UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 31, 1990

NOTE TO: D. Lanham
NUDOCS

FROM: H. Smith

SUBJECT: R. LICCIARDO - DIFFERING PROFESSIONAL OPINION (DPO) CONCERNING
(A) ZION 1/2 CONTAINMENT ISOLATION VALVES, AND
(B) METHODOLOGY USED FOR CALCULATING OFFSITE DOSES

Enclosed is a Chronology (Enclosure 1) related to the October 19, 1989 memorandum from R. Licciardo to J. Taylor, EDO, which consists of a Differing Professional Opinion (DPO) concerning Zion. Enclosure 2 are records pertaining to the DPO. Please make the Chronology and the enclosed records available in Central Files and the Public Document Room. Mr. Blaha, EDO, agrees with this action. Please contact me on 21287 if additional information is needed.

HS

Hazel Smith, NRR

Enclosures:
1. Chronology - DPO
2. Records related to DPO, as listed on Enclosure 1

cc w/Encl 1:
R. Licciardo
J. Blaha
D. Ross
T. Murley
J. Sniezek
J. Partlow
F. Miraglia
F. Gillespie
J. Larkins
V. Wilson
A. Thadani
G. Holahan
S. Varga
C. Patel
C. McCracken

DFOA Add: Hazel Smith Ltr Encl
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Enclosure 1

CHRONOLOGY

Pertaining to the DPO filed by Mr. Licciardo with the EDO on Zion

Item No.	Description

1. Memo dated 10/19/89 from R. Licciardo to J. Taylor, EDO, which constitutes a formal submission of a Differing Professional Opinion (DPO) concerning Zion isolation valves, and methodology used for calculating offsite doses. (1 page memo; 4 page attachment) the following items are enclosures to the DPO.

   Enclosure 1 Memo (proposed by R. Licciardo) dated 5/11/89 from J. Wermiel to D. Muller, subject: Offsite Radiological Consequences of LOCA During Containment Purge Proposed in Technical Specification (TS) Changes for Zion 1 and 2," with SER and SALP input (memo 2 pages; SER 3 pages; SALP 1 page).

   Enclosure 2 Memo dated 5/10/89 from J. Wermiel to D. Muller, subject: "Proposed TS Changes on Purge/Vent Operation" (2 page memo; 3 page SER; 1 page SALP input; page 202a of TS).

   Enclosure 3 Memo dated 5/11/89 from R. Licciardo to T. Murley, which submits the differing professional view regarding Zion and offsite doses, w/memo (proposed by R. Licciardo) dated 5/11/89 from Wermiel to Muller attached (9 pages).

   Enclosure 4 Memo dated 7/20/89 from R. Licciardo to F. Miraglia submitting information requested by F. Miraglia memo dated 5/11/89 (70 pages).

   Enclosure 5 Zion/FSAR Section 9.10, "Plant Ventilation," Design Basis for Auxiliary Building Ventilation and Containment Purge Systems (35 pages).

   Enclosure 6 Memo dated 9/13/89 from T. Murley to R. Licciardo consisting of his conclusions in regard to Mr. Licciardo's DPV. (1 page) with the 8/31/89 Review Panel memo (3 pages), and References 1-6 as listed below:

      Reference 1 Management Response to Oversight Committee. (57 pages)

      Reference 2 Background Information Related to DPV (28 pages), including Branch Tech. Position CSB 6-4 and Section 4.2 of NUREG-0800

      Reference 3 The memo dated 7/20/89 described in Enclosure 4.

Reference 4 Memo dated 8/11/89 from A. Thadani to F. Miraglia, DPV concerning containment isolation valves at Zion (19 pages).

Reference 5 Note from A. Thadani to F. Miraglia dated 8/24/89, DPV concerning containment isolation valves at Zion, w/memo dated 8/23/89 from W. Hodges and 8/21/89 from N. Lauben. (14 pages)

Reference 6 Note from A. Thadani to F. Miraglia dated 8/29/89, with memo from R. Jones dated 8/25/89; DPV to T. Murley dated 5/11/89 with listing of 10 memoranda; 5/18/89 memo from T. Murley to Licciardo acknowledging receipt of DPV; 5/25/89 memo from R. Licciardo to T. Murley naming panel members; 5/26/89 memo from T. Murley to Panel (Miraglia, Rossi, Congel); 6/2/89 memo from F. Miraglia to R. Licciardo; 6/23/89 memo from F. Miraglia to R. Licciardo; 6/30/89 memo from R. Licciardo to F. Miraglia; 7/14/89 memo from R. Licciardo to F. Miraglia; 7/14/89 memo from R. Licciardo to F. Miraglia; memo 7/14/89 from R. Licciardo to F. Miraglia correcting a date appearing in the initial 7/14/89 memo; 7/21/89 memo from F. Miraglia to R. Licciardo with chronology.

2. Memo dated 11/8/89 from J. Taylor to R. Licciardo acknowledging receipt of 10/19/89 DPO. This memorandum names Dr. Ross of RES as head of independent review groups. T. Murley and L. Soffer are also members of this review group.

3. Memo dated 11/30/89 from T. Murley to R. Licciardo stating that the staff will complete its review of the Zion Technical Specifications for containment purge and vent systems after Dr. Ross' group completes its revie of this issue.

4. Memo dated 1/2/90 from J. Taylor to R. Licciardo, subject: Disposition of DPO - An Independent, Outside, Qualified Review

5. Memo dated 1/8/90 from J. Taylor to T. Murley, subject: operational usage of large purge system valves-PWR's; re-examination of NRR's safety policy and practices

Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 11, 1989

Docket Nos. 50-295
and 50-304

MEMORANDUM FOR: Daniel Muller, Director
Project Directorate III-2
Division of Reactor Projects III, IV, V
and Special Projects

FROM: Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: OFFSITE RADIOLOGICAL CONSEQUENCES OF LOCA DURING CONTAINMENT PURGE PROPOSED IN TS CHANGES FOR ZION 1 AND 2

Reference: Letter to H. R. Denton (NRC) From P. C. Leonard dated February 2, 1986, Subject: Zion Nuclear Power Station, Units 1 and 2 Proposed Amendment to Facility Operating License No. DPR-39 and DPR-48

Plant Name: Zion Nuclear Power Station, Units 1 and 2
Licensee: Commonwealth Edison Company
TAC Nos.: 55417 and 55418
Review Status: Complete

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units; see reference 1.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

The enclosed Safety Evaluation Report has been prepared by the technical reviewer initially assigned to this task, namely Robert B. A. Licciardo.

The licensee's proposal is to allow full power operation of the facility with the 42" purge supply and exhaust containment isolation valves open to a limited position of 50°, and capable of isolation within seven (7) seconds of the commencement of a LOCA.

The review concludes that the 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequence of the offsite dose to thyroid (from iodine) during a LOCA is unacceptable high; whole body dose has not been evaluated: The least value for the additional offsite dose which may be proposed within the licensing basis is 64,000 rem over the first seven (7) seconds.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of resulting source terms.

~~8906140120~~ 3pp.

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

Enclosures:
As stated

cc w/enclosures:
C. Patel

CONTACT: R. Licciardo
X20876

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

Enclosures:
As stated

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C. Patel

CONTACT: R. Licciardo
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RLicciardo;cf	JKudrick	JWermiel
5/11/89	5/ /89	5/ /89

5520 NAME: Zion TACs 55417/8 Licciardo

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PLANT SYSTEMS BRANCH
OFFSITE RADIOLOGICAL CONSEQUENCE OF LOCA DURING
CONTAINMENT PURGE
ZION NUCLEAR POWER STATION, UNITS 1 AND 2
DOCKET NOS. 50-295 and 50-304

1.0 INTRODUCTION

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

2.0 EVALUATION

Background review shows that the facility was evaluated on the basis of normally closed purge valves so that these consequences were never included in the Zion SER. Further, that a letter from Westinghouse (W) to Commonwealth Edison Company dated October 22, 1976 on the subject of "Offsite Doses During LOCA and Containment Purge" (Ref. 2) has never been evaluated by the NRC. Subsequent to the TMI-2 event, the operability and automatic control of these valves was evaluated leading to the request for the required TS, but the Radiological Assessment was left as a "long(er) term issue" (Ref. 3) which was intended to be resolved in a subsequent probabilistic risk assessment which definitively excluded it from consideration without any justification (Ref. 4).

The W analyses undertaken under Commonwealth Edison instruction, uses an RCS operational inventory of 60 uc/gm equivalent I 131 at the time of the accident with a resulting site boundary thyroid dose due to iodine (during closure of the valves), of 52 rem, and which added to the containment leakage dose of 123 rem gives a total 175 rem which is within the 10 CFR 100 limit of 300 rem. The total iodine inventory of the RCS is assumed to be released into containment on initiation of the LOCA; a 50% plate out is assumed leaving the residual 50% as part of containment inventory for discharge out through both fully open containment purge lines for a total of seven (7 seconds).

However, when reviewed against the BTP CSB 6-4, Item B.5.a requires that:

> "The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concommitment release of fission products, and the fission product activity in the primary coolant."

~~8411020127~~ 4pp

Further: SRP 4.2 identifies fuel failure with infringement of DNBR criteria, with the related requirement that gap activity be considered as part of the source term, and Regulatory Guide 1.77 recommends that under similar circumstances, gap activity should be assumed at 10% of core activity. Fuel damage criteria also includes the occurrence of center line melting with measures of additional activity release also guided by Regulatory Guide 1.77, but the Zion SAR shows this does not occur.

Revising the source term to Appendix K calculations [in which all fuel goes to DNBR in ½ second] with related release of all gap activity into containment, with limited blowdown to offsite during the related 7 seconds closure time and absent a 50% plate out of iodine as can be interpreted from the above referenced item B.5.a, increases offsite dose due to containment purge above by a factor of 3400 to 176,000 rem and would thereby be completely unacceptable. Limiting the purge line valves to an opening of 50° could reduce offsite dose to 64,000 rem and represents the least value which may be proposed within the licensing basis.

Note: The BTP CSB 6-4 proposing that valve closure within 5 seconds will ensure purge valves are closed before the onset of fuel failures has since been extended by the staff on a plant-specific basis to 15 seconds. Further, the writer cannot find any safety evaluation report supporting these positions. These positions cannot be sustained for Zion since a) DNBR infringement (from Appendix K calculations) and hence fuel failure and gap activity release [Ref. SRP 4.2) of 10% of core inventory (Ref. Regulatory Guide 1.77) occur within ½ second of the initiation of the LOCA, b) related maximum clad temperatures of 1750°F occur immediately and never reduce below 1400°F, c) RCS pressure in the region of the core rapidly reduces from 2250 psia to 900 psia in 7 seconds increasing potential pressure drop across the cladding for release of gap activity to the RCS inventory, d) the massive bulk boiling and blowdown surrounding the failed fuel ultimately discharges 270,000 lbs of RCS inventory into the containment at 7 seconds into the event increasing containment pressure from 0.3 psig to 23.8 psig (in these 7 seconds), and e) causes 15,000 lbs of the resulting containment inventory to be discharged to the environment through 2x42" fully open lines, or 5400 lbs for the same lines with valve closed to 50°.

3.0 CONCLUSION

The 42" valves at Zion should remain closed in Modes 1, 2, 3, and 4 because the consequences of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body dose has not been evaluated. The least value for offsite dose to the thyroid which may be proposed within the existing licensing basis is 64,000 rem.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of the resulting source terms.

References

1. Letter from P. C. Blond (CECo) to H. R. Denton (NRC); Subject: Zion, Units 1 and 2, Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 dated February 21, 1986.

2. Letter from R. L. Kelley (W) to C. Reed (CECo); Subject: Offsite Dose During LOCA and Containment Purge, dated October 22, 1986.

3. Letter to L. O. DelGeorge (CECo) from S.A. Varga (NRC); Subject: Generic Concerns of Purging and Venting Containments, dated September 9, 1981.

4. Memo for F. H. Robinson from R. W. Houston, Subject: "Evaluation of the Risk at Zion," dated August 14, 1985.

Enclosure 2

SPLB SALP INPUT

Plant Name: Zion Nuclear Generating Stations, Units 1 and 2
SER Subject: Containment Purge and Vent Valve Operation
TAC Nos.: 55417/8

Summary of Review/Inspection Activities

The licensee provided an evaluation of offsite doses undertaken in 1976. This was undertaken with a methodology and source term chosen by the licensee. The licensee did not present results from alternative more detailed methodologies which could be considered enforceable under existing regulatory positions and the related circumstances.

Narrative Discussion of Licensee Performance - Functional Area

The single only methodology used by the licensee is not an acceptable approach for estimating doses under the proposed circumstances and especially since alternate detailed evaluations required by the SRP give greatly increased values beyond 10 CFR Part 100 limits. A prudent approach would have recognized the deficiencies and risks in the single methodology adopted with resulting substantively different recommendations to ensure public health and safety.

Author: Robert B. A. Licciardo

Date: May 11, 1989

Enclosure 2

May 10, 1989

Docket Nos. 50-295
and 50-304

MEMORANDUM FOR: Daniel Muller, Director
Project Directorate III-2
Division of Reactor Projects III, IV, V
and Special Projects

FROM: Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: PROPOSED TS CHANGES ON PURGE/VENT OPERATION

Reference: Zion Nuclear Power Station, Units 1 and 2 Proposed Amendment to Facility Operating License No. DPR-39 and DPR-48, letter to H. R. Denton (NRC) From P. C. Leonard dated February 2, 1986

Plant Name: Zion Nuclear Power Station, Units 1 and 2
Licensee: Commonwealth Edison Company
Review Status: Complete

The Plant Systems Branch has reviewed Commonwealth Edison's proposed changes to the Technical Specifications on containment purge and vent valve operation for Zion Units 1 and 2, as described in a letter dated February 21, 1986. The proposed changes are either administrative in nature or are to comply with the generic concerns of MPA B-24 as it is related to demonstration of containment purge and vent valve operability. Based on the enclosed safety evaluation report (Enclosure 1), the Plant Systems Branch concludes that the proposed Technical Specifications are acceptable.

There is one possible follow-up item that should be clarified with the licensee, however. There is some question as to how the licensee intends to preclude opening the purge/vent valve beyond the 50 degree angle as specified in the TS. Discussions with the Mechanical Engineering Branch (MEB) have indicated that a positive stop is required on the valve to prevent opening beyond the TS angle. Operational procedures, by themselves, are not acceptable. Since none of the incoming information addresses how the opening will be limited, the Project Manager should verify with the licensee that a positive stop has been installed on the valve. If this is not the case, this issue should be pursued with MEB.

~~8905180081~~ XA 2pp.

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

[illegible]

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

Enclosures:
As stated

cc w/enclosures:
C. Patel

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5520 NAME: Zion TACs 55417/8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PLANT SYSTEMS BRANCH
PROPOSED TECHNICAL SPECIFICATIONS
CONTAINMENT PURGE
ZION NUCLEAR POWER STATION, UNITS 1 AND 2
DOCKET NOS. 50-295 and 50-304

1.0 INTRODUCTION

Commonwealth Edison, the owner of the nuclear power plants Zion Units 1 and 2, proposed in a letter to H. Denton dated February 21, 1986, an amendment to Facility Operating License Nos. DPR-39 and DPR-48. The amendment proposed changes to the Technical Specifications (TS) related to vent and purge operations as well as restricting the maximum purge valve position. These changes were in response to an NRC request in a Safety Evaluation Report dated April 3, 1984. Simply stated, the request was to reflect the permissible operation of the purge and vent valves into the TS. The submittal contains the requested changes.

2.0 EVALUATION

The proposed changes related to restrictions in purge and vent operations. Specifically, they include the allowable angle the purge supply and exhaust valves can be opened, the number of valves that can be used at one time, the valve closure time, and the goal for purging time in one year. Each of these changes will be discussed below.

However, before the individual TS changes are discussed, there is one surviellance test that was recommended in the staff SER that was not added to the proposed TS. The staff had recommended the periodic leakage testing of the valves with resilient seals. The frequency was to be once per three months during operating Modes 1 through 4, if the valves were considered to be active.

In response to this request, the licensee indicated that the additional surveillance requirement was not needed for the valves at Zion because the isolation valve seal water system and penetration pressurization system are designed to continuously detect any leakage during plant operation. If leakage is detected, an alarm is sounded in the control room. The staff has reviewed the licensee's justification for not performing the added leakage tests. As part of their justification, the licensee, in the bases Section 3.4 of the TS, indicated that the seal water is introduced at a pressure of 50 psig. This pressure is slightly higher than the peak containment post accident pressure. Further, the seal water system and penetration pressurization system are included in TS Section 3.9.1 and 3.9.2 which includes limiting condition for operation (LCO) and surveillance requirements.

~~8911020170~~ 4pp

Based on the above, the staff concludes that the continuous leakage detection systems now in place at Zion Units 1 and 2 satisfy the requirements of the surveillance leakage tests referenced in the staff's SER. In addition, the current TS on the leakage systems meets the intended purpose of the suggested added TS. Therefore, the staff concurs with the licensee that no additional surveillance testing or added TSs are necessary.

The proposed TS indicating that the purge supply and exhaust valves shall not be opened more than 50 degrees is consistent with the staff's SER dated April 3, 1984. Therefore the staff finds the proposed TS acceptable. The acceptance of the allowable opening angle is based, in part, on the demonstration of acceptable stresses within the valve. An equally important parameter in determining the closure stresses is the closure time. The staff concluded, as documented in the April, 1984 SER, that acceptable closure times range between 5 and 8 seconds. The proposed TS change, in this regard, is to change the surviellance test value from the current 60 seconds to 7 seconds. The revised closure time reflects the acceptable stress analysis and is therefore acceptable.

Another proposed change is to assure that the containment purge valves shall not be open concurrently with the containment vent valves. This operational restriction is consistent with the guidelines set forth in SRP Section 6.2.4 to minimize the number of pathways open at any one time. Based on this compliance with the SRP, the staff finds the operational guidance provided for vent and purge operation acceptable.

An important consideration in the development of an effective program is the selection of a usage factor as well as the reasons for vent and purge operation. The licensee has proposed a goal of 2000 hours per year. This time has been established based upon the licensee's estimate to limit the concentration of radioactive materials in the containment atmosphere to less than 100 times the maximum permissible concentration per 10 CFR 20. After review of the purging criteria, the staff has concluded that the program including the goal established by the licensee is acceptable. However, due to the importance the staff has placed on the need to minimize purging or venting of the containment, the staff believes that additional clarification should be added to the TS to ensure that purging be performed only for safety related reasons. A marked up copy of the appropriate TS page is enclosed which the staff would find acceptable. The licensee has agreed to the staff's proposed markup in a series of telephone conferences. Based on the verbal agreement of the marked up changes, the staff finds the proposed use of the purge and vent systems acceptable.

An additional consideration must be included in the overall evaluation of the purging program, in light of the fact that large diameter valves are being used for time periods greater than 90 hours. For these conditions, SRP Section 6.2.4 indicates that the radiological consequences of a LOCA concurrent with the purge/vent valves assumed open at time zero must be calculated. The analysis should show that 10 CFR Part 100 limits are not exceeded.

Guidance is provided in the SRP concerning the source term to be used for calculating the dose consequences due to the release through the valves until closure. The guide indicates that for valve closure times within five seconds, isolation is assured prior to onset of fuel failure. This has been interpreted by the staff to mean that only the pre-existing iodine spike need to be considered in determining primary coolant activity without the need for further justification. For closure times slightly beyond 5 seconds, the staff has evaluated the merits of assuming no fuel failure on a case by case basis. Consideration has included the transport times necessary to sweep the source from the failed fuel into the reactor coolant, from the fuel pins to the postulated pipe rupture, from the pipe rupture to the nearest pipe inlet of the open purge line, and finally through the duct to the isolation valve. Based on this rationale, the staff has concluded that there will be a substantial time delay between the onset of fuel failure and the actual release of products from the containment as a result of the fuel failure. Additionally, there will be a finite minimum time before initiation of fuel failure can occur. Using the above rationale, the staff has concluded that a more reasonable upper bound of valve closure time for which no source term contribution due to fuel failure can be conservatively assumed is 15 seconds.

Therefore, for the Zion closure time of seven seconds, the staff has concluded that fuel failure need not be considered. Based on the above, the staff has concluded that only the pre-existing iodine spike need be considered.

The licensee has computed the dose consequences considering the above source term. The results show that using a 60 uc/gm equivalent I-131 spike at the time of the accident, the site boundary thyroid dose due to iodine up until valve closure is 52 rem. When added to the containment leakage dose of 123 rem yields a total dose of 175 rem. This is well within 10 CFR 100 requirements of 300 rem.

The staff has performed an independent calculation of the dose contribution due to releases through the purge/vent pathways. The results confirm the licensee's value. Based on this agreement, the staff finds that the dose consequences due to purging operations are acceptable and within 10 CFR 100 limits.

3.0 CONCLUSION

Based on the above evaluation, the staff concludes that the proposed changes to the Zion Units 1 and 2 Technical Specifications for limitation on purge and vent valve operation above cold shutdown are more restrictive than current TSs and consistent with the commitments identified in the staff SER on the same subject. Therefore, the staff finds the proposed changes acceptable.

5520 NAME: Zion TACS 55417/8

Enclosure 2

SPLB SALP INPUT

Plant Name: Zion Nuclear Generation Stations, Units 1 and 2
SER Subject: Containment Purge and Vent Valve Operation
TAC Nos.: 55417/8

Summary of Review/Inspection Activities

The licensee initially proposed Technical Specification changes for containment purge and vent valve operation needed revision. However, data revisions adequately addressed the concerns.

Narrative Discussion of Licensee Performance - Functional Area

The licensee's approach for resolution of generic concerns related to the demonstration of containment purge and vent valve was viable and sound from a safety standpoint.

Authors: J. Kudrick and C. Li

Date: May 10, 1989

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.9.6 CONTAINMENT VENTILATION SYSTEM	**4.9.6 CONTAINMENT VENTILATION SYSTEM**
A. The purge supply and exhaust isolation valves shall be limited to a maximum opening of 50 degrees and may only be opened for safety-related reasons.	A. 1. The position of the containment purge supply and exhaust isolation valves shall be determined at the beginning of purge or venting operations and at least once per week while venting or purging.
B. The containment vent line shall be isolated whenever a containment purge line is open and may only be opened for safety-related reasons.	2. When purge isolation valve position is being controlled by regulating the air pressure to the valve operator, the air pressure shall be measured at the beginning of the purge operation and daily while purging.
APPLICABILITY: Modes 1, 2, 3, 4, and 7	3. The cumulative gaseous radioactive effluent release shall be determined once per month for the purpose of verifying compliance with the gaseous effluent release limits.
ACTION: a. With the purge supply or exhaust isolation valve(s) open greater than 50 degrees, return the valve(s) to an acceptable position within 1 hour or terminate purge operations and close and deactivate at least one in-series purge isolation valve or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	4. At least once per 18 months valves RV0001, RV0002, RV0003, RV0004, RV0005 and RV0006 shall be closed manually from the control room and the closing time measured. Performance will be acceptable if the valves close within seven seconds.
b. With the containment vent isolation valves and purge isolation valves open simultaneously, isolate one of the flow paths within 1 hour or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	

Enclosure 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 11, 1989

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance of SER to Zion 1/2 allowing full power operation with open 42" containment isolation valves.

b) Methodology used for calculating related offsite doses.

The writer submits a Differing Professional View (DPV) in accordance with the provisions of NRC Manual Chapter 4125.

This issue has arisen out of the Safety Evaluation Report (SER) undertaken for the Zion Units 1 and 2 as prepared by the writer; see Attachment.

The principal issue is the prudent and conservative calculation of the additions to offsite dose which may result from a LOCA at a facility during the use of open purge supply and exhaust valves at full power.

The licensee for Zion 1/2 has proposed full power operation of the facility with the 42" purge supply and exhaust containment isolation valves open to a limited position of 50°, and capable of isolation within seven (7) seconds of the commencement of a LOCA.

The writers SER concludes that the 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequence of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body has not been evaluated. The least value for the additional offsite dose which may be proposed within the licensing basis is 64,000 rem over the first seven (7) seconds of the LOCA. Management staff has disagreed with the writer's methodology and conclusion and plans issuance of a separate SER permitting the operation requested. The writer requests non-issuance of the related SER to the licensee. He also proposes probability of a generic action on other facilities which have been granted such licenses based on the staff's current methodology.

In general, the management staff has adopted a criterion described in SRP BTP CSB 6-4 which is that providing the maximum time for closure of these containment isolation valves does not exceed 5 seconds (and by plant-specific exception, up to 15 seconds), then the valves would be closed before the onset of fuel failure following a LOCA so that the only contribution to offsite dose is from RCS operational levels of fission product directly discharged into containment during this period, and then through the open containment isolation valves before closure.

~~8909140128~~ XA 2pp.

In evaluating the consequence for Zion, the writer has used an alternate Criterion in BTP CSB 6-4 which states that:

"The following analyses should be performed to justify the containment purge system design:

An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values."

Using these related guidelines for Zion, the fuel performance over the 0-7 seconds is detailed and shows that fuel failure (by infringement of DNBR criteria) occurs within ½ seconds of the commencement of the LOCA, and together with other licensing basis responses including fission product release from the fuel gap and the thermal hydraulic conditions in the core, containment and discharge nozzle, result in a substantive discharge of fission products to the environment of far greater consequence than are calculated by the staff.

The relative consequences of these differing approaches are that whereas the staff methodology gives additions to offsite dose resulting in total doses within 10 CFR Part 100 limits, the alternate approach used by the writer shows a substantially increased offsite dose exceeding 10 CFR Part 100 limits, with completely unacceptable consequences to Public Health and Safety.

The writer requests review of the Differing Professional View in a timely manner in accordance with the provisions of NRC Manual Chapter 4125.

R. Licciardo

Robert B. A. Licciardo
Registered Professional Engineer California
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cc: J. Sniezek
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J. Wermiel
J. Kudrick

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 11, 1989

Attachment

Docket Nos. 50-295
and 50-304

MEMORANDUM FOR: Daniel Muller, Director
Project Directorate III-2
Division of Reactor Projects III, IV, V
and Special Projects

FROM: Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: OFFSITE RADIOLOGICAL CONSEQUENCES OF LOCA DURING CONTAINMENT PURGE PROPOSED IN TS CHANGES FOR ZION 1 AND 2

Reference: Letter to H. R. Denton (NRC) From P. C. Leonard dated February 2, 1986, Subject: Zion Nuclear Power Station, Units 1 and 2 Proposed Amendment to Facility Operating License No. DPR-39 and DPR-48

Plant Name: Zion Nuclear Power Station, Units 1 and 2
Licensee: Commonwealth Edison Company
TAC Nos.: 55417 and 55418
Review Status: Complete

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units; see reference 1.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

The enclosed Safety Evaluation Report has been prepared by the technical reviewer initially assigned to this task, namely Robert B. A. Licciardo.

The licensee's proposal is to allow full power operation of the facility with the 42" purge supply and exhaust containment isolation valves open to a limited position of 50°, and capable of isolation within seven (7) seconds of the commencement of a LOCA.

The review concludes that the 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequence of the offsite dose to thyroid (from iodine) during a LOCA is unacceptable high; whole body dose has not been evaluated: The least value for the additional offsite dose which may be proposed within the licensing basis is 64,000 rem over the first seven (7) seconds.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of resulting source terms.

~~8909140120~~ XA 3pp.

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

Enclosures:
As stated

cc w/enclosures:
C. Patel

CONTACT: R. Licciardo
X20876

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

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As stated

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5520 NAME: Zion TACs 55417/8 Licciardo

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PLANT SYSTEMS BRANCH
OFFSITE RADIOLOGICAL CONSEQUENCE OF LOCA DURING
CONTAINMENT PURGE
ZION NUCLEAR POWER STATION, UNITS 1 AND 2
DOCKET NOS. 50-295 and 50-304

1.0 INTRODUCTION

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

2.0 EVALUATION

Background review shows that the facility was evaluated on the basis of normally closed purge valves so that these consequences were never included in the Zion SER. Further, that a letter from Westinghouse (W) to Commonwealth Edison Company dated October 22, 1976 on the subject of "Offsite Doses During LOCA and Containment Purge" (Ref. 2) has never been evaluated by the NRC. Subsequent to the TMI-2 event, the operability and automatic control of these valves was evaluated leading to the request for the required TS, but the Radiological Assessment was left as a "long(er) term issue" (Ref. 3) which was intended to be resolved in a subsequent probabilistic risk assessment which definitively excluded it from consideration without any justification (Ref. 4).

The W analyses undertaken under Commonwealth Edison instruction, uses an RCS operational inventory of 60 uc/gm equivalent I 131 at the time of the accident with a resulting site boundary thyroid dose due to iodine (during closure of the valves), of 52 rem, and which added to the containment leakage dose of 123 rem gives a total 175 rem which is within the 10 CFR 100 limit of 300 rem. The total iodine inventory of the RCS is assumed to be released into containment on initiation of the LOCA; a 50% plate out is assumed leaving the residual 50% as part of containment inventory for discharge out through both fully open containment purge lines for a total of seven (7 seconds).

However, when reviewed against the BTP CSB 6-4, Item B.5.a requires that:

> "The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concommitment release of fission products, and the fission product activity in the primary coolant."

~~8911020127~~ 4pp.

Further: SRP 4.2 identifies fuel failure with infringement of DNBR criteria, with the related requirement that gap activity be considered as part of the source term, and Regulatory Guide 1.77 recommends that under similar circumstances, gap activity should be assumed at 10% of core activity. Fuel damage criteria also includes the occurrence of center line melting with measures of additional activity release also guided by Regulatory Guide 1.77, but the Zion SAR shows this does not occur.

Revising the source term to Appendix K calculations [in which all fuel goes to DNBR in ½ second] with related release of all gap activity into containment, with limited blowdown to offsite during the related 7 seconds closure time and absent a 50% plate out of iodine as can be interpreted from the above referenced item B.5.a, increases offsite dose due to containment purge above by a factor of 3400 to 176,000 rem and would thereby be completely unacceptable. Limiting the purge line valves to an opening of 50° could reduce offsite dose to 64,000 rem and represents the least value which may be proposed within the licensing basis.

Note: The BTP CSB 6-4 proposing that valve closure within 5 seconds will ensure purge valves are closed before the onset of fuel failures has since been extended by the staff on a plant-specific basis to 15 seconds. Further, the writer cannot find any safety evaluation report supporting these positions. These positions cannot be sustained for Zion since a) DNBR infringement (from Appendix K calculations) and hence fuel failure and gap activity release [Ref. SRP 4.2) of 10% of core inventory (Ref. Regulatory Guide 1.77) occur within ½ second of the initiation of the LOCA, b) related maximum clad temperatures of 1750°F occur immediately and never reduce below 1400°F, c) RCS pressure in the region of the core rapidly reduces from 2250 psia to 900 psia in 7 seconds increasing potential pressure drop across the cladding for release of gap activity to the RCS inventory, d) the massive bulk boiling and blowdown surrounding the failed fuel ultimately discharges 270,000 lbs of RCS inventory into the containment at 7 seconds into the event increasing containment pressure from 0.3 psig to 23.8 psig (in these 7 seconds), and e) causes 15,000 lbs of the resulting containment inventory to be discharged to the environment through 2x42" fully open lines, or 5400 lbs for the same lines with valve closed to 50°.

3.0 CONCLUSION

The 42" valves at Zion should remain closed in Modes 1, 2, 3, and 4 because the consequences of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body dose has not been evaluated. The least value for offsite dose to the thyroid which may be proposed within the existing licensing basis is 64,000 rem.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of the resulting source terms.

References

1. Letter from P. C. Blond (CECo) to H. R. Denton (NRC); Subject: Zion, Units 1 and 2, Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 dated February 21, 1986.

2. Letter from R. L. Kelley (W) to C. Reed (CECo); Subject: Offsite Dose During LOCA and Containment Purge, dated October 22, 1986.

3. Letter to L. O. DelGeorge (CECo) from S.A. Varga (NRC); Subject: Generic Concerns of Purging and Venting Containments, dated September 9, 1981.

4. Memo for F. H. Robinson from R. W. Houston, Subject: "Evaluation of the Risk at Zion," dated August 14, 1985.

Enclosure 2

SPLB SALP INPUT

Plant Name: Zion Nuclear Generating Stations, Units 1 and 2
SER Subject: Containment Purge and Vent Valve Operation
TAC Nos.: 55417/8

Summary of Review/Inspection Activities

The licensee provided an evaluation of offsite doses undertaken in 1976. This was undertaken with a methodology and source term chosen by the licensee. The licensee did not present results from alternative more detailed methodologies which could be considered enforceable under existing regulatory positions and the related circumstances.

Narrative Discussion of Licensee Performance - Functional Area

The single only methodology used by the licensee is not an acceptable approach for estimating doses under the proposed circumstances and especially since alternate detailed evaluations required by the SRP give greatly increased values beyond 10 CFR Part 100 limits. A prudent approach would have recognized the deficiencies and risks in the single methodology adopted with resulting substantively different recommendations to ensure public health and safety.

Author: Robert B. A. Licciardo

Date: May 11, 1989

Enclosure 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 20, 1989

MEMORANDUM FOR: Frank J. Miraglia, Associate Director
for Inspection and Enforcement

FROM: Robert B. A. Licciardo, Reactor Engineer
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: DIFFERING PROFESSIONAL VIEW (DPV) CONCERNING CONTAINMENT ISOLATION VALVES AT ZION

On May 11, 1989, The writer submitted a memo on the subject:

Differing Professional View Concerning

a) Issuance Of SER To Zion 1/2 Allowing Full Power Operation With Open 42" Containment Isolation Valves

b) Methodology Used For Calculating Related Offsite Doses

By memo of May 11, 1989, from F. J. Miraglia to R. Licciardo, the writer was asked to clarify certain aspects of the regulatory positions used in the analyses including the time to failure used in LOCA analyses and mechanisms for the transport of fission products from the primary (system) to the containment.

The writer was also asked to provide a view as to the safety significance of the Amendment proposed by management and the safety significance of my concern regarding LOCA analyses.

In response to the above request, I am pleased to submit the enclosed document which analyzes for your specific concerns and presents the related conclusions in Section 4.

Regarding the safety significance of the existing Zion Amendment proposed by management. Use of that Amendment and required Regulatory Guide 1.4 criteria would result in a contribution to thyroid dose over seven (7) secs. of 158,000 rem; using DNBR failure criteria with 10% fission product gap release would reduce this to 64,000 rem. Use of DNBR failure and equilibrium gap activity only would contribute 27,000 rem.

It would take a fuel failure of only 0.2% of the existing rods releasing 10% gap activity only to increase offsite doses to 10 CFR 100 limits.

~~8909140111~~XA 2pp.

It must be recognized that allowing the containment purge valves to remain open for seven (7) secs. following a LOCA, multiplies by 194,000 the amount of fission product that would otherwise be release by leakage over the same period of seven (7) secs. from an isolated containment. It becomes a direct contradiction of the regulatory need for effective containment and limited leakage.

In summary: Proceeding with the existing Amendment proposed by management would be in direct violation of regulatory requirements.

The writer's SER of May 11 issued with his DPV of that date remains the writer's safety conclusions and recommendations in this matter i.e.:

"The 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequences of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body dose has not been evaluated. The least value for offsite dose to the thyroid which may be proposed within the existing licensing basis is 64,000 rem.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel failure and the quantification and treatment of the resulting source terms."

R. Licciardo

Robert B. A. Licciardo
Registered Professional Engineer California
Nuclear Engineering License No. NU 001056
Mechanical Engineering License No. M 015380

Enclosure:
As stated

cc: J. Sniezek
C. Rossi
F. Congel
H. Smith

Enclosure

AN EVALUATION OF THE CRITERIA FOR

AND

THE CALCULATION OF OFFSITE DOSES DERIVING FROM
OPEN CONTAINMENT PURGE VALVES DURING
A LOCA AT ZION UNITS 1 & 2

DATED JULY 20, 1989

PREPARED BY

ROBERT B. A. LICCIARDO
REGISTERED PROFESSIONAL ENGINEER CALIFORNIA
NUCLEAR ENGINEERING LICENSE NO. NU 001056
MECHANICAL ENGINEERING LICENSE NO. M015380

~~8909140189~~ 68 pp.

INTRODUCTION

On May 11, 1989, the writer submitted a memo on the subject:

DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance Of SER to Zion 1/2 Allowing Full Power Operation With Open 42" Containment Isolation Valves.

b) Methodology Used For Calculating Related Offsite Doses.

By memo of May 11, 1989, from F. J. Miraglia to R. Licciardo, the writer was asked to clarify certain aspects of the regulatory positions used in his analysis including: a) Time to failure used in LOCA analysis and b) mechanisms for the transport of fission products from the primary (system) to the containment. The writer was also asked to provide his view as to the safety significance of the Amendment proposed by management, and the safety significance of his concerns regarding LOCA analysis.

This material was prepared in response to that request and is in adjunct to his D.P.V which is attached to this document as Attachment 1.

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1 FISSION PRODUCT RELEASED FROM FUEL AND CONTAINMENT USED IN ACCIDENT ANALYSES

1.1 Radiological Source Terms Within The Core

Exhibit 1 shows core and gap activities for Zion for iodine.

- Calculated levels of iodine in the fuel clad gap are given to show a total I-131 EQU of 24.09×10^5 curies
- Total iodine in the core as I-131 EQU is 15.79×10^7 curies.

1.2 LOCA: Reg. Guide 1.4 Criteria: Application to Zion

Branch Technical Position CSB 6-4 (Ref. 25) states that:

> "The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, <u>the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes</u>.
>
> <u>The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences</u> and assuring emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR Part 100 guideline values. Also the maximum time for valve closure should not exceed five seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Similar concerns apply to vent system designs."

This is interpreted by the writer as specifying that the large 42" purge and vent lines (PVLs) should be closed except in Modes 5 and 6. And if purging is necessary in Modes 1, 2, 3 and 4, then smaller lines (8" and 10") should be considered and the source term to be used for evaluating offsite dose is that of Reg. Guide 1.4 which uses TID 14844 source terms as the fission product available for release to containment.

RG 1.4.C Regulatory Position (Ref. 30) requires the following under related subsection No.:

"1a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides."

i.e., 25% of the radioactive iodine inventory from exhibit 1 is specified to be immediately available inside primary containment for leakage to the atmosphere. For Zion this would represent approximately 25 percent of 15.79×10^7 curies of I-131 EQU in the core i.e., 3.9×10^7 curies immediately available inside containment for leakage to atmosphere.

"1c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account."

With half lives for iodine (I) varying from 3.16×10^3 secs for I-134 to 6.95×10^5 secs for I-131, released immediately on a LOCA, and a time to valve closure of seven (7) seconds, there is no time for significant radioactive decay of any iodine isotope before it is discharged to atmosphere.

It is to be noted that the actual first stage of fission product release during a LOCA occurs with the infringement of DNBR for the fuel rod, leading to overheating of the clad and fuel failure according to SRP 4.2 (Ref. 26) by perforation (or loss of hermeticity). For Zion, this is specified to occur 0.1 sec's into the event in the Appendix K evaluation of the LOCA event; the off-site calculations for this submittal have been made for a DNBR infringement of 1/2 sec. and are therefore less conservative.

"1d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis."

During the first 7 seconds, there are no engineered safety features (ESF) fission product clean up systems available for reducing fission product content prior to discharge to the environment. Engineered safety feature containment sprays are initiated after 45 secs. Any filtration systems on the 42" inlet and outlet penetrations are not designed to ESF requirements. Recirculating filter systems provided by W for fission product control of containment atmosphere during normal operations are not ESF equipment.

Containment volume of 2 million cubic feet originally containing 144,000 lbs of air reduces fission product discharged from the RCS by prior dilution through mixing. Exhibits 3 and 4, and 3A and 4A show the circumstances for containment and the discharging reactor coolant system.

The containment has an initial mass of air of 144,000 lbs (at atmospheric pressure). On a LOCA, the initial rate of discharge from the RCS into containment is 75,000 lbs/sec and over a period of seven (7) seconds prior to containment valve closure, a total of 270,000 lbs is so discharged. This increases total mass in containment to 420,000 lbs, increasing total pressure in containment to 23.7 psig; at the same time a total mass of 15,000 lbs [valves fully open] to 2,860 lbs (valves partly open) of mixed containment inventory is discharged to the atmosphere.

If it is assumed that all fission product released from the core is immediately available to containment as in RG 1.4, then total mixing of this product should be assumed to occur on initiation of the LOCA. (The data presented show the results for a release ½ second after the LOCA, but the differences are not significant for the intent of this submittal.) As a result, containment inventory discharged contains a uniform concentration of a decreasing curie content over the first 7 seconds, and the net result is a release to outside containment of 4.38% of the source term fission product inventory Q, released from the core on occurrence of the LOCA. (A reduced amount of 1.57% is released for partly closed valves). Exhibit 2A shows that for the RG 1.4 source term, this gives a total release from containment over the first 7 seconds of 1.7 x 10^6 curies direct to atmosphere. Related offsite dose is 490,000 rem for 2 x fully open valves. Partially open valves reduce this to the value shown in Exhibit 2 of 612,000 curies and 156,000 rem.

It should be recognized that the thermal-hydraulic, including energy conditions, are such that fluid is discharging from both the RCS and the containment at very high energy levels, with associated pressure levels giving sonic discharge velocities into containment of the order of 1000 fps. Under these conditions it takes only hundredths of a seconds for RCS fluid to reach the containment isolation valves from the RCS system. This is no comparison with the very low transport rates from the top of a fuel pool to containment isolation valves for a fuel handling accident inside containment as discussed in Section 1.3.3.5 of this submittal; values of up to 15 secs. have been considered appropriate for these circumstances.

If is assumed that the core fission product source term is instead uniformly mixed with the RCS fluid prior to its dicharge to containment, (less conservative than R.G. 1.4) curie content discharged to atmosphere is reduced from 4.38% Q to 1.9% Q where Q is the total term source released from the core by the LOCA and related source terms and related offsite doses are reduced by the same amount.

These are not unrealistic assumptions, for conservative purposes. The LOCA causes sudden pressure drops in the RCS, to saturation pressures for the prevailing temperatures of the RCS inventory, causing steam release from violent boiling throughout the system. This would cause substantial vibration of the fuel rods and movement of the prevailing damaged UO_2 pellets, facilitating the mass transfer of fission product gases to and through the gap to the locally faulted cladding, followed by blowdown through the clad defects at high rates because of the prevailing pressure drops, between the gap and the core.

Over the first seven seconds of the event, heat is being tranferrred from the core to containment by steam formation at the core and subsequent mass transfer to the RCS system and break, and discharge to the containment, at the very high rates discussed earlier in this subsection. Since fission product gases are released from the cladding, (and probably at the hottest sections) the transport of fission products released from the gap would be within the same steam and entrained liquid transport system to the break and then containment.

Within containment, unless special provisions have been made, there is no guarantee that a certain percentage of high concentrations of fission product inventory being released by RCS discharge is not being bypassed directly to the open containment isolation valves from its main path to principal containment volume. In this sense, assuming an immediate release of all fission product to the containment on DNBR would help offset the potential non-conservatism of this bypass.

"1e. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours [0.1 percent per day], and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing."

Except for dilution through mixing discussed under 1d above, there is complete bypass of containment for 7 secs through the 2 x 42" open valves.

The magnitude of discharge to the environment with related offsite doses has been discussed under 1d above. In reviewing these figures, it should be recognized that for a normal leakage of 0.1%/day from containment, 8×10^{-6}% of Containment Inventory (Q), would be released in the same time frame of 7 seconds. When compared with 4.38%, this represents a dose reduction factor of 541,000 and would reduce the 7 second dose from 489,000 rem to 0.9 rem.

Over a two hour time frame, and making allowance for 38 seconds without spray, followed by an iodine removal coefficient of 54/hr with a maximum reduction factor of 100, gives an approximate reduction in discharge by a factor of 32,000 leading to a calculated dose of 15 rem.

These reduction factors in offsite dose of 489,000 for the first seven seconds by effective early containment at 0.1%/day, and of 32,000 in the first 2 hours by effective containment at 0.1% per day and an iodine cleanup factor of 100, manifest the real significance of effective containment and containment spray in fission product containment.

1.3 LOCA: BTP CSB 6-4, B5 Criteria

The Reg. 1.4 source terms of 1.2 above, are based upon the Regulatory requirement of 10 CFR 100.11, (a) footnote 1 (Ref. 36) that:

"The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

However, Branch Technical Position CSB 6-4 (Ref. 25) provides another basis to justify containment purge design and which is less conservative than the Regulatory position. This is given in related section B-5, as:

"5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values."

To gain further regulatory interpretation of the meaning of fuel failure within this context, the writer's DPV (Ref. 42) refers to SRP 4.2 FUEL SYSTEM DESIGN, I (AREAS OF REVIEW), 2nd para. (Ref. 26) which states that, in respect of postulated accidents:

"The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 38), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents."

The underscored lines show that fue rod failure in the context of this paragraph must be evaluated for postulated accidents and this evaluation must be conservative. Fuel Rod Failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has therefore been breached; these failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 36) for postulated accidents.

Coolability is addressed as a separate criterion.

1.3.1 Characteristics of Fuel Failure Giving Fission Product Release During Postulated Accidents

Regulatory clarification of fuel rod failure is given in SRP 4.2.II.A.2. (Ref 26) This is abstracted as follows for the circumstances of postulated accidents in particular:

"2. FUEL ROD FAILURE

This subsection applies to [normal-operation,-anticipated-operational occurrences,-and] postulated-accidents. [Paragraphs-(a)-through-(c)-address

~~failure mechanisms that are more limiting during normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.~~] Paragraphs (d) through (h) address failure mechanisms that are more limiting during (anticipated operational occurrences and) postulated accidents, [~~and the information to be reviewed will usually be contained in Chapter 15 of the Safety Analysis Report. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere~~]

To meet the requirements of [~~(a) General Design Criterion 10 as it relates to Specified Acceptable~~ Fuel Design Limits for normal operation, including antici ~~pated operational occurrences, and (b)~~] 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. [~~Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences.~~] Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Fuel failure criteria should address the following to be complete.

Only those failure mechanisms that are more limiting for postulated accidents are abstracted here:

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs [~~and CPR for BWRs)~~] are satisfied. [~~The review of these criteria is detailed in SRP Section 4.4. For normal operation and anticipated operational occurrences, violation of the thermal margin criteria is not permitted.~~] For postulated accidents, the total number of fuel rods that exceed the cri teria has been assumed to fail for radiological dose calculation purposes.

Although a thermal margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: [~~It-has-also-been-traditional-practice-to-assume-that-failure-will-occur-if-centerline-melting-takes-place.--This-analysis-should-be-performed-for-the-maximum-linear-heat-generation-rate-anywhere-in-the-core,-including-all-hot-spots-and-hot-channel-factors,-and-should-account-for-the-effects-of-burnup-and-composition-on-the-melting-point.--For-normal-operation-and-anticipated-operational-occurrences,-centerline-melting-is-not-permitted.~~] For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. [~~The-centerline-melting-criterion-was-established-to-assure-that-axial-or-radial-relocation-of-molten-fuel-would-neither-allow-molten-fuel-to-come-into-contact-with-the-cladding-nor-produce-local-hot-spots.~~] The assumption that centerline melting results in fuel failure is conservative.

(f) Excessive Fuel Enthalpy: [~~For-a-severe-reactivity-initiated-accident-(RIA)-in-a-BWR-at-zero-or-low-power,-fuel-failure-is-assumed-to-occur-if-the-radially-averaged-fuel-rod-enthalpy-is-greater-than-170-cal/g-at-any-axial-location.~~] For full-power RIAs in a BWR and all RIAs in a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 (Ref. 6) as it relates to fuel rod failure. [~~The-170-cal/g-enthalpy-criterion-is-primarily-intended-to-address-cladding-overheating-effects,-but-it-also-indirectly-address-pellet/cladding-interactions-(PCI).~~] Other criteria may be more appropriate for an RIA, but continued approval of [~~this-enthalpy-criterion-and-the~~ thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude

PCI failures. (1) The uniform strain of the cladding should not exceed 1%. [~~In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded.~~] Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

(h) Bursting: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to incidence of rupture during a LOCA, [~~a rupture temperature correlation must be used in the LOCA-ECCS analysis.~~] Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. [~~Although fuel suppliers may use different rupture temperature vs differential pressure curves, an acceptable curve should be similar to the one described in Ref. 10.~~]

(i) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from seismic and LOCA analysis (Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events."

Summary:

Failure Mechanisms include:

(a) Infringment of DNBR criteria during postulated accidents which causes overheating of the cladding of the fuel rod, and is assumed to cause failure of the clad, and release of contained fission products from the gap as a source term for the calculation of radiological doses.

(b) If postulated accident conditions cause calculated values of fuel pellet temperature to reach the melting point for the uranium dioxide at the centerline of the pellet, it is assumed that all such rods shall fail (and release fission products from the pellets - as well as the gap) for the calculation of radiological doses.

1.3.2 Characteristics of Fission Product Released From Failed Fuel During Postulated Accidents

1.3.2.1 General

Fission product release as source terms for postulated accidents relevant to the above fuel failure criteria are specified as:

SRP 4.2, Section I, last paragraph (Ref. 26) states that:

> "All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in SRP Section 4.4. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Evaluation Branch for use in estimating the radiological consequences of plant releases."

SRP 4.2.C.3(h) (Ref. 26) states that:

> "Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 [~~Refs. 6, 28-30~~] as they relate to fission product release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by CPB [~~see Ref. 31~~]."

The criteria from these Reg Guides are considered separately in the following subsections of this submittal in order to examine for general guidelines which may be applied to BTP CSB 6-4 B5 Criteria.

1.3.2.2 Regulatory Guide (RG) 1.25: Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

RG 1.25 (Ref. 31) covers the Fuel Handling Accident inside containment.

RG 1.25 page 25.1 under Section B, second para. provides for an immediate release of all activity from the fuel rod gap of the damage rods:

> "The number and exposure histories of fuel assemblies assumed to be damaged determine the total amount of radioactive material available for <u>immediate release into the water</u> during a fuel handling accident."

The same Section B, fourth para. provides that:

> "Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation would be available for immediate release into the water in the event of clad damage. (Migration of fission products is a function of several variables including operating temperature, burnup, and isotopic half life taken into consideration in establishing the release fractions listed in this guide.)"

RG 1.25 also assumes that 10% of the total radioactive iodine in the rod (with calculated peak activity) is contained in the gap for release. (See page 25.2, Item C.1.d):

> "All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident.

Released iodine rises to the surface of the related pool with a decontamination factor of 100, provided a minimum depth of 25 ft exists, and gap pressure is no greater than 1200 psig. Subsequent treatment of the source term is typified by the guidelines of SRP 15.7.4 Radiological Consequences of Fuel Handling Accidents (Ref. 28) which requires (under Section III.4, second and third para's that:

> "The reviewer should assess the time required to isolate the containment. This should include the instrument line sampling time (where appropriate), detector response time and containment purge isolation valve actuation and closure time. The containment is considered isolated only when the purge isolation valves are fully closed. The applicant's analysis should be reviewed regarding the travel time of any activity release starting from its release point above the refueling cavity or transfer canal and including travel time in ducts or ventilation systems up to the inner containment purge isolation valve."
>
> "The time required for the release to reach the inner isolation valve is compared to the time required to isolate the containment. If the time required for the release to reach the isolation valve is longer than the time required to isolate containment, then essentially no release to the atmosphere occurs, and the reviewer's assessment should reflect this. If the time required for the release to reach the isolation valve is less than that required to isolate containment, and no mixing or dilution credit can be given, the reviewer should assume that the entire activity release escapes from the containment in evaluating the consequences. Claims for credit for dilution or mixing of a release due to natural or forced convection inside containment are reviewed and assessed. References [4] and [5] should be consulted and used by the reviewer for guidance in estimating dilution and mixing. Where mixing and dilution can be demonstrated within containment, the radiological consequences will be reduced by the degree of mixing and dilution occurring prior to containment isolation."

Related references [4] and [5] are:

"4. Evaluation of Fission Product Release and Transport for a Fuel Handling Accident by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, revised October 5, 1971.

5. Industrial Ventilation/A Manual of Recommended Practice - American Conference of Governmental Industrial Hygienists."

These circumstances relate to a set of containment environmental conditions in which mixing energy is virtually absent, being provided by low energy containment purge and exhaust ventilation fans, and virtually no additional energy from the very small mass of fission product gas released from the damaged fuel elements, after travelling through a minimum depth of 23 ft. Under certain conditions, this could provide for the total activity released (after decontamination in the pool) to be discharged directly to atmosphere outside containment.

For Zion, the fundamental set of values for the thermal hydraulic parameters covering the above circumstances, are completely different to those governing the release and disbursement of fission products to the environment from a LOCA.

1.3.2.3 Regulatory Guide 1.77: Assumptions Used for Evaluating a Control Rod Ejection Accident For Pressurized Water Reactors

Fundamentally, this Guide provides for an evaluation of the Thermal Hydraulic and Power conditions within the core, during the accident, to determine a) the extent of DNBR infringement and b) the amount of fuel exceeding the initiation temperature of fuel melt (approximately 5150°F).

For Source Terms, RG 1.77, Appendix B1 (Ref. 32) proposes that:

"a. The case resulting in the largest source term should be selected for evaluation.

b. The nuclide inventory in the fuel elements potentially breached should be calculated, and it should be assumed that all gaseous constituents in the fuel-clad gaps are released.

c. The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life, assuming continuous maximum full power operation.

d. No allowance should be given for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

e. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature for fuel melting (typically 2842°C) at any time during the course of the accident should be calculated, and 100% of the noble gases and 25% of the iodine contained in this fraction should be assumed to be available for release from the containment."

Summarily: The source term from molten fuel is the same as for RG 1.4. The source term release from the gap is the same as for the fuel handling accident.

The subsequent effects of the release path on the ultimate source terms from containment are evaluated for each of two release paths, as if the other did not exist. These release paths are:

(1) By effectively immediate release of all source terms to containment to be followed by the following cleanup and decay provisions which are the same as those normally accounted for in a LOCA in RG 1.4 (Ref. 30). RG 1.77, App. B1 (Ref. 32) provides that:

"f. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

g. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on a case-by-case basis.

h. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing."

Additionally SRP 15.4.8, Section III.3 (Ref. 27), further specifies that:

"For releases via the containment building, 100% of the noble gases and 25% of the iodines contained in the fuel which is estimated to reach initiation of melting are assumed to be available for release from the containment."

Summarily: For the release path to containment, these are effectively the provisions of RG 1.4 in respect of the treatment of Fission Product Source Terms after release from the core.

(2) By release of fission products to the secondary system as per RG 1.77, Appendix B, Items 1i, j and k (Ref. 32).

There are not considered in this submittal, as they do not apply to a release to containment.

1.3.2.4 Summary (of General Positions on Fission Product Releases Deriving from RG 1.25 and 1.77)

(a) For failure of fuel cladding by either DNBR infringement or fuel handling accident:

For iodine, 10% of the fuel rod inventory is released from the gap. For the control rod ejection accident this release is assumed to be available immediately inside containment for leakage.

(b) For failure by centerline melting of the fuel pellet:

25% of the iodine inventory of any fuel rod which reaches or exceeds the initiation temperature of fuel melting is assumed to be immediately available inside containment for release. This is the same assumption applied in RG 1.4 for fuel melt deriving from a LOCA.

1.4 LOCA: BTP CSB 6-4/B5 Criteria: Application to Zion

Zion Fuel temperatures during normal operation at maximum power prior to a LOCA very from 2500°F to 4100° for approximately 15% of the core (Exhibit 23). There will be a substantial increase in temperature of the whole core over a period of up to 7 seconds following a LOCA and Exhibit 6 shows the related average cladding temperatures. Considering the correlation of fission product release as a function of temperature shown in Exhibit 22, there is a high probability of a substantial increase in fission product activity in the gap over that of the equilibrium activity level represented on Exhibit 1, during these first seven (7) seconds of the accident, so that an increase in gap activity level from the equilibrium values shown in Exhibit 1 to the value of 10% used in the other postulated accidents is not an unreasonably conservative regulatory position to adopt for this event. On this basis, the iodine source term deriving from fuel rod failure by overheating of the fuel cladding by DNB infringement at Zion at 0.1 second into the event would be 157.9×10^5 curies of I-131 EQU and is the value adopted by the writer in conformance to the related BTP. In respect of fuel rod failure by centerline melting, the Zion FSAR (Ref. 33) does not provide detailed information on fuel pellet temperatures except for the general statement that the safety injection system prevents core meltdown Ref. 33, page 14.3-46, Revision 1 second para.; provision for related fission product release from melted fuel rods is therefore not necessary for this evaluation to the guidance of the related BTP.

On the basis of BTP CSB 6-4, B5 therefore, a total iodine fission product release of 157.9×10^5 curie I-131 EQU from the core, would be available to inside containment at 0.1 second into the LOCA. By reference to the conditions inside containment discussed in detail in Section 1.2, items 1d and 1e above, it can be shown that, the release of 157.9×10^5 curies of I-131 EQU from the core as a source term will result in the discharge of 692,000 curies of I-131 EQU to atmosphere with an offsite dose of 176,000 rem with 2 x 42" fully open for 7 seconds, see Exhibit 2A, item 5. With valves partly closed this is reduced to 249,000 curies I-131 EQU and 63,400 rem, see Exhibit 2 item 5.

It is noted that in its recent revision to the FSAR (Ref. 34) page 14.3-38 Revision 1. W has calculated an offsite dose from the LOCA on a non-Reg. Guide 1.4 basis, by also using the entire inventory of fission products contained in the pellet cladding gap, but has assumed the equilibrium values only, as listed in Exhibit 1. This is equal to 24.09×10^5 I-131 EQU which is 1.52% of the core activity as compared with the 10% exemplified in other NRC criteria and used by the writer. Effective doses that would be obtained using equilibrium gap activity only are also presented in Exhibits 2A and 2 under items 4 and show offsite doses to thyroid are reduced to 27,000 rem for 2 fullopen valves and 9,700 rem for 2 partially closed valves.

2 OFFSITE DOSE CONSEQUENCES: SUMMARY

2.1 Basis for Calculations

Based on discussions in section 1, radiological releases and related offsite consequences are shown in Exhibit 2A item 6 for 2 x 42" fully open (90°) valves and Exhibit 2 item 6 for 2 x 42" values at a limited opening of 50°.

All calculations are based on valves closing in 7 seconds from commencement of a LOCA. Doses are based upon valves being in the open position for a full 7 seconds as required by the SRP. Valves will be required by technical specifications to close within seven (7) seconds of commencement of the LOCA.

For the sake of example only, source terms are restricted to iodine in terms of I-131 EQU, and thyroid dose only has been calculated. Dose is calculated at the site boundary (exclusion distance) of 415 meters. Each dose is calculated independently of each other and are to be added to the LOCA leakage dose (over 2 hours) of 123 rem as appropriate.

An additional dose due to RCS inventory discharged into the containment would also need to be added, for all non-RG 1.4 calculations. These are given in Exhibits 2A and 2 under items 2 at 132 rem for 2x fully open valves, and 48 rem for 2 partially opened valves.

For the diffusion coefficient, a value of 5×10^{-4} sec/cm^3 applicable to leakage conditions over a 2 hour period has been used. In fact we have a high energy puff release of 7 seconds giving a potential finite cloud in travel to the enclosure boundary instead of a low leakage release diffusing into a cloud; as a result, the offsite dose under actual conditions is likely to be increased. For the 0-2 hour leakage, the licensee has used a more conservative value than the NRC of 9.2×10^{-4} sec/cm^3 and this would increase dose by a factor of 1.84.

2.2 Offsite Doses

2.2.1 RG 1.4 Source Terms Released Immediately on LOCA

Exhibit 2A, item 6, shows that for fully (90°) open 42" valves, the offsite dose for a RG 1.4 source term is calculated at 489,000 rem. And Exhibit 2, item 6, shows that for partially (50°) open 42" values, these doses are reduced to 156,000 rem.

2.2.2 10% Gap Activity Released on DNBR

Exhibit 2A (item 5) shows offsite doses reduced to 176,000 rem for fully open valves, and Exhibit 2 (item 5) shows reduction to 63,000 rem for partially open valves.

2.2.3 Equilibrium Gap Activity Released on DNBR

Exhibit 2A (item 4) shows offsite dose is reduced to 27,000 rem for fully open valves and Exhibit 2 (item 4) shows reduction to 9,700 rem for valves partially open.

2.2.4 RCS @ 60 μc/gm Activity; All Released To Containment Immediately On A LOCA.

Exhibit 2A (item 2) shows offsite dose contribution is 132 rem for fully open valves and Exhibit 2 (item 2) shows a reduction to 48 rem for partially open valves.

