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

August 31, 1989

#18

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

FROM: DPV Standing Review Panel

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

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

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

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

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

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

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

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

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

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

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

The Panel recommends that:

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

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

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

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

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

Frank J Congel
Frank J. Congel

Charles E Rossi
Charles E. Rossi

Frank J. Miraglia Jr
Frank J. Miraglia, Jr.

Enclosures:
As stated

cc: R. Licciardo
J. Larkins

Ref 1

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

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

Response:

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

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

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

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

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

b. <u>SRP Section 4.2 "Fuel System Design"</u>

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

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

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

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

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

Response:

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

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

Response:

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

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

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

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

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

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

Document Name:
MGMT RESP. TO OVERSIGHT COMMIT

Requestor's ID:
FAIRCLOT

Author's Name:
j. wermiel

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

TABLE 14.3.2-1

LARGE BREAK - 1984 ANALYSIS - SEQUENCE OF EVENTS

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

Note: All times in seconds

* With Replacement Reactor Containment Fan Coolers as installed 1985

Attachment I

ECCS Analysis from Zion Updated FSAR

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Revision 1
June 26, 1986

TABLE 14.3.2-2

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

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

Inputs

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

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

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

TABLE 14.3.2-4

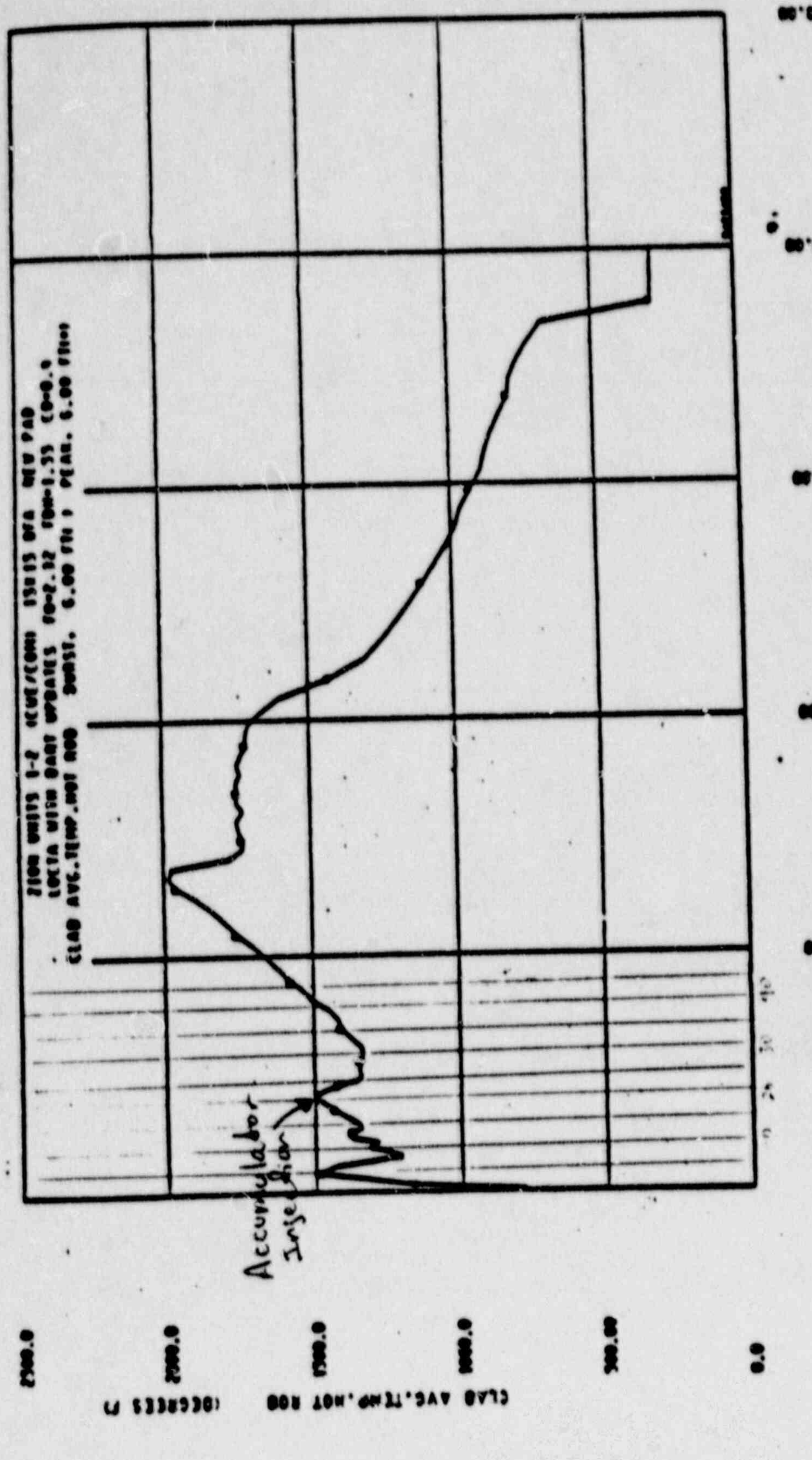
SMALL BREAK RESULTS - 1984 ANALYSIS - 6" BREAK CASE

<u>Results</u>

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

<u>Inputs</u>

Core Power - 102% of ESDR	=	3390 MWt
Peak Linear Power - 102% of	=	See Figure 14.3.2-7
Accumulator Water Volume	=	900 ft³.



PEAK CLAD TEMPERATURE
DECLG(CD = 0.4)

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Figure 14.3.2-4G

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

6.2.4 CONTAINMENT ISOLATION SYSTEM

REVIEW RESPONSIBILITIES

Primary - Containment Systems Branch (CSB)

Secondary - None

I. AREAS OF REVIEW

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

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

1. The design of containment isolation provisions, including:

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

   b. The actuation and control features for isolation valves.

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

   d. The valve actuation signals.

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

   f. The mechanical redundancy of isolation devices.

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

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

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

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

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

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

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

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

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

II. ACCEPTANCE CRITERIA

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

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

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

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

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

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

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

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

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

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

[1]Locked closed isolation valves are defined as sealed closed barriers (see Item II.3.f).

[2]A simple check valve is not normally an acceptable automatic isolation valve for this application.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3. The system is designated seismic Category I.

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

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

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

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

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

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

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

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

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

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

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

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

III. REVIEW PROCEDURES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

IV. EVALUATION FINDINGS

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

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

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

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

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

c. meeting the regulatory position in BTP_______;

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

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

2. Repeat discussion for each regulation cited above.

V. IMPLEMENTATION

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

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

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

VI. REFERENCES

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

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

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

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

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

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

7. 10 CFR Part 50, Appendix A, General Design Criterion 56, "Primary Containment Isolation."

8. 10 CFR Part 50, Appendix A, General Design Criterion 57, "Closed System Isolation Valves."

9. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

10. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

11. Regulatory Guide 1.29, "Seismic Design Classification."

12. Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems."

13. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation," attached to this SRP section.

14. 10 CFR Part 100, "Reactor Site Criteria."

15. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

16. NUREG-0737, "Classifications of TMI Action Plan Requirements."

17. NUREG-0718, "Licensing Requirements for Pending Application for Construction Permits and Manufacturing License."

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the lines associated with the containment purge and vent systems. It supplements the position taken in SRP Section 6.2.4.

While the containment purge and vent systems provide plant operational flexibility, their designs must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on their use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously. There is similar disparity in the need for, and use of, containment vent systems at operating plants.

Containment purge systems have been used in a variety of ways; for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. Containment vent systems are typically used to relieve the initial containment pressure buildup caused by the heat load imposed on the containment atmosphere during reactor power ascension, or to periodically relieve the pressure buildup due to the operation of pneumatic controllers. However, the purge and vent lines provide an open path from the contaiment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR Part 100 guidelines values.

The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the contaiment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines.

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

The size of the purge lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR Part 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR, or HTGR.

B. BRANCH TECHNICAL POSITION

The systems used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. General Design Criterion 54 requires that the reliability and performance capabilities of containment isolation valves reflect the importance of safety of isolating the systems penetrating the containment boundary. Therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in Branch Technical Position MEB-2, "Pump and Valve Operability Assurance Program." (Also see SRP Section 3.10.) The design basis for the valves and actuators should include the build-up of containment pressure for the LOCA break spectrum, and the supply line and exhaust line flows as a function of time up to and during valve closure.

   b. The number of supply and exhaust lines that may be used should be limited to one supply line and one exhaust line, to improve the reliability of the isolation function as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in the evaluation of the emergency core cooling system effectiveness and 10 CFR Part 100 regarding offsite radiological consequences.

   c. The size of the lines should not exceed about eight inches in diameter, unless detailed justification for larger line sizes is provided, to improve the reliability and performance capability of the isolation and containment functions as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in evaluating the emergency core cooling system effectiveness and 10 CFR Part 100 regarding the offsite radiological consequences.

d. As required by General Design Criterion 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria, to reflect the importance to safety of isolating these lines. General Design Criterion 56 establishes explicit requirements for isolation barriers in purge system lines.

e. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. Furthermore, if energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can effect the isolation function.

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds, to facilitate compliance with 10 CFR Part 100 regarding offsite radiological consequences.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves during reactor operation.

