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AUTH.NAME AUTHOR AFFILIATION
UTLEY,E.E. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Forwards response to NRC 810521 ltr re safety evaluation for environ qualification of safety-related electrical equipment.Methodology used to prepare response based on info obtained through NRC 810707-10 meeting & telcons. see RPT

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	WILLIAMS,M.H.	22	1	1				
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CP&L
Carolina Power & Light Company

August 31, 1981

FILE: NG-3514(R)

Serial No.: NO-81-1432

Office of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
United States Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED SEP 03 1981 U.S. NUCLEAR REGULATORY COMMISSION

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED
ELECTRICAL EQUIPMENT

Dear Mr. Varga:

Please find attached Carolina Power & Light Company's (CP&L) response to your letter of May 21, 1981 which provided the NRC's Safety Evaluation for the Environmental Qualification of Safety-Related Electrical Equipment at H. B. Robinson, Unit No. 2. The methodology used in preparing this response is based on information obtained at the NRC Meeting on this subject held July 7 - 10, 1981, and through telephone conversations held with your staff.

If you have any questions on this response, please contact our staff.

Yours very truly,

M A McDuffie

for E. E. Utley
Executive Vice President
Power Supply and
Engineering & Construction

A048
S
1/1

JJS/lr (0060)

cc: J. D. Neighbors (NRC)

8109040119 810831
PDR ADOCK 05000261
P PDR

teville Street • P. O. Box 1551 • Raleigh, N. C. 27602

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261, LICENSE NO. DPR-23

RESPONSE TO THE NRC ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT - SAFETY EVALUATION REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION EQUIPMENT QUALIFICATION BRANCH FOR CAROLINA POWER & LIGHT COMPANY H. B. ROBINSON UNIT 2.

AUGUST, 1981

CAROLINA POWER & LIGHT COMPANY
RALEIGH, N. C.

SECTION I

Response to the NRC Environmental Qualification of Safety-Related Electrical Equipment - Safety Evaluation Report by the Office of Nuclear Reactor Regulation Equipment Qualification Branch for Carolina Power & Light Company H. B. Robinson Unit 2, Docket No. 50-261.

1.0 Introduction

Carolina Power & Light Company (CP&L) has reviewed the Safety Evaluation Report (SER) issued by the NRC Office of Nuclear Reactor Regulation Equipment Qualification Branch, and transmitted by NRC letter dated May 21, 1981, and submits this response within the prescribed 90-day period stating our positions, critiques, and actions to be taken in meeting the requirements of NRC IE Bulletin 79-01B and subsequent revisions. The basis for our evaluations to date are the DOR Guidelines as applied to an operating license (OL) plant and limited to electrical equipment in the harsh environments as defined within the DOR Guidelines and subsequent revisions.

The format of our response is in keeping with the document entitled Expected Licensee Response to the Equipment Qualification SERs introduced during the NRC's Equipment Qualification Meeting held July 7 - 10, 1981 at Bethesda, Maryland. It is divided into four sections: Section I, Response to the NRC Evaluation, follows the issued SER paragraph numbering and titling which requested additional information or solicited some position by CP&L; Section II, Updated Summary Sheets, contains a re-submittal of System Component Evaluation Work Sheets (SCEWS) originally sent as part of our Revision 3, 90-day Report, dated February 3, 1981 to aid in evaluation of our position on listed deficiencies per the SER; Section III, Proposed Corrective Actions for Outstanding Items, contains a listing of actions already taken, as well as a restating of actions previously reported within our 90-day responses pertaining to equipment upgrading and component changeout when required; Section IV, Justification for Continued Operation, reiterates CP&L's original position and NRC's acceptance concerning continued operation of H. B. Robinson Unit 2 during the continued efforts being expended to meet NRC IE Bulletin 79-01B, its revisions, and existing licensing orders pertaining to equipment qualification.

2.0 Background

This SER paragraph and subparagraphs provide information only and require no response.

3.0 Staff Evaluation

This SER paragraph provides information only and requires no response.

3.1 Completeness of Safety-Related Equipment

The reference to display instrumentation requirements within the SER appears to limit the current need to instrumentation within the harsh environment and/or as mentioned in the LOCA and HELB emergency procedures. Instrumentation within harsh environment would be limited to those transmitters, switches, and RTD's already reported on SCEWS.

At present, CP&L is to develop new Emergency Instructions incorporating TMI emergency guidelines as part of a TMI action item requirement. These new instructions are scheduled for 1983 implementation. When reviewed and accepted for plant operational use, they will be reviewed for components and display instrumentation within plant harsh environment. If new instrumentation is required to meet these emergency procedures or if additinal instrumentation is referenced by these procedures, a list will be compiled and SCEWS's provided to the NRC.

3.2 Service Conditions

The SER requires the licensee to verify that the containment spray system is not subjected to a disabling single-component failure and therefore satisfies the requirements of Section 4.2.1 of the DOR Guidelines. H. B. Robinson FSAR Section 6.4.3, Design Evaluation, describes the capabilities of the containment spray system and addresses the single failure analysis for this system within Table 6.4-4 entitled, Single Failure Analysis - Containment Spray System, (See Appendix A).

As this system is not subject to a single component failure affecting its performance the MSLB accident environment is not the limiting parameter for qualification per the DOR Guidelines Section 4.2.1. To further support this conclusion a study performed for NRC IE Bulletin 80-04, Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition indicates a maximum containment pressure of 34.4 psig and a temperature of 257^{o}F for the feedwater augmented MSLB.

3.3 Temperature, Pressure, and Humidity Conditions Inside Containment

The SER questions the pressure value as stated in our submittals and found within the H. B. Robinson FSAR. Apparently, a comparison with other plant's specified pressure values is used as the basis for this questioning. As reported in our prior responses to IE Bulletin 79-01B, the H. B. Robinson containment volume is 2.1 X 10^6 ft^3. This effectively accounts for both a lower pressure value and radiation level when defining LOCA parameters. Therefore, it is not practical to use comparisons in determining H. B. Robinson adequacy. Our FSAR has been reviewed by the U. S. Atomic Energy Commission and after several exchanges requiring amendments to this document, a Safety Evaluation Report was issued on May 18, 1970 accepting the LOCA evaluation as presented for operation. Further review of LOCA conditions was performed by the AEC when the plant requested permission to increase power. A Safety Evaluation Report was issued on May 20, 1974 accepting the LOCA conditions and approving power increase. Per the DOR Guidelines Section 4.1.1 Service Conditions Inside Containment for a Loss of Coolant Accident (LOCA), Temperature and Pressure Steam Conditions, the FSAR analysis for containment

temperature and pressure conditions is used for establishing the qualification of electrical equipment located within containment.

The service condition analysis was performed by Westinghouse Electric Company and the method utilized was reported to the AEC and subsequently enclosed as a Status Report by the Directorate of Licensing, dated October 15, 1974 within the Safety Evaluation Report by the Directorate of Licensing, U. S. Atomic Energy Commission for CP&L H. B. Robinson Unit #2, dated December 27, 1974. (See Appendix B, Westinghouse ECCS Evaluation Model).

Of further note, the H. B. Robinson calculated maximum pressure associated with a double-ended break is 38 psig (peaking at 12 seconds after LOCA and reducing to 32 psig after 3000 seconds). The stated 42 psig within our IE Bulletin 79-01B responses refers to the containment design pressure; therefore, a margin of approximately 11% above the conservative value of the blowdown peaks is available. H. B. Robinson Unit 2 FSAR, Section 14.3.4, Containment Integrity Evaluation, contains data and parameters associated with containment pressure. See Figure 14.3.4-2, Containment Pressure Transients for a Range of Break Sizes, Figure 14.3.4-4, Structural Heat Transfer Coefficient, and Figure 14.2.4-6, Containment Capability Study, All Available Energy for graphic presentation of H. B. Robinson LOCA profiles.

Per the above, we do not believe it is justified to recalculate the LOCA pressure and temperature profiles. The conservatisms involved with the values and the tested values for equipment as recorded on the System Component Evaluation Work Sheets indicate there are no conflicting or questionable pressure and/or temperature values involved.

Addressing stratification within Containment, the upper regions where stratification may affect temperature do not contain any instrumentation or equipment related to IE Bulletin 79-01B. Only the containment fans are located on the crane deck level (elevation 275′) and their test temperature exceeds LOCA temperature by 16% (11% if the saturation temperature associated with LOCA is considered).

3.4 Temperature, Pressure, and Humidity Conditions Outside Containment

As stated in the SER, ambient temperature conditions have been used in some areas outside containment. As H. B. Robinson 2 has master listed equipment located on the turbine deck which is an open air area, it is impractical to utilize an arbitrary value such as ambient saturation temperature for qualification purposes. Therefore, the System Component Work Sheets listing turbine Deck Area location will not be summarily revised.

Considering the Auxiliary Building, the following quote from the Report entitled: H. B. Robinson No. 2 Postulated Pipe Failure Analysis Outside of Containment, Section 10.0 Description of Compartment Environmental Effects Analysis, indicates only one enclosed volume subject to pressure and temperature buildup following rupture. This is the pipe penetration gallery. An analysis indicates a calculated pressure buildup of 0.2 psi and a maximum temperature increase of 2.4^{o}F. This consequence is the result of the limiting postulated steam generator blowdown line rupture. These limiting

environmental conditions will have no effect on the structural adequacy of the auxiliary building or on plant operation. It would be arbitrary to assume the ambient saturation temperature for equipment qualification purposes. Review of the System Component Work Sheets designated within the Reactor Auxiliary Building show qualification temperatures well in excess of the area requirement. Therefore, the System Component Work Sheets for the Reactor Auxiliary Building will not be changed to reflect a screening number when actual conditions have been calculated.

3.5 Submergence

The safety-related level transmitters (LT 459, LT 460, and LT 461) referred to in this paragraph provide pressurizer water level indication and are mounted in an instrument rack on the shield wall at elevation 230 ft within containment. These transmitters have been replaced with Rosemount Model 1153A transmitters during the August, 1980 outage at H.B. Robinson Unit 2. Instruction was given to remount as high as possible and practical in the instrument rack. A new measurement is required for re-evaluation of submergence level for these instruments. Additional study indicates their need time within the LOCA to be the first 30 minutes.

A study will be performed to evaluate the effect of new mounting, the rate of flood to determine useful time, and the effects of new emergency procedures on these instrumentation reading requirements. At this time, the statement that their assumed failure under submergence will not affect accident mitigation is still a valid one. This study will be completed by the date established by the NRC for completion of qualification of safety-related electrical equipment.

Submergence of equipment outside of containment will also be studied. New modification of Auxiliary Building areas due to fire protection requirements need to be evaluated to establish drain paths and/or water accumulations. Existing reports indicate no detrimental water buildups due to HELB or small pipe breaks, but they do not account for building modification performed in recent months. A report on submergence of safety-related electrical equipment within harsh environment areas in the Auxiliary Building will be completed and submitted by the date established by the NRC for completion of qualification of safety-related electrical equipment.

3.6 Chemical Spray

The chemical spray consists of sodium hydroxide, boric acid, and refueling water. Mixing of the refueling water from the refueling water storage tank, the boric acid from the boric acid tank, the borated water contained within the accumulators, and primary coolant will bring the concentration of sodium hydroxide in the containment to approximately 0.6 weight percent solution caustic and 1.7 weight percent boric acid. This maintains a pH of at least 9.3. Spray additive eductors are designed to provide enough sodium hydroxide in the mixture so as not to exceed pH 10 during the injection phase.

3.7 Aging

Electrical equipment identified within harsh environments were reviewed against existing test data to determine if a 40 year life was established. When so established, this was recorded on the Component Evaluation Work Sheet. Also noted was the period within the test program when aging was addressed if this information was available from the test reports.

Section 4.0, Conclusions, of our 90-day response, Revision 3, dated February 1, 1981 reviewed the actions to be taken when less than 40 year life is realized for components and/or equipments. These actions are again summarized within Section III of this report.

To assure that aging is adequately covered under equipment qualification, it is our intent to establish a qualified life for all components in a harsh environment and to identify the component part or parts that limit qualified life. This will be the plant data base into which new safety equipment required by future modifications and replacement parts required by operation will be entered when evaluated and approved. This data base will form the baseline for a component inspection and replacement program. Comparisons of predicted age vs. actual age will modify the limiting life of either components or equipments. Factored into this program will be supportive elements such as elastomers, lubricants, mountings, and supplier conditions which are necessary to assure both operational life and qualification level.

This program will be compiled, evaluated, reviewed, approved, and operational by the date established by the NRC for completion of qualification of safety-related electrical equipment.

3.8 Radiation (Inside Containment)

The SER second paragraph for Section 3.8 states that the values submitted within CP&L's H. B. Robinson 90-day response do not envelop the DOR Guidelines (4×10^7 rads) requirements and therefore are not acceptable.

Radiation values listed on the submitted System Component Evaluation Work Sheets within the specification column reflect a series of calculations based upon containment volume, internal shielding, and instrumentation/equipment location. These calculations follow the procedures referenced as acceptable in the DOR Guidelines and provided within Appendix B. Sample calculations and representative nomogram use were presented as Appendix A within CP&L 90-day submittals; Rev. 0, Rev. 1, Rev. 2, and Rev. 3.

Of note is the reduced level radiation number due to oversized containment volume (2.1×10^6 ft^3) and shield wall thickness (36 inches) when using the nomograms. Each instrument or equipment as represented on the work sheets was dimensioned by level within containment or Reactor Auxiliary Building and located by compartment or shield wall to determine the maximum radiation level experienced under LOCA conditions. This figure was added to the normal operating radiation dose (40 year life) and a margin assigned. Section 1.3.2 of the H. B. Robinson 90-day Rev. 3 report provides additional insight into the assigned radiation levels. Individual dosages used on the

work sheets are summarized and listed in Table 1.3.3 of the above mentioned report. For review purposes, response submitted figures and tables are included in this section to aid in evaluation of our radiation assignments.

When operating time for equipment/instrumentation was less than one (1) hour, a minimum of one (1) hour was picked for establishing dosage reduction based on the nomogram entitled 30 Day Dose Connection Factor vs. Time Required to Remain Functional (HRS). This should establish sufficient margin and encompass existing test data.

For items located close to sump water flooding levels an additional radiation dosage was assigned based on actual operating time. As stated in Table 1.3.3 Notes (8) & (9), data used can be found in NUREG-0588, Appendix D, Table D-8, Containment Sump Gamma Dose Rates and Integrated Dose Versus Time.

Beta radiation was considered using Appendix D, Table D-6, Beta Dose Rates and Integrated Doses at the Containment Center Versus Time in Air. Based on the time of operation, equipment location, shield wall absorption, compartment wall absorption, insulation thickness, instrumentation housing absorption, motor case shielding, et.al., beta contribution is less than 10% of the total gamma dose experienced by the listed equipment. This is a conservative assumption based on the DOR Guidelines requirement of the beta dose to be less than or equal to 10% of the total gamma dose to which an item of equipment has been qualified.

The above is the basis, assumptions, and basic analysis of the option chosen to justify the choice of lower service conditions than the generalized screening radiation service value stated in the SER and presented at the NRC 79-01B meeting held in Bethesda, Maryland on July 7-10, 1981. Sample calculations as included in our 90-day, Revision 3 response to IE Bulletin 79-01B are repeated as Appendix C of this report.

4.0 Qualification of Equipment

The H. B. Robinson #2 SER for Environmental Qualification of Safety-Related Electrical Equipment, dated May 21, 1981, separated master list hardware submitted by 90-day responses Revision 0, Revision 1, Revision 2 and Revision 3 into three (3) categories: first, equipment requiring immediate corrective action (Appendix A); second, equipment requiring additional qualification information and/or corrective action (Appendix B); third, equipment considered acceptable (Appendix C). Descriptive NRC evaluation within each category was addressed in SER sections 4.1, 4.2, and 4.3 respectively. The following response to each category evaluation will collect data previously reported within the varied CP&L 90-day responses and relay current status of on-going programs to support our interpretation of the listed equipment qualification status.

4.1 Equipment Requiring Immediate Corrective Action (Appendix A)

Conductor Pigtails of Electrical Penetrations, B-1, B-2, B-5, B-9, C-1, C-2, C-3, C-4, C-6, C-8, C-9, D-1, D-2, D-3, D-5, D-8, D-9
Manufacturer: Continental Wire and Cable Company
Jacket Material: PVC

Test Laboratory: Wyle Laboratories, Huntsville, Alabama
Qualification Test Plan: 45307-1 dated January 12, 1981

Status: All test sequences stated within the Final Qualification Test Plan (45307-1) were completed on July 17, 1981. The test cables, test control cables and splices were removed from the LOCA test chamber on July 20, 1981 and are undergoing evaluation. A test report is scheduled for issuance September 1, 1981. Review of this report will be made before final disposition of this matter will be made to the NRC. Preliminary data indicates no failure of the cable or splices which would require any immediate corrective action by the plant. Therefore the statement found at the end of the SER Appendix A. . . . "Licensee integrated dose assessment provides justification for contained operation until testing is completed and analyzed." remains valid.

4.2 Equipment Requiring Additional Information and/or Corrective Action (Appendix B)

The bulk of master list equipment was placed in this category and our review of deficiencies is based upon the following:

- Deletion of deficient listed equipment due to replacement programs carried out and reported in our 90-day, Revision 3 response dated February 1, 1981.

- Re-evaluation of CP&L's updated submittals by NRC Region II, Atlanta, GA, and reported in Environmental Qualification of Safety-Related Electrical Equipment IEB 79-01B. Technical Evaluation Report - Docket No. 50-261; Plant: H. B. Robinson 2, dated November 6, 1980 and Revised November 11, 1980.

  Telecons with NRC, Bethesda, MD reviewing personnel July 29, 1981, et al.

Equipment deleted due to replacement programs completed at the plant. (Deficiency listings will not be addressed as these equipments are no longer in place or use at the plant).

Equipment Description	Manufacturer	Model No.	Location
1. level transmitter	Fisher & Porter	10B2496	Containment
2. pressure transmitter	Fisher & Porter	50EP1041BCXA-NS	Containment
3. flow transmitter	Rosemount	1151	Containment
4. level transmitter	Fisher & Porter	13D2495	Containment
5. solenoid valve	ASCO	LB8211C32	Containment
6. solenoid valve	ASCO	LB8316B25	Containment
7. solenoid valve	ASCO	LB8316B15	Containment
8. solenoid valve	ASCO	LB8316B14	Containment
9. level transmitter	Fisher & Porter	13B2496	Containment

Equipment reported as qualified within NRC TER Revision 1, dated November 11, 1980:

Equipment Description	Manufacturer	Model No.	Location
1. pump motor	Westinghouse	506UPZ	outside containment
2. motor operator	Limitorque	SMB-00	containment
3. motor operator	Limitorque	SMB-00	outside containment
4. motor operator	Limitorque	SMB-1	outside containment
5. flow transmitter	Fisher & Porter	10B2496 PBBABBB	outside containment
6. pressure transmitter	Fisher & Porter	50EP1041 BCXA	outside containment
7. fan motor	Westinghouse	685.5-S	containment
8. cable	Continental Wire & Cable	CC2115	containment
9. cable	Kerite	HT FR	containment
10. solenoid valve[1]	ASCO	NP831665E	containment
11. solenoid valve[1]	ASCO	NP8316E35E	containment
12. solenoid valve[1]	ASCO	206-381-2U	containment
13. cable splices[2]	Raychem	1000-12N	containment
14. cable splices[2]	Raychem	500-12N	containment
15. cable splices[2]	Raychem	300-12N	containment
16. cable splices[2]	Raychem	200-12N	containment
17. cable splices[2]	Raychem	115-6N	containment
18. cable splices[2]	Raychem	070-6N	containment
19. cable terminals[2] and splices	AMP	53548-1	containment

Notes:

[1]As reported in qualification test programs, components and/or materials will require replacement on a designated schedule to maintain qualification.