This activity release is equivalent to DNBR infringement of only .08% of the fuel in the core. based on Gap Activity [illegible]

2.2.5 RCS @ 60 μc/gm Activity; Released Progressively To Containment On RCS Discharge From A LOCA

Exhibit 2A (item 3) shows offsite dose contribution is 58 rem and Exhibit 2 (item 3) shows a reduction to 21 rem for partially open valves.

2.2 Conclusions

(1) According to Reg. Guide 1.4 criteria the offsite doses are completely unacceptable.

(2) LOCA calculations for Zion show no fuel melt; however, for DNBR infringement only, an evaluation of offsite dose based on release of 10% gap activity from 100% fuel still shows completely unacceptable circumstances.

Although this is in conformance with SRP 6-4, BTP, CSB B5 criteria, it is not in conformance with 10 CFR 100.11 (a) footnote 1 requirements which states that:

> "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

(3) Partially closing the valve to 50° from 90° is not successful in reducing the offsite dose to acceptable values.

(4) With valves partially open at 50°; fuel failures by DNBR infringement on a LOCA would have to be limited to 0.2% of the core to limit total doses to 10 CFR 100 limits.

3 APPENDIX K EVALUATIONS, FUEL FAILURE, AND FISSION PRODUCT RELEASE

10 CFR 50.46 (Ref. 37), acceptance criteria for emergency core cooling system for light water nuclear power reactors, requires that during a LOCA, cladding temperatures, cladding oxidation, and hydrogen generation, are limited and such that the core remains amenable to cooling in the short run from the initial break through reflood, and also for long term post accident cooling.

10 CFR 50.46 does not include a requirement to evaluate the earliest time at which fission products could be released by local failure of the fuel cladding as fuel rod conditions rapidly change, challenge and exceed the limiting features of design which ensures fuel clad (and rod integrity) under Normal Operating Conditions and Transient Occurrences. These limiting features are described as specified acceptable fuel design limits (SAFDLs) and are required under 10 CFR Part 50, Appendix A, Criterion 10.

A principal feature of the Appendix K evaluation is that it is designed to identify that rupture which causes a maximum post rupture cladding temperature within the fuel assembly being evaluated; and it is this time to rupture which is reported in the FSAR. The Appendix K evaluation is not designed to report the earliest rupture that can occur.

3.1 Preliminary

In evaluating 10 CFR 50.46 criteria through the use of the Appendix K evaluation model (Ref. 39), licensees are required to undertake a detailed evaluation of the items to be discussed below throughout the complete LOCA event, i.e., from time 0 through 50-60 seconds, to determine that the clad rupture meeting the Appendix K criteria does not occur in the first 10-15 seconds of the event, and which is the region of special interest for this review. In the time available for this research, a search of the UFSAR and the related reference material on the docket does not disclose many of essential the details of this calculation (Ref's 1-24). We therefore use the limited information available to draw conclusions.

3.2 Review

Appendix K calculations are undertaken on that fuel element assembly which ultimately provides the maximum clad temperature after (post) clad rupture.

Generic work by W (Ref. 17) proposes that maximum calculated temperatures (post rupture) occurs in the low burn up (third region) fuel assembly.

Exhibit 6 shows the average clad temperatures deriving from Appendix K calculations from the Zion FSAR, Figure 14 F. 2-19a, (Ref. 33). This shows that on infringement of DNBR at 1/10 second, average clad temperature increases very rapidly from a normal operating value of 720°F to at least 1350°F, and then to 1750°F, over a total period of seven seconds; thereafter temperature reduces rapidly to 1000°F at about 15 secs. from which it sharply increases ultimately to approx 2200°F.

Exhibit 10 shows that W fuels are designed to require a yield strength of 45,000 psi a minimum for normal operations, and an ultimate tensile strength of 57,000 psi as a damage limit, as specified acceptable fuel design limits (SAFDL). Exhibit 11 shows that as temperatures increase above 850°F, the available mechanical properties can be reduced below both these limits so that fuel clad cannot therefore be considered reliable in terms of protection against fission product release.

Exhibit 10 also shows that W fuels require a design limit of 1% on cladding strain as a design limit, and 1.7% as a damage limit. The work of this Section 3 will show how both these limits can be exceeded inside the seven seconds on infringement of DNBR during the course of a LOCA, so that again, fuel clad cannot be considered reliable in terms of protection against fission product release.

Exhibit 15, shows how a temperature range of 1350°-1750°F traverses a range of Zircalloy metallurgical phases (transitions), α to ($\alpha + \beta$) to β phases, during which ys = UTS and structural stability under stress is dependent upon mechanical/strength properties which are a function of temperature and related time and stress at temperature. Under the circumstance of the transient expected from Appendix K calculations with rapid changes of both temperature and stress, their is a need for empirical tests to determine swelling and burst (rupture) characteristics under these same dynamic conditions. Exhibit 15 represents results from such a series of tests (Ref. 13).

Such conditions are also represented in Exhibit 16 for Engineering Hoop Stress and temperature at rupture, for particular heating rates, and in conjunction with the information in Exhibit 20 on related rates of circumferential strain on rupture, at the given rupture temperatures.

What are the expected operating pressure differentials across the clad under these LOCA conditions:

Reference information shows that internal clad pressure under normally operating conditions is of the order of 1400 psig for new fuel and expected to increase to 2250 psig at the end of the 3rd cycle (for the fuel). On this basis, we evaluate a gap pressure of 1500 psig at approximately 1/3 burnup into the first cycle, at which burnup maximum calculated clad temperatures are expected on a LOCA.

It is proposed that, immediately on a LOCA as clad temperature increases to 1350°F, gap pressure will increase by 20%, to 1800 psig. Exhibit 12 shows that at this time, core pressure has reduced to 1500 psig giving a pressure drop across the clad of 300 psi which according to Exhibit 13 will give a hoop stress of approximately 2460 psi.

At 7 seconds into the event, clad temperature has increased further to 1750°F, a total increase of 1030°F from the normal operating condition. From this, it can be proposed that gap pressure for the complete rod can increase by 36% over its normal operating value to 2100 psig. Exhibit 12 shows that at this time, core pressure has reduced to 950 psig so that the pressure drop across the clad is now 2100-950 i.e., 1150 psi which according to Exhibit 13 will give a hoop stress of 9400 psi.

When the above values of pressure and temperature are plotted on a particular Hoop Stress vs Burst Temp curve (Exhibit 14) from reference 1, at one sec the clad does not rupture, but at seven seconds the clad is well into the rupture regime.

In its calculation of clad strain during Appendix K calculations, W uses results from tests by Hardy (Ref. 13). Exhibit 15 is a set of results from one such test at 100°C/sec heat up rate (the heat up rate between 720°F and 1750°F in 7 seconds = 150F°/second [or 84C°/second]). This exhibit shows that these Appendix K values over the first 7 seconds bracket the range from zero (0) expansion at 1350°F to the burst regime at 1750°F. In respect to these values, W has assumed that if clad strain reaches 10%, the clad will rupture; see Exhibit 18 from Ref. 3. Note that the SAFDLs of 1% and 1.7% on cladding strain can both be exceeded in the first seven seconds of DNBR infringement in the course of the LOCA.

The NRC, in its clad strain and rupture models uses the data shown in Exhibit 16 to determine when rupture is likely to occur for given rates of increase in temperature. It is proposed by the NRC that the 28°C/S (=50F°/second) test points apply also to larger values (of rate of temperature increase). Exhibit 16 shows that the Appendix K values again bracket the complete set of experimental data and significantly at the higher temperatures of the transient.

Exhibit 20 shows the circumferential strain that can occur at given rupture temperatures, and the curve proposed by the NRC for Appendix K calculations. Prime facie; maximum strain gives maximum blockage leading to maximum calculated temperatures for cladding after the burst. In fact, W has established that maximum post rupture cladding temperature does not necessarily occur with a maximum circumferential strain at rupture, due apparently to direct radiation influences from fuel rods exposed by rupture at lesser values. Providing rupture is expected by the data of Exhibit 16, the related strain is to be given by the NRC curve on Exhibit 20 (or lesser value giving maximum temperature). It should be noted that with this information there would be a very high probability of rupture at 1750°F down to 1500°F, with the probability decreasing, but still present at lower temperature.

Note that Exhibits 16 and 20 do show that fuel temperatures and pressures could rupture the cladding over a whole range of conditions. However, the purpose of the Appendix K evaluation is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirements of 10 CFR 50.46 and for this end, it models, and uses factors, to conservatively calculate values for the related parameters. Its purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released. An example can be seen from Exhibit 16. The test points can show marked deviations from what are apparently best estimate curves for the various rates of temperature increase. For conservatism in estimating the first occurrence of fuel rupture, one would have presumed the use of a boundary curve at the lower temperatures and pressures of each heating rate and Exhibit 20 would not have been required.

Note that Exhibit 15 does show that even though rupture may not occur with a detailed re-evaluation, cladding strain is most likely to exceed the 1% strain used by W (Ref 33, P. 3.2-39) as a SAFDL to meet the regulatory requirements of Ref. 38.

The writer would be concerned about the relevance of the hoop stress, strain/rupture data of Exhibits 16 and 20 to the power generation and heat transfer conditions inside a reactor. These tests were done on electrically resistance heated cladding tubes. They do not simulate the heat transfer from central fuel rod pellets at high temperatures through a realistic gas gap of varying geometry, fuel pellet-clad contact, and pellet fracture/fragmentation to a cladding which is 12 ft long and which is likely to have a much smaller ratio of rupture length to clad length and gap volume than the test specimens. The most revealing feature of Exhibit 16 is the data from the only test undertaken under much more realistic conditions, on a nuclear fuel rod using Zircalloy cladding in the TREAT reactor at ORNL; this information shows ruptures at very much reduced stress levels than the rest of the data.

3.3 Summary

1. Conditions within the core as currently evaluated by the Appendix K model, show that over the first seven (7) seconds following a LOCA, the following significant events occur:

1.1 DNBR for the whole core is infringed at 1/10 sec requiring gap activity at 10% core inventory for the whole core to be assumed as a source inside containment.

1.2 The temperature of the fuel clad, and the pressure drops across the same fuel clad, infringe specified acceptable fuel design limits (SADL) for normal operation and operational occurrences, required by 10 CFR 50 Appendix A, Criterion 10. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

1.3 The temperature of the fuel clad and the related pressure drops show conditions in which substantial deformation of the fuel clad by strain, can exceed the design and damage SAFDL values for cladding strain. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

1.4 The temperature of the fuel clad and the related pressure drops show conditions which could result in fuel rupture. This conclusion would need to be subject to detailed verification using the Appendix K model.

1.5 For Zion, fuel rods do not reach the melting point of the fuel pellets so that under minimum engineered safeguard conditions, additional fission product release from the fuel rods would not occur.

2. The writer proposes that the purpose of Appendix K is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirement of 10 CFR 50.46 and for this end it models, and uses factors, to calculate values for the related purposes. The purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released from the fuel summary a LOCA.

4 CONCLUSIONS

1. Conditions within the core as currently evaluated by the Appendix K model, show that over the first seven (7) seconds following a LOCA, the following significant events occur:

   1.1 DNBR for the whole core is infringed at 1/10 sec requiring gap activity at 10% core inventory for the whole core to be assumed as a source inside containment.

   1.2 The temperature of the fuel clad, and the pressure drops across the same fuel clad, infringe specified acceptable fuel design limits (SADL) for normal operation and operational occurrences, required by 10 CFR 50 Appendix A, Criterion 10. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

   1.3 The temperature of the fuel clad and the related pressure drops show conditions in which substantial deformation of the fuel clad by strain, can exceed the design and damage SAFDL values for cladding strain. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

   1.4 The temperature of the fuel clad and the related pressure drops show conditions which could result in fuel rupture. This conclusion would need to be subject to detailed verification using the Appendix K model.

   1.5 For Zion, fuel rods do not reach the melting point of the fuel pellets so that under minimum engineered safeguard conditions, additional fission product release from the fuel rods would not occur.

2. The writer proposes that the purpose of Appendix K is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirement of 10 CFR 50.46 and for this end it models, and uses factors, to calculate values for the related purposes.

The purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released from the fuel summary a LOCA.

3. As a result of the above

3.1 Fission product release from the fuel gap is a realistic consideration over the first seven seconds and prudent conservatism at this time should consider release from the whole core.

3.2 Reg Guide 1.4 deriving from Regulatory Requirement 10 CFR 100 requires consideration of substantial molten fuel as a design for the source term.

4. The writer proposes that Regulatory philosophy recognized the possibility of Beyond Design Basis Events as the realism of a substantial commercial industry and therefore required protection against this occurrence and made provision in the Regulations for this purpose.

Considering the energy exchanges occurring in the core, and the insight of the Appendix K evaluations, it is not difficult to foresee significant fuel melt with potential additional substantive release of fission products from the fuel pellets over this time frame. The question of the separate consideration of the timing of this additional contribution to the source term inside containment however must be moot. Uncontrollable release through open 42 inch CIVs is out of the question so that steps taken to correct that problem by effective isolation do resolve the unanswered philosophical question as to when fission products released by fuel melt should be more realistically and conservatively established.

4.1 A review of available fuel failure criteria, and the thermal-hydraulics aspects of the movement of fission gases from the clad to the environment over the first seven seconds of the event shows that:

(a) The assumption of an immediate release to the containment is the only available conservative basis for use at this time, and that

(b) The physics of the large energy releases from the core clad through the RCS to containment, and through the open isolation valves, shows effective mass transfer of fission product release from the clad to the environment within the same (7) secs.

5. Fully open purge valves for a period of seven (7) secs. discharge 1.7×10^6 curies of I^{131} EQU to the environment giving an offsite dose of 489,000 rem to thyroid.

   An isolated containment leaking at the safety analyses and TS limit of 0.1% over 24 hrs, releases 3.14 curies of I^{131} EQU over the same seven seconds with a contribution to offsite dose of 0.9 rem.

   The effectiveness of containment isolation and effective leak tightness in achieving a clean up factor of 541,000 over the first seven seconds of the LOCA is manifest.

6. The offsite dose to thyroid for fully (90°) open 42" valves using RG 1.4 source terms is calculated at 489,000 rem. For partially (50°) open 42" valves, these doses are reduced to 156,000 rem. Reduction of source terms from RG 1.4 to 10% gap activity released on DNBR infringement reduces offsite dose to 176,000 rem for fully open valves with a reduction to 63,000 rem for partially open valves.

   Since the allowable limit for thyroid under 10 CFR 100 is 300 rem for 2 hrs at the Exclusion Boundary, these circumstances are unacceptable. Therefore the 42" valves at Zion 1 and 2 should remain closed in Operational Modes 1, 2, 3 and 4.

7. The stress/temperature relationships used to calculate fuel clad rupture to 10 CFR 50.46 are derived from test environments which are substantively non-realistic when compared with actual fuel rod conditions in a reactor

during a LOCA. The only in-reactor tests known to the writer at this time with the closest simulation of a real fuel condition gives ruptures at very much reduced pressures for given rupture temperatures. This comparison needs to be revisited to more thoroughly evaluate the reasons for the differences and thereby improve our detailed knowledge of the total heat transfer environment which can lead to improvements in the calculational models of the fuel assemblies used in the Appendix K evaluations. This can help in a improved definition of the limiting features of the circumstances and lead to ways and means of improving fuel clad design and performance for these circumstances.

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4. Letter from E. P. Rahe (W) to M. Lauben: Subject: Millstone Large Break Results for Increasing Fuel Burnup. July 20, 1981.

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EXHIBITS

OF

BACKGROUND INFORMATION RELATED TO

DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance of SER to Zion 1/2 allowing full power operation with open 42" containment isolation valves.

b) Methodology used for calculating related offsite doses.

ZION

CORE AND GAP ACTIVITIES (IODINE ONLY)

Assumptions: Operation at 3391 MWt for 500 days

Isotope	Curies in the Core ($X 10^7$)	I 131 EQU $x 10^7$	Equilibrium Percent of Core Activity in the Gap	Curies in the Gap ($X 10^5$)	I 131 EQU ($X10^5$)
I-131	8.35	8.35	2.3	19.2	19.2
I-132	12.75	.46	0.26	3.3	.12
I-133	19.09	5.16	0.79	15.1	4.08
I-134	23.01	.39	0.16	3.8	.06
I-135	17.05	1.43	0.43	7.5	.63
		15.79			24.09

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES OPEN 50°

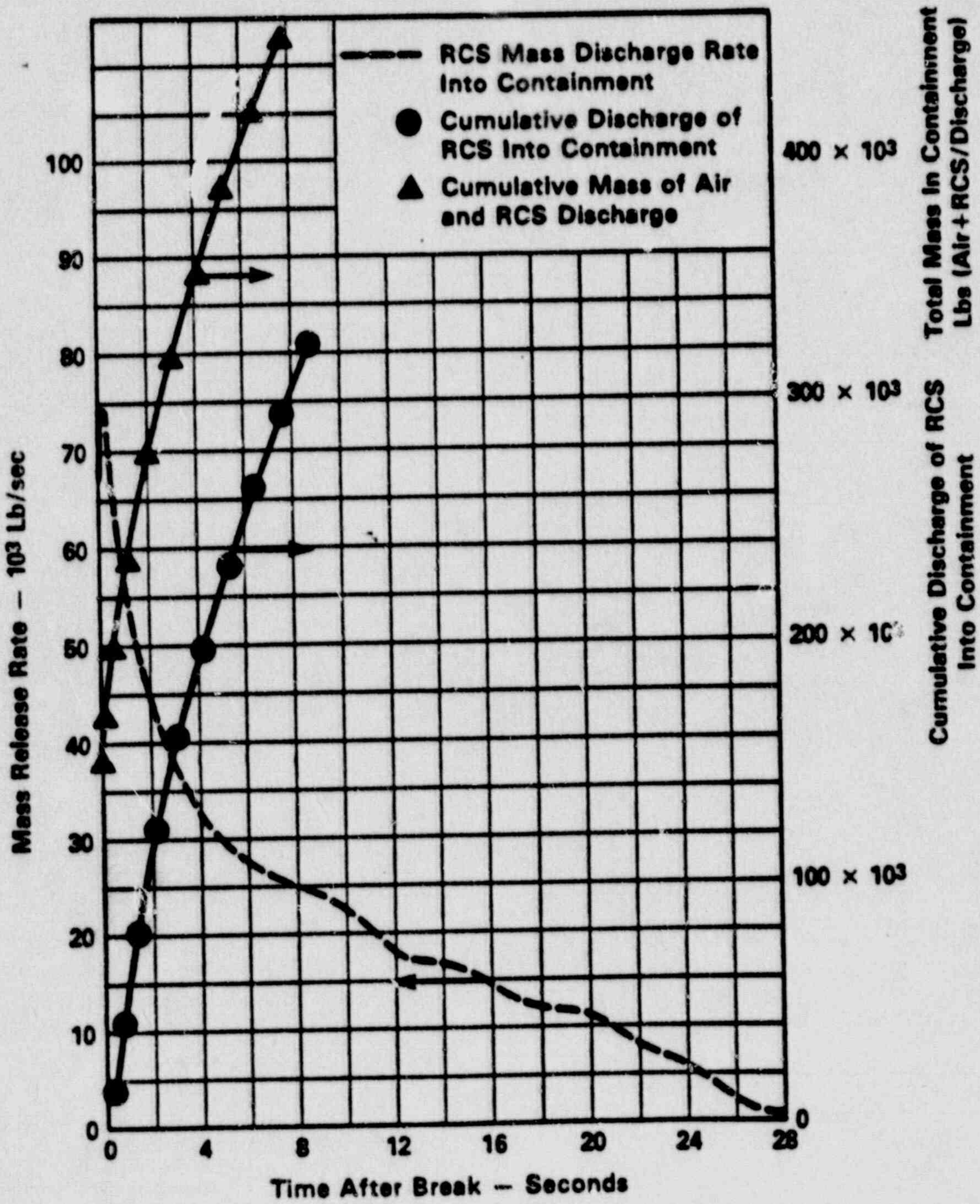
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Item No.	Source	Radiological Sources	Curies Discharged I 131 EQ	Site/Excl. Boundary Dose (Thyroid (REM)
1	Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	73.5	18.7
2	RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119×10^5 curies]	188	48
3	RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	82	21
4	RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	38,000	9676
5	RL	I 131 EQ; SRP Gap activity at 10% Total Activity (SRP calc.) [157.9×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	248,950	63,400
6	RL	I 131 EQ; Reg. Guide 1.4 at 25% Total Activity [390×10^5 curies of I 131 EQ into cont. on LOCA]	611,500	155,700

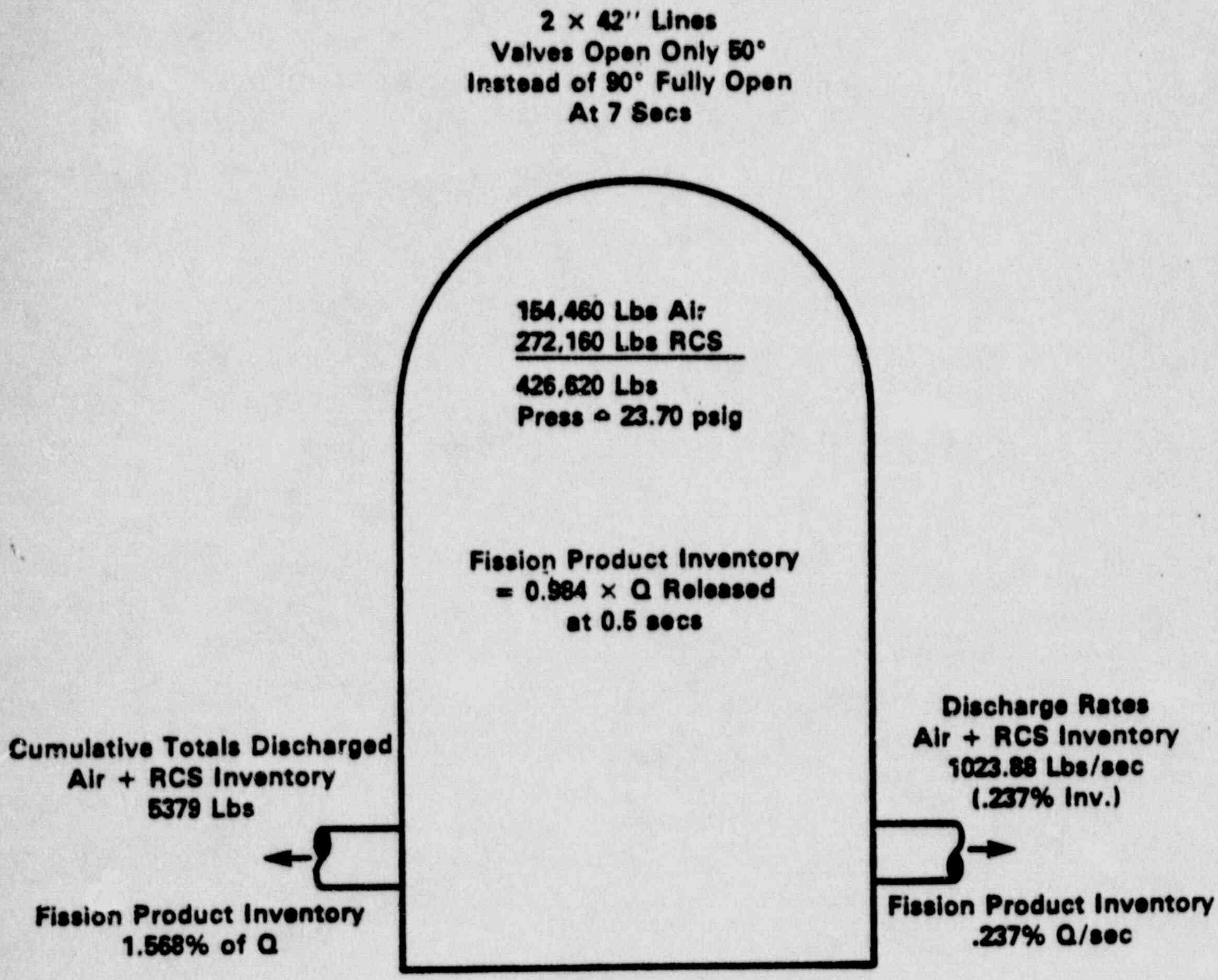
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES
DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON
FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines.
Valves open 50°

Given Q = total inventory of fission products in RCS at T=0.5 secs after LOCA

- If Q is released instantaneously to the total containment volume:

  Fission product inventory discharged outside containment over 7 secs = 1.568% Q

- If Q is released over time with RCS inventory and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 0.561% Q

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES FULLY OPEN (90°)

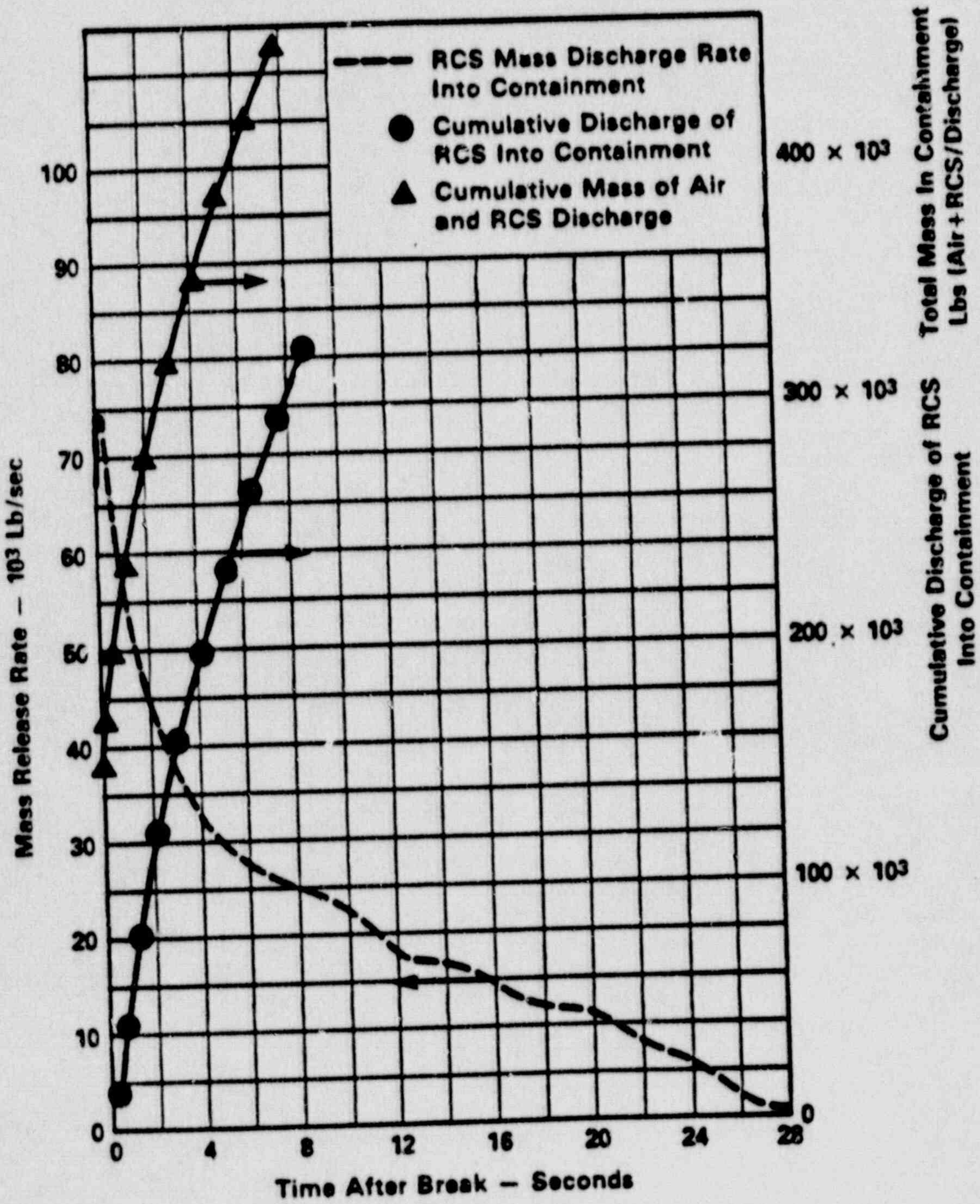
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Item No.	Source	Radiological Sources	Curies Discharged I 131 EQ	Site/Excl. Boundary Dose (Thyroid) (REM)
1	Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	204.3	52
2	RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119 x 10^5 curies]	522	132
3	RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	227	58
4	RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	105,600	26,878
5	RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	691,520	176,010
6	RL	I 131 EQ; Reg. Guide 1.4 at 25% Total Activity [390 x 10^5 curies of I 131 EQ into cont. on LOCA]	1,698,592	488,911

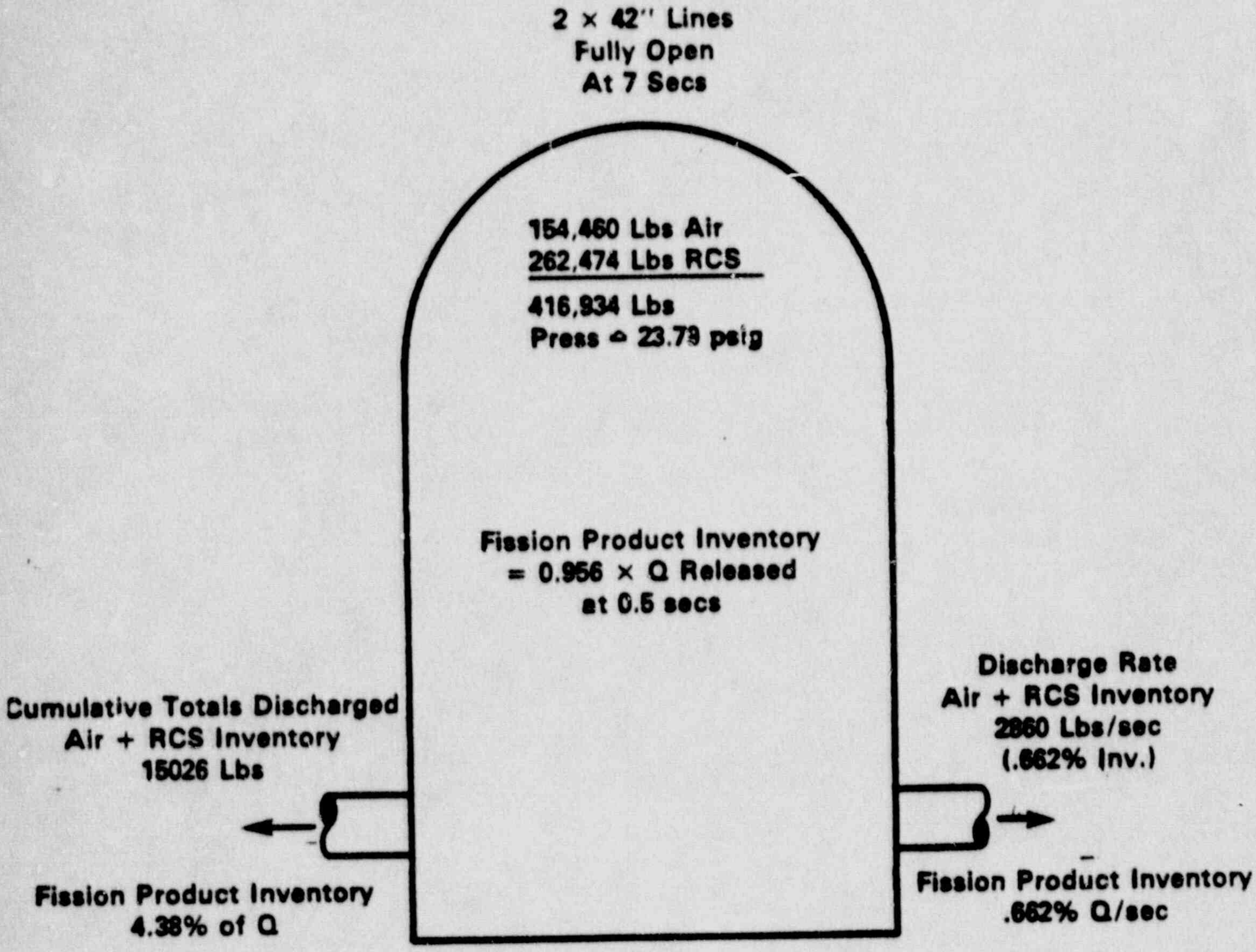
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9 x 10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES
DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

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FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines
fully open (90°).

Given Q = Total inventory of fission products in RCS at T=0.5 sec after LOCA.

- If Q is released instantaneously to the total containment volume

  Fission product inventory discharged outside containment over 7 secs = 4.38% Q

- If Q is released over time with RCS inventory, and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 1.90% Q

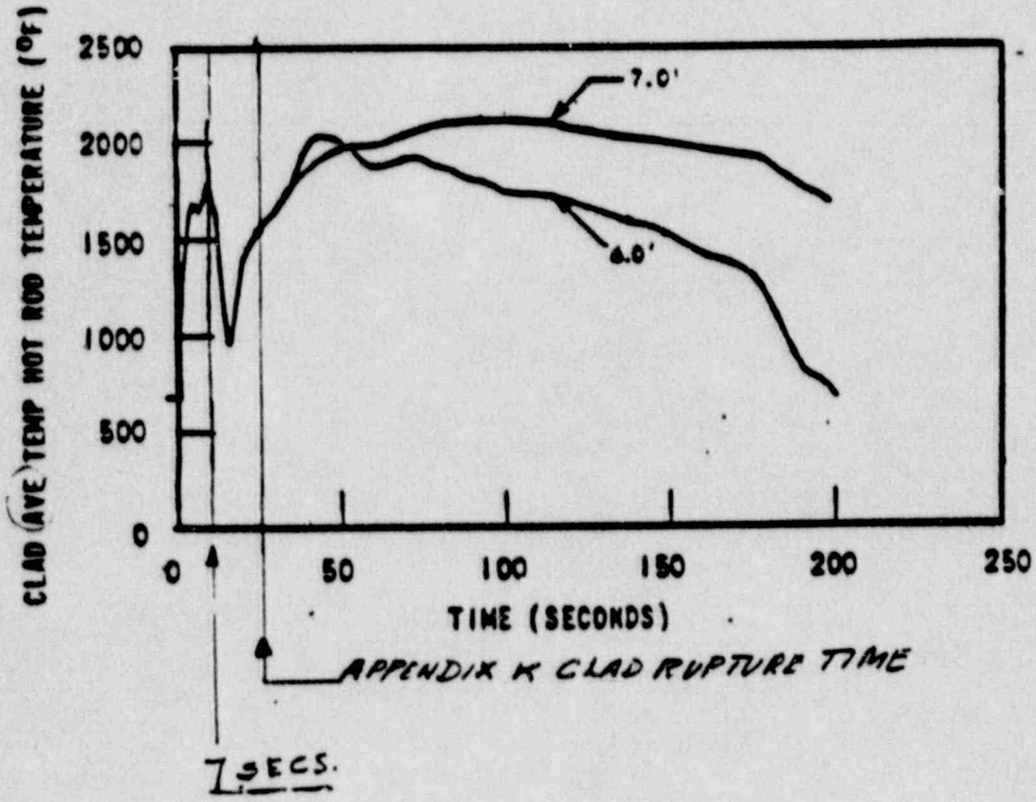

Figure 14 F.2-19a Peak Clad Temperature - DECLG (C_D=1.0)
(Unit 1)

3.1.3.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is [1.30*].

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB). DNB causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength- 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minimum values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerable below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under normal operating conditions is well in excess of the maximum design pressure. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The end of life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history, however it does not exceed the design limit of 2250 psia.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1 [as 720°F].

Previous experience with removable rods has been attained at Saxton, Yankee and Zorita; and additional experience will be acquired at the San Onofre Cycle 2 and Surry Unit 1. Over 300 fuel rods were removed and re-inserted into assemblies during the Saxton re-constitution without evidence of failure. Leak detection tests were performed on the assemblies after all rods were re-inserted, and no leakage was detected. An equally large number of Saxton rods have been successfully removed, examined and re-inserted into over 12 3x3 subassemblies at Saxton. In addition, 28 full length Yankee rods were removed, examined and re-inserted into Yankee Core V special assemblies. Similiar handling of 22 removable rods was sucessfully completed during the first Zorita refueling. All such fuel handlings have been done routinely and without difficulty.

The same fuel rod design limits indicated in section 3.2.3 fuel temperature and internal pressure, are maintained for these removable rods and there is no reduction in margin to DNB. Their inclusion in the initial Zion Unit 1 core loading introduces no additional safety considerations and in no way changes the safeguard analyses and related engineering information presented in previously submitted material in support of the license application.

3.2.3.5 Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods is calculated by a code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 per cent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests. See additional details in Section 3.3.3.1.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature and fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of high burnup uranium dioxide** fuel element behavior indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considers the effect of burnup, temperature distribution, and internal voids. It is an empirical model which has been checked with data from Bettis, Yankee, CVTR, Saxton and others. The pellet densities for the three regions are listed in Table 3.2.3-1.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength- 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minmium values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under normal operating conditions is well in excess of the maximum design pressure. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The end of life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history, however it does not exceed the design limit of 2250 psia.

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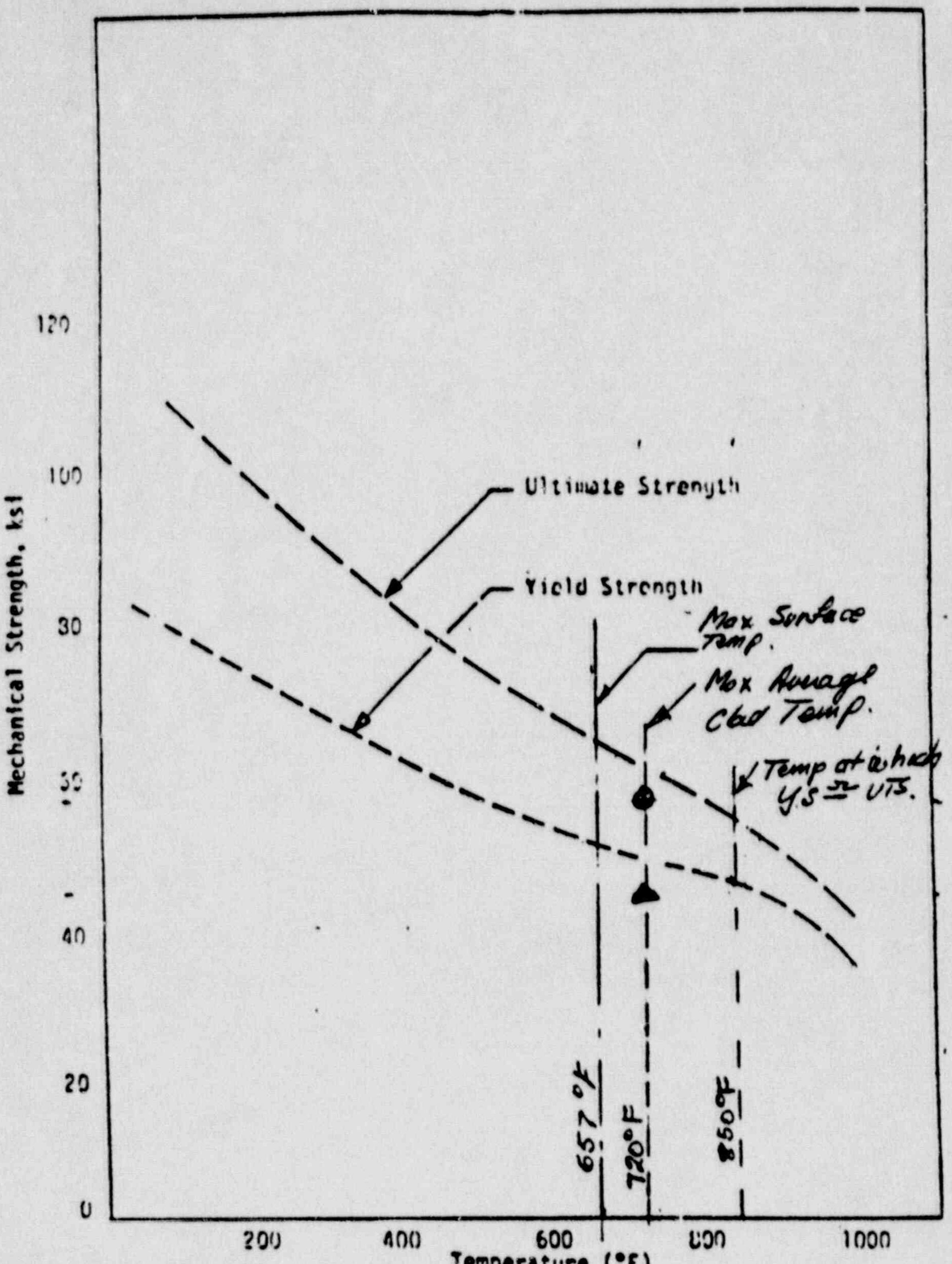
   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1.



MECHANICAL STRENGTH OF ... ROD TUBING VERSUS TEMPERATURE

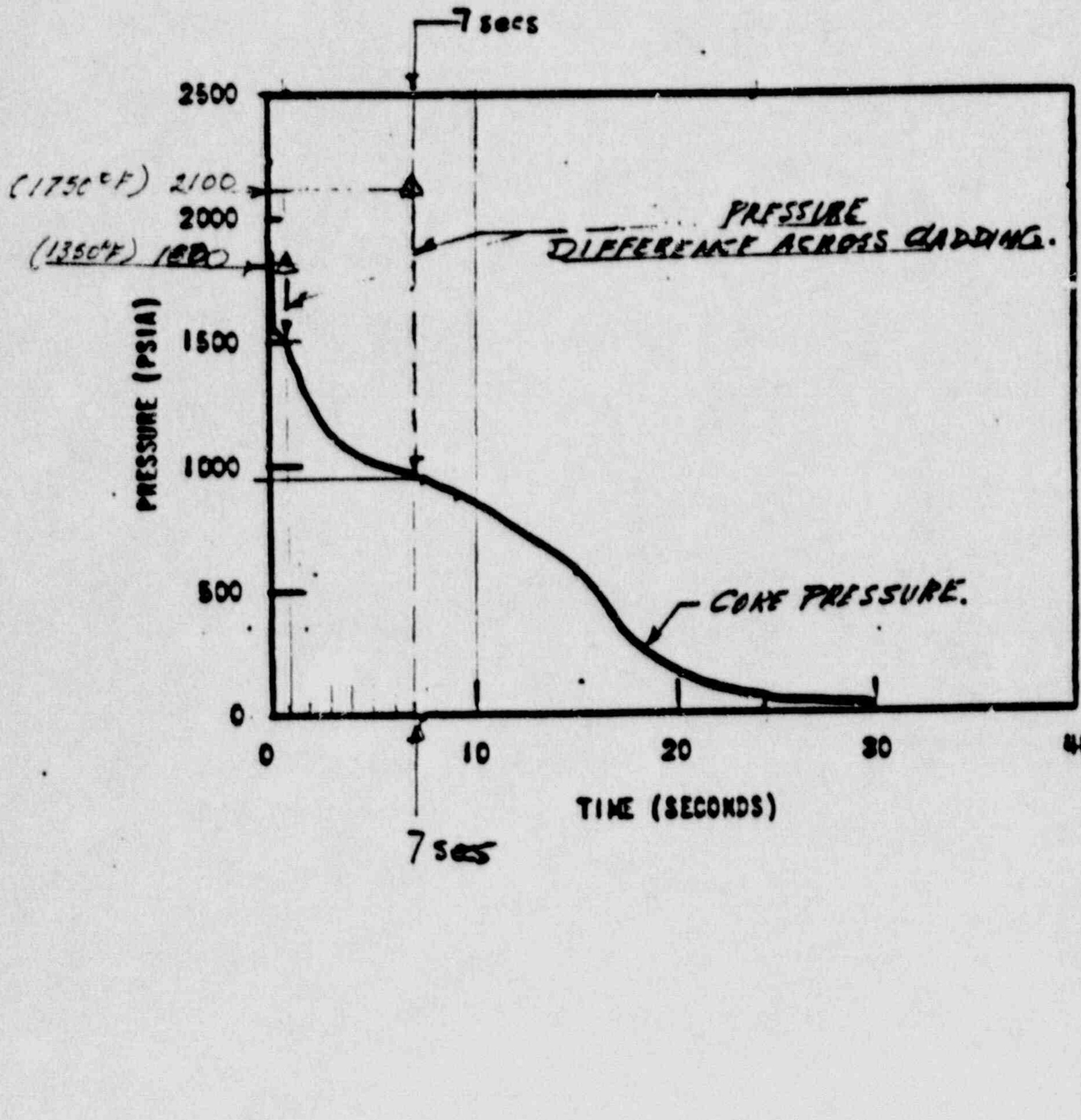


Figure 14 F.2-10a Core Pressure - DECLG (C_D=1.0)
(Unit 1)

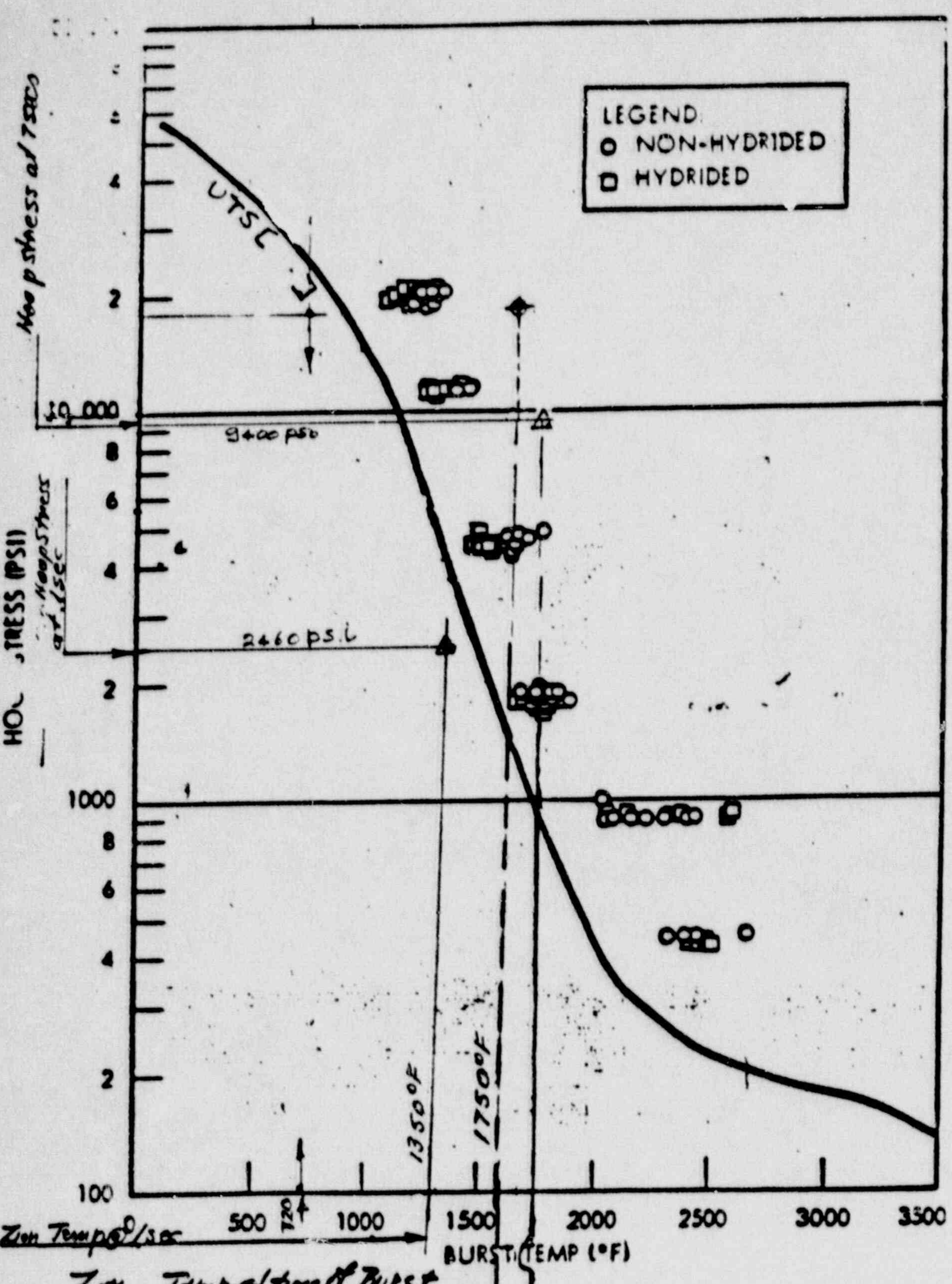
TABLE 1

Engineering Hoop Stress as a Function of Internal Fuel-Rod Gas Pressure and Fuel Vendor Design

Design	Hoop Stress (psi) for a 600 psi Differential Across the Cladding Wall
B&W 15x15	4570
B&W 17x17	4540
C-E 16x16	4280
W 15x15	4910
W 17x17	4690
GE 8x8	4050
ENC 15x15*	3940
ENC 8x8**	3880

* D. C. Cook, Unit 1

** Oyster Creek



Burst Temperature versus Stress at Burst

Zirc Peak Temp during 0-7500

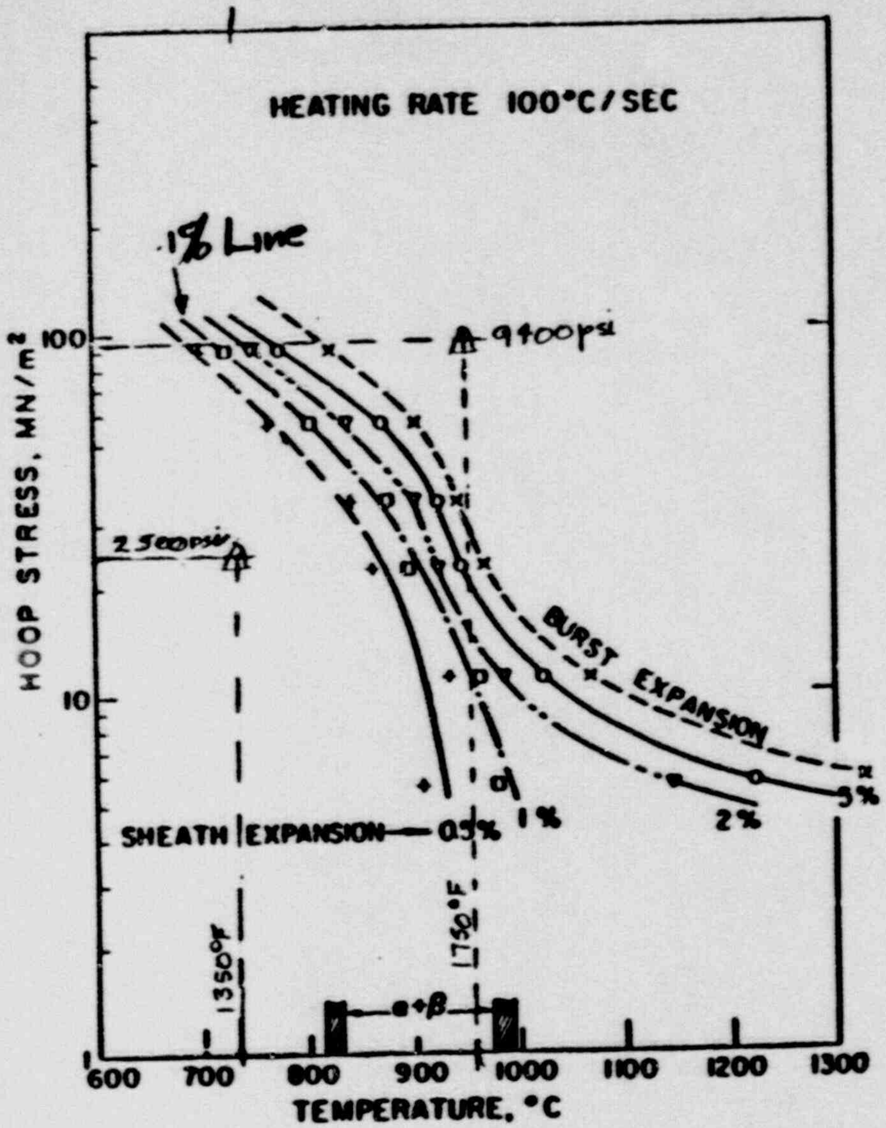


FIGURE 10 (HARDY)

Isostrain and rupture curves plotted as a function of hoop stress and temperature for tubes heated at 100°C/sec.

263

1 MN/m² = 142.9 psi

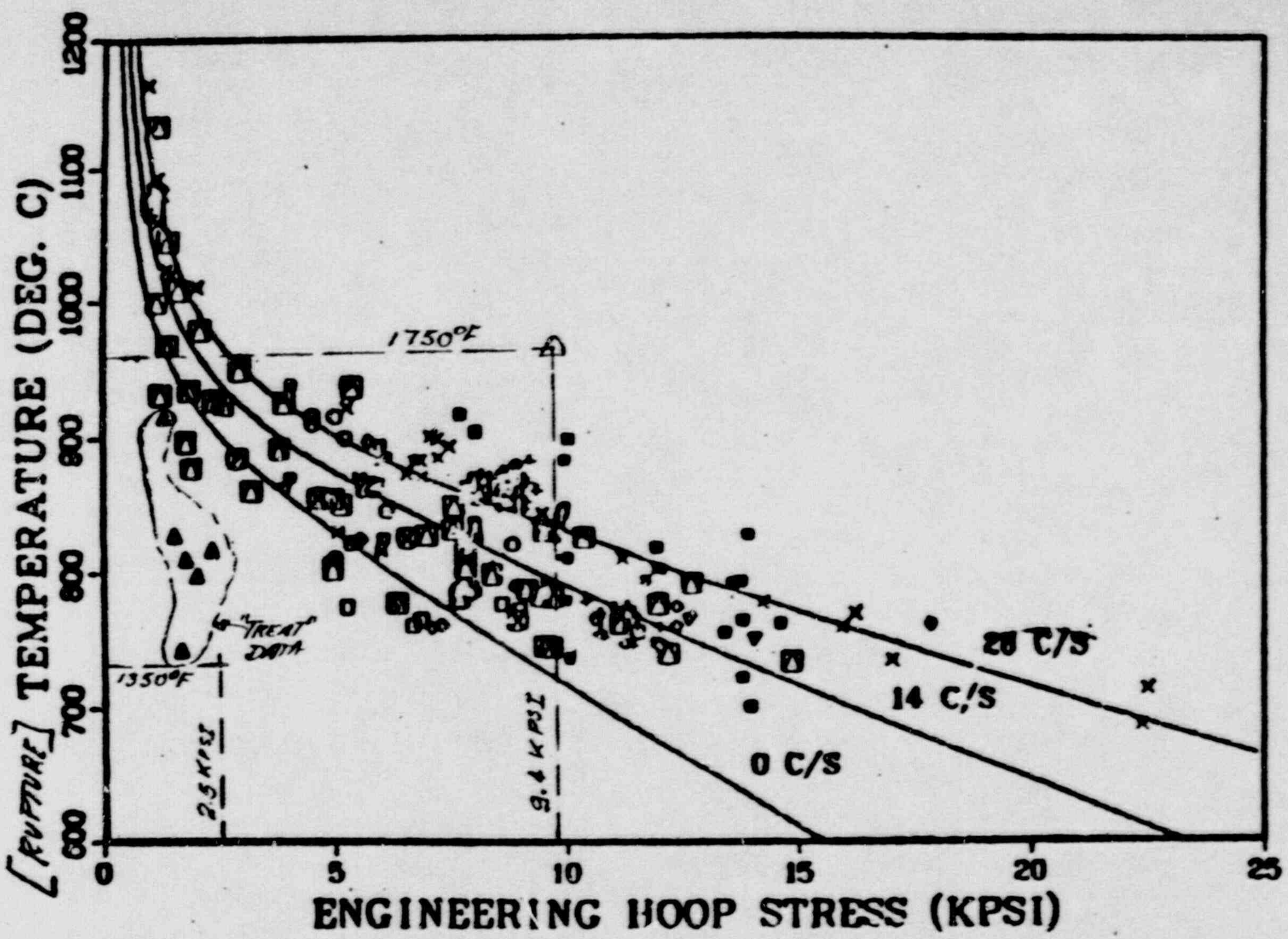


Fig. 3 ORNL correlation of rupture-temperature as a function of engineering hoop stress and temperature-ramp rate with data from internally heated Zircaloy cladding in aqueous atmospheres.

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16

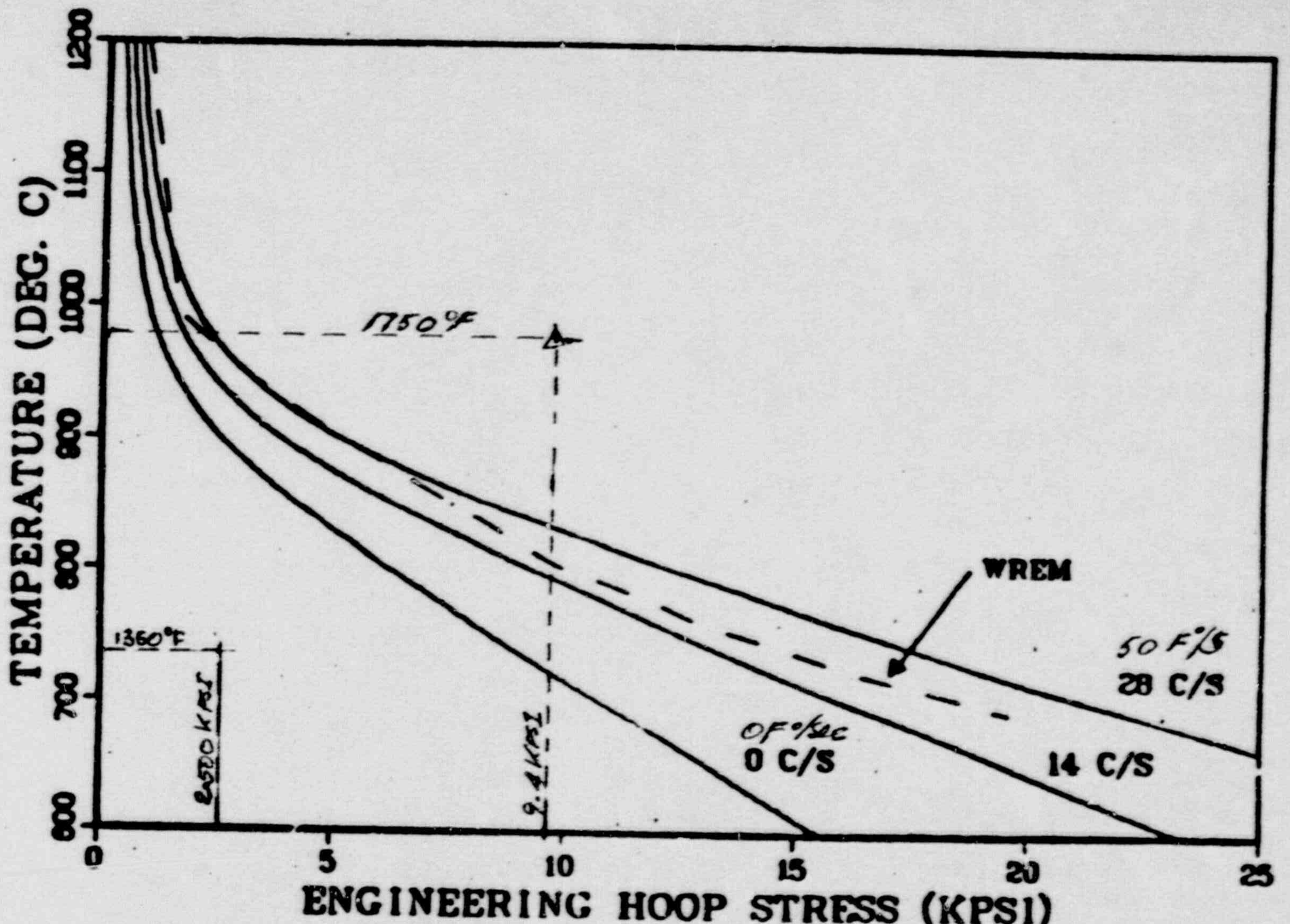

Fig. 17 WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

17
Rev 1

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3.5 Clad Swelling and Rupture Model

During a LOCA the clad is assumed to strain uniformly and plastically in the radial direction provided that both the temperature and the differential pressure across the clad are sufficiently high. If the strain exceeds [10%] or the clad temperature exceeds the burst temperature (determined as a function of the instantaneous stress) the clad is assumed to burst and an additional local strain is added to the burst node. (a,c)

Three empirical models are employed to evaluate the clad swelling and rupture behavior.