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

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

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters, and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The maximum allowable leak rate of the purge isolation valves should be specified on a case-by-case basis giving appropriate consideration to valve size, maximum allowable leakage rate for the containment (as defined in Appendix J to 10 CFR Part 50), and where appropriate, the maximum allowable bypass leakage fraction for dual containments.

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 1), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents. "Coolability," in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, §50.46 (Ref. 3).

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

The fuel system review covers the following specific areas.

A. Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that damage will be limited to acceptable levels. The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. Design Evaluation

The performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents is reviewed to determine if all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. On-line fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by CPB to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

Specific criteria necessary to meet the requirements of 10 CFR Part 50, §50.46; General Design Criteria 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 identified in subsection I of this SRP section are as follows:

A. Design Bases

The fuel system design bases must reflect the four objectives described in subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

1. Fuel System Damage

This subsection applies to normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Such damage criteria should address the following to be complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the ASME Code (Ref. 4) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles (Ref. 5). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the Safety Analysis Report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

2. Fuel Rod Failure

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

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

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

(a) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 7) for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in

Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown ([illegible]) to be insufficient for primary hydride formation.

(b) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

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

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

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

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

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

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

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

3. Fuel Coolability

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addresses the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Control rod insertability criteria are also addressed in this subsection. Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, §50.46, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800°F previously described in Reference 11 is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of Appendix K of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data (such as Refs. 10, 12, and 13) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 2200°F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

B. Description and Design Drawings

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

Type and metallurgical state of the cladding
Cladding outside diameter
Cladding inside diameter
Cladding inside roughness
Pellet outside diameter
Pellet roughness
Pellet density
Pellet resintering data
Pellet length
Pellet dish dimensions
Burnable poison content
Insulator pellet parameters
Fuel column length
Overall rod length
Rod internal void volume
Fill gas type and pressure
Sorbed gas composition and content
Spring and plug dimensions
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawing have also been found necessary for an acceptable fuel system description:

Fuel assembly cross section
Fuel assembly outline
Fuel rod schematic
Spacer grid cross section
Guide tube and nozzle joint
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Burnable poison rod assembly cross section
Burnable poison rod assembly outline
Burnable poison rod schematic
Orifice and source assembly outline

C. Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed. Those methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the Safety Analysis Report by reference.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed when practical to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested prior to irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

Spacer grid structural tests
Control rod structural and performance tests
Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
Fuel assembly growth
Fuel assembly bowing
Channel box wear and distortion
Fuel rod ridging (PCI)
Crud formation
Fuel rod integrity
Holddown spring relaxation
Spacer grid spring relaxation
Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished prior to operation of a full core of that design. This inability to perform in-reactor testing may result from an incompatability of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see subsection II.D below).

3. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical metnods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient*

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 14 and 15). Examples of previous fuel performance code reviews are given in References 16 through 20.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core

*Although needed in fuel performance codes, this model is reviewed as described in SRP Section 4.4.

power distributions (power spiking, see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR, see SRP Section 4.4), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in Reference 21 and in Regulatory Guide 1.126 (Ref. 22). To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References 23 and 24.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 25. Interim methods that may be used prior to compliance with this guidance are given in Reference 26. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References 10, 12, and 13.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate, the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27). For non-LOCA applications, other correlations may be used if justified.

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

D. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

1. <u>Testing and Inspection of New Fuel</u>

Testing and inspection plans for new fuel should include verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Details of the manufacturer's testing and inspection programs should be documented in quality control reports, which should be referenced and summarized in the Safety Analysis Report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. Where the overall testing and inspection programs are essentially the same as for previously approved plants, a statement to that effect should be made. In that case, the details of the programs need not be included in the Safety Analysis Report, but an appropriate reference should be cited and a (tabular) summary should be presented.

2. <u>On-line Fuel System Monitoring</u>

The applicant's on-line fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. References 32 and 33 evaluate several common detection methods and should be utilized in this review.

Surveillance is also needed to assure that B_4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as described in Reference 34 are acceptable.

3. <u>Post-irradiation Surveillance</u>

A post-irradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered, i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features.

For a fuel design like that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, or crud deposition. There should also be a commitment in the program to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, there should exist a continuing fuel surveillance effort for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with prototype testing discussed in subsection II.C.2. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

III. REVIEW PROCEDURES

For construction permit (CP) applications, the review should assure that the design bases set forth in the Preliminary Safety Analysis Report (PSAR) meet the acceptance criteria given in subsection II.A. The CP review should further determine from a study of the preliminary fuel system design that there is reasonable assurance that the final fuel system design will meet the design bases. This judgment may be based on experience with similar designs.

For operating license (OL) applications, the review should confirm that the design bases set forth in the Final Safety Analysis Report (FSAR) meet the acceptance criteria given in Subsection II.A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have reviewed the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review is dependent should be referenced so that a completely documented safety evaluation is contained in the plant safety evaluation report. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Certain generic reviews have also been performed by CPB reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References 9, 11, 21, 31, 32, 35, and 36, and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (Refs. 6, 22, 28-30, and 41) should also be mentioned in the plant safety evaluation reports. Deviation from these guides or positions should be explained. After briefly discussing related previous reviews, the plant safety evaluation should concentrate on areas where the application is not identical to previously reviewed and approved applications and areas related to newly discovered problems.

Analytical predictions discussed in Subsection II.C.3 will be reviewed in PSARs, FSARs, or licensing topical reports. When the methods are being reviewed, calculations by the staff may be performed to verify the adequacy of the analytical methods. Thereafter, audit calculations will not usually be performed to check the results of an approved method that has been submitted in a Safety Analysis Report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time the clear need arises to reconfirm the adequacy of the method.

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information has been provided to satisfy the requirements of this SRP section and that the evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

The staff concludes that the fuel system of the ______________ plant has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents and thereby meets the related requirements of 10 CFR Part 50, §50.46; 10 CFR Part 50, Appendix A, General Design Criteria 10, 27 and 35; 10 CFR Part 50, Appendix K; and 10 CFR Part 100. This conclusion is based on the following:

1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection (PWR) or drop (BWR), and fuel densification have been performed in accordance with (a) the guidelines of Regulatory Guides 1.60, 1.77, and 1.126, or methods that the staff has reviewed and found to be acceptable alternatives to those Regulatory Guides, and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A to SRP Section 4.2.

2. The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform on-line fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100. In meeting these requirements, the applicant has (a) used the fission-product release assumptions of Regulatory Guides 1.3 (or 1.4), 1.25, and 1.77 and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77 or with methods that the staff has reviewed and found to be an acceptable alternative to Regulatory Guide 1.77.

V. IMPLEMENTATION

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

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

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

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. 10 CFR Part 50, §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

4. "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section III, 1977.

5. W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nucl. Sci. Eng. 20, 1 (1964).

6. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

7. "Standard Specification for Sintered Uranium Dioxide Pellets," ASTM Standard C776-76, Part 45, 1977.

8. K. Joon, "Primary Hydride Failure of Zircaloy-Clad Fuel Rods," Trans. Am. Nucl. Soc. 15, 186 (1972).

9. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

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

11. "Technical Report on Densification of Light Water Reactor Fuels," AEC Regulatory Staff Report WASH-1236, November 14, 1972.

12. F. Erbacher, H. J. Neitzel, H. Rosinger, H. Schmidt, and K. Wiehr, "Burst Criterion of Zircaloy Fuel Claddings in a LOCA," ASTM Fifth International Conference on Zirconium in the Nuclear Industry, August 4-7, 1980, Boston, Massachusetts.

13. R. H. Chapman, "Multirod Burst Test Program Progress Report for January-June 1980," Oak Ridge National Laboratory Report NUREG/CR-1883, March 1981.

14. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, "User's Guide for GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1897, November 1975.

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17. "Fuel Evaluation Model," Combustion Engineering Report CENPD-139-A, July 1974 (Approved version transmitted to NRC April 25, 1975).

18. "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," AEC Regulatory Staff Report, December 14, 1973.

19. "Technical Report on Densification of Exxon Nuclear PWR Fuels," AEC Regulatory Staff Report, February 27, 1975.

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

21. R. O. Meyer, "The Analysis of Fuel Densification," USNRC Report NUREG-0085, July 1976.

22. Regulatory Guide 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification."

23. Memorandum from V. Stello, NRC, to R. C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 14, 1975.

24. Memorandum from D. F. Ross, NRC, to R. C. DeYoung, Subject: CEPAN -- Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.

25. Memorandum from D. F. Ross, NRC, to D. B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

26. Memorandum from D. F. Ross and D. G. Eisenhut, NRC, to D. B. Vassallo and K. R. Goller, Subject: Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, dated February 16, 1977.

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28. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."

29. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of Loss-of-Coolant Accident for Pressurized Water Reactors."

30. 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."

31. "The Role of Fission Gas Release in Reactor Licensing," USNRC Report NUREG-75/077, November 1975.

32. B. L. Siegel and H. H. Hagen, "Fuel Failure Detection in Operating Reactors," USNRC Report NUREG-0401, March 1978.

33. W. J. Bailey, et al., "Assessment of Current Onsite Inspection Techniques for LWR Fuel Systems," Battelle Pacific Northwest Laboratory Report NUREG/CR-1380, Vol. 1, July 1980, Vol. 2, January 1981.