[2]These items also included in conductor pig-tail qualification test (Wyle Qualification Test Plan 45307-1) to supply acceptable aging, radiation, and LOCA test data for H. B. Robinson qualification.

The above items were discussed with NRC Region II personnel during their evaluation and many of their concerns were addressed by revision to their original TER, dated October 24, 1980, which stated their acceptance as qualified, within the constraints as noted (i.e. age limited components, replacement schedules, etc.) for this equipment.

Individual deficiency categories which appear generalized are Radiation (R), Aging (A), and Containment Spray (CS). Review of our submitted System Component Evaluation Work Sheets (SCEWS) listed as deficient equipments do not indicate a generic problem with these items. Radiation levels as established within Section 3.8 of this response are below the reported qualification levels for all equipment. Containment spray exposure, as reported per qualification tests performed, covered the plant designated pH and was introduced during the appropriate cycle. Aging is reported per qualification tests performed and represents the 40 year requirements when listed. Items with acknowledged aging deficiencies will be handled per the program outlined within Section 3.7, Aging, of this report.

The items within Appendix B not fully covered by either TER classification or SCEWS's are as follows:

Equipment Description	Manufacturer	Model No.	Location
Motor Operator	Limitorque	SMB-3	containment
Transmitter	Rosemount	1153A	containment
Level Switch	Madison	5602	containment
Silicon Rubber Tape	3M	Scotch 70	containment

The Limitorque motor operators of type SMB-3 are operators with motor brakes. Two are in the master list - V744A and V744B. These operators were required by Westinghouse to be upgraded to special service by Limitorque before installation. Data is in file denoting the requirements and certification that these operators were modified for their intended service. Under review is the currently available SB Model Limitorque Operators which perform similar function without a motor brake. When completed, a recommendation and disposition will be made to the NRC. At the present time, the installed SMB-3 operators are considered adequate for their service and qualified for their environment.

The Rosemount 1153A Transmitters are currently qualified to IEEE 323-1971. As reported in our 90-day responses these transmitters require a changeout cycle of ten (10) years with an O-Ring replacement after each calibration check to maintain their qualification level. CP&L is part of a utilities group which is underwriting transmitter qualification to IEEE-323-1974 standards. As these tests are ongoing, no recommendations or changeout program is being formulated at this time. Upon test completion, CP&L will determine any added transmitter changeout or modification program and report to the NRC its actions.

The level switch reported is for containment sump level indication and has been superceded in place by two GEMS level transmitters Model No. XM36495 as part of the TMI lessons learned action items. This system is currently reported in our response to the TMI Equipment requirement submitted in February, 1981.

The Scotch 70, silicon rubber tape used as part of our splicing procedure has been tested within the containment cable qualification test program reported for Appendix A listed equipment. Complete splice materials, per CP&L splice procedure M-521 (Rev. 0), were installed on both containment cable under test and known qualified control cable to establish qualification per H. B. Robinson environmental parameters. When this test report is reviewed, CP&L will inform the NRC of any additional actions, if any are required concerning Scotch 70 tape.

Included in this section is a re-worked listing representing our equipment list evaluation in response to the issued SER.

4.3 Equipment Considered Acceptable or Conditionally Acceptable (Appendix C)

No equipment was listed in this category within the SER.

CONTAIN PRESSURE TRANSIENTS
FOR A RA OF BREAK SIZES (MIN. SAFETY FEATURES OPERATING)

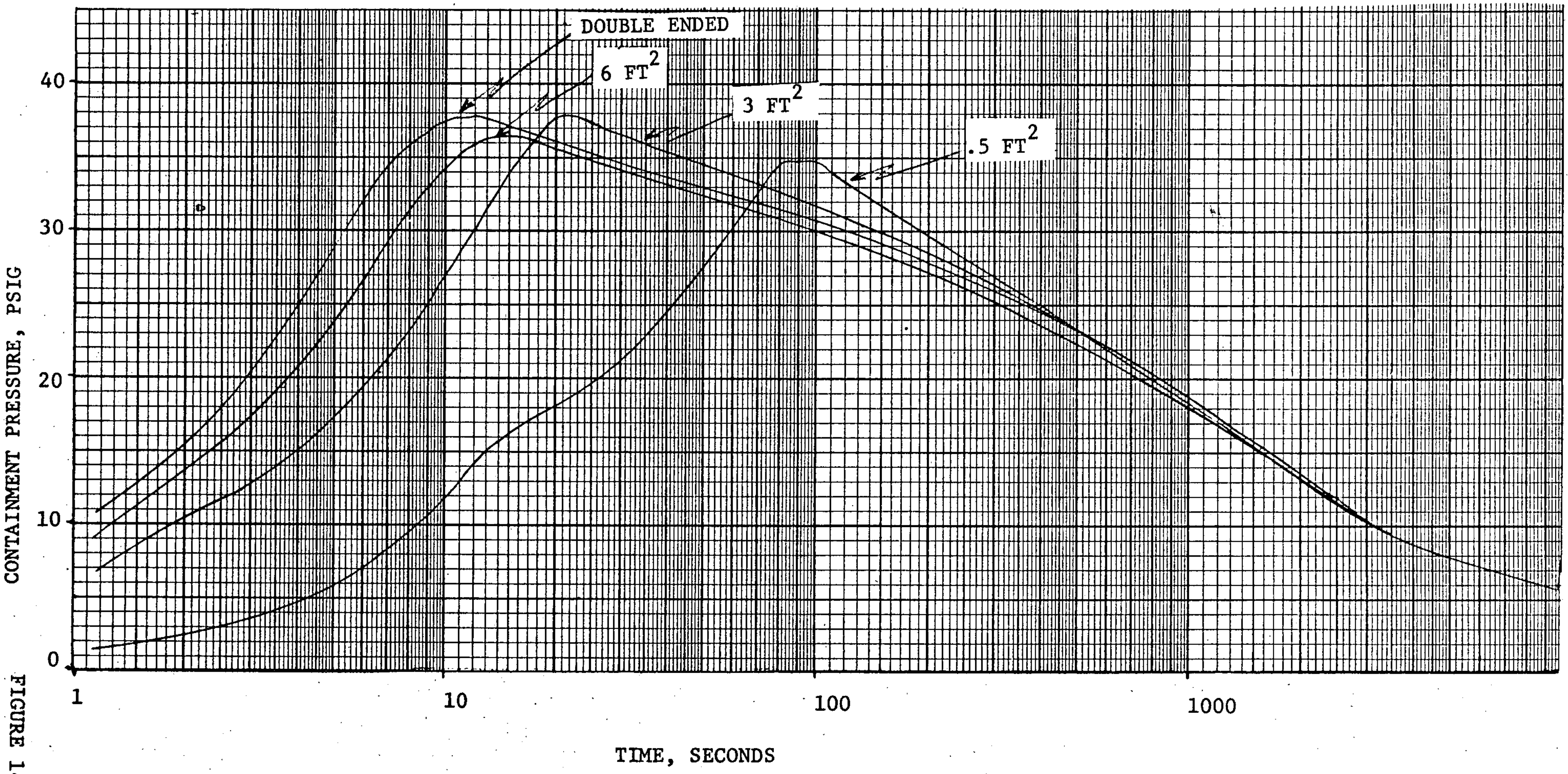


FIGURE 14.3.4-2

STRUCTURAL HEAT TRA COEFFICIENT

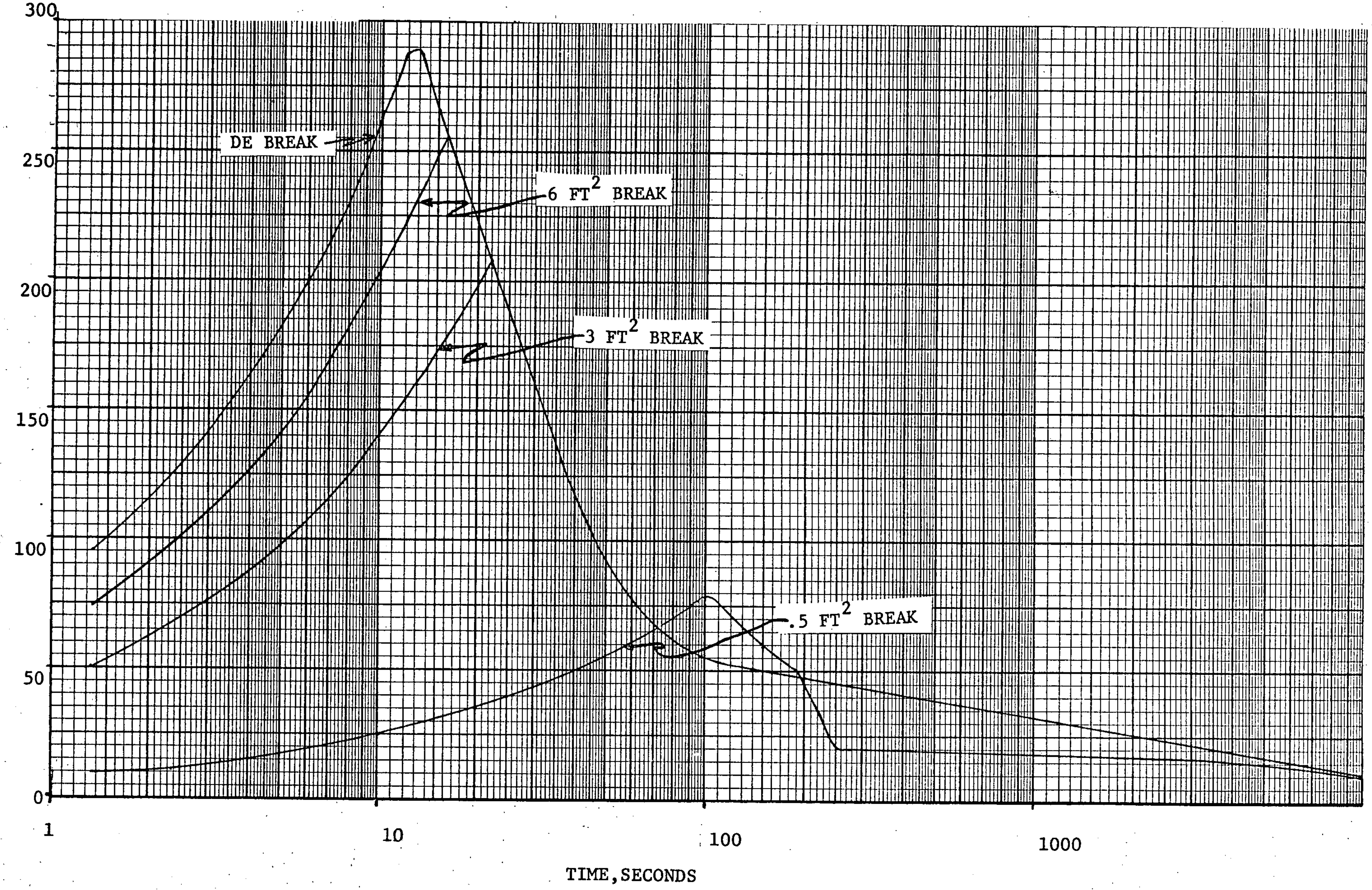


FIGURE 14.3.4-4

CONTAINMENT CAPABILITY STUDY

ALL AVAILABLE ENERGY

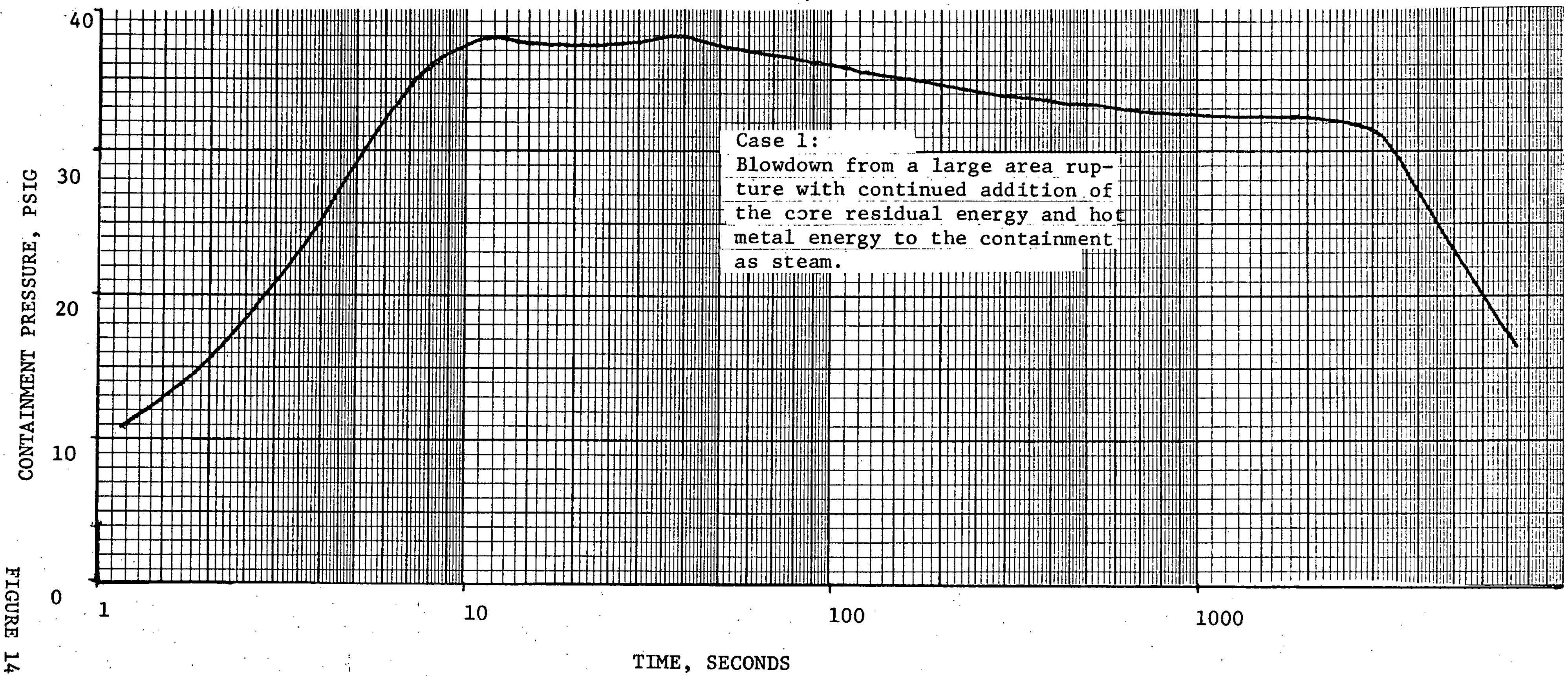
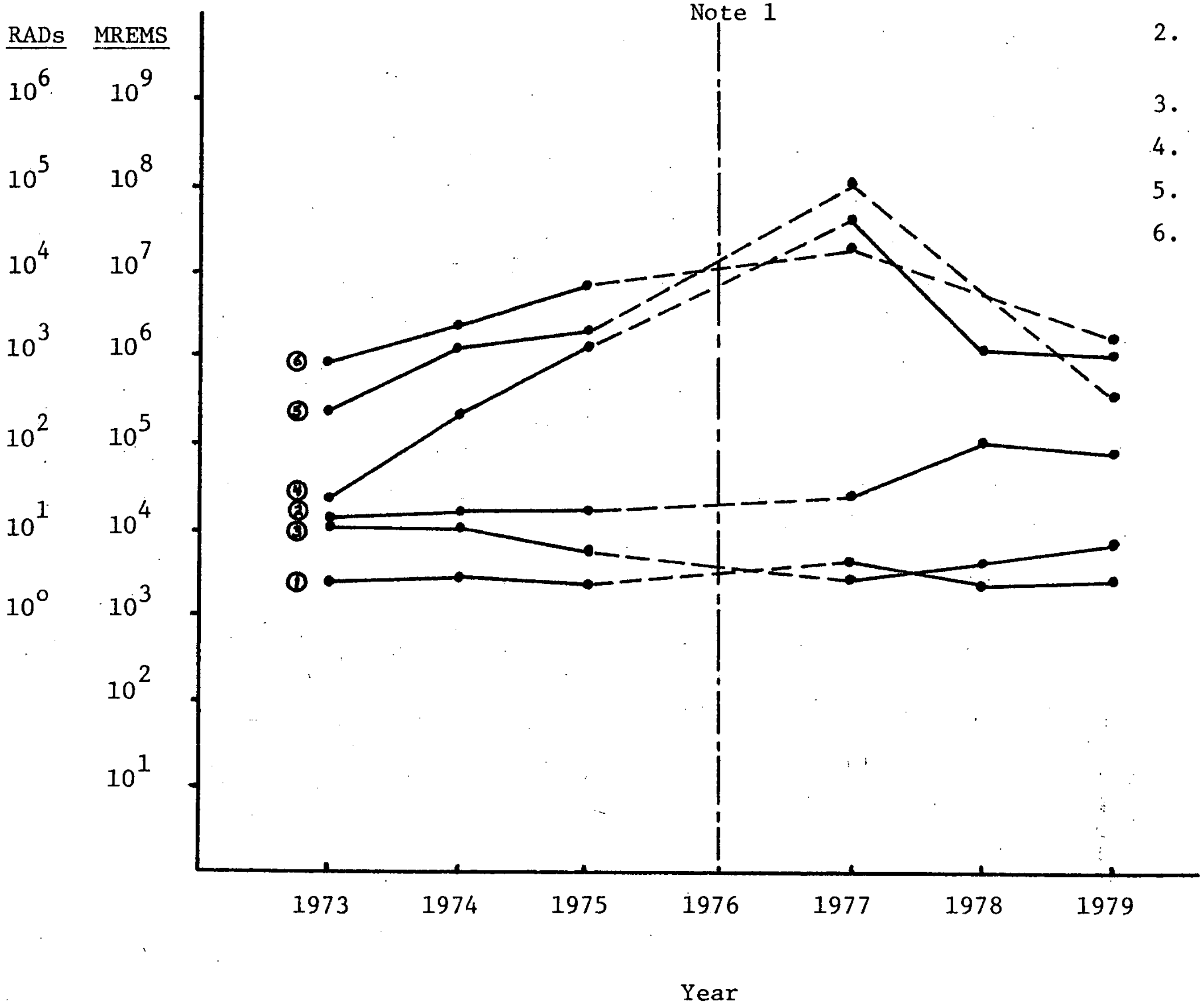


FIGURE 14.3.4-6

H. B. ROBINSON CONTAINMENT RADIATION LEVEL MEASUREMENTS



Recording Stations

1. CV Operating Deck (pressurizer)
2. CV Lower Level Polar Crane Wall (Regen. Heat Exchanger)
3. CV Second Level - Seal Table Room
4. Reactor Coolant Pump BAY-A
5. Reactor Coolant Pump BAY-B
6. Reactor Coolant Pump BAY-C