3.5.1 Clad Swelling Prior to Rupture

Hardy[24] performed a series of tests in which rods with constant internal pressure were ramped to a series temperatures at various constant ramp rates. The pressures reported by Hardy were converted to hoop stresses by the formula

(3-69)

and the strain at a given temperature and ramp rate were correlated as functions of the derived hoop stress. The equation developed which best describes the data is

(a,c)

(3-70)

where:

(a,c)

1974: Later Revised

WESTINGHOUSE

(a,c)

(a,c)

(a,c)

(a,c)

3.5.2 Clad Burst

(a,c)

Clad is assumed to burst if it reaches [10%] hoop strain based on the swelling model described above or if the clad temperature in the burst node reaches the burst temperature. [Burst temperature is calculated as a function of hoop stress based on correlation of the Westinghouse single rod burst test data shown in Figure 3-1. The best estimate curve from figure 3-1 is used and pressure is converted to hoop stress by the relationship described in Equation 3-69 using original test specimen geometry. This best estimate curve is described by the equation

Fig 3.1 [27]

(a,b,c)

$$T_{burst} = \left[\qquad \right] \tag{3-71A}$$

3.53 Local Hoop Strain After Burst

The localized diametral swelling that occurs very rapidly at the time of burst is calculated from a correlation of single rod burst test data of Westinghouse and others. Figure 3-2 shows the correlation and the ranges of the data used. Expressed in terms of hoop stress the correlation gives

(a,b,c)

$$\frac{\Delta d^*}{d_o} \left[\qquad \right] \tag{3-71B}$$

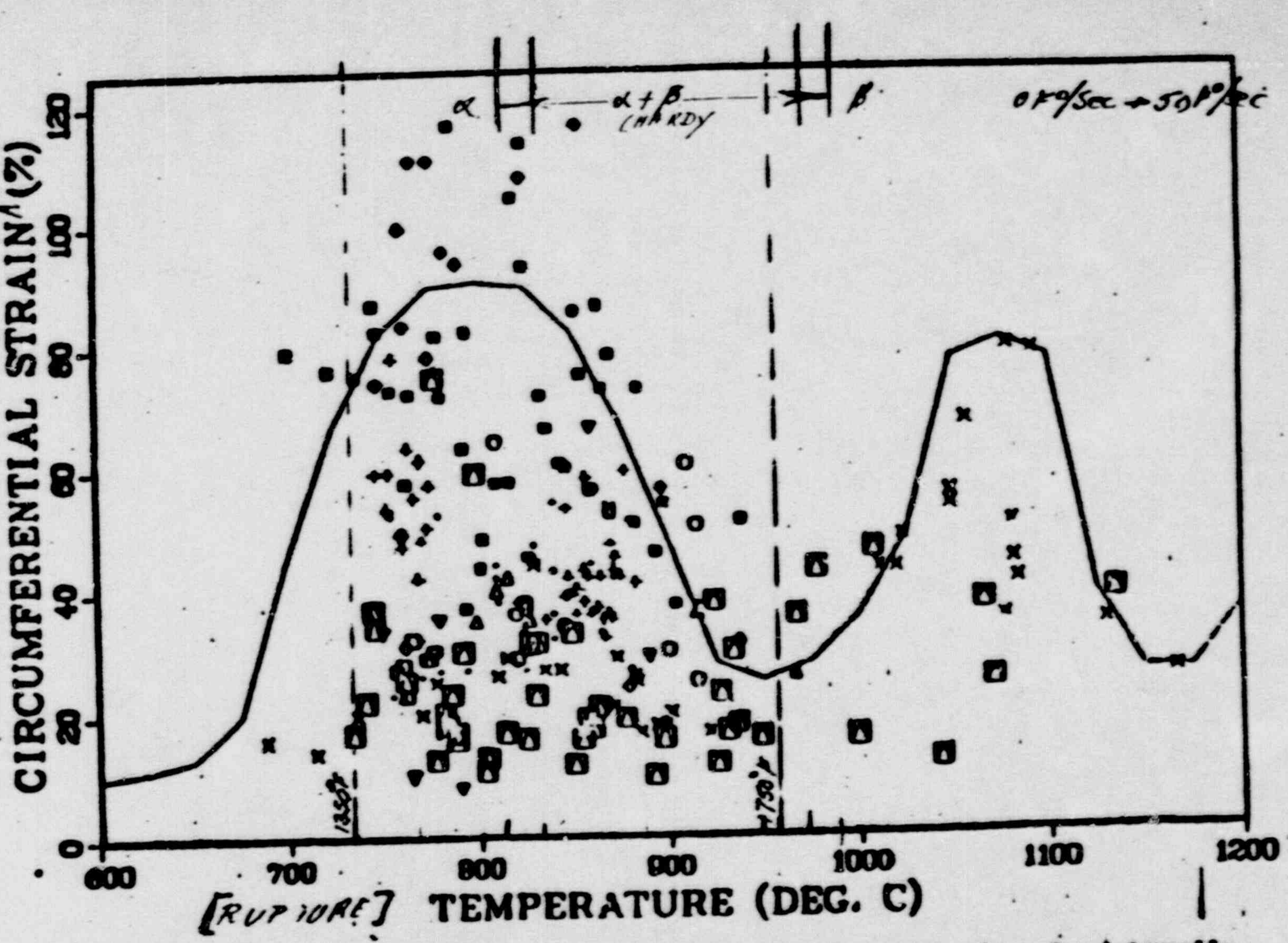


Fig. 8 ~~Maximum circumferential strain as~~ a function of ~~rupture temperature~~ for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

NUREG-75/077

THE ROLE OF FISSION GAS RELEASE IN REACTOR LICENSING

CORE PERFORMANCE BRANCH

U. S. NUCLEAR REGULATORY COMMISSION

NOVEMBER 1975

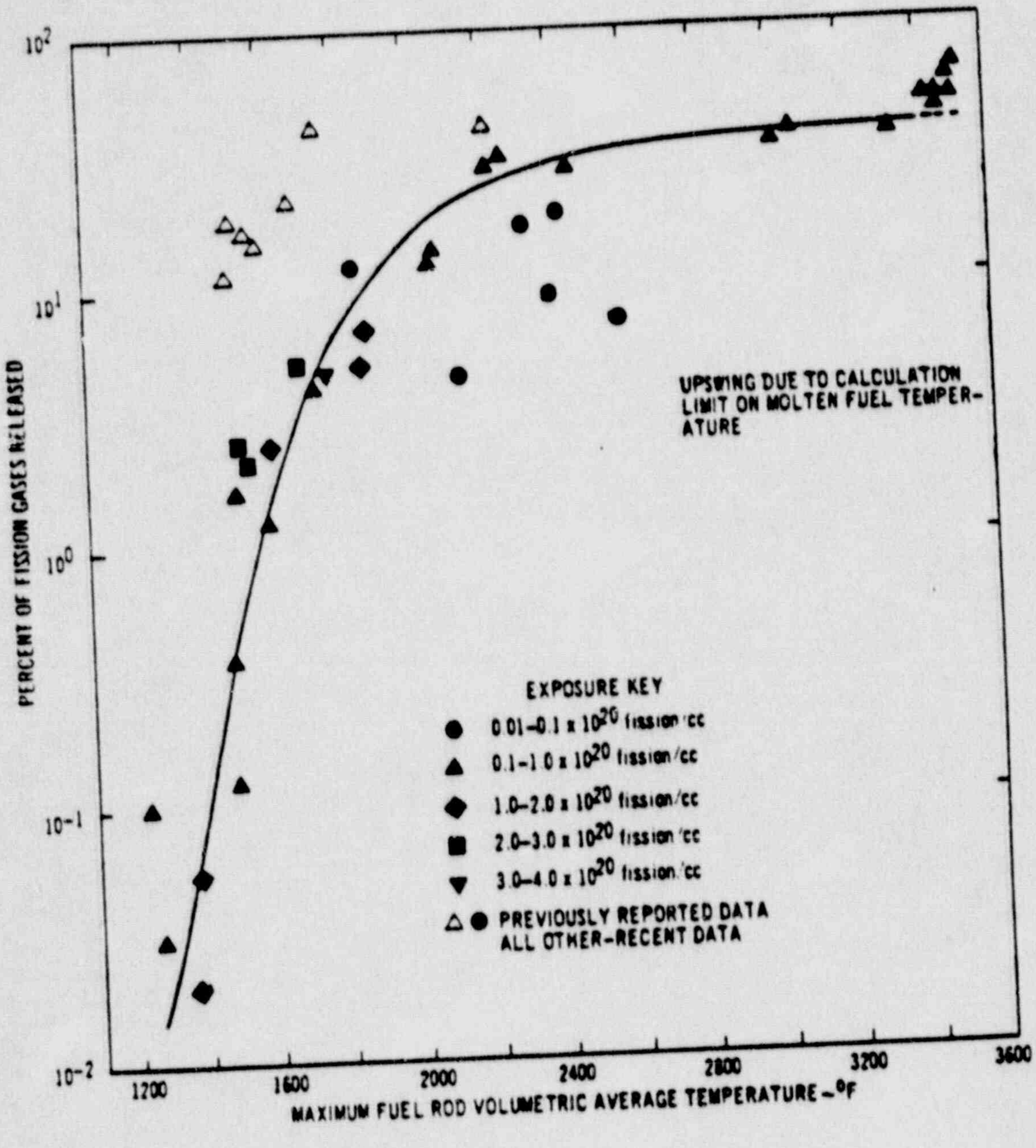


Fig. 2. The Hoffman & Coplin correlation of fission gas release as a function of temperature (from Ref. 35).

ZION

CORE TEMPERATURE DISTRIBUTION

Assumptions: Operation at 3391 MWt for 500 days

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
14.5	2500

9.10 PLANT VENTILATION

9.10.1 AUXILIARY BUILDING VENTILATION AND CONTAINMENT PURGE SYSTEMS

9.10.1.1 Design Basis

The Auxiliary Building Ventilation System serves all plant areas of the Auxiliary Building including the Fuel Handling Building, but excludes the control room, computer room, auxiliary electrical equipment room, and miscellaneous rooms and laboratories in the Auxiliary Building which are served by other independent environmental control systems. The Auxiliary Building Ventilation System also incorporates individual cubicle cooler units to provide supplementary cooling to specific safeguard equipment cubicles.

The Auxiliary Building ventilation system is designed to provide a continuous source of filtered, temperature conditioned outdoor air to maintain a thermal environment in accordance with the maximum ambient temperature for the operating equipment in the various areas served by this system as described above. During normal operation the temperature in equipment areas is limited to 105°F. In the event of a loss of outside power, this temperature is limited to 115°F, except in cubicles with unit coolers where the temperature is limited to 105°F. Conditioned air is supplied to clean areas and is routed to areas of progressively greater contamination potential. Pressure differential control dampers are employed as required to maintain a nominal 1/4" negative pressure in potentially contaminated cubicles or pipe chases. All exhaust air is routed through a return duct system, is passed through HEPA filters and is discharged to two ventilation stacks which are directed up the side of each containment building. In addition, normally bypassed charcoal filters are provided and subject to high radiation, exhaust air from the fuel handling building, potentially contaminated equipment cubicles, or the pipe tunnel can be routed through the charcoal filters prior to discharge to the ventilation stacks.

The containment purge system is designed to assure safe, continuous access (40 hours/week) to the containment within three hours after a planned or unplanned reactor shutdown by reducing the airborne particulates of the containment atmosphere. Prior to activating the purge system, the particulate and gas monitors will indicate the system activity levels inside the containment and is used as a guide for routing release from the building.

The purge system can also be used to purge the containment after a loss-of-coolant accident after the activity in the containment has reached a level low enough to allow its exhaust to the atmosphere (See Section 14.3).

9.10.1.2 System Design and Operation

9.10.1.2.1 Containment Purge System

The containment purge system provides 40,000 cfm of filtered, heated as required, outside air which is delivered to the reactor containment around the periphery of the reactor refueling pool. The air is discharged through linear grills to create a fluid boundary, or air curtain, between access areas surrounding the pool and air with a potential tritium concentration. This purge rate provides approximately 1-1/2 air changes per hour and will permit safe access to the containment three hours after a planned or unplanned shutdown. The purge exhaust is routed through HEPA filters prior to discharge to the ventilation stacks.

The purging system exhaust and air supply connections through the containment are provided with two tight-seating, 125 psig butterfly valves with one located inside and one located outside the containment. These valves are normally closed during plant operation, with the space between the valves pressurized by the penetration pressurization system.

Interlocks to the radiation monitoring system and containment pressure sensors automatically close the butterfly valves upon a high containment activity or containment isolation signal.

All actuators are remotely controlled from the control room by the operator, except where automatic interlocks are involved as described above. All actuators are designed to fail in the position required for post accident operation upon loss of electric or pneumatic power. Instruments will be located in the control room to provide information to allow proper remote operation of the system. Additionally a low air temperature alarm on the purge air supply system is provided at a local panel in the purge room.

Exhausting the air in the containment prior to entrance for maintenance or refueling will be a normal, but intermittent, source of radioactivity in airborne effluents. This radioactivity will stem from evaporation of refueling water during refueling, the desorption of gases when equipment is disassembled during cleaning or maintenance, leakage through valves and pumps, and the activation of argon in the containment air.

The amount of radioactivity in the effluent from the containment cannot be precisely predicted. This amount will vary with fuel failure levels, equipment leakage and the delay time prior to opening the containment exhaust valves. However, the emmisions from these sources normally does not exceed an annual average of 300 μCi/sec.

The isotopic composition of these emissions will also vary depending upon the sources of leakage; when in time the refueling occurs, the extent of maintenance required, and upon the decay time prior to opening containment exhaust valves. A typical breakdown might indicate that tritium comprises about 80 percent of the radioactivity from these sources. The only other radionuclides which would then be expected in quantities sufficient to effect dose calculations would be isotopes of Krypton and Xenon (Xe), with XE-133 the dominant noble gas. Iodine and particulates will be rendered insignificant in the containment exhaust due to prior circulation through charcoal and particulate filters and filtration through particulate filters during discharge.

Assuming conservatively that all the activity released from the containment ventilation system is Xe-133, then the annual whole body dose as the site boundary is approximately 1 mrem, which is well within the 10 mrem limit proposed in Appendix I of 10CFR50.

9.10.1.2.2 Auxiliary Building Ventilation System

The Auxiliary Building ventilation system is shown on Figures 9.10.1-1 and 9.10.1-2. The supply sysyem filters 100% outdoor air in two stages where the final stage has a nominal efficiency of 85% based upon the NBS atmospheric dust spot test. The filtered air is heated or cooled as required to maintain a nominal supply air temperature of 75°F ± 10°F. The system incorporates three, 50% capacity supply fans; two which normally operate and one which is standby. During normal operation (full power or shutdown), two supply fans are in operation. In the event of a loss of off-site power, the system is designed to operate with one (1) ventilation fan which is connected to the emergency power supply. During normal operation, the two operating supply fans are controlled to maintain a constant supply volume in the main supply duct.

Supply air is ducted to various areas in the auxiliary and fuel handling buildings and in general is delivered to clean areas which are normally accessible. The volume of air delivered in each area is based on the quantity of heat to be dissipated and/or to provide sufficient air change for personnel occupancy. All of the ventilation air flows to areas of progressively greater contamination potential where it is returned through a duct system to the Auxiliary Building Ventilation System equipment room (El.642). Pressure differential control dampers are located as required to maintain potentially contaminated areas at a negative pressure.

All exhaust air which is returned from the auxiliary building and fuel handling building is filtered through HEPA filters which are tested on site for a nominal bank efficiency of 99.0% based on 0.3 micron DOP tests.

The HEPA filters are arranged in three separate banks. The main bank, which filters return air from general areas in the auxiliary building, consists of six filter modules; five which operate normally and one module is standby. A second bank of HEPA filters for filtering exhaust air from the auxiliary building equipment cubicles consist of three filter modules; two which operate normally and one which is standby. A third HEPA filter bank filters exhaust air from the fuel handling building and consists of two filter modules; one normally operates and one which is standby. The spare exhaust filter modules are provided to permit the replacement of expended filter elements without interrupting the normal exhaust air capacity. The filtered exhaust air is then routed up the ventilation stacks.

Two banks of charcoal filters have been provided and on detection of high radiation, air from potentially contaminated equipment cubicles, or the pipe tunnels will automatically be routed through the charcoal filters prior to exhaust. In addition, exhaust air from the fuel handling building is routed through the charcoal filters during the refueling operation. A maximum of three (3) out of four (4) charcoal booster fans will automatically start to account for the increase in system resistance in accordance with the required flow through the charcoal filters. The exhaust air from cubicles of potentially high radiation which can be passed through the charcoal filters include the following:

a. Boric acid evaporator feed pumps
b. Hold-up tank recirculating pump
c. Auxiliary building equipment drain tank and pumps
d. Auxiliary building sump A
e. Residual heat removal pumps
f. Diesel residual heat removal pumps (future)
g. Cavity fill pumps (future)
h. Gas decay tank
i. Auxiliary building sump B
j. Containment spray pumps
k. Residual heat exchangers
l. Safety injection pumps
m. Charging pumps
o. Boric acid and rad waste evaporators

In addition to the exhaust filters described above, a local HEPA filter is provided to filter exhaust air leaving the instrument calibration room. An air flow bypass with manual dampering is provided to permit replacement of the filter elements without interrupting the exhaust air flow.

A local filter system is provided for the drumming station exhaust air. Normally, the exhaust air from the drumming station bypasses the local filter. Whenever the drumming station is in operation, the exhaust air is passed through a HEPA filter and charcoal filter before being routed to the main ventilation exhaust system.

Six (6) auxiliary building exhaust fans are provided; four (4) which normally operate and two (2) which are standby. The exhaust fans are arranged such that two (2) out of three (3) exhaust fans discharge the air flow to each of two stacks. In the event of a loss of off-site power, the system is designed to operate with two (2) exhaust fans.

Equipment located in the safeguard equipment cubicles does not normally operate. These cubicles are connected to the ventilation system in such a manner so as to provide a nominal amount of ventilation during operation. Whenever the equipment within an individual cubicle is operating, supplementary cooling is required. To meet this function, auxiliary building cubicle unit coolers are installed in the cubicles. The unit coolers are each designed to limit the maximum ambient temperature to 105°F. Each cooler is equipped with either two, three, or four ventilating fans arranged to operate in parallel with a single section cooling coil housed in a common cabinet.

One of the fans on each cubicle unit cooler is a spare. However, whenever cooling is required, all fans will automatically start, and consequently, the design allows for the loss of one fan without the loss of design cooling capacity. The auxiliary building cubicle unit coolers are connected to the emergency power supply and are designed to operate under any normal or abnormal plant condition. Cubicle unit coolers have been provided for the following equipment:

a. Residual heat removal pumps
b. Safety injection pumps
c. Containment spray pumps
d. Charging pumps

Provision for future installation of cubicle unit coolers has been made in the diesel driven RHR and cavity flood pumps cubicles.

All essential operating functions are monitored and controlled from the control room. Each supply and exhaust fan may be manually started and stopped from the control room.

Operation of the auxiliary building cubicle unit coolers is completely automatic and these units start whenever the equipment in the respective cubicle operates.

All system temperature control is maintained from a local panel and the supply temperature is maintained between the limits of 65°F and 85°F.

The supply fans are controlled to maintain constant air supply to the auxiliary building. The exhaust fans are controlled to maintain the auxiliary building at a nominal 1/4" negative pressure with respect to the outdoors. The pressure in potentially contaminated areas is controlled for approximately 1/4" negative pressure with respect to adjacent clean areas in the auxiliary building.

The auxiliary building exhaust air routing through the charcoal filters, as described above, is automatically controlled by high radiation signals from the previously described areas. Exhaust air from the fuel handling building is passed through the charcoal filters during refueling.

System variables pertaining to normal operation are indicated on the main control room panel. Abnormal conditions, such as high temperature, low temperature, low building differential pressure, high pressure drop across filters, and high radiation are annunciated either locally or on the main control panel.

9.10.1.3 System Components

All equipment will be factory inspected and tested in accordance with the applicable equipment specifications. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, the system was balanced for the design air and water flows. Control on each system was checked, adjusted and tested to insure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

9.10.1.3.1 Supply Filters

The auxiliary building ventilation supply filters are composed of banks of prefilters and high efficiency filters installed in series. Each filter unit has a rated flow of 300,000 cfm. Each prefilter bank contains 143 individual filter elements rated at 10% efficiency based on the NBS atmospheric dust spot test. Each high efficiency filter bank contains 143 individual filter elements rated at 85% efficiency based on the NBS atmospheric dust spot test.

9.10.1.3.2 Heating Coils

The auxiliary building ventilation heating coils are composed of twelve (12) coil sections arranged in 2 sets in parallel, each of six (6) coils. The heating coils are designed to heat 300,000 cfm from -10°F to 65°F when supplied with hot water at 270°F. The total coil capacity is 24.3×10^6 Btuh.

9.10.1.3.3 Supply Fans

The auxiliary building ventilation supply fans are of the direct-driven vane axial type located downstream of the heating coils. Each fan is rated at 150,000 cfm at a total pressure of 6.5" H_2O and is driven by a nominal 200HP motor.

9.10.1.3.4 Cooling Coils

The auxiliary building cooling coils are composed of twelve (12) sections arranged in 2 sets in parallel, each of six (6) coils. The cooling coils are designed to cool 300,000 cfm from 98.5°F to 81.5°F when supplied with water at a temperature of 78°F. This guarantees the minimum required capacity to supply air at 85°F in case of a loss of one coil section. The total coil capacity is 5.53×10^6 Btuh.

9.10.1.3.5 Exhaust Fans

The auxiliary building vent system exhaust fans are of the direct-driven vane axial type. Each fan is rated at 75,000 cfm at a total pressure of 9" H_2O and is driven by a nominal 150 HP motor.

9.10.1.3.6 Charcoal Booster Fans

The auxiliary building vent system charcoal booster fans are of the direct-driven vane axial type. Each fan is rated at 22,000 cfm at a total pressure of 3.5" H_2O and is driven by a nominal 15 HP motor. These fans are designed to overcome the additional resistance of the charcoal adsorbers when the air from the cubicles, fuel handling building or the pipe tunnel is routed through the charcoal adsorbers as a result of high radiation or during refueling.

9.10.1.3.7 Fuel Handling Building Exhaust Filters

The fuel handling building exhaust filters are composed of banks of prefilters and HEPA filters installed in series. Each filter unit has a rated flow of 22,000 cfm. Each prefilter bank contains 24 individual filter elements rated at 85% efficiency based on the NBS atmospheric dust spot test. Each HEPA filter bank contains 24 individual filter elements each having a nominal efficiency of 99.7% based on the DOP test.

9.10.1.3.8 Cubicles Exhaust Filters

The auxiliary building cubicles exhaust filters are composed of banks of prefilters and HEPA filters installed in series. Each filter unit has a rated flow of 20,000 cfm. Each prefilter bank contains 24 individual filter elements rated at 85% efficiency based on the NBS atmospheric dust spot test. Each HEPA filter bank contains 24 individual filter elements each having a nominal efficiency of 99.7% based on the DOP test.

9.10.1.3.9 Main Exhaust Filters

The auxiliary building vent exhaust filters are composed of banks of prefilters and HEPA filters installed in series. Each filter unit has a rated flow of 48,000 cfm. Each prefilter bank contains 48 individual filter elements rated at 85% efficiency based on the NBS atmospheric dust spot test. Each HEPA filter bank contains 48 individual filter elements each having a nominal efficiency of 99.7% based on the DOP test.

9.10.1.3.10 Charcoal Adsorbers

The auxiliary building charcoal exhaust system is composed of banks of charcoal adsorbers rated at 32,000 cfm. Each unit contains 91 individual elements which are tested at least once every 18 months. A laboratory analysis is run, and when charcoal adsorber efficiency to adsorb methyl iodide decreases to $\leq$ 95% at 95% RH, the charcoal adsorbers are replaced.

9.10.1.3.11 Cubicle Unit Coolers

The auxiliary building cubicle unit coolers are designed to limit the maximum ambient temperature to 105°F. Each cooler is equipped with either two, three, or four ventilating fans arranged to operate in parallel with a single section cooling coil housed in a common cabinet. One of the fans on each cubicle unit cooler is a spare. However, whenever cooling is required, all fans will automatically start, and consequently, the design allows for the loss of one fan without the loss of design cooling capacity. The unit coolers have the capacities as listed on Table 9.10.2-1.

9.10.1.3.12 Miscellaneous Vent System

Vent lines from various tanks containing radioactive or potentially contaminated wastes are headered together and pass through a prefilter, a HEPA filter and charcoal adsorber before discharging to the atmosphere. The filters have a rated flow of 3000 cfm.

9.10.1.3.13 Instrument Calibration Room Filter

The instrument calibration room filter consists of one prefilter element having a nominal efficiency of 35% based on the NBS atmospheric dust spot test, and one HEPA filter element having a nominal efficiency of 99.7% based on the DOP test. The filters have a rated flow of 1,000 cfm.

9.10.1.3.14 Sample Room Exhaust

Ventilation air from the Sample Room is discharged into the Auxiliary Building Ventilation System which is filtered and monitored as described in Section 9.10.1.2.

9.10.1.3.15 Drumming Station Exhaust System

The drumming station exhaust filters are composed of banks of prefilters, HEPA filters and charcoal filters. Each has a rated flow of 4,000 cfm. The prefilter bank contains 4 individual filter elements having a nominal efficiency of 35% based on the NBS atmospheric dust spot test. The HEPA filter bank contains 4 individual filter elements having a nominal efficiency of 99.7% based on the DOP test. The charcoal bank contains 12 individual adsorber elements. Laboratory analysis on samples of charcoal are run at lease once every 18 months. The charcoal adsorbers are replaced when the bank efficiency to adsorb methyl iodide decreases to 90% at 95% RH.

9.10.2 CONTROL ROOM HEATING, VENTILATING AND AIR CONDITIONING SYSTEM

9.10.2.1 System Design and Operation

The control room heating, ventilating and air conditioning system including ductwork is designed to function during the Design Basis Earthquake (DBE). The design consists of two, 100% capacity equipment trains. One system will operate normally and one system is standby. All of the electric equipment required for maintaining a continuously operating, heating, ventilating and air conditioning system is connected to the emergency electrical system. The normal occupancy of the control room will be three to five people.

The design of the heating, ventilating and air conditioning for the control room is based on a system with the capability to provide air filtration, heating and/or cooling, with humidification and/or dehumidification with continuous occupancy under any normal or abnormal condition to permit a safe shutdown of the plant as may be required.

Interconnected areas (such as computer room, auxiliary electric equipment, offices, cold and process laboratories, etc.) are served by a separate air conditioning system as described in Section 9.10.4.

The control room HVAC system arrangement shown on Figure 9.10.2-1 consists of high efficiency filters, a charcoal filter for odor removal, a cooling and heating coil package, a humidifier, and a direct-driven vane axial fan. Air is supplied through a louverall ceiling. Return air passes through the control boards into a return duct system which is connected to outlets on top of the boards. The return air is mixed with outside air as required to meet minimum ventilation requirements and/or room cooling requirements. The control room HVAC system equipment room and outside air intake is located at El. 617'-0".

A fixed minimum quantity of make-up air is supplied which is sufficient to maintain a positive operating pressure in the control room with respect to adjacent areas to prevent inleakage. The system will be controlled to maintain a nominal 75°F and 40% RH.

The normal make-up air supply is the outside air supply which is brought in through a missile-protected wall opening. In the event of high radiation detection in the outside air supply, the normal outside air inlet is closed, which allows make-up air for pressurizing the control room to be introduced from the turbine room. Also, the emergency make-up filter fan is automatically started, and make-up air is introduced through HEPA filters and charcoal adsorbers for the removal of potential radioactive contamination. In all other respects, the system operates as for normal operation.

In the event of smoke in the control boards, smoke detectors will annunciate in the control room. Concurrently, all of the supply air delivered to the conditioned spaces will pass through a normally bypassed charcoal filter for smoke and odor absorption.

Mechanical cooling for the control room system is provided by means of two (2) 100% capacity water-cooled refrigeration units. Each refrigeration unit will be connected to its respective air handling unit.

Each air conditioning system has a local control panel and each is independently controlled. Important operating functions are controlled and monitored from the main control room.

Instrumentation is provided to monitor important variables associated with normal operation. Instruments to alarm abnormal conditions are provided in the control room. All of the facilities discussed above are structurally and physically arranged within the control room complex.

9.10.2.2 Normal Operation

Each control room HVAC system is controlled to provide ventilation, heating and/or cooling on a year-round basis. Two (2) full capacity air handling units are provided; one of which is standby.

Each air handling unit is connected to a full capacity refrigeration unit. During normal operation a minimum amount of outdoor air is introduced to the system for the purpose of odor and smoke (cigarette) dilution, and system pressurizing. Return air from the conditioned spaces is ducted back through a return fan and is recirculated through the air handling unit or is exhausted to the outdoors as conditions dictate. A standby return air fan is provided. Each system is provided with separate controls.

The HVAC systems are controlled to maintain the following conditions:

Outside Air Ventilation:	2000 cfm
Control Room Temperature:	75 $\pm$ 1°F
Control Room Humidity:	40 $\pm$ 5% RH
Control Room Pressure Relative to Adjacent Exterior Spaces:	0.25 $\pm$ .03 in H_2O.

9.10.2.3 Control Room Ventilation Isolation

9.10.2.3.1 General

The control room is designed to accommodate the normal occupancy of three to five people. The control room, cable room, computer room and other interconnecting areas are, from a ventilation standpoint, isolated completely from the remainder of the auxiliary building. The control room has two (2), 100% capacity HVAC systems together with adequate shielding which is independent from the HVAC system for the remainder of the interconnecting areas. All HVAC equipment for the control room has the necessary standby capacity to accommodate an equipment failure and is powered from the emergency electrical system so that adequate ventilation is maintained at all times.

9.10.2.3.2 Radiation Level Control

A pressure differential controller, with sensing elements in the control room and the auxiliary and turbine buildings, modulates a control damper in the relief duct to maintain a positive pressure within the control room. This will prevent the inleakage of any airborne activity from outside the control room area. All cable, piping and miscellaneous penetrations through the biological barrier are sealed to minimize the magnitude of leakage. Personnel doors will be tight-fitting and gasketed.

A radiation monitor in the outside air intake is set to sense an abnormal level of activity. On an increase in activity approximately 100 times above background, the HVAC system automatically takes make-up air from the turbine building and automatically starts one (1) of two (2) 100% capacity emergency filter make-up air fans and routes the make-up air through two (2) HEPA filter and charcoal adsorbers arranged in series. The minimum quantity of outdoor air introduced into the system under all conditions replaces air leakage for system pressurizing. The quantity of make-up air to the control room is more than sufficient to satisfy personnel requirements. In all other respects, the system operates as for normal operation.

The guiding principle of this design is to have alternate fresh air intake points such that either, but probably not both, could be contaminated by containment leakage following a LOCA. However, in any event, inlet filtration capability is provided, in addition to the recirculation filters in normal use, to ensure the system's capability to provide clean air.

A re-evaluation of this system in terms of post-LOCA thyroid doses was based on the following set of assumptions:

a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core is immediately available for leakage from the primary reactor containment - as per AEC Safety Guide 4, paragraph C.1.d.

b. Initial gross iodine and tellurium inventories used are slightly higher than given by the procedure outlined in TID 14844 due to an allowance for higher yields from plutonium fissions in ripe fuel (8.2×10^8 curies of iodine isotopes, 2.8×10^8 curies of tellurium isotopes).

c. One percent of the initial tellurium inventory is immediately available for leakage from the primary reactor containment. Residual tellurium in the fuel is allowed to decay into iodine, one fourth of which then becomes available for leakage.

d. Leakage from the primary reactor containment is assumed to average 0.1 percent for the first day and 0.045 percent per day thereafter.

e. The plume of escaping isotopes is assumed to travel in the worst possible direction; - directly into the turbine building from which fresh air make-up is drawn.

f. Circulation in the turbine building through convective ventilation out of the upper windows is assumed to continue after the LOCA at 2.5 x 10^4 cfm.

g. Free volume of turbine building is taken to be 12,000,000 ft^3.

h. Free volume of control room - 132,500 ft^3.

i. Tellurium and iodine decay is allowed to proceed during residence in the turbine building and control room.

j. The eight hour breathing rate from AEC Safety Guide 4, paragraph 2.c (10 $meters^3$/8 hours) is used throughout.

k. Iodine dose conversion factors as given by ICRP Pub. 2, Report of Committee II, "Permissible Dose for Internal Radiation," 1959 are used. (See also Table 14.3.5-2).

l. The containment spray removes 90 percent of the iodine present as elemental and particulate iodine. No credit is taken for removal of any of the organic iodides.

m. Ingress and egress from the control room is assumed to take six minutes per shift (2 times 400 meters to site boundary at 4 miles per hour). It is conservatively assumed that the individual breathes air with the same iodine content as that at the control room filter intake.

Results of the re-calculation of control room doses based on the above assumptions are presented in the following table:

Elapsed time after LOCA (hours)	Accumulated Thyroid Dose (Continuous Occupancy) (Rem)	Accumulated Thyroid Dose (1/3 Occupancy*) (Rem)	Iodine Trapped on Filter (milligrams)
36	.165	.125	3
92	.257	.194	6
140	.318	.238	9
308	.462	.347	18
812	.607	.456	44
1484	.632	.474	81
8760	.664	.498	85

* Including allowance for ingress and egress this dose approximates individual exposures for one eight hour shift per day.

These results are very conservative since they are based on the 60 day persistance of the worst possible meteorological condition and include no allowances for plate-out, fallout, washout or similar loss factors. In addition, no iodine intake has been allowed for the activated charcoal filters used for smoke and odor removal in the recirculation part of the control room air conditioning system. Since the tandem HEPA - charcoal filters contain a total of approximately 420 pounds of activated charcoal, the iodine removed from the air supply to the control room represents an extremely light loading. No decrease in breathing rate after eight hours has been assumed, and no use of available respiratory equipment during ingress and egress has been assumed.

9.10.2.3.3 Fire/Smoke Detection and Control

An equipment fire in the control room will not cause abandonment of the control room and will not prevent a safe shutdown of the plant.

Overload and short circuit protection is provided for the electrical control and instrumentation equipment within the control room. There are no power cables in the control room, therefore the fire hazard due to electrical faults is minimized. All electrical wiring and equipment is surrounded by or mounted in metal enclosures. The redundance of the reactor trip, Engineered Safeguards Systems Control circuits and the associated segregation and physical separation afforded for redundant channels (including wiring) allows only isolated damage to electrical equipment.

In the event of a fire, the operators have available portable respiratory equipment and portable fire extinguishers located and used in accordance with National Fire Code and National Fire Protection Association specifications. It is considered that the equipment provided is adequate to control any such fire that could possibly result and prevent a forced abandonment of the control room. The ability to achieve a safe shutdown condition during a fire is discussed in References 1 and 2.

To prevent the spread of fire behind the control boards and interconnected areas, the following provisions are made:

a. Cables used throughout the installation will have an exterior jacket that meets the IPCEA vertical flame test requirements. Power and control cables for application at 480-volt and lower are rated at 600-volts and insulated with oil-based, high temperature rubber and covered with a fire resistant jacket of similar material. Shielded instrumentation cables are insulated with fire resistant chlorosulfurated polyethylene and covered with a jacket of the same material.

b. Structural and finish materials (including furniture) for the control room and interconnecting areas have been selected on the basis of fire resistant characteristics. Structural floors and interior walls are of reinforced concrete. Interior partitions incorporate metal, masonry or gypsum dry walls on metal joints. The control room ceiling, door frames, and doors are metallic. Wood trim was not used.

The design of the HVAC system insures a habitable environment both during an outbreak of fires or smoke and during the period required to bring the condition under control. The HVAC system is arranged to establish a ventilation pattern which routes supply air first to the normally occupied spaces and then exhausts air through normally unoccupied areas to the return duct system. In the control room, conditioned air is supplied to the occupied spaces through a ceiling distribution system. Air is exhausted from the control room through openings in the control boards and up through exhaust ducts to a return header located above the ceiling.

The air outlets from the control boards is provided with dampers to regulate the distribution of air flow to each board section. Since the control boards are under negative pressure, any leakage through board cracks will be from the occupied area to behind the boards.

In the event of fire, smoke, or products of combustion in the control boards, smoke detectors will alert the operators and will automatically position dampers to pass all of the supply air delivered to the conditioned spaces through a normally bypassed charcoal filter for smoke and odor absorption. A manual override is also provided for this function. Also, a smoke detector is located in the emergency make-up air supply from the turbine building. On sensing smoke the detector will generate a signal which will divert all of the supply air delivered to the control room through the charcoal adsorbers. |

9.10.2.4 <u>System Components</u>

All components and their supports meet the requirements for Seismic Class I structure.

9.10.2.4.1 Supply Air Filters

The control room supply air filters are composed of banks of prefilters and high efficiency filters installed in series. Each filter unit has a rated flow of 13,500 cfm. Each prefilter bank contains 12 individual filter elements rated at 35% efficiency based on the NBS atmospheric dust spot test. Each high efficiency filter bank contains 12 individual filter elements rated at 95% efficiency based on the NBS atmospheric dust spot test.

9.10.2.4.2 Air Handling Units

The control room HVAC system air handling units consist of cooling and heating coils and cold deck and hot deck mixing dampers arranged in a housing with an interconnecting direct-driven vane axial fan arranged for blow-through operation. Each air handling unit is 100% design capacity with one unit as a spare. The heating coil is designed to heat 12,400 cfm of air from 62°F to 88°F when supplied with 20 gpm of water at 210°F. The cooling coil is designed to cool 13,500 cfm of air from 81°F DB and 62.5°F WB to 47.5°F DB and

46.8°F WB when supplied with refrigerant at 39°F. The total coil capacity is 610,000 Btu/hr. The air handling unit supply air fans are of the direct-driven vane axial type and are mounted on the inlet of each air handling unit coil cabinet. Each fan is rated at 13,500 cfm at a total pressure of 4.97" H_2O and is driven by a nominal 20 HP motor.

9.10.2.4.3 Return Air Fans

The control room HVAC System return air fans are of the direct-driven vane axial type and are mounted in line in the return ductwork. Each fan is rated at 13,500 cfm at a total pressure of 2.5" H_2O and is driven by a nominal 7.5 HP motor. Two fans are provided for the control room system and each is 100% design capacity with one (1) fan as a spare.

9.10.2.4.4 Odor Filter

The control room HVAC system charcoal odor absorption unit is located downstream of the ventilation particulate filters and upstream of the air handling unit supply fan. Each charcoal bank has a rated air flow of 13,500 cfm and contains 80 individual charcoal absorption elements. Each filter bank is 100% design capacity with one (1) filter unit as a spare.

9.10.2.4.5 Make-Up Air Filter Units

The control room HVAC system make-up air filter units are located in the makeup air duct coming from the turbine room. Each filter unit consists of a prefilter, HEPA filter and two charcoal absorbers arranged in series and are each designed for 99% iodine adsorbtion. Face velocity is 40 feet/min and bed depth is two inches. Laboratory analysis on samples of charcoal are taken at least once every 18 months. The charcoal adsorbers are replaced when the bank efficiency to adsorb methyl iodide decreases to 90% at 95% RH. Each filter unit has a rated air flow of 2,000 cfm. The prefilter bank contains 2 individual filter elements having a nominal efficiency of [illegible]% based on the NBS atmospheric dust spot test. Each HEPA filter bank contains [illegible] filter elements each having a nominal efficiency of 99.7% based on the DOP test.

Based on the 2000 cfm minimum outdoor air flow, and the fact that charcoal filters can absorb an average of 33.3% of its own weight of most odor causing substances, and an average of 16.7% of chlorine and other toxic gases, the make-up charcoal filter has the capability of absorbing 70 lbs. of chlorine and other toxic gases (charcoal weight of 420 lbs.) Assuming that 50% chlorine and 50% other gases would be absorbed by the charcoal filter simultaneously, a total of 35 lbs. of chlorine can be absorbed. The charcoal filters would then be adequate for 12 hours of operation at an inlet chlorine concentration of 162 ppm.

9.10.2.4.6 Make-up Air Filter Unit Fans

The control room HVAC System make-up air filter unit fans are of the direct-driven, centrifugal type and are located on the downstream side of the make-up air filter units. Each fan is rated at 2,000 cfm at a static pressure of 6" H_2O and is driven by a nominal 3 HP motor. Two fans are provided and each is 100% design capacity with one (1) as a spare.

9.10.2.4.7 Refrigeration Units

The control room HVAC system refrigeration units are of the packaged, reciprocating water cooled type. Each refrigeration unit is interconnected to its respective air handling unit. Each condensing unit will develop 51 tons (612,000 Btu/hr) of cooling capacity corresponding to a suction temperature of 49°F and a condensing temperature of 105°F. The condenser on each unit is water cooled and will dissipate 880,000 Btu/hr of heating when supplied with 160 gpm of water at 80°F. Two (2) condensing units are provided for the control room system and each is 100% design capacity with one (1) as a spare.

9.10.2.5 Inspection and Tests

All equipment will be factory inspected and tested in accordance with the applicable equipment specification. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, the system was balanced for the design air and water flows. Controls on each system were checked, adjusted and tested to insure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

9.10.3 CONTAINMENT VENTILATION SYSTEM

9.10.3.1 Performance Objectives

The containment ventilation system as shown in Figure 9.10.3-1 will during normal operation accomplish the following:

a. Limit the average containment thermal environment to 120° maximum and 65°F minimum. The design cooling load is based on all internal and external effects. In particular the ambient air surrounding the reactor coolant pumps is limited to 120°F.

b. Permit cleanup of the containment atmosphere prior to limited personnel access at power.

c. Provide ventilation to remove all the heat generated within the CRD shroud and limit the maximum exhaust air temperature to 155°F.

d. Provide ventilation in the reactor vessel cavity to remove the thermal and gamma heat losses from the reactor vessel and limit the maximum temperature of the biological shield to 150°F. In addition, provide ventilation to the out-of-core instrumentation to limit the maximum surrounding concrete surface temperature to 150°F, and limit the surrounding air temperature to 135°F.

e. Minimize the risk to operators on the manipulator crane bridge from inhaling potential airborne tritium during refueling.

f. Maintain the normal containment atmospheric pressure between -0.1 and +0.3 psig.

g. Provide means to reduce the concentration of particulate and gaseous contamination to assure safe continuous access (40 hours/week) during normal reactor shutdown.

h. Provide the necessary instrumentation and controls to permit the required monitoring and control of all systems from outside the containment for all required modes of operation.

9.10.3.2 <u>Design Description</u>

The reactor containment ventilation system is a recirculation system designed to limit the maximum thermal environment inside the containment and is completely isolated from the atmosphere outside the containment except under the following conditions:

a. During normal operation only when operating pressure is less than or greater than -0.1 psig and +0.3 psig respectively. Isolation is broken by a pressure and vacuum relief system which connects the containment atmosphere and the outside atmosphere to keep the pressure inside the containment within the above range.

b. After normal shut down of the reactor, the containment exhaust air will be purged through HEPA-filters to reduce potential airborne particulates and dilute the concentration of any gaseous constituents.

c. After a loss-of-coolant accident, the containment air will be purged when the activity in the containment has reached a level low enough to allow its exhaust to the atmosphere. (See Section 14).

The reactor containment ventilation system consists of the following sub-systems:

1. Reactor Containment Fan Coolers (RCFC)
2. Containment Activated Charcoal Filter Units
3. Reactor Cavity and Out-of-Core Instrumentation Ventilation
4. Control Rod Drive Mechanism (CRDM) Ventilation
5. Manipulator Crane Ventilation
6. Containment Purge System
7. Pressure and Vacuum Relief System

9.10.3.3 Containment Ventilation Subsystem Design Descriptions

9.10.3.3.1 Reactor Containment Fan Cooler (RCFC) System

The fan cooler units provide cooling so as to limit the air temperature in the closed containment to a maximum temperature of 120°F during all normal modes of operation and a minimum of 65°F during shut down conditions. The system also is designed to remove heat and particulate radioactivity from the containment as required following a loss-of-coolant accident.

The main portion of this system consists of five (5) air handling units (RCFC units) located in the space beween the containment wall and the secondary shielding (crane support wall). Each unit draws air from the containment atmosphere through a return air riser, which extends approximately 50'-0" above the operating floor.

Each unit discharges ventilation air inside the periphery of the secondary shield wall through concrete shafts designed for missile and radiation shield protection. The ventilation air is circulated first to the reactor coolant pump and the steam generator area and then flows upward above the operating floor and is mixed with air in the upper containment atmosphere.

The air flow inside each unit follows either one of two paths:

a. Normal Operation: During normal reactor operation and after reactor safe shutdown, the air is routed from the return air risers directly to the cooling coils bypassing the demister and the HEPA (high efficiency particulate air) filters, and to the fan suction plenum. The fan is operating at approximately 1200 rpm during normal operation.

b. Accident Operation: Immediately following a LOCA ventilation air is automatically rerouted to flow from the return air risers to the demister, the high efficiency particulate filters, cooling coils, and into the fan suction plenum. During accident operation, the fan speed is automatically reduced to approximately 900 rpm.

The reactor containment fan coolers are required to operate after a loss-of-coolant accident, and consequently they are designed to operate in the 47 psig containment pressure resulting from the accident. In addition, every component of each unit is capable of withstanding, without impairing operability, a pressure of 1.5 times the design pressure and the associated temperature of the air-vapor mixture (298°F) for a period of one hour. A detailed description of post accident operation is described in Chapter 6.3.

The following design criteria are common and applicable to filter assemblies, moisture eliminators, cooling coils, fans, RCFC unit housing, and connecting ductwork for each of the five air handling assemblies.

a. Normal design air flow rate is 85,000 cfm/unit, and corresponding to accident operation the flow is reduced to 53,000 cfm/unit.

b. The normal maximum environment is a dry bulb temperature of 120°F and dew point temperature of 80°F. The design maximum accident environment corresponds to a saturated steam air mixture of 271°F at 47 psig, and density of 0.175 lb/cu ft.

c. All components are capable of withstanding differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds.

d. All components and their supports are designed to meet the requirements for Class I (seismic) structures.

e. Each fan is provided with isolators to isolate the fan vibration from the other components.

In addition to the design criteria common to the components stated above, additional design criteria applicable to specific components are discussed in Chapter 6.3.

9.10.3.3.2 Containment Activated Charcoal Filter Units System

This system is provided to permit cleanup of the containment atmosphere prior to limited personnel access at power and prior to personnel access for refueling.

This system consists of two charcoal filter units located on the refueling floor in the space between the containment wall and the crane support wall equally spaced around the containment perimeter. Each unit is provided with the following components:

a. High efficiency particulate air (HEPA) filters
b. Activated charcoal filters
c. Circulating fan

All components and their supports meet the requirement for seismic Class II structures. Each unit is capable of circulating 8,000 cfm in the normal containment atmosphere conditions. Operation of these two units for approximately 32 hours will permit two hours access to the containment at full power under normal operating conditions. These units are not part of the engineered safeguards system and are not designed to operate after a loss-of-coolant accident.

9.10.3.3.3 Reactor Cavity and Out-of-Core Instrumentation Ventilation System

This system is provided to remove gamma and thermal heat from the biological shield wall around the reactor vessel and to supply ventilation to cool the out-of-core instrumentation cavities. Cooling air supplied to the reactor cavity will be drawn from outside the crane wall areas and will be discharged around the nozzles and up to the refueling pool area.

This system consists of two (2) full capacity fans located at El. 568'-0" between the containment wall and the crane wall. Either fan draws relatively cool air from the above location and the discharge is ducted to the reactor cavity where it flows into the following paths:

1. Through eight ducts into the eight out-of-core instrumentation cavities to pick up the heat from gamma radiation and thermal conduction, and then flows upward around the cable junction boxes and out to the refueling floor area.

2. Upward through the gap between the biological shield and the reactor vessel where part of the flow will escape around the sealing plate, and the balance of the air will flow out through the gaps around the eight reactor vessel nozzles.

Each fan is capable of delivering 20,000 cfm of which 2,000 cfm flows to the out-of-core instrumentation cavities (250 cfm per each cavity) and the balance flows upward around the vessel.

This system is not part of the engineered safeguards system and is not required to operate after a loss-of-coolant accident. All components and their supports meet the requirements for Class II (seismic) structures.

9.10.3.3.4 Control Rod Drive Mechanism (CRDM) Ventilation System

This system is designed to supply cooling air to the control rod drive mechanism shroud and to exhaust ventilation air which has absorbed heat within the shroud.

This system consists of four (4) 1/3 capacity control rod drive ventilation booster fans which induce air from the reactor coolant pump area where the air is relatively cool. The discharge from each fan is directed towards the top of the CRDM cooling shroud where ventilation air is drawn into the CRDM shroud.

Each fan is capable of delivering 25,000 cfm, or a total of 75,000 cfm during normal operation. These fans are not part of the engineered safeguards system and are not required to operate during reactor shut down or after a LOCA; this system will operate only during reactor operation.

Two (2) full capacity control rod drive ventilation fans, one which is normally operating, and one which is standby, induce air from the control rod drive mechanism cooling shroud and discharge it vertically to the upper atmosphere of the containment above the refueling floor. Each fan is designed to deliver 75,000 cfm.

These fans are not part of the engineered safeguards system and are not required to operate during reactor shut down or after a LOCA. All components and their supports meet the requirements for seismic Class II structures.

9.10.3.3.5 Manipulator Crane Ventilation System

This system is provided to minimize the risk of the operators working on the manipulator crane from inhaling air mixed with water vapor from the refueling water where the possibility of higher concentration of tritium exists in the immediate vicinity of the refueling water.

This system consists of two (2) 50% capacity fans mounted on the top of the manipulator crane which induce air from the containment atmosphere and discharge this air into a plenum having a perforated distribution plate. Air flows downward from the perforated plate and delivers a curtain of air above the operators standing on the crane platform. This downward flow of air will minimize the potential of the operators inhaling water vapor from the refueling water. This system operates in conjunction with the purge supply air system which delivers outdoor air around the periphery of the refueling pool.

This system is not a part of the engineered safeguards system and is designed to operate during the refueling period only. All components and their supports meet the requirements for seismic Class II structures.

9.10.3.3.6 Containment Purge System

This system is described in Section 9.10.1.1.

9.10.3.3.7 Pressure and Vacuum Relief System

This system is provided to handle the normal pressure changes in the containment which result from containment air temperature changes, barometric pressure changes, instrument air bleeds, and inleakage from the Penetration Pressurization System. The containment pressure will be maintained between -0.1 psig and +0.3 psig during normal operation. This is to facilitate personnel access during normal operation. The pressure relief line will discharge to the ventilation stack.

This system consists of one 10" line penetration through the containment with fast-acting, butterfly valves located outside the containment. These valves are normally closed and are designed to fail closed in the event of an incident or failure of control power.

When the containment pressure increases to the high set point pressure, the corresponding pressure switch sounds an alarm in the main control room. Concurrently, the corresponding diverting butterfly damper opens to line up the discharge path to the stack. As a result of the alarm (and after a containment air sample has been obtained and the appropriate release form approved), the operator will open the two isolation valves (gate type) which allows the containment air to be released to the ventilation stack. The operator will close the isolation valves when the containment pressure has equalized (reaches steady state value as indicated on the corresponding pressure indicator at the main control board). Similarly, when the containment pressure decreases to the low pressure set point, the corresponding pressure switch sounds an alarm in the main control room and concurrently the corresponding diverting butterfly damper opens to line up the intake path through the purge supply system. When the operator opens the isolation valves, tempered and filtered outside air is allowed inside the containment through the relief line to equalize the pressure. The operator will close the isolation valves when the containment pressure reaches a steady state value. | 1

In the case of vacuum in the containment, after the two pressure and vacuum relief containment isolation valves have been opened, outside air is automatically routed through the purge supply system high efficiency filters and heating coil to assure a source of filtered and heated (in winter) air delivery to the containment.

The isolation valves used in this system are designed for fail safe operation. The valves will close either on a containment isolation signal or on a high radiation level in the containment, or on loss of instrument or electrical power source.

The isolation valves are Seismic Class I, the ductwork and instrumentation are Seismic Class II.

As the system is not an engineered safety feature, single failure criterion is not incorporated.

9.10.3.3.8 Hydrogen Purge System

The hydrogen purge exhaust system consists of two separate 100% capacity fans each fed from one of two separate lines from the containment. Either fan may be fed from the main 42 inch purge line or from the pressure and vacuum relief line. The fans and lines are Seismic Class I and redundant power operated components are supplied from separate essential busses. The charcoal filter units are Seismic Class I and consist of roughing, HEPA, and charcoal filters in series, with a rated flow of 400 CFM. Two such units are provided for each Zion containment.

9.10.3.4 <u>System Components</u>

9.10.3.4.1 Reactor Containment Fan Cooler (RCFC) System

Each RCFC unit is provided with the following components:

a. Demister or moisture separator
b. High efficiency particulate air filters
c. Cooling coils
d. Two speed fan

A description of these components is given in Section 6.3.2.

9.10.3.4.2 Containment Activated Charcoal Filter Unit Systems

The containment charcoal filter unit circulating fans are of the direct-driven, vane axial type and are mounted on the outlet of the respective containment activated charcoal filter housing. Each fan is rated at 8,000 cfm at a total pressure of 2.5" H_2O and is driven by a nominal 5 hp motor. Two (2) fans are required for maximum clean-up with no spares.

The containment activated charcoal filter units are composed of banks of HEPA filters and charcoal absorbers installed in series. Each filter unit has a rated flow of 8,000 cfm. Each filter bank contains nine (9) individual HEPA elements rated at 99.97% efficiency based on the DOP test. The charcoal absorbing bank has 24 drawer-type elements each nominally rated at 8,000 cfm and having a nominal efficiency of 99% on the removal of elemental iodine at 75°F and 90% RH. Two (2) filter units are required for maximum clean-up with no spares.

9.10.3.4.3 Reactor Cavity and Out-of-Core Instrumentation Ventilation System

The reactor cavity vent fans are of the direct-driven, vane axial type and are mounted in-line in ductwork to discharge air to the reactor ventilation cavity. Each fan is rated at 20,000 cfm at a total pressure of 4.5" H_2O and is driven by a nominal 25 hp motor. Two (2) fans are provided for each containment unit as each is 100% design capacity with one (1) fan as spare.

9.10.3.4.4 Control Rod Drive Mechanism (CRDM) Ventilation System

The CRDM ventilation booster fans are of the direct-driven, vane axial type and are mounted in-line in ductwork and are arranged to discharge air toward the CRDM shroud. Each fan is rated at 25,000 cfm at a total pressure of 2.5" H_2O and is driven by a nominal 15 hp motor. Four (4) fans are provided for each containment unit and each is 1/3 design capacity with one (1) fan as spare.

The CRDM ventilation fans are of the direct-driven, vane axial type and are mounted on a plenum taking suction from the CRDM mechanism shroud. Each fan is rated at 75,000 cfm at a total pressure of 7.5" H_2O and is driven by a nominal 150 hp motor. Two (2) fans are provided for each containment unit and each is 100% design capacity with one (1) fan as spare.

9.10.3.4.5 Manipulator Crane Ventilation System

The manipulator crane ventilation fans are of the direct-driven vane axial type and are mounted to discharge air into a plenum on top of the manipulator crane bridge. Each fan is rated at 5,000 cfm at a total pressure of 2.25" H_2O and is driven by a nominal 3 hp motor. Two (2) fans are provided for each containment unit and each fan is 50% design capacity.

9.10.3.4.6 Containment Purge System

The containment purge system supply filters are located in the supply air inlet, and are composed of banks of prefilters and high efficiency filters. The filters are arranged in two (2) parallel independent modules and each module has a rated air flow of 20,000 cfm. The prefilter bank in each module contains 20 individual filter elements having a nominal efficiency of 35% based on the NBS atmospheric dust spot test. The high efficiency filter bank in each module contains 20 filter elements each having a nominal efficiency of 85% based on the NBS atmospheric dust spot test.

The containment purge system supply air preheating coil is designed to heat 40,000 cfm of air from -10°F to 68°F when supplied with 67 gpm of water at 270°F. The total coil capacity is 3.37×10^6 Btu/hr and consists of 3 finned tube sections supported and arranged for plenum mounting.

The containment purge system supply air fans are of the direct-driven, vane axial type and are mounted on the outlet of the respective supply air filter and coil plenum. Each fan is rated at 40,000 cfm at a total pressure of 6.5" H_2O and is driven by a nominal 60 hp motor. Two (2) supply air fans are provided for each containment unit, each of 100% design capacity with one (1) fan as spare. They are Seismic Class II.

The containment purge exhaust air filters are composed of banks of prefilters and HEPA filters installed in series. The filters are arranged in two (2) parallel independent modules and each module has a rated flow of 20,000 cfm. The prefilter bank in each module contains 21 individual prefilter elements having a nominal efficiency of 85% based on the NBS atmospheric dust spot test. The HEPA filter bank in each module contains 21 individual filter elements having a rated efficiency of 99.97% based on the DOP test. The arrangement of the two (2) filter modules in parallel permits replacement of elements in the module without interruption of the purge exhaust ventilation.

The containment purge system exhaust air fans are of the direct-driven, vane axial type and are mounted on the outlet of the respective exhaust air filter and coil plenum. Each fan is rated at 40,000 cfm at a total pressure of 7.5" H_2O and is driven by a nominal 75 hp motor. Two (2) exhaust air fans are provided for each containment unit, each of 100% design capacity with one (1) fan as spare. They are Seismic Class II.

In addition a containment mini-purge system is installed on both Units 1 and 2. The mini-purge system is operated as necessary to lower the containment airborne activity prior to maintenance or other needs to increase the stay time in containment. The mini-purge is used to purge the containment when the containment activity is too high to permit the use of the main containment purge system.

The mini-purge system on each unit consists of one mini-purge supply fan and one mini-purge exhaust fan, each rated at 3000 cfm. The mini-purge fans are arranged in parallel with the containment purge fans. A variable flow damper is installed on the discharge of the mini-purge exhaust fan and on the suction of the mini-purge supply fan. This variable flow damper can be manually adjusted locally to provide a variable mini-purge fan flow rate. A flowmeter is installed on the discharge of both the mini-purge exhaust fan and mini-purge supply fan. On high mini-purge exhaust flow, the mini-purge exhaust fan will automatically trip.

The containment purge supply and exhaust air ducts are each equipped with two (2) isolation valves in series.The isolation valves are Seismic Class I.

9.10.3.4.7 Pressure and Vacuum Relief System

The pressure and vacuum relief system components are described in section 1
9.10.3.3.7.

9.10.3.4.8 Hydrogen Purge System

The hydrogen purge exhaust venting fans, each of which provides a flow rate of 360 CFM, were sized based on the following assumptions:

a. The hydrogen generation rate in the containment after an LOCA is such that the hydrogen level reaches 3.0% in 55 days.

b. The purge rate was selected to match the hydrogen production rate at 55 days so that the hydrogen level will slowly decrease from 3.0% as the production rate decreases.

c. The entire quantity of hydrogen required to be removed from the containment at 55 days can be purged in a period of one hour. This allows the operator to choose the optimum purge period as a function of the prevailing meteorological conditions on each day that the purge system is operated.

The hydrogen purge exhaust charcoal filters have the following parameters:

Airflow	400 ft^3/min
Depth of Charcoal Bed	2 inches
Face velocity	40 ft/min
Efficiency	Methyl iodide removal efficiency $\geq$ 99% at 70% RH

These are the only two components unique to the Hydrogen Control Systems for the Zion Station.

9.10.3.4.9 Steam Pipe Tunnel Ventilation System

The steam pipe tunnel ventilation fans are of the direct-driven, vane axial type and are mounted in the ductwork in the respective pipe tunnel penthouse.

Each fan is rated at 27,500 cfm at a total pressure of 1.55 inches H_2O and is driven by a nominal 10 hp motor. Four (4) steam pipe tunnel ventilation fans are provided for each containment unit and each is rated at 25% design capacity.

9.10.3.4.10 Tendon Access Tunnel Ventilation System

The tendon access tunnel ventilation fans are of the direct-driven, propeller type and are mounted in closure panels within the tendon access tunnel. Each fan is rated at 300 cfm at a static pressure of 0.15" H_2O and is driven by a nominal 1/10 hp motor. Two (2) fans are required for each containment unit and each is rated at 50% design capacity.

