50-346

REPORT: P187R40

DOCUMENTATION MANAGEMENT
DOCUMENT ON-LINE CONTROL SYSTEM
TRANSMITTAL / RECEIPT ACKNOWLEDGMENT

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REPORT: P187R40

DOCUMENTATION MANAGEMENT
DOCUMENT ON-LINE CONTROL SYSTEM
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TRANSMITTAL NUMBER: 0004-00809
TRANSMITTAL DATE : 04/04/2000
TRANSMITTAL PAGE : 002

COPY NUMBER : 1665

	TYPE	DOCUMENT			STATUS
TOC EPIB			SHT/SEC:		
REMOVE:	TOC	EPIB	SHT/SEC:	REV/SUB: 0049	REVISED
INSERT:	TOC	EPIB	SHT/SEC:	REV/SUB: 0050	APPROVED
MANUAL: EPIB			SHT/SEC:		
PROC RA-EP-02320			SHT/SEC:		
REMOVE:	PROC	RA-EP-02320	SHT/SEC:	REV/SUB: 0001	APPROVED
	PC	C97-2100			APPROVED
	PC	C97-2879			APPROVED
	PC	C99-0015			APPROVED
INSERT:	PROC	RA-EP-02320	SHT/SEC:	REV/SUB: 0001	APPROVED
	PC	C97-2100			APPROVED
	PC	C97-2879			APPROVED
	PC	C99-0015			APPROVED
	PC	C99-2329			APPROVED

END OF TRANSMITTAL

MANUAL TABLE OF CONTENTS
DB EMERGENCY PLAN IMPLEMENTING PROCEDURES VOL B
MANUAL: EPIB REVISION: 50

PROCEDURE DEVELOPMENT FORM
ED 7635-14

SHEET	PROCEDURE ACTIVITY TRACKING NO.
1 OF 1	99-2329

SECTION 1 - IDENTITY

PROCEDURE NO. / CURRENT REVISION	PROPOSED REVISION NO.	PROPOSED CHANGE NO.			
RA-EP-02320 / R01	N/A	4	☐ NEW ☐ REVISION ☒ CHANGE ☐ TEMPORARY	☐ CANCELLATION ☐ INACTIVATION ☐ REACTIVATION APPROVAL	☐ RESTRICTED ESTIMATED EXPIRATION ----- (DATE / EVENT)
SUPERSEDES					

PROCEDURE TITLE
Emergency Technical Assessment

PROCEDURE CLASSIFICATION
☒ SR ☐ QR ☐ N-QR CHANGE TO? ☐ YES ☒ NO

PAT / PCR NOS. CLOSED OUT
None ☐ CONTINUED

ACTIVITY SUMMARY / PURPOSE
Incorporate changes required due MOD 93-0021.

☐ CONTINUED

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS
☐ CONTINUED

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST
None		

SECTION 3 - REVIEW ORGANIZATIONS

REQUIRED

None

☐ CONTINUED

REQUESTED

☐ CONTINUED

PROCEDURE PREPARER	DATE	QUALIFIED REVIEWER	DATE
David B. Kelley	3/2/2000	D R Blakely	3/2/00

SECTION 4 - ATTACHMENTS

COMPLETED AND ATTACHED

YES	N/A		YES	N/A		YES	N/A	
☐	☑	VALIDATION CHECKLIST	☑	☐	SAFETY REVIEW	☐	☑	REVISED CROSS REFERENCES LIST
☑	☐	COMMITMENT VERIFICATION SUMMARY	☐	☑	DOCUMENT REVIEW SHEETS	☐	☑	PCRs
☐	☑	SAFETY EVALUATION NO.				☐	☑	OTHER

FINAL QUALIFIED REVIEWER CONCURRENCE	DATE
D R Blakely D.R. Blakely	3/2/00

SECTION 5 - TEMPORARY APPROVAL

MANAGEMENT SRO	DATE	APPROVAL AUTHORITY	DATE
N/A		N/A	

SECTION 6 - CONCURRENCE / FINAL APPROVAL

	DATE		DATE
PROCEDURE SPONSOR J.L. Freels	3/3/00	☐ MANAGER - BUSINESS SERVICES (BSA) N/A	
☐ DIRECTOR – SUPPORT SERVICES (NA) N/A		☒ SRB	3/15/00
☐ DIRECTOR – TECHNICAL SERVICES (E&S) N/A		☒ PLANT MANAGER	3/15/00
☐ DIRECTOR – WORK MANAGEMENT (WMD) N/A		APPROVAL AUTHORITY S.P. Moffitt	3/20/00

SECTION 7 - TRAINING / PROCEDURE EFFECTIVITY

TRAINING COMPLETE	ALTERATION EFFECTIVE DATE	PROCEDURE SPONSOR	DATE
☐ YES ☒ N/A	4/4/00	James L. Freels / DHK	3-28-00

PROCEDURE DEVELOPMENT FORM
ED 7635-13

SHEET 1 OF 1

PROCEDURE ACTIVITY TRACKING NO. C99-0015

SECTION 1 - IDENTITY

PROCEDURE NO. / CURRENT REVISION	PROPOSED REVISION NO.	PROPOSED CHANGE NO.		
RA-EP-02320 R1	N/A	3	☐ NEW ☐ REVISION ☒ CHANGE ☐ TEMPORARY APPROVAL	☐ RESTRICTED
SUPERSEDES N/A			☐ CANCELLATION ☐ INACTIVATION ☐ REACTIVATION	EXPIRES ____ (DATE / EVENT)

PROCEDURE TITLE
Emergency Technical Assessment

PROCEDURE CLASSIFICATION
☒ SR ☐ QR ☐ N-QR ☐ CHANGE TO? ☐ YES ☒ NO

PAT / PCR NOS. CLOSED OUT
NONE ☐ CONTINUED

ACTIVITY SUMMARY / PURPOSE
To add guidance to help the TSC personnel assess LOCA's and the need for boron dilution.
This implements committments A19420, A19423, A19426

☐ CONTINUED

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS

☐ CONTINUED

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST
None		

SECTION 3 - REVIEW ORGANIZATIONS

REQUIRED

Ops !

☐ CONTINUED

REQUESTED

☐ CONTINUED

PROCEDURE PREPARER	DATE	QUALIFIED REVIEWER	DATE
DR Blakely	1/18/99	Daniel B Kelley	1/18/99

SECTION 4 - ATTACHMENTS

COMPLETED AND ATTACHED

YES	N/A		YES	N/A		YES	N/A	
☐	☒	VALIDATION CHECKLIST	☒	☐	SAFETY REVIEW	☐	☒	REVISED CROSS REFERENCES LIST
☒	☐	COMMITMENT VERIFICATION SUMMARY	☒	☐	DOCUMENT REVIEW SHEETS	☐	☒	PCRs
☐	☒	SAFETY EVALUATION NO. ____				☒	☐	OTHER TSC Guidance for Implementing DB-01-2000 Step 10.14.1 (ICAQR 1998-1641)

FINAL QUALIFIED REVIEWER CONCURRENCE	DATE
Daniel B Kelley	1/18/99

SECTION 5 - TEMPORARY APPROVAL

MANAGEMENT SRO	DATE	APPROVAL AUTHORITY	DATE

SECTION 6 - CONCURRENCE / FINAL APPROVAL

	DATE		DATE
PROCEDURE SPONSOR Mgr Reg Affairs [signature]	1/18/99	☐ MANAGER - BUSINESS SERVICES	
☐ DIRECTOR - NA		☒ SRB Robert B Cook	1/20/99
☐ DIRECTOR - E&S		☒ PLANT MANAGER [signature]	1/20/99
☐ DIRECTOR - NSS		APPROVAL AUTHORITY Dir. Eng. & Ser. R L Donnelly	1/21/99

SECTION 7 - TRAINING / PROCEDURE EFFECTIVITY

TRAINING COMPLETE	ALTERATION EFFECTIVE DATE	PROCEDURE SPONSOR	DATE
☒ YES ☐ N/A	JAN 22 1999	[signature]	1/21/99

PROCEDURE DEVELOPMENT FORM
ED 7635-12

SHEET 1 OF 2

PROCEDURE ACTIVITY TRACKING NO. C97-2879

SECTION 1 - IDENTITY

PROCEDURE NO.	~~PROPOSED~~ REVISION NO.	PROPOSED CHANGE NO.		
RA-EP-02320 (Imd 1-28-98)	1	2	☐ NEW ☐ REVISION ☒ CHANGE ☐ TEMPORARY APPROVAL	☐ CANCELLATION ☐ INACTIVATION ☐ REACTIVATION
SUPERSEDES: RA-EP-02320 R1 C1				☐ RESTRICTED EXPIRES ______ (DATE/EVENT)

PROCEDURE TITLE
Emergency Technical Assessment

PROCEDURE CLASSIFICATION
☒ SR ☐ QR ☐ N-QR CHANGE TO? ☐ YES ☒ NO

PAT/PCR NOS. CLOSED OUT
PCR PA96-2297 RFA 97-0085 RFA97-0218 ☐ CONTINUED

ACTIVITY SUMMARY / PURPOSE
1. The purpose of this change is to:
   1. Add responsibility for Severe Accident Management Guideline implementation to TSC duties.
   2. Add responsibility for evaluating RCP restart criteria to TSC duties, including an Attachment which explains the criteria.
   3. Add a caution regard the potential for a radiation release path while taking PASS samples
   4. Revise the curie content of the core to reflect a 24 month cycle.
5. This alteration does not change the procedure classification.

☐ CONTINUED

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS ☐ CONTINUED

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST
DB-OP-02000	RPS, SFAS, SFRCS TRIP or SG Tube Rupture	A TA will be issued on the effective date.

SECTION 3 - REVIEW ORGANIZATIONS

REQUIRED: ~~SRB~~ (1/15/98 DBK) Chem' RP' ~~Stat~~ (1/15/98 DBK) Ops' ☐ CONTINUED

REQUESTED: EP' ☐ CONTINUED

PROCEDURE PREPARER	DATE	QUALIFIED REVIEWER	DATE
Dennis R Blakely	12/11/97	DR Blakely	12/11/97

SECTION 4 - ATTACHMENTS

COMPLETED AND ATTACHED

YES	N/A		YES	N/A		YES	N/A	
☐	☒	VALIDATION CHECKLIST	☒	☐	SAFETY REVIEW	☐	☒	REVISED CROSS REFERENCES LIST
☒	☐	COMMITMENT VERIFICATION SUMMARY	☒	☐	DOCUMENT REVIEW SHEETS	☒	☐	PCRs
☐	☒	SAFETY EVALUATION NO. ______	☐	☒	P COMMITMENT DOCUMENTATION	☒	~~☒~~ DBK 1/16/98	OTHER RFA's listed above

FINAL QUALIFIED REVIEWER CONCURRENCE	DATE
David B. Kelley	1/16/98

SECTION 5 - TEMPORARY APPROVAL

MANAGEMENT SRO	DATE	APPROVAL AUTHORITY	DATE

SECTION 6 - CONCURRENCE / FINAL APPROVAL

	DATE		DATE
PROCEDURE SPONSOR MGR - Reg Affairs: David H. Lockwood FOR J. FREELS	1-19-98	☒ SRB: J. W. Stevens, Jr	1/22/98
☐ DIRECTOR - NA		☒ PLANT MANAGER Approval Authority: J H Lash	1/22/98
☒ DIRECTOR - E&S Approval Authority: Robert E Donnelly	1/26/98	APPROVAL AUTHORITY	
☐ DIRECTOR - NSS			

