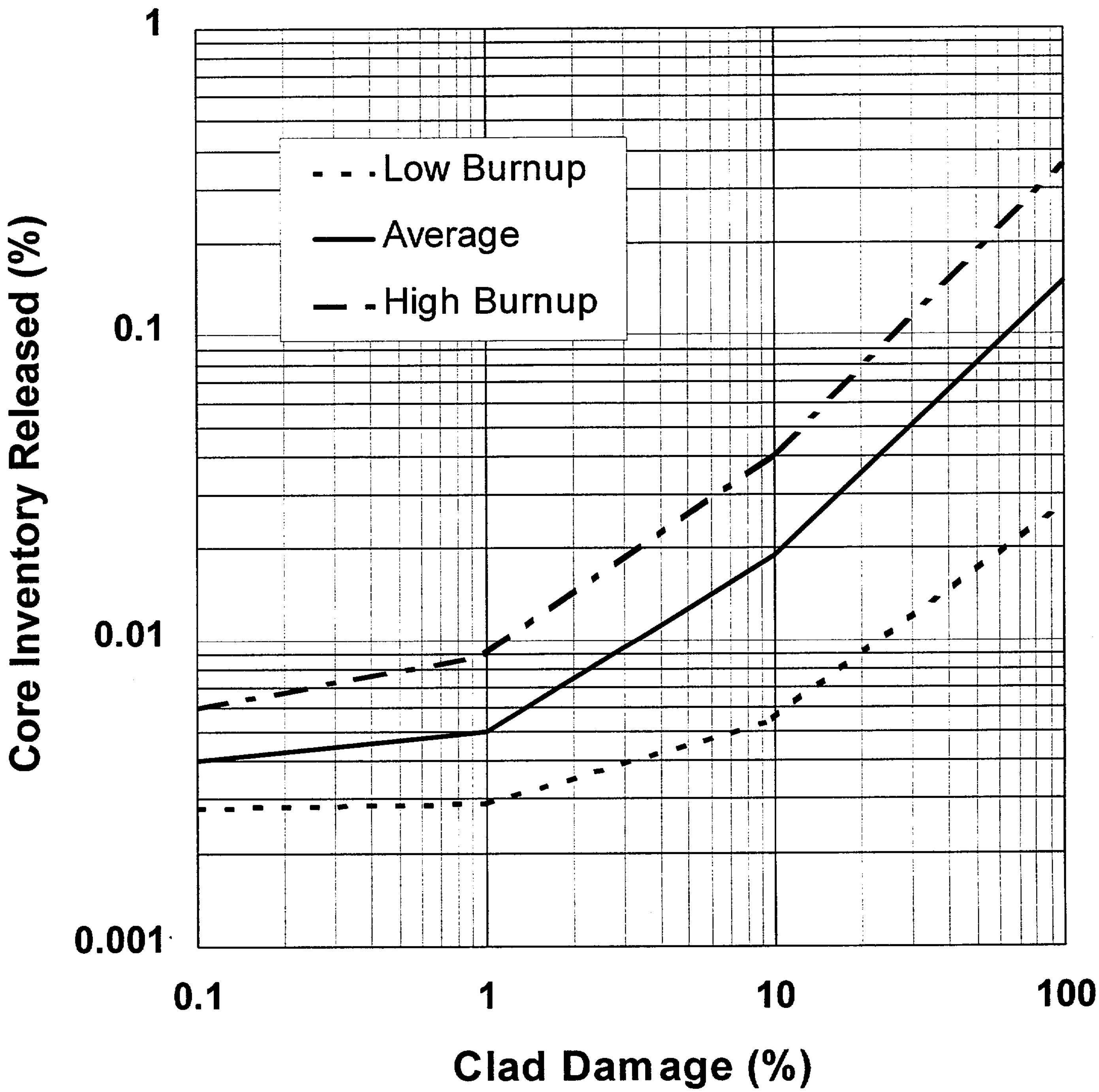
ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE



FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

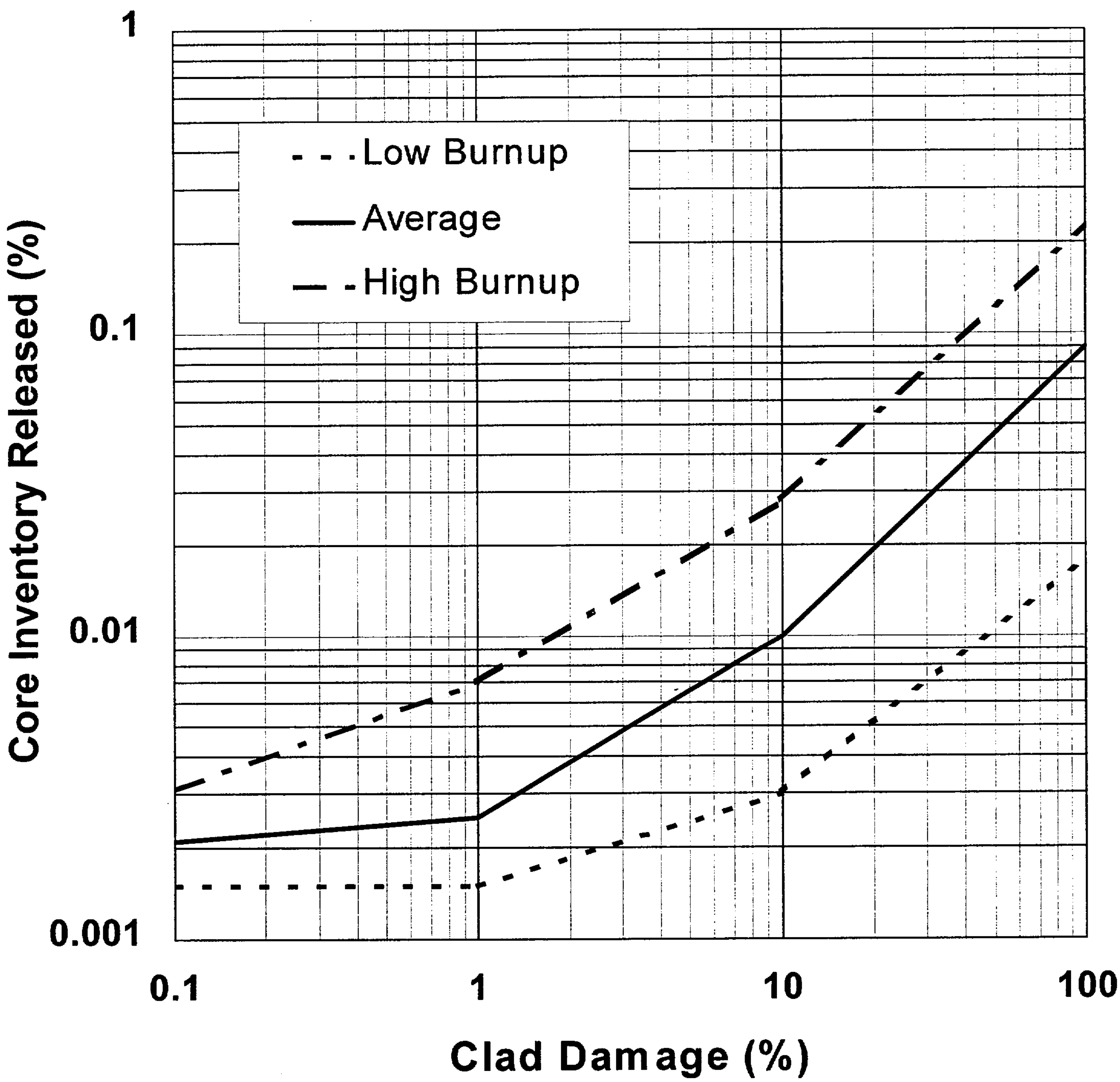
XE-131M



FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

XE-133

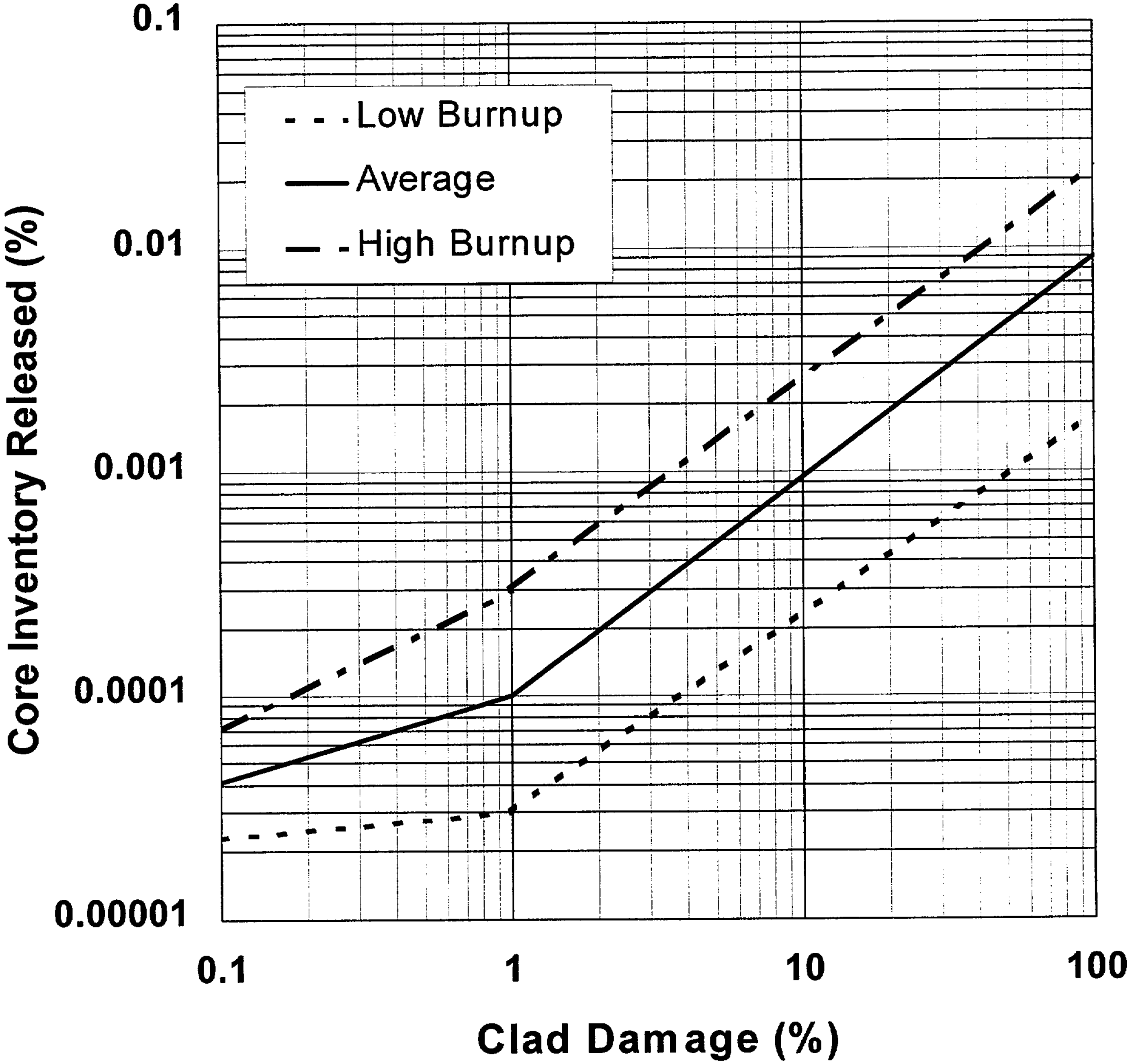


ATTACHMENT 20 (Con't)
Page 7 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF CLAD DAMAGE