Notes:

1 No data recorded for the year 1976

.______. Known data

._ _ _ _. Extrapolated data

Figure 1.3.2

TABLE 1.3.1

H. B. ROBINSON CALCULATED RADIATION ACCUMULATION

AREA (1)	YR. ACCUM. (2)	40 YR. ACCUM. (2)	ELEV.(ft)
1. CV Operating Deck (Pressurizer)	4.8×10^{0}	1.9×10^{2}	280
2. CV Lower Level Polar Crane	5.7×10^{1}	2.3×10^{3}	233
3. CV Second Level-Seal Table Rm.	8.5×10^{0}	3.4×10^{2}	254
4. Reactor Coolant Pump - Bay A	1.1×10^{4}	4.4×10^{5}	243
5. Reactor Coolant Pump - Bay B	2.8×10^{4}	1.1×10^{6}	243
6. Reactor Coolant Pump - Bay C	9.6×10^{3}	3.9×10^{5}	243
	7.2×10^{3} (3)	2.9×10^{5} (3)	

| R1

(1) See figure 1.3.1 for locations.
(2) Calculations in (RADs)
(3) Total Containment (Averaged)

TABLE 1.3.2

REACTOR COOLANT SYSTEM DOSES

LOCATION	DOSE r/hr
PIPE CENTER	820
PIPE ID	470
PIPE OD	200
GENERAL AREA	50

TABLE 1.3.3

EQUIPMENT TOTAL RADIATION ACCUMULATION BY LOCATION AND LOCA OPERATING TIME

Component	Location	Level (ft) (Approx.)	Time Of Operation	Radiation Exp. (40 yrs) (1)	Accident (3) Radiation Exp.	Margin (10%)	Total Anticipated Radiation Exposure	
TRANSMITTERS								
PT-444 (2)	CV	231.5	30 MIN. (4)	2.3×10^3	9.5×10^5	-	1.0×10^6 (8)	
PT-445 (2)	CV	231.5	30 MIN. (4)	2.3×10^3	9.5×10^5	-	1.0×10^6 (8)	
PT-456 (2)	CV	231.5	30 MIN. (4)	2.3×10^3	9.5×10^5	-	1.0×10^6 (8)	R1, R2
PT-457 (2)	CV	231.5	30 MIN. (4)	2.3×10^3	9.5×10^5	-	1.0×10^6 (8)	
PT-455	CV	231.5	30 MIN. (4)	2.3×10^3	9.5×10^5	-	1.0×10^6 (8)	
LT-474	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-475	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-476	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-477	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-484	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-485	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-486	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-487	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-494	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-495	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-496	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-497	CV	233	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	3.8×10^6	
LT-459 (2)	CV	230	30 MIN. (4)	2.3×10^3	9.5×10^5	-	9.5×10^5	
LT-460 (2)	CV	230	30 MIN. (4)	2.3×10^3	9.5×10^5	-	9.5×10^5	R1
LT-461 (2)	CV	230	30 MIN. (4)	2.3×10^3	9.5×10^5	-	9.5×10^5	
FT-474	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	
FT-475	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	
FT-484	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	R2
FT-485	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	
FT-494	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	
FT-495	CV	231.5	1 DAY	2.3×10^3	3.5×10^6	3.5×10^5	5.0×10^6 (9)	
FT-940	RAB	230	30 DAYS	———	1.0×10^6 (6)	1.0×10^5	1.1×10^6	
FT-943	RAB	230	30 DAYS	-	1.0×10^6 (6)	1.0×10^5	1.1×10^6	
PT-934	RAB	230	30 DAYS	-	1.0×10^6 (6)	1.0×10^5	1.1×10^6	
PT-940	RAB	230	30 DAYS	-	1.0×10^6 (6)	1.0×10^5	1.1×10^6	
PT-943	RAB	230	30 DAYS	-	1.0×10^6 (6)	1.0×10^5	1.1×10^6	
MOV								
V-866A	CV	241	1 HR.	2.3×10^3	9.5×10^5	9.5×10^4	1.0×10^6	
V-866B	CV	241	1 HR.	2.3×10^3	9.5×10^5	9.5×10^4	1.0×10^6	
V869	RAB	241	30 DAYS	-	1.0×10^6	1.0×10^5	1.1×10^6	
V-744A	CV	240	5 MIN. (4)	2.3×10^3	9.5×10^5	-	9.5×10^5	
V-744B	CV	240	5 MIN. (4)	2.3×10^3	9.5×10^5	-	9.5×10^5	

TABLE 1.3.3 (Continued)

EQUIPMENT TOTAL RADIATION ACCUMULATION BY LOCATION AND LOCA OPERATING TIME

Component	Location	Level (ft) (Approx.)	Time Of Operation	Radiation Exp. (40 yrs)(1)	Accident(3) Radiation Exp.	Margin (10%)	Total Anticipated Radiation Exposure
V-860A	RAB	212	30 DAYS	-	1.0×10^6	1.0×10^5	1.1×10^6
V-860B	RAB	212	30 DAYS	-	1.0×10^6	1.0×10^5	1.1×10^6
V-861A	RAB	212	30 DAYS	-	1.0×10^6	1.1×10^5	1.1×10^6
V-861B	RAB	212	30 DAYS	-	1.0×10^6	1.1×10^5	1.1×10^6
V-863A	RAB	212	30 DAYS	-	1.0×10^6	1.1×10^5	1.1×10^6
V-863B	RAB	212	30 DAYS	-	1.0×10^6	1.1×10^5	1.1×10^6
CVC-381	RAB	240	30 DAYS	-	1.0×10^6	1.1×10^5	1.1×10^6
<u>SOLENOIDS</u>							
V12-7	CV	233	5 MIN.(4)	2.3×10^3	9.5×10^5	-	9.5×10^5
V12-9	CV	233	5 MIN.(4)	2.3×10^3	9.5×10^5	-	9.5×10^5
V12-11	CV	233	5 MIN.(4)	2.3×10^3	9.5×10^5	-	9.5×10^5
V12-13	CV	233	5 MIN.(4)	2.3×10^3	9.5×10^5	-	9.5×10^5
<u>MOTORS</u>							
HVH-1	CV	275	3 HRS.	1.9×10^2	3.1×10^6	3.1×10^5	3.4×10^6
HVH-2	CV	275	3 HRS.	1.9×10^2	3.1×10^6	3.1×10^5	3.4×10^6
HVH-3	CV	275	3 HRS.	1.9×10^2	3.1×10^6	3.1×10^5	3.4×10^6
HVH-4	CV	275	3 HRS.	1.9×10^2	3.1×10^6	3.1×10^5	3.4×10^6
<u>ELECTRICAL PENETRATIONS</u>							
Type 2	CV	234 -246	30 DAYS	2.3×10^3	-	-	1.4×10^7(5)
<u>TEMPERATURE ELEMENTS</u>							
TE-412B	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)
TE-412D	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)
TE-422B	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)
TE-422D	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)
TE-432B	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)
TE-432D	CV	243	(7)	1.1×10^6	-	-	1.5×10^7(5)

(1) Extrapolated from plant data (See Table 1.3.1)

(2) Equipment located in instrument cabinets.

(3) Calculation based on IE Bulletin 79-01B, Appendix B. CHARTS/GRAPHS, Procedures for Evaluating Gamma Radiation Service Conditions.

(4) Charts/Graphs per IE Bulletin 79-01B, Appendix B allow calculation to a minimum of 1 hour exposure. This figure is conservative--no margin required.

(5) Total Integrated Radiation for accident condition (30 days) per IE Bulletin 79-01B, Appendix B. CHARTS/GRAPHS.

(6) Calculation based on Accident Radiation figure - 2×10^7 RADS.

(7) Not required for DBE--used only for outside containment MSLB protection.

(8) Includes added 7.9×10^4 RADS for 1 hour total integrated gamma dose at the surface of containment sump water (Per Appendix D, Table D-8, NUREG 0588). R2

(9) Includes added 1.15×10^6 RADS for 1 day total integrated gamma dose at the surface of containment sump water (Per Appendix D, Table D-8, NUREG 0588).

APPENDIX B

Equipment Requiring Additional Information
and/or Corrective Action
(Category 4.2)

LEGEND:
Designation for Deficiency

R - Radiation
T - Temperature
QT - Qualification Time
RT - Required Time
P - Pressure
H - Humidity
CS - Chemical spray
A - Material aging evaluation, replacement schedule, ongoing equipment surveillance
S - Submergence
M - Margin
I - HELB evaluation outside containment not completed
QM - Qualification method
RPN - Equipment relocation or replacement, adequate schedule not provided
EXN - Exempted equipment justification inadequate
SEN - Separate effects qualification justification inadequate
QI - Qualification information being developed
RPS - Equipment relocation or replacement schedule provided

	Equipment Description	Manufacturer	Model No.	Location[1]	Deficiency
A.	Flow Transmitter	Fischer & Porter	10B2496PBBABBB	2	R,QT,A,QM
A.	Pressure Transmitter	Fischer & Porter	50EP1041BCXA	2	R,QT,A,QM
A.	Motor Operator	Limitorque	SMB-00	1	R,CS,A
A.	Motor Operator	Limitorque	SMB-00	2	R,RT,A
A.	Motor Operator	Limitorque	SMB-3	1	R,CS,A
A.	Motor Operator	Limitorque	SMB-1	2	R,RT,A
A.	Pump Motor	Westinghouse	506UPZ	2	R,T,QT,RT,P,H,A,M

[1] Location (1) Containment Building
Location (2) Auxiliary Building

[A] Items reported as qualified within NRC Region II revised TER, dated 11/7/80.

[B] ~~Items reported as qualified in part, with additional testing underway to establish overall equipment qualification, within NRC Region II revised TER, dated 11/7/80.~~

APPENDIX B (cont'd)

	Equipment Description	Manufacturer	Model No.	Location[1]	Deficiency
A.	Temperature Element	Rosemount	176KF	1	R,CS,A
B.	Fan Motor	Westinghouse	685.5-S	1	R,CS,A
B.	Electrical Penetration	Crouse-Hinds	1.2.2 (745)	1	R,QT,RT,CS,A
B.	Electrical Penetration	Crouse-Hinds	1.2.2 (747)	1	R,QT,RT,CS,A
B.	Electrical Penetration	Crouse-Hinds	1.2.4 (749)	1	R,QT,RT,CS,A
B.	Electrical Penetration	Crouse-Hinds	1.2.5 (751)	1	R,QT,RT,CS,A
A.	Cable	Continental Wire & Cable	CC2115	1	R,RT,P,H,CS,A
A.	Cable	Kerite	High temp, fire resistant	1	R,RT,P,H,CS,A
	Transmitter	Rosemount	1153A	1	R,H,CS,A,QM,S
A.	Solenoid Valve	ASCO	NP831665E	1	R,CS,A
A.	Solenoid Valve	ASCO	NP8316E35E	1	R,CS,A
A.	Solenoid Valve	ASCO	206-381-2U	1	R,CS,A
A.	Cable Splices	Raychem	1000-12N	1	R,T,RT,P,H,CS,A
A.	Cable Splices	Raychem	500-12N	1	R,T,RT,P,H,CS,A
A.	Cable Splices	Raychem	300-12N	1	R,T,RT,P,H,CS,A
A.	Cable Splices	Raychem	200-12N	1	R,T,RT,P,H,CS,A
A.	Cable Splices	Raychem	115-6N	1	R,T,RT,P,H,CS,A
A.	Cable Splices	Raychem	070-6N	1	R,T,RT,P,H,CS,A
	~~Level Transmitter~~	~~Fischer & Porter~~	~~10B2496~~	~~1~~	~~QT,RT,CS,A,QM, S,RPS~~

[A] Items reported as qualified within NRC Region II revised TER, dated 11/7/80.

[B] Items reported as qualified in part, with additional testing underway to establish overall equipment qualification, within NRC Region II revised TER, dated 11/7/80.

Equipment Description	Manufacturer	Model No.	Location[1]	Deficiency
~~Pressure Transmitter~~	~~Fischer & Porter~~	~~50EP1041BCXA NS~~	~~1~~	~~CS,A,QM,S,RPS~~
~~Flow Transmitter~~	~~Rosemount~~	~~1151~~	~~1~~	~~QT,P,CS,A,QM, S,RPS~~
~~Level Transmitter~~	~~Fischer & Porter~~	~~13D2495~~	~~1~~	~~QT,CS,A,QM,S, RPS~~
~~Solenoid Valve~~	~~ASCO~~	~~LB8211C32~~	~~1~~	~~R,T,QT,P,H,CS, A,QM,S,RPS~~
~~Solenoid Valve~~	~~ASCO~~	~~LB8316B25~~	~~1~~	~~R,T,QT,P,H,CS, A,QM,S,RPS~~
~~Solenoid Valve~~	~~ASCO~~	~~LB8316B15~~	~~1~~	~~R,T,QT,P,H,CS, A,QM,S,RPS~~
~~Solenoid Valve~~	~~ASCO~~	~~LB8316B14~~	~~1~~	~~R,T,QT,P,H,CS, A,QM,S,RPS~~
Level Switch	Madison	5602	1	R,T,QT,RT,P,H, CS,A,QM
~~Level Transmitter~~	~~Fischer & Porter~~	~~13B2496~~	~~1~~	~~QT,CS,A,QM,S, RPS~~
(Repeated Item) Cable for Instrumentation	Continental Wire & Cable	CC2115	1	R,RT,P,H,CS, A,QI
Silicon Rubber Tape	3M/Electric Products Division	Scotch 70	1	R,T,QT,RT,P,H, CS,A,QM,QI
A. Cable Terminals and Splices	AMP	53548-1	1	R,T,RT,P,H,CS, A,QM,QI

Section II - Updated Summary Sheets

Resubmitted System Component Evaluation Work Sheets originally sent as part of CP&L Environmental Qualification of Electrical Equipment, H. B. Robinson E. G. Plant Unit 2, 90-Day Report, Revision 3; dated February 1, 1981.

For easier comparison purposes the accident pressure and temperature values have been entered in the Summary Work Sheets. The figures represent values stated within paragraph 3.3 of this report.

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: SAFETY INJECTION

Plant ID No. FT-940 (1)

Component:
FLOW TRANSMITTER

Manufacturer:
FISHER & PORTER

Model Number:
10B2496PBBABBB

Function:
SAFETY INJECTION

Accuracy: Spec: ± 1/2%
Demon:

Service:
HEADER FLOW (Hot Leg)

Location:
REACTOR AUXILIARY BLDG.

Flood Level Elev: (2)
Above Flood Level: Yes
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
Parameter	Specification	Qualification	Specification	Qualification (4)			
Operating Time	30 DAYS	2 HRS.	38	17	SIMULTANEOUS TEST	NONE	
Temperature (°F)	AMBIENT	287	38	17	SIMULTANEOUS TEST	NONE	
Pressure (PSIA)	ATMOS.	75	38	17	SIMULTANEOUS TEST	NONE	
Relative Humidity (%)	AMBIENT	100	38	17	SIMULTANEOUS TEST	NONE	
Chemical Spray	NOT REQUIRED	-	-	-	-		
Radiation	1.1×10^6	2×10^8	(3)	17	SEQUENTIAL TEST (5)	NONE	R2
Aging							
Submergence	NOT REQUIRED	-	-	-	-		

(1) Transmitter not exposed to DBE - Long-term mitigation radiation exposure only
(2) Not involved in containment flood postulation
(3) See Section 1.3.2
(4) See Section 3.2.2 for evaluation.
(5) Test performed after LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (4)		
System: SAFETY INJECTION Plant ID No. FT-943 (1)	Operating Time	30 DAYS	2 HRS.	38	17	SIMULTANEOUS TEST	NONE
Component: FLOW TRANSMITTER	Temperature (°F)	AMBIENT	287	38	17	SIMULTANEOUS TEST	NONE
Manufacturer: FISHER & PORTER	Pressure (PSIA)	ATMOS.	75	38	17	SIMULTANEOUS TEST	NONE
Model Number: 10B2496PBBABBB	Relative Humidity (%)	AMBIENT	100	38	17	SIMULTANEOUS TEST	NONE
Function: SAFETY INJECTION Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	-	-	-	-	
Service: HEADER FLOW (Cold Leg)	Radiation	1.1×10^6	2×10^8	(3)	17	SEQUENTIAL TEST (5)	NONE R2
Location: REACTOR AUXILIARY BUILDING	Aging						
Flood Level Elev: (2) Above Flood Level: Yes No	Submergence	NOT REQUIRED	-	-	-	-	

(1) Transmitter not exposed to DBE - Long-term mitigation radiation exposure only
(2) Not involved in containment flood postulation
(3) See Section 1.3.2
(4) See Section 3.2.2 for evaluation.
(5) Test performed after LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: SAFETY INJECTION

Plant ID No. PT-934 (1)

Component:
PRESSURE TRANSMITTER

Manufacturer:
FISHER & PORTER

Model Number:
50EP1041BCXA

Function:
BORON INJECTION

Accuracy: Spec:
Demon:

Service:
TANK HEADER PRESSURE

Location:
REACTOR AUXILIARY BLDG.