SECTION 7 - TRAINING / PROCEDURE EFFECTIVITY

TRAINING COMPLETE	ALTERATION EFFECTIVE DATE	PROCEDURE SPONSOR	DATE
☒ YES ☐ N/A	1-30-98	David H. Lockwood FOR J. L. FREELS	1-27-98

PROCEDURE DEVELOPMENT FORM
ED 7635-12

SHEET 2 OF 2

PROCEDURE ACTIVITY TRACKING NO. C97-2879

SECTION 1 - IDENTITY

PROCEDURE NO. RA-EP-02320

SUPERSEDES: RA-EP-02320 R1 C1

~~PROPOSED~~ 2nd 1-28-98 REVISION NO. 1

PROPOSED CHANGE NO. 2

☐ NEW ☐ REVISION ☒ CHANGE ☐ TEMPORARY APPROVAL ☐ CANCELLATION ☐ INACTIVATION ☐ REACTIVATION

☐ RESTRICTED

EXPIRES ______ (DATE/EVENT)

PROCEDURE TITLE: Emergency Technical Assessment

PROCEDURE CLASSIFICATION: ☒ SR ☐ QR ☐ N-QR CHANGE TO? ☐ YES ☒ NO

PAT/PCR NOS. CLOSED OUT: PAT 96-2297 RFA 97-0085 RFA 97-0218 ☐ CONTINUED

ACTIVITY SUMMARY / PURPOSE

A. The purpose of this change is to:
1. Add responsibility for Severe Accident Management Guideline implementation to TSC duties.
2. Add responsibility for evaluating RCP restart criteria to TSC duties, including an Attachment which explains the criteria.
3. Add a caution regard the potential for a radiation release path while taking PASS samples
4. Revise the curie content of the core to reflect a 24 month cycle.

B. This alteration does not change the procedure classification.

☐ CONTINUED

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS ☐ CONTINUED

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST
DB-OP-02000	RPS, SFAS, SFRCS Trip or SG Tube Rupture	A TA will be issued on the effective date.

SECTION 3 - REVIEW ORGANIZATIONS

REQUIRED: ~~SRB~~ 1/15/98 DBK, Chem', RP', ~~Stat~~ 1/15/98 DBK, Ops' ☐ CONTINUED

REQUESTED: EP' ☐ CONTINUED

PROCEDURE PREPARER	DATE	QUALIFIED REVIEWER	DATE
Dennis R Blakely	12/11/97	DR Blakely	12/11/97

SECTION 4 - ATTACHMENTS

COMPLETED AND ATTACHED

YES	N/A		YES	N/A		YES	N/A	
☐	☒	VALIDATION CHECKLIST	☒	☐	SAFETY REVIEW	☐	☒	REVISED CROSS REFERENCES LIST
☒	☐	COMMITMENT VERIFICATION SUMMARY	☒	☐	DOCUMENT REVIEW SHEETS	☒	☐	PCRs
☐	☒	SAFETY EVALUATION NO. ______	☐	☒	P COMMITMENT DOCUMENTATION	☒	☒ DBK 1/16/98	OTHER RFA's listed above

FINAL QUALIFIED REVIEWER CONCURRENCE	DATE
David B. Kelley	1/16/98

SECTION 5 - TEMPORARY APPROVAL

MANAGEMENT SRO	DATE	APPROVAL AUTHORITY	DATE

SECTION 6 - CONCURRENCE / FINAL APPROVAL

	DATE		DATE
PROCEDURE SPONSOR MGR - Reg Affairs: David H. Lockwood FOR J. Freels	1-19-98	☒ SRB: J. W. Stoner, Jr.	1/22/98
☐ DIRECTOR - NA		☒ PLANT MANAGER Approval Authority: J H Lash	1/22/98
☒ DIRECTOR - E&S Approval Authority: Robert E Donnelly	1/26/98	APPROVAL AUTHORITY	
☐ DIRECTOR - NSS			

SECTION 7 - TRAINING / PROCEDURE EFFECTIVITY

TRAINING COMPLETE	ALTERATION EFFECTIVE DATE	PROCEDURE SPONSOR	DATE
☐ YES ☐ N/A			

PROCEDURE DEVELOPMENT FORM
ED 7635-11

SHEET 1 OF 1

PROCEDURE ACTIVITY TRACKING NO. 97-2100

SECTION 1 - IDENTITY

PROCEDURE NO.	PROPOSED REVISION NO.	PROPOSED CHANGE NO.		
RA-EP-02320 R01 OLD NO. (For No. Change)	~~+~~ KSG 11/3/97	1	☐ NEW ☐ CANCELLATION ☐ REVISION ☐ INACTIVATION ☒ CHANGE ☐ REACTIVATION ☐ TEMPORARY APPROVAL	☐ RESTRICTED EXPIRES ______ (DATE/EVENT)

PROCEDURE TITLE
Emergency Technical Assessment

PROCEDURE CLASSIFICATION
☒ SR ☐ QR ☐ N-QR CHANGE TO? ☐ YES ☒ NO

PAT/PCR NOS. CLOSED OUT
97-1969

☐ CONTINUED

ACTIVITY SUMMARY / PURPOSE
To delete figure 1-1 and reference Figure 2 in DB-OP-02000 (PCAQR 97-~~1051~~ 1051)

☐ CONTINUED

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS

☐ CONTINUED

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST
N/A		

SECTION 3 - REVIEW ORGANIZATIONS

DUE: 9/11/97

REQUIRED
Chem OPS
NED RP
☐ CONTINUED

REQUESTED
☐ CONTINUED

PROCEDURE PREPARER	DATE	QUALIFIED REVIEWER	DATE
[signature]	8/29/97	[signature]	9/2/97

SECTION 4 - ATTACHMENTS

COMPLETED AND ATTACHED

YES	N/A		YES	N/A		YES	N/A	
☐	☒	VALIDATION CHECKLIST	☒	☐	SAFETY REVIEW	☐	☒	REVISED CROSS REFERENCES LIST
☒	☐	COMMITMENT VERIFICATION SUMMARY	☒	☐	DOCUMENT REVIEW SHEETS	☒	☐	PCRs
☐	☒	SAFETY EVALUATION				☐	☐	OTHER ______

FINAL QUALIFIED REVIEWER CONCURRENCE	DATE
[signature]	10/2/97

SECTION 5 - TEMPORARY APPROVAL

MANAGEMENT SRO	DATE	APPROVAL AUTHORITY	DATE
N/A		N/A	

SECTION 6 - CONCURRENCE / FINAL APPROVAL

	DATE		DATE
PROCEDURE SPONSOR [signature]	10/3/97	☒ SRB [signature]	10/15/97
☐ DIRECTOR - NA N/A		☒ PLANT MANAGER [signature]	10/16/97
☐ DIRECTOR - E&S N/A		APPROVAL AUTHORITY [signature]	10/24/97

SECTION 7 - TRAINING / PROCEDURE EFFECTIVITY

TRAINING COMPLETE	ALTERATION EFFECTIVE DATE	PROCEDURE SPONSOR	DATE
☐ YES ☒ N/A	NOV 6 1997	[signature]	10/27/97

SECTION 1 - IDENTITY (CONTINUED)

SECTION 2 - CONCURRENT EFFECTIVE DOCUMENTS (CONTINUED)

DOCUMENT NO. / REVISION	DOCUMENT TITLE	PAT NO. / CHANGE REQUEST

SECTION 3 - REVIEW ORGANIZATIONS (CONTINUED)

REQUIRED	REQUESTED

Davis-Besse Nuclear Power Station

EMERGENCY PLAN IMPLEMENTING PROCEDURE

RA-EP-02320

(Supersedes RA-EP-02320 R0)

EMERGENCY TECHNICAL ASSESSMENT

REVISION 1

Prepared by: Dennis J Gordon 6/8/95
Date

Sponsor: W. D. O'Connor 6/14/95
Manager - Regulatory Affairs Date

Approved by: S. C. Jain / WO 6/14/95
Director - Engineering & Services Date

Approved by: [signature] 7/14/95
Plant Manager Date

Effective Date: JUL 28 1995

Procedure Classification:

X Safety Related

____ Quality Related

____ Non-Quality Related

LIST OF EFFECTIVE PAGES

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1.0 PURPOSE

The purpose of this procedure is:

1.1 To describe necessary technical and engineering assessment activities during an emergency response to a major off-normal event involving the reactor systems.

1.2 To provide the methodology for estimating reactor core damage using the radiochemistry sample analysis, Emergency Containment Radiation Plots and other available parameters such as incore thermocouple readings and containment hydrogen concentration.

2.0 REFERENCES

2.1 Developmental

2.1.1 NUREG-0654/FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

2.1.2 NUREG 0956 (draft) Radionuclide Release Under Specific LWR Accident Conditions, January 1983

2.1.3 NUREG 0737, Clarification of TMI Action Plan Requirements

2.1.4 Davis-Besse Nuclear Power Station Emergency Plan

2.1.5 Bechtel to TED Letter dated October 22, 1984, Log No. BT 15004

2.1.6 Rogovin Report. Three Mile Island, A Report to the Commissioners and to the Public. Volume II, Part 2

2.1.7 Requirements for Post-Accident Sampling and Analysis - Frank Witt, USNRC. Presented at the 1983 ANS Winter Meeting

2.1.8 Westinghouse Owners' Group Post Accident Core Damage Assessment Methodology, Rev. 1, March 1984

2.1.9 Operator Training - Degraded core recognition and mitigation TRG-81-3, B&W, May 1981

2.1.10 J. Steward Bland, Containment Radiation Levels for Various Accident Conditions and Emergency Classification for the Davis-Besse Nuclear Power Station, January, 1983

2.1.11 Davis-Besse Nuclear Power Station Shift Manager Training Lesson Plan; STA-EOP-I021.

2.2 Implementing

2.2.1 RA-EP-01500, Emergency Classification

2.2.2 RA-EP-02310, TSC Activation and Response

2.2.3 DB-CH-06000, Post Accident Sampling System Operation and Analysis

2.2.4 DB-CH-06001, Emergency Containment Grab Sampling System Operation and Analysis

2.2.5 DB-OP-02000, RPS, SFRCS Trip or SG Tube Rupture

2.2.6 DB-OP-06502, Containment Hydrogen Dilution and Hydrogen Purge System

2.2.7 INPO 86-032, INPO Emergency Resources Manual

2.2.8 Framatome Technologies, INS-97-1040, letter from Len Barker to BWOG Operations Support Committee, dated March 18, 1997.

3.0 DEFINITION

3.1 POST ACCIDENT SAMPLING SYSTEM (PASS) - A system designed to obtain highly radioactive samples from containment building atmosphere, pressurizer liquid space, letdown system, decay heat loops one and two, and reactor coolant system cold leg 2-1.

4.0 RESPONSIBILITIES

4.1 The Technical Support Center (TSC) Engineering Manager is responsible for implementing this procedure.

4.2 The Emergency Plant Manager is responsible for making the final decision to obtain PASS samples.

4.2.1 Determination of timing and location of a PASS sample shall be made in consultation with the following individuals:

- TSC Engineering Manager
- Emergency Radiation Protection (RP) Manager
- TSC Operations Supervisor
- Core Thermal Hydraulics (CTH) Engineer

4.3 The CTH Engineer is responsible for assessing core damage.

4.4 TSC management and supervision shall be aware of activities involving core damage assessment.

5.0 INITIATING CONDITIONS

This procedure shall be initiated when an emergency classification of Alert or greater has been declared and the TSC has been activated.

