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

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.

TABLE OF CONTENTS

	Page
1 FISSION PRODUCT RELEASE FROM FUEL AND CONTAINMENT USED IN ACCIDENT ANALYSES	1-1
1.1 Radiological Source Terms Within the Core	1-1
1.2 LOCA: Reg. Guide 1.4 Criteria: Application to Zion	1-1
1.3 LOCA: BTP CSB 6-4 B5. Criteria	1-6
1.3.1 Characteristics of Fuel Failure Giving Fission Product Release During Postulated Accidents	1-8
1.3.2 Characteristics of Fission Product Released From Failed Fuel During Postulated Accidents	1-12
1.3.2.1 General	1-12
1.3.2.2 Regulatory Guide 1.25	1-13
1.3.2.3 Regulatory Guide 1.77	1-15
1.3.2.4 Summary	1-17
1.4 BTP CBS 6-4 B5 Criteria: Application to Zion	1-18
2 OFFSITE DOSE CONSEQUENCES: SUMMARY	2-1
2.1 Basis for Calculations	2-1
2.2 Offsite Doses	2-2
2.2.1 RG 1.4 Source Terms Released Immediately on LOCA	2-2
2.2.2 10% Gap Activity Released on DNBR	2-2
2.2.3 Equilibrium Gap Activity Released on DNBR	2-2

TABLE OF CONTENTS (Continued)

Page

2.2.4 RCS @ 60 μC/gm Activity: Released to Containment Immediately on a LOCA 2-2

2.2.5 RCS @ 60 μC/gm Activity: Released Progressively to Containment on RCS Discharge from a LOCA 2-2

2.3 Conclusions 2-3

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

3.1 Preliminary 3-1

3.2 Review 3-2

3.3 Summary 3-5

4 CONCLUSIONS 4-1

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 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 <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 <u>LOCA: BTP CSB 6-4, B5 Criteria</u>

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 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.

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 (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 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 x 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 concequences 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/cm³ 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/cm³ 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. 29), 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 (α + β) 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.

REFERENCES

1. Letter from E. P. Rahe (W) to James R. Miller (NRC): Subject: WCAP 9220-P-A. Rev. 1 "Westinghouse ECCS Evaluation Model 1981 Version."

2. Safety Evaluation Report on the 1981 Version of the Westinghouse Large Break ECCS Evaluation Model.

3. Westinghouse Electric Corporation: Locta-IV Program, Loss of Coolant Transient Analysis. Bordelon et al., WCAP8301. June 1974.

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.

6. Letter from T. M. Anderson (W) to J. R. Miller, dated May 15, 1981. Proposed 1981 version of the Westinghouse Appendix K Evaluation Model.

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).

8. USNRC: Safety Evaluation Report on the 1981 Version of the Westinghouse Large Break ECCS Evaluation Model.

9. Letter from J. Stoltz (NRC) to T. Anderson (W); Subject: Safety Evaluation of WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version." February 1978.

10. "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests," WCAP-7379-L, Volume I (Proprietary) and Volume II (Non-proprietary), September 1969.

11. "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Multi-Rod Tests," WCAP-7495-L, Proprietary, Volume I - Test setup and Results; Volume II - Analyses of Results.

12. Lorenz, Hobson & Parker. ORNL. Fuel Rod Failure Under Loss-of-Coolant Conditions In TREAT. Nuclear Technology, Vol II, May 1971-August 1971.

13. D. G. Hardy: High Temperature Expansion and Rupture Behavior of Zircaloy Tubing, Atomic Energy of Canada Ltd. Topical Meeting on Water Reactor Safety, Salt Lake City, Utah, March 26-28, 1973.

14. Letter from G. F. Owsley (EXXON) to J. R. Miller (NRC); Subject: Acceptance for Referencing of Topical Report XN-76-47(P), dated Nov. 5, 1981.

15. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," USNRC Report NUREG-0630, April 1980.

16. D. H. Risher, et al., "Safety Analysis for Revised Fuel Rod Internal Pressure Design Bases," Westinghouse Report WCAP-8963-P-A, August 31, 1978.

17. Letter from C. Eichelding (W) to J.F. Stolz (NRC) concerning "Review of WCAP-8963(P)," dated May 18, 1977.

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).

19. Letter from J. F. Stolz, USNRC, to T. M. Anderson, Westinghouse, Subject: Safety Evaluation of WCAP-8720, dated February 9, 1979.

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.

24. WCAP-8785: "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations."

25. NUREG-0800 Standard Review Plan for the Review of Safety Analyses Report for Nuclear Power Plants. July 1981, Section 6.2.4, Containment Isolation System, Branch Technical Position CSB 6-4, Containment Purging During Normal Plant Operations.

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)

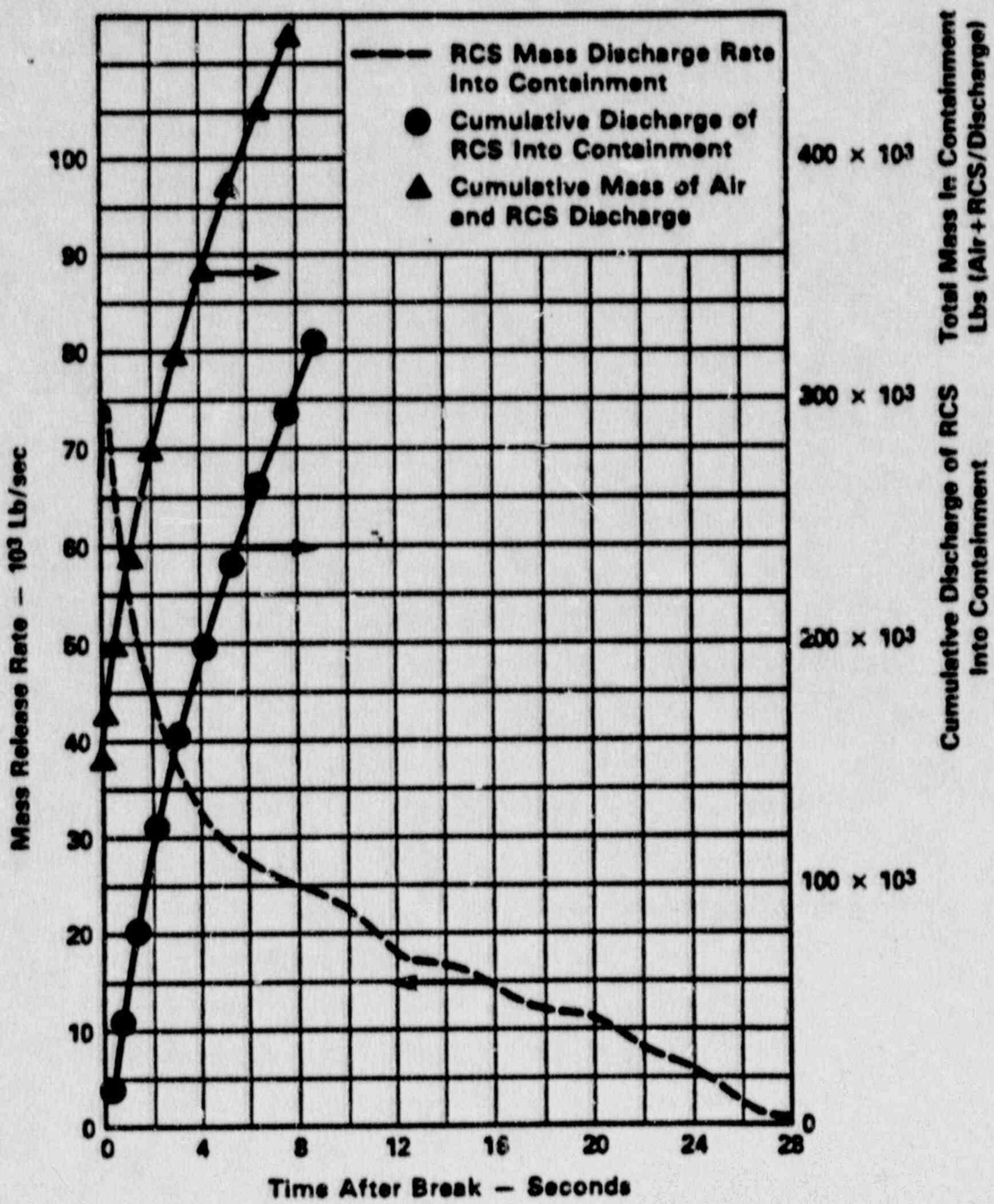
Item No.	Source	Radiological Sources	Curies Discharged I 131 EQ	Site/Excl. Boundary Dose (Thyroid (REM)
1	Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	73.5	18.7
2	RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119×10^5 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