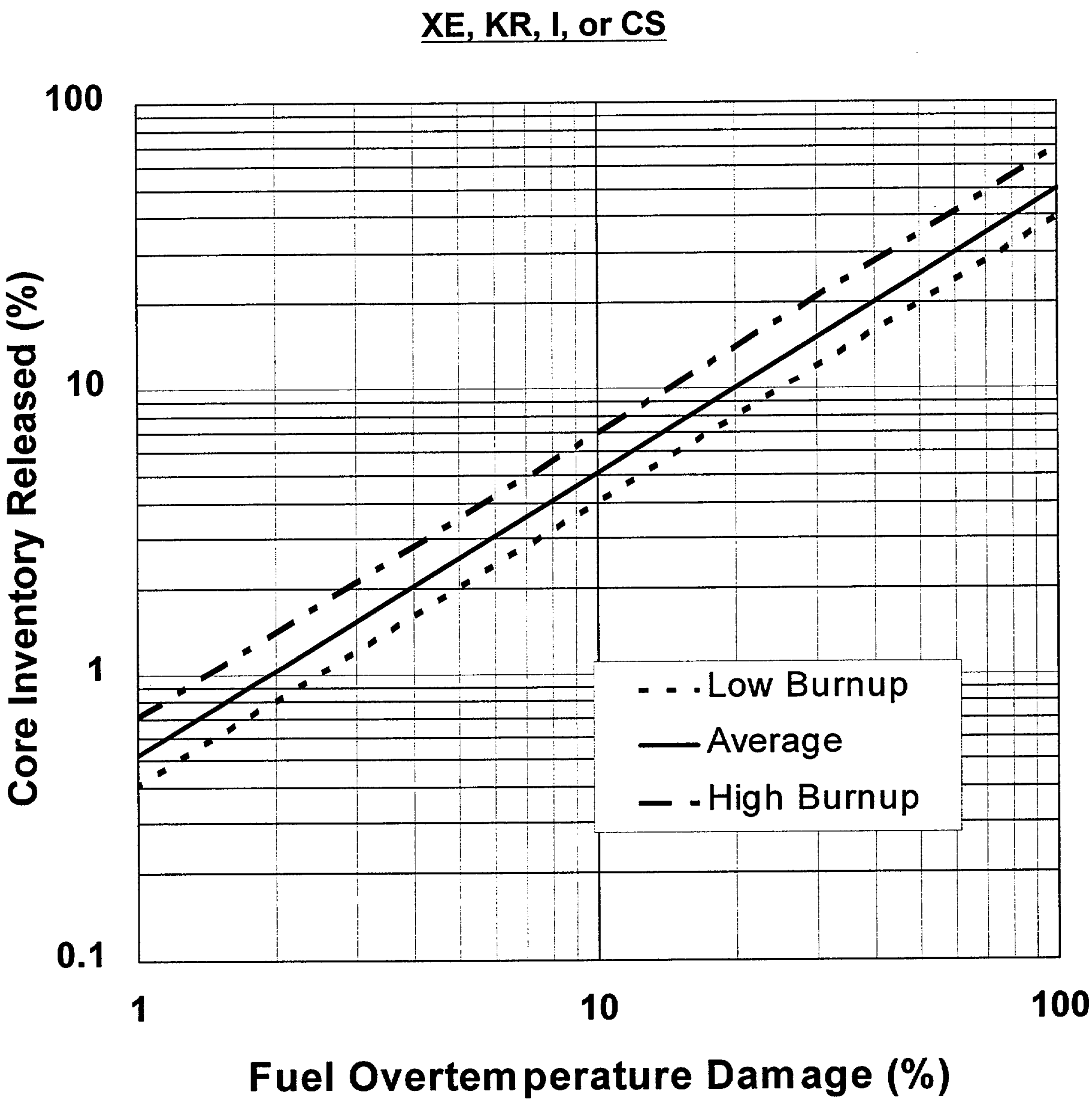
KR-87



ATTACHMENT 20 (Con't)
Page 8 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF FUEL PELLET OVER-TEMPERATURE DAMAGE