9.10.3.5 Control and Instrumentation

Hand switches are provided on the main control panel for control of the following equipment:

a. Reactor containment fan cooler fans
b. Reactor cavity ventilation fans
c. Control rod drive ventilation fans
d. Containment charcoal filter unit fans
e. Normal flow and bypass dampers in each RCFC unit
f. Containment pressure relief line isolation valves

The following conditions are alarmed on the main control panel:

a. Containment charcoal filter unit fan trip
b. Control rod drive ventilation fan trip
c. Reactor cavity ventilation fan trip
d. Containment charcoal filter unit high temperature
e. Containment ventilation local control panel system trouble
f. Normal operation containment high/low pressure

The following indication is provided on the main control panel:

a. Normal operation containment pressure
b. RCFC fan motor current (high and low speed)
c. RCFC normal flow and bypass dampers position indicating lights
d. RCFC fan vibration indication and test
e. Containment atmosphere dewpoint temperature (3 points)
f. Containment atmosphere dry bulb temperature (3 points)

The following variables are measured and inputted to the computer:

a. Out of core neutron monitor temperature (8 points)
b. RCFC return air temperature (5 points)
c. Control rod drive booster fan discharge temperature (4 points)
d. Discharge temperature around reactor vessel nozzles (4 points)
e. Reactor cavity air inlet temperature (1 point)
f. Control rod drive shroud air inlet temperature (1 point)
g. Control rod drive shroud air outlet temperature (1 point)

9.10.3.6 Inspection and Tests

All equipment was factory inspected and tested in accordance with the applicable equipment specifications. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, each system was balanced for the design air and water flows. Controls on each system was checked, adjusted and tested to insure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

9.10.4 AUXILIARY ELECTRIC EQUIPMENT ROOM AND COMPUTER ROOM HEATING, VENTILATING AND AIR CONDITIONING SYSTEM

9.10.4.1 System Design

The auxiliary electric equipment room and computer room heating, ventilating and air conditioning system including ductwork to the electric equipment room is designed to seismic Class 1. Equipment for this system is connected to the emergency electrical system. The design consists of two, 50% equipment trains. Two (2) systems will operate normally and one (1) system is shutdown on a loss of off-site power. This system may be manually restarted by the operator when emergency power is available. All of the electric equipment required for maintaining heating, ventilating and air conditioning to the auxiliary electric equipment room is connected to the emergency power supply.

The auxiliary electric equipment room is normally unoccupied, and therefore, there is no special provision to provide outside air make-up during post LOCA or loss of off-site power operation.

The design of the heating, ventilating and air conditioning for the auxiliary electric equipment room is based on a system with the capability to provide air filtration, heating and/or cooling, and humidification and/or dehumidification under any normal or abnormal condition to permit a safe shutdown of the plant as may be required.

Interconnected areas which are normally served by this system (such as computer room, offices, cold and process laboratories, etc.) are shut off from the supply of conditioned air after a loss of off-site power. Therefore, a source of cooling is maintained only to the auxiliary electric equipment room under this condition.

Each HVAC equipment train shown on Figure 9.10.4-1 consists of high efficiency filters, a cooling and heating coil package, humidifier, and a direct-driven vane axial fan. Air is supplied via a dual duct distribution system and room or zone control is achieved by means of terminal mixing boxes.

Return air is ducted back to the HVAC equipment room (El 617'-0") where it is mixed with outside air as required to meet the normal minimum ventilation requirements and/or room cooling requirements.

During normal operation, a minimum quantity of outside air is supplied to maintain a positive operating pressure in the conditioned areas with respect to adjacent areas to prevent inleakage. The system will be controlled to maintain a nominal 75°F and 40% RH.

The outside air supply will be brought in through a missile-protected wall opening. In the event of abnormal outside radiation detection, or loss of off-site power. the normal outside air inlet will close and the system will operate on a 100% recirculation basis. Under these conditions, the filtered make-up supply on the Control Room HVAC System (See Section 9.10.2) will provide positive contamination control in the areas served by the auxiliary electric equipment room system as a result of leakage from the control room through the auxiliary electric equipment room.

In the event of smoke or fire in the auxiliary electric equipment room, smoke detectors will annunciate in the control room.

Mechanical cooling for the auxiliary electric equipment room and computer room system is provided by means of two (2), 50% water-cooled refrigeration units. Each refrigeration unit will be connected to its respective air handling unit. Each air handling unit is also rated at 50% of the total design cooling capacity.

Each air conditioning equipment train is independently controlled. Important operating functions are performed from the main control room.

Instrumentation is provided to monitor important normal operating variables. Instruments to alarm abnormal operating conditions are provided.

With regard to radiological exposure, all of the facilities discussed above are structurally and physically arranged within the control room complex. This complex is protected at the outer boundries by concrete shielding which reduces radiation exposure to less than the design limit of 1/2 rem/8 hours subsequent to a LOCA. This shield wall includes the heating, ventilating and cooling equipment room serving the subject areas. Missile protected openings are provided into the latter controlled area; one for the make-up air intake, and one for the exhaust air. By including the equipment room within the controlled area, access may be gained to the equipment under any condition.

9.10.4.2 <u>System Operation</u>

9.10.4.2.1 Normal Operation

Each auxiliary electric equipment room and computer room HVAC equipment train is controlled to provide heating and/or cooling on a year-round basis. Two (2), 50% capacity air handling units are provided, one of which is shut down on loss of off-site power. In the event of failure of one equipment train during normal operation. the remaining operating unit functions to maintain cooling to the auxiliary electric equipment room. Cooling to the cable rooms is shut off under these conditions.

Each air handling unit is connected to a 50% capacity refrigeration unit. During normal operation a minimum amount of outdoor air is introduced to the system for the purpose of odor and smoke (cigarette) dilution, exhaust air make-up and system pressurizing. Return air from the conditioned spaces is directed back through two (2), 50% capacity return fans and is recirculated through the air handling units or is exhausted to the outdoors as conditions dictate. Each system is provided with separate controls.

9.10.4.2.2 Abnormal Operation

The auxiliary electric equipment room is normally unoccupied, but is designed to accommodate an occupancy of three to five people. The cable room, computer room and other interconnecting areas served by this system are automatically isolated from the conditioned air supply for circumstances previously described. The electric equipment room HVAC system has two (2), 50% HVAC equipment trains together with adequate shielding and which is independent from the HVAC system for the control room. All HVAC equipment for the electric equipment room has the necessary standby capacity to accommodate an equipment failure and is powered from the emergency electrical system so that adequate ventilation is maintained at all times.

A radiation monitor in the outside air intake is set to sense an abnormal level of activity. On an increase in activity approximately 100 times above background, the HVAC system automatically closes a damper in the make-up air duct from the outdoors. Under these conditions, the HVAC system will operate on a 100% recirculation basis. Leakage of make-up air supplied to the control room system is sufficient to satisfy personnel requirements and space pressurization. In all other respects, the system operates as for normal operation.

In the event of a loss of off-site power, the HVAC system will operate on a 100% recirculation basis. In addition, one of the two HVAC equipment trains is shut down and the system operates to maintain cooling to the electric equipment rooms only, and cooling to the computer room and cable room is automatically shut down.

9.10.4.3 System Components

9.10.4.3.1 Supply Air Filters

The system supply air filters are composed of banks of prefilters and high efficiency filters installed in series. Each filter unit has a rated flow of 22,800 cfm. Each prefilter bank contains 12 individual filter elements rated at 35% efficiency based on the NBS atmospheric dust spot test. Each high efficiency filter bank contains 12 individual filter elements rated at 85% efficiency based on the NBS atmospheric dust spot test.

9.10.4.3.2 Air Handling Units

The air handling units consist of cooling and heating coils and cold deck and hot deck arranged in a housing with an interconnecting direct-driven vane axial fan arranged for blow-through operation. Each air handling unit is 50% design capacity. The heating coil is designed to heat 17,000 cfm of air from 66.5°F to 90.5°F when supplied with 25 gpm of water at 210°F. The cooling coil is designed to cool 22,800 cfm of air from 80.6°F DB and 62.5°F WB to 50°F DB and 40°F WB when supplied with refrigerant at 43°F. The total coil capacity is 872,000 Btu/hr. The air handling units supply air fans are of the direct-driven vane axial type and are mounted on the inlet of each air handling unit coil cabinet. Each fan is rated at 22,800 cfm at a total pressure of 5.3" H_2O and is driven by a nominal 25 HP motor.

The return air fans are of the direct-driven vane axial type and are mounted in line in the return ductwork. Each fan is rated at 22,800 cfm at a total pressure of 3" H_2O and is driven by a nominal 15 HP motor. Two fans are provided, each is design capacity. | 1

9.10.4.3.3 Refrigeration Condensing Units

The refrigeration condensing units are of the packaged, reciprocating water cooled type. Each condensing unit is interconnected to its respective air handling unit. Each condensing unit will develop 73 tons (875,000 Btu/hr) of cooling capacity corresponding to a suction temperature of 53°F and a condensing temperature of 105°F.

The condenser on each unit is water cooled and will dissipate 1,210,000 Btu/hr of heating when supplied with 220 gpm of water supplied at 80°F. Two (2) condensing units are provided for the system and each is 50% design capacity.

9.10.4.3.4 Exhaust Air Fans

The exhaust air fans are of the direct-driven centrifugal type. Each fan is rated at 4050 cfm at a static pressure of 6" and is driven by a nominal 75 HP motor. Two fans are provided, each is 100% design capacity with one (1) fan as spare.

9.10.4.3.5 Hot Laboratory Exhaust Filter

The hot laboratory exhaust filter is located in the duct between the hot laboratory hood and the exhaust fans. The filter is composed of prefilters and HEPA filters installed in series. The filter unit has a rated flow of 1500 cfm. Each prefilter bank contains two (2) individual filter elements rated at 35% efficiency based on the NBS atmospheric dust spot test. Each HEPA filter bank contains two (2) filter elements each having a nominal efficiency of 99.7% based on the DOP test.

9.10.4.4 Inspection and Tests

All equipment was factory inspected and tested in accordance with the applicable equipment specifications. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, the system was balanced for the design air and water flows. Controls on each system were checked, adjusted, and tested to ensure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

9.10.5 SERVICE WATER PUMP ROOM VENTILATION

9.10.5.1 Design Basis

The service water pump ventilation system is designed to limit the maximum ambient temperature to 105°F and minimum ambient temperature to 65°F on a year-round basis.

9.10.5.2 System Design and Operation

The service water pump ventilation system is shown on Figure 9.10.8-1 and consists of six (6), 25% capacity fans for both Units 1 and 2. Whenever the out-door temperature is above 40°F, four (4) of six (6) fans operate to mix out-door air with recirculated air as required to limit the maximum pump room ambient temperature to 105°F. Below 40°F, only two (2) of six (6) fans operate as decribed above.

This system is connected to the essential power supply and is designed to operate after a LOCA and/or loss of off-site power. All components and their supports are designed to meet the requirements for Class I (seismic) structures.

9.10.5.3 System Components

9.10.5.3.1 Circulating Water Pump Area Ventilation

Nine (9) propeller-type supply air fans are provided. Each fan is 1/9 capacity (no standby) and is rated for 40,000 cfm at a static pressure of 1" H_2O and is driven by a nominal 7.5 hp motor.

9.10.5.3.2 Service Water Pump Area Ventilation

Two (2) sets of three (3) 25% capacity vane axial fans, each rated for 37,500 cfm at a total pressure of 2.5" H_2O, are provided and are driven by a nominal 20 hp motor.

9.10.5.4 Inspection and Tests

All equipment was factory inspected and tested in accordance with the applicable equipment specifications. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, each of the systems was balanced for the design air and water flows. Controls on each system were checked, adjusted, and tested to ensure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

9.10.6 DIESEL GENERATOR AREA VENTILATION

9.10.6.1 Design Basis

The diesel generator building ventilation systems are designed to maintain a thermal environment in each of the equipment areas in the diesel generator building, and/or to provide the air change rate as required to ventilate the space.

9.10.6.2 System Design and Operation

9.10.6.2.1 Diesel Oil Room Ventilation System

Each diesel oil room ventilation system is designed to introduce air from the turbine building to the diesel oil rooms and to exhaust this air to the outdoors. The rate of purge is designed to prevent the accumulation of flammable fumes.

The diesel oil room ventilation system for Unit 1 as shown on Figure 9.10.6-1 consists of two (2) full capacity exhaust fans, each capable of exhausting 6,000 cfm total from three oil rooms. One fan normally operates and one is standby. A similar system is provided for Unit 2. Air is drawn from the turbine room through openings protected by fire dampers. The exhaust duct penetration leaving each room is also protected by a fire damper.

This system is not part of the engineered safeguards system and is not required to operate after a LOCA. All components and their supports meet the requirements for Class I (seismic) structures.

9.10.6.2.2 Diesel Generator Room Ventilation System

Each diesel generator room ventilation system is designed to introduce outside air to the room to limit the maximum room ambient to 115°F and to supply the diesel engine with the necessary combustion air. Exhaust air is relieved to the turbine building.

An independent ventilation system is provided for each diesel generator room. Each system as shown on Figure 9.10.6-1 has one fan with a design capacity of 70,000 cfm which induces outside air through a missile protected outside air plenum. Each inlet to the fan is provided with a fire damper and each fan has a variable inlet vane to modulate the amount of ventilation required. The fan variable inlet vane is controlled by a proportional room thermostat. Exhaust air from each room is relieved to the turbine building through a relief damper protected by a fire damper.

This system is part of the engineered safeguards and is required to operate for all loss of off-site power conditions. Each ventilation system starts automatically and controls room temperature whenever the respective diesel generator starts. System operation is interlocked with the CO_2 fire protection system and will automatically shut down whenever the fire protection system is activated. All components and their supports meet the requirements for Class I (seismic) structures.

9.10.6.2.3 Switch Gear and M-G Set Ventilation System

Each switch gear or M-G set ventilation system is designed to introduce air to the respective room to limit the maximum room ambient temperature to 105°F. Exhaust air is relieved to the turbine building.

Switch Gear Rooms El. 617' Ventilation System -

Each ventilation system as shown on Figure 9.10.6-1 consists of one fan for each switch gear room at El. 617'-0" having a capacity of 10,500 cfm. Each fan induces outside air through the same missile protected inlet for the diesel generator room. Exhaust air is relieved to the turbine building. Penetrations for ventilation in each room are protected by fire dampers.

This system will be required to operate after a LOCA or loss of off-site power. All components and their supports meet the requirements for Class I (seismic) structures.

Switch Gear Rooms El. 642' Ventilation System -

Each ventilation system as shown on Figure 9.10.6-1 consists of two (2) 50% capacity fans and each is capable of delivering 6,000 cfm. One system is provided for Unit 1 and one system is provided for Unit 2. This system is not part of the engineered safeguards and will not be required to operate after a LOCA or loss of off-site power. All components and their supports meet the requirements for Class II (seismic) structure.

In normal operation two (2) fans operate. In the event of fan failure. partial ventilation capacity can be maintained. Outside air is induced by the fans and discharged to the switch gear rooms. All exhaust air is relieved to the turbine building. All wall penetrations used for ventilation openings are protected by fire dampers.

Each ventilation system as shown on Figure 9.10.6-1 consists of one fan having a capacity of 12,500 cfm. One system is provided for Unit 1 and one system is provided for Unit 2.

This system is not part of the engineered safeguards and will not be required to operate after a LOCA or loss of off-site power. All components and their supports meet the requirements for Class II (seismic) structures.

MG Set Room Vent Systems

In normal operation, the 100% capacity fan operates continuously - no standby is provided. Outside air is induced by the fan and discharged to the MG set room. All exhaust air is relieved to the turbine building, and all wall penetrations used for ventilation openings are protected by fire dampers.

9.10.6.3 System Components

9.10.6.3.1 Main Turbine Building Area Vent System

All inlet and outlet windows have pneumatic operators.

9.10.6.3.2 Fuel Oil Room Exhaust System

Two (2), 100% capacity vane axial fans per unit (Equipment Nos. 1TV012-1A, 1TV013-1B, 2TV012-2A, 2TV013-2B), each fan rated at 6000 cfm at a total pressure of 2.25" H_2O and driven by a nominal 3 hp motor.

9.10.6.3.3 Diesel Generator Room Vent System

One (1) vane axial fan per each room (Equipment Nos. 1TV009, 1TV010, 0TV011, 2TV009, 2TV010), each fan rated at 70,000 cfm at a total pressure of 3" H_2O and driven by a nominal 50 hp motor.

9.10.6.3.4 Switch Gear Room Elevation 617' Vent System

One (1) vane axial fan per each room (Equipment Nos. 1TV001, 1TV002, 1TV003, 2TV001, 2TV002, 2TV003), each fan rated for 10,500 cfm at a total pressure of 2.25" H_2O and driven by a nominal 5 hp motor.

9.10.6.3.5 Switch Gear Room Elevation 642' Vent System

Two (2), 50% capacity vane axial fans per each room (Equipment Nos. 1TV004, 1TV005, 2TV004, 2TV005). Each fan is rated for 6,000 cfm at a total pressure of 2.25" H_2O and is driven by a nominal 3 hp motor.

9.10.6.3.6 M-G Set Room Vent System

One (1) vane axial fan per each room (Equipment Nos. 1TV006, 2TV006). Each fan is rated at 12,500 cfm at a total pressure of 2.5" H_2O and is driven by a nominal 7.5 hp motor.

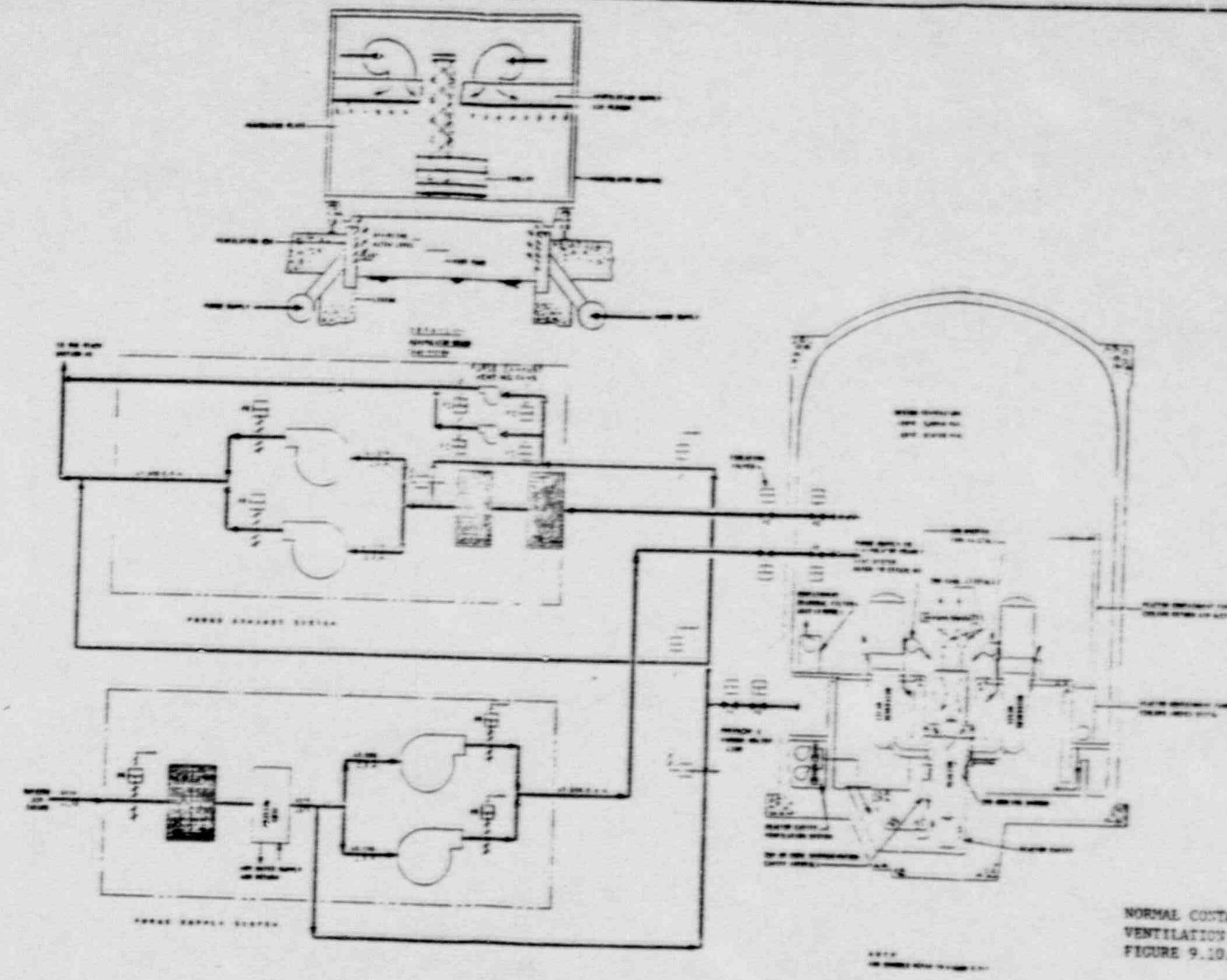
9.10.6.4 Tests and Inspections

All equipment was factory inspected and tested in accordance with the applicable equipment specifications. System ductwork and erection of equipment was inspected in accordance with the respective specifications. On completion of construction tests, each of the systems was balanced for the design air and water flows. Controls on each system were checked, adjusted and tested to ensure the proper sequence of operation under all normal and abnormal conditions. A final integrated test was conducted with all equipment and controls operational to verify that system performance and operation met all design requirements.

TABLE 9.10.2-1

CUBICLE COOLERS HEAT REMOVAL CAPACITIES

	Number of Units	Heat Transfer Capability (Btu/hr/unit)	Number of Fans	Fan Motor HP
Residual Heat Removal Pump Rooms	4	120,000	3	1
Safety Injection Pump Rooms	4	120,000	3	1
Containment Spray Pump Rooms	4	330,000	4	3
Reciprocating Charging Pump Rooms	2	60,000	2	1.5
Centrifugal Charging Pump Rooms	4	180,000	3	3



NORMAL CONTAINMENT
VENTILATION SYSTEM
FIGURE 9.10.3-1

Enclosure 6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 1 8 1989

MEMORANDUM FOR: Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Systems Technology

FROM: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

SUBJECT: DIFFERING PROFESSIONAL VIEW (DPV)

References:
1. Memorandum, Licciardo to Murley, dtd May 11, 1989
2. Memorandum Murley to Miraglia, et al, dtd May 26, 1989
3. Memorandum and Report from DPV Standing Review Panel to Murley dtd August 31, 1989

On May 11, 1989 (Reference 1), you sent me a Differing Professional View regarding containment isolation valves (42" purge supply and exhaust valves) at Zion. Pursuant to NRC Manual Chapter 4125, and NRR Office Letter No. 300, I established a Panel to review your Differing Professional View and make recommendations to me regarding appropriate disposition of your concerns (Reference 2).

I have reviewed the Panel's report and have discussed these issues with you in a meeting in my office on September 8, 1989. Based on my review of the Panel's report and on our discussions, I have concluded that the following actions should be taken:

1. The staff should issue its evaluation of the proposed Zion Technical Specifications. (Action--Projects, A/D for Region III & V Reactors)

2. The staff should ensure that pressure and temperature effects during a LOCA are considered in the review of new and advanced fuel designs. (Action--Reactor Systems Branch)

3. The staff should revise the Standard Review Plan to clarify the relationship between DNBR and fuel failure. Such clarifications may be made during the normal SRP update process. (Action--Inspection and Licensing Program Branch)

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure: Memorandum and Report
from Panel dtd 8/31/89

cc: F. Miraglia
F. Gillespie
J. Partlow

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 31, 1989

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: DPV Standing Review Panel

SUBJECT: DIFFERING PROFESSIONAL VIEW (DPV) CONCERNING ZION BY ROBERT LICCIARDO

The subject DPV was submitted to you on May 11, 1989 (Enclosure 1). The DPV was handled in accordance with NRR Office Letter No. 300, Revision 1, and NRC Manual Chapter 4125. On May 26, 1989, the Standing Review Panel of Frank J. Miraglia, C. E. Rossi and Frank J. Congel was established to review the DPV.

This memorandum summarizes the activities of the Panel and provides our recommendation regarding the subject DPV.

On June 12, 1989, the Panel met with Ashok Thadani and J. Wermeil regarding the subject DPV. The Panel requested that (1) copies of all references to staff criteria in the DPV be provided, (2) the results of an Appendix K LOCA analysis for Zion be provided, and (3) the staff opinion on the safety significance of the DPV for Zion, and to other power plants, be provided. Mr. Wermeil responded to the Panel (Reference 1).

On June 16, 1989, the Panel met with Mr. Licciardo. Mr. Licciardo provided the Panel with background material (Reference 2). Based on that meeting, Mr. Licciardo's concern regarding calculation of allowable closure times for containment purge valves was primarily based on a belief that fuel rods would rupture early in a LOCA induced accident and that entry into DNBR also occurred early, and thereby significant fission product inventory would be present in the containment in less than one second. When these results are coupled with conservative radiological dose models in the SRP's, large radiological consequences are projected.

The Panel requested Mr. Licciardo to provide clarification of his position. Mr. Licciardo provided a response to the Panel on July 20, 1989 (Reference 3).

On July 27, 1989 the Panel requested the staff to provide the following information: (1) the temperature and pressure effects experienced by fuel early in a LOCA event, and (2) why entry into DNBR does not result in fuel failure. The staff responded on August 11, 1989 (References 4 and 5). Mr. Licciardo indicated that these References did not appropriately address his concerns. The Panel requested the staff to re-examine their response. On August 29, 1989 the staff reaffirmed their original views (Reference 6).

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Based upon our review of the subject DPV, and reference material, the Panel concludes the following:

1. Test data provide reasonable assurance that fuel clad integrity will be maintained for more than 7-15 seconds into a LOCA event for current fuel designs. (Advanced fuel designs may need further evaluation.)

2. Entry into DNBR is not equated to fuel failure. (Clarification in the SRP's would be helpful.)

3. The proposed Zion License Amendment on containment purge valve operation can be issued based on the staff safety evaluation.

The Panel recommends that:

1. The staff evaluation of the proposed Zion Technical Specifications be issued.

2. The Reactor Systems Branch be requested to review new and advanced fuel designs to assure that pressure and temperature effects during a LOCA are considered.

3. Revision of the SRP's not be undertaken in view of resource restraints.

In accordance with NRR Office Letter No. 300, Revision 1, copies of the enclosed list of memoranda and references are in the official Office file being maintained by Chief, Planning, Program and Management Support Branch, PMAS.

The Standing Review Panel is prepared to brief you on the subject matter if you desire.

Frank J. Congel

Frank J. Congel

Charles E. Rossi

Charles E. Rossi

Frank J. Miraglia, Jr.

Frank J. Miraglia, Jr.

Enclosures:
As stated

cc: R. Licciardo
J. Larkins

List of References

1. Management Response to Oversight Committee Regarding DPV of R. Licciardo dated May 11, 1989.

2. Background Information Related to Differing Professional View.

3. Memorandum to F. Miraglia from R. Licciardo dated July 20, 1989 enclosing "An Evaluation of the Criteria for and the Calculation of Offsite Doses Deriving from Open Containment Purge Valves During a LOCA at Zion Units 1 and 2."

4. Note to F. Miraglia from A. Thadani dated August 11, 1989, subj: "DPV Concerning Containment Isolation Valves at Zion."

5. Note to F. Miraglia from A. Thadani dated August 24, 1989, subj: "DPV Concerning Containment Isolation Valves at Zion."

6. Note to F. Miraglia from A. Thadani dated August 29, 1989, subj: "DPV Concerning Containment Isolation Valves at Zion."

Ref 1

Management Response to Oversight Committee
Regarding DPV of R. Licciardo dated May 11, 1989

1. Provide copies of the references to staff criteria included in the above DPV. Indicate DEST management view on their applicability to the issue.

Response:

Mr. Licciardo refers to three staff criteria documents as the basis for his alternative dose calculation with the containment purge valves open. These are SRP Section 6.2.4 (specifically BTP CSB 6-4), SRP Section 4.2, and Regulatory Guide 1.77. These are attached.

a. SRP Section 6.2.4 "Containment Isolation System," BTP CSB 6-4 "Containment Purging During Normal Plant Operations"

BTP CSB 6-4 provides the applicable staff guidelines for use of the containment vent/purge valves during power operation and specifically identifies the need to perform an analysis to ensure that radiological consequences for a loss-of-coolant accident occurring at the time the purge valves are open will be within 10 CFR Part 100 limits. It states (page 6.2.4-15, Position B.5.a):

"An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values."

In his DPV, Mr. Licciardo refers only to the third sentence in the above paragraph. He does not employ the above guidance fully which indicates consideration of a pre-existing iodine spike. Instantaneous release of fission products from projected failed fuel ignores that transport that must take place, i.e., release from fuel into the primary system, release to the containment, and subsequent release from the containment. The use of the spiked coolant activity specified by the SRP 6.4 BTP was intended to bound the maximum activity that could exist in the coolant at the onset of the LOCA. As an alternative, Mr. Licciardo refers to SRP Section 4.2 and Regulatory Guide 1.77.

b. SRP Section 4.2 "Fuel System Design"

SRP Section 4.2 provides the staff guidelines for analyses to ensure acceptable fuel performance (limited damage, maintaining coolability, and ensuring control rod insertion). It applies to normal operation, anticipated operational occurrences and postulated accidents. It does not, however, apply to the design basis LOCA. 10 CFR 50.46

[illegible] [illegible] / [illegible]
[illegible] [illegible], [illegible] [illegible] [illegible] [illegible]
[illegible] [illegible] [illegible]

criteria are employed when evaluating fuel performance following a LOCA. The indicated use in the DPV is "SRP 4.2 identifies fuel failure with infringement of DNBR criteria, with related requirement that gap activity be considered as part of the source term,...." By satisfying the requirements of 10 CFR 50.46, Zion assures negligible fuel damage per GDC 35 for a LOCA.

c. <u>Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"</u>

RG 1.77 identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in a PWR. These assumptions were not intended to be used for a LOCA evaluation. The DPV refers to the guidance in Appendix B of Regulatory Guide 1.77, "Radiological Assumptions," which states that "the amount of activity accumulated in the fuel clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated..." For the rod ejection accident, limited, localized, DNB caused, fuel failure is assumed (typically 10% of fuel pins) and the source term as specified in Appendix B is assumed to be instantaneously dumped and uniformly mixed into the primary coolant. The rod ejection accident results in releases to the environment through two paths: leakage from the primary vessel to the containment and subsequent leakage from the containment, and second, through primary-to-secondary leakage in the steam generators. While the rod ejection transient itself is rapid (within 2 seconds) releases of fission products through these two paths is assumed to occur over a period of several hours and the rod ejection accident assumptions are intended to bound the expected rod ejection doses. Because of the assumed accident duration for the rod ejection accident is several hours, the assumptions used in the evaluation of the rod ejection accident obviously ignore any transport time for fission products. This is not the case for the purge contribution to the LOCA dose. The timing of the valve closure (15 seconds or less) is very important to limiting the releases and, as stated in item a (above), a pre-existing iodine spike (one which was the result of fission product activity existing in the fuel at the time of the LOCA and not the result of subsequent LOCA fuel failures) was used to bound the expected dose consequences.

2. Provide the results of an Appendix K LOCA analysis which indicates when the onset of fuel failure occurs.

Response:

Attachment 1 is a copy of the ECCS Analysis for Zion from the updated FSAR. It gives the results of the LOCA analysis (per 10 CFR 50.46 and Appendix K) for a spectrum of breaks. Note that in no case does fuel failure, "hot rod burst" occur before 34.8 seconds.

3. What is the staff opinion on the safety significance of this issue for Zion and generically.

Response:

a. The safety significance of this issue for Zion specifically is as follows:

   1) By imposition of more restrictive technical specification surveillance requirements for the purge valve closure time from 60 seconds to 7 seconds, potential radiological releases are reduced. While there is some probability of failure of the redundant valves in series to close, the staff views it to be sufficiently unlikely, concurrent with a LOCA to require continuous purge valve closure at power. In spite of this, some restrictions are imposed on the allowable hours of purge valve operation.

   2) As indicated in the staff guidance, use of the purge valves is intended to be minimized, however, purging is necessary for relief of containment pressure due to air leakage from pneumatic controllers, and reducing airborne activity levels to facilitate containment access. The detrimental effects that these problems could have on equipment operability (e.g., ability to do maintenance while at power) is outweighed by the negligible decrease in offsite release probability resulting from continuous purge valve closure.

   3) The DPV unreasonably assumes instantaneous (within ½ second) fuel failure and transport of the resulting gap activity to the site boundary before the 7 second purge valve closure time. The LOCA analysis (Attachment 1) indicates that the purge valves would be closed long before fuel failure would occur (approximately 34 seconds). Additional time is needed to transport the release to the purge line opening.

The staff concludes, therefore, that the concerns in the DPV are not safety significant and do not justify a change in staff position.

b. The generic safety significance of this issue is similar to the above discussion for Zion. While there are plant-specific differences in purge valve closure time and time to fuel failure following a LOCA, the staff believes significant margin exists and the probability of an unacceptable release is very small.

Document Name:
MGMT RESP. TO OVERSIGHT COMMIT

Requestor's ID:
FAIRCLOT

Author's Name:
j. wermiel

Document Comments:
resp. to oversight committee re; DPV of R. Licciardo of 5/11

TABLE 14.3.2-1

LARGE BREAK - 1984 ANALYSIS - SEQUENCE OF EVENTS

	DECLG C_D = 0.4	DECLG C_D = 0.6	DECLG C_D = 0.8	* DECLG C_D = 0.6
Start	0.0	0.0	0.0	0.0
Rx Trip Signal	0.747	0.737	0.732	0.737
S.I. Signal	1.89	1.52	1.34	1.52
Accumulater Injection	20.10	15.00	12.80	15.00
Pump Injection	26.89	26.52	26.34	26.52
End of Blowdown	38.99	30.14	30.86	30.14
End of Bypass	38.99	30.14	30.86	30.14
Bottom of Core Recovery	53.75	43.34	45.17	43.34
Accumulators Empty	66.67	60.37	58.88	60.33

Note: All times in seconds

* With Replacement Reactor Containment Fan Coolers as installed 1985

Attachment 1

ECCS Analysis from Zion Updated FSAR

Revision 1
June 26, 1986

0411T
0189A

TABLE 14.3.2-2

LARGE BREAK RESULTS - 1984 ANALYSIS - USING MODIFIED 1981 MODEL (With BART)

Results	DECLG C_D = 0.4	DECLG C_D = 0.6	DECLG C_D = 0.8	* DECLG C_D = 0.6
Peak Clad Temperature (°F)	1986	2016	1983	2159
Peak Clad Temp. Elevation (ft.)	6.0	6.0	5.5	6.0
Max Local Zr/H_2O Reaction (%)	4.584	4.182	4.065	6.94
Max Local Zr/H_2O Rxn Elevation (ft.)	6.0	6.0	5.5	6.0
Total Zr/H_2O Reaction (%)	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Elevation (ft.)	6.0	6.0	5.5	6.0
Hot Rod Burst Time (sec.)	42.8	35.2	34.8	34.8

Inputs

NSSS Power - 102% of	= 3250 MWt
Peak Linear Power - 102% of	= 15.575 Kw/ft.
Local Peaking Factor (at licensed rating)	= 2.32
Accumulator Water Volume	= 888 ft³/tank
Steam Generator Tube Plugging Level	= 10% (uniform)

* With replacement RCFC and corrected data transfer methodology between WREFLOOD and BART

Note: Values for DECLG C_D = .4 and .8 reflect original RCFC
DECLG C_D = .6 case reanalyzed with new RCFC because this represents the limiting case

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TABLE 14.3.2-4

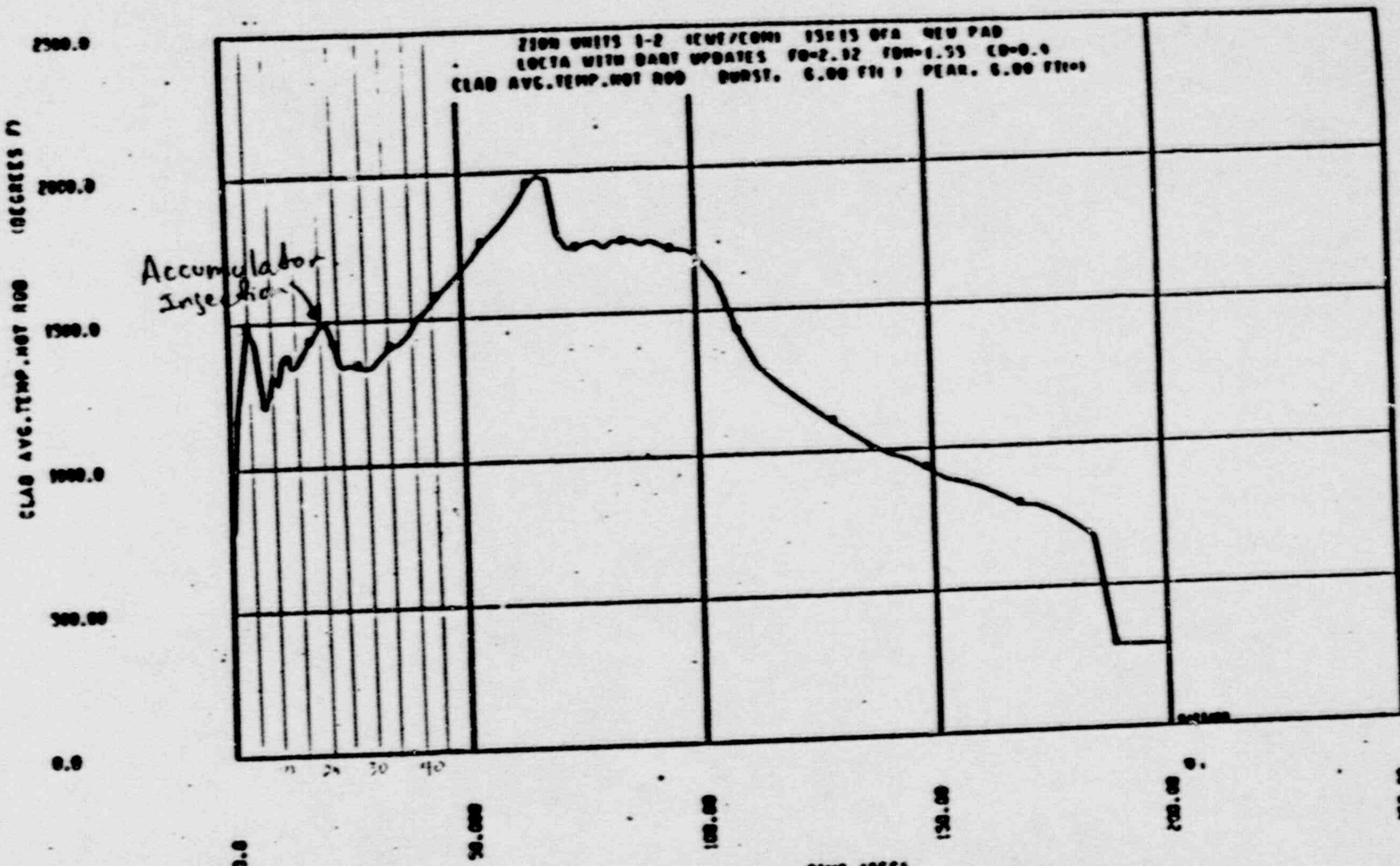
SMALL BREAK RESULTS - 1984 ANALYSIS - 6" BREAK CASE

<u>Results</u>

Peak Clad Temperature	=	1747 °F
Peak Clad Temp. Elevation	=	10.75 ft.
Max Local Zr/H_2O Reaction	=	1.45%
Max Local Zr/H_2O Rxn Elevation	=	11.00 ft.
Total Zr/H_2O Reaction	=	<0.3
Hot Rod Burst Elevation	=	11.00 ft.
Hot Rod Burst Time	=	313.59 sec.

<u>Inputs</u>

Core Power - 102% of ESDR	=	3390 MWt
Peak Linear Power - 102% of	=	See Figure 14.3.2-7
Accumulator Water Volume	=	900 ft^3.



PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

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Figure 14.3.2-4G

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

6.2.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - None

I. AREAS OF REVIEW

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. This SRP section, therefore, is concerned with the isolation of fluid systems which penetrate the containment boundary, including the design and testing requirements for isolation barriers and actuators. Isolation barriers include valves, closed piping systems, and blind flanges.

The CSB review of the applicant's safety analysis report (SAR) regarding containment isolation provisions covers the following aspects:

1. The design of containment isolation provisions, including:

   a. The number and location of isolation valves, i.e., the isolation valve arrangements and the physical location of isolation valves with respect to the containment.

   b. The actuation and control features for isolation valves.

   c. The positions of isolation valves for normal plant operating conditions (including shutdown) postaccident conditions, and in the event of valve operator power failures.

   d. The valve actuation signals.

   e. The basis for selection of closure times of isolation valves.

   f. The mechanical redundancy of isolation devices.

g. The acceptability of closed piping systems inside containment as isolation barriers.

2. The protection provided for containment isolation provisions against loss of function of missiles, pipe whip, and earthquakes.

3. The environmental conditions inside and outside the containment that were considered in the design of isolation barriers.

4. The design criteria applied to isolation barriers and piping.

5. The provisions for detecting a possible need to isolate remote-manual-controlled systems, such as engineered safety features systems.

6. The design provisions for and technical specifications pertaining to operability and leakage rate testing of the isolation barriers.

7. The calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs.

CSB will coordinate other branch evaluations that interface with the overall review of the containment isolation system, as follows: The Mechanical Engineering Branch (MEB) will review the system seismic design and quality group classification as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2, respectively. The Structural Engineering Branch (SEB) and the MEB will review the mechanical and structural design of the containment isolation system as part of their primary review responsibilities for SRP Sections 3.8 and 3.9, respectively, to ensure adequate protection against a breach of integrity, missiles, pipe whip, jet impingement and earthquakes. The Instrumentation and Control Systems Branch (ICSB), as part of its primary responsibility for SRP Section 7.5, will evaluate the actuation and control features for isolation valves. The Equipment Qualification Branch (EQB), as part of its primary review responsibility for SRP Sections 3.10 and 3.11, will evaluate the qualification test program for electric valve operators, and sensing and actuation instrumentation of the plant protection system located both inside and outside of containment; and the operability assurance program for containment isolation valves. The Accident Evaluation Branch (AEB), as part of its primary review responsibility for SRP Section 15.6.5, will review the radiological dose consequence analysis for the release of containment atmosphere prior to closure of containment isolation valves in lines that provide a direct path to the environs. The Reactor Systems Branch (RSB), as part of its primary review responsibilities for SRP Section 15.6.5, will review the closure time for containment isolation valves in lines that provide a direct path to the environs, with respect to the prediction of onset of accident-induced fuel failure. The review of proposed technical specifications, at the operating license stage of review, pertaining to operability and leakage rate testing of the isolation barriers, and the closure time for containment isolation valves, is performed by the Licensing Guidance Branch (LGB), as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The CSB will accept the containment isolation system design if the relevant requirements of General Design Criteria 1, 2, 4, 16, 54, 55, 56, and 57 and Appendix K to 10 CFR Part 50 are met. The relevant requirements are as follows:

1. General Design Criteria 1, 2, and 4 as they relate to systems important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed; systems being designed to withstand the effects of natural phenomena (e.g., earthquakes) without loss of capability to perform their safety functions; and systems being designed to accommodate postulated environmental conditions and protected against dynamic effects (e.g., missiles, pipe whip, and jet impingement), respectively.

2. General Design Criterion 16 as it relates to a system, in concert with the reactor containment, being provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment.

3. General Design Criterion 54, as it relates to piping systems penetrating the containment being provided with leak detection, isolation, and containment capabilities having redundant and reliable performance capabilities, and as it relates to design provision incorporated to permit periodic operability testing of the containment isolation system, and leak rate testing of isolation valves.

4. General Design Criteria 55 and 56 as it relates to lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or connect directly to the containment atmosphere being provided with isolation valves as follows:

   a. One locked closed isolation valve[1] inside and one locked closed isolation valve outside containment; or

   b. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

   c. One locked closed isolation valve inside and one automatic isolation valve[2] outside containment; or

   d. One automatic isolation valve inside and one automatic isolation valve[2] outside containment.

5. General Design Criterion 57 as it relates to lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere being provided with at least one locked closed, remote-manual, or automatic isolation valve[2] outside containment.

[1]Locked closed isolation valves are defined as sealed closed barriers (see Item II.3.f).
[2]A simple check valve is not normally an acceptable automatic isolation valve for this application.

6. Appendix K to 10 CFR Part 50 as it relates to the determination of the extent of fuel failure (source term) used in the radiological calculations.

The General Design Criteria identified above established requirements for the design, testing, and functional performance of isolation barriers in lines penetrating the primary containment boundary and, in general, required that two isolation in series be used to assure that the isolation function is maintained assuming any single active failure in the containment isolation provisions. However, containment isolation provisions that differ from the explicit requirements of General Design Criteria 55 and 56 are acceptable if the basis for the difference is justified.

Specific criteria necessary to meet the relevant requirements of the regulations identified above and guidelines for acceptable alternate containment isolation provisions for certain classes of lines are as follows:

a. Regulatory Guide 1.11 describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines that are closed both inside and outside containment, are designed to withstand the pressure and temperature conditions following a loss-of-coolant accident, and are designed to withstand dynamic effects, are acceptable without isolation valves.

b. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems may include remote-manual valves, but provisions should be made to detect possible leakage from these lines outside containment.

c. Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but provisions should be made to detect possible leakage from these lines outside containment.

d. Containment isolation provisions for lines in the systems identified in items b and c normally consist of one isolation valve inside, and one isolation valve outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve arrangement, the valve nearest the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP Section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

e. Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I standards, classified Safety

Class 2 (Ref. 9), and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested, unless it can be shown that the system integrity is being maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in a leak tight or controlled leakage housing. If, in lieu of a housing, conservative design of the piping and valve is assumed to preclude a breach of piping integrity, the design should conform to the requirements of SRP Section 3.6.2. Design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and terminate the leakage.

f. Sealed closed barriers may be used in place of automatic isolation valves. Sealed closed barriers include blind flanges and sealed closed isolation valves which may be closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed isolation valves should be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator.

g. Relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure.

h. Item II.E.4.2 of NUREG-0737 and NUREG-0718 requires that systems penetrating the containment be classified as either essential or nonessential. Regulatory Guide 1.141 will contain guidance on the classification of essential and nonessential systems. Essential systems, such as those described in items b and c, may include remote-manual containment isolation valves, but provisions should be made to detect possible leakage from the lines outside containment. Item II.E.4.2 of NUREG-0737 and NUREG-0718 also requires that nonessential systems be automatically isolated by the containment isolation signal.

i. Isolation valves outside contaiment should be located as close to the containment as practical, as required by General Design Criteria 55, 56, and 57.

j. In meeting the requirements of General Design Criteria 55 and 56, upon loss of actuating power, automatic isolation valves should take the position that provides greater safety. The position of an isolation valve for normal and shutdown plant operating conditions and postaccident conditions depends on the fluid system function. If a fluid system does not have a postaccident function, the isolation valves in the lines should be automatically closed. For engineered safety features or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. The position of an isolation valve in the event of power failure to the valve operator should be the "safe" position. Normally this position would be the postaccident valve position. For lines equipped with motor-operated valves, a loss of actuating power will leave the affected valve in the "as is" position, which may be the open position; however, redundant isolation barriers assure that the isolation function for the line is satisfied. All power operated isolation valves should have position indication in the main control room.

k. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, Item II.E.4.2 of NUREG-0737 and NUREG-0718 requires that the containment setpoint pressure that initiates containment isolation for nonessential penetrations be reduced to the minimum value compatible with normal operating conditions.

l. There should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the requirement of General Design Criterion 54 for reliable isolation capability.

m. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, system lines which provide an open path from the containment to the environs (e.g., purge and vent lines which are addressed in Item II.E.4.2 of NUREG-0737 and NUREG-0718) should be equipped with radiation monitors that are capable of isolating these lines upon a high radiation signal. A high radiation signal should not be considered one of the diverse containment isolation parameters.

n. In meeting the requirements of General Design Criterion 54 the performance capability of the isolation function should reflect the importance to safety of isolating system lines. Consequently, containment isolation valve closure times should be selected to assure rapid isolation of the containment following postulated accidents. The valve closure time is the time it takes for a power operated valve to be in the fully closed position after the actuator power has reached the operator assembly; it does not include the time to reach actuation signal setpoints or instrument delay times, which should be considered in determining the overall time to close a valve. System design capabilities should be considered in establishing valve closure times. For lines which provide an open path from the contaiment to the environs; e.g., the containment purge and vent lines, isolation valve closure times on the order of 5 seconds or less may be necessary. The closure times of these valves should be established on the basis of minimizing the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and assure that emergency core cooling system (ECCS) effectiveness is not degraded by a reduction in the containment backpressure. Analyses of the radiological consequences and the effect on the containment backpressure due to the release of containment atmosphere should be provided to justify the selected valve closure time. Additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby and hot shutdown) is provided in Branch Technical Position CSB 6-4 (Ref. 13). For plants under review for operating licenses or plants for which the Safety Evaluation Report for construction permit application was issued prior to July 1, 1975, the methods described in Section B, Items B.1.a, b, d, e, g, f, and g, B.2 through B.4, and B.5.b, c, and d of Branch Technical Position CSB 6-4 should be implemented. For these plants, BTP Items B.1.c and B.5.a, regarding the size of the purge system used during normal plant operation and the justification by acceptable dose consequence analysis, may be waived if the applicant commits to limit the use of the purge system to less than 90 hours per year while the plant is in the startup, power, hot standby and hot shutdown modes of operations. This commitment should be incorporated into the Technical Specifications used in the operation of the plant.

Item II.E.4.2 of NUREG-0737 and NUREG-0718 requires that containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP Section 6.2.4, Item II.3.f during operational conditions 1, 2, 3 and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position appears as Attachment 1 to Item II.E.4.2 in NUREG-0737.)

o. The use of a closed system inside containment as one of the isolation barriers will be acceptable if the design of the closed system satisfies the following requirements:

1. The system does not communicate with either the reactor coolant system or the containment atmosphere.

2. The system is protected against missiles and pipe whip.

3. The system is designated seismic Category I.

4. The system is classified Safety Class 2 (Ref. 12).

5. The system is designed to withstand temperatures at least equal to the containment design temperature.

6. The system is designed to withstand the external pressure from the containment structure acceptance test.

7. The system is designed to withstand the loss-of-coolant accident transient and environment.

Insofar as CSB is concerned with the structural design of containment internal structures and piping systems, the protection of isolation barriers against loss of function from missiles, pipe whip, and earthquakes will be acceptable if isolation barriers are located behind missiles barriers, pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment, and the isolation barriers, including the piping between isolation valves, are designated seismic Category I, i.e., designed to withstand the effects of the safe shutdown earthquake, as recommended by Regulatory Guide 1.29.

p. In meeting the requirements of General Design Criteria 1, 2, 4 and 54, appropriate reliability and performance considerations should be included in the design of isolation barriers to reflect the importance to safety of assuring their integrity; i.e., containment capability, under accident conditions. The design criteria applied to components performing a containment isolation function, including the isolation barriers and the piping between them, or the piping between the containment and the outermost isolation barrier, are acceptable if:

1. Group B quality standards, as defined in Regulatory Guide 1.26 are applied to the components, unless the service function dictates that Group A quality standards be applied.

2. The components are designated seismic Category I, in accordance with Regulatory Guide 1.29.

q. General Design Criterion 54 requires reliable isolation capability. Therefore, when considering remote manual isolation valves, the design of the containment isolation system is acceptable if provisions are made to allow the operator in the main control room to know when to isolate fluid systems that are equipped with remote manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.

r. General Design Criterion 54 specifies the requirements for the containment isolation system. Therefore, to satisfy General Design Criterion 54, provisions should be made in the design of the containment isolation system for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. The acceptance criteria for the leakage rate testing program for containment isolation barriers are presented in SRP Section 6.2.6.

s. General Design Criterion 54 requires reliable isolation capability. To satisfy this requirement, provisions should be made in the design of the containment isolation system to reduce the possibility of isolation valves reopening inadvertently following isolation. In this regard, Item II.E.4.2 of NUREG-0737 and NUREG-0718 requires that the design of the control systems for automatic containment isolation valves be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves should require deliberate operator action. In addition, ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criterion continue to be satisfied.

Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this design requirement.

III. REVIEW PROCEDURES

The procedures described below provide guidance on review of the containment isolation system. The reviewer selects and emphasizes material from the review procedures as may be appropriate for a particular case. Portions of the review may be done on a generic basis for aspects of containment isolation common to a class of containments, or by adopting the results of previous reviews of plants with essentially the same containment isolation provisions.

Upon request from the primary reviewer, other review branches will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

The CSB determines the acceptability of the containment isolation system by comparing the system design criteria to the design requirements for an engineered safety feature. The quality standards and the seismic design classification of the containment isolation provisions including the piping penetrating the containment, are compared to Regulatory Guides 1.26 and 1.29, respectively.

The CSB also ascertains that no single fault can prevent isolation of the containment. This is accomplished by reviewing the containment isolation provisions for each line penetrating the containment to determine that two isolation barriers in series are provided, and in conjunction with the PSB by reviewing the power sources to the valve operators.

The CSB reviews the information in the SAR justifying containment isolation provisions which differ from the explicit requirements of General Design Criteria 55, 56, and 57. The CSB judges the acceptability of these containment isolation provisions based on a comparison with the acceptance criteria given in subsection II of this SRP section.

The CSB reviews the position of isolation valves for normal and shutdown plant operating conditions, postaccident conditions, and valve operator power failure conditions as listed in the SAR. The position of an isolation valve for each of the above conditions depends on the system function. In general, power-operated valves in fluid systems which do not have a postaccident safety function (nonessential systems, as defined in Regulatory Guide 1.141) should close automatically. In the event of power failure to a valve operator, the valve position should be the position of greater safety, which is normally the postaccident position. However, special cases may arise and these will be considered on an individual basis in determining the acceptability of the prescribed valve positions. The CSB also ascertains from the SAR that all power-operated isolation valves have position indication capability in the main control room.

The CSB reviews the signals obtained from the plant protection system to initiate containment isolation. In general, there should be a diversity of parameters sensed; e.g., abnormal conditions in the reactor coolant system, the secondary coolant system, and the containment, which generate containment isolation signals. Since plant designs differ in this regard and many different combinations of signals from the plant protection system are used to initiate containment isolation, the CSB considers the arrangement proposed on an individual basis in determining the overall acceptability of the containment isolation signals. The CSB will use the guidance presented in Item II.E.4.2 of NUREG-0737 for its review of the containment setpoint pressure that initiates containment isolation for nonessential penetrations. This pressure setpoint should be the minimum value that is compatible with normal operating conditions.

The CSB reviews isolation valve closure times. In general, valve closure times should be less than one minute, regardless of valve size. (See the acceptance criteria for valve closure times in subsection II of this SRP section.) Valves in lines that provide a direct path to the environs, e.g., the containment purge and ventilation system lines and main steam lines for direct cycle plants, may have to close in times much shorter than one minute. Closure times for these valves may be dictated by radiological dose analyses or ECCS performance considerations. The CSB will request the AEB or RSB to review analyses justifying valve closure times for these valves as necessary.

The CSB determines the acceptability of the use of closed systems inside containment as isolation barriers by comparing the system designs to the acceptance criteria specified in subsection II of this SRP section.

The MEB and SEB have review responsibility for the structural design of the containment internal structures and piping systems, including restraints, to assure that the containment isolation provisions are adequately protected against missiles, pipe whip, and earthquakes. The CSB determines that for all containment isolation provisions, missile protection and protection against loss of function from pipe whip and earthquakes were design considerations. The CSB reviews the system drawings (which should show the locations of missile barriers relative to the containment isolation provisions) to determine that the isolation provisions are protected from missiles. The CSB also reviews the design criteria applied to the containment isolation provisions to determine that protection against dynamic effects, such as pipe whip and earthquakes, was considered in the design. The CSB will request the MEB to review the design adequacy of piping and valves for which conservative design is assumed to preclude possible breach of system integrity in lieu of providing a leak tight housing.

Systems having a postaccident safety function (essential systems, as defined in Regulatory Guide 1.141) may have remote-manual isolation valves in the lines penetrating the containment. The CSB reviews the provisions made to detect leakage from these lines outside containment and to allow the operator in the main control room to isolate the system train should leakage occur. Leakage detection provisions may include instrumentation for measuring system flow rates, or the pressure, temperature, radiation, or water level in areas outside the containment such as valve rooms or engineered safeguards areas. The CSB bases its acceptance of the leakage detection provisions described in the SAR on the capability to detect leakage and identify the lines that should be isolated.

The CSB determines that the containment isolation provisions are designed to allow the isolation barriers to be individually leak tested. This information should be tabulated in the safety analysis report to facilitate the CSB review.

The CSB determines from the descriptive information in the SAR that provisions have been made in the design of the containment isolation system to allow periodic operability testing of the power-operated isolation valves and the containment isolation system. At the operating license stage of review, the CSB determines that the content and intent of proposed technical specifications pertaining to operability and leak testing of containment isolation equipment is in agreement with requirements developed by the staff.

The CSB verifies that the design of the control system for automatic containment isolation valves is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves, and that ganged reopening of isolation valves is not possible.

IV. EVALUATION FINDINGS

The information provided and the CSB review should support concluding statements similar to the following, to be included in the staff's safety evaluation report:

> The staff concludes that the containment functional design is acceptable and meets the requirements of General Design Criteria 1, 2, 4, 16, 54, 55, 56, and 57 and Appendix K to 10 CFR Part 50. The conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]
>
> 1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation)

by (for each item that is applicable to the review state how it was met and why acceptable with respect to the regulation being discussed):

a. meeting the regulatory positions in NUREG _______ and/or Regulatory Guide(s) ____;

b. providing and meeting an alternative method to regulatory positions in Regulatory Guide _____, that the staff has reviewed and found to be acceptable;

c. meeting the regulatory position in BTP________;

d. using calculational methods for (state what was evaluated) that have been previously reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

e. meeting the provisions of (industry standard number and title) that have been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff plans for using this SRP section.

Except in those cases in which the applicant proposes as acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Basis."

4. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

5. 10 CFR Part 50, Appendix A, General Design Criterion 54, "Piping Systems Penetrating Containment."

6. 10 CFR Part 50, Appendix A, General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

7. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

8. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

9. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

10. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

11. Regulatory Guide 1.29, "Seismic Design Classification."

12. Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems."

13. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation," attached to this SRP section.

14. 10 CFR Part 100, "Reactor Site Criteria."

15. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

16. NUREG-0737, "Classifications of TMI Action Plan Requirements."

17. NUREG-0718, "Licensing Requirements for Pending Application for Construction Permits and Manufacturing License."

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the lines associated with the containment purge and vent systems. It supplements the position taken in SRP Section 6.2.4.

While the containment purge and vent systems provide plant operational flexibility, their designs must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on their use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously. There is similar disparity in the need for, and use of, containment vent systems at operating plants.

Containment purge systems have been used in a variety of ways; for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. Containment vent systems are typically used to relieve the initial containment pressure buildup caused by the heat load imposed on the containment atmosphere during reactor power ascension, or to periodically relieve the pressure buildup due to the operation of pneumatic controllers. However, the purge and vent lines provide an open path from the contaiment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR Part 100 guidelines values.

The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the contaiment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines.

The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR Part 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Similar concerns apply to vent system designs.

The size of the purge lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR Part 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR, or HTGR.

B. BRANCH TECHNICAL POSITION

The systems used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. General Design Criterion 54 requires that the reliability and performance capabilities of containment isolation valves reflect the importance of safety of isolating the systems penetrating the containment boundary. Therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in Branch Technical Position MEB-2, "Pump and Valve Operability Assurance Program." (Also see SRP Section 3.10.) The design basis for the valves and actuators should include the build-up of containment pressure for the LOCA break spectrum, and the supply line and exhaust line flows as a function of time up to and during valve closure.

   b. The number of supply and exhaust lines that may be used should be limited to one supply line and one exhaust line, to improve the reliability of the isolation function as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in the evaluation of the emergency core cooling system effectiveness and 10 CFR Part 100 regarding offsite radiological consequences.

   c. The size of the lines should not exceed about eight inches in diameter, unless detailed justification for larger line sizes is provided, to improve the reliability and performance capability of the isolation and containment functions as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in evaluating the emergency core cooling system effectiveness and 10 CFR Part 100 regarding the offsite radiological consequences.

d. As required by General Design Criterion 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria, to reflect the importance to safety of isolating these lines. General Design Criterion 56 establishes explicit requirements for isolation barriers in purge system lines.

e. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. Furthermore, if energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can effect the isolation function.