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39. R. L. Grubb, "Technical Evaluation of PWR Fuel Spacer Grid Response Load Sensitivity Studies," Idaho National Engineering Laboratory, NUREG/CR-1020, September 1979.

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U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

APPENDIX A

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE
TO EXTERNALLY APPLIED FORCES
TO
STANDARD REVIEW PLAN SECTION 4.2

A. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This Appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. Background material for this Appendix is given in References 37-40.

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system and reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch. Input for the fuel assembly response to a LOCA should include (a) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures; these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis, and (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the SSE; if a linear response is produced, a spectral analysis may be used in accordance with the guidelines of Regulatory Guide 1.60 (Ref. 41).

2. Methods

Analytical methods used in performing structural response analyses should be reviewed. Justification should be supplied to show that the numerical solution techniques are appropriate.

Linear and non-linear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical.

A sample problem of a simplified nature should be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (Ref. 38). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), simplifying assumptions may be made (e.g., one might use a 3-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are probably conservative and input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-convervatisms, two precautions should be taken: (a) If it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased (by about 30%) to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis. (b) Conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for ±10% variations in input amplitude and frequency; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15%. For example, if ±10% variations in input magnitude or frequency produce a maximum resultant increase of 35%, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant-dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

Independent audit calculations for a typical full-sized core should be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (Ref. 38) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of General Design Criterion 2 as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly components should be calculated for each input (i.e., seismic and LOCA) as described above in Paragraph 1, and the resulting loads should be added by the square-root-of-sum-of-squares (SRSS) method. These combined loads should be compared with the component strengths described in Section C according to the acceptance criteria in Section D.

C. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the most damaging mode should be represented in the vendor's laboratory grid strength tests. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (a) the testing impact velocities correspond to expected fuel assembly velocities, and (b) the crushing load P(crit) has been suitably selected from the load-vs-deflection curves. Because of the potential for different test rigs to introduce measurement variations, an evaluation of the grid strength test equipment will be included as part of the review of the test procedure.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P(crit) should be the 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit).

2. Components Other than Grids

Strengths of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95%) of the distribution of component strengths. Therefore, ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects.

D. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR Part 50, §50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisifed if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit) as defined above, then significant deformation of the fuel assembly would not occur and control rod insertion would not be interfered with by lateral displacement of the guide tubes. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability: (a) combined loads on the channel box must remain below the allowable value defined above for components other than grids; otherwise, additional analysis is needed to show that deformation is not severe enough to prevent control blade insertion, and (b) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe Shutdown Earthquake

Two criteria apply for the SSE: (a) fuel rod fragmentation must not occur as a result of the seismic loads, and (b) control rod insertability must be assured. The first criterion is satisfied by the criteria in Paragraph 1. The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1.

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

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ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

A. INTRODUCTION

Section 50.34, "Contents of applications: technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each application for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the potential risk to public health and safety resulting from operation of the facility. General Design Criterion 28, "Reactivity Limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. General Design Criterion 28 also requires that these postulated reactivity accidents include consideration of the rod ejection accident unless such an accident is prevented by positive means.

This guide identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide-fueled pressurized water reactors (PWRs). In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered on an individual basis. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The rate at which reactivity can be inserted into the core of a uranium oxide-fueled water-cooled power reactor is normally limited by the design of the control rod system to a value well below that which would result in serious damage to the reactor system. However, a postulated failure of the control rod system provides the potential for a relatively high rate of reactivity insertion which, if large enough, could cause a prompt power burst. For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant. This is accompanied by the conversion of nuclear energy, deposited as overpower heat in the fuel and in the coolant, to mechanical energy which, in sufficient quantity, could conceivably disarrange the reactor core or breach the primary system.

The Regulatory staff has reviewed the available experimental information concerning fuel failure thresholds. In general, failure consequences for UO_2 have been insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod as a result of a postulated rod ejection accident provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired.

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position.

A sufficient number of initial reactor states to completely bracket all possible operational conditions of interest should be analyzed to assure examination of upper bounds on ultimate damage. In areas of uncertainty, the appropriate minimum or maximum parameters relative to nominal or expected values should be used to assure a conservative evaluation. The initial reactor states should include consideration of at least the following:

Zero power (hot standby)	– Beginning of Life (BOL) and End of Life (EOL);
Low power	– BOL and EOL;
Full power	– BOL and EOL.

The effects of the loss of primary system integrity as a result of the failed control rod housing should be included in the analysis. It should also be shown that failure of one control rod housing will not lead to failure of other control rod housings.

The approach that should be used in the radiological analysis of a control rod ejection accident is to determine the amount of each gaseous radionuclide released to the primary containment and, with this information in conjunction with the procedures set forth in Appendix B of this guide, to determine the radiological consequences of this accident for a pressurized water reactor.

C. REGULATORY POSITION

Acceptable assumptions and evaluation models for analyzing a rod ejection accident in PWRs are presented in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumptions) of this guide. By use of these appendices, it should be shown that:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod.

2. Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.[1]

3. Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

[1]Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, New York 10017.

APPENDIX A

PHYSICS AND THERMAL-HYDRAULICS

The assumptions described below should be applied in evaluating the physics and thermal-hydraulic behavior of the reactor system for a control rod ejection accident.

1. The ejected rod worth should be calculated based on the maximum worth rod resulting from the following conditions: (a) all control banks at positions corresponding to values for maximum allowable bank insertions at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed by operating procedures. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod in each rod group for different control rod configurations, both expected and unexpected. The worth of single rods in rod groups should be evaluated during startup physics tests and compared with values used in the rod ejection analysis. The accident should be reanalyzed if the rod worths used in the initial analysis are found to be nonconservative. Calculated rod worths should be increased, if necessary, to account for calculational uncertainties in parameters such as neutron cross sections and power asymmetries due to xenon oscillations.

2. The reactivity insertion rate due to an ejected rod should be determined from differential control rod worth curves and calculated transient rod position versus time curves. If differential rod worth curves are not available for the reactor state of interest, conservatism should be included in the calculation of reactivity insertion through consideration of the nonlinearity in reactivity addition as the rod passes through the active core. The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

3. The calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*) should be based on the well-known definitions resulting from perturbation theory, such as those described by Henry (Ref. 1), using available experimental delayed neutron data and averaging by the fraction of fission in the various fissionable materials. In cases where the accident is quite sensitive to β_{eff} (where the ejected rod worth $\gg \beta_{eff}$), the minimum calculated value for the given reactor state should be used. For smaller transients, conservatism in the value should include consideration of not only the initial power rise (which increases with decreasing β), but also the power reduction after the trip. Similar considerations should also be applied to determine an appropriately conservative value of ℓ^* to be used.

4. The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen with respect to their influence on the magnitude of the transient. Pressure and temperature are mainly significant with respect to their effect on the amount of reactivity inserted if there exists a positive moderator coefficient.

5. The fuel thermal properties such as fuel-clad gap heat transfer coefficient and fuel thermal conductivity should be conservatively chosen, depending upon the transient phenomenon being investigated. For conditions of a zero or positive moderator coefficient (usually at beginning of life), for example, high heat transfer parameters would reduce the Doppler feedback and increase any positive moderator feedback effects and hence tend to increase the magnitude of the reactivity transient. For a negative moderator coefficient, high heat transfer parameters could cause the magnitude of the transient to decrease if a given quantity of heat produces more feedback in the moderator than in the fuel. In the consideration of pressure pulses which may be generated, high moderator heating rates could cause significant pressure gradients to develop in the moderator channels. In computing the average enthalpy of the hottest fuel pellet during the excursion for power cases, low heat transfer would be conservative.

6. The specific heat of UO_2 has been determined experimentally and is a deterministic factor in the calculated amount of stored energy (enthalpy) in the fuel. Recommended values in the range of 25 to 902°C are the data reported by Moore and Kelly (Ref. 2). In the range of 900 to 2842°C, the data obtained by Hein and Flagella (Ref. 3), Leibowitz, Mishler, and Chasanov (Ref. 4), and Chasanov (Ref. 5) are recommended for the heat capacity of the fuel. These recommended values are for clean core conditions. Possible variation in the specific heat due to burnup should be investigated and appropriate values used, if necessary.

7. The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If no three-dimensional space-time kinetics calculation is performed, the reactivity feedback due to these coefficients should be conservatively weighted to account for the variation in their spatial importance in the missing dimension(s). If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed.

8. The Doppler coefficient should be calculated based on the effective resonance integrals and should include corrections for pin shadowing (Dancoff correction). Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively

with available experimental data such as those of Hellstrand (Ref. 6). Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied value of the Doppler coefficient. If no three-dimensional space-time kinetics calculation is performed, the reactivity effect of spatially weighting the core average temperature rise in both the axial and radial directions should be calculated.

9. Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. If the rod worth curve (reactivity vs. depth of insertion) is not obtained from a "true" representation (i.e., an x, y, z, t or an r, z, t calculation), the conservatism of the approximate calculation should be shown. The difference in the depth of insertion at zero power and at full power should be accounted for in calculating the available scram reactivity.