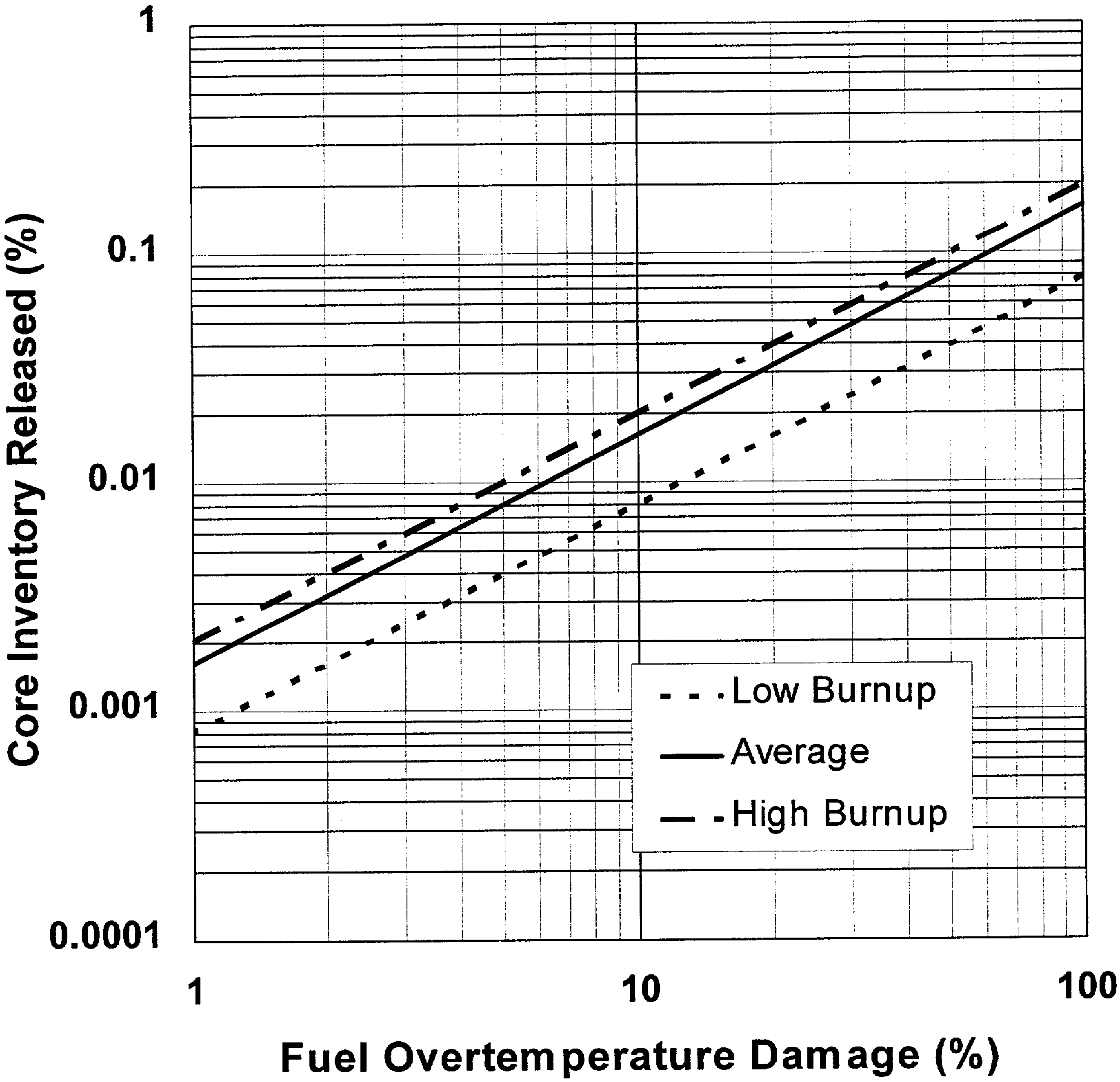


ATTACHMENT 20 (Con't)
Page 9 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF FUEL PELLET OVER-TEMPERATURE DAMAGE