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds, to facilitate compliance with 10 CFR Part 100 regarding offsite radiological consequences.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values.

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters, and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The maximum allowable leak rate of the purge isolation valves should be specified on a case-by-case basis giving appropriate consideration to valve size, maximum allowable leakage rate for the containment (as defined in Appendix J to 10 CFR Part 50), and where appropriate, the maximum allowable bypass leakage fraction for dual containments.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 1), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents. "Coolability," in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, §50.46 (Ref. 3).

All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in SRP Section 4.4. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Evaluation Branch for use in estimating the radiological consequences of plant releases.

The fuel system review covers the following specific areas.

A. Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that damage will be limited to acceptable levels. The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. Design Evaluation

The performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents is reviewed to determine if all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. On-line fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by CPB to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

Specific criteria necessary to meet the requirements of 10 CFR Part 50, §50.46; General Design Criteria 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 identified in subsection I of this SRP section are as follows:

A. Design Bases

The fuel system design bases must reflect the four objectives described in subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

1. Fuel System Damage

This subsection applies to normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Such damage criteria should address the following to be complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the ASME Code (Ref. 4) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles (Ref. 5). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the Safety Analysis Report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

2. Fuel Rod Failure

This subsection applies to normal operation, anticipated operational occurrences, and postulated accidents. Paragraphs (a) through (c) address failure mechanisms that are more limiting during normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report. Paragraphs (d) through (h) address failure mechanisms that are more limiting during anticipated operational occurrences and postulated accidents, and the information to be reviewed will usually be contained in Chapter 15 of the Safety Analysis Report. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere.

To meet the requirements of (a) General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, and (b) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Fuel failure criteria should address the following to be complete.

(a) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 7) for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in

Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown (Ref. 8) to be insufficient for primary hydride formation.

(b) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The review of these criteria is detailed in SRP Section 4.4. For normal operation and anticipated operational occurrences, violation of the thermal margin criteria is not permitted. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes.

Although a thermal margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: It has also been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and anticipated operational occurrences, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

(f) Excess[illegible] Fuel Enthalpy: For a severe reactivity initiated acciden[illegible] [illegible]IA) in a BWR at zero or low power, fuel failure is assumed to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. For full-power RIAs in a BWR and all RIAs in a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 (Ref. 6) as it relates to fuel rod failure. The 170 cal/g enthalpy criterion is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions (PCI). Other criteria may be more appropriate for an RIA, but continued approval of this enthalpy criterion and the thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

(h) Bursting: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to the incidence of rupture during a LOCA, a rupture temperature correlation must be used in the LOCA ECCS analysis. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one described in Ref. 10.

(i) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.

3. Fuel Coolability

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addresses the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Control rod insertability criteria are also addressed in this subsection. Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, §50.46, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800°F previously described in Reference 11 is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of Appendix K of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data (such as Refs. 10, 12, and 13) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 2200°F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

B. Description and Design Drawings

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

Type and metallurgical state of the cladding
Cladding outside diameter
Cladding inside diameter
Cladding inside roughness
Pellet outside diameter
Pellet roughness
Pellet density
Pellet resintering data
Pellet length
Pellet dish dimensions
Burnable poison content
Insulator pellet parameters
Fuel column length
Overall rod length
Rod internal void volume
Fill gas type and pressure
Sorbed gas composition and content
Spring and plug dimensions
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawing have also been found necessary for an acceptable fuel system description:

Fuel assembly cross section
Fuel assembly outline
Fuel rod schematic
Spacer grid cross section
Guide tube and nozzle joint
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Burnable poison rod assembly cross section
Burnable poison rod assembly outline
Burnable poison rod schematic
Orifice and source assembly outline

C. Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed. Those methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the Safety Analysis Report by reference.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed when practical to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested prior to irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

Spacer grid structural tests
Control rod structural and performance tests
Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
Fuel assembly growth
Fuel assembly bowing
Channel box wear and distortion
Fuel rod ridging (PCI)
Crud formation
Fuel rod integrity
Holddown spring relaxation
Spacer grid spring relaxation
Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished prior to operation of a full core of that design. This inability to perform in-reactor testing may result from an incompatability of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see subsection II.D below).

3. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient*

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 14 and 15). Examples of previous fuel performance code reviews are given in References 16 through 20.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core

*Although needed in fuel performance codes, this model is reviewed as described in SRP Section 4.4.

power distributions (power spiking, see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR, see SRP Section 4.4), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in Reference 21 and in Regulatory Guide 1.126 (Ref. 22). To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References 23 and 24.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 25. Interim methods that may be used prior to compliance with this guidance are given in Reference 26. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References 10, 12, and 13.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate, the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27). For non-LOCA applications, other correlations may be used if justified.

(h) Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 (Refs. 6, 28-30) as they relate to fission product release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by CPB (see Ref. 31).

D. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

1. Testing and Inspection of New Fuel

Testing and inspection plans for new fuel should include verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Details of the manufacturer's testing and inspection programs should be documented in quality control reports, which should be referenced and summarized in the Safety Analysis Report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. Where the overall testing and inspection programs are essentially the same as for previously approved plants, a statement to that effect should be made. In that case, the details of the programs need not be included in the Safety Analysis Report, but an appropriate reference should be cited and a (tabular) summary should be presented.

2. On-line Fuel System Monitoring

The applicant's on-line fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. References 32 and 33 evaluate several common detection methods and should be utilized in this review.

Surveillance is also needed to assure that B_4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as described in Reference 34 are acceptable.

3. Post-irradiation Surveillance

A post-irradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered, i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features.

For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, or crud deposition. There should also be a commitment in the program to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, there should exist a continuing fuel surveillance effort for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with prototype testing discussed in subsection II.C.2. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

III. <u>REVIEW PROCEDURES</u>

For construction permit (CP) applications, the review should assure that the design bases set forth in the Preliminary Safety Analysis Report (PSAR) meet the acceptance criteria given in subsection II.A. The CP review should further determine from a study of the preliminary fuel system design that there is reasonable assurance that the final fuel system design will meet the design bases. This judgment may be based on experience with similar designs.

For operating license (OL) applications, the review should confirm that the design bases set forth in the Final Safety Analysis Report (FSAR) meet the acceptance criteria given in Subsection II.A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have reviewed the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review is dependent should be referenced so that a completely documented safety evaluation is contained in the plant safety evaluation report. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Certain generic reviews have also been performed by CPB reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References 9, 11, 21, 31, 32, 35, and 36, and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (Refs. 6, 22, 28-30, and 41) should also be mentioned in the plant safety evaluation reports. Deviation from these guides or positions should be explained. After briefly discussing related previous reviews, the plant safety evaluation should concentrate on areas where the application is not identical to previously reviewed and approved applications and areas related to newly discovered problems.

Analytical predictions discussed in Subsection II.C.3 will be reviewed in PSARs, FSARs, or licensing topical reports. When the methods are being reviewed, calculations by the staff may be performed to verify the adequacy of the analytical methods. Thereafter, audit calculations will not usually be performed to check the results of an approved method that has been submitted in a Safety Analysis Report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time the clear need arises to reconfirm the adequacy of the method.

IV. <u>EVALUATION FINDINGS</u>

The reviewer should verify that sufficient information has been provided to satisfy the requirements of this SRP section and that the evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

The staff concludes that the fuel system of the ________________ plant has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents and thereby meets the related requirements of 10 CFR Part 50, §50.46; 10 CFR Part 50, Appendix A, General Design Criteria 10, 27 and 35; 10 CFR Part 50, Appendix K; and 10 CFR Part 100. This conclusion is based on the following:

1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection (PWR) or drop (BWR), and fuel densification have been performed in accordance with (a) the guidelines of Regulatory Guides 1.60, 1.77, and 1.126, or methods that the staff has reviewed and found to be acceptable alternatives to those Regulatory Guides, and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A to SRP Section 4.2.

2. The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100. In meeting these requirements, the applicant has (a) used the fission-product release assumptions of Regulatory Guides 1.3 (or 1.4), 1.25, and 1.77 and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77 or with methods that the staff has reviewed and found to be an acceptable alternative to Regulatory Guide 1.77.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

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7. "Standard Specification for Sintered Uranium Dioxide Pellets," ASTM Standard C776-76, Part 45, 1977.

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22. Regulatory Guide 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification."

23. Memorandum from V. Stello, NRC, to R. C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 14, 1975.

24. Memorandum from D. F. Ross, NRC, to R. C. DeYoung, Subject: CEPAN -- Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.

25. Memorandum from D. F. Ross, NRC, to D. B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

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U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

APPENDIX A

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE
TO EXTERNALLY APPLIED FORCES
TO
STANDARD REVIEW PLAN SECTION 4.2

A. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This Appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. Background material for this Appendix is given in References 37-40.

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system and reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch. Input for the fuel assembly response to a LOCA should include (a) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures; these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis, and (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the SSE; if a linear response is produced, a spectral analysis may be used in accordance with the guidelines of Regulatory Guide 1.60 (Ref. 41).

2. Methods

Analytical methods used in performing structural response analyses should be reviewed. Justification should be supplied to show that the numerical solution techniques are appropriate.

Linear and non-linear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical.

A sample problem of a simplified nature should be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (Ref. 38). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), simplifying assumptions may be made (e.g., one might use a 3-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are probably conservative and input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-convervatisms, two precautions should be taken: (a) If it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased (by about 30%) to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis. (b) Conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for ±10% variations in input amplitude and frequency; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15%. For example, if ±10% variations in input magnitude or frequency produce a maximum resultant increase of 35%, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant-dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

Independent audit calculations for a typical full-sized core should be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (Ref. 38) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of General Design Criterion 2 as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly components should be calculated for each input (i.e., seismic and LOCA) as described above in Paragraph 1, and the resulting loads should be added by the square-root-of-sum-of-squares (SRSS) method. These combined loads should be compared with the component strengths described in Section C according to the acceptance criteria in Section D.

C. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the most damaging mode should be represented in the vendor's laboratory grid strength tests. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (a) the testing impact velocities correspond to expected fuel assembly velocities, and (b) the crushing load P(crit) has been suitably selected from the load-vs-deflection curves. Because of the potential for different test rigs to introduce measurement variations, an evaluation of the grid strength test equipment will be included as part of the review of the test procedure.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P(crit) should be the 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit).

2. Components Other than Grids

Strengths of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95%) of the distribution of component strengths. Therefore, ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects.

D. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR Part 50, §50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisifed if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit) as defined above, then significant deformation of the fuel assembly would not occur and control rod insertion would not be interfered with by lateral displacement of the guide tubes. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability: (a) combined loads on the channel box must remain below the allowable value defined above for components other than grids; otherwise, additional analysis is needed to show that deformation is not severe enough to prevent control blade insertion, and (b) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe Shutdown Earthquake

Two criteria apply for the SSE: (a) fuel rod fragmentation must not occur as a result of the seismic loads, and (b) control rod insertability must be assured. The first criterion is satisfied by the criteria in Paragraph 1. The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1.

U.S. ATOMIC ENERGY COMMISSION

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REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.77

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

A. INTRODUCTION

Section 50.34, "Contents of applications: technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each application for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the potential risk to public health and safety resulting from operation of the facility. General Design Criterion 28, "Reactivity Limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. General Design Criterion 28 also requires that these postulated reactivity accidents include consideration of the rod ejection accident unless such an accident is prevented by positive means.

This guide identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide-fueled pressurized water reactors (PWRs). In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered on an individual basis. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The rate at which reactivity can be inserted into the core of a uranium oxide-fueled water-cooled power reactor is normally limited by the design of the control rod system to a value well below that which would result in serious damage to the reactor system. However, a postulated failure of the control rod system provides the potential for a relatively high rate of reactivity insertion which, if large enough, could cause a prompt power burst. For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant. This is accompanied by the conversion of nuclear energy, deposited as overpower heat in the fuel and in the coolant, to mechanical energy which, in sufficient quantity, could conceivably disarrange the reactor core or breach the primary system.

The Regulatory staff has reviewed the available experimental information concerning fuel failure thresholds. In general, failure consequences for UO_2 have been insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod as a result of a postulated rod ejection accident provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired.

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position.

A sufficient number of initial reactor states to completely bracket all possible operational conditions of interest should be analyzed to assure examination of upper bounds on ultimate damage. In areas of uncertainty, the appropriate minimum or maximum parameters relative to nominal or expected values should be used to assure a conservative evaluation. The initial reactor states should include consideration of at least the following:

Zero power (hot standby)	– Beginning of Life (BOL) and End of Life (EOL);
Low power	– BOL and EOL;
Full power	– BOL and EOL.

The effects of the loss of primary system integrity as a result of the failed control rod housing should be included in the analysis. It should also be shown that failure of one control rod housing will not lead to failure of other control rod housings.

The approach that should be used in the radiological analysis of a control rod ejection accident is to determine the amount of each gaseous radionuclide released to the primary containment and, with this information in conjunction with the procedures set forth in Appendix B of this guide, to determine the radiological consequences of this accident for a pressurized water reactor.

C. REGULATORY POSITION

Acceptable assumptions and evaluation models for analyzing a rod ejection accident in PWRs are presented in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumptions) of this guide. By use of these appendices, it should be shown that:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod.

2. Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.[1]

3. Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, New York 10017.

APPENDIX A

PHYSICS AND THERMAL-HYDRAULICS

The assumptions described below should be applied in evaluating the physics and thermal-hydraulic behavior of the reactor system for a control rod ejection accident.

1. The ejected rod worth should be calculated based on the maximum worth rod resulting from the following conditions: (a) all control banks at positions corresponding to values for maximum allowable bank insertions at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed by operating procedures. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod in each rod group for different control rod configurations, both expected and unexpected. The worth of single rods in rod groups should be evaluated during startup physics tests and compared with values used in the rod ejection analysis. The accident should be reanalyzed if the rod worths used in the initial analysis are found to be nonconservative. Calculated rod worths should be increased, if necessary, to account for calculational uncertainties in parameters such as neutron cross sections and power asymmetries due to xenon oscillations.

2. The reactivity insertion rate due to an ejected rod should be determined from differential control rod worth curves and calculated transient rod position versus time curves. If differential rod worth curves are not available for the reactor state of interest, conservatism should be included in the calculation of reactivity insertion through consideration of the nonlinearity in reactivity addition as the rod passes through the active core. The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

3. The calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*) should be based on the well-known definitions resulting from perturbation theory, such as those described by Henry (Ref. 1), using available experimental delayed neutron data and averaging by the fraction of fission in the various fissionable materials. In cases where the accident is quite sensitive to β_{eff} (where the ejected rod worth $\geq \beta_{eff}$), the minimum calculated value for the given reactor state should be used. For smaller transients, conservatism in the value should include consideration of not only the initial power rise (which increases with decreasing β), but also the power reduction after the trip. Similar considerations should also be applied to determine an appropriately conservative value of ℓ^* to be used.

4. The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen with respect to their influence on the magnitude of the transient. Pressure and temperature are mainly significant with respect to their effect on the amount of reactivity inserted if there exists a positive moderator coefficient.

5. The fuel thermal properties such as fuel-clad gap heat transfer coefficient and fuel thermal conductivity should be conservatively chosen, depending upon the transient phenomenon being investigated. For conditions of a zero or positive moderator coefficient (usually at beginning of life), for example, high heat transfer parameters would reduce the Doppler feedback and increase any positive moderator feedback effects and hence tend to increase the magnitude of the reactivity transient. For a negative moderator coefficient, high heat transfer parameters could cause the magnitude of the transient to decrease if a given quantity of heat produces more feedback in the moderator than in the fuel. In the consideration of pressure pulses which may be generated, high moderator heating rates could cause significant pressure gradients to develop in the moderator channels. In computing the average enthalpy of the hottest fuel pellet during the excursion for power cases, low heat transfer would be conservative.

6. The specific heat of UO_2 has been determined experimentally and is a deterministic factor in the calculated amount of stored energy (enthalpy) in the fuel. Recommended values in the range of 25 to 902°C are the data reported by Moore and Kelley (Ref. 2). In the range of 900 to 2842°C, the data obtained by Hein and Flagella (Ref. 3), Leibowitz, Mishler, and Chasanov (Ref. 4), and Chasanov (Ref. 5) are recommended for the heat capacity of the fuel. These recommended values are for clean core conditions. Possible variation in the specific heat due to burnup should be investigated and appropriate values used, if necessary.

7. The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If no three-dimensional space-time kinetics calculation is performed, the reactivity feedback due to these coefficients should be conservatively weighted to account for the variation in their spatial importance in the missing dimension(s). If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed.

8. The Doppler coefficient should be calculated based on the effective resonance integrals and should include corrections for pin shadowing (Dancoff correction). Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively

with available experimental data such as those of Hellstrand (Ref. 6). Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied value of the Doppler coefficient. If no three-dimensional space-time kinetics calculation is performed, the reactivity effect of spatially weighting the core average temperature rise in both the axial and radial directions should be calculated.

9. Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. If the rod worth curve (reactivity vs. depth of insertion) is not obtained from a "true" representation (i.e., an x, y, z, t or an r, z, t calculation), the conservatism of the approximate calculation should be shown. The difference in the depth of insertion at zero power and at full power should be accounted for in calculating the available scram reactivity.

10. The reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (a) time required for instrument channel to produce a signal, (b) time for the trip breaker to open, (c) time for the coil to release the rods, and (d) time required before scram rods enter the core if the tips lie above the core-reflector interface.

11. The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (a) incorporation of all major reactivity feedback mechanisms, (b) at least six delayed neutron groups, (c) both axial and radial segmentation of the fuel element, (d) coolant flow provision, and (e) control rod scram initiation on either coolant system pressure or neutron flux.

12. The analytical models and computer codes used should be documented and justified and the conservatism of the models and codes should be evaluated both by comparison with experiment, as available, and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

13. The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

14. The number of fuel rods experiencing clad failure should be calculated and used to obtain the amount of contained fission product inventory released to the reactor coolant system. It should be assumed that clad failure occurs if the heat flux equals or exceeds the value corresponding to the onset of the transition from nucleate to film boiling (DNB), or for other appropriate causes.

The margin to DNB is expressed in terms of a departure from nucleate boiling ratio (DNBR). The DNBR at any position in the hottest channel is the ratio of the DNB heat flux to the actual heat flux. The DNB heat flux should be evaluated using correlations based on recognized studies and experimental heat transfer DNB data. A minimum DNBR should be determined from the evaluation of the experimental data to ensure a 95% probability with a 95% confidence level that DNB has not occurred for the fuel element being evaluated. One example of a correlation which has been used to date is given by Tong (Ref. 7). The use of this correlation and the above probabilities and confidence level yields a minimum DNBR of 1.30. Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data.

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1. A. F. Henry, "Computation of Parameters Appearing in the Reactor Kinetics Equations," WAPD-142, December 1955.

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3. R. A. Hein and P. N. Flagella, "Enthalpy Measurements of UO_2 and Tungsten to 3260°K," GEMP-578, February 1968.

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APPENDIX B

RADIOLOGICAL ASSUMPTIONS

The assumptions given below should be applied in determining a conservative source term and subsequent transport of activity and resulting doses to the public for use in evaluating the radiological consequences of a control rod ejection accident.

1. The assumptions related to the release of radioactive material to the primary containment are as follows:

a. The case resulting in the largest source term should be selected for evaluation.

b. The nuclide inventory in the fuel elements potentially breached should be calculated, and it should be assumed that all gaseous constituents in the fuel-clad gaps are released.

c. The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life, assuming continuous maximum full power operation.

d. No allowance should be given for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

e. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842°C) at any time during the course of the accident should be calculated, and 100% of the noble gases and 25% of the iodine contained in this fraction should be assumed to be available for release from the containment.

f. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

g. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on a case-by-case basis.

h. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.[1] Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.

i. Release of fission products to the secondary system should be computed by assuming that all fission products released from the fuel clad are uniformly mixed in the primary coolant volume.

j. The primary-to-secondary leak rate limitation incorporated or to be incorporated as a technical specification requirement should be assumed to exist until the primary system pressure falls below the secondary system pressure.

k. The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

2. Acceptable assumptions for atmospheric diffusion and dose conversion are:

a. The 0-to-8-hour ground-level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in Section 3-3.5.2 of Meteorology and Atomic Energy 1968 (Ref. 1), should be used only in the 0-to-8-hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor building only.

b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground or for the radiological decay of iodine in transit.

c. For the first 8 hours, the breathing rate of a person offsite should be assumed to be 3.47×10^{-4} m^3/sec. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} m^3/sec. From 24 hours until the end of the accident, the rate should be assumed to be 2.32×10^{-4} m^3/sec. (These values were developed from the average daily breathing rate [2×10^7 cm^3/day] assumed in a report (Ref. 2) of ICRP.[2])

d. The iodine dose conversion factors are also given in Reference 2.

e. External whole body doses should be calculated using "infinite cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for

[1] The effect on containment leakage under accident conditions of features provided to reduce the leakage of radioactive materials from the containment should be evaluated on a case-by-case basis.

[2] International Commission on Radiological Protection.

a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor." (Ref. 3) Editorial additions were made to the quotation so that gamma as well as beta emitting material could be considered. Under these conditions, the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose in air at the cloud center is:

$$_{\beta}D'_{\infty} = 0.457 \bar{E}_{\beta}\chi$$

The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $_{\beta}D'_{\infty} = 0.23\ \bar{E}_{\beta}\chi$). For gamma emitting material, the dose rate in air at the cloud center is:

$$_{\gamma}D'_{\infty} = 0.507 \bar{E}_{\gamma}\chi$$

From a semi-infinite cloud, the gamma dose rate in air is:

$$_{\gamma}D'_{\infty} = 0.25 \bar{E}_{\gamma}\chi$$

where

$_{\beta}D'_{\infty}$	=	beta dose rate from an infinite cloud (rad/sec)
$_{\gamma}D'_{\infty}$	=	gamma dose rate from an infinite cloud (rad/sec)
$\bar{E}_{\beta}$	=	average beta energy per disintegration (Mev/dis)
$\bar{E}_{\gamma}$	=	average gamma energy per disintegration (Mev/dis)
χ	=	concentration of beta or gamma emitting isotope in the cloud (Ci/m³)

f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:

(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance, taking into account special meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site-related characteristics must be evaluated on a case-by-case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground-level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.

(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes (Ref. 4), should be used.

g. The atmospheric diffusion model should be as follows:

(1) The basic equation for atmospheric diffusion from a ground-level point source is:

$$\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$$

where

χ	=	the short-term average centerline value of the ground-level concentration (Ci/m³)
Q	=	amount of material released (Ci/sec)
u	=	windspeed (m/sec)
σ_y	=	the horizontal standard deviation of the plume (meters) [see Figure V-1, Ref. 5].
σ_z	=	the vertical standard deviation of the plume (meters) [see Figure V-2, Ref. 5].

(2) For time periods greater than 8 hours, the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is:

$$\chi/Q = \frac{2.032}{\sigma_z u x}$$

where

x	=	distance from the point of release to the receptor; other variables are as given in paragraph g. (1), above.

(3) The atmospheric diffusion model[3] for ground-level releases is based on the information in the table below.

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, wind speed 1 m/sec, uniform direction
8-24 hours	Pasquill Type F, wind speed 1 m/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, wind speed 3 m/sec (b) 60% Pasquill Type F, wind speed 2 m/sec (c) wind direction – variable within a 22.5° sector.

[3]This model should be used until adequate site meteorological data are obtained. In some cases, available information, such as meteorology, topography, and geographical location, may dictate the use of a more restrictive model to insure a conservative estimate of potential offsite exposures.

Time Following Accident	Atmospheric Conditions
4-30 days	(a) 33.3% Pasquill Type C, wind speed 3 m/sec
	(b) 33.3% Pasquill Type D, wind speed 3 m/sec
4-30 days	(c) 33.3% Pasquill Type F, wind speed 2 m/sec
	(d) Wind direction – 33.3% frequency in a 22.5° sector.

(4) Figures 2(A) and 2(B) give the ground-level release atmospheric diffusion factors based on the parameters given in paragraph g.(3), above.

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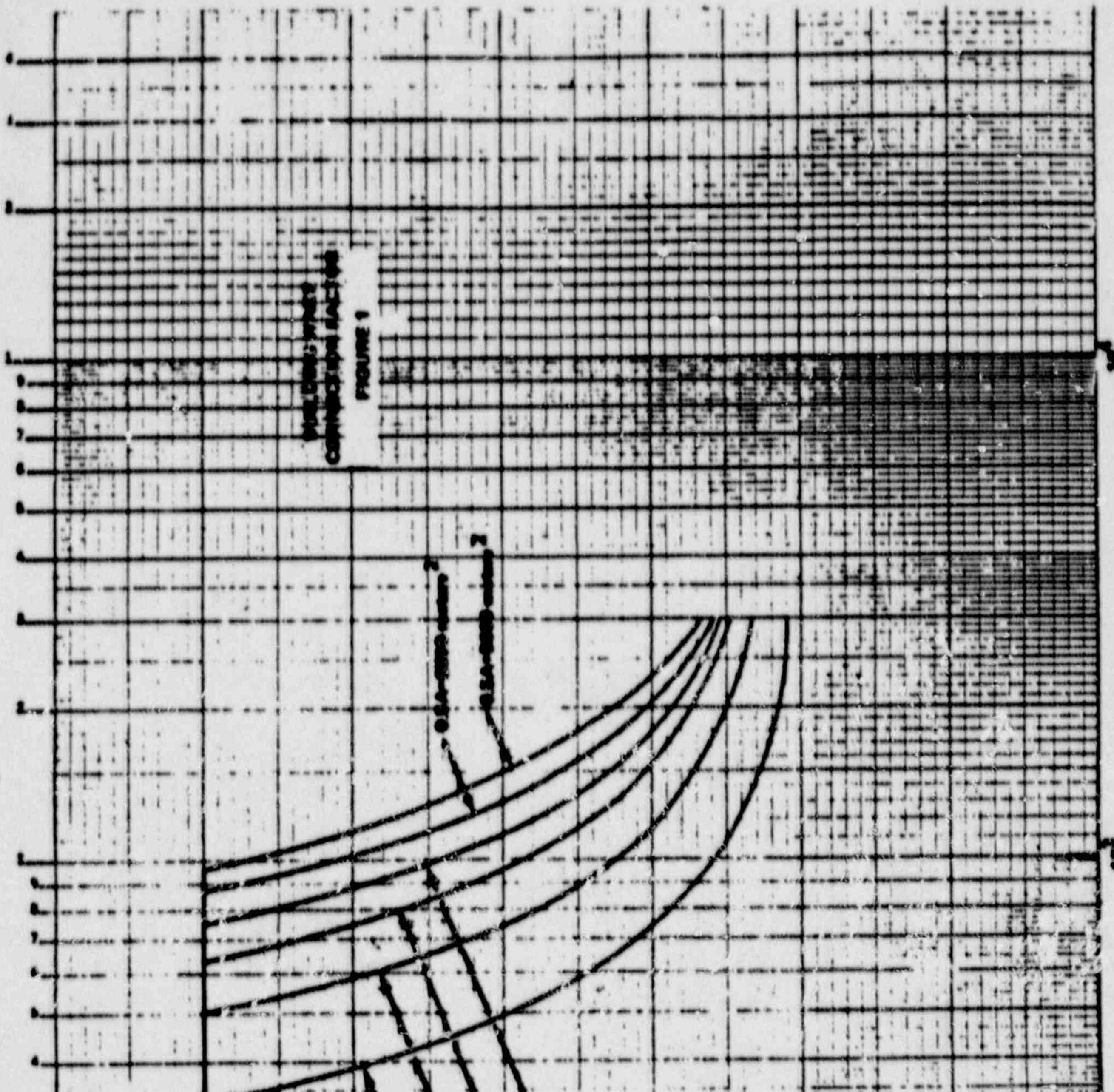
1. D. H. Slade, ed., "Meteorology and Atomic Energy – 1968," TID-24190, Division of Technical Information, U.S. Atomic Energy Commission, July 1968.

2. Report of Committee II, "Permissible Dose for Internal Radiation," ICRP Publication 2, 1959.

3. D. H. Slade, *op. cit.*, Section 7.4.1.1.

4. C. M. Lederer, J. M. Hollander, and I. Perlman, "Table of Isotopes," Sixth Edition, University of California, Berkeley, Lawrence Radiation Laboratory.

5. F. A. Gifford, Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety, Vol. 2, No. 4, June 1961.



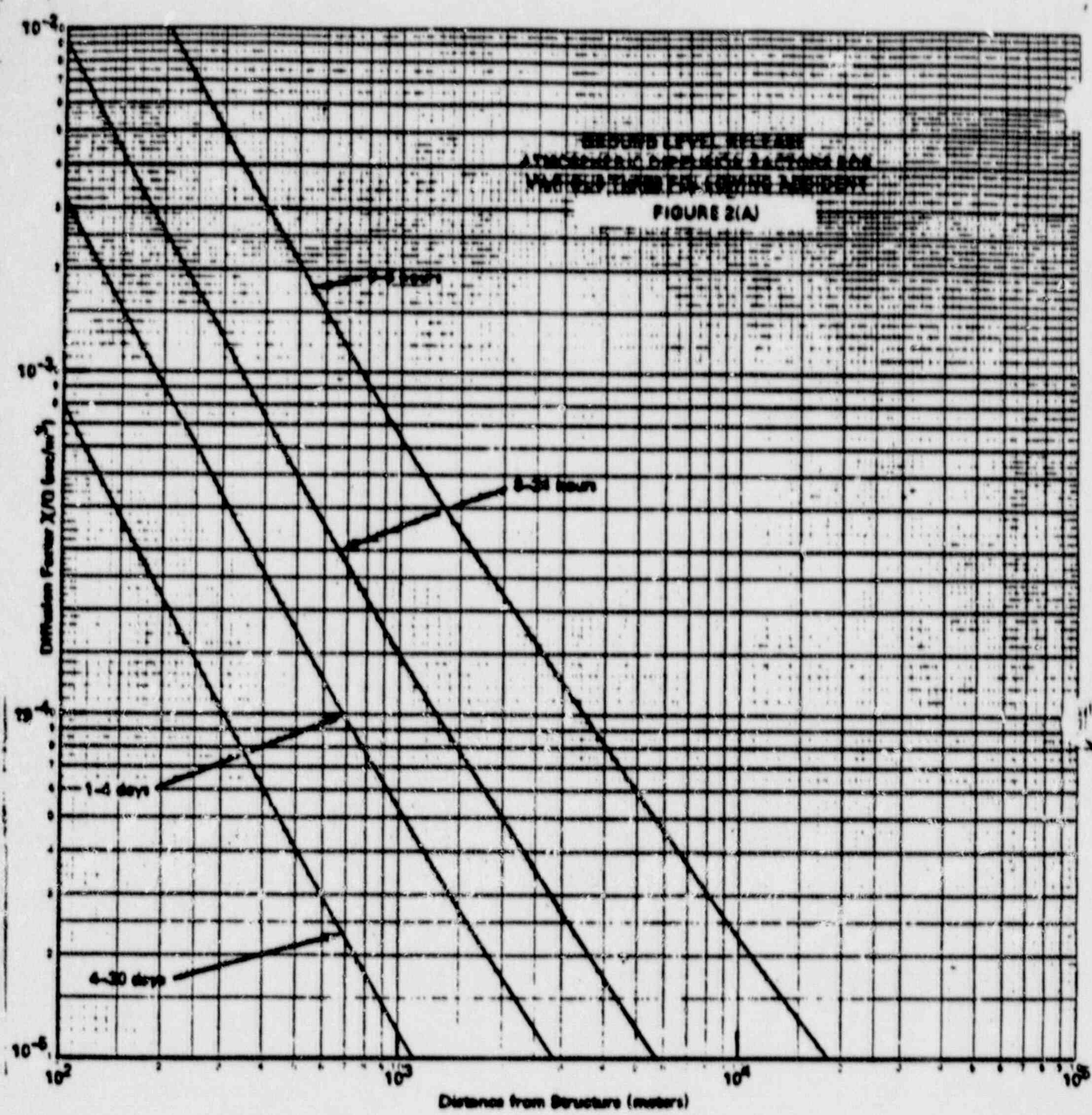
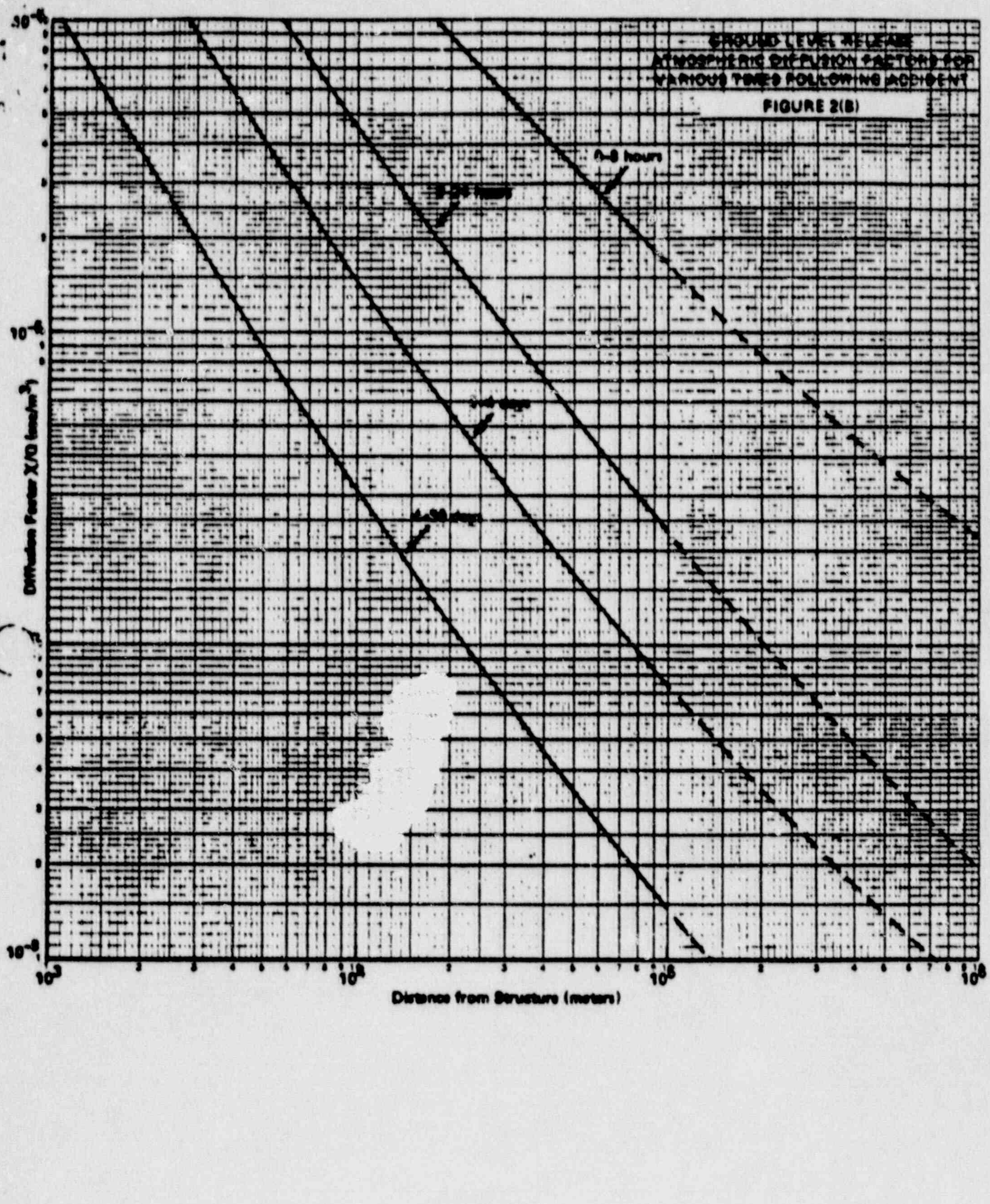


FIGURE 2(A)
8–24 hours
1–4 days
4–30 days
Diffusion Factor X/Q (sec/m³)
Distance from Structure (meters)
10⁻²
10⁻³
10⁻⁴
10⁻⁵
10²
10³
10⁴
10⁵


GROUND LEVEL RELEASE
ATMOSPHERIC DIFFUSION FACTORS FOR
VARIOUS TIMES FOLLOWING ACCIDENT
FIGURE 2(B)
0-8 hours
Diffusion Factor X/Q (sec/m³)
Distance from Structure (meters)

Reference 2

BACKGROUND INFORMATION RELATED TO

DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance of SER to Zion 1/2 allowing full power operation with open 42" containment isolation valves.

b) Methodology used for calculating related offsite doses.

ZION

CORE AND GAP ACTIVITIES

Assumptions: Operation at 3391 MWt for 500 days

Isotope	Curies in the Core ($X\ 10^7$)	I 131 EQU $x\ 10^7$	Percent of Core Activity in the Gap	Curies in the Gap ($X\ 10^5$)	I 131 EQU ($X10^5$)
I-131	8.35	8.35	2.3	19.2	19.2
I-132	12.75	.46	0.26	3.3	.12
I-133	19.09	5.16	0.79	15.1	4.08
I-134	23.01	.39	0.16	3.8	.06
I-135	17.05	1.43	0.43	7.5	.63
		15.79			24.09

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES OPEN 50°

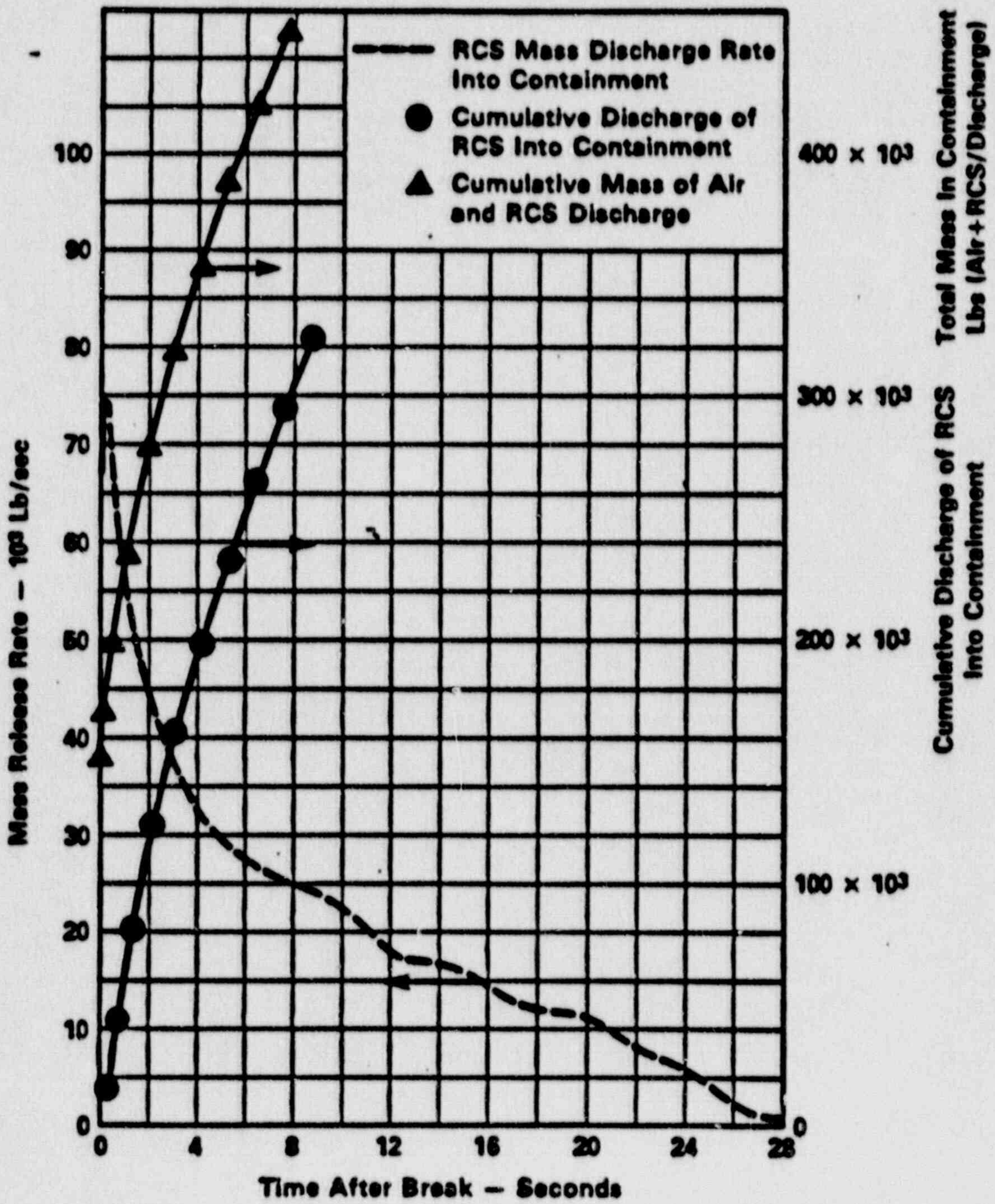
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Source	Radiological Sources	Curies Discharged I 131 EQ	Site Boundary Exposure (REM)
Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	73.5	18.7
RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119 x 10^5 curies]	188	48
RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	82	21
RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	38,000	9676
RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	248,950	63,400

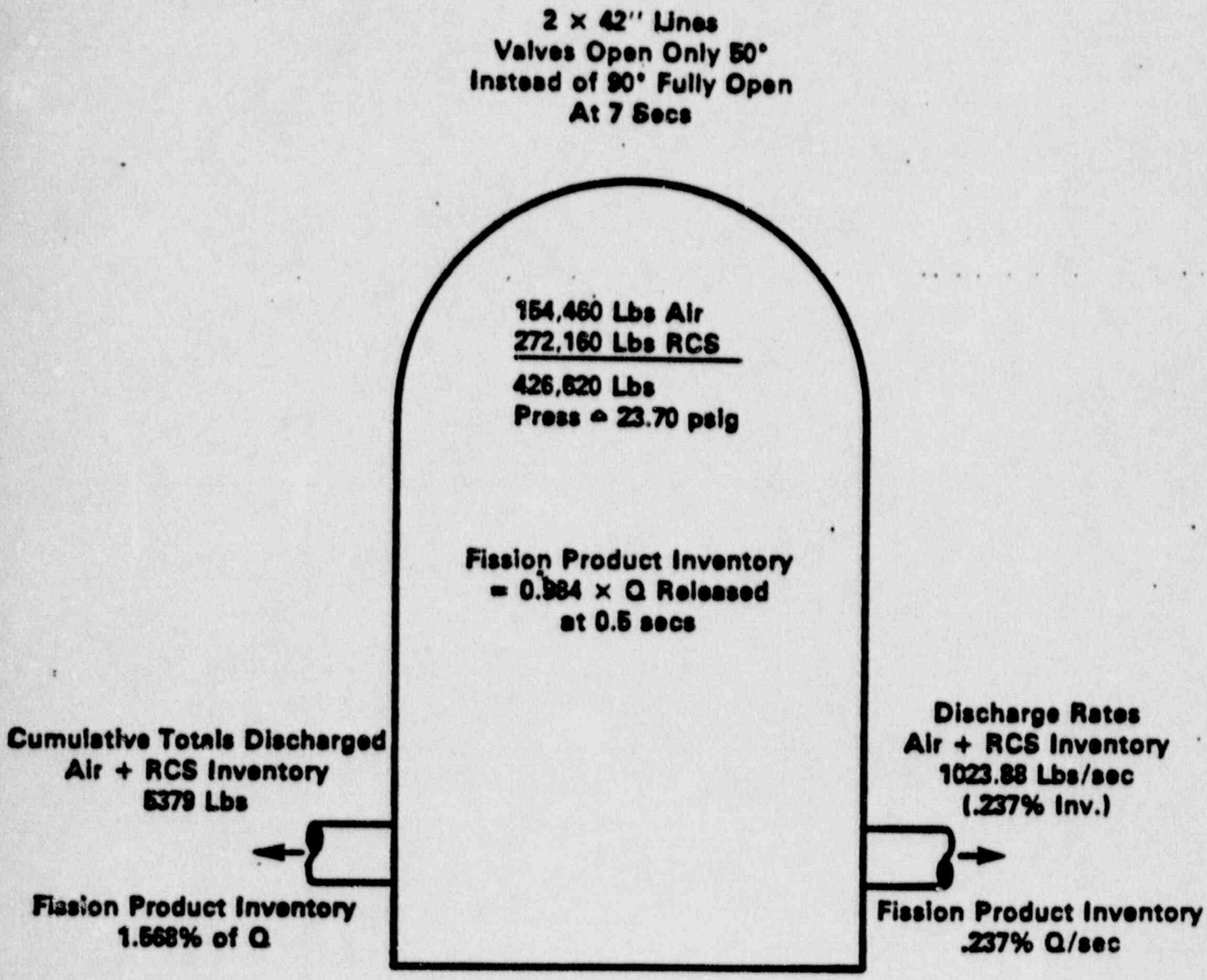
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES
DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines.
Valves open 50°

Given Q = total inventory of fission products in RCS at T=0.5 secs after LOCA

- If Q is released instantaneously to the total containment volume:

  Fission product inventory discharged outside containment over 7 secs = 1.568% Q

- If Q is released over time with RCS inventory and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 0.561% Q

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES FULLY OPEN (90°)

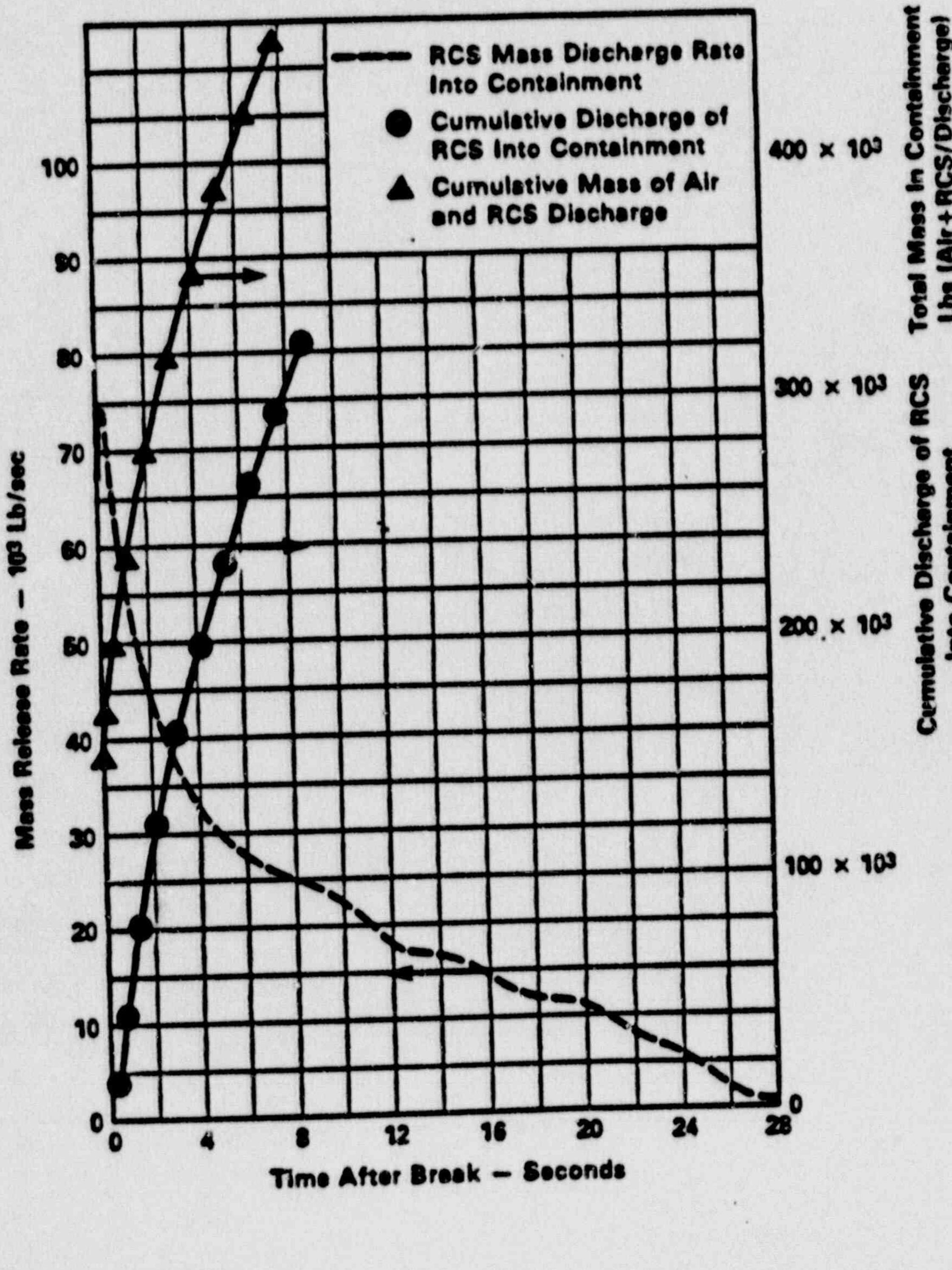
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Source	Radiological Sources	Curies Discharged I 131 EQ	Site Boundary Exposure (REM)
Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	204.3	52
RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119×10^5 curies]	522	132
RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	227	58
RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	105,600	26,878
RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	691,520	176,010

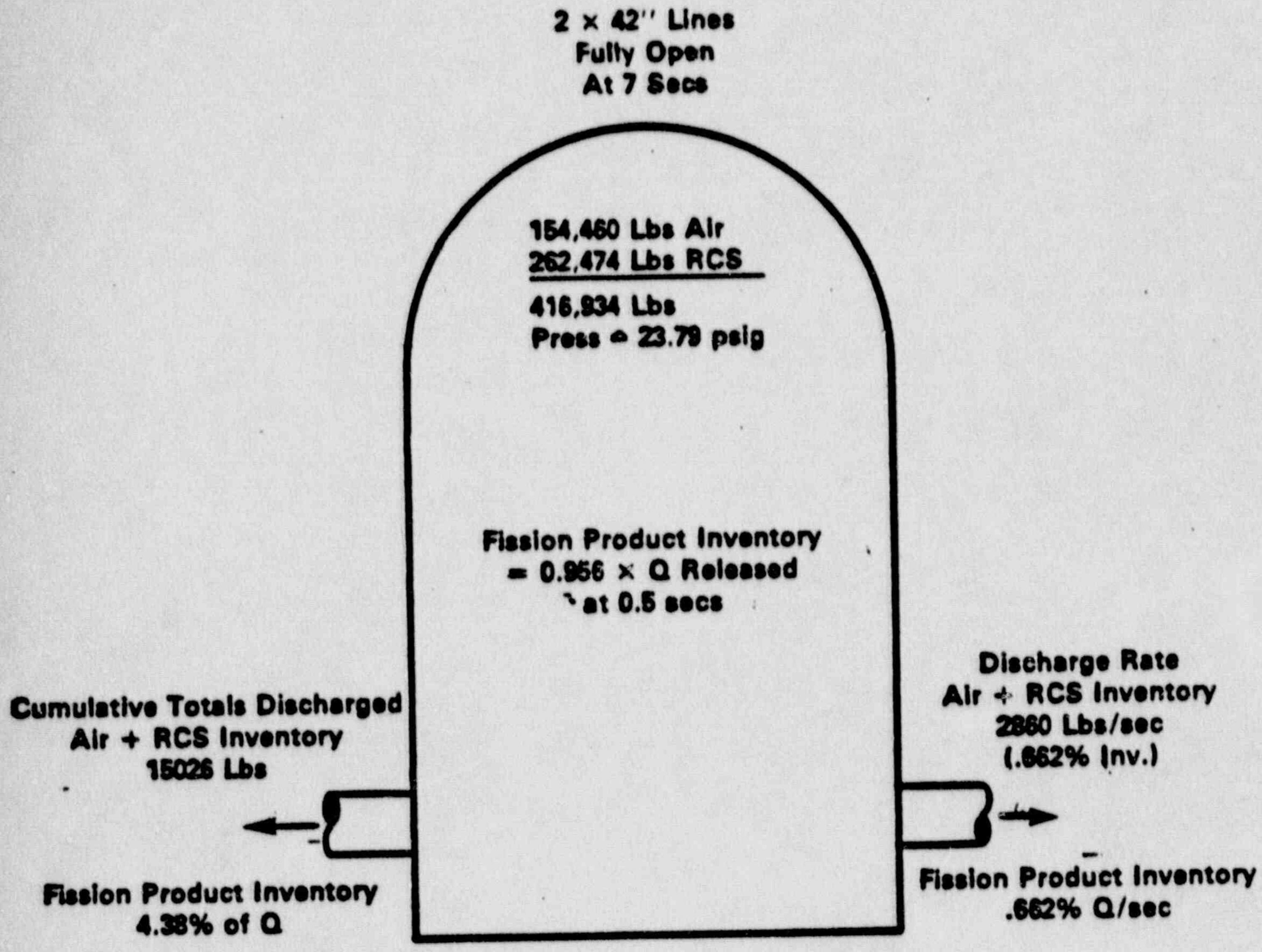
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines
fully open (90°).

Given Q = Total inventory of fission products in RCS at T=0.5 sec after LOCA.

- If Q is released instantaneously to the total containment volume

  Fission product inventory discharged outside containment over 7 secs = 4.38% Q

- If Q is released over time with RCS inventory, and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 1.90% Q

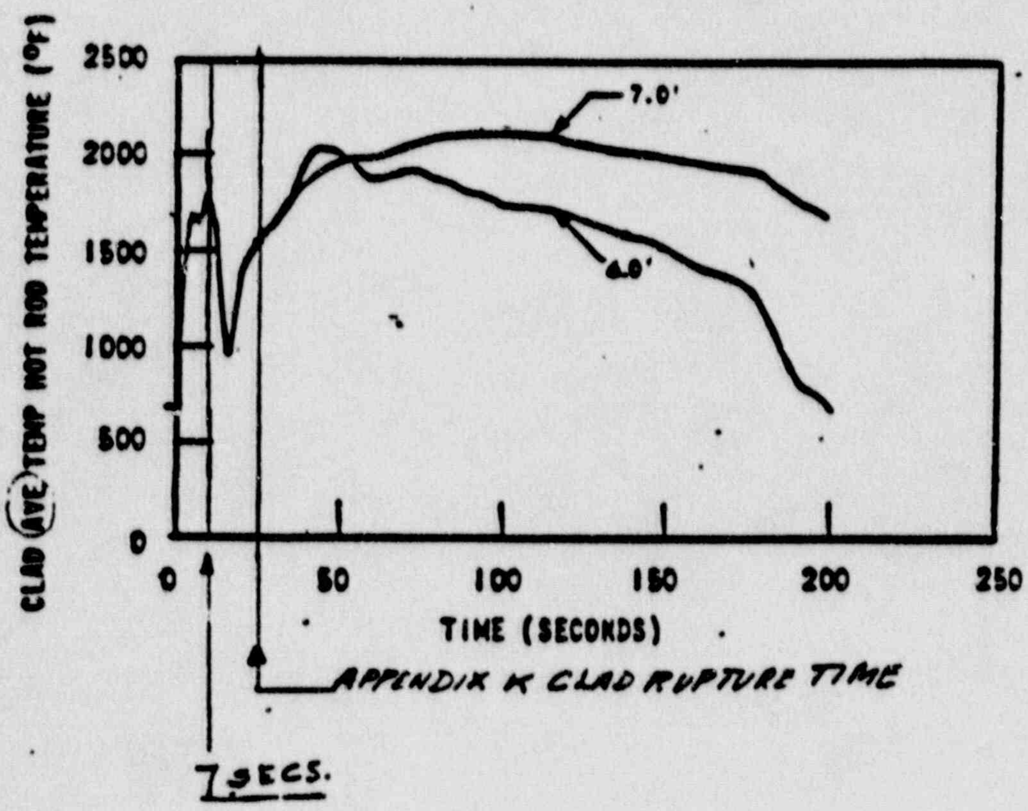


Figure 14 F.2-19a Peak Clad Temperature - DECLG (CD=1.0)
(Unit 1)

3.1.3.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is [1.30*].

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB). DNB causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength- 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minimum values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerable below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under normal operating conditions is well in excess of the maximum design pressure. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The end of life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history, however it does not exceed the design limit of 2250 psia.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1 [as 720°F].

Previous experience with removable rods has been attained at Saxton, Yankee and Zorita; and additional experience will be acquired at the San Onofre Cycle 2 and Surry Unit 1. Over 300 fuel rods were removed and re-inserted into assemblies during the Saxton re-constitution without evidence of failure. Leak detection tests were performed on the assemblies after all rods were re-inserted, and no leakage was detected. An equally large number of Saxton rods have been successfully removed, examined and re-inserted into over 12 3x3 subassemblies at Saxton. In addition, 28 full length Yankee rods were removed, examined and re-inserted into Yankee Core V special assemblies. Simillar handling of 22 removable rods was sucessfully completed during the first Zorita refueling. All such fuel handlings have been done routinely and without difficulty.

The same fuel rod design limits indicated in section 3.2.3 fuel temperature and internal pressure, are maintained for these removable rods and there is no reduction in margin to DNB. Their inclusion in the initial Zion Unit 1 core loading introduces no additional safety considerations and in no way changes the safeguard analyses and related engineering information presented in previously submitted material in support of the license application.

3.2.3.5 Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods is calculated by a code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 per cent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests. See additional details in Section 3.3.3.1.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

3. Burnup:

Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is not expected to be reached until the peak burnup reaches approximately 65,000 MWD/MTU. The peak pellet burnup for fuel in equilibrium cycling is expected to be 50,000 MWD/MTU. The design equilibrium region average discharge burnup is about 33,000 MWD/MTU.

4. Fuel temperature and kw/ft:

At zero burnup, cladding damage is calculated to occur at 31 kw/ft based upon cladding strain reaching the damage limit. At this power rating [illegible]% of the pellet central region is expected to be in the molten condition. The maximum thermal output at rated power is 15.0 kw/ft.

An evaluation of the fuel densification as it affects operating limits for Unit 1 is given in Appendix 3A, and WCAP 8060 Addendum 2 (references 18 [illegible]). WCAP 8122 and its addendum (references 25 and 26) evaluates fuel densification as it applies to Unit 2.

In Appendix 3A, the initial fuel densities of Regions 2 and 3 reflect the actual region densities as amended in the FSAR Table 4.1. The fuel density itself does not have any significant effect on the power transient associated with the rod ejection accident.

In the fuel densification report a 2700°F clad surface temperature limits is used for accidents such as Rod Ejection and Locked Rotor as discussed in WCAP 7855 and in Attachment 13 of Westinghouse letter NS-SL-543 (January 12, 1973) to Dr. D. F. Knuth.

In Appendix 3A, the methods described in WCAP-7422-1 "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident" Section 6.2.3 were used to determine the gap conductance during LOCA except for the initial value. The initial gap conductance in LOCTA was adjusted such that the initial average temperature in LOCTA was equal to the design value of the BOL average temperature at the appropriate Kw/ft plus an additional temperature increase to cover an uncertainty which was equal to 9.375 x (Kw/ft).

Parameters considered important to fuel densification analysis are summarized in Table 3.1 and Table 4.1 of the Zion and Point Beach Unit 2 fuel densification reports. In addition:

a. The initial, as fabricated diametral fuel pellet/clad gap for Zion is 0.0075 inches which is 0.001 inches less than that for Point Beach Unit 2.

b. The time integrated axial power distribution (or fast neutron flux distribution) used in the analyses of gap conductance for both Zion and Point Beach Unit 2 is given in Attachment L, Figure 2 of Westinghouse letter NS-SL-521 (January 4, 1973) to Dr. D. F. Knuth.