6.0 PROCEDURE

6.1 Problem Identification

6.1.1 The TSC Engineering Manager shall obtain the latest emergency status information in accordance with RA-EP-02310, TSC Activation and Response.

6.1.2 The TSC Engineering staff shall continuously determine the status of affected plant equipment and systems.

6.1.3 The TSC Operations staff shall continuously determine the status of the plant operations including:

a. Essential plant parameters and parameter trends
b. System and equipment status
c. Automatic and manual actions being taken to establish and maintain safe shutdown.
d. Containment hydrogen concentration and corrective actions in accordance with DB-OP-06502, Containment Hydrogen Dilution and Hydrogen Purge System.

6.1.4 The Core Thermal Hydraulics (CTH) Engineer shall continuously determine if event history or current plant conditions indicate the potential for core damage. Conditions which may indicate the potential for core damage include:

a. Loss of reactor coolant flow
b. Pressurizer level out of range low
c. Significant loss of reactor coolant
   1. Hot Leg Level Monitoring System (HLLMS) - computer points L722 & L723 - valid indication only when Reactor Coolant Pumps (RCP) are off.
   2. RCP Monitoring Program - Computer points C780, C800, C820, and C840 calculate void fraction when the reactor is tripped, and the RCP(s) are running.
d. High containment sump levels
e. Loss of subcooling margin
f. Reactivity excursions
g. Failure of the Reactor Protection System (RPS) to shutdown the reactor
h. Failure of Emergency Core Cooling System (ECCS) systems
i. High containment hydrogen concentration
j. High core exit thermocouple temperatures
k. High containment radiation or airborne radioactivity
l. High activity levels in the coolant
m. Any other condition which could cause inadequate core cooling
n. Higher than normal post trip source range counts on the excore detectors C-4
o. Unusual indications on Self Powered Neutron Detectors (SPND) (see Attachment 1).

6.2 Reporting Results

The TSC Engineering Manager shall inform the Emergency Plant Manager of any immediate staff recommendations for controlling the emergency situation.

6.3 Control Room Assistance

As directed by the TSC Engineering Manager, the TSC Engineering and Operations staffs shall provide assistance to the Control Room staff by relieving the plant operators of peripheral duties such as:

6.3.1 Plotting cooldown curves, radiation dose rates, or other key parameters to backup or assist in trend analysis.

6.3.2 Verifying and logging instrument indications where continuous technical assessment and monitoring is required to indicate when core degradation is or may be occurring.

6.3.3 Evaluating the adequacy of natural circulation flow, heat sink efficiency or other system operation.

6.3.4 Providing estimates of release rates for the Dose Assessment Staff when the radioactive release pathway is from the secondary side or is unmonitored.

6.3.5 Providing other information, analysis, and recommendations as requested.

6.3.6 Evaluating if Reactor Coolant Pump Restart Criteria (Attachment 9) have been met and recommending restart of RCPs if possible.

6.3.7 Evaluating plant conditions per Attachment 10, Boron Dilution Flowpath Concerns, prior to initiating boron dilution flowpaths.

6.3.8 Assessing plant conditions and recommending actions based on Severe Accident Management Guideines.

6.4 Other Projects

6.4.1 The TSC Engineering Manager shall coordinate the design and installation of short term instrument and control modifications, including preparation of abnormal operating procedures necessary to support the effort.

6.4.2 The TSC Engineering Manager shall arrange for additional engineering support, such as:

a. Reactor Engineering, Transient Assessment, or others, to utilize techniques such as event tree analyses or computer calculations.

b. Engineering development of system modifications necessary to ensure the immediate safe shutdown of the reactor, and any system additions necessary to maintain long-term shutdown capabilities.

6.5 Core Damage Assessment

The CTH Engineer shall perform an initial and continuous assessment of the extent of the core damage.

6.5.1 Evaluate core exit thermocouple (CETC) and self powered neutron detectors (SPND) for indications of possible cladding damage using Attachment 1, Core Damage Assessment Using Core Exit Thermocouples and Self Powered Neutron Detectors. Use Attachment 2, Core Damage Assessment Form, to record the results of the evaluations of the CETC readings.

6.5.2 Evaluate Containment Radiation readings and Containment Hydrogen readings provided on Attachment 2, Core Damage Assessment Form.

a. The Emergency Radiation Protection Manager (ERPM) will provide the radiation and hydrogen readings on a copy of Attachment 2, Core Damage Assessment Form.

b. The Safety Features Actuation System (SFAS) radiation monitor readings RE-2004, RE-2005, RE-2006, and RE-2007 (Computer points R311, R312, R313, and R314) may also be used to indicate a release of radiation to containment.

6.5.3 Evaluate the containment high range radiation monitor readings using Attachment 3, Core Damage Assessment Using Containment Radiation Dose Rates, and determine the percent metal-water reaction using Attachment 4, Core Damage Assessment Using Containment Hydrogen Concentration.

6.5.4 Notify the TSC Engineering Manager and the Emergency Plant Manager that a decision must be made whether or not to take a PASS sample.

6.6 Decision to Take PASS Sample

NOTE 6.6.1

Attachment 5 contains information on PASS Sampling and Analysis.

CAUTION 6.6.1

Using P218, Post Accident Gas Sample Pump, to obtain a containment atmosphere PASS grab sample will cause radiation and contamination levels in the Auxiliary Building Train Bay to increase due to seal leakage.

6.6.1 The CTH Engineer, in conjunction with the individuals listed in Step 4.2.1, shall advise the Emergency Plant Manager on desired PASS sample(s) locations and times (refer to Table 1, Suggested PASS Samples for Various Accidents.

a. There are no strict criteria for determining when a PASS sample should be taken. Generally, if there is indication that core damage exists, it is desirable to eventually quantify the damage as accurately as possible by using PASS sample data. Obtaining a PASS sample need not be considered an urgent or high priority task since the other methods of damage assessment directed by this procedure provide adequate information for all short term actions. NUREG-0737, Paragraph II.B.3, requires that the design of the PASS be capable of drawing and analyzing a sample within a three hour period. This requirement has no effect on the decision on when to draw a sample. The decision to schedule a PASS sample should consider issues such as:

1. Stability of plant conditions.

2. Manpower availability.

3. Radiation levels in the sampling area.

4. Availability of alternate indications.

Table 1. Suggested PASS Samples for Various Accidents

Accident Condition	PASS Sample Suggested	Reference Procedure	Alternate Samples
RCS Intact	RCS* cold leg	DB-CH-06000	Pressurizer
Small LOCA	RCS* cold leg	DB-CH-06000	Pressurizer
	Decay Heat Loops	DB-CH-06000	RCS
	CTMT Atm	DB-CH-06001	
Large LOCA	Decay Heat Loops	DB-CH-06000	RCS
	CTMT Atm	DB-CH-06001	
Steam Line Rupture	RCS* cold leg	DB-CH-06000	Pressurizer
OTSG Tube Rupture	RCS* cold leg	DB-CH-06000	Affected SG

*RCS pressure >75 psi required for PASS operation.

6.6.2 The CTH Engineer shall confer with the ERPM to determine if additional PASS sample information is needed beyond what is on Attachment 6, PASS Sample Analysis Results Form.

6.6.3 The CTH Engineer with concurrence from the ERPM, shall recommend to the Emergency Plant Manager when the PASS sample(s) should be obtained.

6.6.4 The Emergency Plant Manager shall make the final decision to obtain PASS samples.

6.7 PASS Data Evaluation

6.7.1 The ERPM shall record sample data on Attachment 6, PASS Sample Analysis Results Form.

6.7.2 The CTH Engineer shall evaluate sample data in accordance with the guidelines provided in Attachment 7, Core Damage Assessment Using PASS Sample Data.

a. Computer Calculations: The PASS Core Damage Assessment computer program, on DADS under Engineering Aids, is the primary method for data evaluation.

b. Hand Calculations: Worksheet 7-1 of Attachment 7, Core Damage Assessment Using PASS Sample Data, is provided to support and document all hand calculations during the PASS Sample evaluation.

c. Evaluation results shall be recorded on Attachment 8, PASS Sample Core Damage Assessment Form, and the form provided to the TSC Engineering Manager and Dose Assessment Coordinator.

7.0 FINAL CONDITIONS

This procedure shall be terminated when:

7.1 The emergency has been terminated and TSC support is no longer needed.

7.2 All records reviewed by technical personnel and forwarded to the Emergency Preparedness Unit.

7.3 Deficiencies are recorded and reported to the Emergency Preparedness Unit, and equipment returned to its original state of readiness.

8.0 RECORDS

8.1 The following quality assurance records are completed by this procedure and shall be listed on the Nuclear Records List, captured, and submitted to Nuclear Records Management in accordance with NG-NA-00106:

8.1.1 None

8.2 The following non-quality assurance records are completed by this procedure and may be captured and submitted to Nuclear Records Management, in accordance with NG-NA-00106:

8.2.1 Core Damage Assessment Form.

8.2.2 PASS Sample Core Damage Assessment Form.

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES (CETC) AND SELF POWERED NEUTRON DETECTORS (SPND)

1. If core conditions are no longer deteriorating, then use parameter history obtained through DADS to determine worst case conditions during the transient. The core exit thermocouple readings are normally used in conjunction with the PASS sample analysis to evaluate the core damage. However, these readings provide significant information on the core conditions prior to taking PASS samples.

   a. If more than eight core exit thermocouples are available, then calculate the average core exit thermocouple temperature by averaging no less than five of the highest reading thermocouples (computer points T511 to T562). If eight or less thermocouples are available, then use an average of the three highest.

   b. Determine the RCS system pressure as measured by hot leg pressure sensors (P724, P725, P732, and P733).

   c. Use DB-OP-02000, Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC to determine approximate cladding temperature.

   d. Use Table 1-1 to determine possible cladding and fuel damage states.

2. Evaluation

   As the core exit thermocouple temperature increases beyond saturation and approaches the T_{clad} > 1400°F curve, the cladding will experience ballooning and bursting. These failures will start at cladding temperatures above 1300°F (i.e., the saturation curve in DB-OP-02000, Figure 2, Incore T/C Temperature vs. RCS Pressure) and start accelerating at 1400°F.

   In the temperature region between the T_{clad} > 1400°F curve and the T_{clad} > 1800°F curve the cladding will experience major damage. Around 1600°F oxidation of cladding due to Zr-metal water reaction will start generating significant quantities of hydrogen. If the temperature approaches 1800°F, then it can be assumed that there is a major cladding damage and possible fuel overheating.

   If the cladding temperature is beyond 1800°F, then it can be assumed that there is significant fuel overheating and possible fuel melt. Since DB-OP-02000, Figure 2, Incore T/C Temperature vs. RCS Pressure is a correlation for the cladding temperature, it is not possible to determine the extent of fuel overheating or melting.

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES (CETC) AND SELF POWERED NEUTRON DETECTORS (SPND) (Con't)

Core damage estimation from Self Powered Neutron Detectors (SPND)

SPND are also useful in assessing clad temperature thresholds. For example, due to insulation breakdown at approximately 1000°F, a space charge release(+) will cause SPNDs to spike high. If the clad temperature does not exceed 1100°F, the SPNDs output will remain high. However, if cladding temperatures continue to rise, the SPNDs will go from off-scale-high due to space charge release(+) to off-scale-low due to thermionic emission(-).