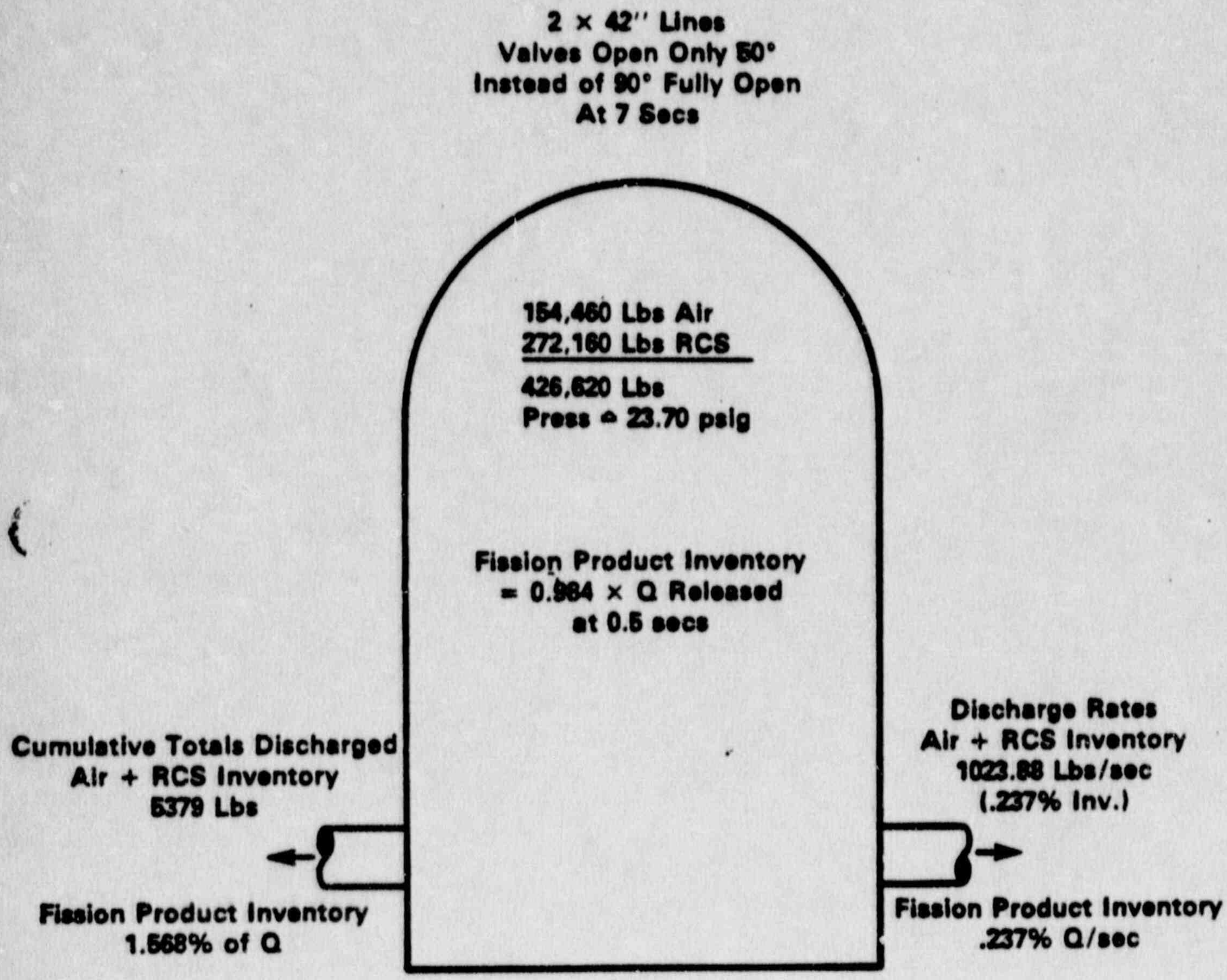
[Licensee has used 9×10^{-4} sec/m³ for SARs]

3

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 proouct 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°)