10. The reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (a) time required for instrument channel to produce a signal, (b) time for the trip breaker to open, (c) time for the coil to release the rods, and (d) time required before scram rods enter the core if the tips lie above the core-reflector interface.

11. The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (a) incorporation of all major reactivity feedback mechanisms, (b) at least six delayed neutron groups, (c) both axial and radial segmentation of the fuel element, (d) coolant flow provision, and (e) control rod scram initiation on either coolant system pressure or neutron flux.

12. The analytical models and computer codes used should be documented and justified and the conservatism of the models and codes should be evaluated both by comparison with experiment, as available, and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

13. The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

14. The number of fuel rods experiencing clad failure should be calculated and used to obtain the amount of contained fission product inventory released to the reactor coolant system. It should be assumed that clad failure occurs if the heat flux equals or exceeds the value corresponding to the onset of the transition from nucleate to film boiling (DNB), or for other appropriate causes.

The margin to DNB is expressed in terms of a departure from nucleate boiling ratio (DNBR). The DNBR at any position in the hottest channel is the ratio of the DNB heat flux to the actual heat flux. The DNB heat flux should be evaluated using correlations based on recognized studies and experimental heat transfer DNB data. A minimum DNBR should be determined from the evaluation of the experimental data to ensure a 95% probability with a 95% confidence level that DNB has not occurred for the fuel element being evaluated. One example of a correlation which has been used to date is given by Tong (Ref. 7). The use of this correlation and the above probabilities and confidence level yields a minimum DNBR of 1.30. Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data.

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6. E. Hellstrand et al., "The Temperature Coefficient of the Resonance Integral for Uranium Metal and Oxide," Nuclear Science and Engineering, Vol. 8, pp. 497-506, 1960.

7. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, pp. 241-248, 1967.

APPENDIX B

RADIOLOGICAL ASSUMPTIONS

The assumptions given below should be applied in determining a conservative source term and subsequent transport of activity and resulting doses to the public for use in evaluating the radiological consequences of a control rod ejection accident.

1. The assumptions related to the release of radioactive material to the primary containment are as follows:

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

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

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

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

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

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

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

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

i. Release of fission products to the secondary system should be computed by assuming that all fission products released from the fuel clad are uniformly mixed in the primary coolant volume.

j. The primary-to-secondary leak rate limitation incorporated or to be incorporated as a technical specification requirement should be assumed to exist until the primary system pressure falls below the secondary system pressure.

k. The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

2. Acceptable assumptions for atmospheric diffusion and dose conversion are:

a. The 0-to-8-hour ground-level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in Section 3-3.5.2 of Meteorology and Atomic Energy 1968 (Ref. 1), should be used only in the 0-to-8-hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor buiding only.

b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground or for the radiological decay of iodine in transit.

c. For the first 8 hours, the breathing rate of a person offsite should be assumed to be 3.47×10^{-4} m^3/sec. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} m^3/sec. From 24 hours until the end of the accident, the rate should be assumed to be 2.32×10^{-4} m^3/sec. (These values were developed from the average daily breathing rate [2×10^7 cm^3/day] assumed in a report (Ref. 2) of ICRP.[2])

d. The iodine dose conversion factors are also given in Reference 2.

e. External whole body doses should be calculated using "infinite cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for

[1] The effect on containment leakage under accident conditions of features provided to reduce the leakage of radioactive materials from the containment should be evaluated on a case-by-case basis.

[2] International Commission on Radiological Protection.

a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor." (Ref. 3) Editorial additions were made to the quotation so that gamma as well as beta emitting material could be considered. Under these conditions, the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose in air at the cloud center is:

$$_{\beta}D'_{\infty} = 0.457\,\overline{E}_{\beta}\chi$$

The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $_{\beta}D'_{\infty} = 0.23\,\overline{E}_{\beta}\chi$). For gamma emitting material, the dose rate in air at the cloud center is:

$$_{\gamma}D'_{\infty} = 0.507\,\overline{E}_{\gamma}\chi$$

From a semi-infinite cloud, the gamma dose rate in air is:

$$_{\gamma}D'_{\infty} = 0.25\,\overline{E}_{\gamma}\chi$$

where

$_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec)
$_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)
$\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis)
$\overline{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis)
χ = concentration of beta or gamma emitting isotope in the cloud (Ci/m^3)

f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:

(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance, taking into account special meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site-related characteristics must be evaluated on a case-by-case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground-level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.

(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes (Ref. 4), should be used.

g. The atmospheric diffusion model should be as follows:

(1) The basic equation for atmospheric diffusion from a ground-level point source is:

$$\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$$

where

χ = the short-term average centerline value of the ground-level concentration (Ci/m^3)
Q = amount of material released (Ci/sec)
u = windspeed (m/sec)
σ_y = the horizontal standard deviation of the plume (meters) [see Figure V-1, Ref. 5].
σ_z = the vertical standard deviation of the plume (meters) [see Figure V-2, Ref. 5].

(2) For time periods greater than 8 hours, the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is:

$$\chi/Q = \frac{2.032}{\sigma_z u x}$$

where

x = distance from the point of release to the receptor; other variables are as given in paragraph g. (1), above.

(3) The atmospheric diffusion model[3] for ground-level releases is based on the information in the table below.

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, wind speed 1 m/sec, uniform direction
8-24 hours	Pasquill Type F, wind speed 1 m/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, wind speed 3 m/sec (b) 60% Pasquill Type F, wind speed 2 m/sec (c) wind direction – variable within a 22.5° sector.

[3]This model should be used until adequate site meteorological data are obtained. In some cases, available information, such as meteorology, topography, and geographical location, may dictate the use of a more restrictive model to insure a conservative estimate of potential offsite exposures.

Time Following Accident	Atmospheric Conditions
4-30 days	(a) 33.3% Pasquill Type C, wind speed 3 m/sec
	(b) 33.3% Pasquill Type D, wind speed 3 m/sec

Time Following Accident	Atomospheric Conditions
4-30 days	(c) 33.3% Pasquill Type F, wind speed 2 m/sec
	(d) Wind direction – 33.3% frequency in a 22.5° sector.

(4) Figures 2(A) and 2(B) give the ground-level release atmospheric diffusion factors based on the parameters given in paragraph g.(3), above.

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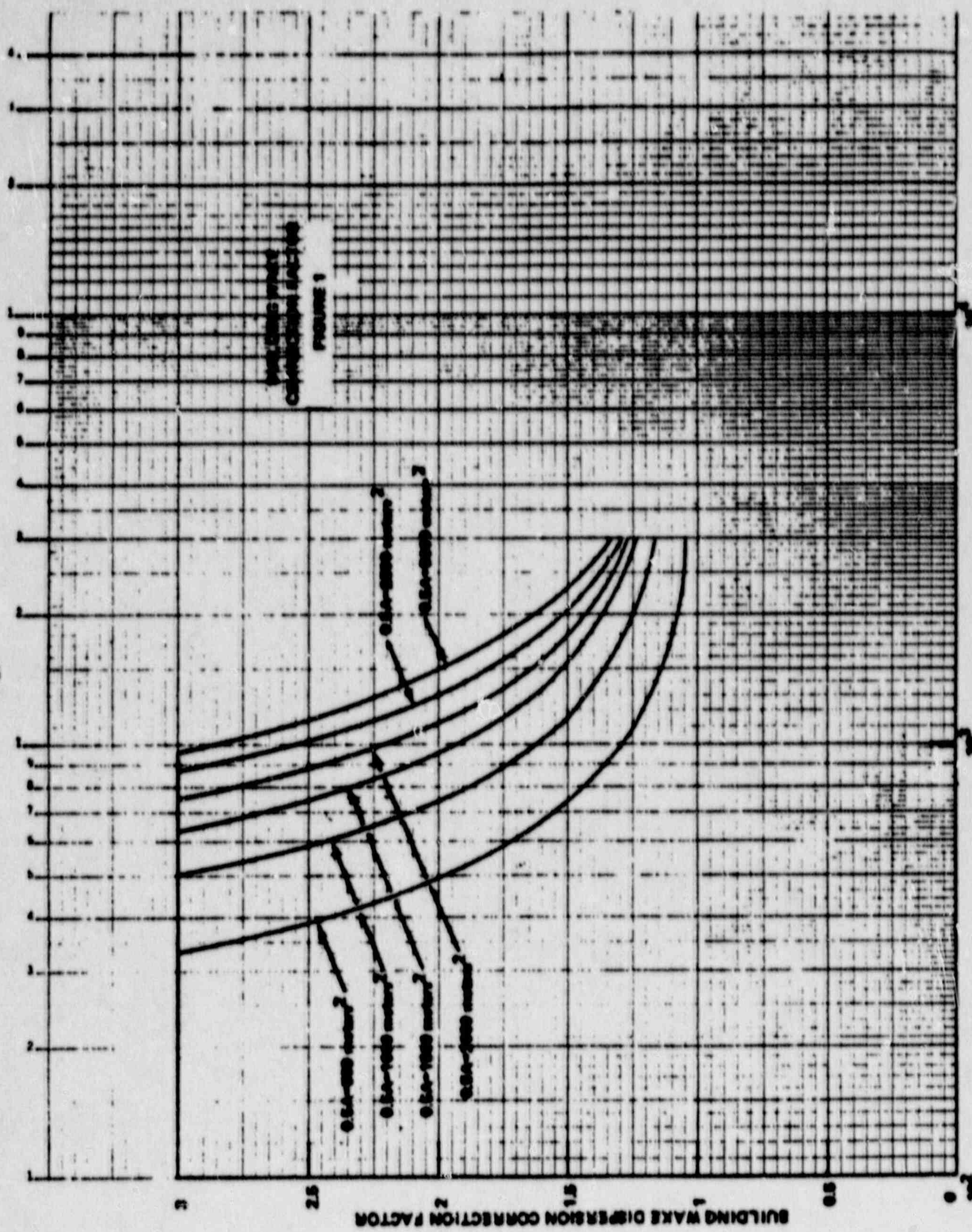
3. D. H. Slade, *op. cit.*, Section 7.4.1.1.