If clad damage is suspected, a review of the SPNDs output, level by level, starting at the upper level (Level 7), should aid in assessing amount and location of clad damage. SPNDs output can be obtained by reviewing computer points R413 - R776.

See reference 2.1.11 for more information.

> **NOTE TABLE 1-1**
>
> Consult DB-OP-02000, Figure 2, Incore T/C Temperature vs. RCS Pressure for ICC when evaluating the cladding temperature and regions

TABLE 1-1

RELATIONSHIP BETWEEN FUEL DAMAGE AND CLAD TEMPERATURE

Cladding Temperature °F	Fuel damage state
≤ Tsat (Region 1)	No damage
> Tsat (Region 2)	Inadequate Core Cooling; Ballooning and Rupture of Zircaloy cladding at Tclad >1300°F
> 1400 (Region 3)	Cladding damage accelerates
> 1600	Oxidation of cladding and hydrogen generation.
> 1800 (Region 4, Severe) Accident Region)	Fuel overheating/melting

The lowest temperature at which UO_2-Zr-ZrO_2 liquifies is approximately 3500°F.

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CORE DAMAGE ASSESSMENT FORM

Containment Radiation				Containment Hydrogen			Core Exit Thermocouples						
Time/ Date	Cont. Rad. (R/hr) RE-4596A / RE-4596B	Region	Core Damage	Time/ Date	Cont. H_2 * (%) A302 / * (%) A303	Metal-Water Reaction (%)	Time/ Date	Core TCs Used	Temp. (°F)	Ave. Temp. (°F)	RCS Pressure (psig)	Clad Temp. (°F)	Fuel Dam-age State

* Verify isolation valves, CV5010 and CV5011 series, are open prior to taking reading.

CORE DAMAGE ASSESSMENT USING CONTAINMENT RADIATION DOSE RATES

Evaluate the extent of core damage using containment radiation dose rates as follows:

1. Obtain readings of RE 4596A and 4596B from any of the following:

   a. DADS (Validyne Computer Point R299)

   b. The Control Room

   c. The Emergency RP Manager

   d. Radiation Trends display on SPDS

2. On Figure 3-1, locate the point which represents the intersection of the highest reading in R/hr from RE 4596A or B, and the time after reactor shutdown. This is point X.

3. The region in which point X is located corresponds to core damage assessment as follows:

REGION	CORE DAMAGE
Below A	<10% gap activity released
$A \le X \le B$	10%-50% gap activity released
$B < X \le C$	>50% gap activity released
Above C	>100% gap activity plus fuel melt.

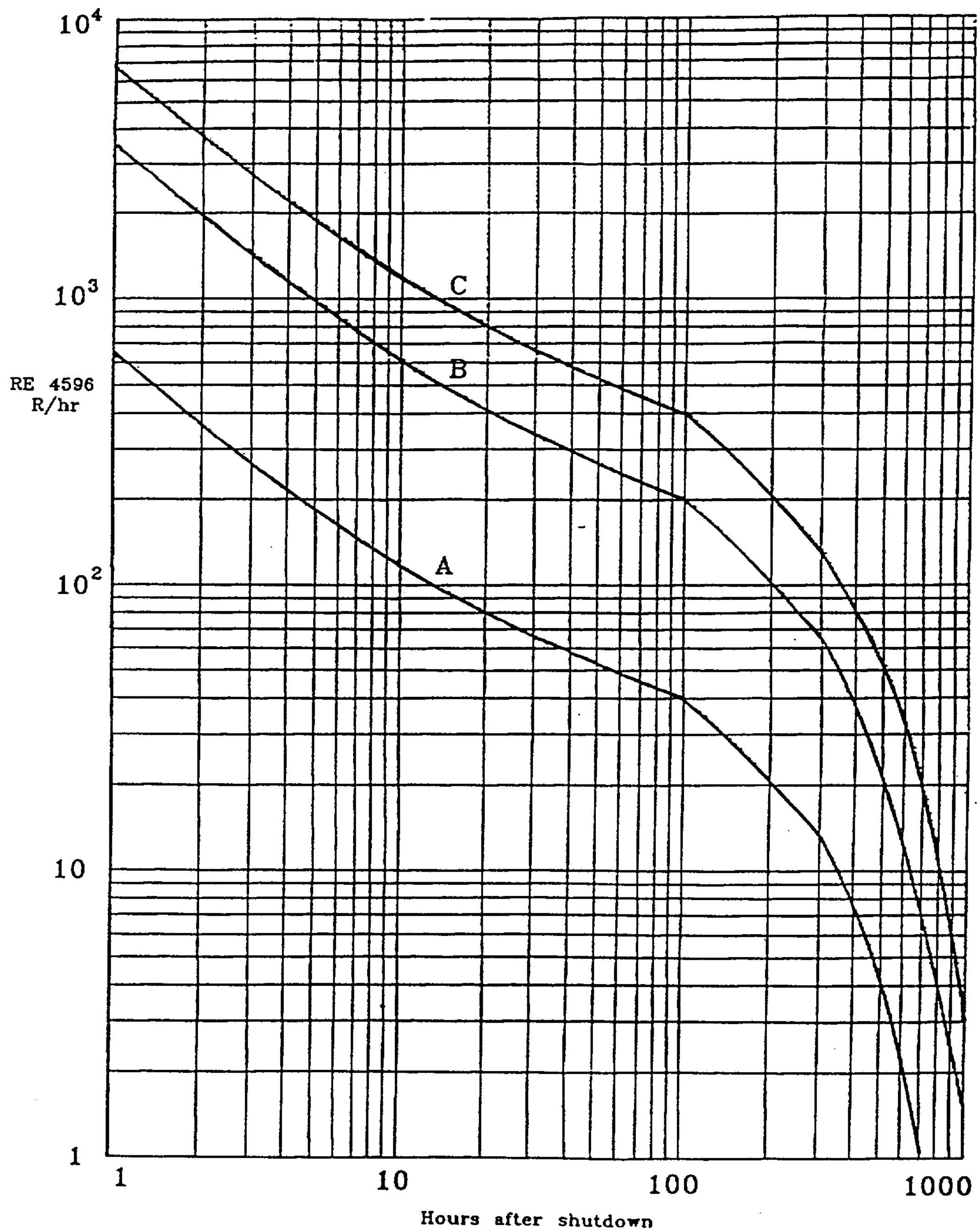
4. Record data on Attachment 2, Core Damage Assessment Form.

5. Background Information:

   Figure 3-1 is based on data from Reference 2.1.10.

CORE DAMAGE ASSESSMENT USING CONTAINMENT RADIATION DOSE RATES (Cont.)

Figure 3-1



Based on data from reference 2.1.10

CORE DAMAGE ASSESSMENT USING CONTAINMENT HYDROGEN CONCENTRATION

1. The Containment Hydrogen Monitors are maintained shutdown and isolated. Prior to taking a reading, verify that the monitors have been placed into service by communication with Control Room personnel.

2. Determine the containment hydrogen concentration by:

   a. Containment Trends display on SPDS

   b. Reading computer points A302 and A303 on DADS

   c. If there is measurable primary containment atmosphere hydrogen concentration, the TSC Engineering Manager should consider the use of the Hydrogen Dilution and Purge System and should recommend that the Emergency Plant Manager initiate steps to obtain a hydrogen recombiner.

      1) Request a hydrogen recombiner from Duquesne Light Company's Beaver Valley Nuclear Station.

      2) The Institute of Nuclear Power Operations (INPO) can be utilized to locate other sources for a hydrogen recombiner. See the INPO Emergency Resource Manual.

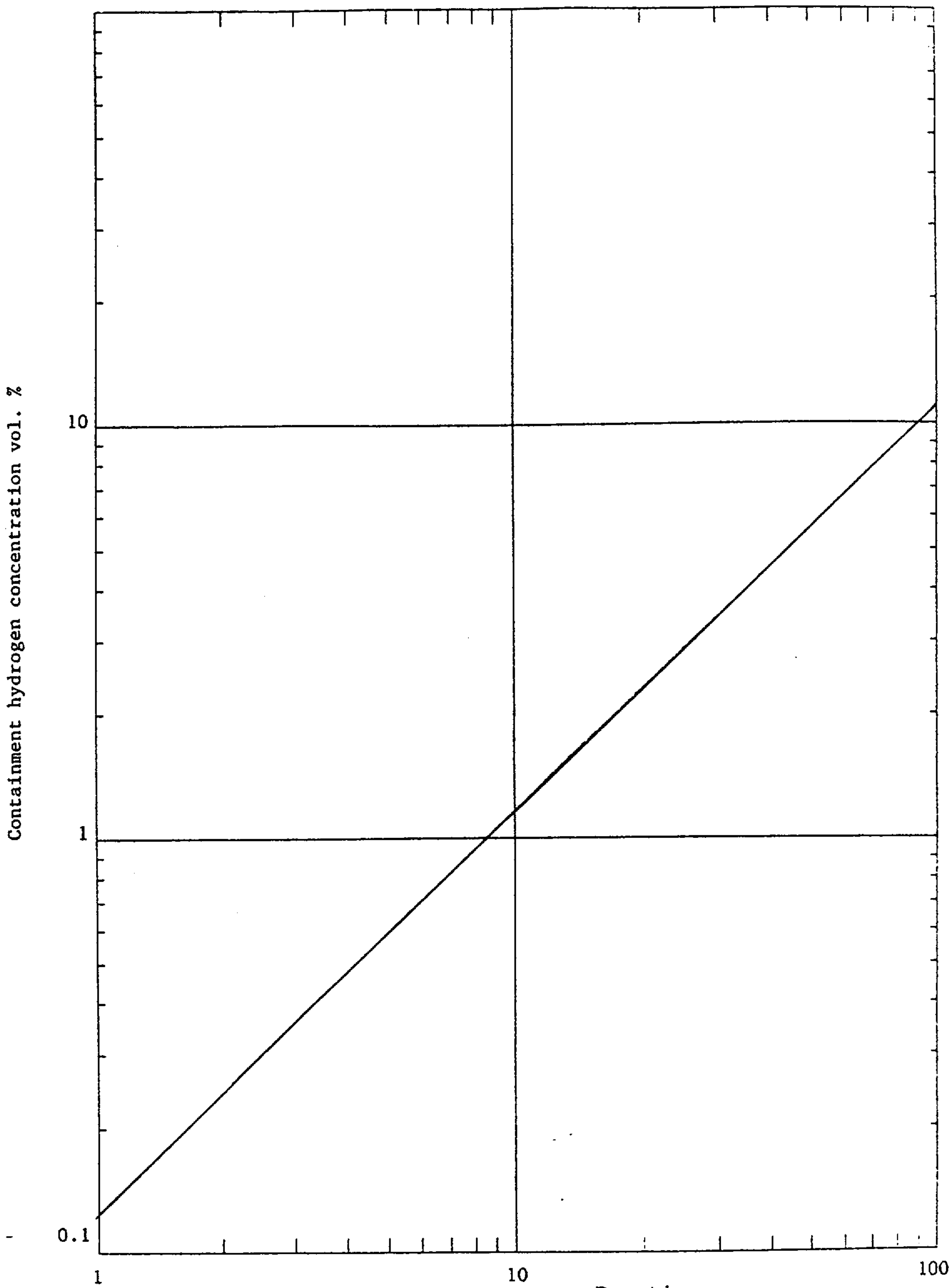
3. Determine the Percent Metal Water Reaction by plotting the containment hydrogen concentration (vol %) on Figure 4-1 on the diagonal line.