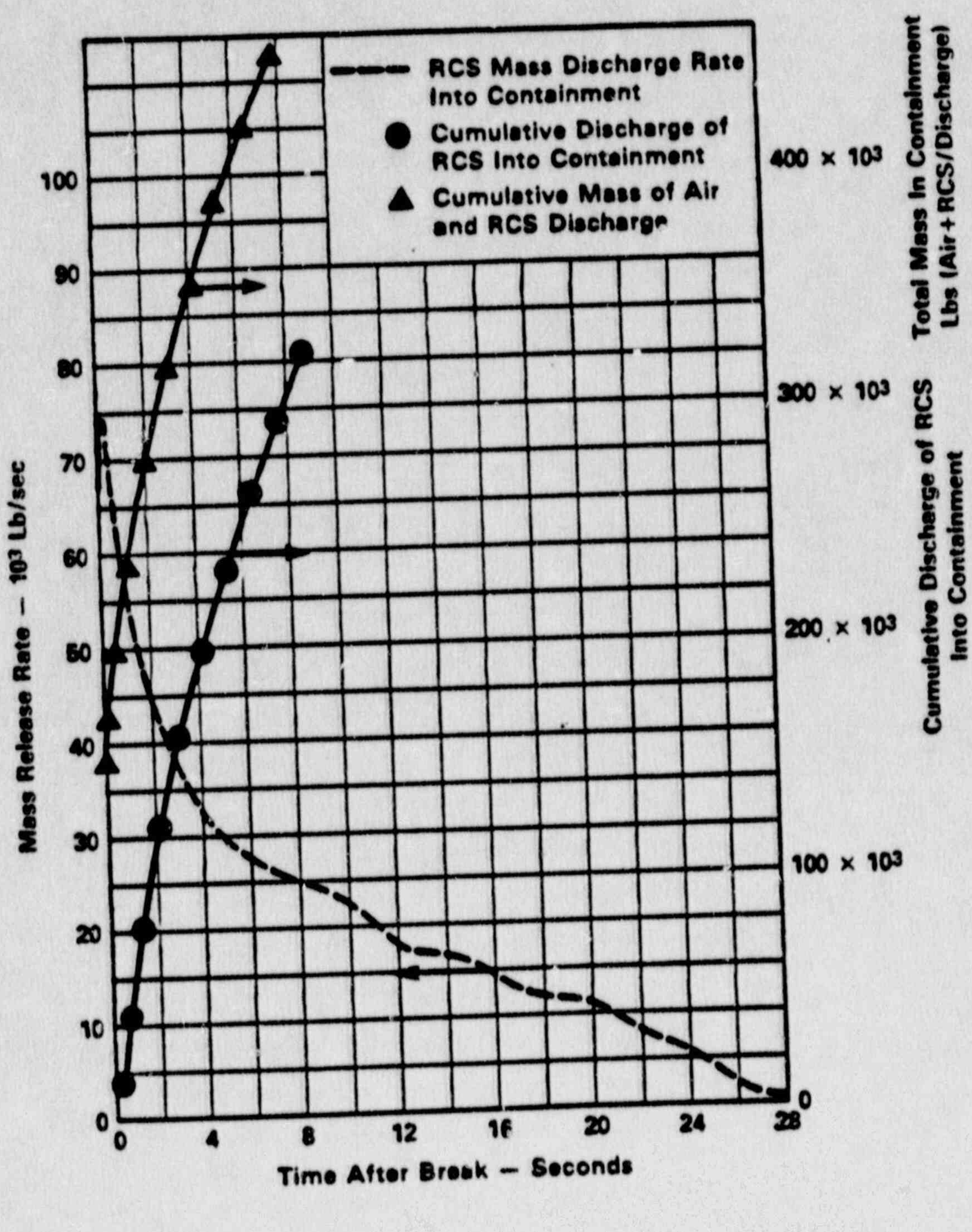
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

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

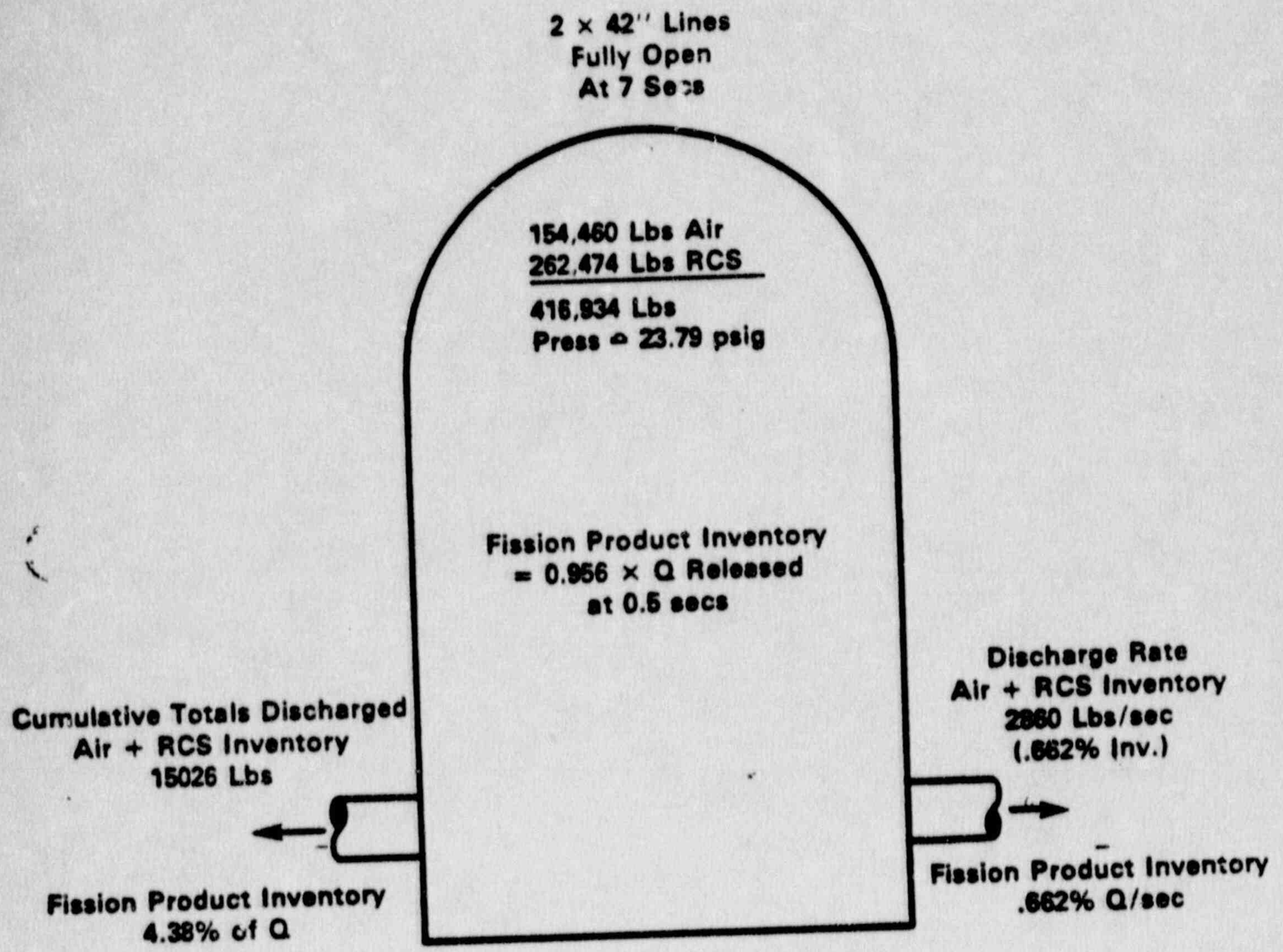
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