4. C. M. Lederer, J. M. Hollander, and I. Perlman, "Table of Isotopes," Sixth Edition, University of California, Berkeley, Lawrence Radiation Laboratory.

5. F. A. Gifford, Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety, Vol. 2, No. 4, June 1961.


BUILDING WAKE DISPERSION CORRECTION FACTOR

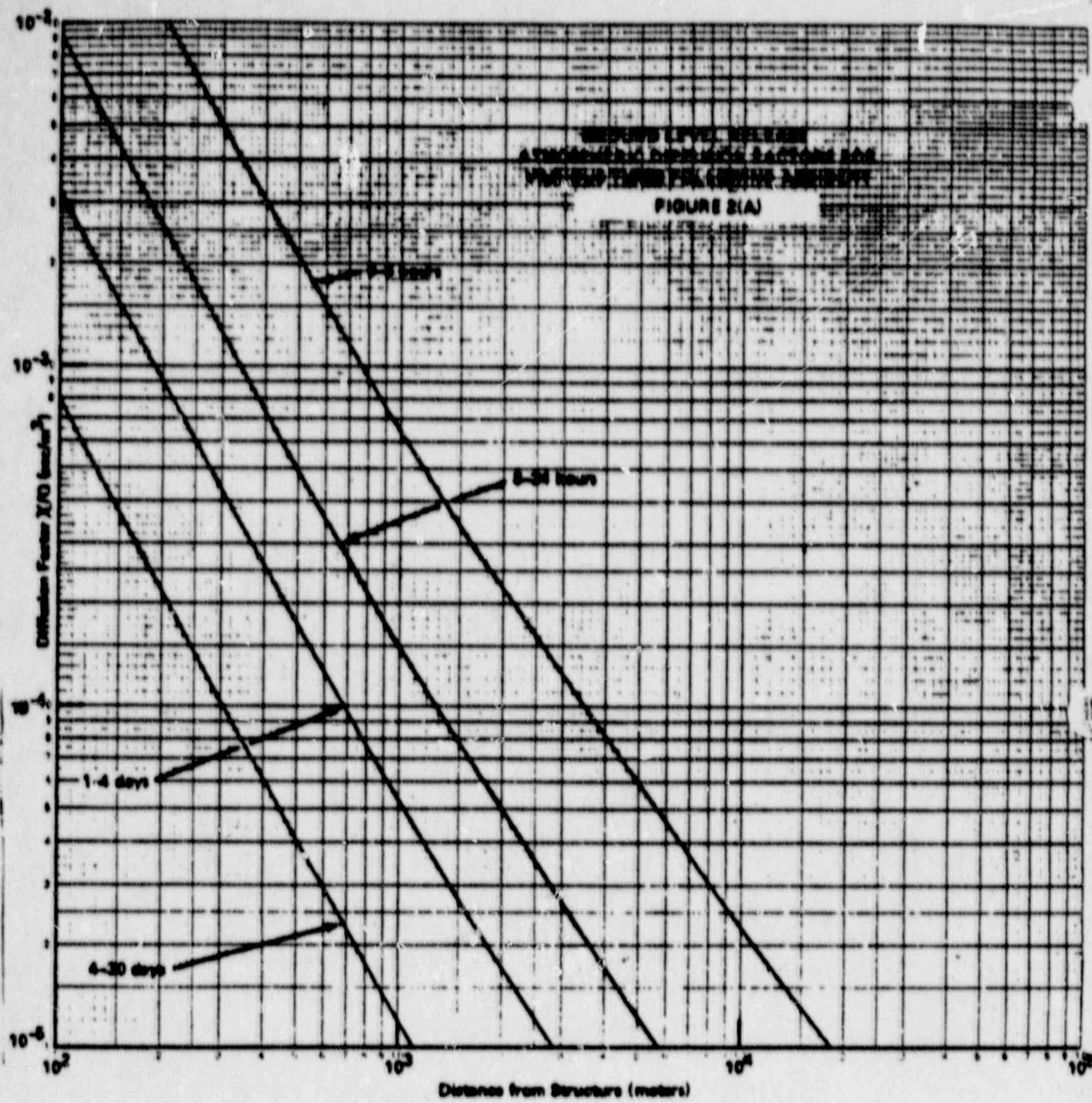
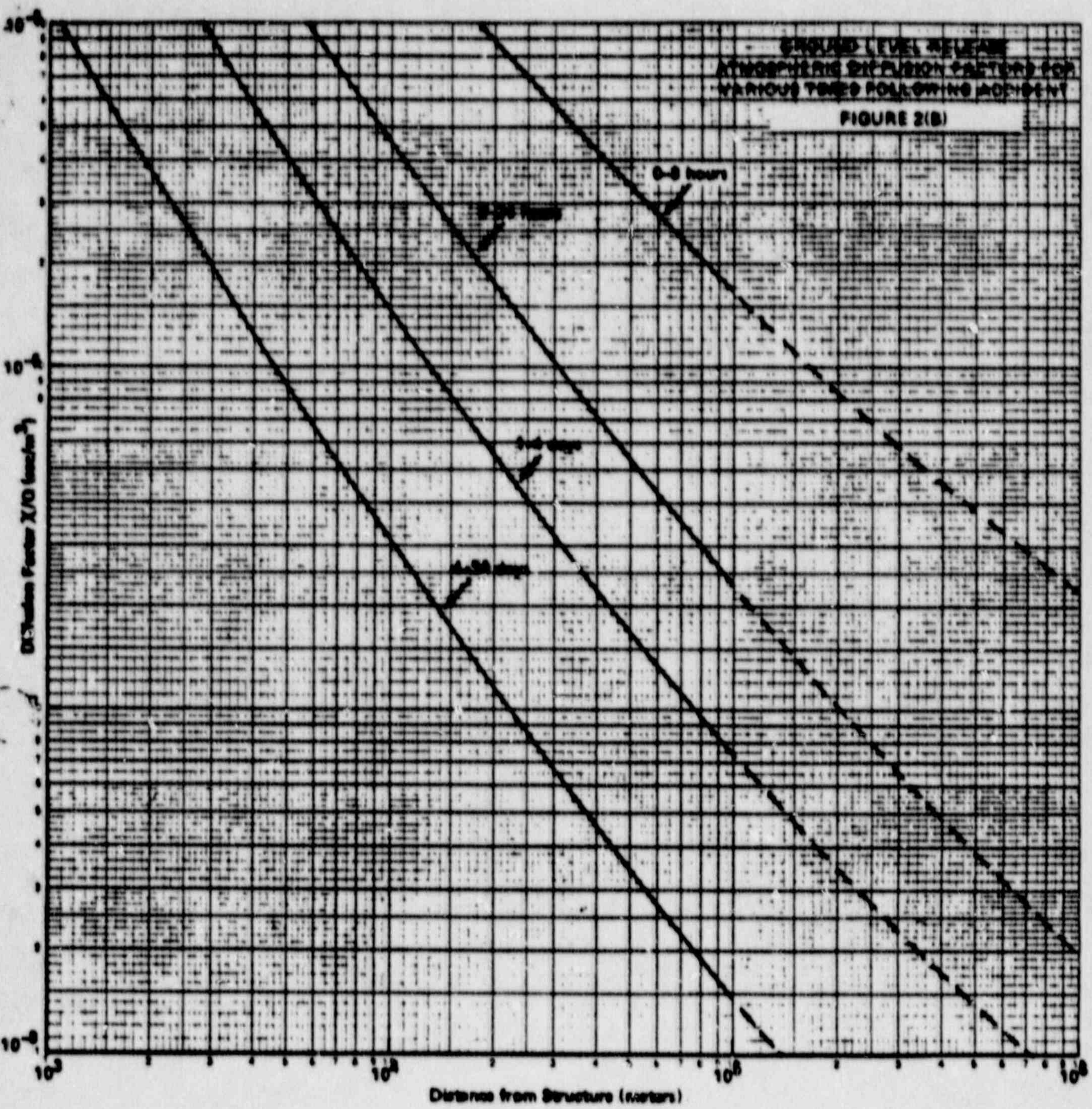


FIGURE 2(A)
8-24 hours
1-4 days
4-30 days
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{2}
10^{3}
10^{4}
10^{5}
Distance from Structure (meters)
Diffusion Factor χ/Q (sec/m³)


GROUND LEVEL RELEASE
ATMOSPHERIC DIFFUSION FACTORS FOR
VARIOUS TIMES FOLLOWING ACCIDENT
FIGURE 2(B)
0-8 hours
10⁻³
10⁻⁴
10⁻⁵
10³
10⁴
10⁵
10⁶
Distance from Structure (meters)

Reference 2

BACKGROUND INFORMATION RELATED TO

DIFFERING PROFESSIONAL VIEW CONCERNING

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

b) Methodology used for calculating related offsite doses.