   The corresponding % Metal Water Reaction is found directly below the point of intersection.

4. Record the results on Attachment 2, Core Damage Assessment Form.

CORE DAMAGE ASSESSMENT USING CONTAINMENT HYDROGEN CONCENTRATION (Cont.)

Figure 4-1. Relationship Between Containment Hydrogen Concentration and Metal Water Reaction



CRITERIA FOR DETERMINING NEED AND LOCATION FOR PASS SAMPLING AND ANALYSIS

The post accident samples that are required to assess the core damage are dependent on the accident scenario. For accidents that do not involve the breach of reactor coolant system (RCS), the sample analysis results from the RCS are adequate to assess the core damage.

For the loss of coolant accidents, samples from the RCS, Emergency Containment Sump (via Decay Heat Loops) and Containment atmosphere are normally required. However, if at the time of sampling the sump water has been recirculated through the core for some time, it can be assumed that the activity in the sump and the RCS are the same and only a sump sample is needed.

The methodology used in this procedure utilizes RCS, Emergency Containment Sump, and Containment air sample results. For a more accurate assessment, samples from additional sample locations (e.g. pressurizer) could be used.

For accidents involving secondary system (e.g. steam generator tube rupture), samples from steam generators would provide additional input to the core damage assessment.

Samples from the emergency sump can be drawn only after the decay heat removal (DHR) system is placed into operation in the sump recirculation mode. For accidents involving small break LOCAs, this mode of operation will not take place for a long time. If it is absolutley essential to perform a core damage assessment prior to placing DHR system into operation, it will be necessary to estimate concentrations in the sump based on the RCS activity, sump water volume and sequence of events during the accident.

PASS SAMPLE ANALYSIS RESULTS FORM

NOTE

Record the total of individual activities, both those dissolved in the liquid, and any gaseous activity that has been stripped from the sample.

		RCS	Sump	Containment	Other
Sample	Date				
	Time				
	Temperature F				
	Pressure PSIA				
System	Temperature F				
	Pressure PSIA				
Activity	µCi/cc				
	Xe-133				
	Kr-88				
	I-131				
	I-133				
	Ba-140				
	Other isotopes				
Sample activities are adjusted for decay to					
	Date				
	Time				

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA

1. Decay Correction to t = 0: PASS radionuclide concentrations must be corrected for radioactive decay which has occurred between the time of reactor trip and PASS sample analysis.

1.1 Evaluate whether the sample results provided have been decay corrected to time of reactor trip. If so, proceed to step 2.0. If not, perform the correction for each radionuclide using the following formulas:

$$A_o = A(e^{\lambda_i \tau_s})$$

where λ_i = decay constant for radionuclide i (hr^{-1}) (see Table 7-1)

τ_s = time elapsed between reactor shutdown and sample analysis (hours)

A_o = Sample activity for radionuclide i at the time of reactor trip

A = Sample activity for radionuclide i at the time of analysis

2. RCS Comparison: Determine if the RCS has been breached. If so, go to step 3. If not, compare the decay-corrected RCS radionuclide activity to the average operating RCS radionuclide activity. If there is no marked increase in the concentration, it can be concluded that no fuel failure has occurred.

2.1 The RCS Iodine concentration(s), particularly I-131, may increase due to spiking during reactor transients (changes in power, temperature, pressure). Spiking typically produces 1 µCi/ml of I-131 or less for 5 average-power fuel rods with defects. This should not be interpreted as fuel damage. If iodine spiking is apparent, use the noble gas radionuclide concentration to evaluate core damage.

3. Total Activity Released: An estimate of the total activity released from the core is divided by the core equilibrium inventory (modified by a power correction factor) to estimate the fraction of core damage. The following steps support this appraisal:

Calculate total activity (T_i) released during the accident for each radionuclide. Use Worksheet 7-1 of this attachment for hand analysis, or the Core Damage Assessment computer program on the DADS under Engineering Aids.

$$T_i = A_i(RCS) + A_i(Sump) + A_i(CAtms) + A_i(Env) - A_i(RCSb)$$

3.1 $A_i(RCS)$ is the activity (Ci) contained in the reactor coolant for each radionuclide. It is calculated by:

$$A_i(RCS) = C_i(RCS)\ \mu Ci/cc \times \text{RCS vol gal} \times D \times$$

$$3.785 \times 10^{-3}\ \frac{cc}{gal}\ \frac{Ci}{\mu Ci}$$

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Where: C_i(RCS) = RCS sample concentration for radionuclide i, decay corrected
RCS vol = 83028 gal
D = RCS density correction factor (Table 7-3)
3.785E-3 = conversion constant

3.2 A_i (Sump) is the activity (Ci) contained in the sump for each radionuclide. It is calculated by:

$$A_i \text{ (Sump)} = C_i \text{ (Sump)} \frac{\mu Ci}{cc} \text{ x Sump water vol in gal. x D x}$$

$$3.785\text{E} - 3 \frac{cc}{gal} \frac{Ci}{\mu Ci}$$

Where: C_i (Sump) = Sump sample concentration for radionuclide i, decay corrected
Sump vol = Sump water vol in gal (Table 7-2 or computer pts L-321 & L-322)
D = Sump density correction factor (Table 7-3)
3.785E-3 = conversion constant

The Sump activity is needed for transients in which a significant quantity of reactor coolant is released. Although the reactor sump is the prime consideration, event scenarios are possible where RCS is lost to other sumps. In this case, all sump activities should be summed.

3.3 A_i (CAtms) is the activity (C_i) contained in the containment free air space for each radionuclide. It is calculated by:

$$A_i \text{ (CAtms)} = C_i \text{ (CAtms)} \frac{\mu Ci}{cc} \text{ x containment volume ft}^3 \text{ x}$$

$$\frac{P_1 (T_2 + 460° F)}{P_2 (T_1 + 460° F)} \text{ x } 2.83 \text{ x } 10^{-2} \frac{cc}{ft^3} \frac{Ci}{\mu Ci}$$

Where: C_i (CAtms) = Containment Atmosphere free air sample concentration for radionuclide i, decay corrected

Cont vol = 2.83E6 ft^3 Containment free air volume

P_1, P_2 = containment and sample pressures (PSIA), respectively.

T_1, T_2 = containment and sample temperatures (°F), respectively.

$2.83 \text{ x } 10^{-2}$ = conversion constant

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

3.4 A_i (Env) is the activity (Ci) of each radionuclide released to the environment. It is calculated by:

A_i (Env) = C_i (Env) Ci/sec x Time min x 60 sec/min

Where: C_i (Env) = the release rate source term for radionuclide i, decay corrected (DADS)
Time = the release duration
60 = conversion constant

3.5 A_i (RCSb) is the normal activity (Ci) of each radionuclide contained in the reactor coolant prior to the transient. It is calculated by:

A_i (RCSb) = C_i (RCSb) μCi/cc x RCS vol gal x D x 3.785E-3 cc/gal Ci/μCi

Where: C_i (RCSb) = RCS routine concentration for radionuclide i (EPRM)
RCS vol = 83028 gal
D = RCS density correction factor (Table 7-3)
3.785E-3 = conversion constant

A_i (RCSb) can be ignored if post transient RCS radioactivity concentrations are significantly larger than pre-transient concentrations.

4. Equilibrium Core Inventory: Obtain the equilibrium core inventory from Table 7-1 for the isotope being used to evaluate core damage.

5. Power Correction Factor: Determine the power correction factor(s) which will be applied to the core equilibrium inventory value(s) by method a. or b., depending on time constraints. Normally the radionuclide concentrations in the core are considered to be in equilibrium if the reactor is operated at a steady state power level for four half lives. It is assumed that a steady state power level is achieved if the power did not change by greater than 10%.

a. Quick Method: Time weighted average

For I-133 and Kr-88

$$PCF = \frac{\text{Time weighted Average Power level for prior 4 days (MWt)}}{2772}$$

For Ba-140, I-131, Xe-133

$$PCF = \frac{\text{Time weighted Average Power level for prior 30 days (MWt)}}{2772}$$

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

b. Long Method: Detailed analysis

The Power Correction Factor (PCF) for each radionuclide can be calculated using the following formula:

$$PCF_i = \frac{\sum_j P_j \left(1-e^{-\lambda_i t_j}\right) e^{-\lambda_i t_{dj}}}{2772 \left(1-e^{-\lambda_i t_o}\right)}$$

Where: P_j = average power level (MWT) during operating time period t_j

λ_i = decay constant for radionuclide i in days^{-1} (Table 7-1)

t_j = operating period in days during which power did not change $\pm 10\%$ (P_j)

t_{dj} = time in days between shutdown and power change (i.e., end of time step t_j)

t_o = total operating time in days

6. Calculation of Core Fraction Released: Calculate the fraction of core inventory (F_i) released for each radionuclide.

$$F_i = \frac{T_i}{PCF_i \times Core_i}$$

Where: T_i = total activity released from radionuclide i

PCF_i = power correction factor for radionuclide i

$Core_i$ = the core inventory for full power operation for radionuclide i (See Table 7-1).

7. Using core release fractions calculated from Step 6, go to Figures 7-1, 7-2 and 7-3 to determine:

a. The percent cladding failure, by arbitrarily attributing all activity to cladding failure (Figure 7-1).

b. The percent fuel overheat, by arbitrarily attributing all activity to fuel overheat (Figure 7-2).

c. The percent fuel melt, by arbitrarily attributing all activity to fuel melt (Figure 7-3).

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

An engineering judgement based on all the available information (Attachment 2) should be employed when estimating the percent clad failure, fuel overheat, and fuel melt. This evaluation is necessary because actual events may consist of varying combinations of all three.

a. Generally, the presence of Ba-140 indicates fuel overheat or melt.

b. Cladding temperatures indicate the probability of cladding failures, fuel overheat or fuel melt.

c. Core exit thermocouple temperature distribution provides an indication of the extent of damage.

d. The curves used in the estimation of core damage assume uniform distribution of activity in the core. However, in reality, the activity in various regions of the core would be different. The estimated core damage may be adjusted based on core exit thermocouple temperature distribution, and burnup or neutron flux information for various regions of the core.

e. Hydrogen concentration in the containment will provide an indication on maximum amount of zirconium reacted.

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Worksheet 7-1

Date / /
Time :

Radionuclide ______________________

1. RCS

( )RCS $\frac{\mu Ci}{cc}$ x ( )RCS vol gal x ( )D x 3.785E-3 $\frac{cc}{gal}$ $\frac{Ci}{\mu Ci}$ = ( )Ci
PASS 8.30E4 gal Table 7-3

2. Sump

( )Sump $\frac{\mu Ci}{cc}$ x ( )Sump vol gal x ( ) D x 3.785E-3 $\frac{cc}{gal}$ $\frac{Ci}{\mu Ci}$ = ( )Ci
PASS Table 7-2 Table 7-3

3. Containment

( )Cont $\frac{\mu Ci}{cc}$ x ( )Cont Vol ft3 x $\frac{P_1(T_2+460°F)}{P_2(T_1+460°F)}$ x 2.83E-2 $\frac{cc}{ft^3}$ $\frac{Ci}{\mu Ci}$ = ( )Ci
PASS 2.83E6 ft3

4. Environment

( )Env Rel $\frac{Ci}{sec}$ x ( )time min x 60 $\frac{sec}{min}$ = ( )Ci
DADS

5. Normal RCS = - ( )Ci

Total Core Release ( )Ci

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

TABLE 7-1

RADIONUCLIDE INVENTORIES AT FULL POWER AND DECAY CONSTANT

The core inventories are based on 690 EFPD equilibrium cycle (24-month core). These numbers vary ±15% based on plutonium power fraction. For a more in depth analysis, specific calculations for the fuel cycle of interest may be needed.