[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

? 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

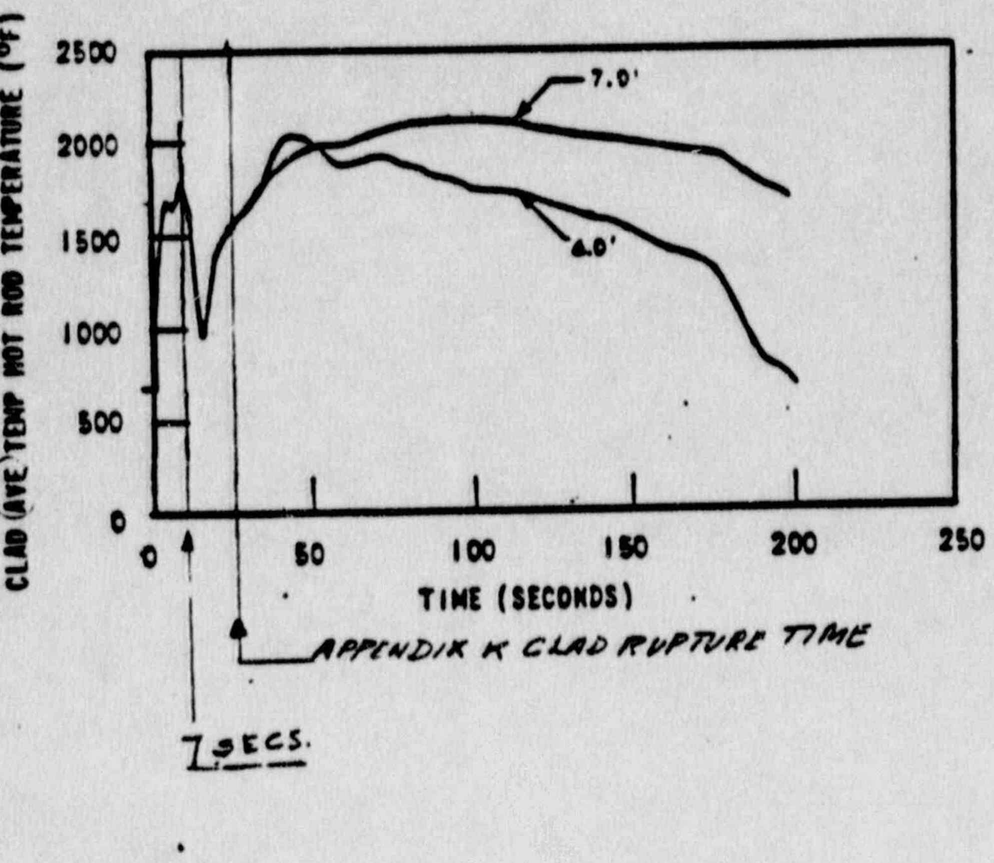


Figure 14 F.2-19a Peak Clad Temperature - DECLG (C_D=1.0)
(Unit 1)

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:

	<u>Damage Limit</u>	<u>Design Limit</u>
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.

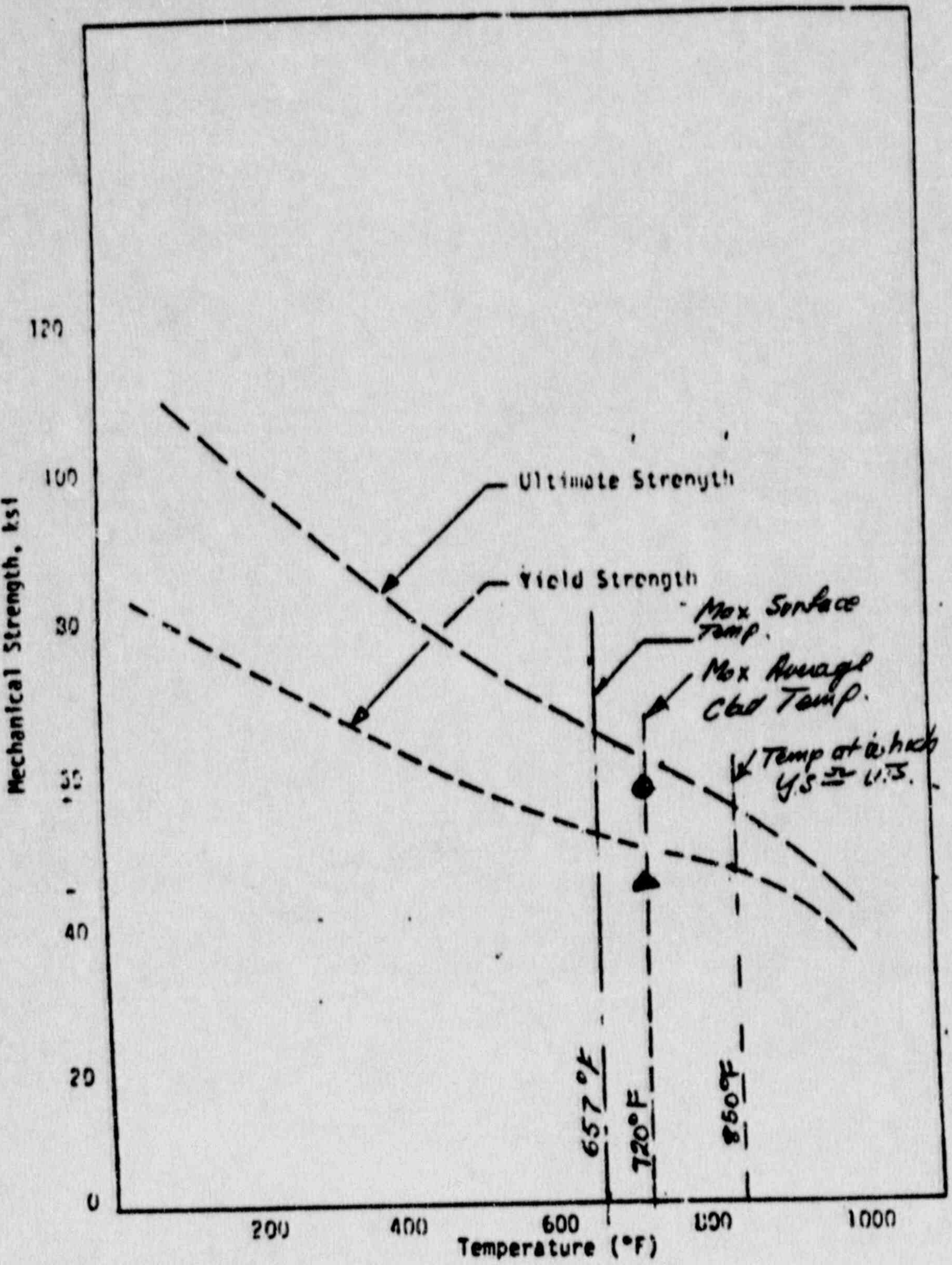
The other parameters having an influunce on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