ZION

CORE AND GAP ACTIVITIES

Assumptions: Operation at 3391 MWt for 500 days

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

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

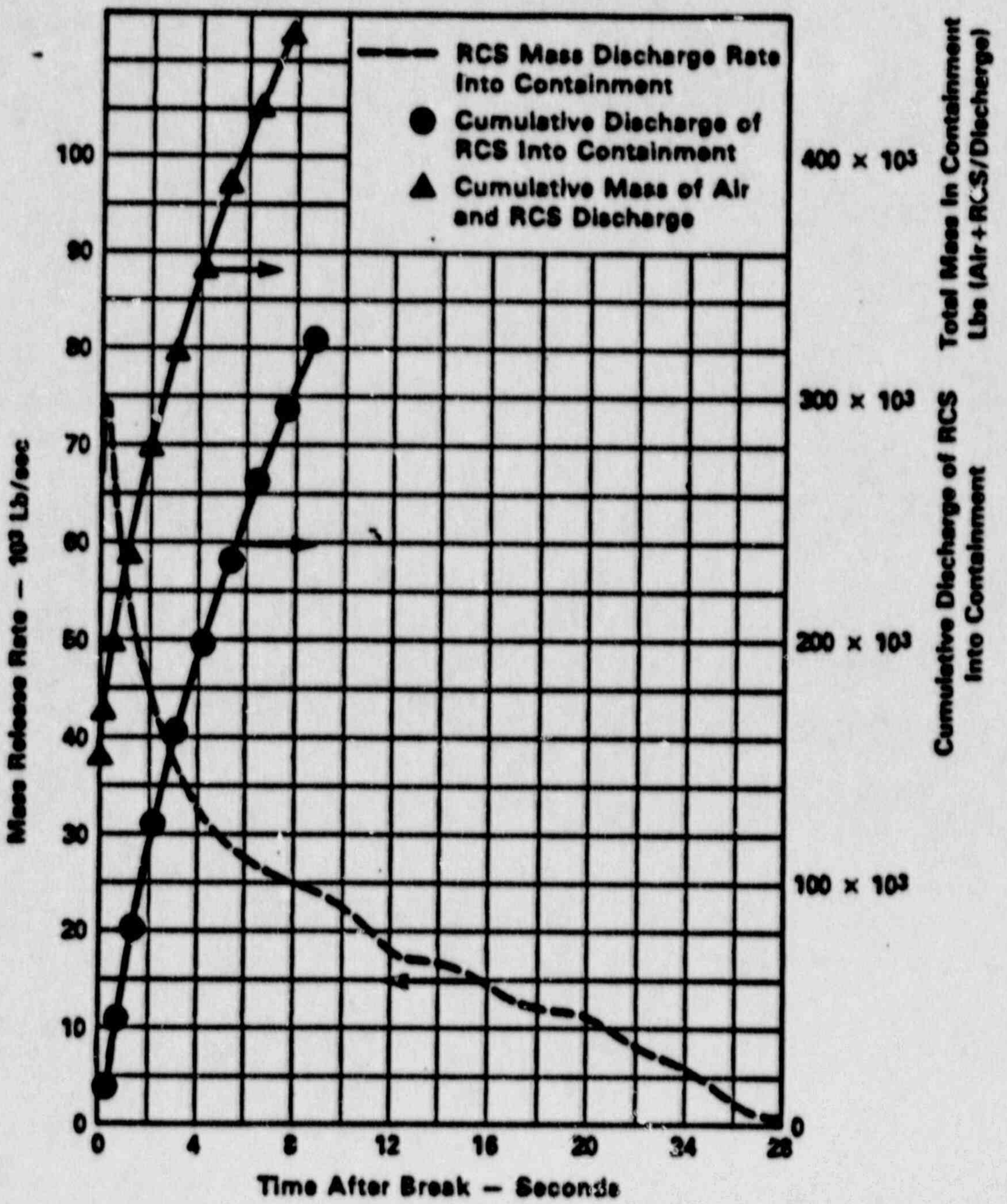
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE (LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Source	Radiological Sources	Curies Discharged I 131 EQ	Site Boundary Exposure (REM)
Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	73.5	18.7
RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119×10^5 curies]	188	48
RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	82	21
RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	38,000	9676
RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9×10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	248,950	63,400

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

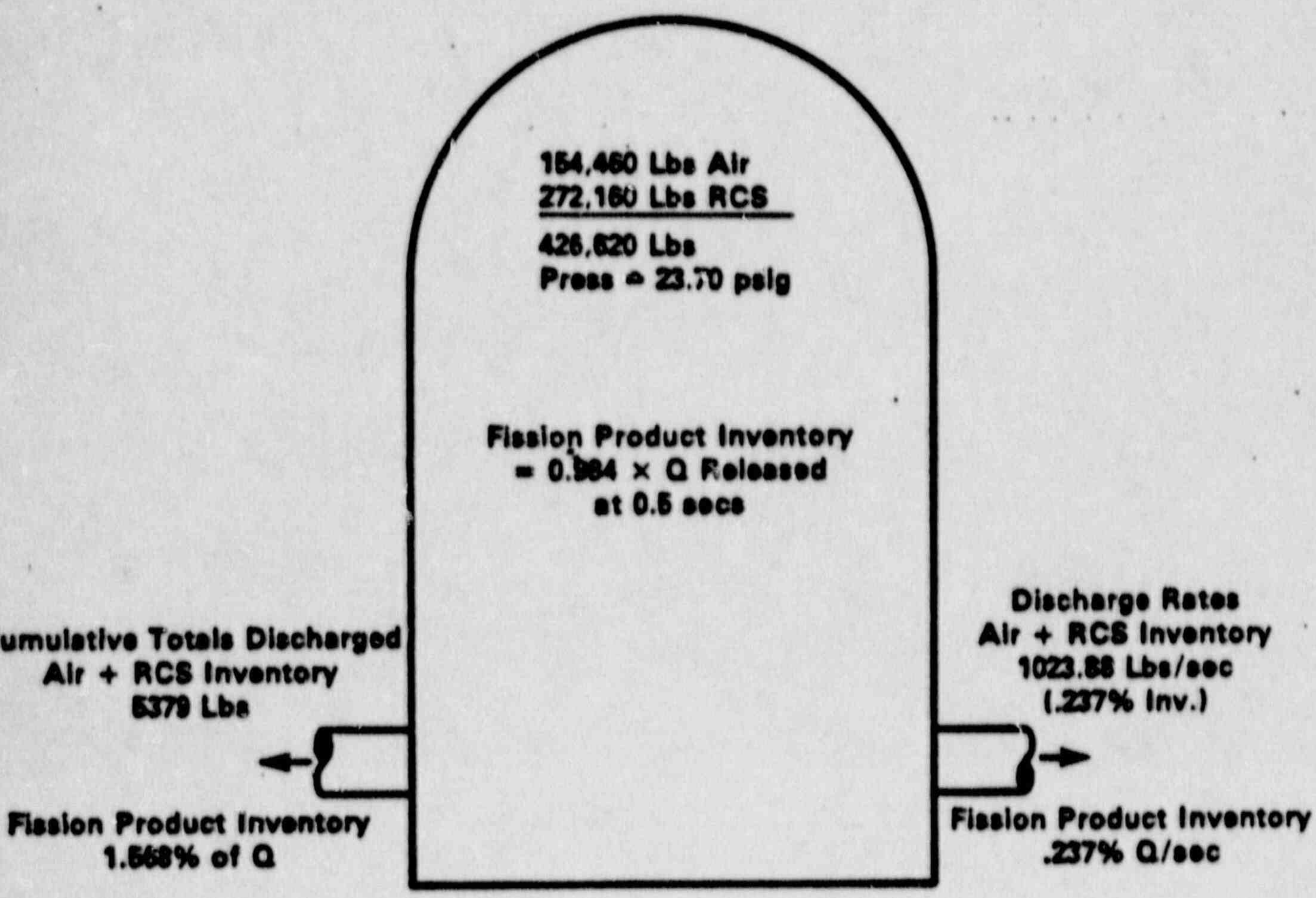
[Licensee has used 9×10^{-4} sec/m³ for SARs]

ZION 1 & 2
CONTAINMENT INVENTORIES DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES

2 × 42″ Lines
Valves Open Only 50°
Instead of 90° Fully Open
At 7 Secs



(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°)