Flood Level Elev: (2)
Above Flood Level: Yes
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Specification	Qualification	Specification	Qualification (4)		
Operating Time	30 DAYS	2 HRS.	38	17	SIMULTANEOUS TEST	NONE
Temperature (°F)	AMBIENT	287	38	17	SIMULTANEOUS TEST	NONE
Pressure (PSIA)	ATMOS.	75	38	17	SIMULTANEOUS TEST	NONE
Relative Humidity (%)	AMBIENT	100	38	17	SIMULTANEOUS TEST	NONE
Chemical Spray	NOT REQUIRED	-	-	-	-	-
Radiation	1.1×10^6	2×10^8	(3)	17	SEQUENTIAL TEST (5)	NONE
Aging						
Submergence	NOT REQUIRED	-	-	-	-	-

|R2

(1) Transmitter not exposed to DBE - Long-term mitigation radiation exposure only
(2) Not involved in containment flood postulation
(3) See Section 1.3.2
(4) See Section 3.2.2 for evaluation.
(5) Test performed after LOCA simulated environmental expusure

|R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: SAFETY INJECTION

Plant ID No. PT-940 (1)

Component:
PRESSURE TRANSMITTER

Manufacturer:
FISHER & PORTER

Model Number:
50EP1041

Function:
SAFETY INJECTION

Accuracy: Spec:
Demon:

Service:
HEADER PRESSURE (Hot Leg)

Location:
REACTOR AUXILIARY BUILDING

Flood Level Elev: (2)
Above Flood Level: Yes
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Specification	Qualification	Specification	Qualification (4)		
Operating Time	30 DAYS	2 HRS.	38	17	SIMULTANEOUS TEST	NONE
Temperature (°F)	AMBIENT	287	38	17	SIMULTANEOUS TEST	NONE
Pressure (PSIA)	ATMOS.	75	38	17	SIMULTANEOUS TEST	NONE
Relative Humidity (%)	AMBIENT	100	38	17	SIMULTANEOUS TEST	NONE
Chemical Spray	NOT REQUIRED	-	-	-	-	-
Radiation	1.1×10^6	2×10^8	(3)	17	SEQUENTIAL TEST (5)	NONE
Aging						
Submergence	NOT REQUIRED	-	-	-	-	-

R2

(1) Transmitter not exposed to DBE - Long-term mitigation radiation exposure only
(2) Not involved in containment flood postulation
(3) See Section 1.3.2
(4) See Section 3.2.2 for evaluation.
(5) Test performed after LOCA simulated environmental exposure

R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (4)		
System: SAFETY INJECTION (1) Plant ID No. PT-943	Operating Time	30 DAYS	2 HRS.	38	17	SIMULTANEOUS TEST	NONE
Component: PRESSURE TRANSMITTER	Temperature (°F)	AMBIENT	287	38	17	SIMULTANEOUS TEST	NONE
Manufacturer: FISHER & PORTER	Pressure (PSIA)	ATMOS.	75	38	17	SIMULTANEOUS TEST	NONE
Model Number: 50EP1041BCXA	Relative Humidity (%)	AMBIENT	100	38	17	SIMULTANEOUS TEST	NONE
Function: SAFETY INJECTION Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	-	-	-		
Service: HEADER PRESSURE (Cold Leg)	Radiation	1.1×10^6	2×10^8	(3)	17	SEQUENTIAL TEST (5)	NONE
Location: REACTOR AUXILIARY BUILDING	Aging						
Flood Level Elev: (2) Above Flood Level: Yes No	Submergence	NOT REQUIRED	-	-	-	-	

| R2

(1) Transmitter not exposed to DBE.- Long-term mitigation radiation exposure only
(2) Not involved in containment flood postulation
(3) See Section 1.3.2
(4) See Section 3.2.2 for evaluation. | R2
(5) Test performed after LOCA simulated environmental exposure

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: SAFETY INJECTION

Plant ID No. V-866A (1)

Component: MOTOR OPERATOR

Manufacturer: LIMITORQUE

Model Number: SMB-00

Function: HOT LEG INJECTION

Accuracy: Spec:
Demon:

Service: MOTOR OPERATED VALVE-SIS

Location: CONTAINMENT 241

Flood Level Elev: 231.2'
Above Flood Level: Yes X
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
Parameter	Specification	Qualification	Specification	Qualification (5)			
Operating Time	1 HR	7 DAYS	38	14	SIMULTANEOUS TEST	NONE	
Temperature (°F)	265 (2)	308	35, 38	14, 17	SIMULTANEOUS TEST	NONE	
Pressure (PSIA)	38 (3)	75	35	14, 17	SIMULTANEOUS TEST	NONE	
Relative Humidity (%)	100	100	35	14, 17	SIMULTANEOUS TEST	NONE	
Chemical Spray	H_3BO_3 N_aOH	H_3BO_3 N_aOH		14	SIMULTANEOUS TEST	NONE	R2
Radiation	1.0×10^6	2×10^8	(4)	17	SEQUENTIAL TEST (6)	NONE	R2
Aging		40 YRS		17	SEQUENTIAL TEST (6)	NONE	
Submergence	NOT REQUIRED						R2

NOTES:
(1) Same data this sheet applies to V-866B.
(2) See accident profile - Temperature - Figure 3.1-1.
(3) See accident profile - Pressure - Figure 3.1-2.
(4) See Section 1.3.2.
(5) See Section 3.2.3 for evaluation.
(6) Test performed prior to LOCA simulated environmental exposure R2

Facility: H. B. Robinson #2

Sheet ___ of 25

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: SAFETY INJECTION

Plant ID No. V869

Component: MOTOR OPERATOR

Manufacturer: LIMITORQUE

Model Number: SMB-00

Function: HOT LEG INJECTION BORON INJECTION

Accuracy: Spec:
Demon:

Service: MOTOR OPERATED VALVE

Location:
REACTOR AUXILIARY BLDG.

Flood Level Elev: (4)
Above Flood Level: Yes
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
Parameter	Specification	Qualification	Specification	Qualification (3)			
Operating Time	(1)	7 DAYS	30	14	SIMULTANEOUS TEST	NONE	
Temperature (°F)	AMBIENT	308	35	14,17	SIMULTANEOUS TEST	NONE	
Pressure (PSIA)	ATMOS.	75	35	14,17	SIMULTANEOUS TEST	NONE	
Relative Humidity (%)	AMBIENT	100	35	14,17	SIMULTANEOUS TEST	NONE	
Chemical Spray	NOT REQUIRED	H_3BO_3 NaOH		14,17	SIMULTANEOUS TEST	NONE	R2
Radiation	1.1×10^6	2.0×10^8	(2)	17	SEQUENTIAL TEST (5)	NONE	R2
Aging	-	40 YRS.		17	SEQUENTIAL TEST (5)	NONE	
Submergence	NOT APPLICABLE						

(1) To be used intermittantly during mitigation of LOCA
(2) See Section 1.3.2.
(3) See Section 3.2.3 for evaluation.
(4) Not involved in containment flood postulation
(5) Test performed prior to LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (4)		
System: SAFETY INJECTION Plant ID No. LS-1925A Component: LEVEL SWITCH	Operating Time	CONTINUOUS	NONE	-	-	-	(5)
	Temperature (^{o}F)	265 (2)	NONE	-	-	-	(5)
Manufacturer: MADISON	Pressure (PSIA)	38 (3)	NONE	-	-	-	(5)
Model Number: 5602	Relative Humidity (%)	100	NONE	-	-	-	(5)
Function: CONTAINMENT SUMP WATER LEVEL MEASUREMENT Accuracy: Spec: 1/2' in- Demon: crement	Chemical Spray	H_3BO_3 NaOH	NONE	-	-	-	(5)
	Radiation	1.4×10^7	NONE	-	-	-	(5)
Service: DETECT WATER LEVEL CHANGES Location: CONTAINMENT 228'	Aging		NONE	-	-	-	(5)
Flood Level Elev: 231.2' Above Flood Level: Yes No X	Submergence		NONE				

R2

(1) Same data this sheet applies to LS-1925B
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profie - Pressure - Figure 3.1-2
(4) See Section 3.2.7 for evaluation
(5) Function to be superceded by two channels of analog measurement equipment. No qualification testing required.

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SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
	Parameter	Specification	Qualification	Specification	Qualification (5)			
System: AUXILIARY COOLING Plant ID No. V-744A (1) Component: MOTOR OPERATOR	Operating Time	5 MIN	7 DAYS	40	14	SIMULTANEOUS TEST	NONE	
	Temperature (°F)	265 (2)	308	35, 38	14, 17	SIMULTANEOUS TEST	NONE	
Manufacturer: LIMITORQUE	Pressure (PSIA)	38 (3)	75	35	14, 17	SIMULTANEOUS TEST	NONE	
Model Number: SMB-3	Relative Humidity (%)	100	100	35	14, 17	SIMULTANEOUS TEST	NONE	
Function: REACTOR CORE DELUGE	Chemical Spray	H_3BO_3 $NaOH$	H_3BO_3 $NaOH$		14	SIMULTANEOUS TEST	NONE	R2
Accuracy: Spec: Demon:	Radiation	9.5×10^5	2×10^8	(4)	17	SEQUENTIAL TEST (6)	NONE	
Service: MOTOR-OPERATED VALVE-SIS Location: CONTAINMENT 245'	Aging		40 YRS		17	SEQUENTIAL TEST (6)	NONE	R2
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence							R2

NOTES:
(1) Same data this sheet applies to V-744B.
(2) See accident profile - Temperature - Figure 3.1-1.
(3) See accident profile - Pressure - Figure 3.1-2.
(4) See Section 1.3.2.
(5) See Section 3.2.3 for evaluation.
(6) Test performed prior to LOCA simulated environmental exposure R2

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SYSTEM COMPONEN EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (4)		
System: AUXILIARY COOLING Plant ID No. V860A	Operating Time	(2)	7 DAYS	30	14	SIMULTANEOUS TEST	NONE
Component: MOTOR OPERATOR	Temperature (^{o}F)	AMBIENT	308	35	14,17	SIMULTANEOUS TEST	NONE
Manufacturer: LIMITORQUE	Pressure (PSIA)	ATMOS.	75	35	14,17	SIMULTANEOUS TEST	NONE
Model Number: SMB-1	Relative Humidity (%)	AMBIENT	100	35	14,17	SIMULTANEOUS TEST	NONE
Function: CV SUMP TO RHR SUCTION Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	H_3BO_3 NaOH		14,17	SIMULTANEOUS TEST	NONE
Service: MOTOR OPERATED VALVE	Radiation	1.1×10^6	2.0×10^8	(3)	17	SEQUENTIAL TEST (6)	NONE
Location: REACTOR AUXILIARY BLDG.	Aging	-	40 YRS.		17	SEQUENTIAL TEST (6)	NONE
Flood Level Elev: (5) Above Flood Level: Yes No	Submergence	NOT APPLICABLE	-				

R2

R2

(1) Same data this sheet applies to V860B
(2) To be used intermittantly during mitigation of LOCA.
(3) See Section 1.3.2.
(4) See Section 3.2.3 for evaluation.
(5) Not involved in containment flood postulation
(6) Test performed prior to LOCA simulated environmental exposure

R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT Parameter	ENVIRONMENT Specification	ENVIRONMENT Qualification	DOCUMENTATION REFERENCE Specification	DOCUMENTATION REFERENCE Qualification (4)	QUALIFICATION METHOD	OUTSTANDING ITEMS	
System: AUXILIARY COOLING Plant ID No. V861A (1)	Operating Time	(2)	7 DAYS	30	14	SIMULTANEOUS TEST	NONE	
Component: MOTOR OPERATOR	Temperature (°F)	AMBIENT	308	35	14,17	SIMULTANEOUS TEST	NONE	
Manufacturer: LIMITORQUE	Pressure (PSIA)	ATMOS.	75	35	14,17	SIMULTANEOUS TEST	NONE	
Model Number: SMB-1	Relative Humidity (%)	AMBIENT	100	35	14,17	SIMULTANEOUS TEST	NONE	
Function: CV SUMP TO RHR SUCTION Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	H_3BO_3 NaOH		14,17	SIMULTANEOUS TEST	NONE	R2
Service: MOTOR OPERATED VALVE	Radiation	1.1×10^6	2.0×10^8	(3)	17	SEQUENTIAL TEST (6)	NONE	R2
Location: REACTOR AUXILIARY BLDG.	Aging	-	40 YRS.		17	SEQUENTIAL TEST (6)	NONE	
Flood Level Elev: (5) Above Flood Level: Yes No	Submergence	NOT APPLICABLE						

(1) Same data this sheet applies to V861B
(2) To be used intermittantly during mitigation of LOCA.
(3) See Section 1.3.2.
(4) See Section 3.2.3 for evaluation.
(5) Not involved in containment flood postulation R2
(6) Test performed prior to LOCA simulated environmental exposure

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
	Parameter	Specification	Qualification	Specification	Qualification (4)			
System: AUXILIARY COOLING Plant ID No. V863A	Operating Time	(2)	7 DAYS	30	14	SIMULTANEOUS TEST	NONE	
Component: MOTOR OPERATOR	Temperature (°F)	AMBIENT	308	35	14,17	SIMULTANEOUS TEST	NONE	
Manufacturer: LIMITORQUE	Pressure (PSIA)	ATMOS.	75	35	14,17	SIMULTANEOUS TEST	NONE	
Model Number: SMB-00	Relative Humidity (%)	AMBIENT	100	35	14,17	SIMULTANEOUS TEST	NONE	
Function: RHR DISCHARGE TO SI SPRAY SYSTEM Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	H_3BO_3 NaOH		14,17	SIMULTANEOUS TEST	NONE	R2
Service: MOTOR OPERATED VALVE	Radiation	1.1×10^6	2.0×10^8	(3)	17	SEQUENTIAL TEST (6)	NONE	R2
Location: REACTOR AUXILIARY BLDG.	Aging	-	40 YRS.		17	SEQUENTIAL TEST (6)	NONE	
Flood Level Elev: (5) Above Flood Level: Yes No	Submergence	NOT APPLICABLE						

(1) Same data this sheet applies to V863B
(2) To be used intermittantly during mitigation of LOCA.
(3) See Section 1.3.2.
(4) See Section 3.2.3 for evaluation.
(5) Not involved in containment flood postulation
(6) Test performed prior to LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (5)		
System: AUXILIARY COOLING Plant ID No. RHR-A (1) Component: MOTOR, PUMP Manufacturer: WESTINGHOUSE Model Number: 506UPZ Function: CIRCULATE SUMP WATER & BORATED REFUELING WATER TO REACTOR COOLANT SYSTEM-POST LOCA Accuracy: Spec: Demon: Service: RESIDUAL HEAT REMOVAL PUMP - SIS Location: AUXILIARY BUILDING	Operating Time	CONTINUOUS	CONTINUOUS	34, 35		(4)	(2)
	Temperature (°F)	85 (AVG) AMBIENT	90° C RISE	35, 19		(4)	(2)
	Pressure (PSIA)	15	15	35, 19		(4)	(2)
	Relative Humidity (%)	AMBIENT	AMBIENT	35, 19		(4)	(2)
	Chemical Spray	NOT REQUIRED	NOT REQUIRED				
	Radiation	1.1×10^6	2.0×10^8	19 (3)	18	SEQUENTIAL TEST	NONE
	Aging		40 yrs.		18	SEQUENTIAL TEST	NONE
Flood Level Elev: N/A Above Flood Level: Yes No	Submergence	NOT APPLICABLE					

NOTES:
(1) Same data this sheet applies to RHR-B.
(2) Motor not exposed to DBE, no qualification testing needed.
(3) See Section 1.3.2
(4) Information to be obtained from manufacturer.
(5) See Section 3.2.8 for evaluation

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	Parameter	ENVIRONMENT Specification	ENVIRONMENT Qualification	DOCUMENTATION REFERENCE Specification	DOCUMENTATION REFERENCE Qualification	QUALIFICATION METHOD	OUTSTANDING ITEMS	
System: REACTOR PROTECTION Plant ID No. TE-412B (1)	Operating Time	1 HR.	2 WKS.	21	48	SIMULTANEOUS TEST	NONE (5)	
Component: TEMPERATURE ELEMENT	Temperature (°F)	265 (2)	320	21	48	SIMULTANEOUS TEST	NONE (5)	
Manufacturer: ROSEMOUNT	Pressure (PSIA)	38 (3)	81	21	48	SIMULTANEOUS TEST	NONE (5)	
Model Number: 176KF	Relative Humidity (%)	100	100	21	48	SIMULTANEOUS TEST	NONE (5)	
Function: MAIN STEAM LINE BREAK MONITOR Accuracy: Spec: Demon:	Chemical Spray	-	H_3BO_3 NaOH		48	SIMULTANEOUS TEST	NONE (5)	
Service: T_{AV}-REACTOR COOLANT LOOP #1 SIS GENERATION	Radiation	1.5×10^7	1.0×10^8	(4)	48	SEQUENTIAL TEST (6)	NONE (5)	R2
Location: CONTAINMENT 243'	Aging		40 YRS. + 2 WKS.		48	SEQUENTIAL TEST (6)	NONE	R2
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence	NOT APPLICABLE						R2

NOTES:

(1) Same data this sheet applies to TE-412D
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profile - Pressure - Figure 3.1-2
(4) See Section 1.3.2
(5) Not required for DBE - used only for outside containment Main Steam line Break protection
(6) Test performed prior to LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification		
System: REACTOR PROTECTION Plant ID No. TE-422B (1)	Operating Time	1 HR.	2 WKS.	21	48	SIMULTANEOUS TEST	NONE (5)
Component: TEMPERATURE ELEMENT	Temperature (°F)	265 (2)	320	21	48	SIMULTANEOUS TEST	NONE (5)
Manufacturer: ROSEMOUNT	Pressure (PSIA)	38 (3)	81	21	48	SIMULTANEOUS TEST	NONE (5)
Model Number: 176 KF	Relative Humidity (%)	100	100	21	48	SIMULTANEOUS TEST	NONE (5)
Function: MAIN STEAM LINE BREAK MONITOR Accuracy: Spec: Demon:	Chemical Spray	-	H_3BO_3 NaOH		48	SIMULTANEOUS TEST	NONE (5)
Service: T_{AV}-REACTOR COOLANT LOOP #2 SIS GENERATION	Radiation	1.5×10^7	1.0×10^8	(4)	48	SEQUENTIAL TEST (6)	NONE (5)
Location: CONTAINMENT 243'	Aging		40 YRS. + 2 WK. POST ACCIDENT		48	SEQUENTIAL TEST (6)	NONE
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence	NOT APPLICABLE					

R2

R2

NOTES:

(1) Same data this sheet applied to TE-422D
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profile - Pressure - Figure 3.1-2
(4) See Section 1.3.2
(5) Not required for DBE - only used for outside containment Main Steam Line Break protection
(6) Test performed prior to LOCA simulated environmental exposure

R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: REACTOR PROTECTION

Plant ID No. TE-432B (1)

Component: TEMPERATURE ELEMENT

Manufacturer: ROSEMOUNT

Model Number: 176KF

Function: MAIN STEAM-LINE BREAK MONITOR

Accuracy: Spec:
Demon:

Service: T_{VA}-REACTOR COOLANT LOOP #3 - SIS. GENERATION

Location: CONTAINMENT 243'

Flood Level Elev: 231.2'
Above Flood Level: Yes X
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Specification	Qualification	Specification	Qualification		
Operating Time	1 HR.	2 WKS.	21	48	SIMULTANEOUS TEST	NONE (5)
Temperature (°F)	265 (2)	320	21	48	SIMULTANEOUS TEST	NONE (5)
Pressure (PSIA)	38 (3)	81	21	48	SIMULTANEOUS TEST	NONE (5)
Relative Humidity (%)	100	100	21	48	SIMULTANEOUS TEST	NONE (5)
Chemical Spray	-	H_3BO_3 NaOH		48	SIMULTANEOUS TEST	NONE (5)
Radiation	1.5×10^7	1.0×10^8	(4)	48	SEQUENTIAL TEST (6)	NONE (5)
Aging		40 YRS. + 2 WKS.		48	SEQUENTIAL TEST (6)	
Submergence	NOT APPLICABLE					

R2

R2

NOTES:

(1) Same data this sheet applies to TE-432D
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profile - Pressure - Figure 3.1-2
(4) See Section 1.3.2
(5) Not required for DBE - only used for outside containment main steam line break protection
(6) Test performed prior to LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (5)		
System: HVAC Plant ID No. HVH-1 (1)	Operating Time	3 hrs.	24 hrs. +	36	16	Simultaneous Test	None
Component: MOTOR, FAN	Temperature (°F)	265 (2)	315	36	16	Simultaneous Test	None
Manufacturer: WESTINGHOUSE	Pressure (PSIA)	38 (3)	75-95	36	16	Simultaneous Test	None
Model Number: 685.5-S	Relative Humidity (%)	100	100	36	16	Simultaneous Test	None
Function: TRANSFER HEAT FROM CONTAINMENT TO SERVICE WATER	Chemical Spray	H_3BO_3 NaOh	H_3BO_3 NaOh	34	16	Simultaneous Test	None
Accuracy: Spec: Demon:	Radiation	3.4×10^6	1.41×10^8	(4)	15	Sequential Test (6)	None
Service: CONTAINMENT FAN COOLER Location: CONTAINMENT 275'	Aging		40 yrs.	-	15	Sequential Test (6)	None
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence	NOT APPLICABLE					