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

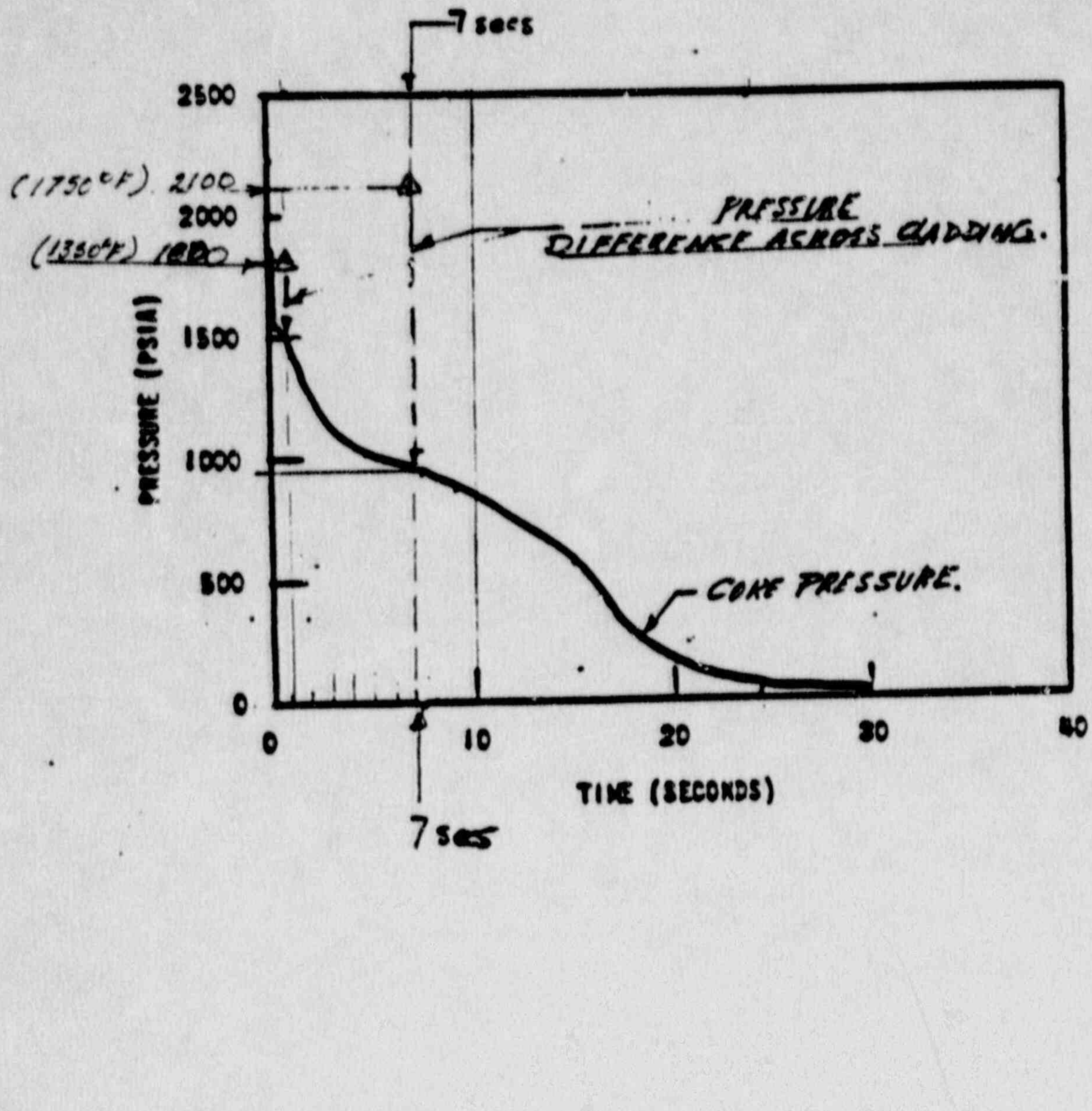


Figure 14 F.2-10a Core Pressure - DECLG (C_D=1.0)
(Unit 1)