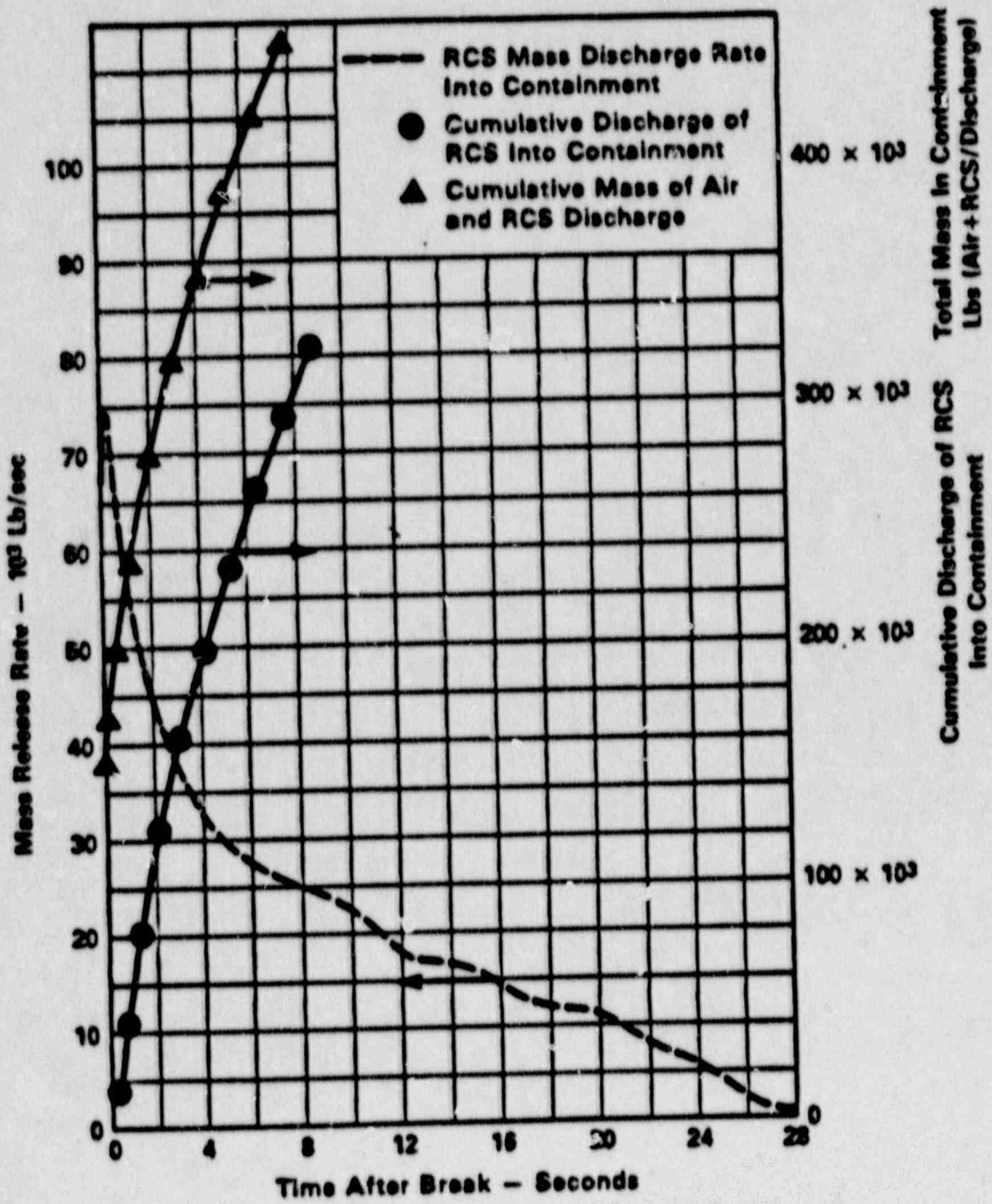
THYROID DOSE AT SITE BOUNDARY RESULTING ONLY FROM
DISCHARGE TO CONTAINMENT OUTSIDE DURING CLOSURE
(LOCA LEAKAGE DOSE (OVER 2 HRS) = +123 REMS)

Source	Radiological Sources	Curies Discharged I 131 EQ	Site Boundary Exposure (REM)
Licensee	I 131 EQ. 60 uc/gm in RCS 50% cleanup in cont. All released to containment on LOCA	204.3	52
RL	I 131 EQ, 60 uc/gm in RCS. All released to cont. on LOCA + 0.5 secs. [Total = 0.119 x 10^5 curies]	522	132
RL	I 131 EQ; 60 uc/gm in RCS. Released progressively to cont. with RCS discharge	227	58
RL	I 131 EQ; equiv gap activity (FSAR calc.) [24.09 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	105,600	26,878
RL	I 131 EQ; SRP Gap activity at 10% Total Activity (FSAR calc.) [157.9 x 10^5 curies of I 131 EQ into cont. on LOCA + 0.5 secs.]	691,520	176,010

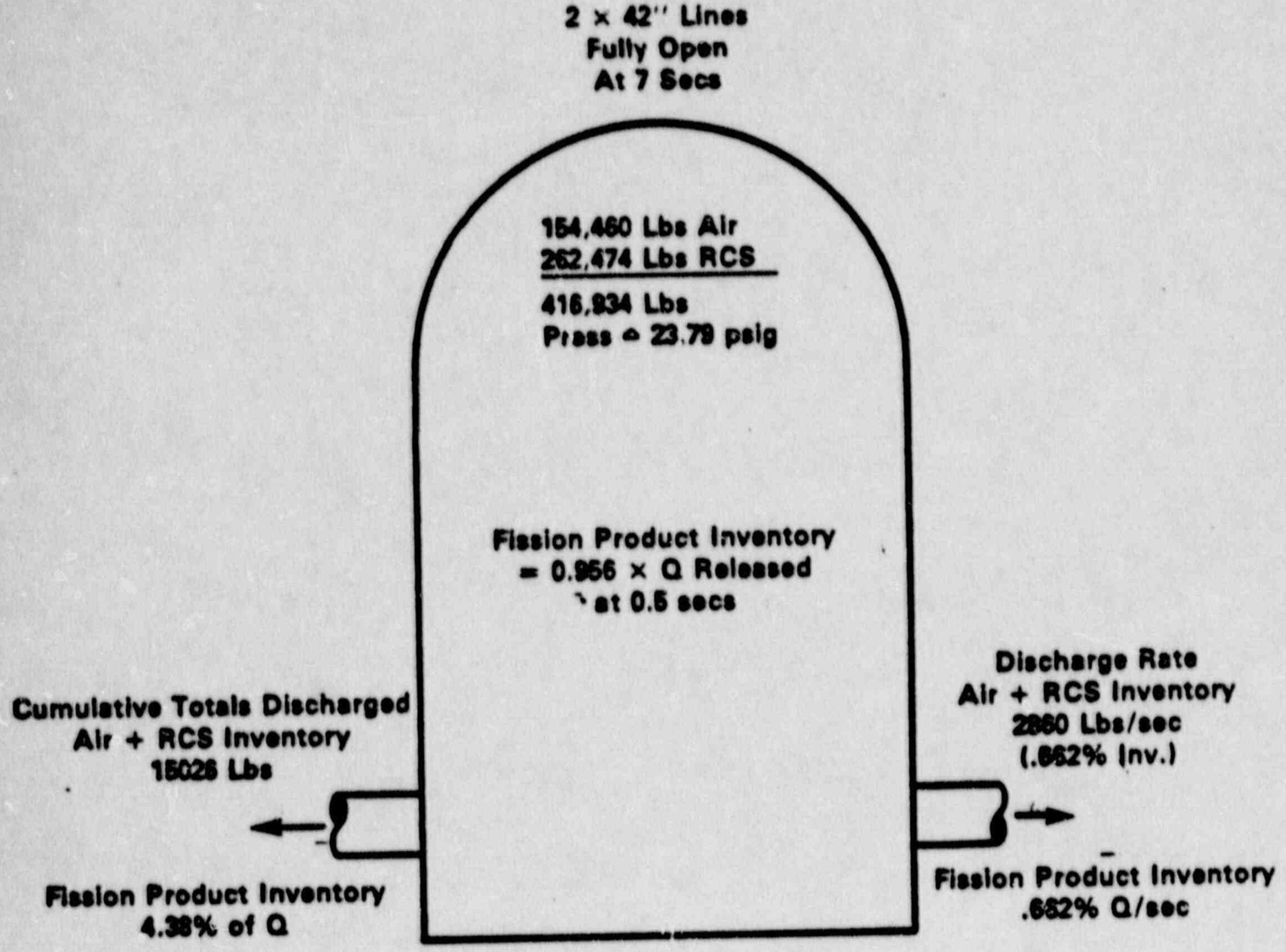
[NRC] $\frac{X}{Q} = 5 \times 10^{-4}$ sec/m³ for 0-2 hrs. at minimum exclusion distance of 415 meters

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

ZION 1 & 2
CONTAINMENT INVENTORIES DURING LOCA BLOW DOWN



ZION 1 & 2
CONTAINMENT THERMAL HYDRAULIC CONDITIONS
FISSION PRODUCT INVENTORIES



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

FISSION PRODUCT DISCHARGED TO OUTSIDE CONTAINMENT

EFFECT OF ASSUMPTIONS ON FISSION PRODUCT RELEASE TO CONTAINMENT

2 x 42" lines
fully open (90°).