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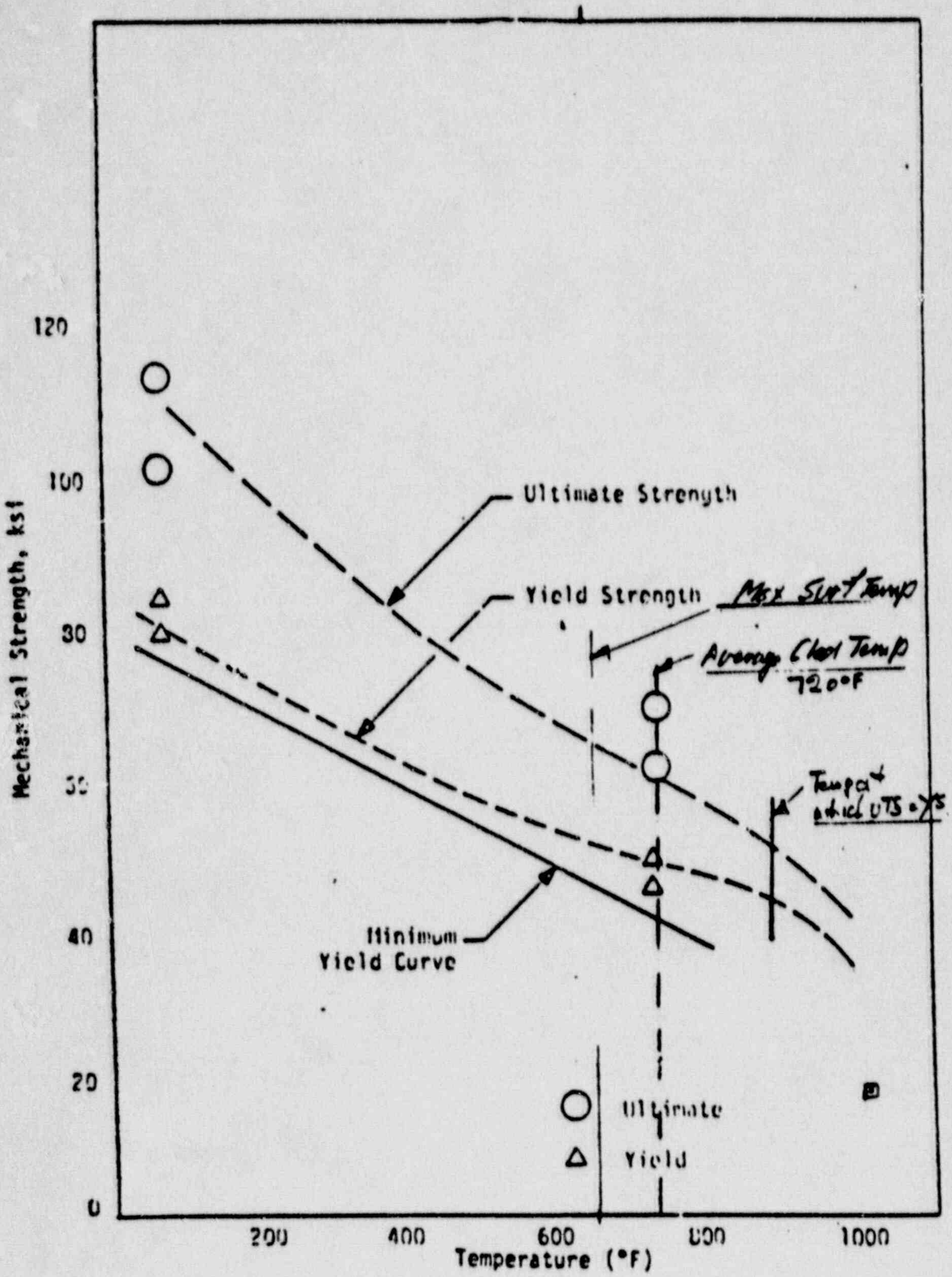


FIGURE ____ MECHANICAL STRENGTH OF ROD TUBING VERSUS TEMPERATURE

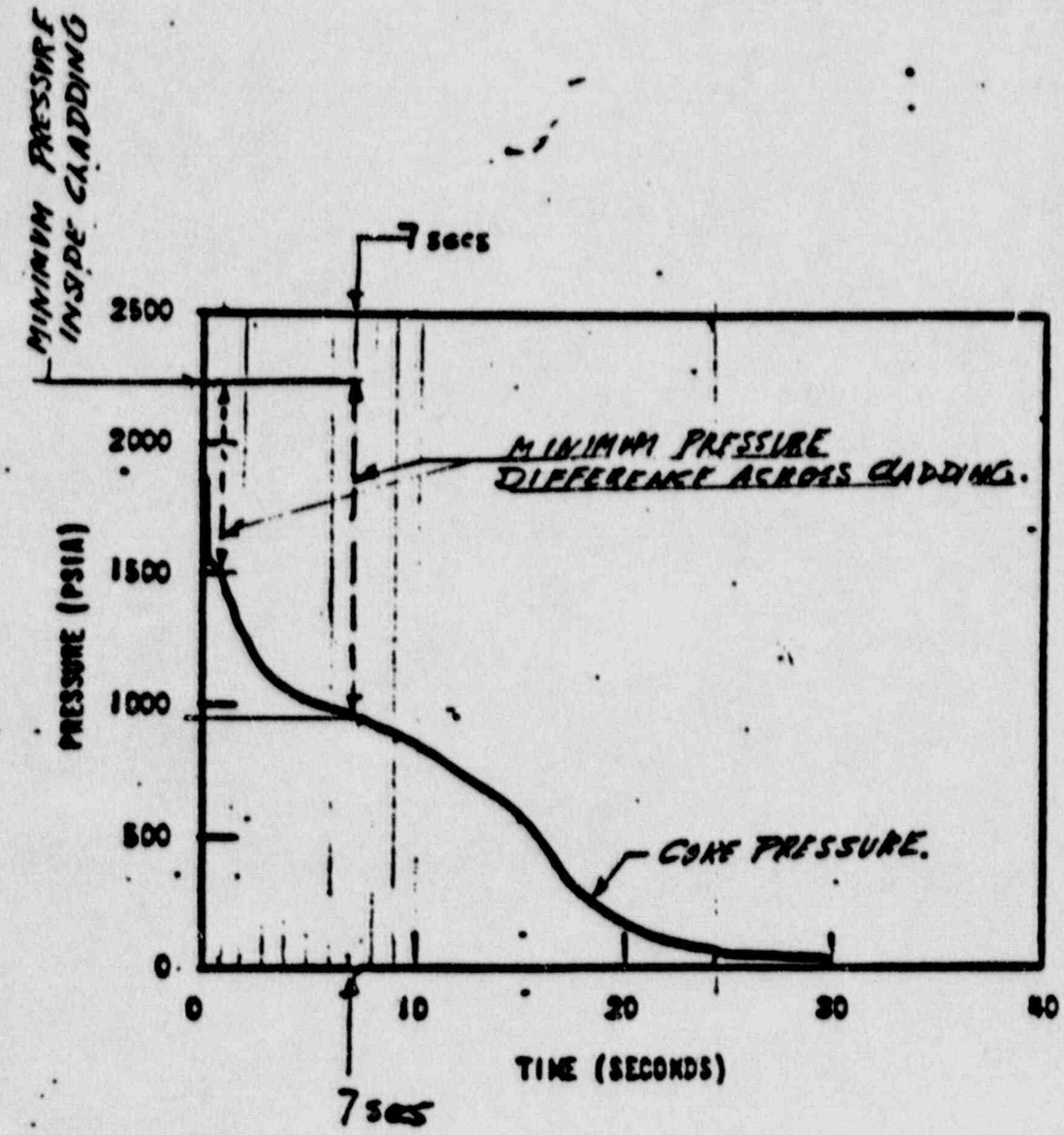


Figure 14 F.2-10a Core Pressure - DECLG (C_D=1.0)
(Unit 1)

TABLE 1

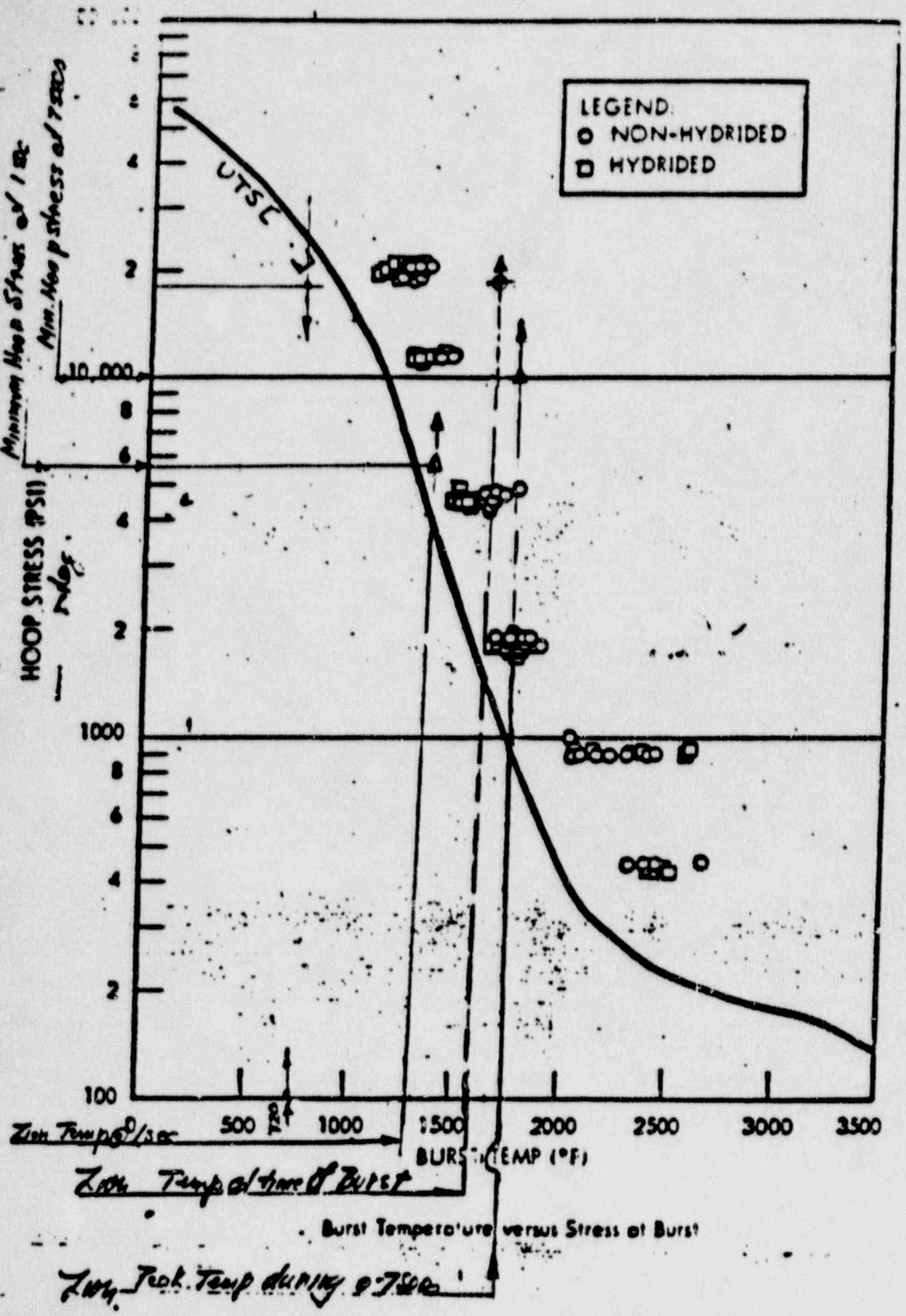
Engineering Hoop Stress as a Function of Internal Fuel Rod

Gas Pressure and Fuel Vendor Design

Design	Hoop Stress (psi) for a 600 psi Differential Across the Cladding Wall
B&W 15x15	4570
B&W 17x17	4540
C-E 16x16	4280
W 15x15	4910 ←
W 17x17	4690
GE 8x8	4050
ENC 15x15*	3940
ENC 8x8**	3880

* D. C. Cook, Unit 1

** Oyster Creek



Burst Temperature versus Stress at Burst

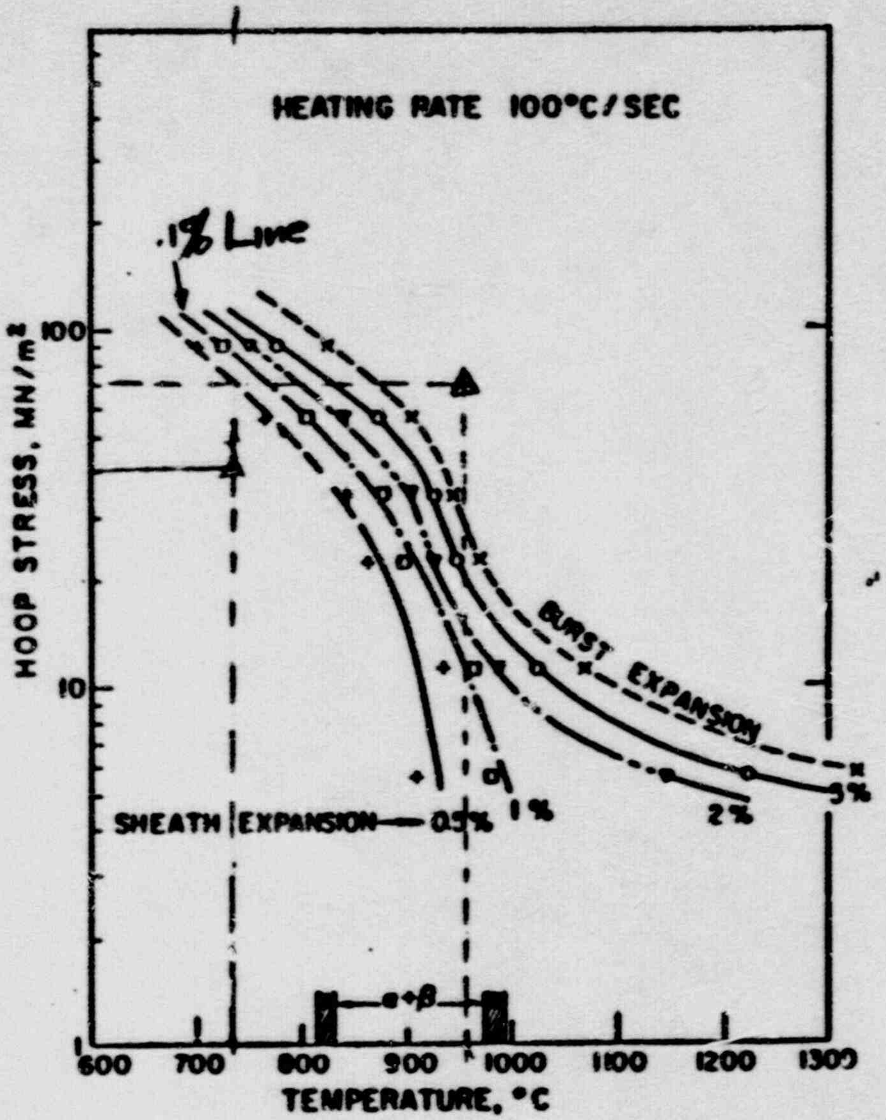


FIGURE 10 (HARDY)

Isostrain and rupture curves plotted as a function of hoop stress and temperature for tubes heated at 100°C/sec.

1 MN/m² = 142.9 psi

1750°F = 954°C 1350°F = 732°C
10,000 psi = 68 MN/m² 5800 psi = 40.6 MN/m²

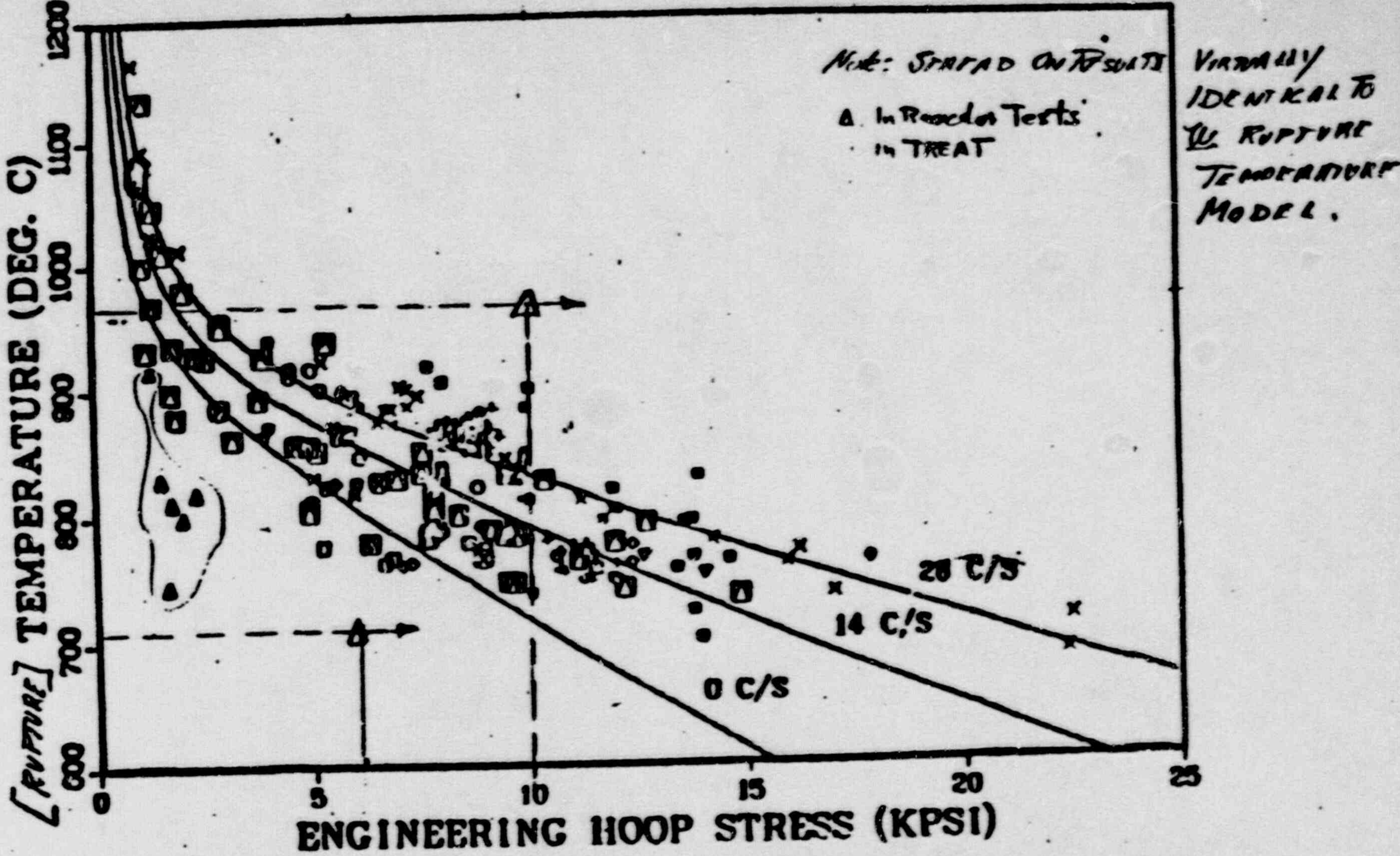


Fig. 3 ORNL correlation of rupture temperature as a function of engineering hoop stress and temperature-ramp rate with data from internally heated Zircaloy cladding in aqueous atmospheres.

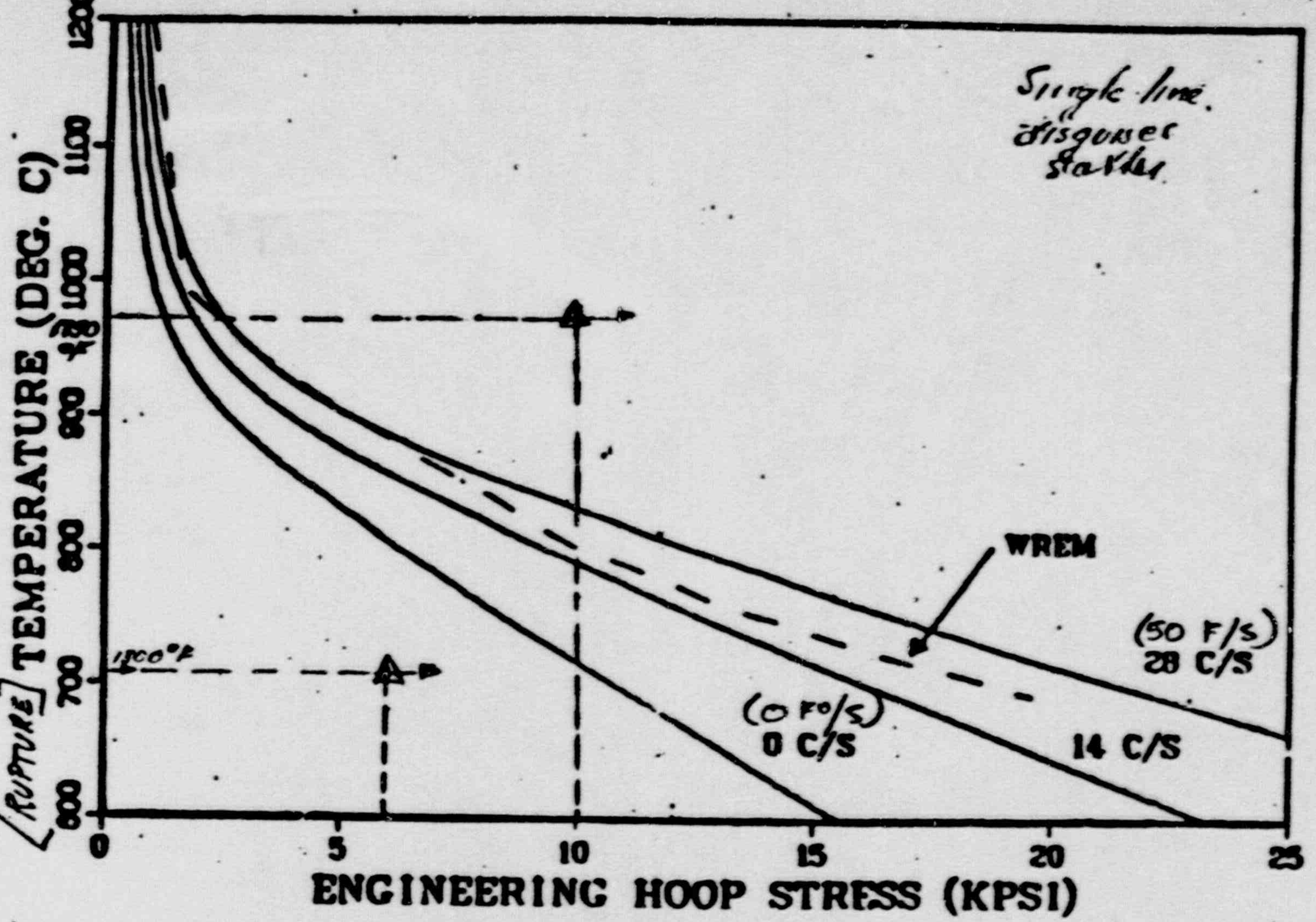


Fig. 17 WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

1974 . WATER REVISED . CLASS

3.5 Clad Swelling and Rupture Model

During a LOCA the clad is assumed to strain uniformly and plastically in the radial direction provided that both the temperature and the differential pressure across the clad are sufficiently high. If the strain exceeds [10%] or the clad temperature exceeds the burst temperature (determined as a function of the instantaneous stress) the clad is assumed to burst and an additional local strain is added to the burst node. (a,c)

Three empirical models are employed to evaluate the clad swelling and rupture behavior.

3.5.1 Clad Swelling Prior to Rupture

Hardy[24] performed a series of tests in which rods with constant internal pressure were ramped to a series temperatures at various constant ramp rates. The pressures reported by Hardy were converted to hoop stresses by the formula

(3-69)

and the strain at a given temperature and ramp rate were correlated as functions of the derived hoop stress. The equation developed which best describes the data is

(a,c)

(3-70)

where:

(a,c)

1974 : Later Revised

WESTINGHOUSE

(a,c)

(a,c)

(a,c)

(a,c)

3.5.2 Clad Burst

Clad is assumed to burst if it reaches [10%] hoop strain based on the swelling model described above or if the clad temperature in the burst node reaches the burst temperature. (a,c) Burst temperature is calculated as a function of hoop stress based on correlation of the Westinghouse single rod burst test [27] data shown in Figure 3-1. The best estimate curve from figure 3-1 is used and pressure is converted to hoop stress by the relationship described in Equation 3-69 using original test specimen geometry. This best estimate curve is described by the equation

(a,b,c)

$$T_{burst} = \left[\qquad \right] \tag{3-71A}$$

3.53 Local Hoop Strain After Burst

The localized diametral swelling that occurs very rapidly at the time of burst is calculated from a correlation of single rod burst test data of Westinghouse and others. Figure 3-2 shows the correlation and the ranges of the data used. Expressed in terms of hoop stress the correlation gives

(a,b,c)

$$\frac{\Delta d^*}{d_o} = \left[\qquad \right] \tag{3-71B}$$

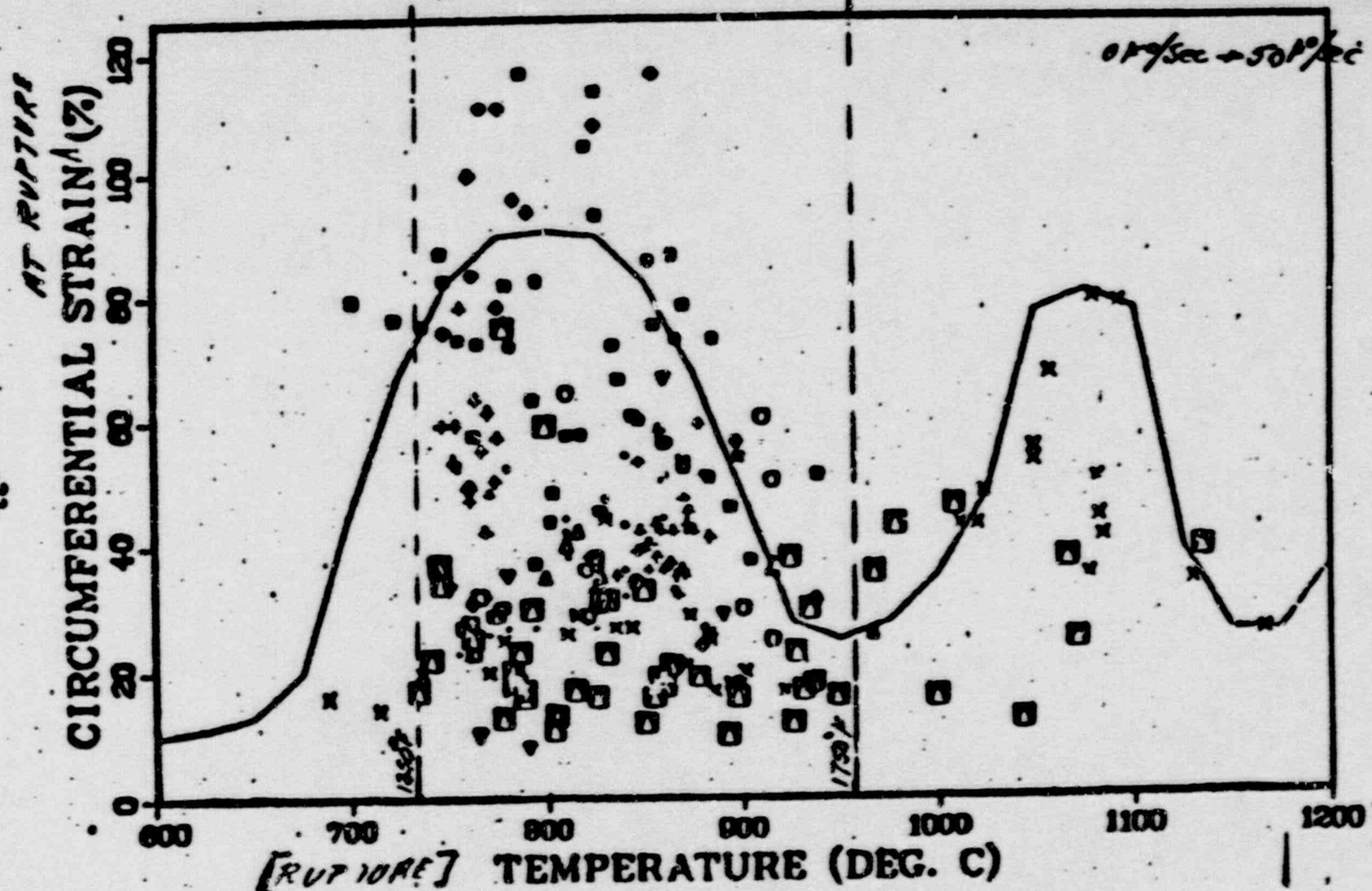


Fig. 8 ~~Maximum circumferential strain~~ as a function of ~~rupture temperature~~ for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

NUREG-75/077

THE ROLE OF FISSION GAS RELEASE IN REACTOR LICENSING

CORE PERFORMANCE BRANCH

U. S. NUCLEAR REGULATORY COMMISSION

NOVEMBER 1975

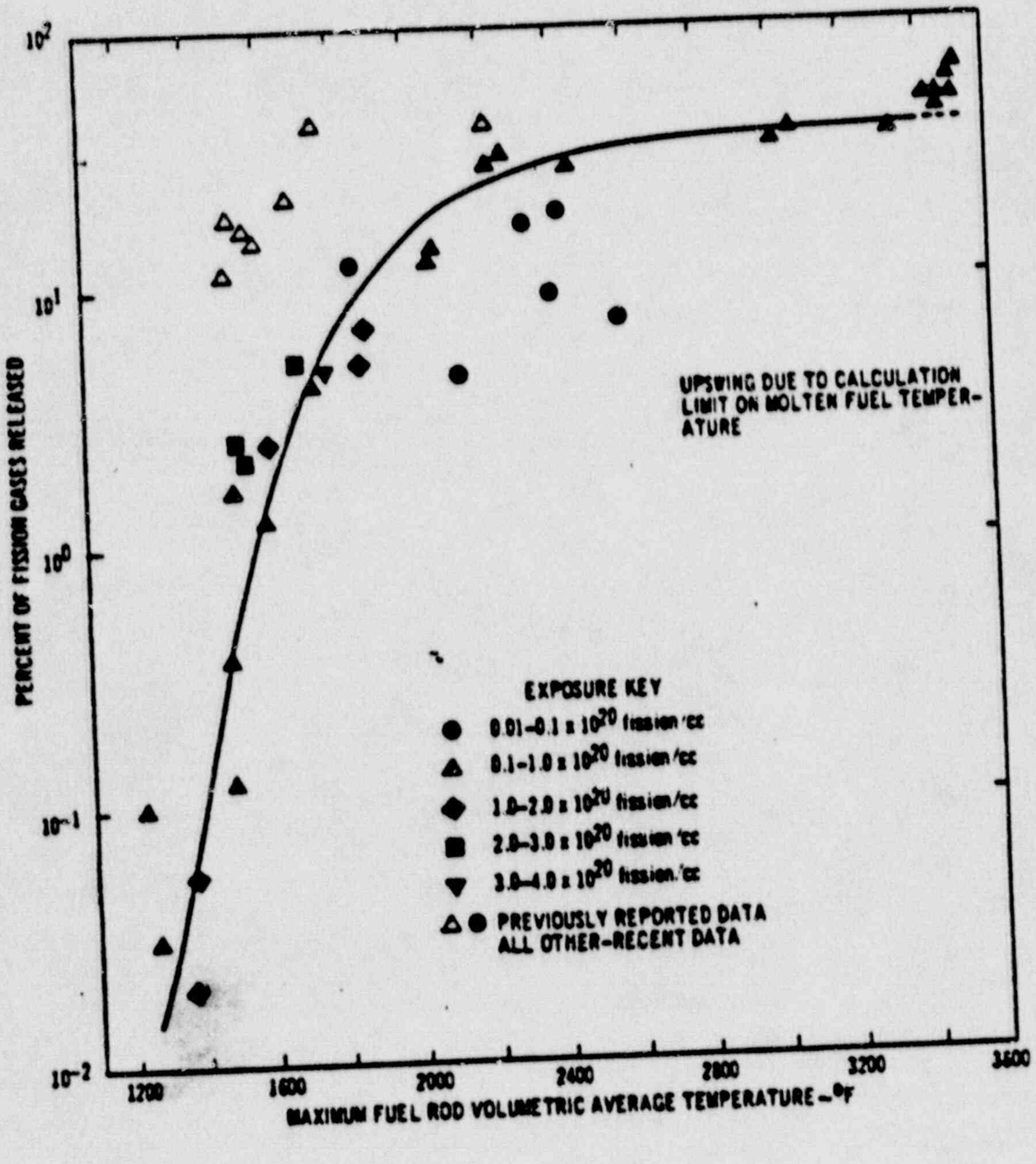


Fig. 2. The Hoffman & Coplin correlation of fission gas release as a function of temperature (from Ref. 35).

ZION

CORE TEMPERATURE DISTRIBUTION

Assumptions: Operation at 3391 MWt for 500 days

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
14.5	2500

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the lines associated with the containment purge and vent systems. It supplements the position taken in SRP Section 6.2.4.

While the containment purge and vent systems provide plant operational flexibility, their designs must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on their use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously. There is similar disparity in the need for, and use of, containment vent systems at operating plants.

Containment purge systems have been used in a variety of ways; for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. Containment vent systems are typically used to relieve the initial containment pressure buildup caused by the heat load imposed on the containment atmosphere during reactor power ascension, or to periodically relieve the pressure buildup due to the operation of pneumatic controllers. However, the purge and vent lines provide an open path from the containment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR Part 100 guidelines values.

The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines.

The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR Part 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Similar concerns apply to vent system designs.

The size of the purge lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR Part 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type, e.g., BWR, PWR, or HTGR.

B. BRANCH TECHNICAL POSITION

The systems used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. General Design Criterion 54 requires that the reliability and performance capabilities of containment isolation valves reflect the importance of safety of isolating the systems penetrating the containment boundary. Therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in Branch Technical Position MEB-2, "Pump and Valve Operability Assurance Program." (Also see SRP Section 3.10.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the supply line and exhaust line flows as a function of time up to and during valve closure.

   b. The number of supply and exhaust lines that may be used should be limited to one supply line and one exhaust line, to improve the reliability of the isolation function as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in the evaluation of the emergency core cooling system effectiveness and 10 CFR Part 100 regarding offsite radiological consequences.

   c. The size of the lines should not exceed about eight inches in diameter, unless detailed justification for larger line sizes is provided, to improve the reliability and performance capability of the isolation and containment functions as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in evaluating the emergency core cooling system effectiveness and 10 CFR Part 100 regarding the offsite radiological consequences.

d. As required by General Design Criterion 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria, to reflect the importance to safety of isolating these lines. General Design Criterion 56 establishes explicit requirements for isolation barriers in purge system lines.

e. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. Furthermore, if energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can effect the isolation function.

[On-Line]

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds, to facilitate compliance with 10 CFR Part 100 regarding offsite radiological consequences.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values.

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters, and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

c. ~~An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.~~

d. The maximum allowable leak rate of the purge isolation valves should be specified on a case-by-case basis giving appropriate consideration to valve size, maximum allowable leakage rate for the containment (as defined in Appendix J to 10 CFR Part 50), and where appropriate, the maximum allowable bypass leakage fraction for dual containments.

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 1), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents. "Coolability," in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, §50.46 (Ref. 3).

All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in SRP Section 4.4. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Evaluation Branch for use in estimating the radiological consequences of plant releases.

The fuel system review covers the following specific areas.

A. Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that damage will be limited to acceptable levels. The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. Design Evaluation

The performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents is reviewed to determine if all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. On-line fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by CPB to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

Specific criteria necessary to meet the requirements of 10 CFR Part 50, §50.46; General Design Criteria 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 identified in subsection I of this SRP section are as follows:

A. Design Bases

The fuel system design bases must reflect the four objectives described in subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

1. Fuel System Damage -

This subsection applies to normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Such damage criteria should address the following to be complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles. fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the ASME Code (Ref. 4) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles (Ref. 5). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the Safety Analysis Report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

2. Fuel Rod Failure

This subsection applies to normal operation, anticipated operational occurrences, and postulated accidents. Paragraphs (a) through (c) address failure mechanisms that are more limiting during normal operation, and the ~~information to be reviewed should be contained in~~ Section 4.2 of the Safety Analysis Report. Paragraphs (d) through (h) address failure mechanisms that are more limiting during anticipated operational occurrences and postulated accidents, and the information to be reviewed will usually be contained in Chapter ~~15~~ ~~of the Safety Analysis Report~~. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere.

To meet the requirements of (a) General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, and (b) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Fuel failure criteria should address the following to be complete.

(a) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 7) for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in

Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown (Ref. 8) to be insufficient for primary hydride formation.

(b) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The review of these criteria is detailed in SRP Section 4.4. For normal operation and anticipated operational occurrences, violation of the thermal margin criteria is not permitted. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes.

Although a thermal margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: It has also been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and anticipated operational occurrences, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

(f) Excessive Fuel Enthalpy: For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location. For full-power RIAs in a BWR and all RIAs in a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 (Ref. 6) as it relates to fuel rod failure. The 170 cal/g enthalpy criterion is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions (PCI). Other criteria may be more appropriate for an RIA, but continued approval of this enthalpy criterion and the thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

(h) Bursting: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to the incidence of rupture during a LOCA, a rupture temperature correlation must be used in the LOCA ECCS analysis. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one described in Ref. 10.

(i) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.

3. <u>Fuel Coolability</u>

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addresses the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Control rod insertability criteria are also addressed in this subsection. Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, §50.46, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800°F previously described in Reference 11 is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of Appendix K of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data (such as Refs. 10, 12, and 13) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 2200°F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

B. Description and Design Drawings

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

Type and metallurgical state of the cladding
Cladding outside diameter
Cladding inside diameter
Cladding inside roughness
Pellet outside diameter
Pellet roughness
Pellet density
Pellet resintering data
Pellet length
Pellet dish dimensions
Burnable poison content
Insulator pellet parameters
Fuel column length
Overall rod length
Rod internal void volume
Fill gas type and pressure
Sorbed gas composition and content
Spring and plug dimensions
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawing have also been found necessary for an acceptable fuel system description:

Fuel assembly cross section
Fuel assembly outline
Fuel rod schematic
Spacer grid cross section
Guide tube and nozzle joint
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Burnable poison rod assembly cross section
Burnable poison rod assembly outline
Burnable poison rod schematic
Orifice and source assembly outline

C. Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed. Those methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the Safety Analysis Report by reference.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed when practical to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested prior to irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

Spacer grid structural tests
Control rod structural and performance tests
Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
Fuel assembly growth
Fuel assembly bowing
Channel box wear and distortion
Fuel rod ridging (PCI)
Crud formation
Fuel rod integrity
Holddown spring relaxation
Spacer grid spring relaxation
Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished prior to operation of a full core of that design. This inability to perform in-reactor testing may result from an incompatability of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see subsection II.D below).

3. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient*

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 14 and 15). Examples of previous fuel performance code reviews are given in References 16 through 20.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core

*Although needed in fuel performance codes, this model is reviewed as described in SRP Section 4.4.

power distributions (power spiking, see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR, see SRP Section 4.4), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in Reference 21 and in Regulatory Guide 1.126 (Ref. 22). To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References 23 and 24.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 25. Interim methods that may be used prior to compliance with this guidance are given in Reference 26. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References 10, 12, and 13.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate, the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27). For non-LOCA applications, other correlations may be used if justified.

(h) Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 (Refs. 6, 28-30) as they relate to fission product release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by CPB (see Ref. 31).

D. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Reference 3

July 20, 1989

MEMORANDUM FOR: Frank J. Miraglia, Associate Director for Inspection and Enforcement

FROM: Robert B. A. Licciardo, Reactor Engineer
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: DIFFERING PROFESSIONAL VIEW (DPV) CONCERNING CONTAINMENT ISOLATION VALVES AT ZION

On May 11, 1989, The writer submitted a memo on the subject:

Differing Professional View Concerning

a) Issuance Of SER To Zion 1/2 Allowing Full Power Operation With Open 42" Containment Isolation Valves

b) Methodology Used For Calculating Related Offsite Doses

By memo of May 11, 1989, from F. J. Miraglia to R. Licciardo, the writer was asked to clarify certain aspects of the regulatory positions used in the analyses including the time to failure used in LOCA analyses and mechanisms for the transport of fission products from the primary (system) to the containment.

The writer was also asked to provide a view as to the safety significance of the Amendment proposed by management and the safety significance of my concern regarding LOCA analyses.

In response to the above request, I am pleased to submit the enclosed document which analyzes for your specific concerns and presents the related conclusions in Section 4.

Regarding the safety significance of the existing Zion Amendment proposed by management. Use of that Amendment and required Regulatory Guide 1.4 criteria would result in a contribution to thyroid dose over seven (7) secs. of 158,000 rem; using DNBR failure criteria with 10% fission product gap release would reduce this to 64,000 rem. Use of DNBR failure and equilibrium gap activity only would contribute 27,000 rem.

It would take a fuel failure of only 0.2% of the existing rods releasing 10% gap activity only to increase offsite doses to 10 CFR 100 limits.

~~8909140111~~ XA 2pp.

It must be recognized that allowing the containment purge valves to remain open for seven (7) secs. following a LOCA, multiplies by 194,000 the amount of fission product that would otherwise be release by leakage over the same period of seven (7) secs. from an isolated containment. It becomes a direct contradiction of the regulatory need for effective containment and limited leakage.

In summary: Proceeding with the existing Amendment proposed by management would be in direct violation of regulatory requirements.

The writer's SER of May 11 issued with his DPV of that date remains the writer's safety conclusions and recommendations in this matter i.e.:

"The 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequences of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body dose has not been evaluated. The least value for offsite dose to the thyroid which may be proposed within the existing licensing basis is 64,000 rem.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel failure and the quantification and treatment of the resulting source terms."

R Licciardo

Robert B. A. Licciardo
Registered Professional Engineer California
Nuclear Engineering License No. NU 001056
Mechanical Engineering License No. M 015380

Enclosure:
As stated

cc: J. Sniezek
C. Rossi
F. Congel
H. Smith

Enclosure

AN EVALUATION OF THE CRITERIA FOR

AND

THE CALCULATION OF OFFSITE DOSES DERIVING FROM OPEN CONTAINMENT PURGE VALVES DURING A LOCA AT ZION UNITS 1 & 2

DATED JULY 20, 1989

PREPARED BY

ROBERT B. A. LICCIARDO
REGISTERED PROFESSIONAL ENGINEER CALIFORNIA
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INTRODUCTION

On May 11, 1989, the writer submitted a memo on the subject:

DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance Of SER to Zion 1/2 Allowing Full Power Operation With Open 42" Containment Isolation Valves.

b) Methodology Used For Calculating Related Offsite Doses.

By memo of May 11, 1989, from F. J. Miraglia to R. Licciardo, the writer was asked to clarify certain aspects of the regulatory positions used in his analysis including: a) Time to failure used in LOCA analysis and b) mechanisms for the transport of fission products from the primary (system) to the containment. The writer was also asked to provide his view as to the safety significance of the Amendment proposed by management, and the safety significance of his concerns regarding LOCA analysis.

This material was prepared in response to that request and is in adjunct to his D.P.V which is attached to this document as Attachment 1.

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1 FISSION PRODUCT RELEASED FROM FUEL AND CONTAINMENT USED IN ACCIDENT ANALYSES

1.1 Radiological Source Terms Within The Core

Exhibit 1 shows core and gap activities for Zion for iodine.

- Calculated levels of iodine in the fuel clad gap are given to show a total I-131 EQU of 24.09×10^5 curies
- Total iodine in the core as I-131 EQU is 15.79×10^7 curies.

1.2 LOCA: Reg. Guide 1.4 Criteria: Application to Zion

Branch Technical Position CSB 6-4 (Ref. 25) states that:

> "The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes.
>
> The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guides 1.3 and 1.4 would not exceed the 10 CFR Part 100 guideline values. Also the maximum time for valve closure should not exceed five seconds to assure that the purge valves would be closed before the onset of fuel failures following a LOCA. Similar concerns apply to vent system designs."

This is interpreted by the writer as specifying that the large 42" purge and vent lines (PVLs) should be closed except in Modes 5 and 6. And if purging is necessary in Modes 1, 2, 3 and 4, then smaller lines (8" and 10") should be considered and the source term to be used for evaluating offsite dose is that of Reg. Guide 1.4 which uses TID 14844 source terms as the fission product available for release to containment.

RG 1.4.C Regulatory Position (Ref. 30) requires the following under related subsection No.:

"1a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides."

i.e., 25% of the radioactive iodine inventory from exhibit 1 is specified to be immediately available inside primary containment for leakage to the atmosphere. For Zion this would represent approximately 25 percent of 15.79×10^7 curies of I-131 EQU in the core i.e., 3.9×10^7 curies immediately available inside containment for leakage to atmosphere.

"1c. The effects of radiological decay during holdup in the containment or other buildings should be taken into account."

With half lives for iodine (I) varying from 3.16×10^3 secs for I-134 to 6.95×10^5 secs for I-131, released immediately on a LOCA, and a time to valve closure of seven (7) seconds, there is no time for significant radioactive decay of any iodine isotope before it is discharged to atmosphere.

It is to be noted that the actual first stage of fission product release during a LOCA occurs with the infringement of DNBR for the fuel rod, leading to overheating of the clad and fuel failure according to SRP 4.2 (Ref. 26) by perforation (or loss of hermeticity). For Zion, this is specified to occur 0.1 sec's into the event in the Appendix K evaluation of the LOCA event; the off-site calculations for this submittal have been made for a DNBR infringement of 1/2 sec. and are therefore less conservative.

"1d. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis."

During the first 7 seconds, there are no engineered safety features (ESF) fission product clean up systems available for reducing fission product content prior to discharge to the environment. Engineered safety feature containment sprays are initiated after 45 secs. Any filtration systems on the 42" inlet and outlet penetrations are not designed to ESF requirements. Recirculating filter systems provided by W for fission product control of containment atmosphere during normal operations are not ESF equipment.

Containment volume of 2 million cubic feet originally containing 144,000 lbs of air reduces fission product discharged from the RCS by prior dilution through mixing. Exhibits 3 and 4, and 3A and 4A show the circumstances for containment and the discharging reactor coolant system.

The containment has an initial mass of air of 144,000 lbs (at atmospheric pressure). On a LOCA, the initial rate of discharge from the RCS into containment is 75,000 lbs/sec and over a period of seven (7) seconds prior to containment valve closure, a total of 270,000 lbs is so discharged. This increases total mass in containment to 420,000 lbs, increasing total pressure in containment to 23.7 psig; at the same time a total mass of 15,000 lbs [valves fully open] to 2,860 lbs (valves partly open) of mixed containment inventory is discharged to the atmosphere.

If it is assumed that all fission product released from the core is immediately available to containment as in RG 1.4, then total mixing of this product should be assumed to occur on initiation of the LOCA. (The data presented show the results for a release ½ second after the LOCA, but the differences are not significant for the intent of this submittal.) As a result, containment inventory discharged contains a uniform concentration of a decreasing curie content over the first 7 seconds, and the net result is a release to outside containment of 4.38% of the source term fission product inventory Q, released from the core on occurrence of the LOCA. (A reduced amount of 1.57% is released for partly closed valves). Exhibit 2A shows that for the RG 1.4 source term, this gives a total release from containment over the first 7 seconds of 1.7×10^6 curies direct to atmosphere. Related offsite dose is 490,000 rem for 2 x fully open valves. Partially open valves reduce this to the value shown in Exhibit 2 of 612,000 curies and 156,000 rem.

It should be recognized that the thermal-hydraulic, including energy conditions, are such that fluid is discharging from both the RCS and the containment at very high energy levels, with associated pressure levels giving sonic discharge velocities into containment of the order of 1000 fps. Under these conditions it takes only hundredths of a seconds for RCS fluid to reach the containment isolation valves from the RCS system. This is no comparison with the very low transport rates from the top of a fuel pool to containment isolation valves for a fuel handling accident inside containment as discussed in Section 1.3.3.5 of this submittal; values of up to 15 secs. have been considered appropriate for these circumstances.

If is assumed that the core fission product source term is instead uniformly mixed with the RCS Fluid prior to its dicharge to containment, (less conservative than R.G. 1.4) curie content discharged to atmosphere is reduced from 4.38% Q to 1.9% Q where Q is the total term source released from the core by the LOCA and related source terms and related offsite doses are reduced by the same amount.

These are not unrealistic assumptions, for conservative purposes. The LOCA causes sudden pressure drops in the RCS, to saturation pressures for the prevailing temperatures of the RCS inventory, causing steam release from violent boiling throughout the system. This would cause substantial vibration of the fuel rods and movement of the prevailing damaged UO_2 pellets, facilitating the mass transfer of fission product gases to and through the gap to the locally faulted cladding, followed by blowdown through the clad defects at high rates because of the prevailing pressure drops, between the gap and the core.

Over the first seven seconds of the event, heat is being tranferrred from the core to containment by steam formation at the core and subsequent mass transfer to the RCS system and break, and discharge to the containment, at the very high rates discussed earlier in this subsection. Since fission product gases are released from the cladding, (and probably at the hottest sections) the transport of fission products released from the gap would be within the same steam and entrained liquid transport system to the break and then containment.

Within containment, unless special provisions have been made, there is no guarantee that a certain percentage of high concentrations of fission product inventory being released by RCS discharge is not being bypassed directly to the open containment isolation valves from its main path to principal containment volume. In this sense, assuming an immediate release of all fission product to the containment on DNBR would help offset the potential non-conservatism of this bypass.

"1e. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours [0.1 percent per day], and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing."

Except for dilution through mixing discussed under 1d above, there is complete bypass of containment for 7 secs through the 2 x 42" open valves.

The magnitude of discharge to the environment with related offsite doses has been discussed under 1d above. In reviewing these figures, it should be recognized that for a normal leakage of 0.1%/day from containment, 8×10^{-6}% of Containment Inventory (Q), would be released in the same time frame of 7 seconds. When compared with 4.38%, this represents a dose reduction factor of <u>541,000</u> and would reduce the 7 second dose from 489,000 rem to 0.9 rem.

Over a two hour time frame, and making allowance for 38 seconds without spray, followed by an iodine removal coefficient of 54/hr with a maximum reduction factor of 100, gives an approximate reduction in discharge by a factor of <u>32,000</u> leading to a calculated dose of 15 rem.

These reduction factors in offsite dose of 489,000 for the first seven seconds by effective early containment at 0.1%/day, and of 32,000 in the first 2 hours by effective containment at 0.1% per day and an iodine cleanup factor of 100, manifest the real significance of effective containment and containment spray in fission product containment.

1.3 LOCA: BTP CSB 6-4, B5 Criteria

The Reg. 1.4 source terms of 1.2 above, are based upon the Regulatory requirement of 10 CFR 100.11, (a) footnote 1 (Ref. 36) that:

"The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

However, Branch Technical Position CSB 6-4 (Ref. 25) provides another basis to justify containment purge design and which is less conservative than the Regulatory position. This is given in related section B-5, as:

"5. The following analyses should be performed to justify the containment purge system design:

a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values."

To gain further regulatory interpretation of the meaning of fuel failure within this context, the writer's DPV (Ref. 42) refers to SRP 4.2 FUEL SYSTEM DESIGN, I (AREAS OF REVIEW), 2nd para. (Ref. 26) which states that, in respect of postulated accidents:

"The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 38), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents."

The underscored lines show that fue rod failure in the context of this paragraph must be evaluated for postulated accidents and this evaluation must be conservative Fuel Rod Failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has therefore been breached; these failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 36) for postulated accidents.

Coolability is addressed as a separate criterion.

1.3.1 Characteristics of Fuel Failure Giving Fission Product Release During Postulated Accidents

Regulatory clarification of fuel rod failure is given in SRP 4.2.II.A.2. (Ref 26) This is abstracted as follows for the circumstances of postulated accidents in particular:

"2. FUEL ROD FAILURE

This subsection applies to [~~normal operation, anticipated operational occurrences, and~~] postulated accidents. [~~Paragraphs (a) through (c) address~~

~~failure mechanisms that are more limiting during normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.~~] Paragraphs (d) through (h) address failure mechanisms that are more limiting during (anticipated operational occurrences and) postulated accidents, [~~and the information to be reviewed will usually be contained in Chapter 15 of the Safety Analysis Report. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere~~]

To meet the requirements of [~~(a) General Design Criterion 10 as it relates to Specified Acceptable~~ Fuel Design Limits for normal operation, including antici~~pated operational occurrences, and (b)~~] 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. [~~Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences.~~] Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Fuel failure criteria should address the following to be complete.

Only those failure mechanisms that are more limiting for postulated accidents are abstracted here:

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs [and ~~CPR for BWRs)~~] are satisfied. [~~The review of these criteria is detailed in SRP Section 4.4. For normal operation and anticipated operational occurrences, violation of the thermal margin criteria is not permitted.~~] For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes.

Although a thermal margin criterion is sufficient to demonstrate the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: [~~It has also been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and anticipated operational occurrences, centerline melting is not permitted.~~] For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. [~~The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots.~~] The assumption that centerline melting results in fuel failure is conservative.

(f) Excessive Fuel Enthalpy: [~~For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/g at any axial location.~~] For full-power RIAs in a BWR and all RIAs in a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 (Ref. 6) as it relates to fuel rod failure. [~~The 170 cal/g enthalpy criterion is primarily intended to address cladding overheating effects, but it also indirectly address pellet/cladding interactions (PCI).~~] Other criteria may be more appropriate for an RIA, but continued approval of [~~this enthalpy criterion and the~~ thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude

PCI failures. (1) The uniform strain of the cladding should not exceed 1%. [~~In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady state creepdown and irradiation growth are excluded.~~] Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

(h) Bursting: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to incidence of rupture during a LOCA, [~~a rupture temperature correlation must be used in the LOCA-ECCS analysis.~~] Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. [~~Although fuel suppliers may use different rupture temperature vs differential pressure curves, an acceptable curve should be similar to the one described in Ref. 10.~~]

(i) Mechanical Fracturing: A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core plate motion. Cladding integrity may be assumed if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from seismic and LOCA analysis (Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events."

Summary:

Failure Mechanisms include:

(a) Infringment of DNBR criteria during postulated accidents which causes overheating of the cladding of the fuel rod, and is assumed to cause failure of the clad, and release of contained fission products from the gap as a source term for the calculation of radiological doses.

(b) If postulated accident conditions cause calculated values of fuel pellet temperature to reach the melting point for the uranium dioxide at the centerline of the pellet, it is assumed that all such rods shall fail (and release fission products from the pellets - as well as the gap) for the calculation of radiological doses.

1.3.2 Characteristics of Fission Product Released From Failed Fuel During Postulated Accidents

1.3.2.1 General

Fission product release as source terms for postulated accidents relevant to the above fuel failure criteria are specified as:

SRP 4.2, Section I, last paragraph (<u>Ref. 26</u>) states that:

> "All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in SRP Section 4.4. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Evaluation Branch for use in estimating the radiological consequences of plant releases."

SRP 4.2.C.3(h) (<u>Ref. 26</u>) states that:

> "Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 [~~Refs. 6, 28-30~~] as they relate to fission product release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by CPB [~~see Ref. 31~~]."

The criteria from these Reg Guides are considered separately in the following subsections of this submittal in order to examine for general guidelines which may be applied to BTP CSB 6-4 B5 Criteria.

1.3.2.2 Regulatory Guide (RG) 1.25: Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

RG 1.25 (Ref. 31) covers the Fuel Handling Accident inside containment.

RG 1.25 page 25.1 under Section B, second para. provides for an immediate release of all activity from the fuel rod gap of the damage rods:

> "The number and exposure histories of fuel assemblies assumed to be damaged determine the total amount of radioactive material available for <u>immediate release into the water</u> during a fuel handling accident."

The same Section B, fourth para. provides that:

> "Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation would be available for immediate release into the water in the event of clad damage. (Migration of fission products is a function of several variables including operating temperature, burnup, and isotopic half life taken into consideration in establishing the release fractions listed in this guide.)"

RG 1.25 also assumes that 10% of the total radioactive iodine in the rod (with calculated peak activity) is contained in the gap for release. (See page 25.2, Item C.1.d):

> "All of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident.

Released iodine rises to the surface of the related pool with a decontamination factor of 100, provided a minimum depth of 25 ft exists, and gap pressure is no greater than 1200 psig. Subsequent treatment of the source term is typified by the guidelines of SRP 15.7.4 Radiological Consequences of Fuel Handling Accidents (Ref. 28) which requires (under Section III.4, second and third para's that:

"The reviewer should assess the time required to isolate the containment. This should include the instrument line sampling time (where appropriate), detector response time and containment purge isolation valve actuation and closure time. The containment is considered isolated only when the purge isolation valves are fully closed. The applicant's analysis should be reviewed regarding the travel time of any activity release starting from its release point above the refueling cavity or transfer canal and including travel time in ducts or ventilation systems up to the inner containment purge isolation valve."

"The time required for the release to reach the inner isolation valve is compared to the time required to isolate the containment. If the time required for the release to reach the isolation valve is longer than the time required to isolate containment, then essentially no release to the atmosphere occurs, and the reviewer's assessment should reflect this. If the time required for the release to reach the isolation valve is less than that required to isolate containment, and no mixing or dilution credit can be given, the reviewer should assume that the entire activity release escapes from the containment in evaluating the consequences. Claims for credit for dilution or mixing of a release due to natural or forced convection inside containment are reviewed and assessed. References [4] and [5] should be consulted and used by the reviewer for guidance in estimating dilution and mixing. Where mixing and dilution can be demonstrated within containment, the radiological consequences will be reduced by the degree of mixing and dilution occurring prior to containment isolation."

Related references [4] and [5] are:

"4. Evaluation of Fission Product Release and Transport for a Fuel Handling Accident by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, revised October 5, 1971.

5. Industrial Ventilation/A Manual of Recommended Practice - American Conference of Governmental Industrial Hygienists."

These circumstances relate to a set of containment environmental conditions in which mixing energy is virtually absent, being provided by low energy containment purge and exhaust ventilation fans, and virtually no additional energy from the very small mass of fission product gas released from the damaged fuel elements, after travelling through a minimum depth of 23 ft. Under certain conditions, this could provide for the total activity released (after decontamination in the pool) to be discharged directly to atmosphere outside containment.

For Zion, the fundamental set of values for the thermal hydraulic parameters covering the above circumstances, are completely different to those governing the release and disbursement of fission products to the environment from a LOCA.

1.3.2.3 Regulatory Guide 1.77: Assumptions Used for Evaluating a Control Rod Ejection Accident For Pressurized Water Reactors

Fundamentally, this Guide provides for an evaluation of the Thermal Hydraulic and Power conditions within the core, during the accident, to determine a) the extent of DNBR infringement and b) the amount of fuel exceeding the initiation temperature of fuel melt (approximately 5150°F).

For Source Terms, RG 1.77, Appendix B1 (Ref. 32) proposes that:

"a. The case resulting in the largest source term should be selected for evaluation.

b. The nuclide inventory in the fuel elements potentially breached should be calculated, and it should be assumed that all gaseous constituents in the fuel-clad gaps are released.

c. The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life, assuming continuous maximum full power operation.

d. No allowance should be given for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

e. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature for fuel melting (typically 2842°C) at any time during the course of the accident should be calculated, and 100% of the noble gases and 25% of the iodine contained in this fraction should be assumed to be available for release from the containment."

Summarily: The source term from molten fuel is the same as for RG 1.4. The source term release from the gap is the same as for the fuel handling accident.

The subsequent effects of the release path on the ultimate source terms from containment are evaluated for each of two release paths, as if the other did not exist. These release paths are:

(1) By effectively immediate release of all source terms to containment to be followed by the following cleanup and decay provisions which are the same as those normally accounted for in a LOCA in RG 1.4 (Ref. 30). RG 1.77, App. B1 (Ref. 32) provides that:

"f. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

g. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on a case-by-case basis.

h. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing."

Additionally SRP 15.4.8, Section III.3 (Ref. 27), further specifies that:

"For releases via the containment building, 100% of the noble gases and 25% of the iodines contained in the fuel which is estimated to reach initiation of melting are assumed to be available for release from the containment."

Summarily: For the release path to containment, these are effectively the provisions of RG 1.4 in respect of the treatment of Fission Product Source Terms after release from the core.

(2) By release of fission products to the secondary system as per RG 1.77, Appendix B, Items 1i, j and k (Ref. 32).

There are not considered in this submittal, as they do not apply to a release to containment.

1.3.2.4 Summary (of General Positions on Fission Product Releases Deriving from RG 1.25 and 1.77)

(a) For failure of fuel cladding by either DNBR infringement or fuel handling accident:

For iodine, 10% of the fuel rod inventory is released from the gap. For the control rod ejection accident this release is assumed to be available immediately inside containment for leakage.

(b) For failure by centerline melting of the fuel pellet:

25% of the iodine inventory of any fuel rod which reaches or exceeds the initiation temperature of fuel melting is assumed to be immediately available inside containment for release. This is the same assumption applied in RG 1.4 for fuel melt deriving from a LOCA.