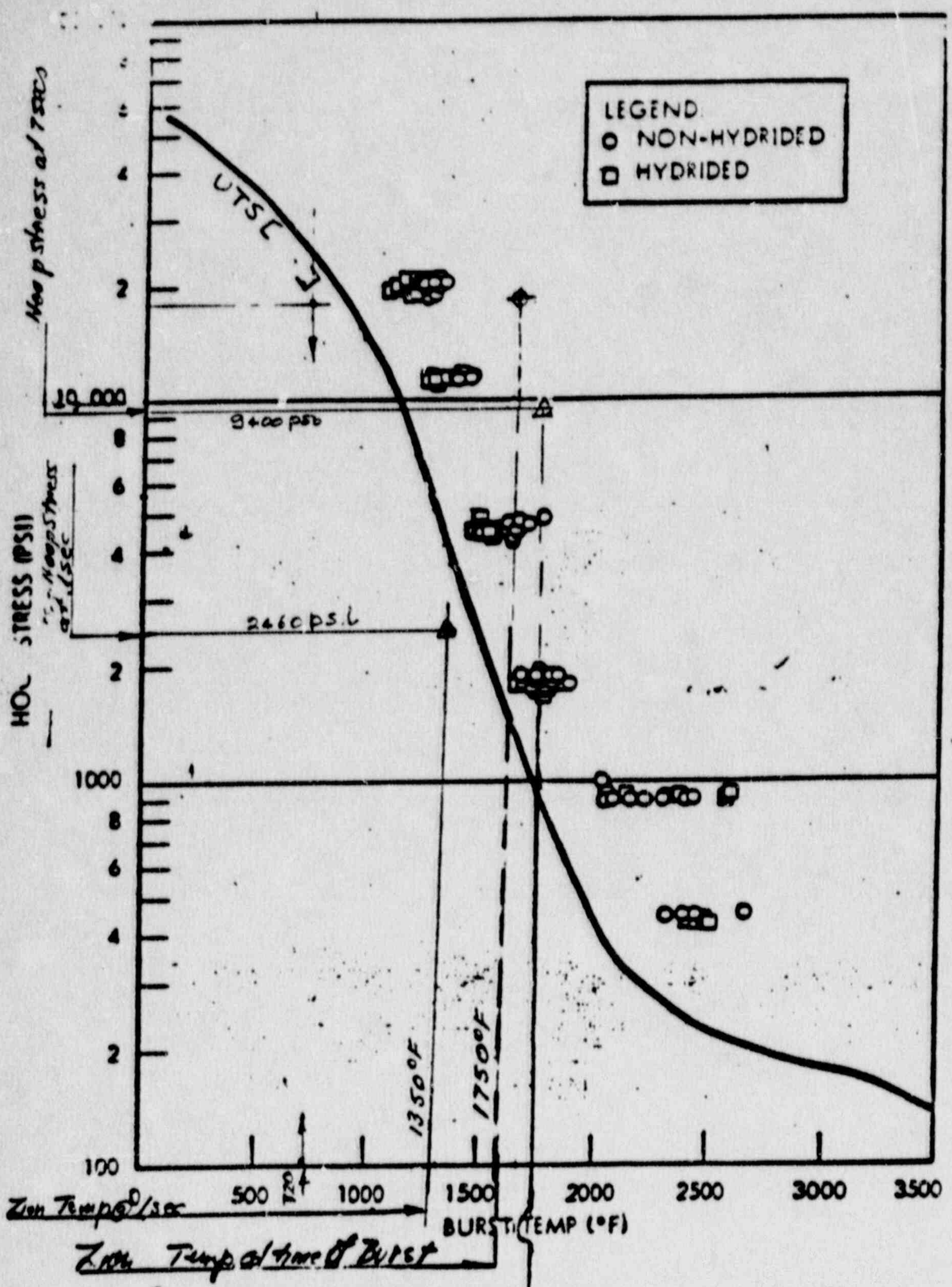
TABLE 1

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

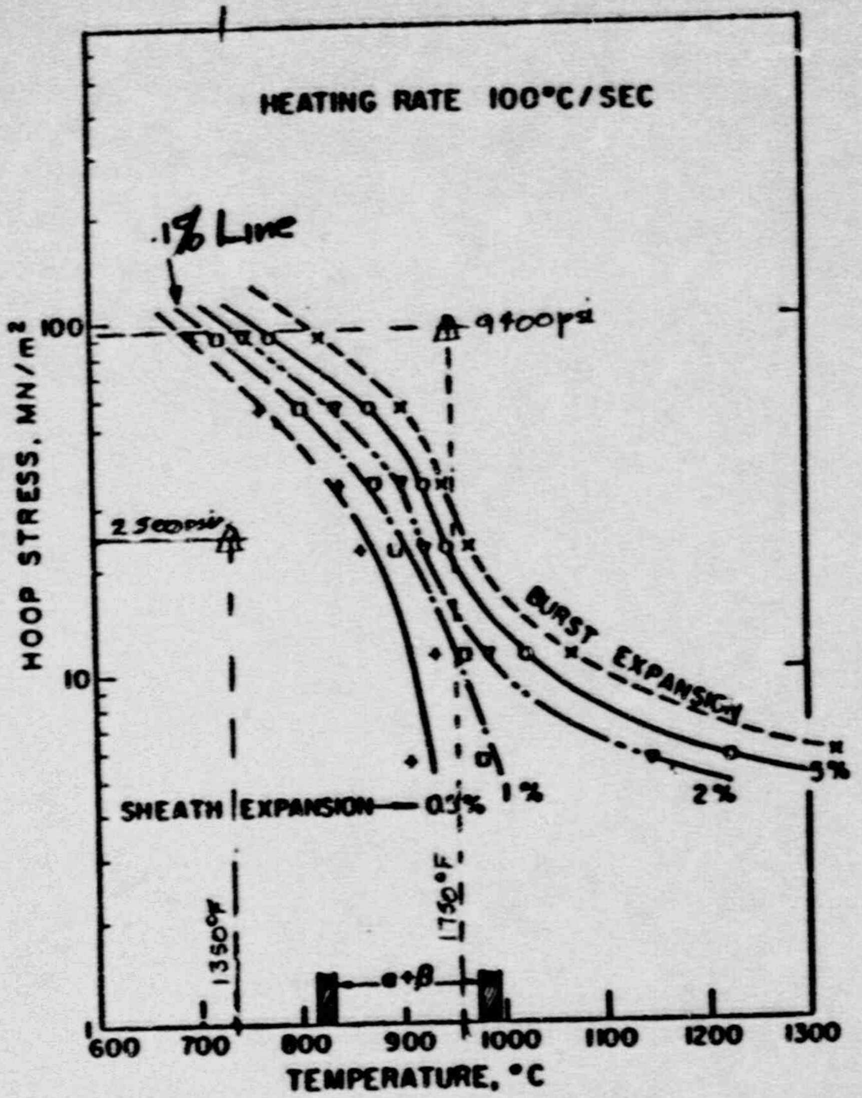
Design	Hoop Stress (psi) for a 600 psi Differential Across the Cladding Wall
B&W 15x15	4570
B&W 17x17	4540
C-E 16x16	4280
W 15x15	4910
W 17x17	4690
GE 8x8	4050
ENC 15x15*	3940
ENC 8x8**	3880

* D. C. Cook, Unit 1

** Oyster Creek



Burst Temperature versus Stress at Burst



263

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

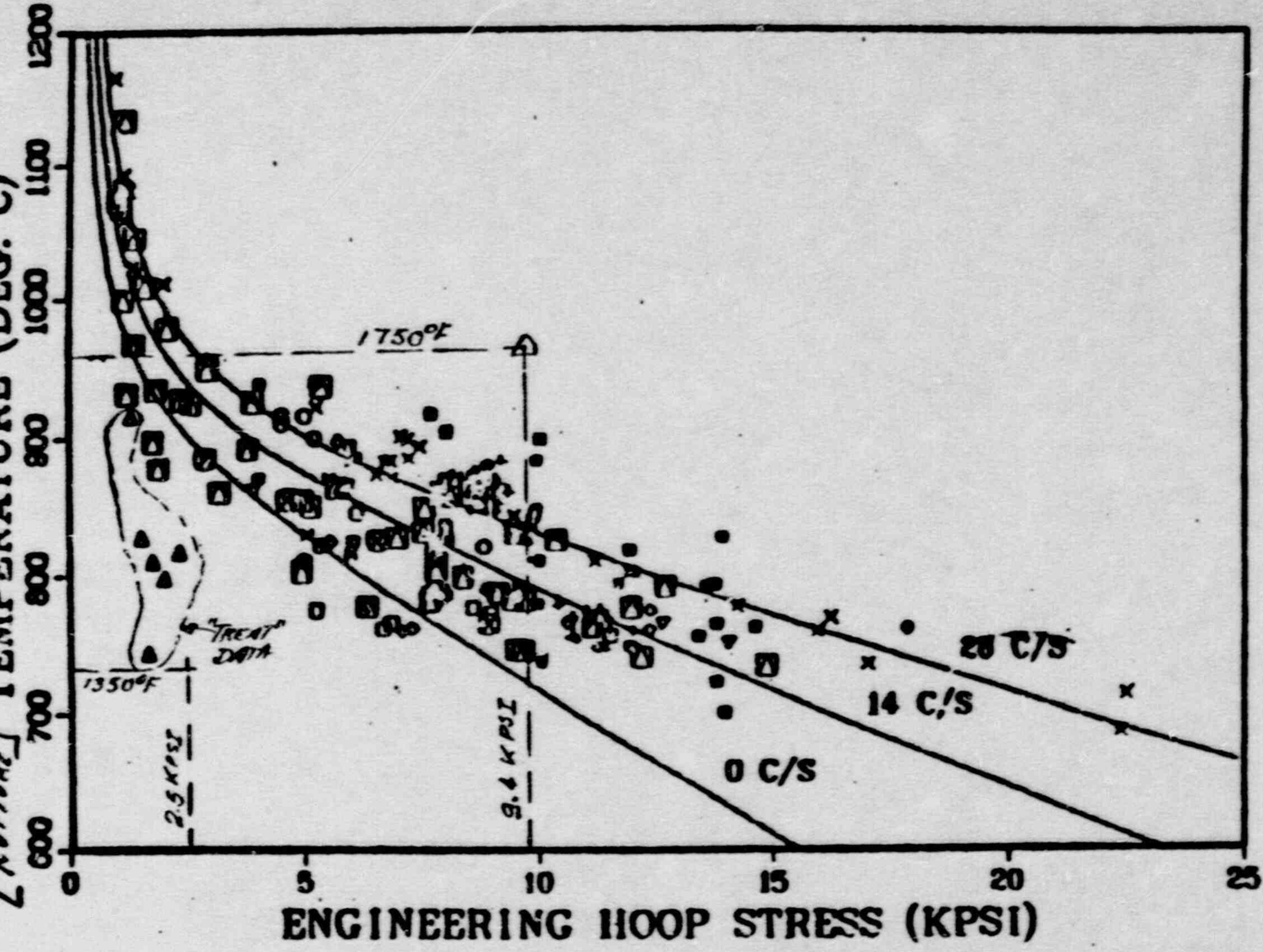


Fig. 3 ORNL correlation of rupture-temperature as a function of engineering hoop stress and temperature-ramp rate with data from internally heated Zircaloy claddinn in aqueous atmospheres.