BA or SR

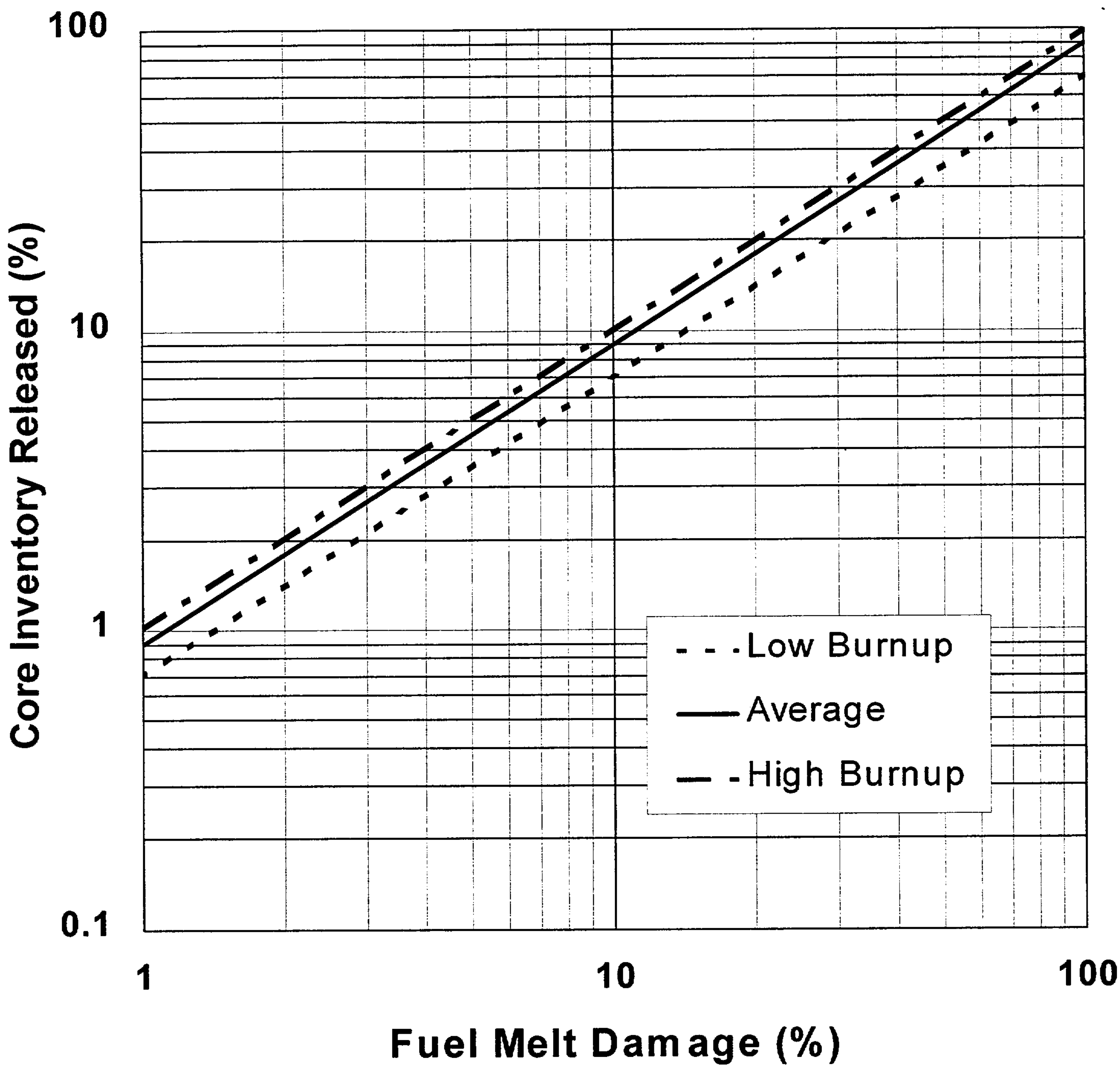


ATTACHMENT 20 (Con't)
Page 10 of 12

FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF FUEL MELT DAMAGE

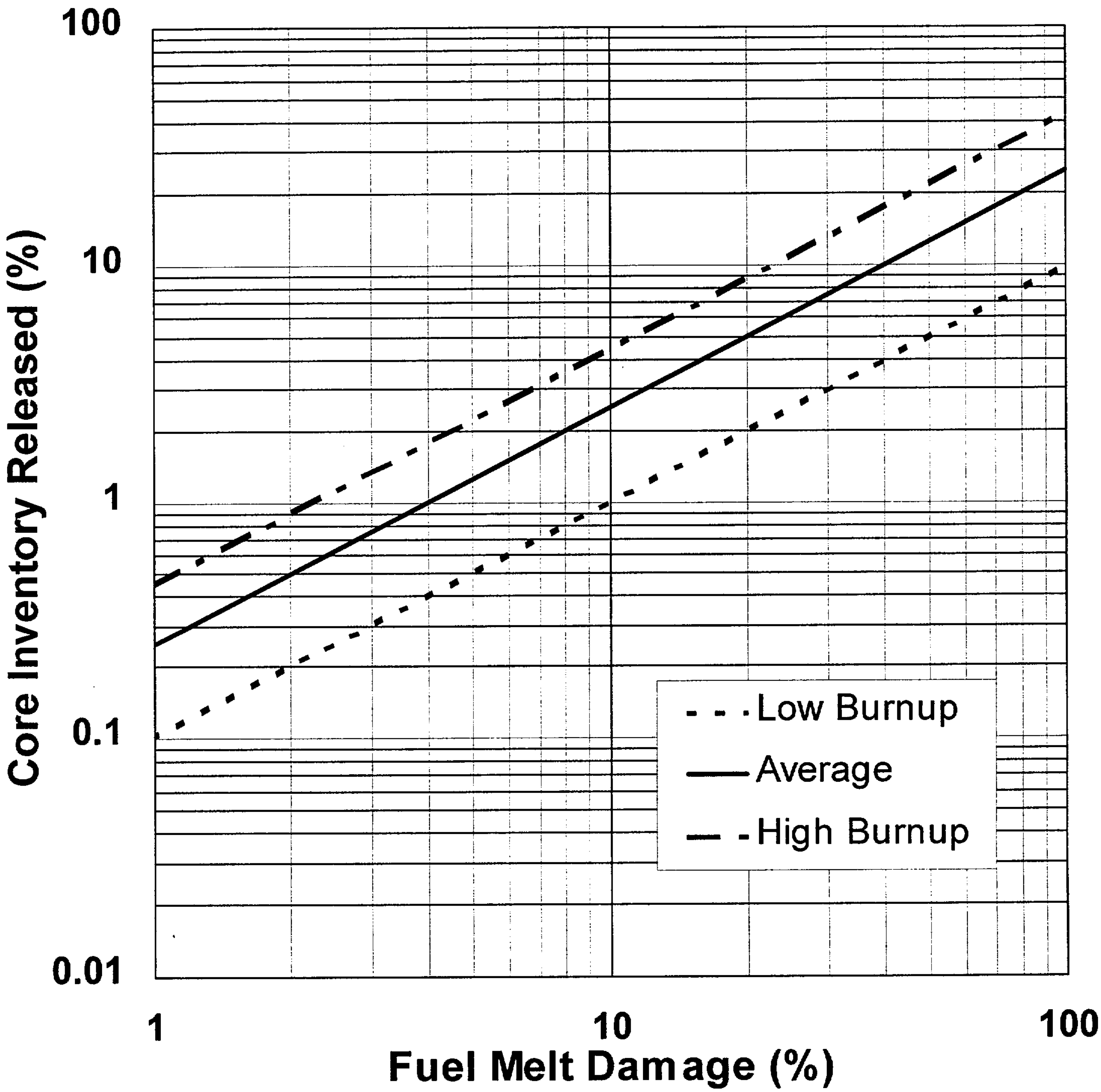
XE, KR, I, CS, or TE



FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF FUEL MELT DAMAGE

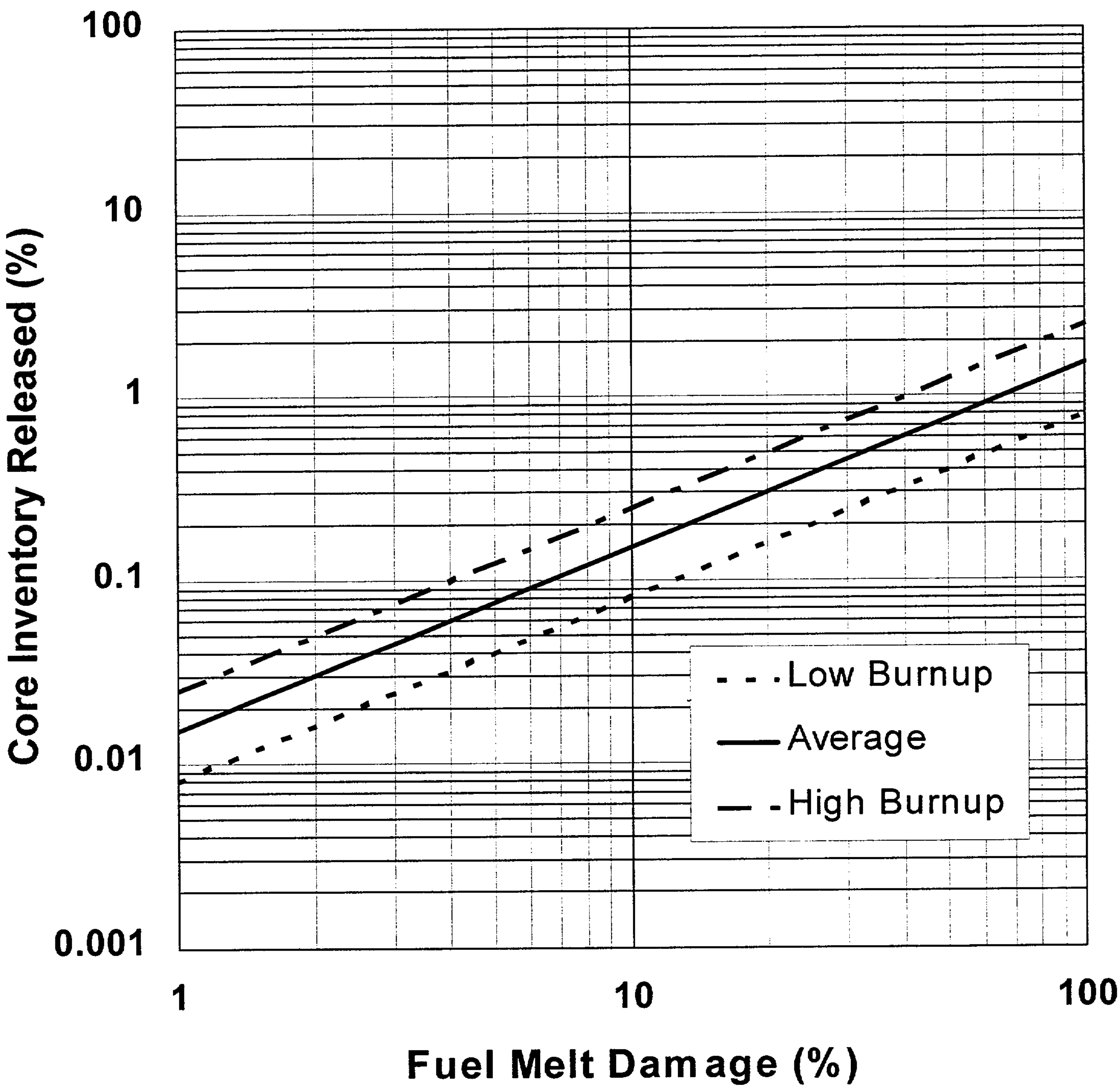
BA or SR



FIGURES USED IN SECTION 5.2

ISOTOPIC CONCENTRATION VS. INDICATION OF FUEL MELT DAMAGE

PR



ATTACHMENT 21
Page 1 of 1

ISOTOPIC ACTIVITY RATIOS OF FUEL PELLET AND GAP

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

***Adapted from "Isotopic Activity Ratios of Fuel Pellet and Gap" Table 2-6 of Reference 2**

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

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

If Calculated Ratio is:	Postulated Damage
Less Than the Gap Activity Ratio	None
Greater Than the Gap Activity Ratio and **Less Than** the Fuel Pellet Activity Ratio	Clad Damage
Greater Than the Fuel Pellet Activity Ratio	Fuel Pellet Over-temperature