R2

R2

NOTES:

(1) Same data this sheet applies to HVH-2, HVH-3, HVH-4
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profile - Pressure - Figure 3.1-2
(4) See Section 1.3.2.
(5) See Section 3.2.8 for evaluation.
(6) Test performed on selected motor components - not part of LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
	Parameter	Specification	Qualification	Specification	Qualification			
System: ALL Plant ID No. SEE NOTE(1)	Operating Time	CONTINUOUS	105 hrs.	1	2,43	SIMULTANEOUS TEST	NONE	
Component: ELECTRICAL PENETRATION	Temperature (°F)	265 (2)	340	1	2,3,4,43	SIMULTANEOUS TEST	NONE	
Manufacturer: CROUSE-HINDS	Pressure (PSIA)	38 (3)	75	1	2,3,4,43	SIMULTANEOUS TEST	NONE	
Model Number: 1.2.2 (745) 1.2.5 (751) 1.2.2 (747) 1.2.4 (749)	Relative Humidity (%)	100	100	1	2,4,43	SIMULTANEOUS TEST	NONE	
Function: ACCIDENT CONDITION MONITORING	Chemical Spray	-	H_3BO_3 NaOH		43	SIMULTANEOUS TEST	NONE	
Accuracy: Spec: Demon:	Radiation	1.4×10^7	2.13×10^8	(6)	43	SEQUENTIAL TEST (7)	NONE (5)	
Service: PROVIDE CABLE CONTINUITY THROUGH CONTAINMENT SHELL Location: CONTAINMENT 234' - 246'	Aging	40	524 hrs. @ 150 C (40 yrs)	1	43	SEQUENTIAL TEST (7)	NONE	R2
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence	NOT APPLICABLE						R2

NOTES:

(1) Data this sheet applies to penetrations B-1,B-2,B-5,B-9,C-1,C-2,C-3,C-4,C-6,C-8,C-9,D-1,D-2,D-3,D-5,D-8,D-9
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) See Section 3.2.1 for evaluation
(5) Qualification established for penetration cartridge only. Pigtail cable requires separate testing as reported in Section 3.2.1
(6) See Section 1.3.2
(7) Test performed prior to LOCA simulated environmental exposure R2

ity: H. B. Robinson #2

Sheet of 25

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification (6)		
System: ALL Plant ID No. SEE NOTE(1)	Operating Time	1 HR.-1 DAY	67 HRS.	38	23	SIMULTANEOUS TEST	NONE
Component: TRANSMITTER	Temperature (°F)	265 (2)	350	38	23	SIMULTANEOUS TEST	NONE
Manufacturer: ROSEMOUNT	Pressure (PSIA)	38 (3)	135	38	23	SIMULTANEOUS TEST	NONE
Model Number: 1153A	Relative Humidity (%)	100	100		23	SIMULTAN-TEST	NONE
Function: REPLACEMENT COMPONENT	Chemical Spray	H_3BO_3 NaOH	H_3BO_3 NaOH		23,41	SIMULTANEOUS TEST	NONE
Accuracy: Spec: ± ¼% Demon:	Radiation	5.0×10^6	4.4×10^7	(5)	23	SEQUENTIAL TEST (7)	(4)
Service: Location: CONTAINMENT	Aging	-	NOT WITHIN MFGR. TEST PROGRAM				(4)
Flood Level Elev: 231.2' Above Flood Level: Yes No	Submergence						

R2

R2

NOTES:

(1) Replacement transmitter to be supplied for: PT-444, PT-445, PT-455, PT-456, PT-457, LT-474, LT-475, LT-476, LT-477, LT-484, LT-486, LT-487, LT-494, LT-495 LT-496, LT-497, LT-459, LT-460, LT-461, FT-474, FT-475, FT-484, FT-485, FT-494, FT-495, LT-485

(2) See accident profile - Temperature - Figure 3.1-1

(3) See accident profile - Pressure - Figure 3.1-2

(4) Replacement transmitters tested under IEEE 323-1971 format, Rosemount currently performing transmitter testing to meet IEEE-323-1974 requirements.

(5) See Section 1.2.3

(6) See Section 3.2.1 for evaluation

(7) Test performed prior to LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: ALL

Plant ID No. SEE NOTE (1)

Component:
SOLENOID, VALVE

Manufacturer: ASCO

Model Number:
NP831665E
NP8316E35E
206-381-2U

Function:
REPLACEMENT COMPONENT

Accuracy: Spec:
Demon:

Service:

Location:
CONTAINMENT 283'

Flood Level Elev: 231.2'
Above Flood Level: Yes X
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Specification	Qualification	Specification	Qualification (5)		
Operating Time	5 min.	30 days	40	47	Simultaneous Test	None
Temperature (°F)	265 (2)	346	40	47	Simultaneous Test	None
Pressure (PSIA)	38 (3)	125	40	47	Simultaneous Test	None
Relative Humidity (%)	100	100	40	47	Simultaneous Test	None
Chemical Spray	H_3BO_3 NaOH	H_3BO_3 NaOH		47	Simultaneous Test	None
Radiation	9.5×10^5	2.0×10^8	(4)	47	Sequential Test (6)	None
Aging	-	40 yrs. & 4.4 yrs.	(5)	47	Sequential Test (6)	None
Submergence	Not Applicable					

R2

NOTES:

(1) Replacement solenoid valves to be supplied for: V12-7, V12-9, V12-11, V12-13, CVC-200A, CVC-200B, CVC-200C R2
(2) See accident profile - Temperature - Figure 3.1-1
(3) See accident profile - Pressure - Figure 3.1-2
(4) See Section 1.3.2
(5) See Section 3.2.6 for evaluation
(6) Test performed prior to LOCA simulated environmental exposure R2

Facility: H. B. Robinson #2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification		
System: ALL Plant ID No.	Operating Time	CONTINUOUS	240 hrs.		46	SIMULTANEOUS TEST	NONE
Component: CABLE 4/C #16, 2/C #16, Shielded	Temperature (°F)	265 (2)	340	5	46	SIMULTANEOUS TEST	NONE
Manufacturer: CONTINENTAL WIRE & CABLE	Pressure (PSIA)	38 (3)	115		46	SIMULTANEOUS TEST	NONE
Model Number: CC2115	Relative Humidity (%)	100	100		46	SIMULTANEOUS TEST	NONE
Function: FIELD CABLE	Chemical Spray		H_3BO_3		46	SEQUENTIAL TEST	NONE
Accuracy: Spec: Demon: Service: INSTRUMENTATION	Radiation	1.4×10^7	1.0×10^8	(1)	46	SEQUENTIAL TEST	NONE
Location: CONTAINMENT	Aging			5	(4)		
Flood Level Elev: 231.2' Above Flood Level: Yes No	Submergence	NOT APPLICABLE					

R2

NOTES:

(1) See Section 1.3.2
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) See Section 3.2.4 for evaluation

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: ALL

Plant ID No.

Component: CABLE
3/C #16, 2/C #16,
500 MCM, 3/C 19/#22

Manufacturer: KERITE

Model Number:
HIGH TEMP, FIRE RESISTANT

Function: FIELD CABLE

Accuracy: Spec:
Demon:

Service: CONTROL AND LOW POWER

Location: CONTAINMENT

Flood Level Elev: 231.2'
Above Flood Level: Yes
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD (4)	OUTSTANDING ITEMS
Parameter	Specification	Qualification	Specification	Qualification		
Operating Time	CONTINUOUS	50 DAYS		49	SIMULTANEOUS TEST	NONE
Temperature (^oF)	265 (2)	346	6	49	SIMULTANEOUS TEST	NONE
Pressure (PSIA)	38 (3)	128		49	SIMULTANEOUS TEST	NONE
Relative Humidity (%)	100	100		49	SIMULTANEOUS TEST	NONE
Chemical Spray		H_3BO_3 NaOH		49	SIMULTANEOUS TEST	NONE
Radiation	1.4×10^7	2.0×10^8	(1)	49	SIMULTANEOUS TEST	NONE
Aging		40 YEARS	6	49	SEQUENTIAL TEST	NONE
Submergence	NOT APPLICABLE					

R1

R2

NOTES:

(1) See Section 1.3.2
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) See Section 3.2.4 R1

Facility: H. B. Robinson #2 Sheet [illegible] of 25

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
	Parameter	Specification	Qualification	Specification	Qualification (3)			
System: CHEMICAL & VOLUME Plant ID No. CVC-381	Operating Time	(1)	7 DAYS	30	14	SIMULTANEOUS TEST	NONE	
Component: MOTOR OPERATOR	Temperature (°F)	AMBIENT	308	35	14,17	SIMULTANEOUS TEST	NONE	
Manufacturer: LIMITORQUE	Pressure (PSIA)	ATMOS.	75	35	14,17	SIMULTANEOUS TEST	NONE	
Model Number: SMB-00	Relative Humidity (%)	AMBIENT	100	35	14,17	SIMULTANEOUS TEST	NONE	
Function: REACTOR COOLANT PUMP SEAL WATER RETURN Accuracy: Spec: Demon:	Chemical Spray	NOT REQUIRED	H_3BO_3 NaOH		14,17	SIMULTANEOUS TEST	NONE	R2
Service: MOTOR OPERATED VALVE	Radiation	1.1×10^6	2.0×10^8	(2)	17	SEQUENTIAL TEST (5)	NONE	R2
Location: 240' REACTOR AUXILIARY BLDG.	Aging		40 YRS.		17	SEQUENTIAL TEST (5)	NONE	
Flood Level Elev: (4) Above Flood Level: Yes No	Submergence	NOT APPLICABLE						

(1) To be used intermittantly during mitigation of LOCA
(2) See Section 1.3.2
(3) See Section 3.2.3 for evaluation
(4) Not involved in containment flood postulation
(5) Test performend prior to LOCA simulated environmental exposure

R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION

System: ALL

Plant ID No.

Component: CABLE SPLICES (1)

Manufacturer: RAYCHEM

Model Number:
1000-12N, 500-12N,
300-12N, 200-12N,
115-6N, 070-6N

Function: SINGLE CONDUCTOR AND MULTICONDUCTOR CABLE SPLICING

Accuracy: Spec:
Demon:

Service: ELECTRICAL PENETRATIONS

Location:
CONTAINMENT 234' - 246'

Flood Level Elev: 231.2'
Above Flood Level: Yes X
No

ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
Parameter	Specification	Qualification	Specification	Qualification			
Operating Time	CONTINUOUS	30 days		44	SIMULTANEOUS TEST	NONE	
Temperature (^{o}F)	265 (2)	357		44	SIMULTANEOUS TEST	NONE	
Pressure (PSIA)	38 (3)	85		44	SIMULTANEOUS TEST	NONE	
Relative Humidity (%)	100	100		44	SIMULTANEOUS TEST	NONE	
Chemical Spray		H_3BO_3 NaOH		44	SIMULTANEOUS TEST	NONE	
Radiation	1.4×10^7	2.0×10^8	(5)	44	SEQUENTIAL TEST (6)	NONE	R2
Aging		7 days @ $302^{o}F$ 5×10^7 RAD		44	SIMULTANEOUS TEST	NONE	
Submergence	NOT APPLICABLE						R2

NOTES:

(1) Plant procedure developed and approved for installation and checkout - M-521 (Revision 0)
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) See Section 3.2.5 for evaluation
(5) See Section 1.3.2
(6) Test performed prior to (5×10^7R) and after (1.5×10^8R) LOCA simulated environmental exposure R2

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS	
	Parameter	Specification	Qualification	Specification	Qualification			
System: ALL Plant ID No.	Operating Time	CONTINUOUS	4 days		45	SIMULTANEOUS TEST	NONE	R2
Component: TERMINALS, CABLE (1)	Temperature (°F)	265 (2)	350		45	SIMULTANEOUS TEST	NONE	R2
Manufacturer: AMP	Pressure (PSIA)	38 (3)	137		45	SIMULTANEOUS TEST	NONE	R2
Model Number: 53548-1 (wire size - 16)	Relative Humidity (%)	100	100		45	SIMULTANEOUS TEST	NONE	
Function: CONDUCTOR BUTT SPLICE Accuracy: Spec: Demon:	Chemical Spray		H_3BO_3 NaOH		45	SIMULTANEOUS TEST	NONE	
Service: ELECTRICAL PENETRATIONS	Radiation	1.4×10^7	2.0×10^8	(5)	45	SEQUENTIAL TEST (6)	NONE	R2
Location: CONTAINMENT 234' - 246'	Aging				(7)			
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence	NOT APPLICABLE						R2

NOTES:

(1) Plant procedure developed and approved for installation and checkout - M-521 (Revision 0)
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) See Section 3.2.5 for evaluation
(5) See Section 1.3.2
(6) Test performed prior to LOCA simulated environmental exposure
(7) Butt splice connection to be qualified during Wyle Lab test of PVC cable R2

Facility: H. B. Robinson #2

Sheet of 25

SYSTEM COMPONENT EVALUATION WORK SHEET

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCE		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification		
System: ALL Plant ID No.	Operating Time	CONTINUOUS			(4)		(6)
Component: TAPE, SILICON RUBBER	Temperature (°F)	265 (2)			(4)		(6)
Manufacturer: 3M/ELECTRO PRODUCTS DIVISION	Pressure (PSIA)	38 (3)			(4)		(6)
Model Number: SCOTCH 70	Relative Humidity (%)	100			(4)		(6)
Function: CABLE TERMINATION PROTECTION	Chemical Spray				(4)		(6)
Accuracy: Spec: Demon:	Radiation	1.4×10^7		(1)	(4)		(6)
Service: Location: CONTAINMENT	Aging				(4)		(6)
Flood Level Elev: 231.2' Above Flood Level: Yes X No	Submergence				(5)		

R2

NOTES:

(1) See Section 1.3.2
(2) See accident profile - Temperature - Figure 3.1.1
(3) See accident profile - Pressure - Figure 3.1.2
(4) Qualification performed in conjunction with Kerite cable testing per IEEE 323-1974
(5) Not required
(6) Qualification per H.B. Robinson parameters to be performed by Wyle Labs. in conjunction with PVC cable test program

SECTION III - Proposed Corrective Actions for Outstanding Items

The major item at H. B. Robinson which could require replacement or other means of accommodation is the PVC pigtail conductors used for designated electrical penetrations. This was reported by LER to the NRC and a test program initiated to evaluate this cable. A complete IEEE 323-1974 test program was established per H. B. Robinson LOCA profiles and performed at Wyle Laboratories, Huntsville, Alabama. A test report is scheduled for completion and review on September 1, 1981. After evaluation, a report will be made to the NRC concerning test results and any further action(s) required. Tentatively, our response to the NRC should be no later than December 1, 1981.

Included in this qualification test were complete splices consisting of: AMP terminals and splices (53548-1), silicon rubber tape (Scotch 70), and Raychem splicing kits (WCSF-N); these items were installed per CP&L procedure M-521 (Rev. 0). Test evaluations and reports to the NRC on these items will be included in our NRC submittal.

Electronic transmitters qualified to IEEE-323-1974 are not currently available. CP&L is a participating member of the Wisconsin Electric Transmitter Evaluation Program to test and qualify electronic transmitters for nuclear plant use. Presently, CP&L has upgraded its master list transmitters within containment to Rosemount 1153A models which are qualified to IEEE-323-1972. After the current test program is complete - which extends into the fourth quarter of 1982 - CP&L will evaluate the results and decide on action(s) to be taken to provide fully qualified equipments. In the interim, CP&L has established a program which requires changeout of O-rings when yearly calibration checks are performed and a complete instrument changeout on ten-year cycles after installation dates, to maintain qualification level.

Additional commitments to meet the various requirements established by the SER have been covered within this response under the corresponding SER paragraphs. It is anticipated that all of the items reported within this 90-day SER response will be complete by the date established by the NRC for completion of qualification of safety-related electrical equipment.

Section IV - Justification for Continued Operation

Based upon our review of the items listed within Appendix B of the NRC issued Environmental Qualification of Safety-Related Electrical Equipment - Safety Evaluation Report as herein reported (Section I) and noting that no deficiency resolution alters the status of the equipment covered, the concluding statement within the SER should continue to be valid:

"The staff further concludes that there is reasonable assurance of continued safe operation of this facility pending completion of these corrective actions. This conclusion is based on the following:

(1) That there are no outstanding items which would require immediate corrective action to assure safety of plant operation.

(2) Some of the items found deficient have been or are being replaced or relocated, thus improving the facility's capability to function following a LOCA or HELB.

(3) The harsh environmental conditions for which this equipment must be qualified result from low-probability events; events which might reasonably be anticipated during this very limited period would lead to less demanding service conditions for this equipment."

CP&L is confident the master listed and reviewed equipment will work in an accident environment, we have documented our opinion, and qualification testing will be completed by the date established by the NRC for conclusion of qualification of safety-related electrical equipment.

TABLE 6.4-4

SINGLE FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

Component	Malfunction	Comments and Consequences
Spray Nozzles	Clogged	Large number of nozzles (116) renders clogging of a significant number of nozzles as incredible.
Pumps		
1) Containment Spray Pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to two out of four containment cooling fans operating during injection phase.
2) Residual Heat Removal Pump	Fails to start	Two provided. Evaluation based on operation of one pump and no containment cooling fans operating during recirculation phase.
Service Water Pump	Fails to start	Four provided. Operation of two pumps during recirculation required.
4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
Automatically operated Valves: (Open on coincidence of two - 2/3 high [HiHi] containment pressure signals)		
1) Containment spray pump discharge isolation valve	Fails to open	Two provided. Operation of one required (per header).
Valves Operated From Control Room		
(a) Injection		
1) Spray Additive Tank outlet isolation valve	Fails to open	Two provided. Operation of one required. 5

TABLE 6.4-4 (Continued)

Component	Malfunction	Comments and Consequences
5 (b) Recirculation		
1) Containment sump recirculation isolation	Fails to open	Two lines in parallel, each with two valves in series. One line required.
2) Containment spray pump isolation valve from residual heat exchangers	Fails to open	Two valves provided. Operation of one required.
3) Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.

Appendix B

STATUS REPORT

BY THE

DIRECTORATE OF LICENSING

IN THE MATTER OF

WESTINGHOUSE ELECTRIC COMPANY

ECCS EVALUATION MODEL

CONFORMANCE

TO 10 CFR 50, APPENDIX K

OCT 1 5 1974

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

1.0 INTRODUCTION

This staff Safety Evaluation Status Report is divided into eight major sections. Section 1.0 presents a history of the evolution of the new criteria and describes the general limitations of the staff review. Section 2.0 describes the methods of analysis whereby a Westinghouse PWR plant's ECCS performance is evaluated with respect to the five criteria presented in paragraph (b) of 10 CFR 50.46. Section 3.0 presents the results of the staff independent calculations and a comparison to analogous Westinghouse calculations. Section 4.0 addresses each requirement of Appendix K, discusses conformance by Westinghouse, and indicates the acceptability of the analytical methods employed in the model. Section 5.0 provides a separate discussion of the staff's evaluation of the Westinghouse small break model and highlights the important differences with the large break model. Section 6.0 provides a complete status and acceptability of current documentation available on the new Westinghouse ECCS evaluation model. Section 7.0 provides a list of references utilized by the staff during its review. Appendices A, B and C provide the reader with greater detail of the staff's review of the containment back pressure calculation, the staff's independent calculations, and densification of Westinghouse PWR fuels.