Isotope	Decay Constant Hr. $^{-1}$	Decay Constant Day $^{-1}$	Inventory at full power ($Core_i$) curies
Kr-87	5.47E-1	1.31E1	3.69E7
Kr-88	2.48E-1	5.95	5.18E7
Xe-133	5.48E-3	1.32E-1	1.51E8
I-131	3.59E-3	8.62E-2	7.71E7
I-133	3.41E-2	8.18E-1	1.56E8
I-135	1.04E-1	2.50	1.46E8
Ba-140	2.26E-3	5.42E-2	1.33E8

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

TABLE 7-2

CORRELATION BETWEEN CONTAINMENT WATER LEVEL AND WATER VOLUME IN SUMP

Assume linear relationship between water level and volume in between the incremental steps.

Water Level Elevation (Ft.)	Water Volume (Gallons)	BWST Level (Ft.)
		43.8 (Full)
544	3.43E4	41.2
546	3.91E4	40.8
548.5	4.73E4	40.2
550.5	5.53E4	39.5
553	6.96E4	38.4
557	9.42E4	36.5
561	1.16E5	34.8
563.5	1.33E5	33.5
565	1.48E5	32.4
567.5	3.13E5	19.6
568.5	3.84E5	14.1
570.9	5.26E5	0.0 (Empty)
572	6.44E5	

TABLE 7-3

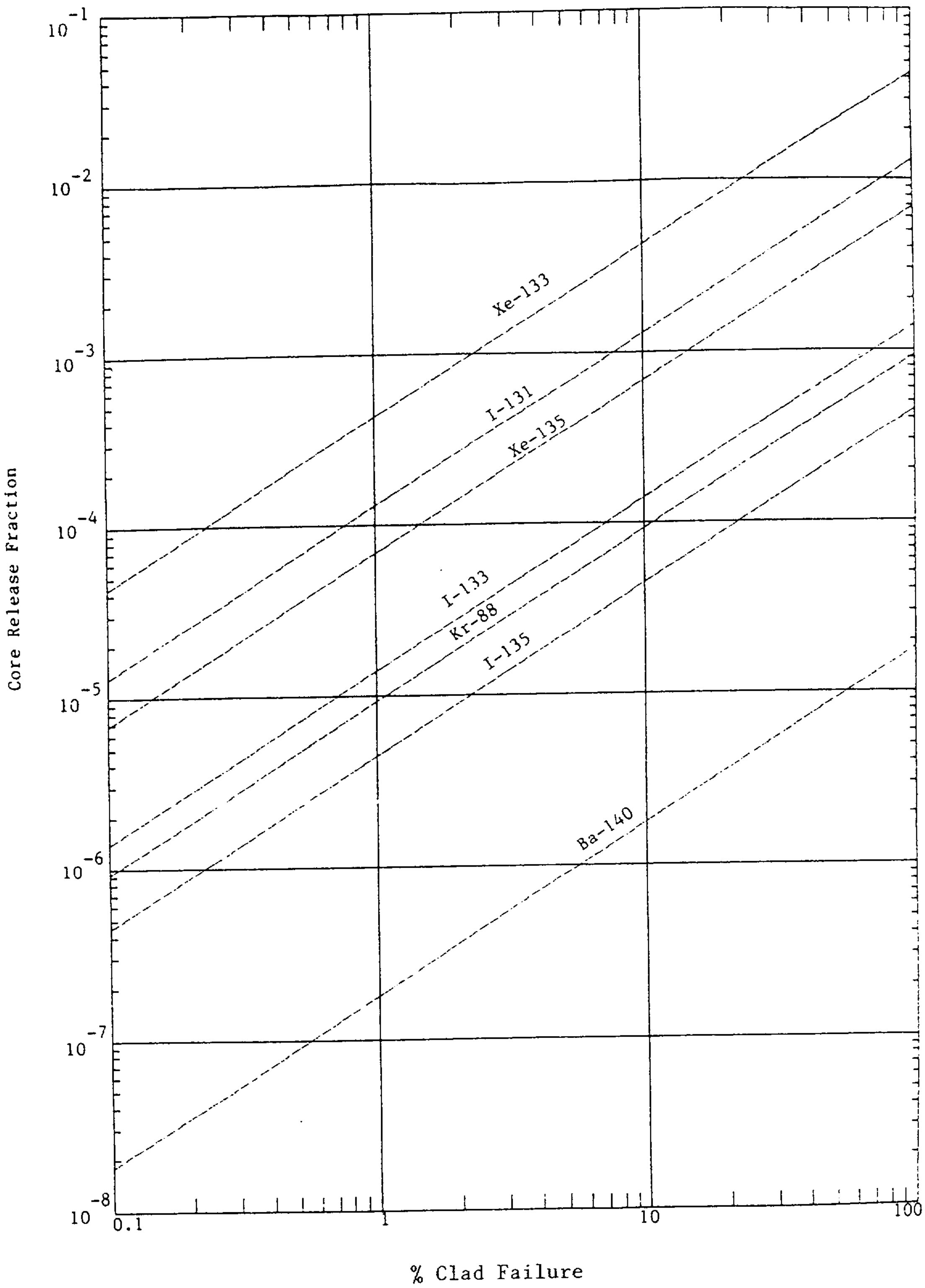
DENSITY CORRECTION FACTORS

The temperature correction factors are applied based on the temperature of the sample source (e.g., RCS, Sump, etc.).

Temp. (°F)	Density correction factor, D
100	1
150	0.987
200	0.97
250	0.949
300	0.924
350	0.897
400	0.865
450	0.830
500	0.790
540	0.752
560	0.731
580	0.708
600	0.682

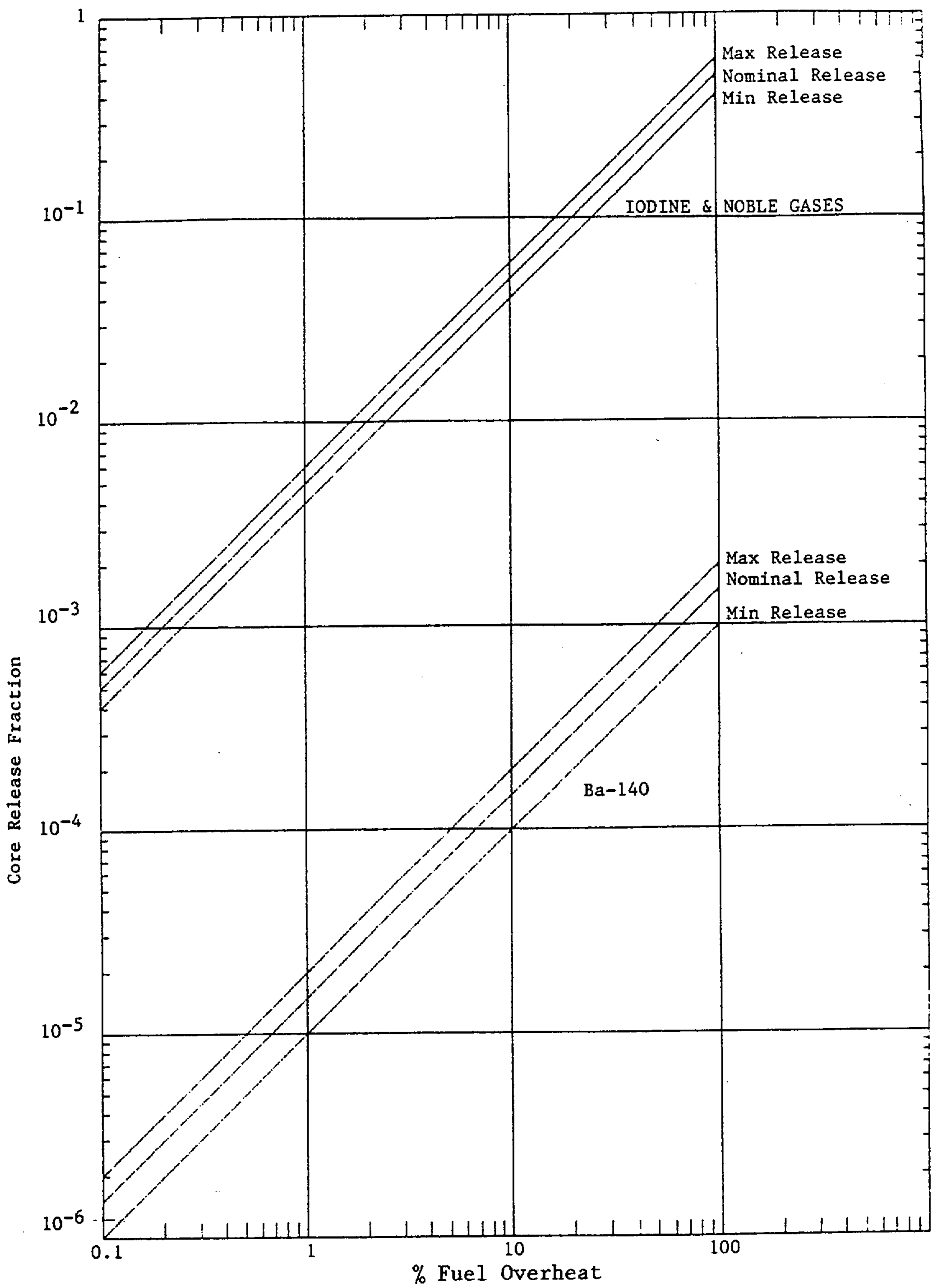
CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Figure 7-1 - Relationship Between Cladding Failures and Core Release Fraction