Rev 1
16

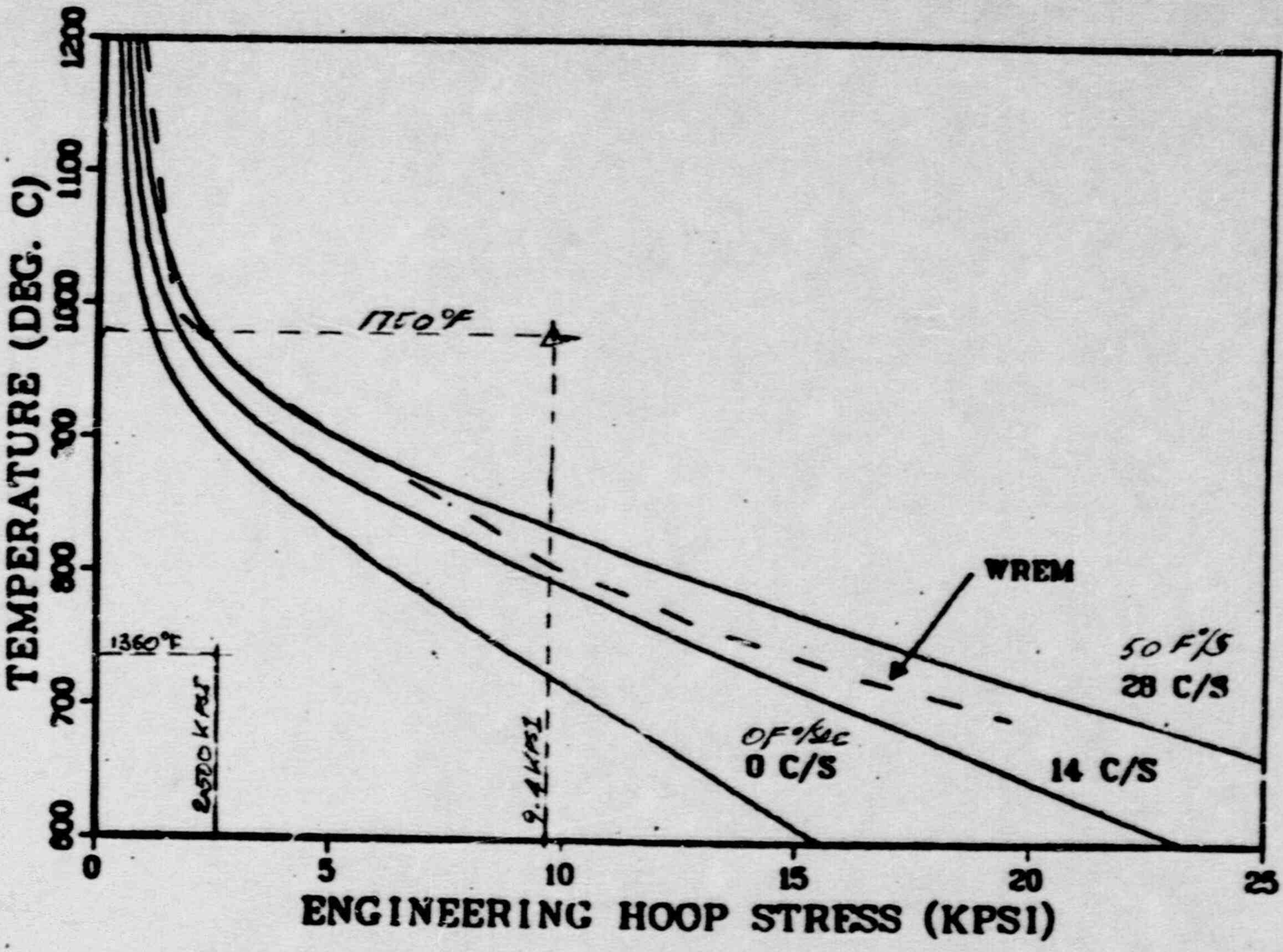


Fig. 17 WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

17
Rev 1

1974 . LATER 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 (a,c)
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 W estinghouse 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)

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

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)