Efforts to assess the impact of specific open items in the Westinghouse model (unresolved or unacceptable) are currently under way. Advice from the Advisory Committee on Reactor Safeguards will be sought prior to reaching final decisions. This final assessment will be published in a separate report.

1.1 General

On June 29, 1971, the Atomic Energy Commission published an Interim Statement of Policy establishing acceptance criteria for emergency core cooling systems for light-water-cooled nuclear power reactors. These criteria, which were adopted following a review by the Commission's Regulatory staff and the Advisory Committee on Reactor Safeguards, provided the basis of reasonable assurance that such systems would be effective in the highly unlikely event of a loss-of-coolant accident (LOCA). On November 30, 1971, the Commission announced their decision to hold a legislative-type rule-making hearing for the purpose of determining whether or not the Interim Policy Statement should be retained as issued, or whether these criteria should be adopted in another form. The hearings lasted a total of 125 days and generated a record of more than 22,000 pages of transcript and thousands of pages of written direct testimony and exhibits.

The Regulatory staff filed its Concluding Statement after considering the entire evidentiary record of the proceeding as well as arguments contained in the Concluding Statements filed by other participants. The

staff set forth a Proposed Rule in this statement. After oral arguments, the Commission published the new rule on December 28, 1973.

The principal changes from the Interim Policy Statement (IPS) are as follows: The IPS criterion specifying that the zircaloy clad temperature shall not exceed 2300°F is replaced by two criteria; lowering the peak zircaloy temperature limit to 2200°F and providing a limit on the maximum allowed local oxidation. The other three criteria of the IAC are retained, with some modification of the wording. These three criteria limit the hydrogen generation from metal-water reactions, require maintenance of a coolable geometry, and provide for long-term cooling of the quenched core.

The acceptance criteria for emergency core cooling system effectiveness are as follows:

1. Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2. Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. Maximum Hydrogen Generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. Long-Term Cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The most important effect of the changes in the required features of the evaluation models is that swelling and bursting of the cladding must now be taken into consideration when they are calculated to occur, and that the maximum temperature and oxidation criteria must be applied to the region of clad swelling or bursting when the maximum temperature and oxidation are calculated to occur there. Another important change is the requirement that, in the steady-state operation just before the accident, the thermal conductance of the gap between the fuel pellets and the cladding should be calculated taking into consideration any increase in gap dimensions resulting from such phenomena as fuel densification, and should also consider the effects of the presence of fission gases. When these effects are taken into consideration, a higher stored energy may be calculated.

The development of an appropriate model to account for stored energy changes resulting from the effects of fuel densification has been completed and the Regulatory staff review, which was reported in

"Technical Report on Densification of Westinghouse PWR Fuel," dated May 14, 1974, is included as Appendix C of this report.

Prior to the formulation of the current ECCS criteria, the Interim Acceptance Criterion was applied to the analyses of ECCS performance. Concurrent with the development and formulation of the current ECCS criteria, technological advances in such areas as fuel densification, core flow redistribution during blowdown and accumulator water bypass, have led to a greater degree of understanding and analytical sophistication. This sophistication has modified the ECCS models to such an extent that it is not surprising that some of the arbitrary conservatisms embodied in the IAC models have been replaced by well-justified quantities capable, in some cases, of affecting an overall decrease in calculated peak clad temperatures. These lower calculated peak clad temperatures do not imply a laxity of criteria...on the contrary, the current ECCS criteria have become more limiting in such areas as peak clad temperatures (2300°F versus 2200°F now) and maximum allowed local oxidation. The availability of increasing quantities of meaningful data and the growing technical expertise in many complex facets of the ECCS evaluation models are permitting these analytical tools to be improved to a greater precision than previously attained. Therefore, errors in prejudgment may result when comparing the results from the present ECCS model which satisfies Appendix K requirements to previously analyzed plant designs using the IAC models.

Since the new rule was published on January 4, 1974, the Regulatory staff has met with Westinghouse on several occasions to discuss the Westinghouse ECCS Evaluation Model as its development progressed. Table 1.1 indicates the review chronology.

The staff had previously held ECCS Evaluation Model meetings with Westinghouse in anticipation of the publication of the new rule. Preliminary submittals by Westinghouse permitted the staff to preview the proposed Westinghouse model and furnished an opportunity to question all phases of the model development effort. On April 26, 1974, Westinghouse met with the Advisory Committee on Reactor Safeguards to report their progress on the development of their new model. On May 7, 1974, Westinghouse met with the Directorate of Regulation management to report their progress of the development of their new model. On July 12, 1974, Westinghouse (W) officially filed the following WCAP Topical Reports which constituted a formal description of their ECCS Evaluation Model:

WCAP Report Number	Subject Matter
8202 (P), 8306 (NP)	SATAN-VI Code
8170 (P), 8171 (NP)	WREFLOOD Code
8301 (P), 8305 (NP)	LOCTA-IV Code
8200 Rev 2 (P), 8261 Rev 2 (NP)	WFLASH Code
8327 (P), 8326 (NP)	COCO Code
8341 (P), 8342 (NP)	Sensitivity Studies
8339 (NP)	ECCS Evaluation Model Summary

Table 1.1

CHRONOLOGY OF THE STAFF REVIEW OF THE

WESTINGHOUSE ECCS EVALUATION MODEL

Date	Milestone
Sept. 17, 1973	Discussion of new ECCS evaluation model development at Westinghouse
Dec. 10, 1973	An overview of Westinghouse codes and methods
Jan. 22, 1974	Discussion of implementation of New Acceptance Criterion
Feb. 5, 1974	Technical discussion of Westinghouse evaluation model
Feb. 19, 1974	Technical discussion of Westinghouse evaluation model
March 12, 1974	Technical discussion of Westinghouse evaluation model
April 2, 1974	Swelling and rupture model in Westinghouse ECCS evaluation model
April 3, 1974	Technical discussion of Westinghouse evaluation model
April 23, 1974	Technical discussion of Westinghouse evaluation model
April 26, 1974	Westinghouse progress report to ACRS
May 7, 1974	Westinghouse presentation to Directorate of Regulatory Management
May 8, 1974	Technical discussion of Westinghouse evaluation model
July 12, 1974	Westinghouse official filing of Topical Reports/evaluation model submittal
Aug. 5, 1974	Submittal of remainder of Topical Reports describing Westinghouse evaluation model

On August 5, 1974, additional information was submitted which included:

WCAP Report Number	Subject Matter
8340 (P), 8356 (NP)	Plant Sensitivity Studies
8354 (P), 8355 (NP)	LOTIC Code

WCAP-8340 includes the calculational results of the ECCS evaluation model applied to two-, three-, and four-loop plants. The seventeen volume package submitted to date constitutes the proposed Westinghouse ECCS evaluation model.

1.2 Scope of Review

The review of the Westinghouse ECCS evaluation model was limited to the analytical techniques utilized to predict the course of a postulated loss-of-coolant accident. The design basis loss-of-coolant accident (LOCA) places the most severe requirements upon the emergency core cooling systems. Using the detailed requirements of Appendix K of 10 CFR (50) as a guideline the staff reviewed each relevant aspect of the Westinghouse analytical model and assumptions. Section 4.0 states these requirements, describes how they are met by Westinghouse, and indicates the acceptability of their conformance to the Regulatory staff. An independent calculation was conducted by the Staff using the RELAP-4 EM, FLOOD, and TOODEE codes for blowdown, reflood, and hot channel heatup calculations, respectively. This calculation is described in Section 3.0 and included as Appendix B, Staff Independent Calculations.

1.3 The Westinghouse ECCS Evaluation Model

The Westinghouse ECCS evaluation model for the large and small break LOCA analyses is presented in the following proprietary (P) and nonproprietary (NP) topical reports:

WCAP-8170 (P)	WREFLOOD Code
-8171 (NP)	WREFLOOD Code
-8200, Rev 2 (P)	WFLASH Code
-8261, Rev 1 (NP)	WFLASH Code
-8301 (P)	LOCTA-IV Code
-8302 (P)	SATAN-VI Code
-8305 (NP)	LOCTA-IV Code
-8306 (NP)	SATAN-VI Code
-8327 (P)	COCO Code
-8326 (NP)	COCO Code
-8339 (NP)	ECCS Evaluation Model Summary
-8340 (P)	Plant Sensitivity Studies
-8341 (P)	Sensitivity Studies
-8342 (NP)	Sensitivity Studies
-8354 (P)	LOTIC Code
-8355 (NP)	LOTIC Code
-8356 (NP)	Plant Sensitivity Studies

Section 2.0 describes the method of analysis whereby a Westinghouse PWR plant's ECCS performance is evaluated with respect to the five criteria presented in paragraph (b) of 10 CFR 50.46.

2.0 OVERVIEW OF THE WESTINGHOUSE ECCS EVALUATION MODEL

The large break LOCA transient is divided into three time periods: blowdown, refill, and reflood. Also there are three distinct physical parts of the transient to be analyzed for each time period: thermal-hydraulic transient in the Reactor Coolant System (RCS), pressure and temperature within the containment, and fuel and clad temperature within the hottest fuel rod. These considerations lead to a system of computer models designed to treat the LOCA transient. The LOCTA-IV[1] code is used throughout the entire transient to compute fuel and clad temperatures in the hottest fuel rod. Likewise, the COCO[2] code is used for the complete containment pressure history for dry containments. The LOTIC[3] code is used for ice containment pressure history. The SATAN-VI code is employed for the thermal-hydraulic transient during blowdown while the WREFLOOD[5] code computes this transient during refill and reflood. See Figure 2.1.

For small breaks, the reactor does not empty and thus the core is recovered during blowdown. For these cases the WFLASH[6] code is employed for the thermal-hydraulic transient while the LOCTA-IV code is again used for calculating the clad temperature. Because the highest clad temperature occurs during blowdown, when the break flow is choked (sonic), containment pressure has no influence on ECCS performance and thus need not be considered.

BREAK SIZE \ PHYSICAL PART	THERMAL-HYDRAULIC TRANSIENT IN REACTOR COOLANT SYSTEM	PRESSURE AND TEMPERATURE IN CONTAINMENT	FUEL AND CLAD TEMPERATURE IN HOTTEST ROD
Large	SATAN and WREFLOOD	COCO or LOTIC	LOCTA IV
Small	WFLASH	Not Required	LOCTA IV

Figure 2-1 Role of Westinghouse Computer Codes in Loss-of-Coolant Analysis

2.1 Large Break Analysis

The SATAN-VI code is the first used in the series of calculations which ultimately result in peak clad temperature. Inputs to this model include reactor power and initial conditions, system geometry and hydraulic data, reactor coolant pump characteristic curves, fuel kinetics data, fuel rod conditions, safety injection (SI) performance, and set-points for reactor trip and safety injection. Containment pressure is input also in the determination of break flow for the period of non-critical flow at the end of blowdown. The fluid model within the SATAN-VI code solves the conservation equation of mass, momentum and energy and the equations of state to determine fluid pressure, enthalpy, density, and mass flow rate as a function of time for each SATAN-VI element (control volume).

SATAN-VI uses a multi-element nodal model to describe the Reactor Coolant System in the Westinghouse evaluation model. This model was determined based on sensitivity studies(7) to SATAN-VI noding in the core, steam generator, reactor vessel, and break.

Other models within the SATAN-VI code simulate quantities of interest such as average and hot assembly core conditions, reactor coolant pump performance, plant power transient, ECCS injection, break flow rate and reactor trip and safety injection signal.

For the purpose of ECCS analysis, items of interest computed during blowdown include fluid conditions entering and within the reactor core - particularly the hot assembly - and the mass and energy flow to the

containment. At the end of the SATAN-VI calculations, it is important to know the RCS and accumulator inventories in order to compute the time required to recover the bottom of the core.

The SATAN-VI code is used from the initiation of the accident to the time designated as "End-of-SATAN". This time is defined as the earliest of either downflow in the downcomer region greater than ECCS flow or zero break flow on the vessel side or bottom of core recovered by ECCS water. An additional restriction for "End-of-SATAN" is that "end-of-bypass" must occur prior to "End-of-SATAN." After this time, the SATAN-VI code is no longer used and the WREFLOOD code is applicable. This is shown in Figure 2.2.

Prior to the "End-of-SATAN", an "end-of-bypass" time is determined as the time when sustained ECCS water begins to go down the downcomer. Refill is considered to begin at "end-of-bypass". The water flow down the downcomer is determined from the total flow with the drift flux model. In particular, liquid flow may be down while steam flow or total flow is up.

The purpose of the "end-of-bypass" time is to provide assurance for Appendix K analyses that all water injected up to that time shall not be included in the calculated reactor vessel inventory subsequent to the end-of-blowdown. Accordingly, the SATAN-VI code includes an accumulator (and SI) bypass model which performs an inventory calculation to determine how much accumulator water must be bypassed according to the Appendix K rule and how much water is actually bypassed in the SATAN-VI calculation.

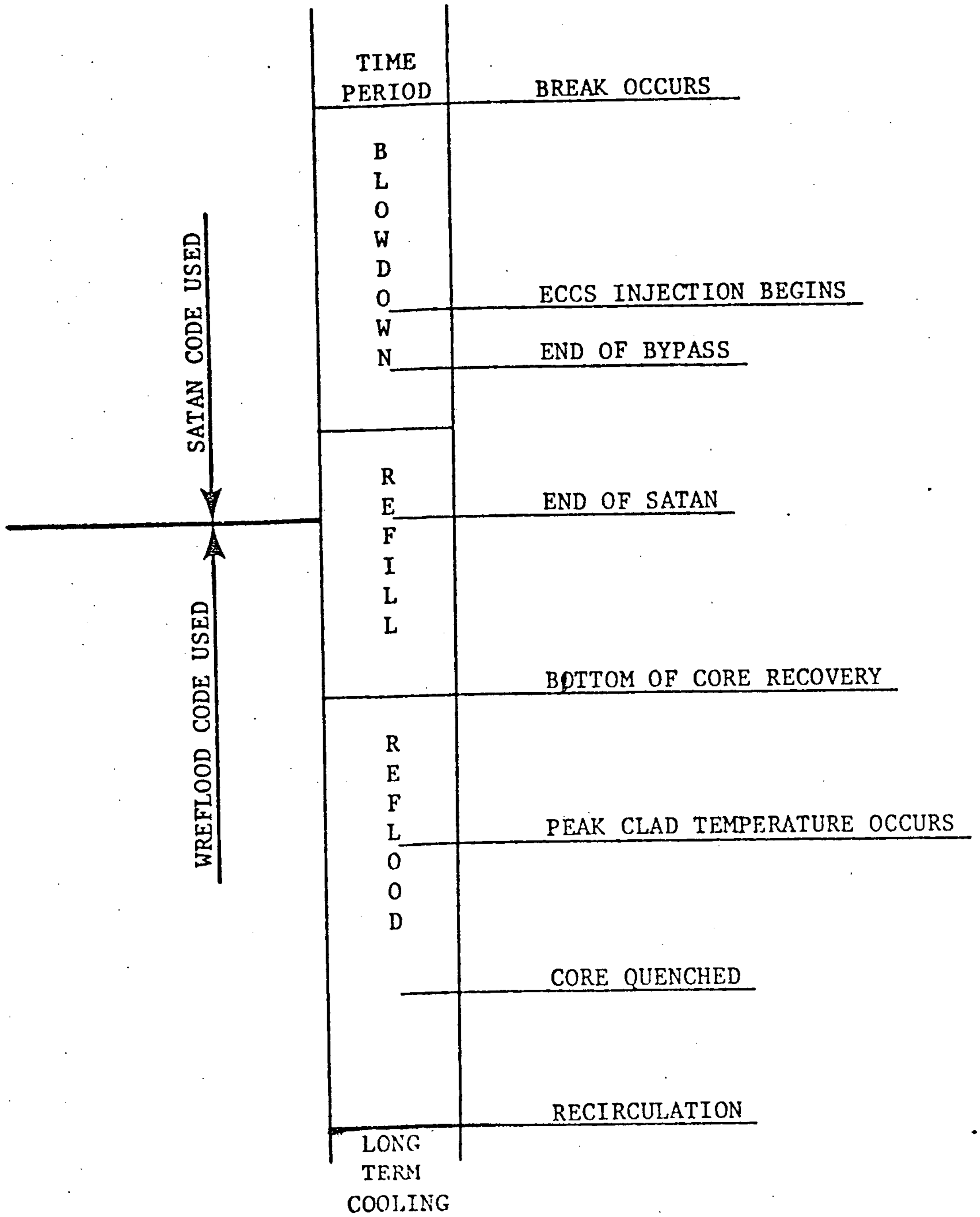


Figure 2-2 - Sequence of Events for Large Break Loss-of-Coolant Analysis

Any deficit in accumulator bypass is subtracted from the vessel inventory at the time of the switch from the SATAN-VI blowdown code to the WREFLOOD reflooding code. The SATAN-VI calculation is not affected by the bypass inventory calculation.

For the refill calculation, WREFLOOD initializes the lower plenum inventory for reflood by determining the available amount of liquid that exists based on "end of SATAN" condition and prevailing containment pressure and subtracts from that the required bypass deficit per the Appendix K rule. Liquid in the intact cold leg pipes and inlet nozzles, broken cold leg nozzle, downcomer and lower plenum is considered available for refill. Negative inventory is disallowed. A fluid transit time from the ECCS injection point to the lower plenum is included. This inventory is increased at a rate determined by the ECCS flow rates until bottom of the core is recovered. At that time reflood begins.

Inputs to the reflood calculation in WREFLOOD include system geometry and hydraulic data, reactor coolant pump characteristic curves, ECCS performance data, core heat flow during reflood as well as steam generator and accumulator conditions at the beginning of the WREFLOOD calculation. The latter two quantities are determined directly from the "End of SATAN" conditions. Reactor coolant pump speed may also be determined directly from SATAN-VI. However, for Appendix K analyses, a locked rotor (zero speed) pump resistance is used. A final quantity determined directly from SATAN-VI is the bypass deficit discussed above.

The primary conservation equation in WREFLOOD is the momentum equation. This equation determines local pressure changes around the reactor coolant loop due to spatial acceleration (area change and density change) and viscous losses (form and friction). Mass velocity is considered uniform except at mixing or separation points.

Enthalpy changes occur due to heating of the water in the lower plenum and downcomer, addition of stored energy and residual heat in the reactor core, addition of heat in the steam generator and mixing at the injection point. Other models within WREFLOOD simulate core heat release, reactor coolant pump performance, residual heat, ECCS injection performance (accumulators and pumps), and break flow.

A key purpose of the WREFLOOD code in ECCS application is to determine the core flooding rate. This is the rate at which liquid enters the bottom of the core. A portion of this liquid is vaporized in the core and this vapor can entrain additional liquid at the phase boundary as it exits the top of the core. The remainder of the liquid accumulates within the core, and the water level is increased. The fluid which exits the top of the core must be vented through the coolant loops and reactor coolant pumps. The driving head for venting is established by the downcomer water level and the core water level. The mass flow rate to be vented is set by the flooding rate and the carryout rate fraction. The volume of steam to be vented depends on the local pressure. Finally, the local pressure depends on the containment pressure. In accordance

with the Appendix K requirements, a conservatively low containment pressure is used for ECCS evaluation.