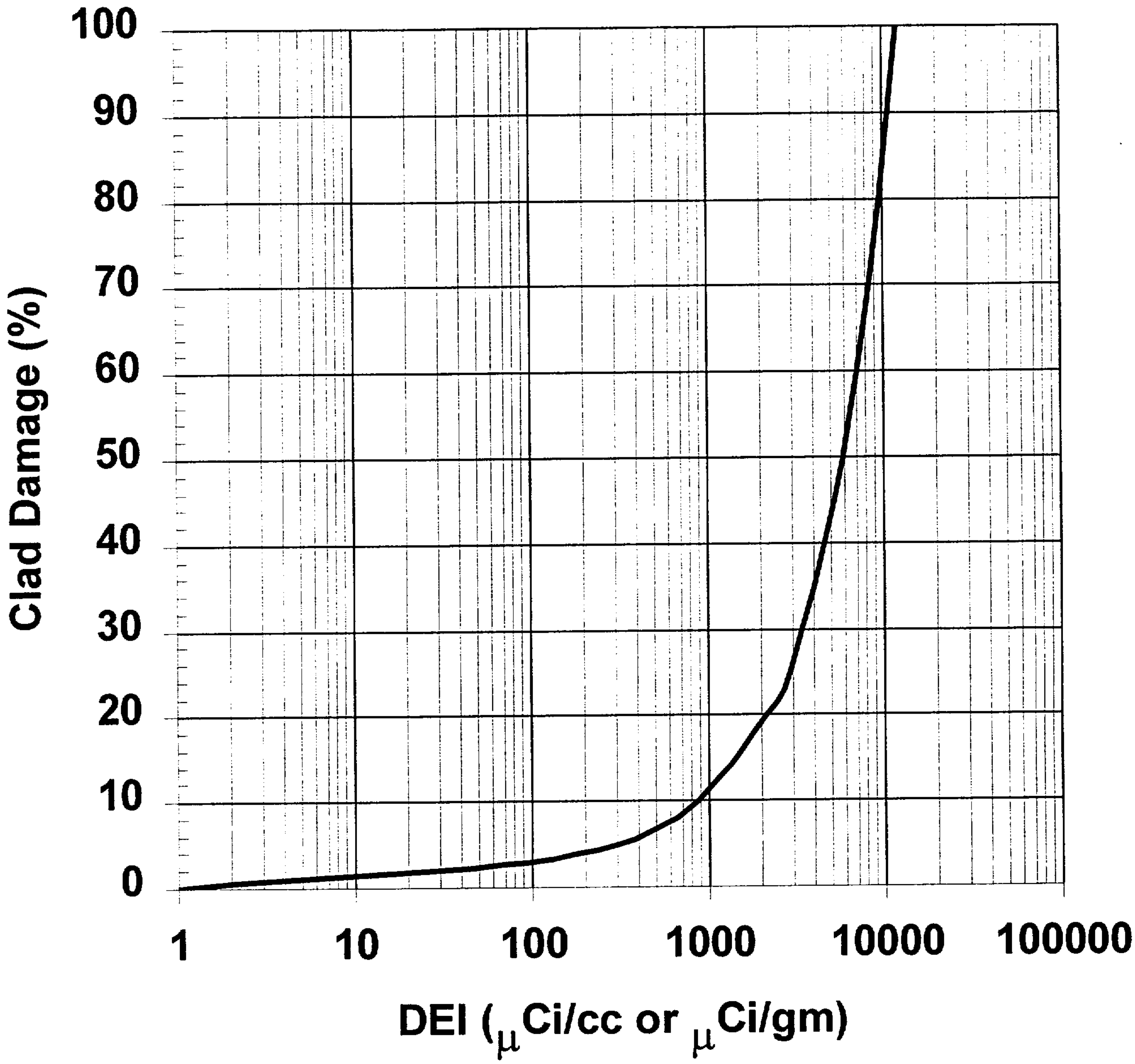
Note:

The use of these ratios for post accident damage assessment is restricted to an attempt to differentiate between fuel pellet over-temperature conditions and fuel cladding failure conditions only.

ATTACHMENT 22
Page 1 of 1

DOSE EQUIVALENT IODINE (DEI)

CLAD DAMAGE VS. DEI



Note:

This figure was created from data found in Reference 6. Due to the very rough correlation between DEI and Clad Damage, the relatively small difference between the DEI in units of μCi/cc and units of μCi/gm does not substantially impact the estimated clad damage.

ATTACHMENT 23

FINAL CORE DAMAGE ASSESSMENT BASED ON ISOTOPIC INVENTORIES AND NOBLE GAS & IODINE RATIOS