1.4 LOCA: BTP CSB 6-4/B5 Criteria: Application to Zion

Zion Fuel temperatures during normal operation at maximum power prior to a LOCA vary from 2500°F to 4100° for approximately 15% of the core (Exhibit 23). There will be a substantial increase in temperature of the whole core over a period of up to 7 seconds following a LOCA and Exhibit 6 shows the related average cladding temperatures. Considering the correlation of fission product release as a function of temperature shown in Exhibit 22, there is a high probability of a substantial increase in fission product activity in the gap over that of the equilibrium activity level represented on Exhibit 1, during these first seven (7) seconds of the accident, so that an increase in gap activity level from the equilibrium values shown in Exhibit 1 to the value of 10% used in the other postulated accidents is not an unreasonably conservative regulatory position to adopt for this event. On this basis, the iodine source term deriving from fuel rod failure by overheating of the fuel cladding by DNB infringement at Zion at 0.1 second into the event would be 157.9×10^5 curies of I-131 EQU and is the value adopted by the writer in conformance to the related BTP. In respect of fuel rod failure by centerline melting, the Zion FSAR (Ref. 33) does not provide detailed information on fuel pellet temperatures except for the general statement that the safety injection system prevents core meltdown Ref. 33, page 14.3-46, Revision 1 second para.; provision for related fission product release from melted fuel rods is therefore not necessary for this evaluation to the guidance of the related BTP.

On the basis of BTP CSB 6-4, B5 therefore, a total iodine fission product release of 157.9 x 10^5 curie I-131 EQU from the core, would be available to inside containment at 0.1 second into the LOCA. By reference to the conditions inside containment discussed in detail in Section 1.2, items 1d and 1e above, it can be shown that, the release of 157.9 x 10^5 curies of I-131 EQU from the core as a source term will result in the discharge of 692,000 curies of I-131 EQU to atmosphere with an offsite dose of 176,000 rem with 2 x 42" fully open for 7 seconds, see Exhibit 2A, item 5. With valves partly closed this is reduced to 249,000 curies I-131 EQU and 63,400 rem, see Exhibit 2 item 5.

It is noted that in its recent revision to the FSAR (Ref. 34) page 14.3-38 Revision 1. W has calculated an offsite dose from the LOCA on a non-Reg. Guide 1.4 basis, by also using the entire inventory of fission products contained in the pellet cladding gap, but has assumed the equilibrium values only, as listed in Exhibit 1. This is equal to 24.09 x 10^5 I-131 EQU which is 1.52% of the core activity as compared with the 10% exemplified in other NRC criteria and used by the writer. Effective doses that would be obtained using equilibrium gap activity only are also presented in Exhibits 2A and 2 under items 4 and show offsite doses to thyroid are reduced to 27,000 rem for 2 fullopen valves and 9,700 rem for 2 partially closed valves.

2 OFFSITE DOSE CONSEQUENCES: SUMMARY

2.1 Basis for Calculations

Based on discussions in section 1, radiological releases and related offsite consequences are shown in Exhibit 2A item 6 for 2 x 42" fully open (90°) valves and Exhibit 2 item 6 for 2 x 42" values at a limited opening of 50°.

All calculations are based on valves closing in 7 seconds from commencement of a LOCA. Doses are based upon valves being in the open position for a full 7 seconds as required by the SRP. Valves will be required by technical specifications to close within seven (7) seconds of commencement of the LOCA.

For the sake of example only, source terms are restricted to iodine in terms of I-131 EQU, and thyroid dose only has been calculated. Dose is calculated at the site boundary (exclusion distance) of 415 meters. Each dose is calculated independently of each other and are to be added to the LOCA leakage dose (over 2 hours) of 123 rem as appropriate.

An additional dose due to RCS inventory discharged into the containment would also need to be added, for all non-RG 1.4 calculations. These are given in Exhibits 2A and 2 under items 2 at 132 rem for 2x fully open valves, and 48 rem for 2 partially opened valves.

For the diffusion coefficient, a value of 5×10^{-4} sec/cm^3 applicable to leakage conditions over a 2 hour period has been used. In fact we have a high energy puff release of 7 secconds giving a potential finite cloud in travel to the enclosure boundary instead of a low leakage release diffusing into a cloud; as a result, the offsite dose under actual conditions is likely to be increased. For the 0-2 hour leakage, the licensee has used a more conservative value than the NRC of 9.2×10^{-4} sec/cm^3 and this would increase dose by a factor of 1.84.

2.2 Offsite Doses

2.2.1 RG 1.4 Source Terms Released Immediately on LOCA

Exhibit 2A, item 6, shows that for fully (90°) open 42" valves, the offsite dose for a RG 1.4 source term is calculated at 489,000 rem. And Exhibit 2, item 6, shows that for partially (50°) open 42" values, these doses are reduced to 156,000 rem.

2.2.2 10% Gap Activity Released on DNBR

Exhibit 2A (item 5) shows offsite doses reduced to 176,000 rem for fully open valves, and Exhibit 2 (item 5) shows reduction to 63,000 rem for partially open valves.

2.2.3 Equilibrium Gap Activity Released on DNBR

Exhibit 2A (item 4) shows offsite dose is reduced to 27,000 rem for fully open valves and Exhibit 2 (item 4) shows reduction to 9,700 rem for valves partially open.

2.2.4 RCS @ 60 μc/gm Activity; All Released To Containment Immediately On A LOCA.

Exhibit 2A (item 2) shows offsite dose contribution is 132 rem for fully open valves and Exhibit 2 (item 2) shows a reduction to 48 rem for partially open valves.

This activity release is equivalent to DNBR infringement of only .08% of the fuel in the core.

2.2.5 RCS @ 60 μc/gm Activity; Released Progressively To Containment On RCS Discharge From A LOCA

Exhibit 2A (item 3) shows offsite dose contribution is 58 rem and Exhibit 2 (item 3) shows a reduction to 21 rem for partially open valves.

2.2 Conclusions

(1) According to Reg. Guide 1.4 criteria the offsite doses are completely unacceptable.

(2) LOCA calculations for Zion show no fuel melt; however, for DNBR infringement only, an evaluation of offsite dose based on release of 10% gap activity from 100% fuel still shows completely unacceptable circumstances.

Although this is in conformance with SRP 6-4, BTP, CSB B5 criteria, it is not in conformance with 10 CFR 100.11 (a) footnote 1 requirements which states that:

> "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

(3) Partially closing the valve to 50° from 90° is not successful in reducing the offsite dose to acceptable values.

(4) With valves partially open at 50°; fuel failures by DNBR infringement on a LOCA would have to be limited to 0.2% of the core to limit total doses to 10 CFR 100 limits.

3 APPENDIX K EVALUATIONS, FUEL FAILURE, AND FISSION PRODUCT RELEASE

10 CFR 50.46 (Ref. 37), acceptance criteria for emergency core cooling system for light water nuclear power reactors, requires that during a LOCA, cladding temperatures, cladding oxidation, and hydrogen generation, are limited and such that the core remains amenable to cooling in the short run from the initial break through reflood, and also for long term post accident cooling.

10 CFR 50.46 does not include a requirement to evaluate the earliest time at which fission products could be released by local failure of the fuel cladding as fuel rod conditions rapidly change, challenge and exceed the limiting features of design which ensures fuel clad (and rod integrity) under Normal Operating Conditions and Transient Occurrences. These limiting features are described as specified acceptable fuel design limits (SAFDLs) and are required under 10 CFR Part 50, Appendix A, Criterion 10.

A principal feature of the Appendix K evaluation is that it is designed to identify that rupture which causes a maximum post rupture cladding temperature within the fuel assembly being evaluated; and it is this time to rupture which is reported in the FSAR. The Appendix K evaluation is not designed to report the earliest rupture that can occur.

3.1 Preliminary

In evaluating 10 CFR 50.46 criteria through the use of the Appendix K evaluation model (Ref. 39), licensees are required to undertake a detailed evaluation of the items to be discussed below throughout the complete LOCA event, i.e., from time 0 through 50-60 seconds, to determine that the clad rupture meeting the Appendix K criteria does not occur in the first 10-15 seconds of the event, and which is the region of special interest for this review. In the time available for this research, a search of the UFSAR and the related reference material on the docket does not disclose many of essential the details of this calculation (Ref's 1-24). We therefore use the limited information available to draw conclusions.

3.2 Review

Appendix K calculations are undertaken on that fuel element assembly which ultimately provides the maximum clad temperature after (post) clad rupture.

Generic work by W (Ref. 17) proposes that maximum calculated temperatures (post rupture) occurs in the low burn up (third region) fuel assembly.

Exhibit 6 shows the average clad temperatures deriving from Appendix K calculations from the Zion FSAR, Figure 14 F. 2-19a, (Ref. 33). This shows that on infringement of DNBR at 1/10 second, average clad temperature increases very rapidly from a normal operating value of 720°F to at least 1350°F, and then to 1750°F, over a total period of seven seconds; thereafter temperature reduces rapidly to 1000°F at about 15 secs. from which it sharply increases ultimately to approx 2200°F.

Exhibit 10 shows that W fuels are designed to require a yield strength of 45,000 psi a minimum for normal operations, and an ultimate tensile strength of 57,000 psi as a damage limit, as specified acceptable fuel design limits (SAFDL). Exhibit 11 shows that as temperatures increase above 850°F, the available mechanical properties can be reduced below both these limits so that fuel clad cannot therefore be considered reliable in terms of protection against fission product release.

Exhibit 10 also shows that W fuels require a design limit of 1% on cladding strain as a design limit, and 1.7% as a damage limit. The work of this Section 3 will show how both these limits can be exceeded inside the seven seconds on infringement of DNBR during the course of a LOCA, so that again, fuel clad cannot be considered reliable in terms of protection against fission product release.

Exhibit 15, shows how a temperature range of 1350°-1750°F traverses a range of Zircalloy metallurgical phases (transitions), α to ($\alpha + \beta$) to β phases, during which ys = UTS and structural stability under stress is dependent upon mechanical/strength properties which are a function of temperature and related time and stress at temperature. Under the circumstance of the transient expected from Appendix K calculations with rapid changes of both temperature and stress, their is a need for empirical tests to determine swelling and burst (rupture) characteristics under these same dynamic conditions. Exhibit 15 represents results from such a series of tests (Ref. 13).

Such conditions are also represented in Exhibit 16 for Engineering Hoop Stress and temperature at rupture, for particular heating rates, and in conjunction with the information in Exhibit 20 on related rates of circumferential strain on rupture, at the given rupture temperatures.

What are the expected operating pressure differentials across the clad under these LOCA conditions:

Reference information shows that internal clad pressure under normally operating conditions is of the order of 1400 psig for new fuel and expected to increase to 2250 psig at the end of the 3rd cycle (for the fuel). On this basis, we evaluate a gap pressure of 1500 psig at approximately 1/3 burnup into the first cycle, at which burnup maximum calculated clad temperatures are expected on a LOCA.

It is proposed that, immediately on a LOCA as clad temperature increases to 1350°F, gap pressure will increase by 20%, to 1800 psig. Exhibit 12 shows that at this time, core pressure has reduced to 1500 psig giving a pressure drop across the clad of 300 psi which according to Exhibit 13 will give a hoop stress of approximately 2460 psi.

At 7 seconds into the event, clad temperature has increased further to 1750°F, a total increase of 1030°F from the normal operating condition. From this, it can be proposed that gap pressure for the complete rod can increase by 36% over its normal operating value to 2100 psig. Exhibit 12 shows that at this time, core pressure has reduced to 950 psig so that the pressure drop across the clad is now 2100-950 i.e., 1150 psi which according to Exhibit 13 will give a hoop stress of 9400 psi.

When the above values of pressure and temperature are plotted on a particular Hoop Stress vs Burst Temp curve (Exhibit 14) from reference 1, at one sec the clad does not rupture, but at seven seconds the clad is well into the rupture regime.

In its calculation of clad strain during Appendix K calculations, W uses results from tests by Hardy (Ref. 13). Exhibit 15 is a set of results from one such test at 100°C/sec heat up rate (the heat up rate between 720°F and 1750°F in 7 seconds = 150F°/second [or 84C°/second]). This exhibit shows that these Appendix K values over the first 7 seconds bracket the range from zero (0) expansion at 1350°F to the burst regime at 1750°F. In respect to these values, W has assumed that if clad strain reaches 10%, the clad will rupture; see Exhibit 18 from Ref. 3. Note that the SAFDLs of 1% and 1.7% on cladding strain can both be exceeded in the first seven seconds of DNBR infringement in the course of the LOCA.

The NRC, in its clad strain and rupture models uses the data shown in Exhibit 16 to determine when rupture is likely to occur for given rates of increase in temperature. It is proposed by the NRC that the 28°C/S (=50F°/second) test points apply also to larger values (of rate of temperature increase). Exhibit 16 shows that the Appendix K values again bracket the complete set of experimental data and significantly at the higher temperatures of the transient.

Exhibit 20 shows the circumferential strain that can occur at given rupture temperatures, and the curve proposed by the NRC for Appendix K calculations. Prime Facie; maximum strain gives maximum blockage leading to maximum calculated temperatures for cladding after the burst. In fact, W has established that maximum post rupture cladding temperature does not necessarily occur with a maximum circumferential strain at rupture, due apparently to direct radiation influences from fuel rods exposed by rupture at lesser values. Providing rupture is expected by the data of Exhibit 16, the related strain is to be given by the NRC curve on Exhibit 20 (or lesser value giving maximum temperature). It should be noted that with this information there would be a very high probability of rupture at 1750°F down to 1500°F, with the probability decreasing, but still present at lower temperature.

Note that Exhibits 16 and 20 do show that fuel temperatures and pressures could rupture the cladding over a whole range of conditions. However, the purpose of the Appendix K evaluation is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirements of 10 CFR 50.46 and for this end, it models, and uses factors, to conservatively calculate values for the related parameters. Its purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released. An example can be seen from Exhibit 16. The test points can show marked deviations from what are apparently best estimate curves for the various rates of temperature increase. For conservatism in estimating the first occurrence of fuel rupture, one would have presumed the use of a boundary curve at the lower temperatures and pressures of each heating rate and Exhibit 20 would not have been required.

Note that Exhibit 15 does show that even though rupture may not occur with a detailed re-evaluation, cladding strain is most likely to exceed the 1% strain used by W (Ref 33, P. 3.2-39) as a SAFDL to meet the regulatory requirements of Ref. 38.

The writer would be concerned about the relevance of the hoop stress, strain/rupture data of Exhibits 16 and 20 to the power generation and heat transfer conditions inside a reactor. These tests were done on electrically resistance heated cladding tubes. They do not simulate the heat transfer from central fuel rod pellets at high temperatures through a realistic gas gap of varying geometry, fuel pellet-clad contact, and pellet fracture/fragmentation to a cladding which is 12 ft long and which is likely to have a much smaller ratio of rupture length to clad length and gap volume than the test specimens. The most revealing feature of Exhibit 16 is the data from the only test undertaken under much more realistic conditions, on a nuclear fuel rod using Zircalloy cladding in the TREAT reactor at ORNL; this information shows ruptures at very much reduced stress levels than the rest of the data.

3.3 Summary

1. Conditions within the core as currently evaluated by the Appendix K model, show that over the first seven (7) seconds following a LOCA, the following significant events occur:

1.1 DNBR for the whole core is infringed at 1/10 sec requiring gap activity at 10% core inventory for the whole core to be assumed as a source inside containment.

1.2 The temperature of the fuel clad, and the pressure drops across the same fuel clad, infringe specified acceptable fuel design limits (SADL) for normal operation and operational occurrences, required by 10 CFR 50 Appendix A, Criterion 10. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

1.3 The temperature of the fuel clad and the related pressure drops show conditions in which substantial deformation of the fuel clad by strain, can exceed the design and damage SAFDL values for cladding strain. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

1.4 The temperature of the fuel clad and the related pressure drops show conditions which could result in fuel rupture. This conclusion would need to be subject to detailed verification using the Appendix K model.

1.5 For Zion, fuel rods do not reach the melting point of the fuel pellets so that under minimum engineered safeguard conditions, additional fission product release from the fuel rods would not occur.

2. The writer proposes that the purpose of Appendix K is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirement of 10 CFR 50.46 and for this end it models, and uses factors, to calculate values for the related purposes. The purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released from the fuel summary a LOCA.

4 CONCLUSIONS

1. Conditions within the core as currently evaluated by the Appendix K model, show that over the first seven (7) seconds following a LOCA, the following significant events occur:

   1.1 DNBR for the whole core is infringed at 1/10 sec requiring gap activity at 10% core inventory for the whole core to be assumed as a source inside containment.

   1.2 The temperature of the fuel clad, and the pressure drops across the same fuel clad, infringe specified acceptable fuel design limits (SADL) for normal operation and operational occurrences, required by 10 CFR 50 Appendix A, Criterion 10. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

   1.3 The temperature of the fuel clad and the related pressure drops show conditions in which substantial deformation of the fuel clad by strain, can exceed the design and damage SAFDL values for cladding strain. Fuel rod failure must therefore be assumed for conservative calculations of offsite dose.

   1.4 The temperature of the fuel clad and the related pressure drops show conditions which could result in fuel rupture. This conclusion would need to be subject to detailed verification using the Appendix K model.

   1.5 For Zion, fuel rods do not reach the melting point of the fuel pellets so that under minimum engineered safeguard conditions, additional fission product release from the fuel rods would not occur.

2. The writer proposes that the purpose of Appendix K is to identify that particular rupture which would have the most conservative effect with respect to meeting the requirement of 10 CFR 50.46 and for this end it models, and uses factors, to calculate values for the related purposes.

The purpose is not to determine and identify when failure by bursting (rupture) first occurs as an otherwise evaluation of when fission product is first released from the fuel summary a LOCA.

3. As a result of the above

3.1 Fission product release from the fuel gap is a realistic consideration over the first seven seconds and prudent conservatism at this time should consider release from the whole core.

3.2 Reg Guide 1.4 deriving from Regulatory Requirement 10 CFR 100 requires consideration of substantial molten fuel as a design for the source term.

4. The writer proposes that Regulatory philosophy recognized the possibility of Beyond Design Basis Events as the realism of a substantial commercial industry and therefore required protection against this occurrence and made provision in the Regulations for this purpose.

Considering the energy exchanges occurring in the core, and the insight of the Appendix K evaluations, it is not difficult to foresee significant fuel melt with potential additional substantive release of fission products from the fuel pellets over this time frame. The question of the separate consideration of the timing of this additional contribution to the source term inside containment however must be moot. Uncontrollable release through open 42 inch CIVs is out of the question so that steps taken to correct that problem by effective isolation do resolve the unanswered philosophical question as to when fission products released by fuel melt should be more realistically and conservatively established.

4.1 A review of available fuel failure criteria, and the thermal-hydraulics aspects of the movement of fission gases from the clad to the environment over the first seven seconds of the event shows that:

(a) The assumption of an immediate release to the containment is the only available conservative basis for use at this time, and that

(b) The physics of the large energy releases from the core clad through the RCS to containment, and through the open isolation valves, shows effective mass transfer of fission product release from the clad to the environment within the same (7) secs.

5. Fully open purge valves for a period of seven (7) secs. discharge 1.7×10^6 curies of I^{131} EQU to the environment giving an offsite dose of 489,000 rem to thyroid.

   An isolated containment leaking at the safety analyses and TS limit of 0.1% over 24 hrs, releases 3.14 curies of I^{131} EQU over the same seven seconds with a contribution to offsite dose of 0.9 rem.

   The effectiveness of containment isolation and effective leak tightness in achieving a clean up factor of 541,000 over the first seven seconds of the LOCA is manifest.

6. The offsite dose to thyroid for fully (90°) open 42" valves using RG 1.4 source terms is calculated at 489,000 rem. For partially (50°) open 42" valves, these doses are reduced to 156,000 rem. Reduction of source terms from RG 1.4 to 10% gap activity released on DNBR infringement reduces offsite dose to 176,000 rem for fully open valves with a reduction to 63,000 rem for partially open valves.

   Since the allowable limit for thyroid under 10 CFR 100 is 300 rem for 2 hrs at the Exclusion Boundary, these circumstances are unacceptable. Therefore the 42" valves at Zion 1 and 2 should remain closed in Operational Modes 1, 2, 3 and 4.

7. The stress/temperature relationships used to calculate fuel clad rupture to 10 CFR 50.46 are derived from test environments which are substantively non-realistic when compared with actual fuel rod conditions in a reactor

during a LOCA. The only in-reactor tests known to the writer at this time with the closest simulation of a real fuel condition gives ruptures at very much reduced pressures for given rupture temperatures. This comparison needs to be revisited to more thoroughly evaluate the reasons for the differences and thereby improve our detailed knowledge of the total heat transfer environment which can lead to improvements in the calculational models of the fuel assemblies used in the Appendix K evaluations. This can help in a improved definition of the limiting features of the circumstances and lead to ways and means of improving fuel clad design and performance for these circumstances.

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4. Letter from E. P. Rahe (W) to M. Lauben: Subject: Millstone Large Break Results for Increasing Fuel Burnup. July 20, 1981.

5. Letter from T. M. Anderson to J. Stolz: Subject: Additions to WCAP9220.

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7. Rahe, E. P., "Westinghouse ECCS Evaluation Model 1981 Version," WCAP-9220-p-A Rev. 1 (Proprietary), WCAP-9221-P-A Rev. 1 (Non-Proprietary).

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12. Lorenz, Hobson & Parker. ORNL. Fuel Rod Failure Under Loss-of-Coolant Conditions In TREAT. Nuclear Technology, Vol II, May 1971-August 1971.

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18. Letter from T. M. Anderson (W) to J. F. Stolz (NRC); Subject WCAP-8963-P-A. Safety Analyses for the Revised Fuel Rod Internal Pressure Design Basis (Prop).

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20. W. D. Leech, et al., "Revised PAD Code Thermal Safety Model," Westinghouse Report WCAP8720, Addenda 2, October 1982.

21. Safety Evaluation of the Westinghouse Electric Corporation Topical Report WCAP-9500. "Reference Core Report 17 x 17 Optimized Fuel Assembly." May 1981, Core Performance Branch, Reactor Systems Branch.

22. Letter from T. N. Anderson (W) to J. R. Miller (NRC); Subject: Proprietary Responses to the NRC request for additional information on WCAP-9500 (Proprietary). January 22, 1981.

23. Letter from Robert L. Tedesco, to Westinghouse Electric Corporation, Attention: T. M. Anderson; Subject: Acceptance for Referencing of Licensing Topical Report WCAP-9500.

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26. NUREG-0800, Standard Review Plan, July, 1981, Section 4.2, Fuel System Design.

27. NUREG-0800, Standard Review Plan, July 1981, Section 15.4.8. Radiological Consequences of a Control Rod Ejection Accident (PWR) Appendix A.

28. NUREG-0800, Standard Review Plan, July 1981, Section 15.7.4. Radiological Consequences of Fuel Handling Accidents.

29. USNRC, NUREG-75/077, The Role of Fission Gas Release in Reactor Licensing, November 1975.

30. USNRC, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors.

31. USNRC, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

32. USNRC, Regulatory Guide 1.77, "Assumptions Used for Evaluating A Control Rod Ejection Accident for Pressurized Water Reactors, May 1974.

33. ZION 1 & 2: Updated Final Safety Analyses Report. Commonwealth Edison.

34. Deleted

35. U.S. Government Printing office, "Reactor Design," Criterion 10, Appendix A, "General Design criteria for Nuclear Power Plants," Part 50, Title 10 Energy, Code of Federal Regulations, January 1, 1979.

36. Ibid., "Reactor Site Criteria," Part 100.

37. Ibid., "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," Part 50.46.

38. Ibid., "Reactor Design", Criterion 10, Appendix A, part 50.

39. Ibid., Appendix K Evaluation, Part 50.

40. Memo from R. Licciardo (NRC) to T. E. Murley (NRC): Subject; Differing Professional View concerning (A) Zion 1/2 Containment Isolation Valves and (B) Methodology Used for Calculating Related Offsite Doses. Dated May 18, 1989.

EXHIBITS

OF

BACKGROUND INFORMATION RELATED TO

DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance of SER to Zion 1/2 allowing full power operation with open 42" containment isolation valves.

b) Methodology used for calculating related offsite doses.

ZION

CORE AND GAP ACTIVITIES (IODINE ONLY)

Assumptions: Operation at 3391 MWt for 500 days

Isotope	Curies in the Core ($X 10^7$)	I 131 EQU $x 10^7$	Equilibrium Percent of Core Activity in the Gap	Curies in the Gap ($X 10^5$)	I 131 EQU ($X10^5$)
I-131	8.35	8.35	2.3	19.2	19.2
I-132	12.75	.46	0.26	3.3	.12
I-133	19.09	5.16	0.79	15.1	4.08
I-134	23.01	.39	0.16	3.8	.06
I-135	17.05	1.43	0.43	7.5	.63
		15.79			24.09

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES OPEN 50°

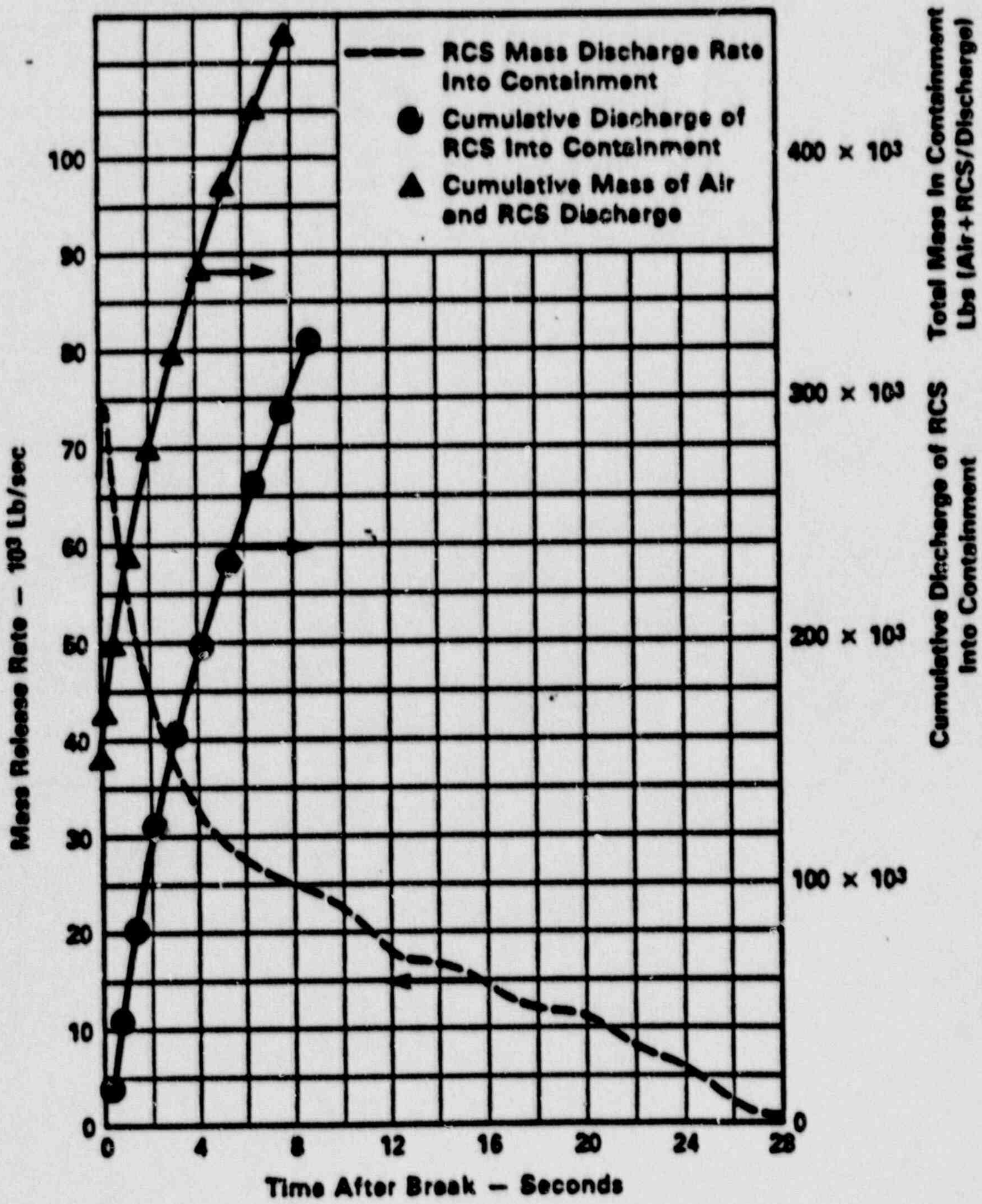
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Item No.	Source	Radiological Sources	Curies Discharged I 131 EQ	Site/Excl. Boundary Dose (Thyroid (REM)
1	Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	73.5	18.7
2	RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119×10^5 curies]	188	48
3	RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	82	21
4	RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	38,000	9676
5	RL	I 131 EQ; SRP Gap activity at 10% Total Activity (SRP calc.) [157.9×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	248,950	63,400
6	RL	I 131 EQ; Reg. Guide 1.4 at 25% Total Activity [390×10^5 curies of I 131 EQ into cont. on LOCA]	611,500	155,700

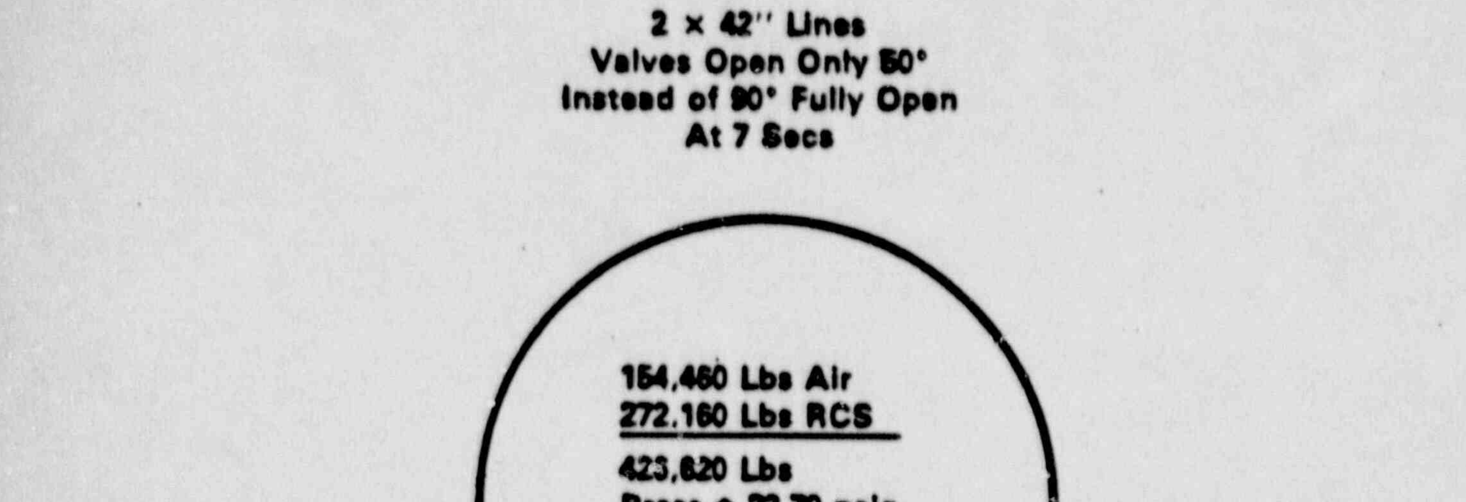
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines.
Valves open 50°

Given Q = total inventory of fission products in RCS at T=0.5 secs after LOCA

- If Q is released instantaneously to the total containment volume:

  Fission product inventory discharged outside containment over 7 secs = 1.568% Q

- If Q is released over time with RCS inventory and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 0.561% Q

ZION: LOCA DURING CONTAINMENT PURGE
USING 2x42" PENETRATIONS - VALVES FULLY OPEN (90°)

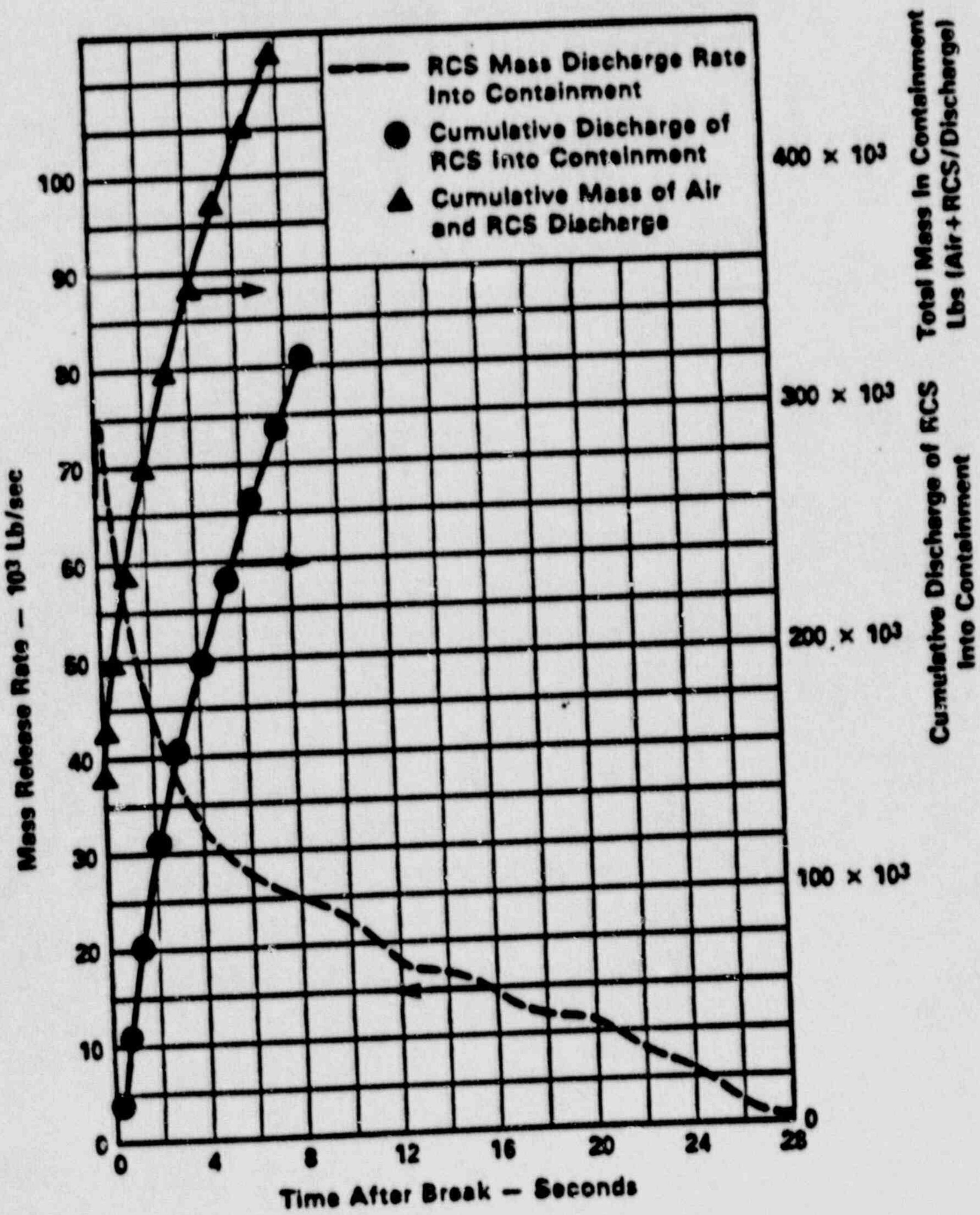
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Item No.	Source	Radiological Sources	Curies Discharged I 131 EQ	Site/Excl. Boundary Dose (Thyroid) (REM)
1	Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	204.3	52
2	RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119 x 10^5 curies]	522	132
3	RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	227	58
4	RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	105,600	26,878
5	RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	691,520	176,010
6	RL	I 131 EQ; Reg. Guide 1.4 at 25% Total Activity [390 x 10^5 curies of I 131 EQ into cont. on LOCA]	1,698,592	488,911

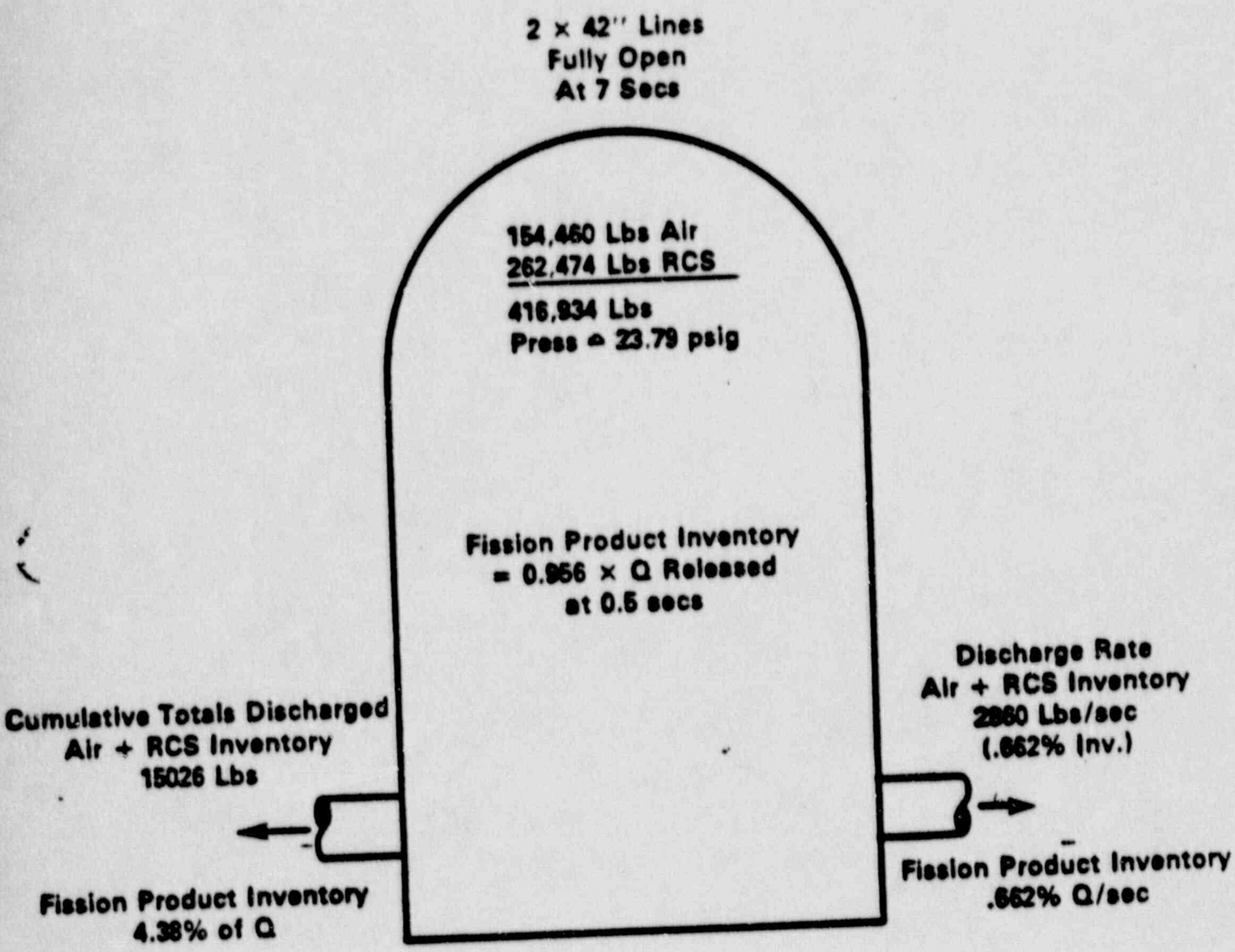
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES
DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



(Q = Fission Product Inventory Released at t = 0.5 secs)

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines
fully open (90°).

Given Q = Total inventory of fission products in RCS at T=0.5 sec after LOCA.

- If Q is released instantaneously to the total containment volume

  Fission product inventory discharged outside containment over 7 secs = 4.38% Q

- If Q is released over time with RCS inventory, and based on a uniform distribution within the inventory:

  Fission product inventory discharged outside containment over 7 secs = 1.90% Q

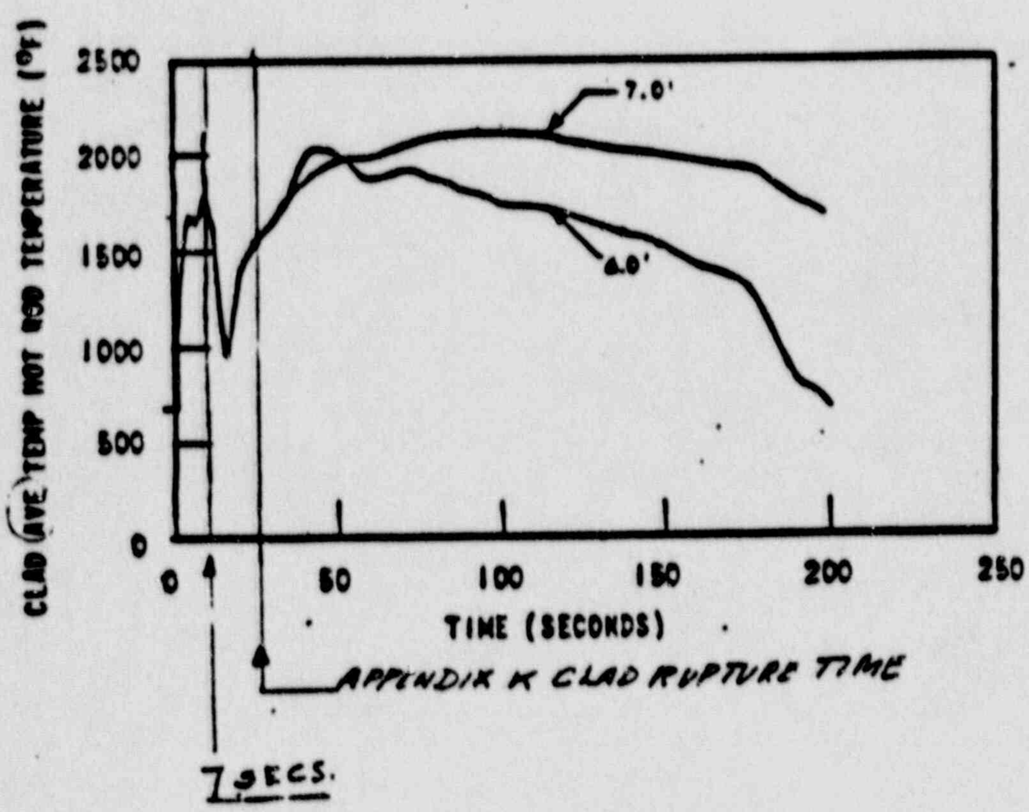


Figure 14 F.2-19a Peak Clad Temperature - DECLG (C_D=1.0)
(Unit 1)

3.1.3.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is [1.30*].

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB). DNB causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength- 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minimum values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerable below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under normal operating conditions is well in excess of the maximum design pressure. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The end of life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history, however it does not exceed the design limit of 2250 psia.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1 [as 720°F].

Previous experience with removable rods has been attained at Saxton, Yankee and Zorita; and additional experience will be acquired at the San Onofre Cycle 2 and Surry Unit 1. Over 300 fuel rods were removed and re-inserted into assemblies during the Saxton re-constitution without evidence of failure. Leak detection tests were performed on the assemblies after all rods were re-inserted, and no leakage was detected. An equally large number of Saxton rods have been successfully removed, examined and re-inserted into over 12 3x3 subassemblies at Saxton. In addition, 28 full length Yankee rods were removed, examined and re-inserted into Yankee Core V special assemblies. Similiar handling of 22 removable rods was sucessfully completed during the first Zorita refueling. All such fuel handlings have been done routinely and without difficulty.

The same fuel rod design limits indicated in section 3.2.3 fuel temperature and internal pressure, are maintained for these removable rods and there is no reduction in margin to DNB. Their inclusion in the initial Zion Unit 1 core loading introduces no additional safety considerations and in no way changes the safeguard analyses and related engineering information presented in previously submitted material in support of the license application.

3.2.3.5 Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods is calculated by a code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 per cent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests. See additional details in Section 3.3.3.1.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

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The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature and fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of high burnup uranium dioxide** fuel element behavior indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considers the effect of burnup, temperature distribution, and internal voids. It is an empirical model which has been checked with data from Bettis, Yankee, CVTR, Saxton and others. The pellet densities for the three regions are listed in Table 3.2.3-1.

The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits and damage limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength- 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minmium values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits. For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

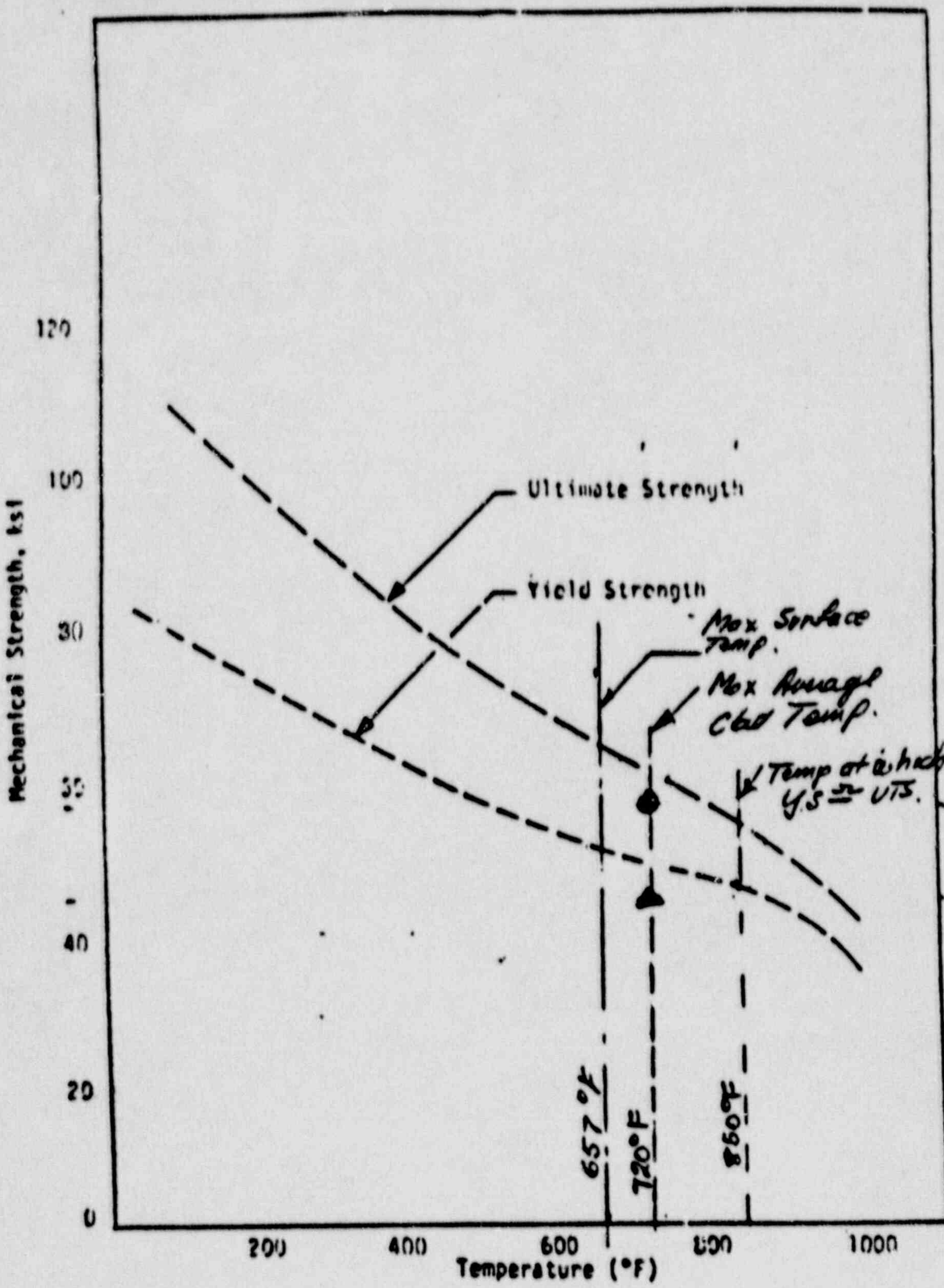
The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure required to produce cladding stresses equal to the damage limit under normal operating conditions is well in excess of the maximum design pressure. The maximum design internal pressure under nominal conditions is 2250 psia which is equal to the coolant pressure. The end of life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history, however it does not exceed the design limit of 2250 psia.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2.2-1.



MECHANICAL STRENGTH OF ... ROD TUBING VERSUS TEMPERATURE

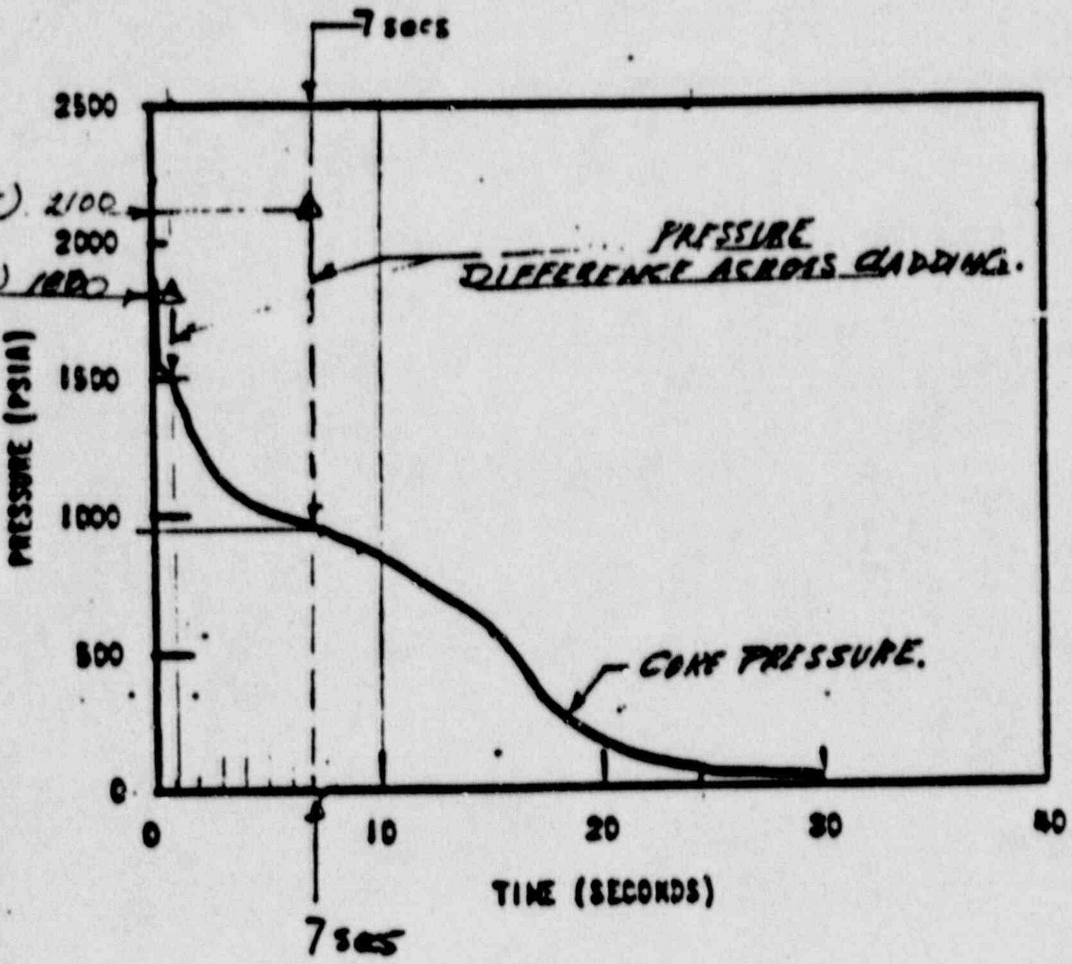


Figure 14 F.2-10a Core Pressure - DECLG (CD=4.0)
(Unit 1)

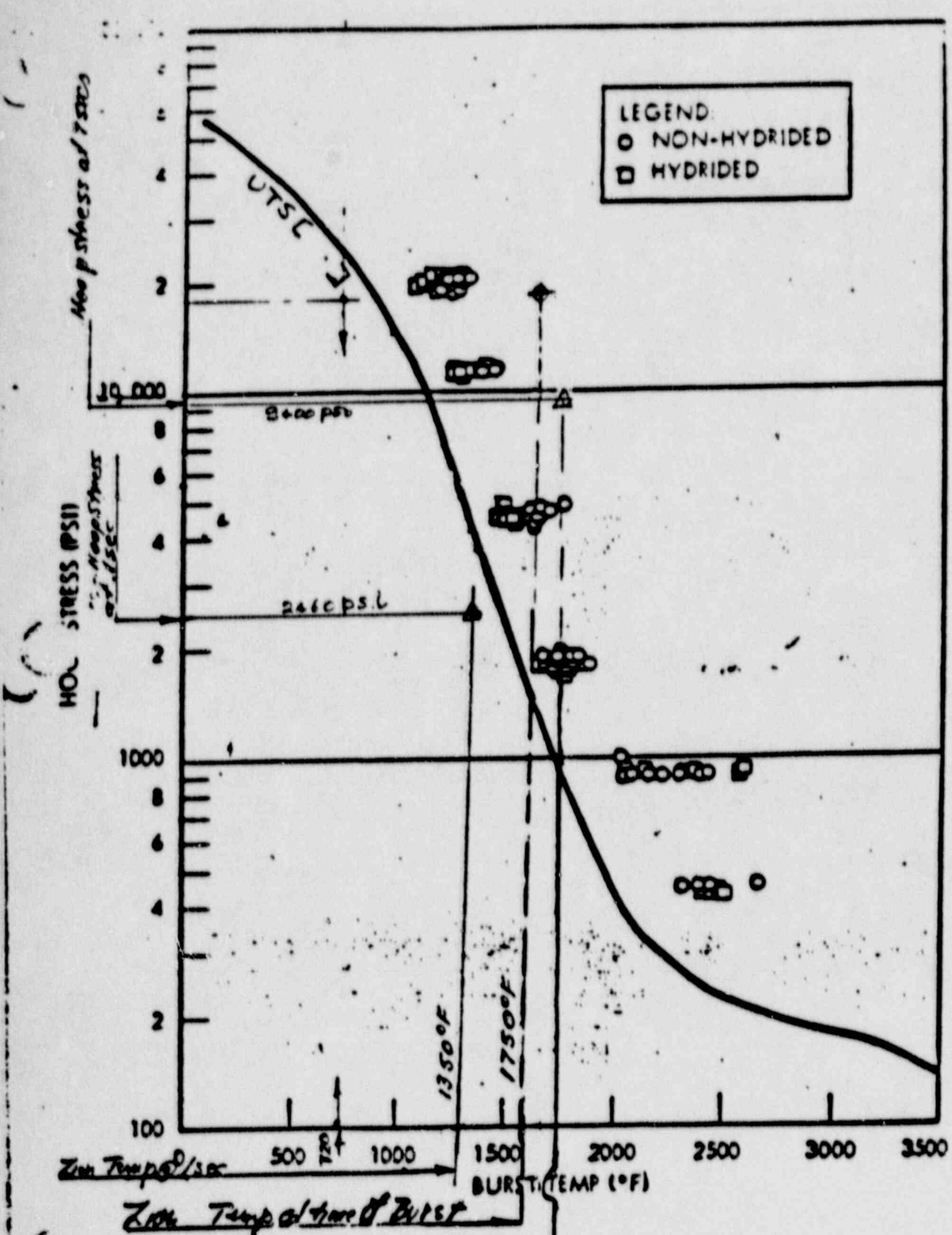
TABLE 1

Engineering Hoop Stress as a Function of Internal Fuel-Rod Gas Pressure and Fuel Vendor Design

Design	Hoop Stress (psi) for a 600 psi Differential Across the Cladding Wall
B&W 15x15	4570
B&W 17x17	4540
C-E 16x16	4280
W 15x15	4910
W 17x17	4690
GE 8x8	4050
ENC 15x15*	3940
ENC 8x8**	3880

* D. C. Cook, Unit 1

** Oyster Creek



Burst Temperature versus Stress at Burst

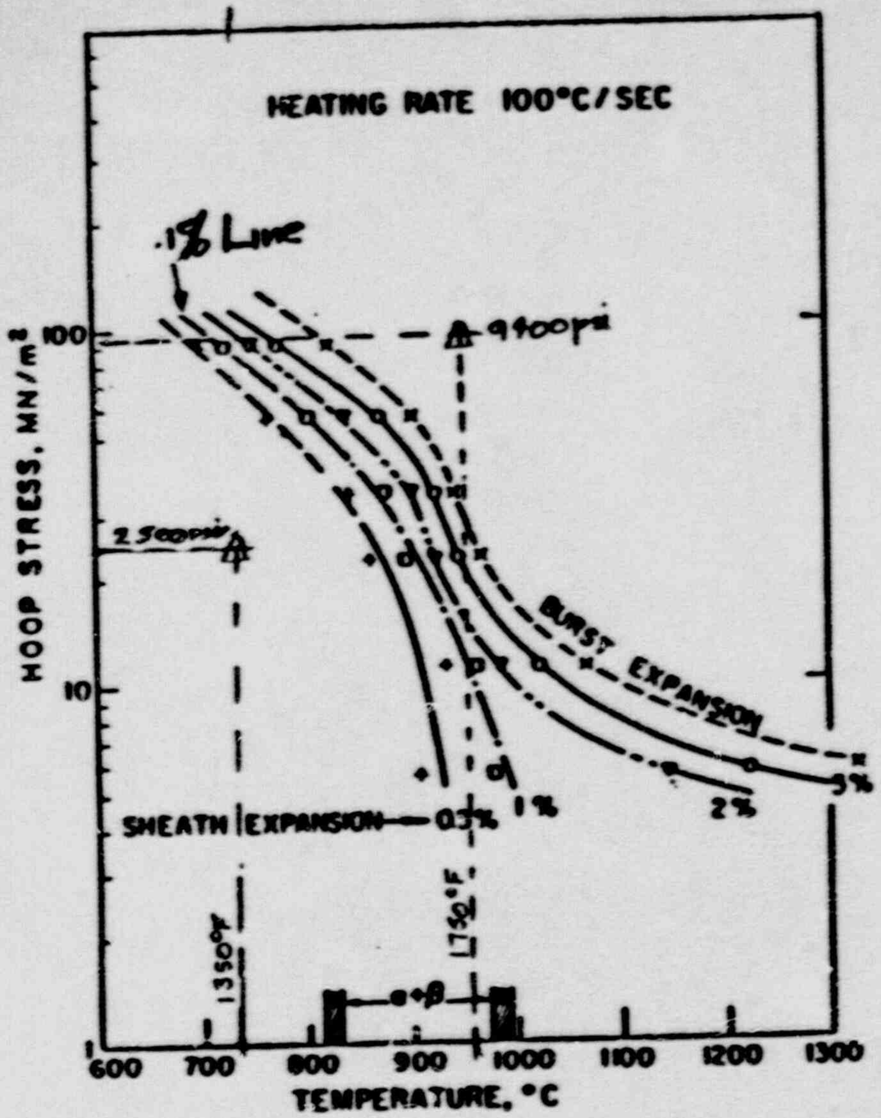


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FIGURE 10 (HARDY)

Isostrain and rupture curves plotted as a function of hoop stress and temperature for tubes heated at 100°C/sec.