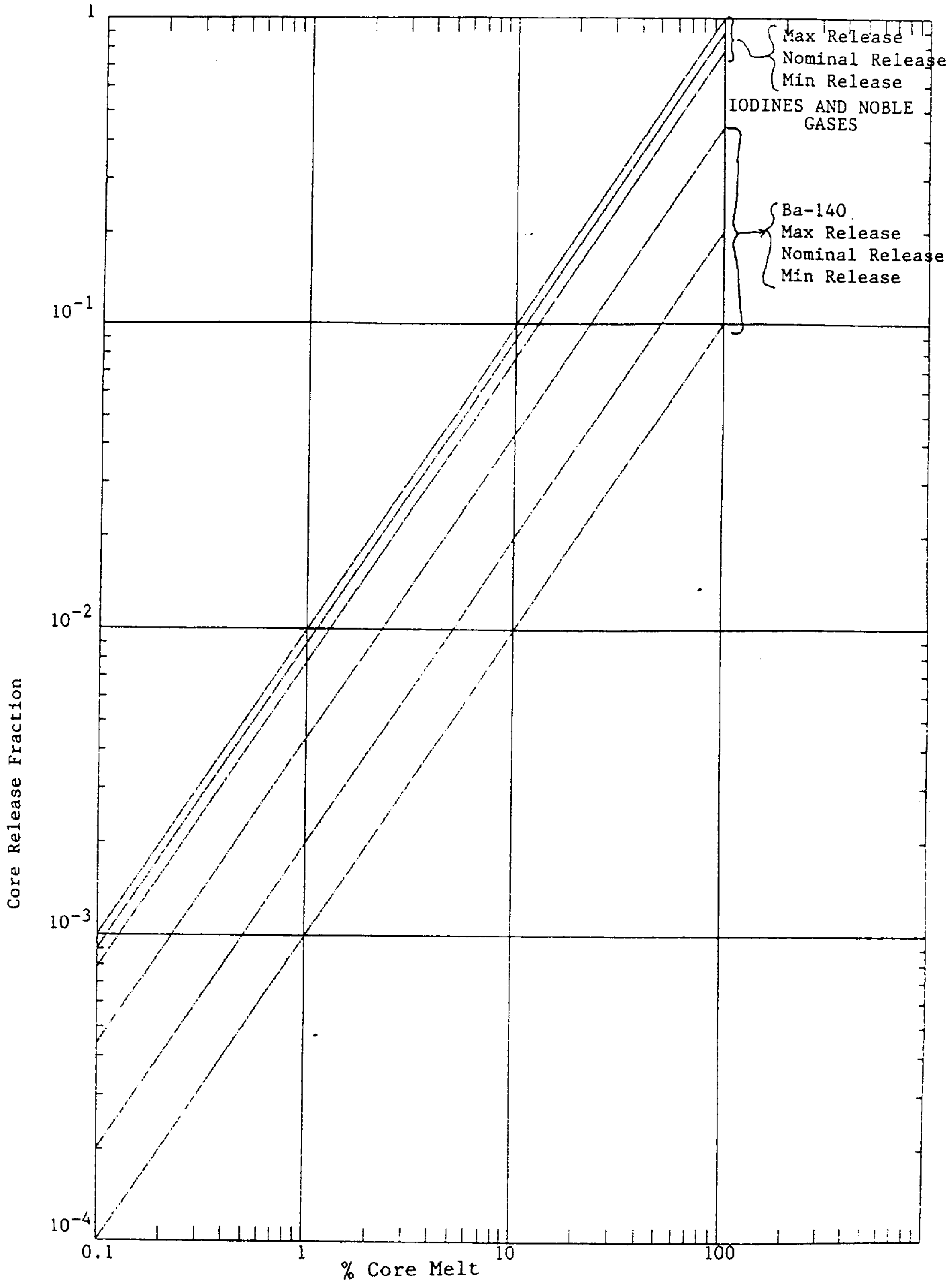


CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Figure 7-2 - Relationship Between Fuel Overheating and Core Release Fraction



CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)
Figure 7-3 - Relationship Between Core Melt
and Core Release Fraction



CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Examples

The following examples illustrate the use of the core damage assessment methodology. These examples assume the following conditions for power history and radiochemistry prior to the transient:

Reactor Coolant Concentrations (μCi/cc)

I-131	0.03
I-132	0.01
I-133	0.026
Xe-133	1.2
Kr-87	0.06
Kr-88	0.19
Ba-140	5.2E-4

Reactor is at power for more than 100 days.

Recent power history prior to shutdown.

Day 100 to Day 31: 2360 MW_t
Day 30 to Day 11: 2772 MW_t
Day 10 to Day 5: 2360 MW_t
Day 4 to Shutdown: 1940 MW_t

Example 1

Reactor experienced a trip due to an instrument malfunction. RCS sample was taken 3 hours following the trip and was counted at 1 hour after obtaining the sample. The following are the sample results.

RCS temperature	400°F
Sample temperature	100°F
I-131	1.3 μCi/cc
I-133	1.2 μCi/cc
Xe-133	2.4 μCi/cc

Adjust RCS concentration for decay. (Step 1.)

$$\text{I-131} = 1.3 \times e^{\underline{(3.59E-3 \times 4)}} = 1.32\ \mu Ci/cc$$

$$\text{I-133} = 1.2 \times e^{\underline{(3.41E-2 \times 4)}} = 1.38\ \mu Ci/cc$$

$$\text{Xe-133} = 2.4 \times e^{\underline{(5.48E-3 \times 4)}} = 2.45\ \mu Ci/cc$$

Comparison of these concentrations with RCS concentrations prior to the transient show that only spiking has occurred due to the transient. No apparent fuel damage. This is also confirmed by no abnormal thermocouple readings.

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Example 2

Reactor experienced a trip due to loss of feedwater flow. During cooldown it was noticed that subcooling margin was lost. A few core exit thermocouples in the center region recorded temperatures above 700°F and the RCS pressure was 1900 PSIG. The transient was brought under control. The containment radiation monitors and sump level instrumentation indicate that there is no breach of RCS boundary. The RCS samples were taken 3 hours from the time subcooling margin was lost and counted after 1 hour. The following are the sample results.

I-131	15.2	µCi/cc
I-133	3.2	µCi/cc
Xe-133	97	µCi/cc
Kr-88	0.4	µCi/cc
Ba-140	0.04	µCi/cc

RCS temperature = 400°F Sample temperature = 100°F

RCS concentration after correcting for decay. (Step 1.)

I-131 = $(15.2)\ e^{(3.59E-3 \times 4)}$ = 15.4 µCi/cc

I-133 = 3.7 µCi/cc

Xe-133 = 99 µCi/cc

Kr-88 = 1.1 µCi/cc

Ba-140 = 4.04E-2 µCi/cc

Comparison of the sample results with radiochemistry samples prior to transient indicate that the concentrations for all radionuclides are higher than normal. This indicates some cladding damage.

The thermocouple readings and corresponding RCS pressure indicate the cladding temperatures are below 1400°F. The cladding failures may not be the result of clad overheating.

Calculation of Cladding Failures

Activity in the RCS (Step 3.1)

RCS density correction factor at 400°F = 0.865 (from Table 7-3)

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

A_i (RCS) for I-131 = 15.4 x 83028 x 3.785E-3 x 0.865 = 4.19E3 Ci

Since the post-transient concentration is significantly greater than the RCS concentration prior to the transient, normal concentrations are ignored.

Radionuclide	Total RCS activity (µCi/cc)		Activity released during transient (Ci)
	After transient	Prior to transient	Step 3.1
I-131	15.4	0.3	4.19E3
I-133	3.7	0.26	1.01E3
Xe-133	99	23	2.69E4
Kr-88	1.1	0.19	2.99E2
Ba-140	0.0404	5.2E-4	1.10E1

Estimation of Power Correction Factors (Step 5.)

$$\begin{aligned} \text{PCF (I-131)} = & (2360 (1\text{-}e^{-0.086 \times 70}) e^{-0.086 \times 30} \\ & +2772 (1\text{-}e^{-0.086 \times 20}) e^{-0.086 \times 10} \\ & +2360 (1\text{-}e^{-0.086 \times 6}) e^{-0.086 \times 4} \\ & +1940 (1\text{-}e^{-0.086 \times 4}) / 2772 (1\text{-}e^{-0.086 \times 100}) \\ & = 0.86 \end{aligned}$$

Using time weighted average power level

$$\text{PCF(I-131)} = \frac{2772 \times 20 + 2360 \times 6 + 1940 \times 4}{2772 \times 30} = 0.93$$

Because the difference is approximately 7% in order to save computation time, time weighted average power level formula could be used.

Radionuclide	PCF
I-131	0.93
I-133	0.70
Xe-133	0.93
Kr-88	0.70
Ba-140	0.93

Calculation of core fraction released (Step 6.)

$$\text{I-131} = \frac{4.19\text{E}3}{7.71\text{E}7 \times 0.93} = 5.84\text{E-}5$$

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Radionuclide	Core inventory at full power (Table 7-1) Curies	Activity released to RCS Curies	Core fraction released
I-131	7.71E7	4.19E3	5.84E-5
I-133	1.56E8	1.01E3	9.25E-6
Xe-133	1.51E8	2.69E4	1.92E-4
Kr-88	5.18E7	2.99E2	8.25E-6
Ba-140	1.33E8	1.1E1	8.89E-8

Estimation Of Cladding Failed (from Figure 7-1) (Step 7.)

I-131 - 0.45%

I-133 - 0.66%

Xe-133 - 0.47%

Kr-88 - 0.89%

Ba-140 - 0.49%

Based on the above, approximately 0.4 to 0.9% cladding failures have occured during the transient.

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Example 3

In this example, it is assumed that plant experienced an accident in which a significant quantity of water was accumulated in the sump and the containment high range radiation monitors indicate high radiation in the containment. There has been no release of radioactivity to the environment. Fifty percent of the core exit thermocouples indicated temperatures greater than 950°F at a corresponding RCS pressure of 1400 PSIG. The samples from RCS, sump and containment are taken at three hours following the accident and counted after one hour. The following are sample results:

Radionuclide	RCS µCi/cc	Sump µCi/cc	Containment µCi/cc
I-131	1.2E4	3.84E3	9E-2
I-133	1.4E4	4.6E3	1E-1
Xe-133	-	-	1.6E2
Kr-88	-	-	1.92E1
Ba-140	57	12	

The following are sample conditions:

Sample	Source Parameters	Sample Parameters
RCS	Temp 300°F	Temp 100°F
Sump	Temp 200°F	Temp 100°F
Containment	Press 20 psia Temp 200°F	Press 14.7 psia Temp 100°F

Containment radiation monitors indicate 5E4 R/hr at one hour following the accident.

The containment hydrogen monitors indicate hydrogen concentration of 5% by volume.

Estimated volume of water in the sump is 250,000 gallons.

RCS concentration after correcting for decay. (Step 1.)

I-131 = 1.2E4(e3.59E-3 x 4) = 1.22E4 µCi/cc

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Radionuclide	RCS (μCi/cc)	Sump (μCi/cc)	Containment (μCi/cc)
I-131	1.22E4	3.9E3	9.14E-2
I-133	1.6E4	5.27E3	1.15E-1
Xe-133	-	-	1.63E2
Kr-88	-	-	5.19E1
Ba-140	5.75E1	1.21E1	-

Presence of high Ba-140 concentrations indicate that some portion of the core might have overheated or melted. See the previous example for the power correction factors used in this example.

Calculate total activity released during accident for each isotope (Step 3.)

I-131 RCS = 1.22E4 x 83028 x 3.785E-3 x 0.924 = 3.54E6 Ci

I-131 Sump = 3.9E3 x 250,000 x 3.785E-3 x 0.97 = 3.58E6 Ci

$$\text{I-131 CB} = 9.14\text{E-}2 \times 8.01\text{E}4 \times \left[\frac{20(100+460)}{14.7(200+460)}\right] = 8.45\text{E}3 \text{ Ci}$$

Radionuclide	RCS (Ci)	SUMP (Ci)	CB (Ci)	TOTAL (Ci)
I-131	3.54E6	3.58E6	8.45E3	7.13E6
I-133	4.65E6	4.84E6	1.06E4	9.50E6
Xe-133	-	-	1.51E7	1.51E7
Kr-88	-	-	4.8E6	4.8E6
Ba-140	1.67E4	1.11E4	-	2.78E4

Since post-accident concentrations are significantly larger than normal concentrations, ignore normal concentrations.