$$\frac{\Delta d^{o}}{d_o} \left[\qquad \right] \qquad (3\text{-}71B)$$

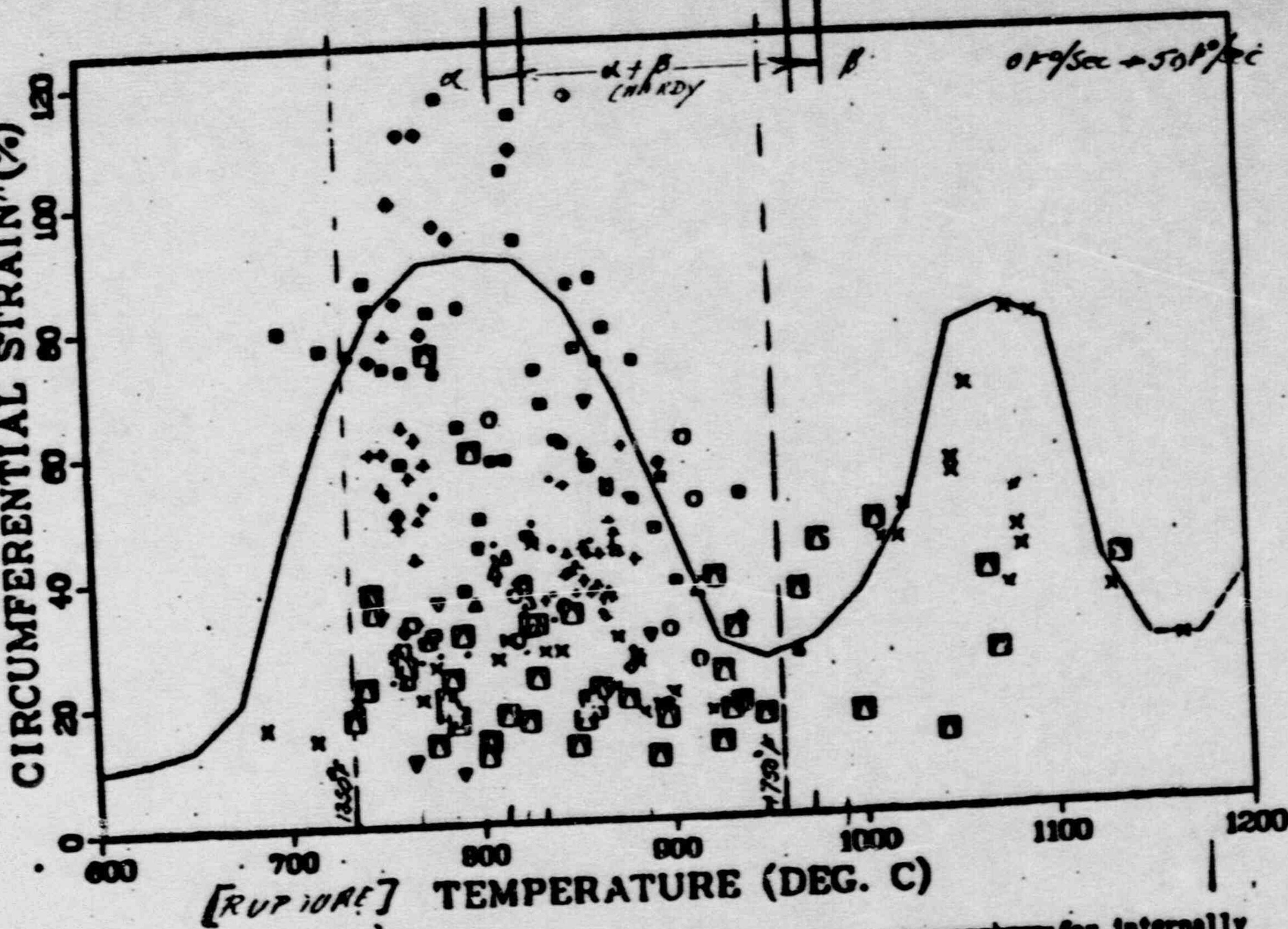


Fig. 8 ~~Maximum circumferential strain as~~ a function of ~~rupture temperature~~ for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

NUREG-75/077

THE ROLE OF FISSION GAS RELEASE IN REACTOR LICENSING

CORE PERFORMANCE BRANCH

U. S. NUCLEAR REGULATORY COMMISSION

NOVEMBER 1975

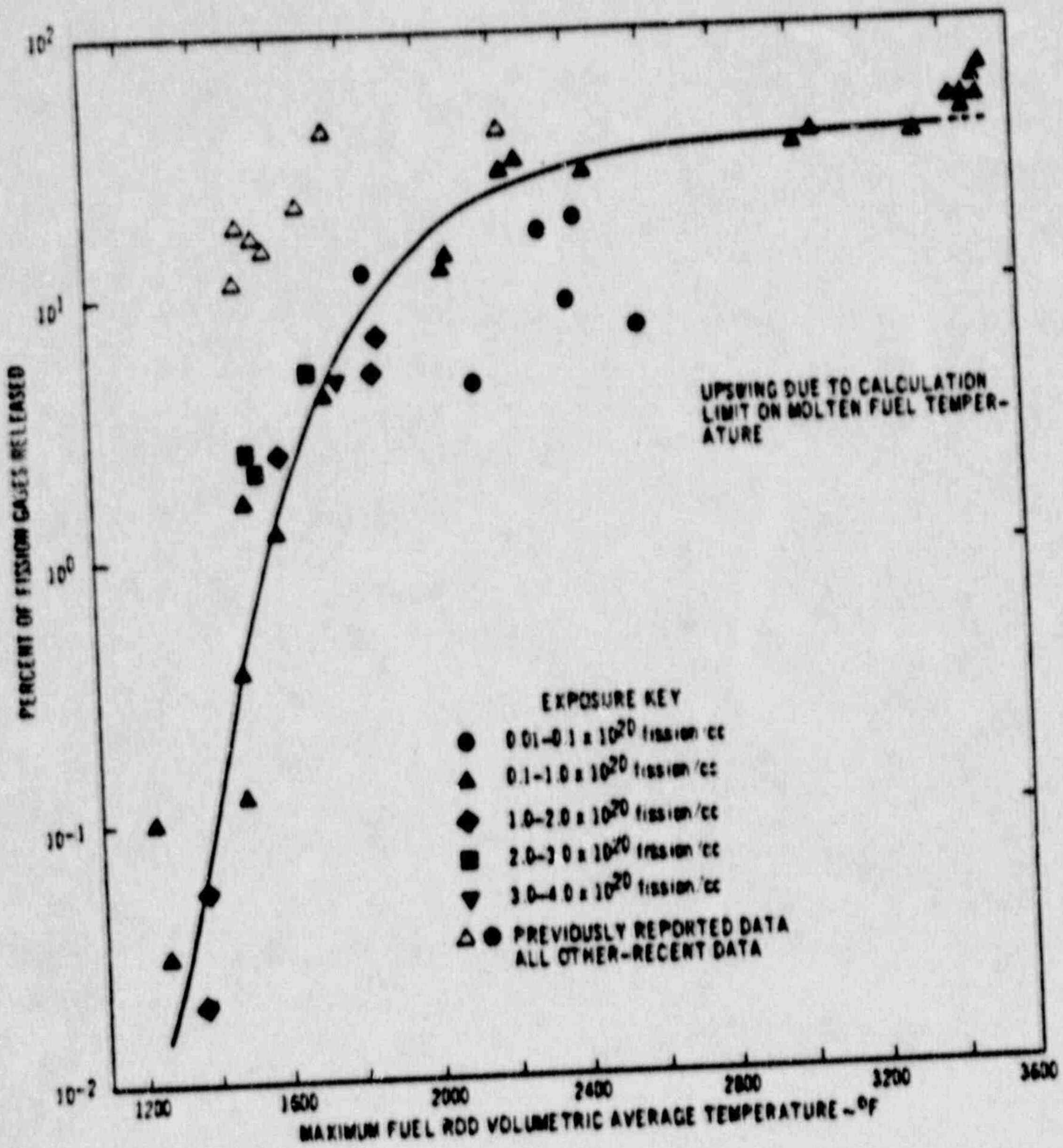


Fig. 2. The Hoffman & Coplin correlation of fission gas release as a function of temperature (from Ref. 35).

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

Thomas E. Murley

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.

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

cc w/enclosures:
C. Patel

CONTACT: R. Licciardo
X20876

DISTRIBUTION
Docket Files
Plant File
JWermiel
JKudrick
RArchitzel
AThadani
LShao
TGody (SALP only)
RLicciardo

RL

SPLB:DEST	SPLB:DEST	SPLB:DEST
RLicciardo;cf	JKudrick	JWermiel
5/11/89	5/ /89	5/ /89

5520 NAME: Zion TACs 55417/8 Licciardo

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."

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