The containment pressure may be provided for use in WREFLOOD via two methods. A constant back pressure may be specified or a simultaneous calculation of containment pressure can be performed using the COCO code. In either case, the value is insured to be conservatively low via appropriate assumptions in the containment pressure analysis.

The linking of WREFLOOD and COCO is accomplished without sacrifice of either accuracy or flexibility. The codes are linked intimately, i.e., both codes are executed simultaneously; problem times for the two codes are locked into phase, and the relevant interface parameters are continuously exchanged for the current problem time. From the user's standpoint, each of the codes is practically unchanged from the form in which it had been previously used. There has been no degradation of the capability or flexibility of either code. Thus, the two codes retain the same mathematical form that they had when they were used separately for LOCA analysis. The parameters exchanged between the codes are indicated in Figure 2.3, which presents interface data.

For the purpose of ECCS analysis, items of interest computed by the WREFLOOD code include the time at which the bottom of the core is recovered, fluid conditions entering the core - particularly the hot assembly - and the mass and energy flow to the containment.

The WREFLOOD code is used from the "end-of-SATAN" until the clad temperature has peaked. The remainder of the transient indicates a

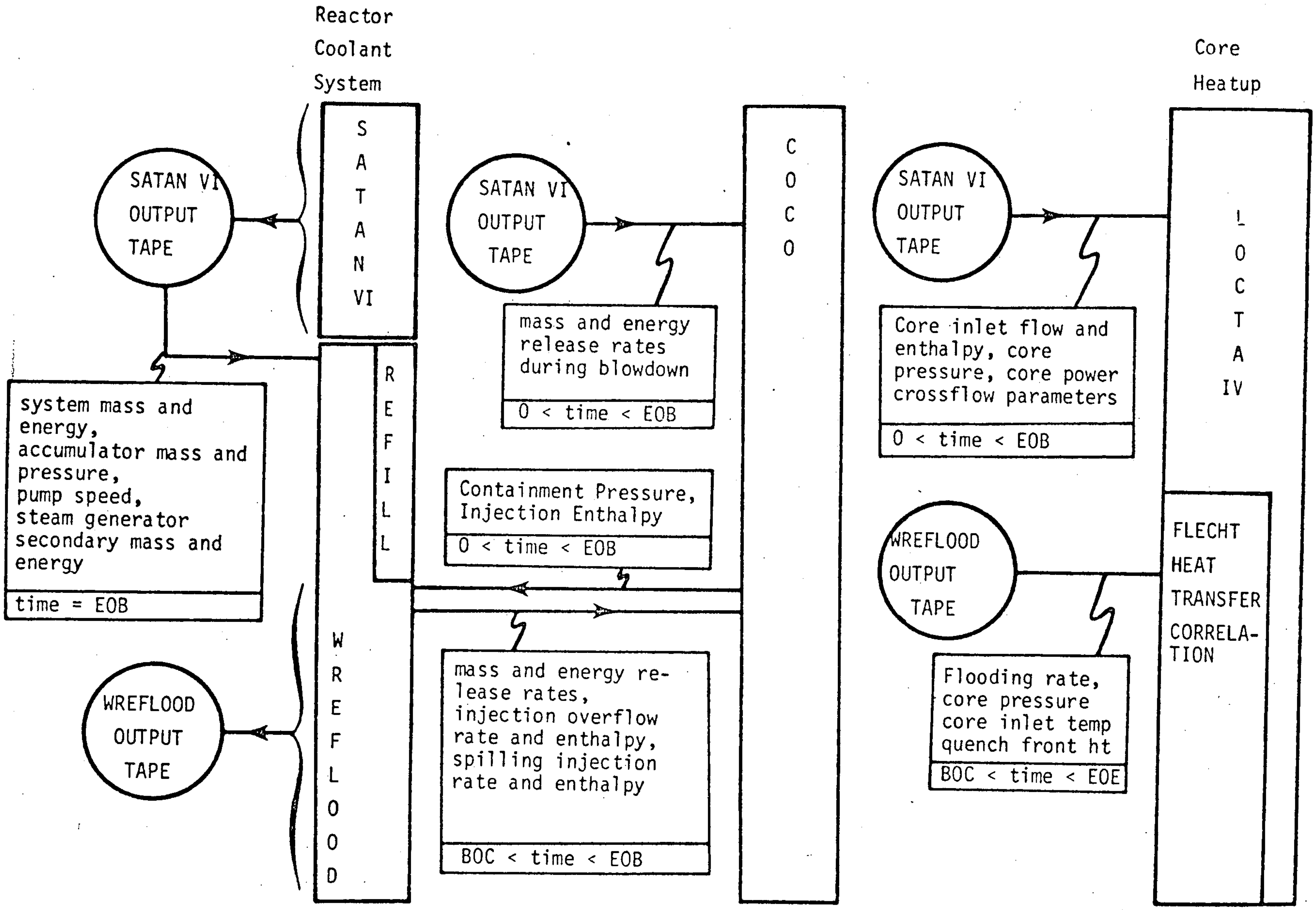


Figure 2-3 - CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL

monotonic reduction in temperature.

The Westinghouse containment pressure transient code, COCO, has been used extensively for containment pressure-temperature design analysis. The application of COCO to the problem of ECCS back pressure analysis is somewhat novel, but requires no major changes in the mathematical formulation of the various models in the code.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into systems. The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam, and subcooled or saturated water. Switching between states is handled automatically by the code. COCO provides analytical models for various containment cooling systems including containment spray, fan coolers, and structural heat sinks.

The overall containment model including containment free volume, spray and fan cooler heat removal capabilities, and containment structural heat sinks will be determined from the individual plant parameters. Suitable conservatism will be applied to each of these parameters on a case by case basis. All containment cooling systems are assumed operable and start times for individual components will be chosen from

the individual plant parameters, on a basis which is consistent with the start time chosen for pumped safety injection. Heat transfer coefficients for structural heat sinks are based on the work of Tagami,[8] with suitable conservatism applied.

The initial containment conditions, pressure, temperature, and relative humidity are the minimum during normal operation. The temperature of water in the refueling water storage tank, and the temperature of service water will be picked on a consistent basis which provides the greatest conservatism for ECCS analysis.

Mass and energy discharge rates during the blowdown portion of the LOCA are available from the SATAN-VI output tape generated for the ECCS blowdown. Thus, blowdown as predicted by the SATAN-VI code directly provides the initialization of the containment pressure at the start of the WREFLOOD - COCO calculation. During the reflooding portion of the transient, mass and energy release rates are transferred to COCO from WREFLOOD on an interactive basis.

The LOCTA-IV code is used to obtain peak clad temperature in the hottest rod. Inputs to this code include initial conditions along the fuel rod including the clad, gap, and pellet. Of particular importance in ECCS analyses are pellet initial temperature and linear power. The pellet initial temperature is chosen at the worst possible time in life. It includes fuel densification and uncertainties per the approved Westinghouse densification model (see Appendix C). The appropriate linear power to be considered in input for ECCS analysis is the maximum

value obtained from operation of the plant within the technical specifications or alternately a parametric study using the ECCS evaluation model may be performed which determines the maximum value of linear power which meets the 10 CFR 50.46 acceptance criteria. The value so determined would constitute an "ECCS limit" for the technical specifications. A choice between the specification methods above would be based on specific plant design and operation.

The hot fuel assembly is divided into three regions. The hot rod is analyzed in order to determine peak clad temperature. A rod adjacent to the hot rod is analyzed to determine the amount of radiation heat transfer from the hot rod to non-burst adjacent rods. The average rod in the hot assembly is analyzed in order that the heat release may be optionally used to determine fluid properties in the hot assembly during blowdown or reflood.

For the determination of hot assembly fluid properties two methods are incorporated in LOCTA-IV. In the first method, fluid properties in the hot assembly are determined from the hot assembly average rod heat release to the fluid. The power in the hot assembly is determined by the assembly peaking factor and the number of fuel rods in the assembly. The fluid properties at the inlet of the hot assembly are taken from SATAN-VI output.

Information from the SATAN-VI code supplied to LOCTA-IV includes hot assembly inlet flows and enthalpies, pressure and depressurization rates, quantities required for the calculation of crossflow, and the

power generated in the fuel during blowdown. The energy and continuity equations are solved at each node for the fluid as it moves up or down the hot assembly, using as boundary conditions SATAN-VI supplied values of flowrate and enthalpy. The following effects are taken into account in the fluid energy equations:

1. energy changes due to heat release from the hot assembly,
2. energy changes due to depressurization,
3. energy changes due to changes in density.

Crossflow due to blockage is calculated from quantities supplied by SATAN-VI and is accounted for in LOCTA-IV as a source term in the continuity equation. The effect of crossflow is thus to add or subtract mass from the hot assembly.

In the second method, SATAN-VI hot assembly fluid properties are used directly as LOCTA-IV hot assembly fluid properties. Since the axial noding of LOCTA-IV is finer than that of SATAN-VI, the mass velocity, pressure, and enthalpy at each LOCTA-IV node are linearly interpolated both in time and space from SATAN-VI information.

Flow rates are defined at each flow path in SATAN-VI. Mass velocity in each LOCTA-IV node is calculated by interpolating this flow rate.

For the pressure, they are calculated in SATAN-VI at the center of each SATAN-VI element (control volume). By interpolation/extrapolation, the pressure at each elevation is calculated.

Enthalpies defined in SATAN-VI elements are considered to be SATAN-VI element enthalpies. From the SATAN-VI enthalpy information at each SATAN-VI point, the enthalpy in each LOCTA-IV node is interpolated.

The hot assembly heat release is used in LOCTA-IV when the core fluid conditions determined in SATAN-VI and WREFLOOD (or WFLASH) are not appropriate for the hot rod clad temperature calculation. This will occur in SATAN-VI (or WFLASH) when the hot assembly is not simulated. It can occur in SATAN-VI when the SATAN-VI hot assembly power is less than the LOCTA-IV hot assembly power. It can occur in WREFLOOD when the flooding rate is less than 1 in/sec and a steam cooling calculation is necessary.

The hot rod, adjacent rod, and hot assembly are modeled with axial nodes placed at intervals along the rods. Additional nodes are placed at 3-inch intervals in the vicinity of the highest power spot. These additional nodes are used to model the burst region.

The fuel rod thermal model solves the transient heat conduction equation for the fuel and cladding. The following effects are taken into account:

1. Power generation and flux depression effects within the fuel.
2. Heat generation within the clad.
3. Variations in fuel and clad thermal properties due to temperature changes and zirconium oxide buildup.

Temperature nodes in the fuel and temperature nodes in the clad are used to calculate the radial temperature distribution within the fuel

rod. Axial conduction in the clad is included in the calculations using the approximation that the axial temperature gradient is that existing at the start of each time step. The LOCTA-IV fuel rod noding model (axial and radial) was determined based on sensitivity studies.

The power assumed to exist in the core at the time of the accident is at least 1.02 times the licensed power level of the plant being analyzed. As mentioned previously, the hot rod peaking factor is the maximum allowed by technical specifications. The axial power distribution assumed to exist in the core at the time of the accident is chosen so as to maximize calculated peak clad temperatures. Power distributions skewed to the top and bottom of the core, as well as the standard cosine power shape, are analyzed for the worst break size.

The burnup which yields the highest calculated stored energy is selected to determine initial values for fuel gap size, gas composition, and gap pressure using standard fuel design methods. These quantities are input to LOCTA-IV which then calculates the corresponding gap conductance and fuel temperature. Additional temperature uncertainties and effects due to densification (in accordance with the Westinghouse Densification Model) are added by increasing the gap width to increase the fuel average temperature. During blowdown prior to burst the gap conductance is calculated as a function of cladding and fuel thermal expansion, elastic deflection due to internal stresses, and temperature and pressure of the gases within the gap. Plastic swelling prior to burst is also included.

Heat generation due to zirconium-water reaction and changes in cladding properties due to oxide buildup are calculated on the outside of the hot adjacent, and average rods using the Baker-Just rate equation. If and when bursting has been calculated to occur, additional metal water reaction and oxide buildup is calculated on the inside of the cladding within a region extending 1.5 inches on either side of the burst point. The rod-to-steam heat transfer regimes considered in LOCTA-IV are:

- Forced Convection to Water
- Nucleate Boiling
- Transition Boiling
- Forced Convection to Steam
- Radiation to Steam
- Reflood Heat Transfer (FLECHT)

In addition, rod-to-rod radiation is considered; this is significant primarily in the burst region.

A summary of the code interfaces (SATAN-VI, WREFLOOD, COCO, and LOCTA-IV) was presented in Figure 2.3.

2.2 Small Break Analysis

For small break analysis, the peak clad temperature occurs during blowdown. Hence, many of the features used for large breaks are not required.

The WFLASH code is similar to SATAN-VI in terms of input data and models. One difference is that phase separation is important for the

longer transients associated with small breaks and this is incorporated in the WFLASH code. Figure 2.4 presents the WFLASH model used in the Westinghouse evaluation model. The interface between WFLASH and LOCTA-IV is shown in Figure 2.5.

The LOCTA-IV code is used to calculate the clad temperature transient in the hot assembly for small breaks from which the peak clad temperature (for small break range only) can be determined. The peak cladding temperature does not exceed 2200°F.

2.3 Compliance with Criterion

2.3.1 Calculation of Peak Clad Temperature

The calculation of peak clad temperature is performed by modelling the hottest fuel assembly (from the reactor core) in the LOCTA-IV code. The hot fuel assembly is subdivided into three regions: (1) the hottest rod, (2) adjacent rod to the hottest rod, and (3) the average fuel channel in the hot assembly. The LOCTA-IV code is used in conjunction with other computer codes (SATAN or WFLASH) which determine necessary thermal-hydraulic boundary conditions for the LOCTA-IV fuel rod heatup analysis.

The method of analysis to determine peak clad temperature is divided into two types of analysis: (1) large break LOCA, and (2) small break LOCA. The method of analysis for large and small break LOCA was compared and described above.

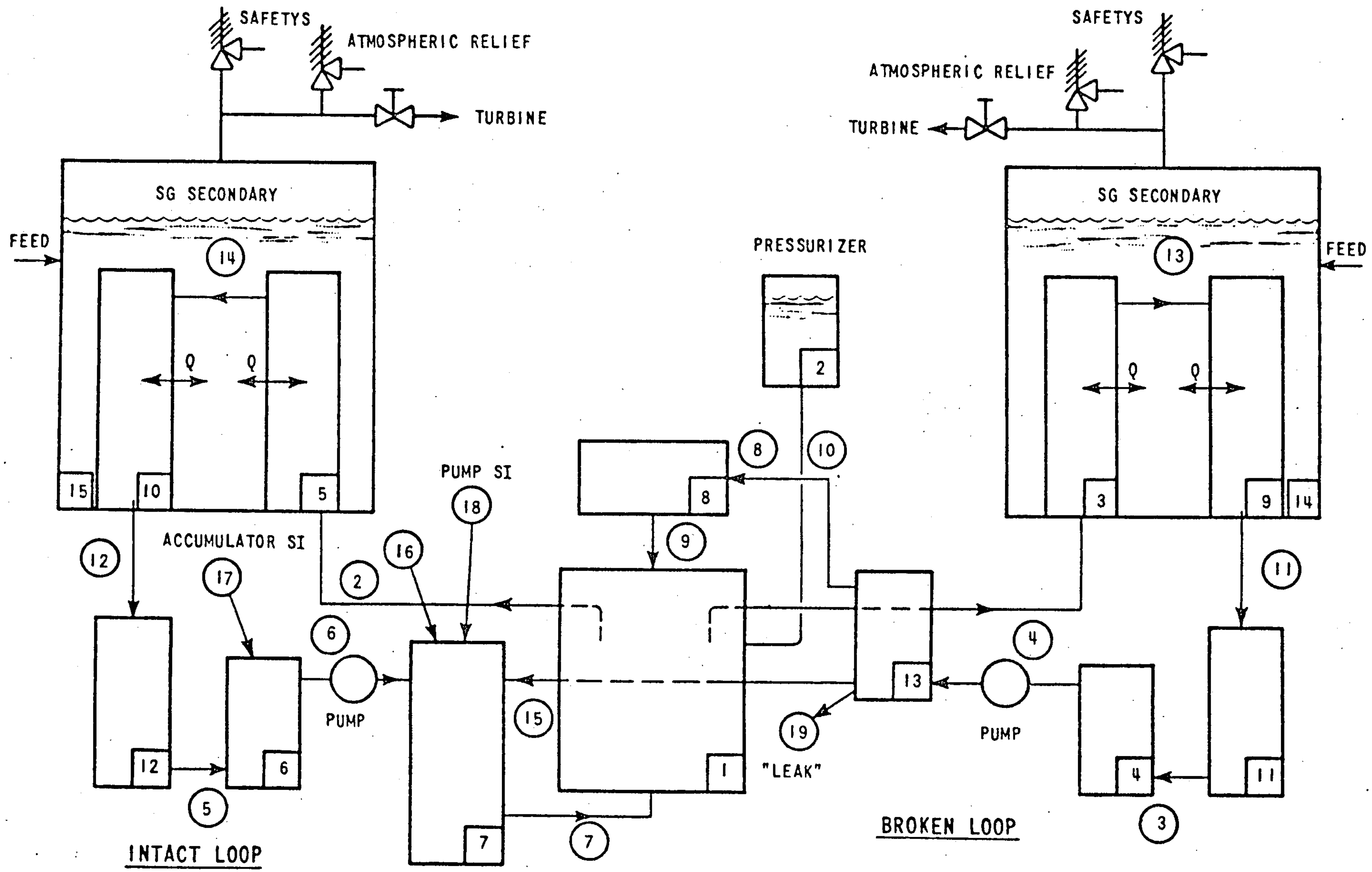
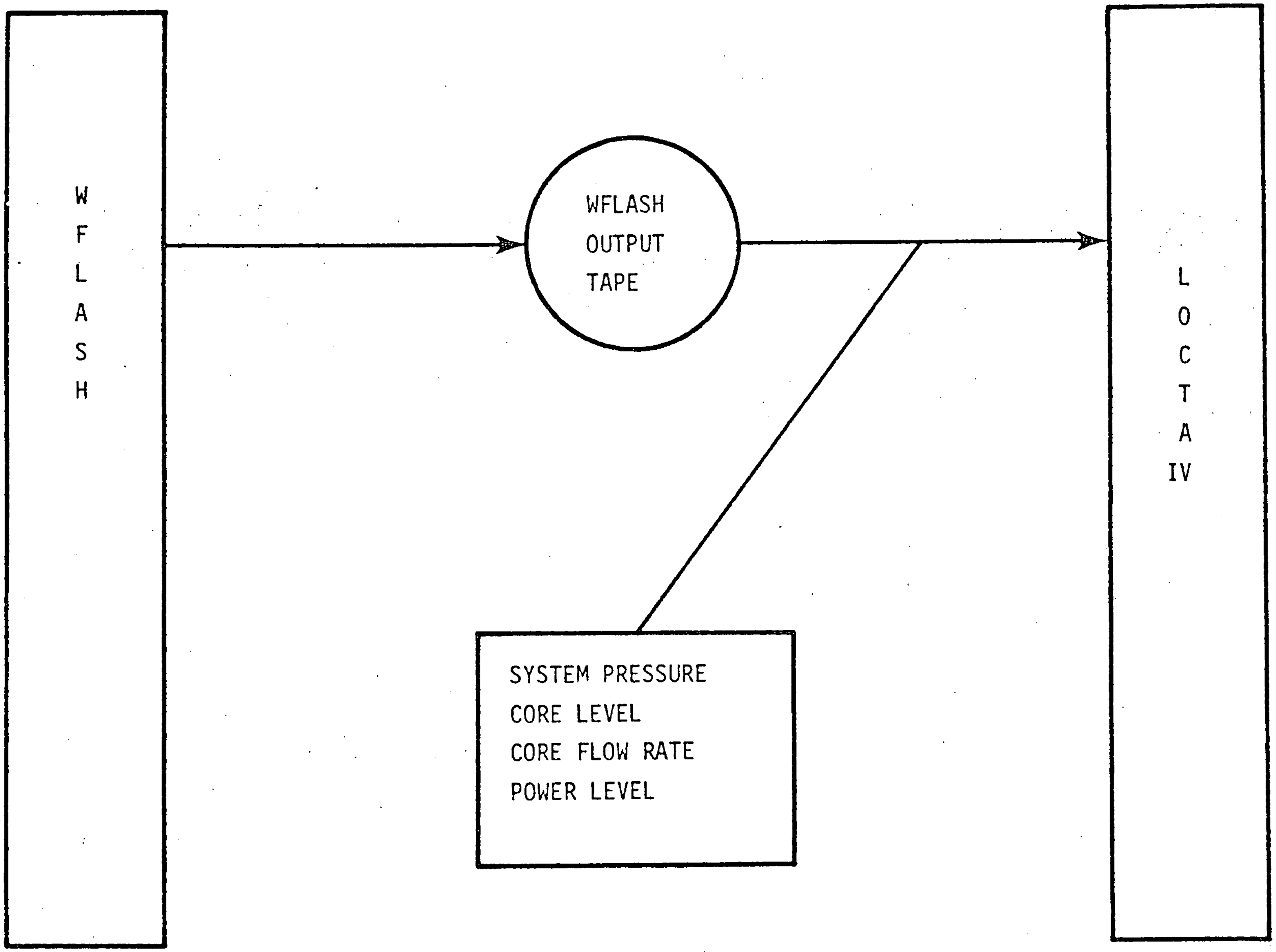