Calculate core release fractions (Step 6.) and % fuel failures (Step 7.)

$$\text{I-131} = \frac{7.22\text{E}6}{0.93 \times 7.71\text{E}7} = 1.01\text{E-}1$$

Radionuclide	Fraction Released	Cladding Failure %	Fuel Over Heat %	Fuel Melt %
I-131	9.94E-2	100	17 to 25	10 to 13
I-133	8.70E-2	100	15 to 22	9 to 11
Xe-133	1.07E-1	100	18 to 27	11 to 14
Kr-88	1.32E-1	100	22 to 33	13 to 17
Ba-140	2.25E-4	100	11 to 23	<0.23

CORE DAMAGE ASSESSMENT USING PASS SAMPLE DATA (Cont.)

Because of the overlap in the release fractions and the fact that a small fraction of overheat or melt would dominate the release, engineering judgment is used to apportion the releases to each type of failure mechanism. Based on Ba-140 release fraction, it can be seen that this release corresponds to less than 0.1% to 0.2% of fuel melting or less than 11% to 20% fuel overheating. Based on core exit thermocouple readings and Attachment 4, it can be seen that the clad temperature did not exceed 1800°F. However, cladding temperatures exceeded cladding failure temperatures and approached temperatures that may result in fuel overheating. Based on the hydrogen concentration measurements, it can be determined that 44% of zirconium in the core has undergone metal water reaction. This indicates significant overheating of core. Based on this available information, a best estimate for core condition can be made.

1. Major fuel cladding damage (greater than 50%) occurred (indicators: release fraction, core exit thermocouple, containment hydrogen concentration).

2. Ten to 20% of fuel overheated (indicators: Ba-140 release fraction (<0.21% fuel melt), core exit thermocouple readings).

3. No melting of fuel (indicators: Ba-140 release fraction, core exit thermocouples).

Core damage estimation from containment high range monitor

The dose rate of 5E4 R/hr at one hour is in the region above curve C which indicates greater than 100% gap activity plus fuel melt.

Use of Radiation Monitors will provide a quick estimate of fuel condition, but may not provide as accurate an assessment as post accident sampling.

PASS SAMPLE CORE DAMAGE ASSESSMENT FORM

PASS SAMPLE CORE DAMAGE ASSESSMENT FORM

Date/ Time	Isotope	RCS Activity Prior to Event (µCi/cc)	Sample Activity (µCi/cc)	Sample Activity Correct-ed for Decay (µCi/cc)	Sample Activity > Normal? (Yes/No)	Total Activity Released During Event Based on all Samples (Ci)	Power Correc-tion Factor	Core Fraction Released (All Samples)	Core Damage Evaluation Cladding Failures (%)	Fuel Overheat (%)	Fuel Melt (%)	Other Data and Comments

Summary of Core Damage Assessment Circle As Applicable

No Damage			
Cladding Damage	<10%	10-50%	>50%
Fuel Overheat	<10%	10-50%	>50%
Fuel Melt	<10%	10-50%	>50%

Attachment 9: Reactor Coolant Pump Restart Criteria

Purpose:

Certain transients can cause regions of the Reactor Coolant System (RCS) to become relatively pure water, containing little or no boron. The restart of a Reactor Coolant Pump (RCP) could flush this deborated water into the core and cause a return to criticality. The conditions which could lead to this are:

1. Boiler Condenser (Reflux) Cooling can cause deborated water to accumulate in the RCP bowl and the lower Steam Generator head and tubes.
2. A SBLOCA can cause the RV Downcomer to become partially voided. HPI flow entering the vessel can cause condensation of steam in this region.
3. Hot leg voiding results in steam voids which can be collapsed by RCP restart.

Since these events all involve a loss of Subcooling Margin (SCM), Reactor Coolant Pump operation will be restricted following a loss of SCM event until certain criteria have been met. Meeting these criteria will avoid a potential return to criticality on RCP start. The criteria are not imposed on RCP operation during mitigation of Inadequate Core Cooling (as defined in DB-OP-02000, Figure 2) since core cooling is the immediate concern and the Reactor Coolant System is highly voided, significantly decreasing the likelihood of a recriticality. The purpose of this Attachment is to provide the criteria for RCP restart for the Technical Support Center (TSC) personnel, so that they can determine if RCP restart is appropriate and provide a recommendation regarding RCP restart to the Control Room when requested.

For additional information on localized boron dilution mechanisms, see Framatome Technologies letter INS-97-1040, dated March 18, 1997 from Len Barker to the B&W Owners Group Operator Support Committee.

Procedure:

A. Loss of RCS Flow, No Loss of Subcooled Margin (SCM)

<u>IF</u>
RCS forced flow was terminated,
<u>AND</u>
Subcooled Margin (as indicated on the Subcooling Monitors using the core exit thermocouples) has stayed greater than +20 ^{0}F,
<u>THEN</u>
RCPs may be restarted as soon as the cause of the loss of flow condition has been resolved. No localized boron dilution will have occurred in the case where the RCS remains subcooled at all times.

B. Loss of RCS Flow Due to Loss of Subcool Margin, No Makeup/HPI Cooling In Progress

<u>IF</u>
1. Stable subcooled two loop natural circulation has been verified to exist for at least 60 minutes cumulative (may be interrupted for periods as long as SCM is maintained and no boron dilution mechanisms have occurred),
<u>AND</u>
Subcooled Margin (as indicated on the Subcooling Monitors using the core exit thermocouples) is greater than or equal to + 50^{0}F,
<u>AND</u>
RCS pressure is greater than 185 PSIG,

<u>OR</u>

2. Stable, subcooled single loop natural circulation has been verified to exist for at least 210 minutes cumulative (may be interrupted for periods as long as SCM is maintained and no boron dilution mechanisims have occurred),
<u>AND</u>
Subcooled Margin (as indicated on the Subcooling Monitors using the core exit thermocouples) is greater than + 45^{0}F,
<u>AND</u>
RCS pressure is greater than or equal to 135 PSIG,
<u>AND</u>
HPI flow has been maintained at or above 900 gpm during the full 210 minutes of natural circulation,
<u>THEN</u>
Start a RCP in a loop that has had natural circulation for the required time.

C. Loss of RCS Flow Due to Loss of SCM, Makeup/HPI Cooling in Progress

1. IF
   Makeup/HPI cooling is in progress,
   AND
   Neither Steam Generator has been fed while RCPs were off,
   AND
   Subcooled Margin (as indicated on the Subcooling Monitors using the core exit thermocouples) is greater than + 20^0F,
   AND
   RCS pressure is LESS THAN 485 PSIG,
   THEN
   A RCP may be started in any loop.

2. IF
   Makeup/HPI cooling is in progress,
   AND
   Steam Generators have been fed while RCPs were off,
   GO TO
   Section B, Loss of RCS Flow Due to Loss of Subcooled Margin, No Makeup/HPI Cooling in Progress, above.

Attachment 10: Boron Dilution Flowpath Concerns

Purpose:

Certain Small Break Loss of Coolant Accidents (SBLOCAs) can leave the Reactor Coolant System (RCS) pressurized above the pressure of the Containment. Connecting the RCS to the Containment, via the Low Pressure Injection/Decay Heat Removal (LPI/DHR) system and the emergency sump in this condition could jeopardize the integrity of the long-term DHR function and the boron dilution function of the LPI/DHR System. This is particularly a concern when LPI/DHR System has been operated in the LPI mode, but is realigned to support a boron dilution flowpath. The purpose of this attachment is to inform Technical Support Center personnel of the conditions that could lead to compromise of the LPI/DHR System and how to deal with those conditions until boron dilution can be safely established.

DB-OP-02000; RPS, SFAS, And SFRCS Trip and SGTR Accident, Step 10.14.1 directs the plant operators to establish a long-term boron dilution flow path using the decay heat drop line. By opening DH-11 and DH-12, a flowpath between the RCS hot leg and the LPI/DHR pump suction is established through the normally open DH-10 and/or DH-26 bypass line isolation valves. These valves allow a small amount of flow around DH-1517 and DH-1518. The small lines include flow orifices to allow monitoring the flow through the line. However, the pressure drop that occurs across the orifice can result in some of the liquid possibly flashing to steam. This steam, or saturated mixture, could then enter the suction of the LPI/DHR pumps, where cavitation may occur. The cavitation could damage the pumps. There is insufficient Net Positive Suction Head (NPSH) provided by the water in the Emergency Sump to preclude this cavitation. It is imperative that the pumps be protected from damage so that the long-term LPI/DHR and boron dilution functions will be maintained.

If a Large Break LOCA has occurred, the pressure of the RCS is equal to the containment pressure. Therefore, the pressure drop across the bypass line flow orifice is small. No flashing or cavitation of liquid is expected in the drop line boron dilution flowpath under these conditions. Therefore, if it can be determined if a LBLOCA has occurred, there is no restriction on initiating a long-term boron dilution flow path. The TSC personnel only need to inform the plant operators that a LBLOCA has occurred and that the flowpath can be initiated. Symptoms of a LBLOCA are provided below to aid in this decision.

During a Small Break LOCA (SBLOCA), if the RCS pressure remains greater than Containment pressure, the large differential pressure would lead to the flashing and cavitation problem discussed above. In this event, the TSC personnel will not grant permission to establish the LPI/DHR boron dilution flowpath. Other mechanisms will aid in maintaining a non-saturated boric acid condition.

It is noted that, if the core is in a subcoooled condition, boric acid will not concentrate in the core. Consequently when subcooled conditions exist, there is no actual need to establish a boron dilution flowpath. However, DB-OP-02000 gives adequate direction for this condition, so that no specific action is required for this condition.

In the event of a SBLOCA, boron dilution will not be required as promptly as with a LBLOCA. Consequently, time for depressurization of the RCS is available. This also allows time for re-establishment of Steam Generator cooling, thereby obviating the need for boron dilution. Care should be exercised though, due to the concerns addressed in Attachment 9 of this procedure.

Procedure:

1. Determine if a Large Break LOCA has occurred
   Symptoms:
   a) Core exit thermocouples indicating within 20 degrees of containment saturation temperature (this is within the thermocouple accuracy).
   b) SFAS Level 3 (RCS Low Low Pressure) or Level 4 (CTMT Pressure HI HI) is actuated.
   c) RCS pressure steadily decreases (without a plateau) from the time of the break.
   d) RCS pressure reaches a steady, low value (less than 70 psig) quickly (<1 minute).
   e) RCS pressure stable by the time transfer of pump suctions from the BWST to the Emergency Sump is required.

2. IF

   A LBLOCA has occurred

   THEN

   Inform the Control Room that long-term boron dilution, per DB-OP-02000, Step 10.14.1, may be initiated as required.

3. IF

   A Small Break LOCA has occurred,

   AND

   RCS pressure is greater than Containment pressure,

   THEN

   Continue RCS cooldown and depressurization as directed by operations procedures. Do NOT grant permission to initiate long-term boron dilution via the decay heat drop line.

4. IF
A Small Break LOCA has occurred,

AND
RCS is equal to OR less than Containment pressure,

THEN

Grant permission to initiate long-term boron dilution via the decay heat drop line, in accordance with DB-OP-02000, Step 10.14.1.

COMMITMENTS

Step Number	Reference	Comments
Entire Procedure	TERMS O 06880	Use of diverse core damage indicators
Entire Procedure	TERMS Q 03111	Emergency Organization and Individual authority
Attachment 3	TERMS O5648	Core damage as a function of Containment Radiation Readings
Step 6.3.7, Attachment 10	TERMS A 19420 TERMS A 19426	Information to discriminate between large break and small break LOCAs.
Step 6.3.7, Attachment 10	TERMS A 19423	Provide guidance on establishing a boron dilution flowpath.

END