1 MN/m² = 142.9 psi

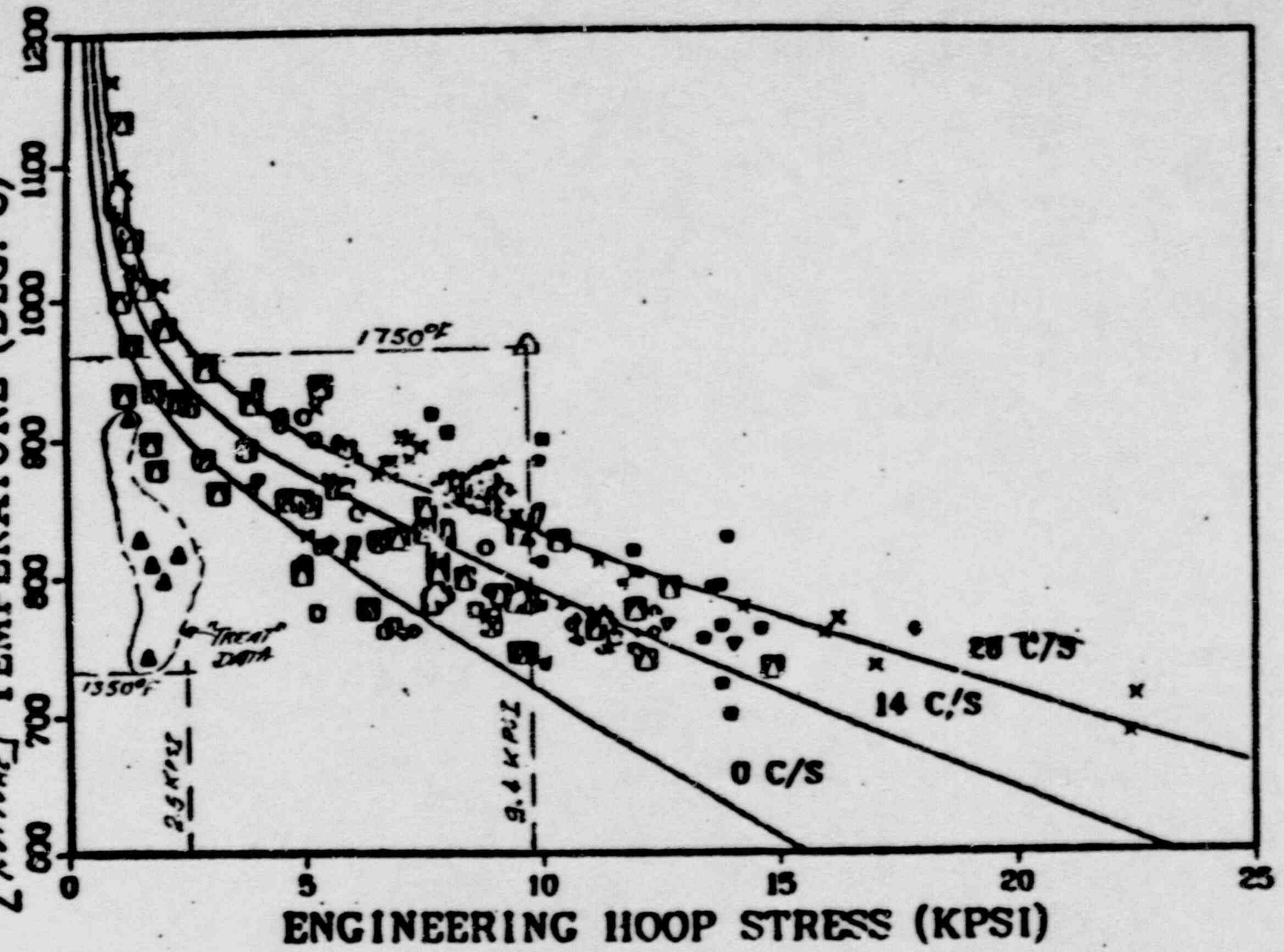

Fig. 3 ORNL correlation of rupture-temperature as a function of ~~engineering hoop stress~~ and ~~temperature-ramp rate~~ with data from internally heated Zircaloy claddinn in aqueous atmospheres.

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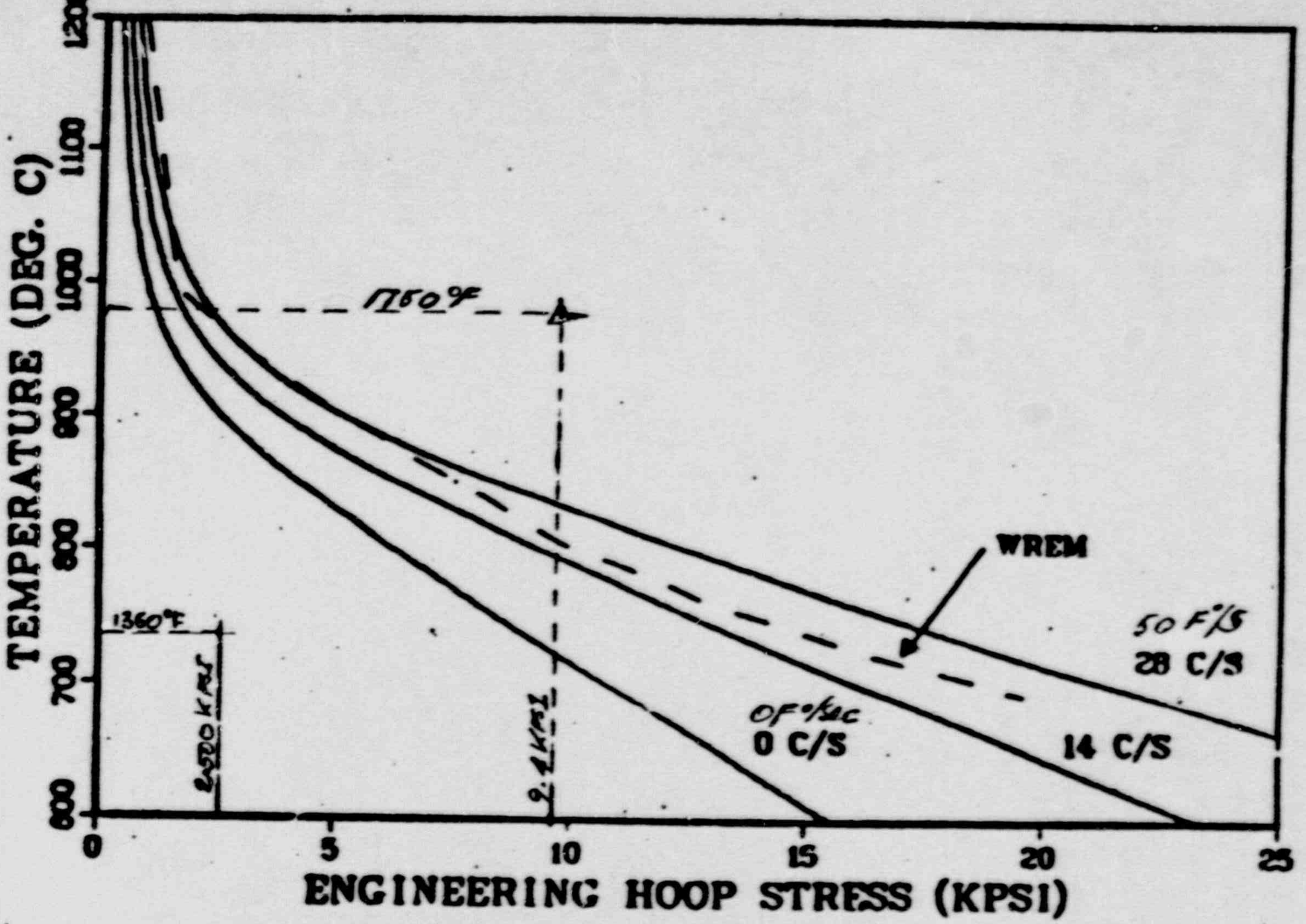

Fig. 17 WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

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3.5 Clad Swelling and Rupture Model

During a LOCA the clad is assumed to strain uniformly and plastically in the radial direction provided that both the temperature and the differential pressure across the clad are sufficiently high. If the strain exceeds [10%] or the clad temperature exceeds the burst temperature (determined as a function of the instantaneous stress) the clad is assumed to burst and an additional local strain is added to the burst node. (a,c)

Three empirical models are employed to evaluate the clad swelling and rupture behavior.

3.5.1 Clad Swelling Prior to Rupture

Hardy[26] performed a series of tests in which rods with constant internal pressure were ramped to a series temperatures at various constant ramp rates. The pressures reported by Hardy were converted to hoop stresses by the formula

(3-69)

and the strain at a given temperature and ramp rate were correlated as functions of the derived hoop stress. The equation developed which best describes the data is (a,c)

(3-70)

where: (a,c)

000

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WESTINGHOUSE

(a,c)

(a,c)

(a,c)

(a,c)

3.5.2 Clad Burst

Clad is assumed to burst if it reaches [10%] hoop strain based on the swelling (a,c) model described above or if the clad temperature at the burst node reaches the burst temperature. ⌈Burst temperature is calculated as a function of hoop stress based on correlation of the Westinghouse single rod burst test[27] data shown in Figure 3-1. The best estimate curve from figure 3-1 is used and pressure is converted to hoop stress by the relationship described in Equation 3-69 using original test specimen geometry. This best estimate curve is described by the equation

(a,b,c)

$$T_{burst} = \left[\qquad \right] \tag{3-71A}$$

3.53 Local Hoop Strain After Burst

The localized diametral swelling that occurs very rapidly at the time of burst is calculated from a correlation of single rod burst test data of Westinghouse and others. Figure 3-2 shows the correlation and the ranges of the data used. Expressed in terms of hoop stress the correlation gives

(a,b,c)

$$\frac{\Delta d}{d_o} \left[\qquad \right] \tag{3-71B}$$

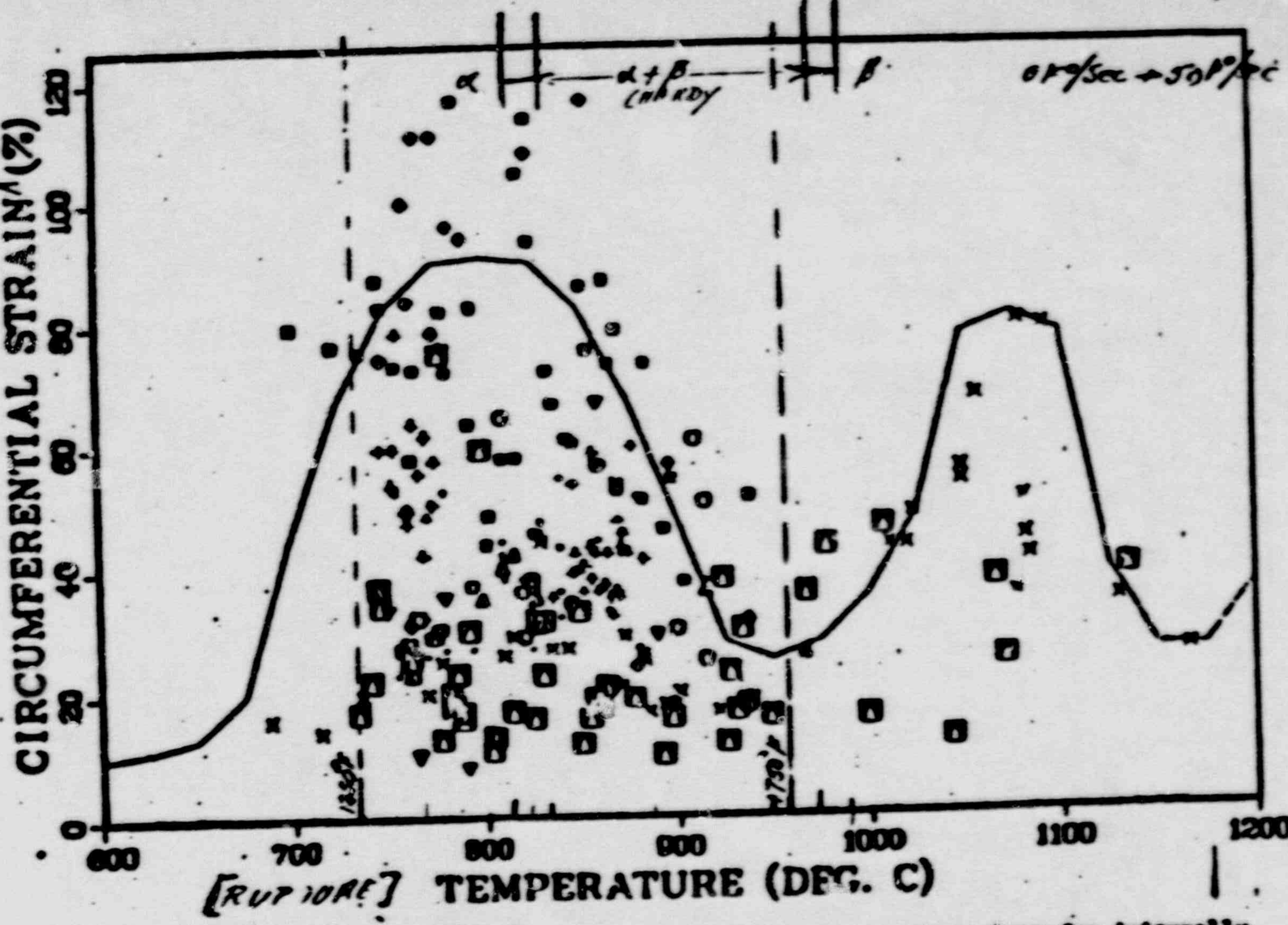

Fig. 8 ~~Maximum circumferential strain~~ as a function of ~~rupture temperature~~ for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

NUREG-75/077

THE ROLE OF FISSION GAS RELEASE IN REACTOR LICENSING

CORE PERFORMANCE BRANCH

U. S. NUCLEAR REGULATORY COMMISSION

NOVEMBER 1975

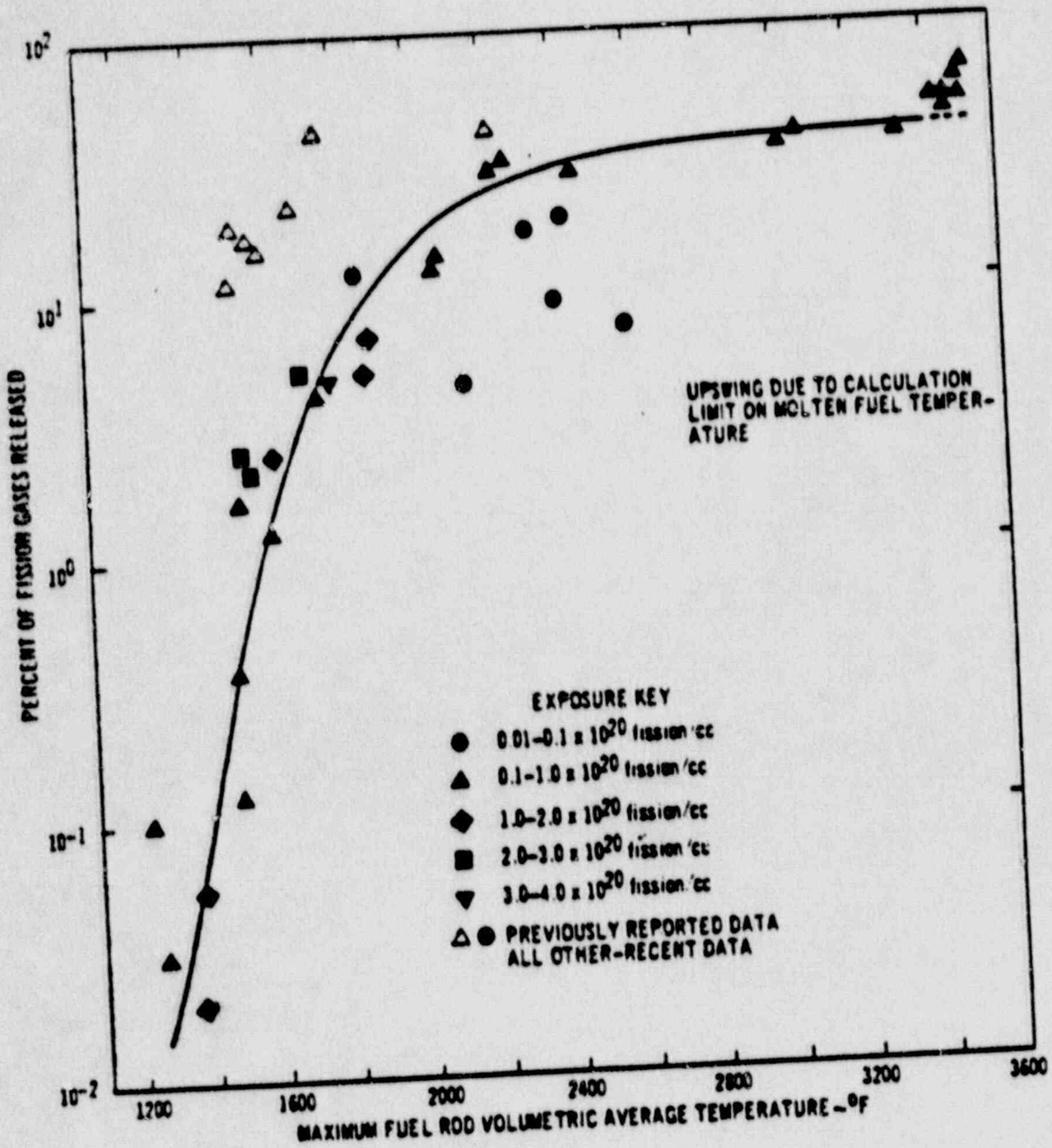

Fig. 2. The Hoffman & Coplin correlation of fission gas release as a function of temperature (from Ref. 35).

ZION

CORE TEMPERATURE DISTRIBUTION

Assumptions: Operation at 3391 MWt for 500 days

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
14.5	2500

Attachment

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 11, 1989

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. A. Licciardo, Reactor Engineer (Nuclear)
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: DIFFERING PROFESSIONAL VIEW CONCERNING

a) Issuance of SER to Zion 1/2 allowing full power operation with open 42" containment isolation valves.

b) Methodology used for calculating related offsite doses.

The writer submits a Differing Professional View (DPV) in accordance with the provisions of NRC Manual Chapter 4125.

This issue has arisen out of the Safety Evaluation Report (SER) undertaken for the Zion Units 1 and 2 as prepared by the writer; see Attachment.

The principal issue is the prudent and conservative calculation of the additions to offsite dose which may result from a LOCA at a facility during the use of open purge supply and exhaust valves at full power.

The licensee for Zion 1/2 has proposed full power operation of the facility with the 42" purge supply and exhaust containment isolation valves open to a limited position of 50°, and capable of isolation within seven (7) seconds of the commencement of a LOCA.

The writers SER concludes that the 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequence of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body has not been evaluated. The least value for the additional offsite dose which may be proposed within the licensing basis is 64,000 rem over the first seven (7) seconds of the LOCA. Management staff has disagreed with the writer's methodology and conclusion and plans issuance of a separate SER permitting the operation requested. The writer requests non-issuance of the related SER to the licensee. He also proposes probability of a generic action on other facilities which have been granted such licenses based on the staff's current methodology.

In general, the management staff has adopted a criterion described in SRP BTP CSB 6-4 which is that providing the maximum time for closure of these containment isolation valves does not exceed 5 seconds (and by plant-specific exception, up to 15 seconds), then the valves would be closed before the onset of fuel failure following a LOCA so that the only contribution to offsite dose is from RCS operational levels of fission product directly discharged into containment during this period, and then through the open containment isolation valves before closure.

~~8909140118~~ XA 2pp.

In evaluating the consequence for Zion, the writer has used an alternate Criterion in BTP CSB 6-4 which states that:

> "The following analyses should be performed to justify the containment purge system design:
>
> An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR Part 100 guideline values."

Using these related guidelines for Zion, the fuel performance over the 0-7 seconds is detailed and shows that fuel failure (by infringement of DNBR criteria) occurs within ½ seconds of the commencement of the LOCA, and together with other licensing basis responses including fission product release from the fuel gap and the thermal hydraulic conditions in the core, containment and discharge nozzle, result in a substantive discharge of fission products to the environment of far greater consequence than are calculated by the staff.

The relative consequences of these differing approaches are that whereas the staff methodology gives additions to offsite dose resulting in total doses within 10 CFR Part 100 limits, the alternate approach used by the writer shows a substantially increased offsite dose exceeding 10 CFR Part 100 limits, with completely unacceptable consequences to Public Health and Safety.

The writer requests review of the Differing Professional View in a timely manner in accordance with the provisions of NRC Manual Chapter 4125.

R. Licciardo

Robert B. A. Licciardo
Registered Professional Engineer California
Nuclear Engineering License No. NU 001056
Mechanical Engineering License No. M 015380

cc: J. Sniezek
D. Muller
S. Varga
C. Patel
F. Miraglia
L. Shao
A. Thadani
J. Wermiel
J. Kudrick

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 11, 1989

Attachment

Docket Nos. 50-295
and 50-304

MEMORANDUM FOR: Daniel Muller, Director
Project Directorate III-2
Division of Reactor Projects III, IV, V
and Special Projects

FROM: Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

SUBJECT: OFFSITE RADIOLOGICAL CONSEQUENCES OF LOCA DURING CONTAINMENT PURGE PROPOSED IN TS CHANGES FOR ZION 1 AND 2

Reference: Letter to H. R. Denton (NRC) From P. C. Leonard dated February 2, 1986, Subject: Zion Nuclear Power Station, Units 1 and 2 Proposed Amendment to Facility Operating License No. DPR-39 and DPR-48

Plant Name: Zion Nuclear Power Station, Units 1 and 2
Licensee: Commonwealth Edison Company
TAC Nos.: 55417 and 55418
Review Status: Complete

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units; see reference 1.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

The enclosed Safety Evaluation Report has been prepared by the technical reviewer initially assigned to this task, namely Robert B. A. Licciardo.

The licensee's proposal is to allow full power operation of the facility with the 42" purge supply and exhaust containment isolation valves open to a limited position of 50°, and capable of isolation within seven (7) seconds of the commencement of a LOCA.

The review concludes that the 42" valves at Zion should remain closed in Modes 1, 2, 3 and 4 because the consequence of the offsite dose to thyroid (from iodine) during a LOCA is unacceptable high; whole body dose has not been evaluated: The least value for the additional offsite dose which may be proposed within the licensing basis is 64,000 rem over the first seven (7) seconds.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of resulting source terms.

XA
~~8909140120~~ 3pp.

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

Enclosures:
As stated

cc w/enclosures:
C. Patel

CONTACT: R. Licciardo
X20876

Daniel Muller

Our SALP input is provided in Enclosure 2. We consider our efforts on TAC Nos. 55417 and 55418 to be complete.

Jared S. Wermiel, Acting Chief
Plant Systems Branch
Division of Engineering and Systems Technology

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5520 NAME: Zion TACs 55417/8 Licciardo

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Enclosure 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
PLANT SYSTEMS BRANCH
OFFSITE RADIOLOGICAL CONSEQUENCE OF LOCA DURING
CONTAINMENT PURGE
ZION NUCLEAR POWER STATION, UNITS 1 AND 2
DOCKET NOS. 50-295 and 50-304

1.0 INTRODUCTION

Zion Units 1 and 2 (CECo) has responded to an NRC request to propose TS to primarily constrain operation of the large (42") containment purge supply and exhaust valves on these units.

The former Plant Systems Branch, Section A, of the Division of PWR Licensing A, requested Section B of the same branch to review the offsite radiological consequences of this proposal.

2.0 EVALUATION

Background review shows that the facility was evaluated on the basis of normally closed purge valves so that these consequences were never included in the Zion SER. Further, that a letter from Westinghouse (W) to Commonwealth Edison Company dated October 22, 1976 on the subject of "Offsite Doses During LOCA and Containment Purge" (Ref. 2) has never been evaluated by the NRC. Subsequent to the TMI-2 event, the operability and automatic control of these valves was evaluated leading to the request for the required TS, but the Radiological Assessment was left as a "long(er) term issue" (Ref. 3) which was intended to be resolved in a subsequent probabilistic risk assessment which definitively excluded it from consideration without any justification (Ref. 4).

The W analyses undertaken under Commonwealth Edison instruction, uses an RCS operational inventory of 60 uc/gm equivalent I 131 at the time of the accident with a resulting site boundary thyroid dose due to iodine (during closure of the valves), of 52 rem, and which added to the containment leakage dose of 123 rem gives a total 175 rem which is within the 10 CFR 100 limit of 300 rem. The total iodine inventory of the RCS is assumed to be released into containment on initiation of the LOCA; a 50% plate out is assumed leaving the residual 50% as part of containment inventory for discharge out through both fully open containment purge lines for a total of seven (7 seconds).

However, when reviewed against the BTP CSB 6-4, Item B.5.a requires that:

> "The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concommitment release of fission products, and the fission product activity in the primary coolant."

~~8911020127~~ 4pp.

Further: SRP 4.2 identifies fuel failure with infringement of DNBR criteria, with the related requirement that gap activity be considered as part of the source term, and Regulatory Guide 1.77 recommends that under similar circumstances, gap activity should be assumed at 10% of core activity. Fuel damage criteria also includes the occurrence of center line melting with measures of additional activity release also guided by Regulatory Guide 1.77, but the Zion SAR shows this does not occur.

Revising the source term to Appendix K calculations [in which all fuel goes to DNBR in ½ second] with related release of all gap activity into containment, with limited blowdown to offsite during the related 7 seconds closure time and absent a 50% plate out of iodine as can be interpreted from the above referenced item B.5.a, increases offsite dose due to containment purge above by a factor of 3400 to 176,000 rem and would thereby be completely unacceptable. Limiting the purge line valves to an opening of 50° could reduce offsite dose to 64,000 rem and represents the least value which may be proposed within the licensing basis.

Note: The BTP CSB 6-4 proposing that valve closure within 5 seconds will ensure purge valves are closed before the onset of fuel failures has since been extended by the staff on a plant-specific basis to 15 seconds. Further, the writer cannot find any safety evaluation report supporting these positions. These positions cannot be sustained for Zion since a) DNBR infringement (from Appendix K calculations) and hence fuel failure and gap activity release [Ref. SRP 4.2) of 10% of core inventory (Ref. Regulatory Guide 1.77) occur within ½ second of the initiation of the LOCA, b) related maximum clad temperatures of 1750°F occur immediately and never reduce below 1400°F, c) RCS pressure in the region of the core rapidly reduces from 2250 psia to 900 psia in 7 seconds increasing potential pressure drop across the cladding for release of gap activity to the RCS inventory, d) the massive bulk boiling and blowdown surrounding the failed fuel ultimately discharges 270,000 lbs of RCS inventory into the containment at 7 seconds into the event increasing containment pressure from 0.3 psig to 23.8 psig (in these 7 seconds), and e) causes 15,000 lbs of the resulting containment inventory to be discharged to the environment through 2x42" fully open lines, or 5400 lbs for the same lines with valve closed to 50°.

3.0 CONCLUSION

The 42" valves at Zion should remain closed in Modes 1, 2, 3, and 4 because the consequences of the offsite dose to thyroid (from iodine) during a LOCA is unacceptably high; whole body dose has not been evaluated. The least value for offsite dose to the thyroid which may be proposed within the existing licensing basis is 64,000 rem.

The conventional treatment of BTP CSB 6-4 which assumes that fuel failure does not occur over the first 5-15 seconds after a LOCA and thereby that only RCS operating inventory of fission products is released to the containment, and then to the environment, cannot in general be sustained against thermal hydraulic analyses for containment response, and licensing basis requirements (including criteria) for the calculation for, and the occurrence of, fuel damage and the quantification and treatment of the resulting source terms.

References

1. Letter from P. C. Blond (CECo) to H. R. Denton (NRC); Subject: Zion, Units 1 and 2, Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 dated February 21, 1986.

2. Letter from R. L. Kelley (W) to C. Reed (CECo); Subject: Offsite Dose During LOCA and Containment Purge, dated October 22, 1986.

3. Letter to L. O. DelGeorge (CECo) from S.A. Varga (NRC); Subject: Generic Concerns of Purging and Venting Containments, dated September 9, 1981.

4. Memo for F. H. Robinson from R. W. Houston, Subject: "Evaluation of the Risk at Zion," dated August 14, 1985.

Enclosure 2

SPLB SALP INPUT

Plant Name: Zion Nuclear Generating Stations, Units 1 and 2
SER Subject: Containment Purge and Vent Valve Operation
TAC Nos.: 55417/8

Summary of Review/Inspection Activities

The licensee provided an evaluation of offsite doses undertaken in 1976. This was undertaken with a methodology and source term chosen by the licensee. The licensee did not present results from alternative more detailed methodologies which could be considered enforceable under existing regulatory positions and the related circumstances.

Narrative Discussion of Licensee Performance - Functional Area

The single only methodology used by the licensee is not an acceptable approach for estimating doses under the proposed circumstances and especially since alternate detailed evaluations required by the SRP give greatly increased values beyond 10 CFR Part 100 limits. A prudent approach would have recognized the deficiencies and risks in the single methodology adopted with resulting substantively different recommendations to ensure public health and safety.

Author: Robert B. A. Licciardo

Date: May 11, 1989

Mtg of Review Panel. Mon, 8/21, @1:30 Due date extended to 8/28. Ref 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 11 1989

8/11/89

NOTE TO: Frank J. Miraglia, Associate Director
for Inspection and Technical Assessment, NRR

FROM: Ashok Thadani, Director
Division of Systems Technology, NRR

SUBJECT: DPV CONCERNING CONTAINMENT ISOLAION VALVES AT ZION

In your note to me dated July 27, 1989, regarding the above subject you asked for information regarding the technical rationale for time to fuel damage from the onset of a LOCA in the Appendex K analysis. Specifically: (1) the temperature and pressure effects experienced by fuel early in a LOCA event; and, (2) why entry into DNBR does not result in fuel failure. Wayne Hodges' note to me dated August 10, 1989, (Enclosure 1) addressed these issues.

With regard to (1) above, analysis indicates that there is potential for fuel pin rupture during the LOCA blowdown (7 seconds) for very high power pins. However, for fuel pin powers that exist for current designs no blowdown rupture is predicted. Thus, fuel pin rupture during blowdown is not a problem for existing designs but should be checked for future designs.

With regard to (2) above, the main contributors to "fuel cladding rupture" are high pressure across cladding and high cladding temperature. While entry into departure from nucleate boiling (DNB) significantly reduces the heat transfer resulting in rapid cladding temperature rise, the heat transfer is not zero and the temperature rise is not instantaneous. Thus, it is not physically possible for the cladding to instantaneously rupture upon entry into DNBR because of LOCA conditions. Experimental data confirms this conclusion.

The fuel criteria described in Chapter 4 of the Standard Review Plan (SRP) could be interpreted to apply to LOCA analyses in the absense of staff practice. However, staff practice has never to our knowledge been to assume fuel failure upon inception of DNB for LOCA analyses. Perhaps, the SRP should be revised to more clearly describe staff practice, but I do not believe the effort to be worth the cost in staff resources.

Based upon these analyses and discussions with several staff experts, I do not believe that rupture of high burnup fuel pins during the blowdown transient to be credible for existing fuel designs. However, it is appropriate to verify that blowdown rupture does not occur for future designs.

You also requested comments regarding the applicability of Reg Guides, SRP's and BTP's cited in the reviews of the Zion amendment. Jack Kudrick, SPLB, and Ted Quay, PD31, looked into this (See Enclosures 2 and 3, respectively).

~~8909140124~~ XA 2pp.

Regarding Reg Guides, SRP"s and BTP's cited in the subject DPV, the major reference within the DPV is BTP CSB 6-4. This BTP is referenced in SRP Section 6.2.4, Containment Isolation Section. The focus of the DPV only addresses the BTP, however, to present a complete picture of the staff's position SRP 6.2.4 and how BTP CSB 6-4 is referenced need be considered. In particular SRP 6.2.4 states that for lines which provide an open path from the containment to the environs; eg., the containment purge and vent lines, isolation valve closure times "on the order of" 5 seconds or less may be necessary. Note that the intent must be taken as a goal but does not preclude closure times greater than 5 seconds.

Subsection n of SRP 6.2.4 is relevant to the DPV regarding dose analysis. Subsection n states:

> "...regarding the size of the purge system used during normal plant operation and the justification by acceptable dose consequence analysis, may be waived if the applicant commits to limit the use of the purge system to less than 90 hours per year while the plant is in the startup, power, hot standby and hot shutdown modes of operations."

Enclosure 3 provides discussion on Reg Guides and the SRP regarding the subject DPV's contentions on the release of fission products to the containment and subsequently to the environment through open purge valves. The bottom line of this discussion is that although the staff has used the "instantaneous" source term in accidents such as LOCA, its use was to ensure that containment isolation features incorporated either fast acting valves or features that would ensure containment integrity was not compromised during operation (e.g., dual doors on personnel locks). This simplified approach was never intended to be applied to purge valves except for those valves that were extremely slow closing (e.g., 2 minutes). No opening in containment during operations could be justified using the simplified instantaneous source term assumption. Specifically, no purge/vent system design could be found acceptable and without such systems, plant operations would be extremely restricted. Although the SRP specifies 5 seconds, the staff accepted closure times up to 15 seconds based on informal discussions we had with Research on their severe accident analyses. We were told that even for closure times up to 20 seconds that no substantial releases would occur.

The above discussion more properly reflects the staff view on purging. It does not indicate that the staff during the development of the SRP believed that the consequences of purging at the time of a LOCA would result in the impact asserted in the DPV.

A. Thadani

Ashok Thadani, Director
Division of Systems Technology

Enclosures:
1. Note from W. Hodges on DPV, dated August 10, 1989
2. Note from J. Kudrick on DPV, dated August 8, 1989
3. Note from T. Quay on DPV, dated August 10, 1989

ENCLOSURE 1

AUG 1 0 1989

NOTE TO: Ashok Thadani, Assistant Director
for Systems
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

FROM: M. Wayne Hodges, Chief
Reactor Systems Branch
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

SUBJECT: DPV CONCERNING CONTAINMENT ISOLATION VALVES AT ZION

The attached memorandum from Norm Lauben addresses most of the technical issues raised in the DPV. The analyses performed by Norm Lauben do indicate the potential for fuel pin rupture during blowdown for very high power pins. However, for pin powers which exist for current fuel designs, no blowdown rupture is predicted. Therefore, pin rupture during blowdown is not a problem for existing designs but should be checked for future designs.

The fuel failure criteria described in Chapter 4 of the Standard Review Plan (SRP) could be interpreted to apply to LOCA analyses in the absence of staff practice. However, staff practice has never (at least not since 1974 when I joined the staff) been to assume fuel failure upon inception of DNB for LOCA analyses. Perhaps, the SRP should be revised to more clearly describe staff practice, but I do not believe the effort to be worth the cost in staff resources.

Based upon the enclosed analyses and discussions with several staff experts (R. Meyer, R. Jones, L. Rubenstein), I do not believe that rupture of high burnup fuel pins during the blowdown transient to be credible for existing fuel designs. However, it is appropriate to verify that blowdown rupture does not occur for future designs.

original signed by
Marvin Hodges

M. Wayne Hodges, Chief
Reactor Systems Branch
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

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ENCLOSURE

UNITED STATES
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MEMORANDUM FOR: M. Wayne Hodges, Chief
Reactor Systems Branch
Division of Engineering & Systems Technology
Office of Nuclear Reactor Regulation

FROM: G. N. Lauben, Section Leader
Accident Management Section
Reactor & Plant Systems Branch
Office of Nuclear Regulatory Research

SUBJECT: COMMENTS ON A DPV CONCERNING EARLY BLOWDOWN CLADDING RUPTURE DURING A LARGE BREAK LOCA

Per your request, I have reviewed certain aspects of the DPV on Containment Isolation Valves at Zion. In particular, I addressed the issues raised with respect to cladding rupture of high burnup high pressure fuel early in blowdown prior to containment isolation (about 7 seconds). The comments are enclosed. If you have any questions, please contact me on x23573.

G. N. Lauben

G. N. Lauben, Section Leader
Accident Management Section
Reactor & Plant Systems Branch
Office of Nuclear Regulatory Research

Enclosures:
As stated

cc: R.B.A Licciardo
A. Thadani

~~9408140174~~ XA 12pp.

Comments on a DPV Concerning Early Blowdown
Cladding Rupture During a Large Break LOCA

In a DPV (reference 1) Bob Licciardo has postulated that PWR fuel rods with high burnup and high internal pressure could sustain cladding rupture within a few seconds of a large break LOCA prior to containment isolation. This is further postulated to lead to large off-site releases. Following is some information which may be helpful in addressing some of the issues in the DPV. Seven issues in the DPV are first addressed, then some preliminary observations are made. The DPV issues are referenced by page number and a quote or summary of the issue.

Issue 1 (p. 3-1) - "Appendix K evaluation is not designed to report the earliest rupture that can occur" (Also on pp. 3-4 and 3-5)

While Appendix K does not specifically require searching for the earliest rupture, early ruptures would always be the worst with respect to 50.46 limits if they were calculated to occur. Vendor analyses in the past have shown that because of the extensive cladding swelling prior to rupture, the resultant low transient gap conductance severely limits blowdown heat removal. As a consequence, vendor evaluation model calculations showed that the 2200°F PCT was always exceeded. Therefore, the vendors would always need to reduce the peak power to avoid early blowdown cladding ruptures. Vendor steady state fuel thermal performance and subsequent LOCA analyses showed that the peak linear heat generation rate (PLHGR) was always low enough to avoid early blowdown swelling and rupture for high burnup pins. These studies were done about 13 to 15 years ago with Appendix K evaluation models which are no longer used. I do not know if analyses with high burnup pins have been done with recently approved fuel performance and LOCA models. The older analyses always showed that low burnup post densification pins were always most limiting, in fact, because the PLHGR was highest and gap conductance was very low. High burnup pins are lowest in PLHGR although the pin pressure is highest. The combination of high cladding temperature and higher internal pressure are needed to cause cladding rupture.

Issue 2 (p. 3-2) - "This shows that on infringement of DNBR at 1/10 second, average clad temperature increase very rapidly from a normal operating value of 720°F to at least 1350°F, and then to 1750°F, over a total period of seven seconds."

1750°F is indeed a very high early blowdown peak cladding temperature (PCT), but virtually impossible for a high burnup pin with a much lower PLHGR. If a high burnup pin reached 1750°F, at 7 seconds it would most likely rupture. More realistic, LOCA analyses have been performed as part of the Code Scaling, Applicability, and Uncertainty program in RES. A best estimate analysis was performed and code uncertainties evaluated for a large break LOCA (reference 2). In order to accomplish this, sensitivity studies were performed which varied gap conductance, peaking factors and several other variables. The plant used was a Westinghouse 4-loop 3411 MWt plant with 17x17 fuel and a low burnup of only 16000 MWD/MTU which resulted in a PLHGR of 9.35 kw/ft. The blowdown peak for the nominal CSAU case was 1103°F (see figure 1). Based on over 250 clad temperature calculations and using Monte Carlo sampling techniques, it was determined that the 95th percentile blowdown PCT was 1447°F. It has been determined that 15x15 pins (as used at Zion) with burnups greater than 40,000 MWD/MTU have PLHGRs no greater than 5.17 kw/ft. Using the CSAU calculated sensitivity of blowdown PCT to LHGR, the value of 1447°F can be extrapolated to approximately 1265°F for the 5.17 kw/ft PLHGR high burnup 15x15 pin. This illustrates that the 1750°F blowdown PCT calculated by Westinghouse is quite conservative, especially for a high burnup pin. I believe that this Westinghouse calculation is probably at least 10 years old.

Issue 3 (p. 3-2) - "Exhibit 10 also shows that W fuels require a design limit of 1% on cladding strain as a design limit, and 1.7% as a damage limit. The work of this Section 3 will show how both of these limits can be exceeded inside the seven seconds on infringement of DNBR during the course of a LOCA, ..."

As exhibit 10 states, these design values are for nominal operation or overpower conditions, not LOCA. Also, DNBR infringement has never been

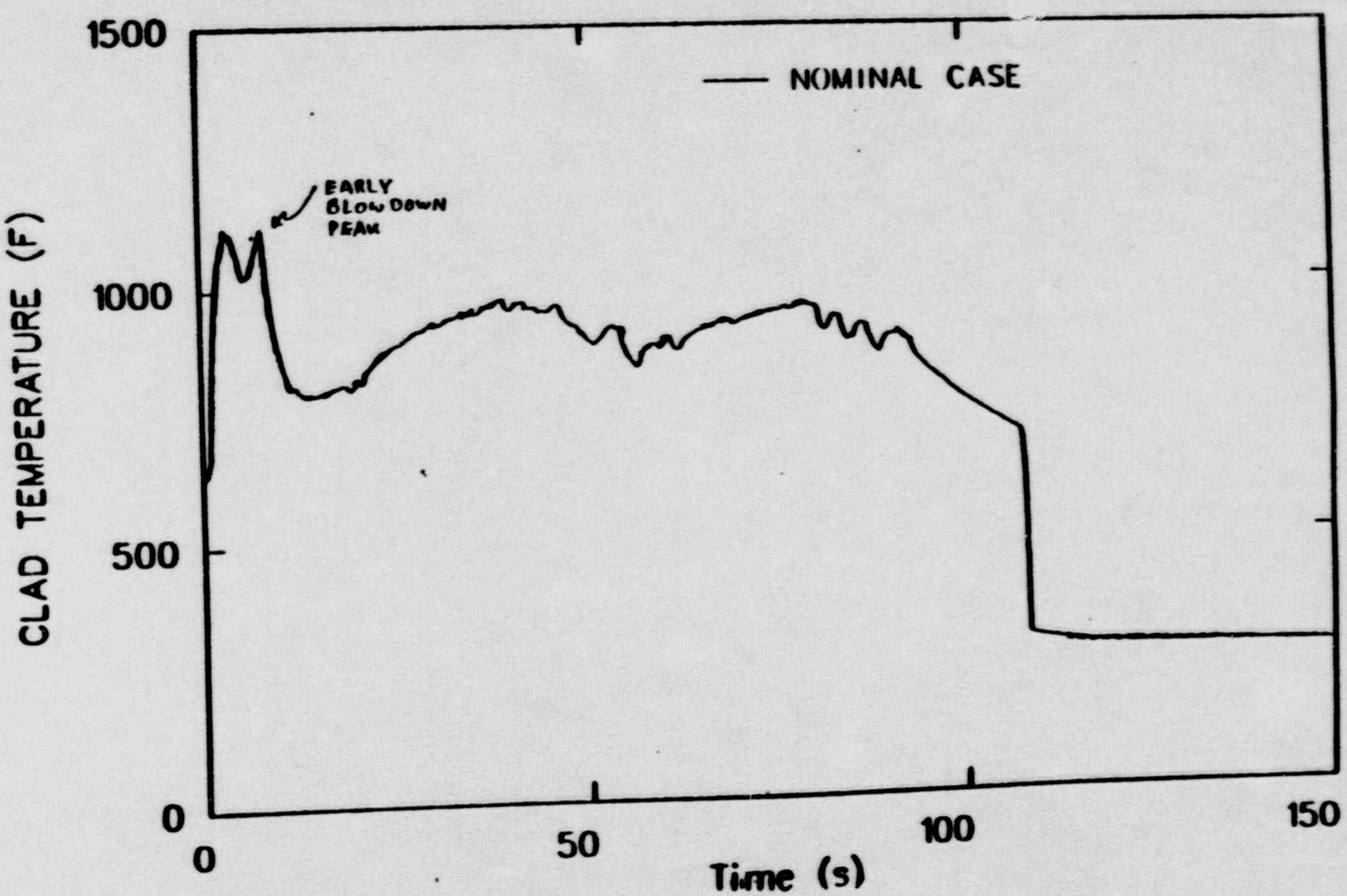

FIGURE 1. Nominal CSAU Case

considered the operant criterion for fuel failure during a LOCA. Although, I am told that this is not as clear as it should be in the standard review plan or any applicable reg. guides. Incidentally, PBF LOCA test do not show DNB occurring until 3-4 seconds for a very severe LBLOCA (reference 3).

Issue 4 (p. 3-3) - ".... there is a need for empirical tests to determine swelling and burst (rupture) characteristics under these same dynamic conditions."
The results of the PBF LOCA tests satisfy this condition and will be discussed as part of Issue 7.

Issue 5 (p. 3-3) - "Reference information shows that internal clad pressure under normally operating conditions is of the order of 1400 psig for new fuel and expected to increase to 2250 psig at the end of the 3rd cycle (for the fuel)."
It is not known what reference information is being invoked here. GAPCON calculations show the following results.

TABLE 1 GAPCON Pin Pressure Calculations

Code	Fuel	PLHGR kw/ft	Burnup MWD/MTU	Pressure (psig)
GAPCON	15x15	15	0	1700
GAPCON	15x15	10	50,000	2700
GAPCON	15x15	5	50,000	2500
GAPCON	17x17	15	0	1900
GAPCON	17x17	10	50,000	3300

The reference 4, GAPCON calculations were performed 9 to 10 years ago. The PAD 3.4 model (reference 5) was approved by the NRC for design and safety analysis in May 1988. Proprietary calculations done with PAD 3.4 showed substantially lower pressures at comparable burnups and PLHGRs. It is well known that the GAPCON fission gas release model is very conservative. The PAD calculations were done at an arbitrarily high PLHGR and would show an even lower pressure at the reduced kw/ft.

Issue 6 (p. 3-3) - "It is proposed that, immediately, on a LOCA as clad temperature increases to 1350°F, gap pressure will increase by 20%, to 1800 psig ..... At 7 seconds into the event, clad temperature has increased further to 1750°F, ... From this, it can be proposed that gap pressure for the complete rod can increase by 36% over its normal operating value to 2100 psig."

The basis for concluding that pin pressure increases during an LBLOCA blowdown is not known and contrary to the evidence. A series of 3 large break LOCA simulations (reference 3) (LOC-3, LOC-5, and LOC-6) were performed in PBF with well instrumented Zircaloy clad UO_2 fuel elements pre-pressurized to simulate low and high burnup PWR fuel. PBF blowdowns are quite severe compared to postulated PWR LBLOCA blowdowns. In PBF, the pressure decrease and rate of mass loss is very rapid. No good reverse flow blowdown heat transfer is evident as is the case in LOFT results or PWR analysis. Figure 2 (reference 6) shows the fuel rod pressure for rod 3 in test LOC-3. Also, shown are FRAP-T6 calculations using two different plastic deformation models. Clearly, pressure decreases throughout the transient. Figure 3is a plot showing measured pressure decrease for Rod 11 in Test LOC-6. A FRAP-T6 characterization calculation was done for a postulated LBLOCA in Zion (reference 7) which also showed a pressure decrease throughout the transient.

Issue 7 (p. 3-5) - Concern is expressed about the relevance of electrically heated rods used in defining the swelling and rupture curves in NUREG-0630. It is suggested that the TREAT data shown in NUREG-0630 (reference 6) would be more realistic. Also, on pp. 4-3 and 4-4, this concern is restated.

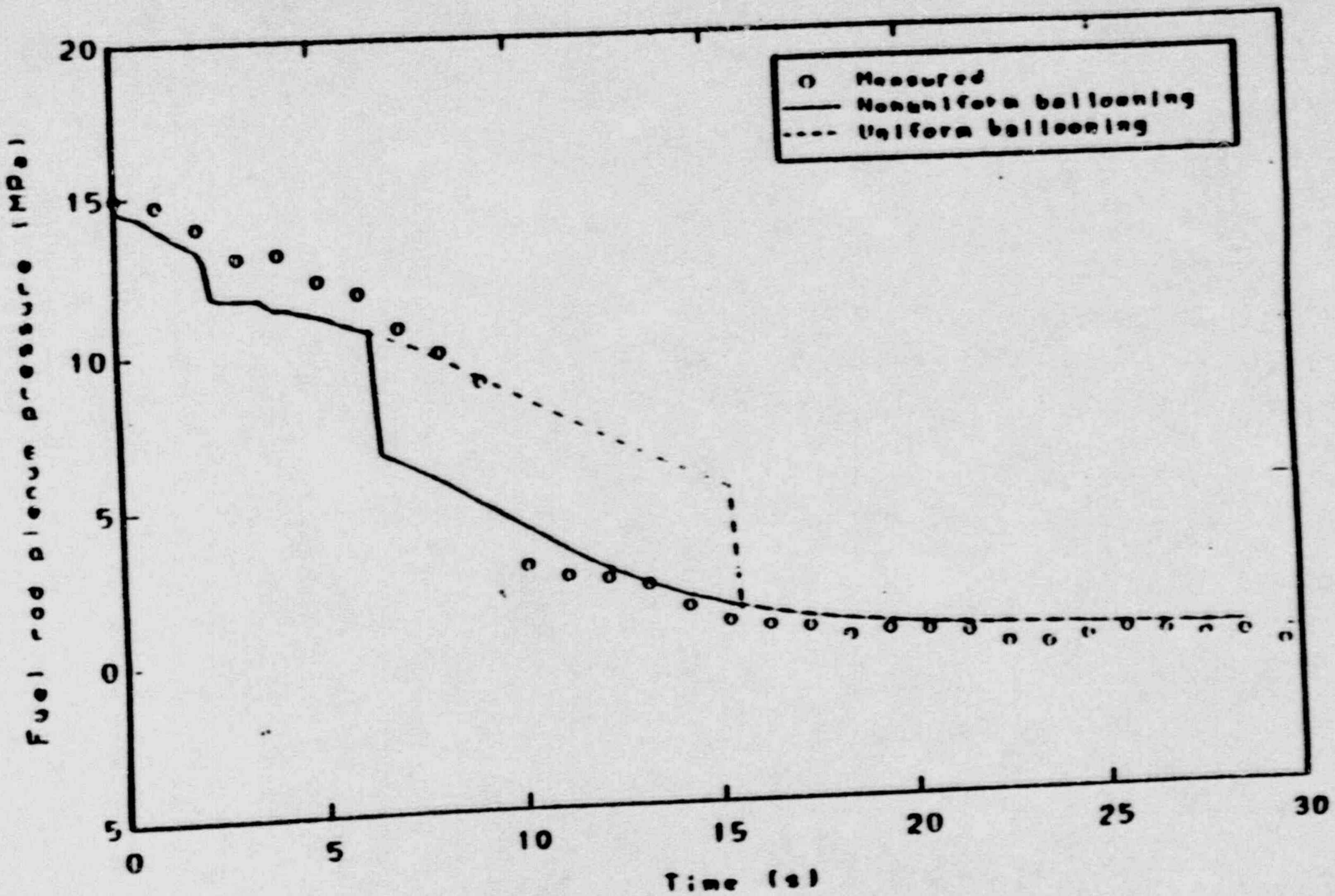

Figure 2. Comparison of measured and calculated fuel rod plenum pressure versus time for Rod 3 of PBF test LOC-3.

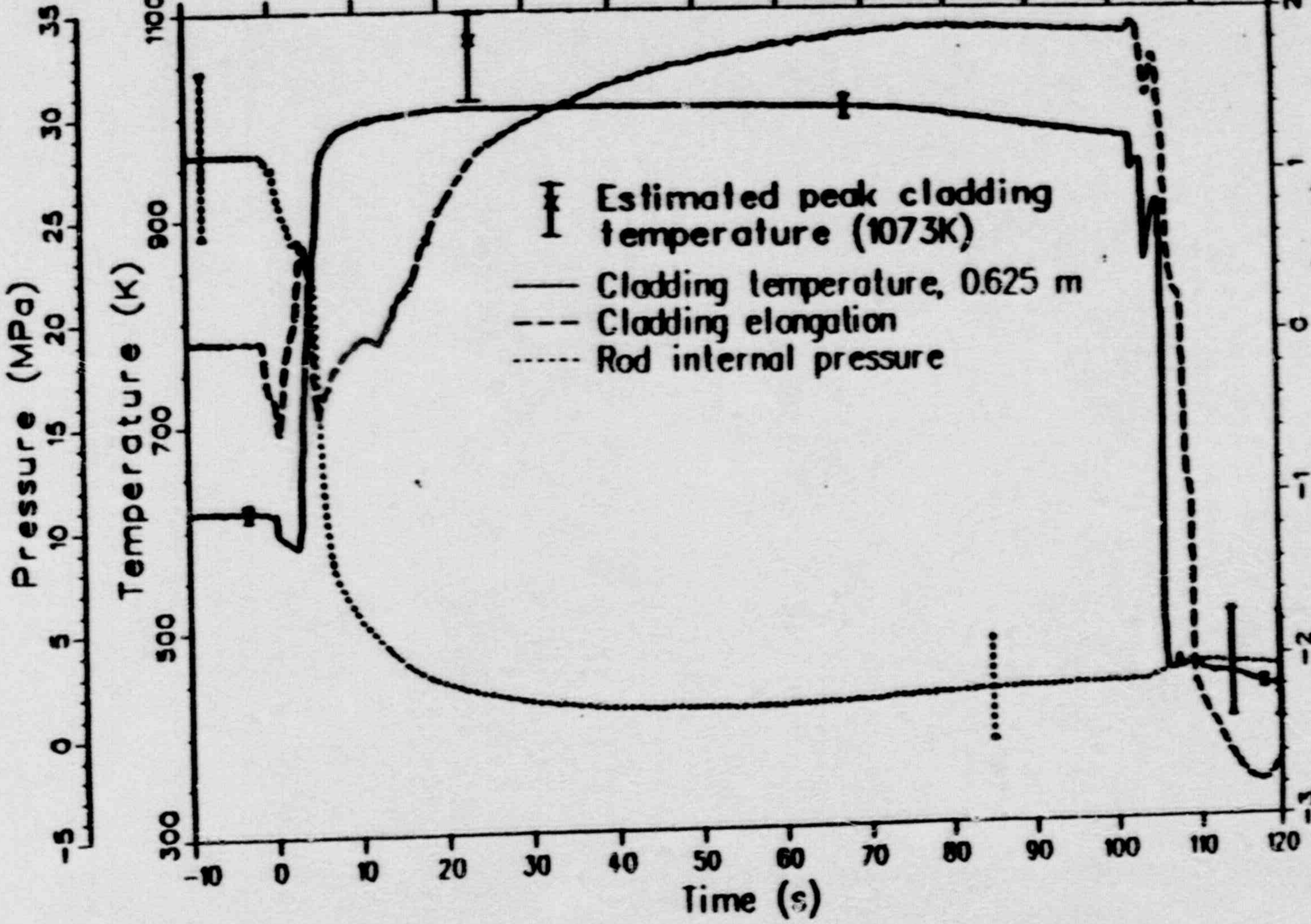

Figure 3. Thermal and mechanical response of Rod 11 during Test LOC-6.

It is clear that TREAT data is anomalous compared to the electrically heated rods and is attributed to difficulties in obtaining accurate temperature data in the burst region. A better source of in-reactor data is the PBF series discussed previously. Figure 4 is a plot from NUREG 0630 (reference 8, exhibit 16). Included are data points with temperature uncertainty for the 9 ruptured rods in the PBF LOC series of tests, and the FRF data from TREAT. It is clear that the more recent PBF data is very consistent with the NUREG-0630 curves.

Observations Regarding LBLOCA Blowdown Rupture of High Burnup Fuel Rods.

The main contributors to fuel cladding rupture are high pressure drop across the cladding and high cladding temperature. Early post-DNB cladding temperatures are determined to a very large degree by pre-accident stored energy which is a function of local peak power (PLHGR), pre-accident gap conductance, effective UO_2 thermal conductivity, blowdown heat transfer, and critical flow model. The CSAU study (reference 2) confirmed this assessment. Of these variables, only PLHGR is controllable by plant operators, and then only to a limited degree. High burnup, third cycle fuel is always placed in low power regions. Pin pressure is determined by pre-pressurization and fission gas release. As shown in reference 3 and 6, pin pressure does not exhibit a direct functional relationship to blowdown cladding temperature.

As noted earlier, the CSAU 17x17 95th percentile PCT of 1447°F (reference 2) could be approximately extrapolated to 1265°F for a high burnup 15x15 pin. The 15x15 PCT calculated at 13.26 kw/ft (reference 7) was 1543°F. The Zion hot pin did not rupture in reference 7. The reference 7 calculation extrapolated to 5.17 kw/ft would result in a PCT of about 1187°F. Therefore, 1265°F determined previously appears to be a good high side estimate of blowdown PCT for a high burnup 15x15 pin. In both reference 7 and reference 2, this blowdown peak occurred between 5 and 9 seconds.

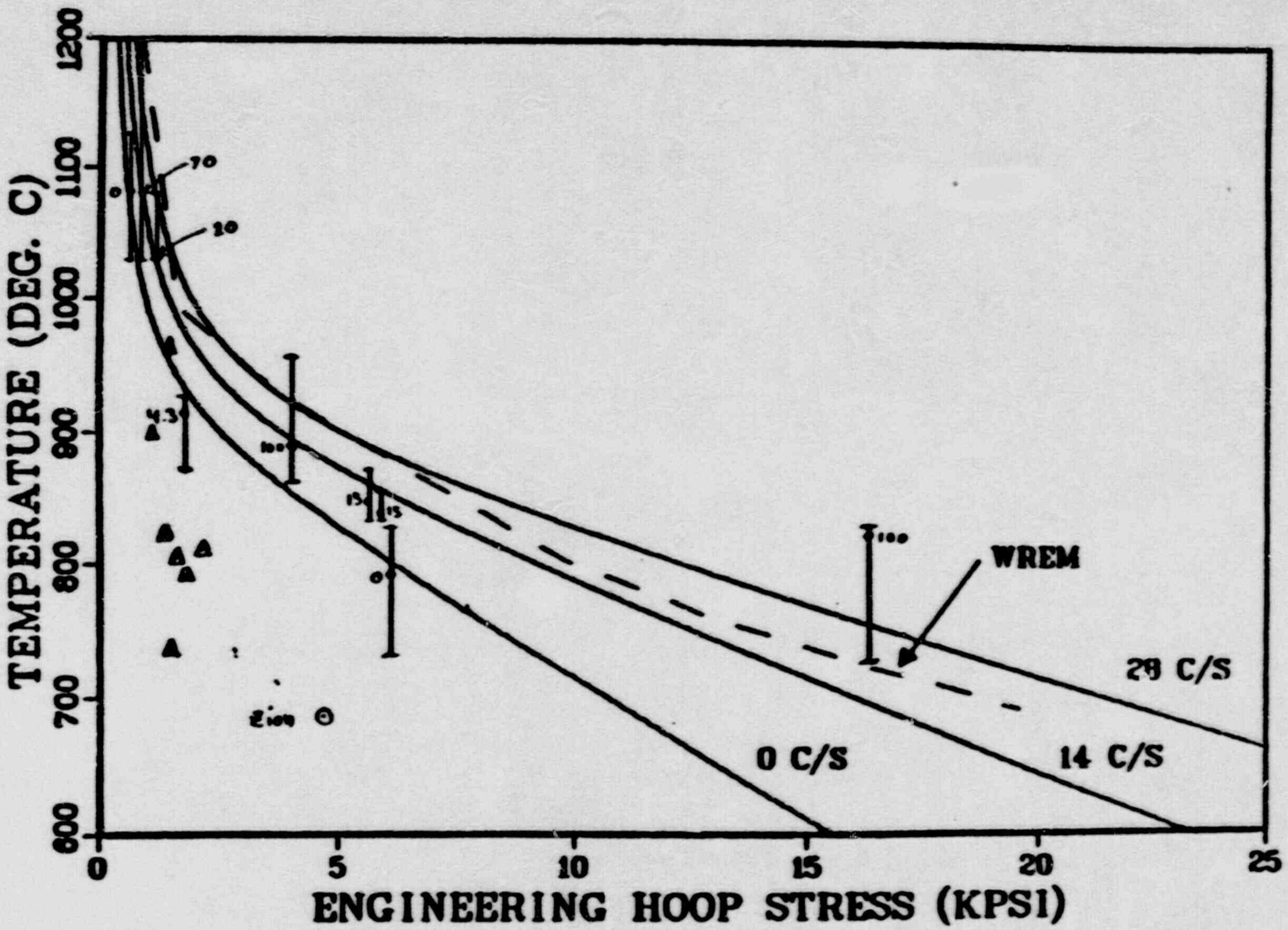

Fig. 4. WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate, with PBF LBLOCA test results.
Numbers adjacent to test data points are ramp rates in °C/sec. TREAT data points are shown as triangles. The Zion high burnup pin estimate is given by a circle.

PAD 3.4 calculations for a 15x15 pin were not performed in reference 5, but by extrapolating 17x17 PAD analyses and the values in Table 1, it is estimated that the pre-accident 15x15 pin pressure at end of cycle 3 would be about 1800 psi. Based on the pressure decrease calculated for the 15x15 pin in the first 5 seconds in reference 7, it is estimated that the pin pressure at 5 seconds for a high burnup 15x15 pin would be 1520 psi. The system pressure at that time was determined to be 920 psi. The pressure drop across the clad is therefore 600 psi and the engineering hoop stress is estimated to be 4.7 KPSI. As shown in Figure 3, this is well below the NUREG-0630 curves and even below the TREAT data. Therefore, it is not expected that any high burnup pins which have low LHGRs would experience any early blowdown ruptures.

It should be noted, however, that this is based on extrapolations, and surely direct calculations based on actual condition would be preferable. Also, if indeed high burnups are expected in the future with higher LHGR, this issue should be revisited. In fact, when significant changes in fuel design models and blowdown LOCA models are proposed, this issue should also be addressed.

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