Figure 2-4 WFLASH Model for PWR


W F L A S H
WFLASH OUTPUT TAPE
L O C T A IV
SYSTEM PRESSURE
CORE LEVEL
CORE FLOW RATE
POWER LEVEL

2.3.2 Calculation of Maximum Cladding Oxidation

Cladding oxidation thickness is calculated in LOCTA IV based on the Baker-Just metal-water reaction as required by Appendix K of 10 CFR 50. The method of analysis for the calculation of cladding oxidation is identical to that presented above and is performed in LOCTA-IV when the clad temperature transient is calculated. This 10 CFR 50.46 requirement is usually less limiting than the peak clad temperature limit. The maximum calculated cladding oxidation occurs on the hot rod of the hot assembly and does not exceed 0.17 times the total cladding thickness before oxidation.

2.3.3 Calculation of Maximum Hydrogen Generation

Hydrogen generation is calculated in LOCTA-IV as a byproduct of the Zr-water (metal-water) reaction using Baker-Just equation. The 10 CFR 50.46 requirement, of $\leq$ 1% metal-water reaction, refers to a core wide basis. The method of analysis for calculating the maximum hydrogen generation on a core wide basis is similar to the methods presented above except that a series of LOCTA-IV calculations are made by varying the radial peaking factors in each calculation such that various representative radial power regions in the core can be analyzed for the local metal-water reaction and hence the hydrogen generation. Each representative radial region is analyzed with one LOCTA-IV calculation.

Each region analyzed uses the transient fluid response calculated for the hottest assembly.

The total core-wide hydrogen generation is calculated by convoluting the results of the radial power region analysis with the appropriate radial power distribution. This 10 CFR 50.46 requirement is usually not limiting compared to the 2200°F limit.

2.3.4 Coolable Geometry

The hottest rod in the entire core is analyzed and shown to have margin between computed peak clad temperature and clad melting point. The majority of the rods in the core are substantially cooler and hence no gross core migration is possible.

Changes in geometry due to bursting is calculated in the Westinghouse Evaluation Model based on experimental data. These regions are also shown to be coolable and thus meet this criteria.

2.3.5 Long-Term Cooling

After successful initial operation of the ECCS, the reactor core is recovered with borated ECCS water. This ECCS water has enough boron concentration to maintain core shutdown. Decay heat is removed by a continuous supply of water from the ECCS. This supply initially comes from the refueling water storage tank (RWST). After RWST is empty the ECCS pumps enter a recirculation mode wherein water is drawn from the containment sump and is cooled in the residual heat removal heat exchangers Hence, long-term cooling of the core is maintained by the ECCS. The core is maintained in a shutdown state by borated water. Description of the residual heat removal system is provided in the individual plant SAR.

APPENDIX C

Calculations per Appendix B of IE Bulletin 79-01B to Determine Total Anticipated Radiation

VOLUMETRIC CALCULATIONS FOR EQUIPMENT COMPARTMENT

Step 1:

Reactor Power Level = 2300 MWth
Containment Volume = 2.1×10^6 ft.^{3}

30-day dose = $\underline{1.4 \times 10^7 \text{ RADS}}$

Step 2:

36" Wall (Concrete Shielding)

Dose = $\underline{1.5 \times 10^3 \text{ RADS}}$

Step 3:

Compartment Volume = 2.8×10^5 ft.^{3}

Correction Factor = $\underline{0.45}$

$$0.45(1.4 \times 10^7) + 1.5 \times 10^3 = 6.3015 \times 10^6$$
$$= \underline{6.3 \times 10^6 \text{ RADS}} \text{ (30-day dose)}$$

Step 4:

1/2 hour Correction Factor = 0.09 $0.09(6.3 \times 10^6) = \underline{5.7 \times 10^5 \text{ RADS}}$

1 hour Correction Factor = 0.15 $0.15(6.3 \times 10^6) = \underline{9.5 \times 10^5 \text{ RADS}}$

24 hour Correction Factor = 0.55 $0.55(6.3 \times 10^6) = \underline{3.5 \times 10^6 \text{ RADS}}$

Time (hrs.)	Dose (RADS)	Dose + 10% Margin (RADS)
1/2	5.7×10^5	
1	9.5×10^5	1.0×10^6
24	3.5×10^6	3.8×10^6

VOLUMETRIC CALCULATIONS FOR OPERATING FLOOR COMPARTMENT

Step 1:

Reactor Power Level = 2300 MWth
Containment Volume = 2.1×10^6 ft.^{3}

30-day dose = 1.4×10^7 RADS

Step 2:

Not Applicable

Step 3:

Compartment Volume = 1.6×10^6 ft.^{3}

Correction Factor = <u>0.80</u>

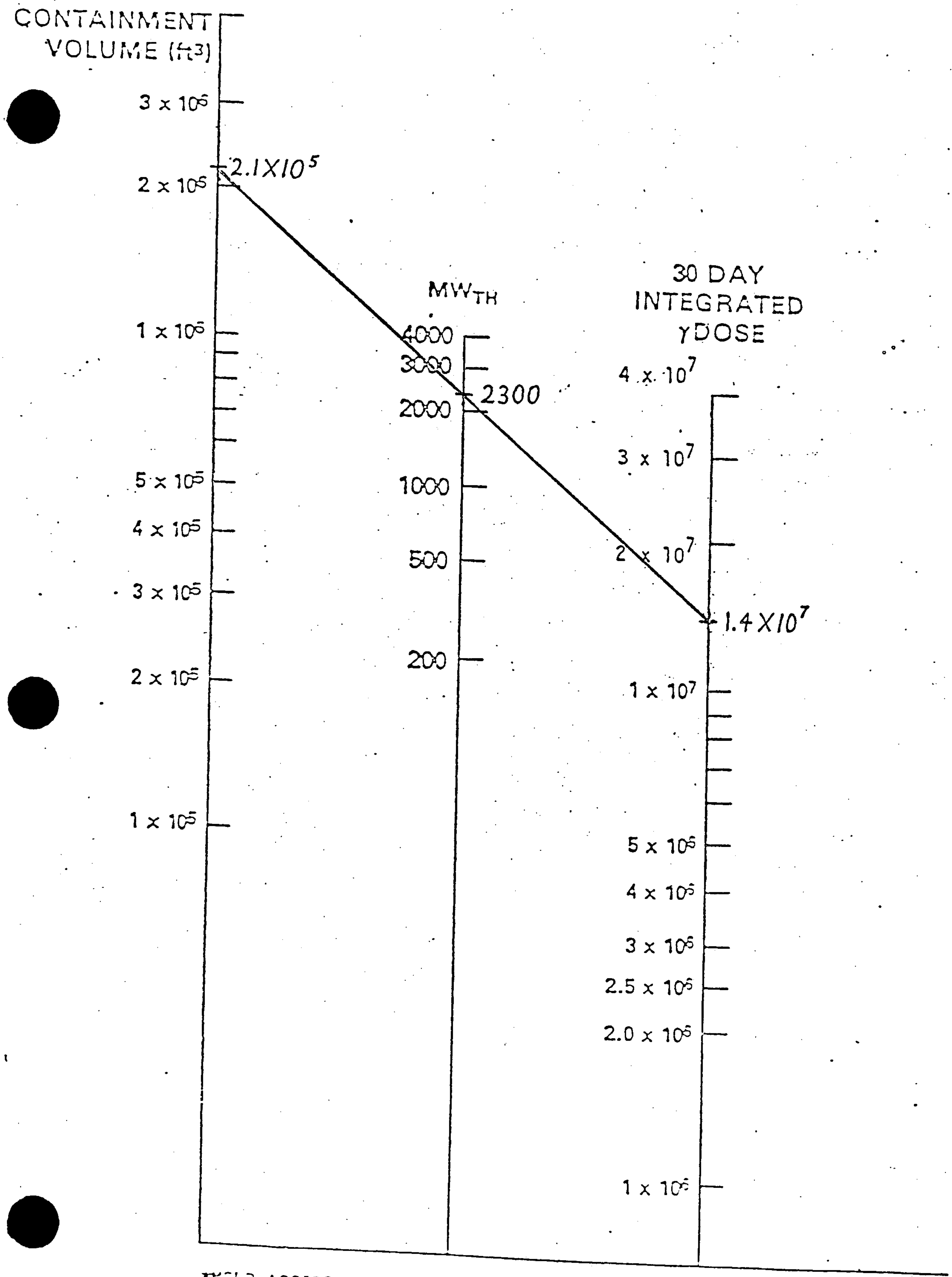
$0.08(1.4 \times 10^7) = \underline{1.12 \times 10^7 \text{ RADS}}$ (30-day dose)

Step 4:

1/2 hour Correction Factor = 0.09 $0.09(1.12 \times 10^7) = \underline{1.0 \times 10^6 \text{ RADS}}$

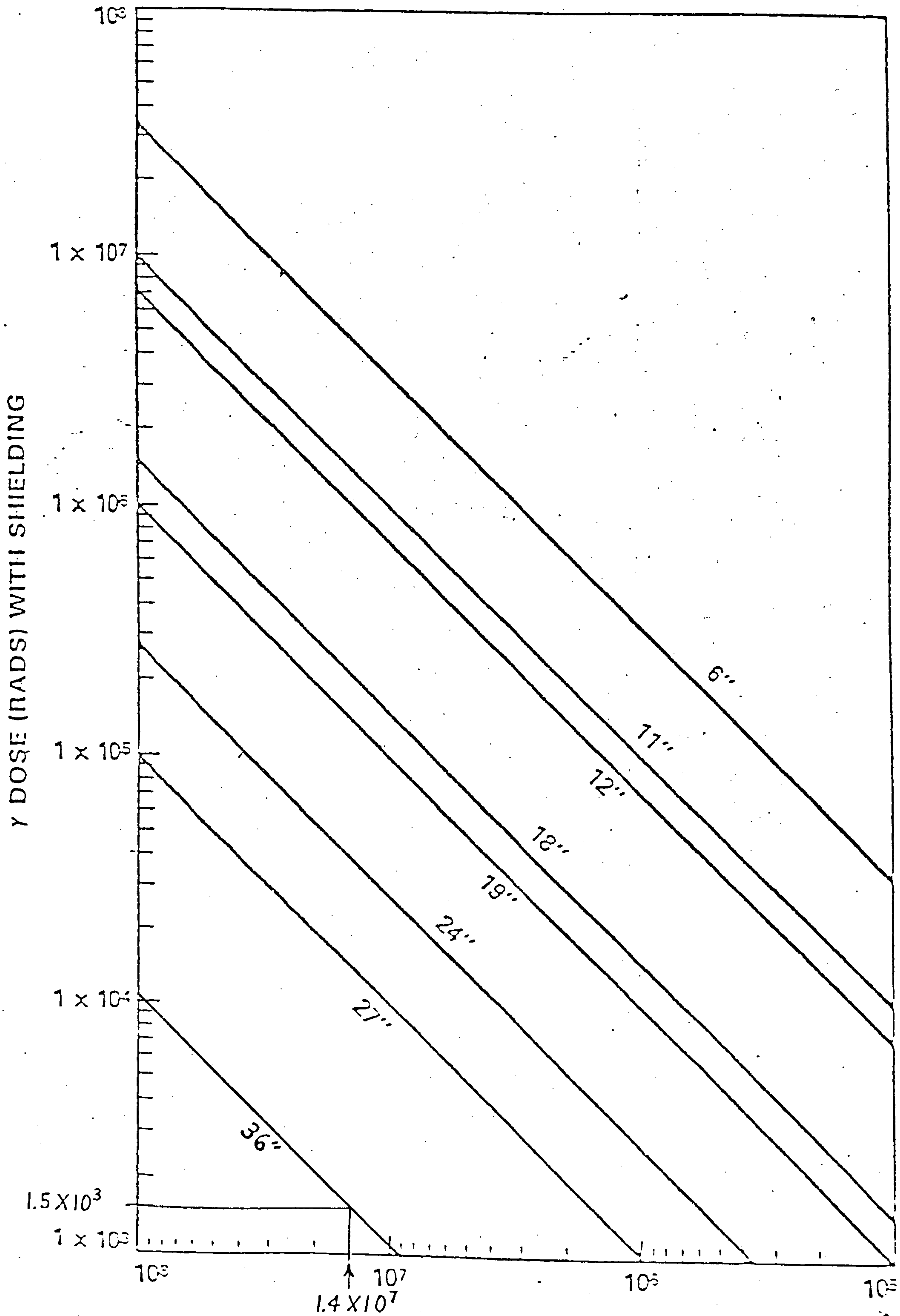
3 hour Correction Factor = 0.28 $0.28(1.12 \times 10^7) = \underline{3.1 \times 10^6 \text{ RADS}}$

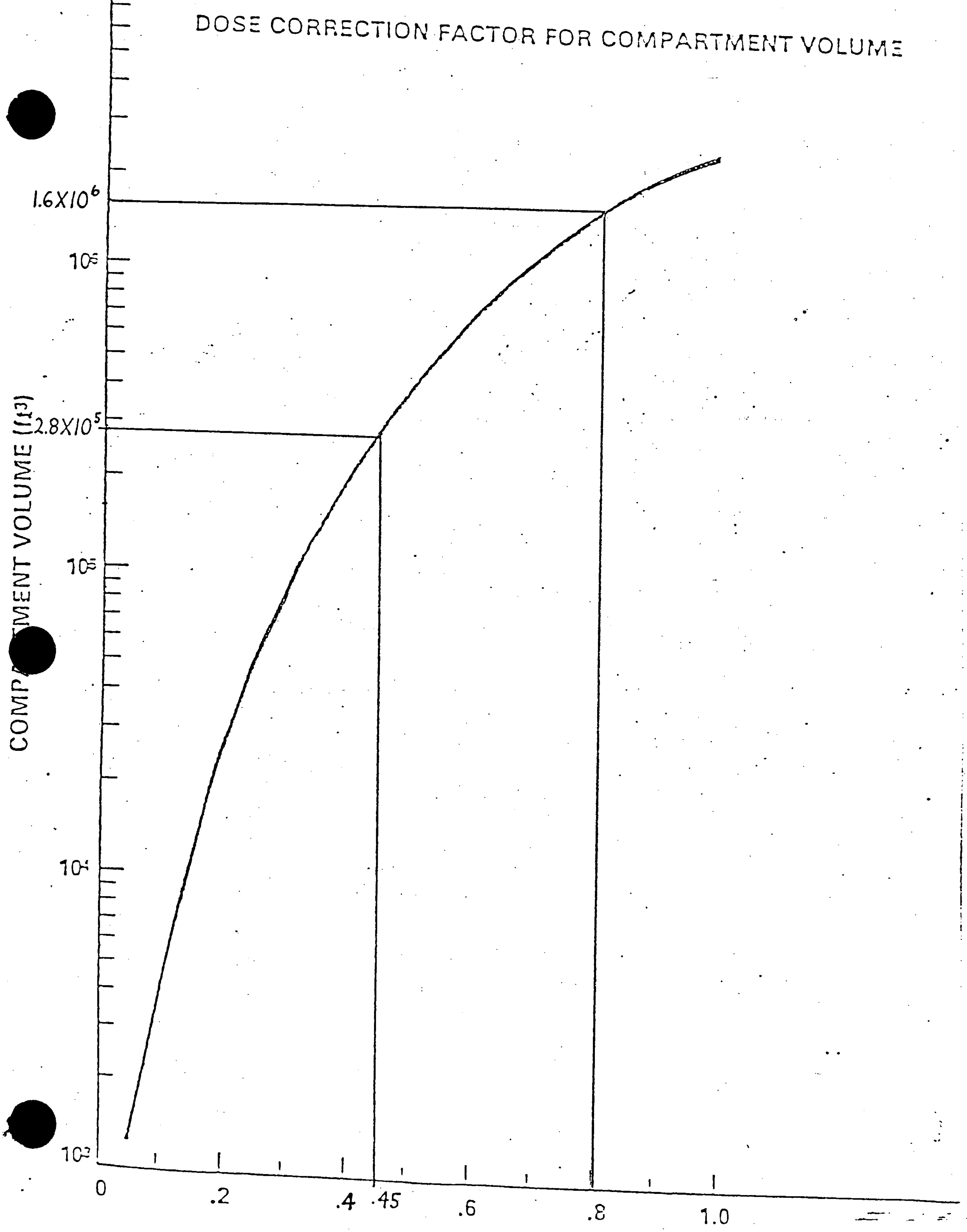
Time (hrs.)	Dose (RADS)	Dose + 10% Margin (RADS)
1/2	1.0×10^6	------
3	3.1×10^6	3.4×10^6



*MSLB ACCIDENT DOSES SHOULD BE READ AS A FACTOR OF 10 LESS

DOSE CORRECTION FACTOR FOR CONCRETE SHIELDING
(γ ONLY)





DOSE CORRECTION FACTOR FOR COMPARTMENT VOLUME
COMPARTMENT VOLUME (ft³)
1.6X10⁶
10⁶
2.8X10⁵
10⁵
10⁴
10³
0
.2
.4
.45
.6
.8
1.0

DOSE CORRECTION FOR TIME REQUIRED TO REMAIN FUNCTIONAL