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

- If Q is released instantaneously to the total containment volume

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

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

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

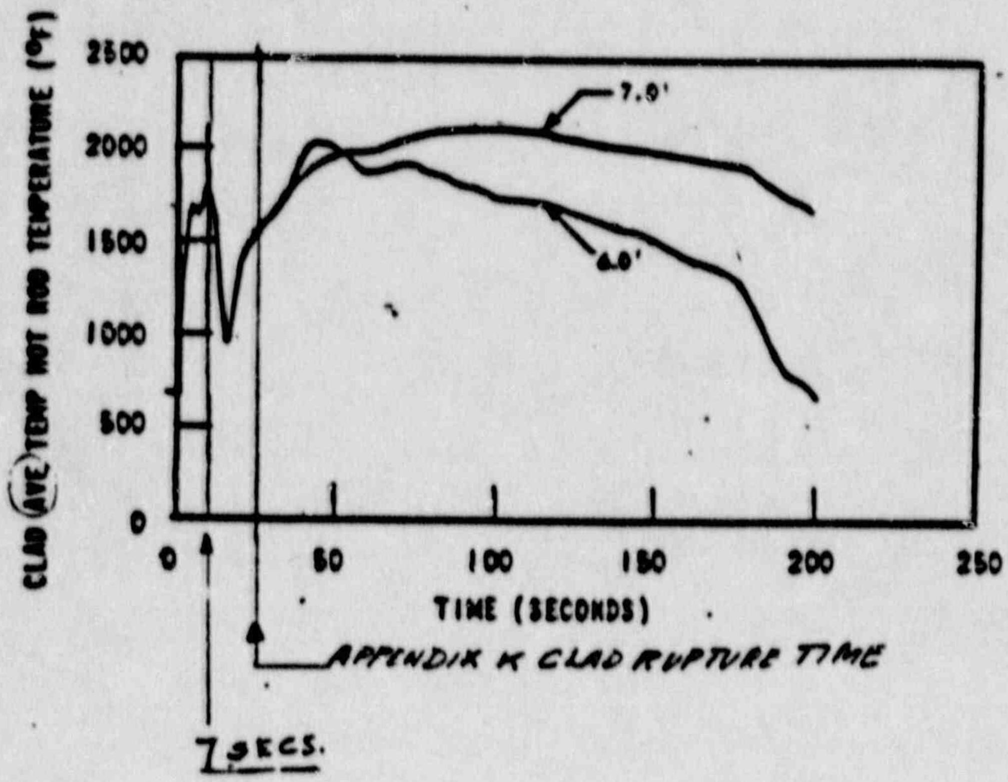


Figure 14 F.2-19a Peak Clad Temperature - DECLG (C_D=1.0) (Unit 1)

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. Simillar handling of 22 removable rods was sucessfully completed during the first Zorita refueling. All such fuel handlings have been done routinely and without difficulty.

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

3.2.3.5 Evaluation of Core Components

Fuel Evaluation

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

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

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

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

3. Burnup:

Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is not expected to be reached until the peak burnup reaches approximately 65,000 MWD/MTU. The peak pellet burnup for fuel in equilibrium cycling is expected to be 50,000 MWD/MTU. The design equilibrium region average discharge burnup is about 33,000 MWD/MTU.

4. Fuel temperature and kw/ft:

At zero burnup, cladding damage is calculated to occur at 31 kw/ft [illegible] upon cladding strain reaching the damage limit. At this power [illegible] % of the pellet central region is expected to be in the molten [illegible]. The maximum thermal output at rated power is 15.0 kw/ft.

An evaluation of the fuel densification as it affects operating limits for Unit 1 is given in Appendix 3A, and WCAP 8063 Addendum 2 ([illegible]). WCAP 8122 and its addendum (references 25 and 26) evaluates fuel densification as it applies to Unit 2.

In Appendix 3A, the initial fuel densities of Regions 2 and 3 reflect the actual region densities as amended in the FSAR Table 4.1. The fuel density itself does not have any significant effect on the power transient associated with the rod ejection accident.

In the fuel densification report a 2700°F clad surface temperature limits is used for accidents such as Rod Ejection and Locked Rotor as discussed in WCAP 7855 and in Attachment 13 of Westinghouse letter NS-SL-543 (January 12, 1973) to Dr. D. F. Knuth.

In Appendix 3A, the methods described in WCAP-7422-L "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident" Section 6.2.3 were used to determine the gap conductance during LOCA except for the initial value. The initial gap conductance in LOCTA was adjusted such that the initial average temperature in LOCTA was equal to the design value of the BOL average temperature at the appropriate Kw/ft plus an additional temperature increase to cover an uncertainty which was equal to 9.375 x (Kw/ft).

Parameters considered important to fuel densification analysis are summarized in Table 3.1 and Table 4.1 of the Zion and Point Beach Unit 2 fuel densification reports. In addition:

a. The initial, as fabricated diametral fuel pellet/clad gap for Zion is 0.0075 inches which is 0.001 inches less than that for Point Beach Unit 2.

b. The time integrated axial power distribution (or fast neutron flux distribution) used in the analyses of gap conductance for both Zion and Point Beach Unit 2 is given in Attachment L, Figure 2 of Westinghouse letter NS-SL-521 (January 4, 1973) to Dr. D. F. Knuth.

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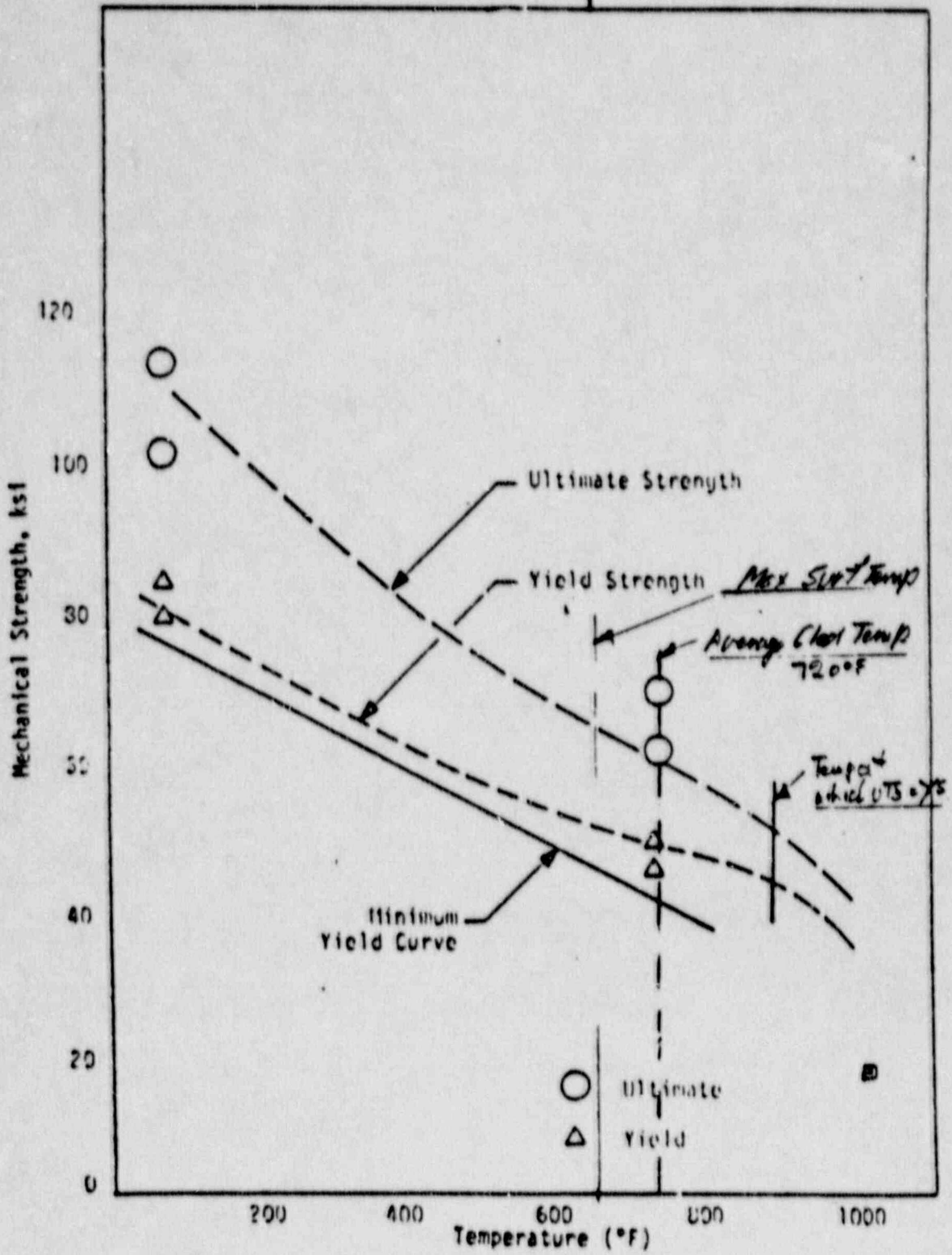


FIGURE ___ MECHANICAL STRENGTH OF ROD TUBING VERSUS TEMPERATURE

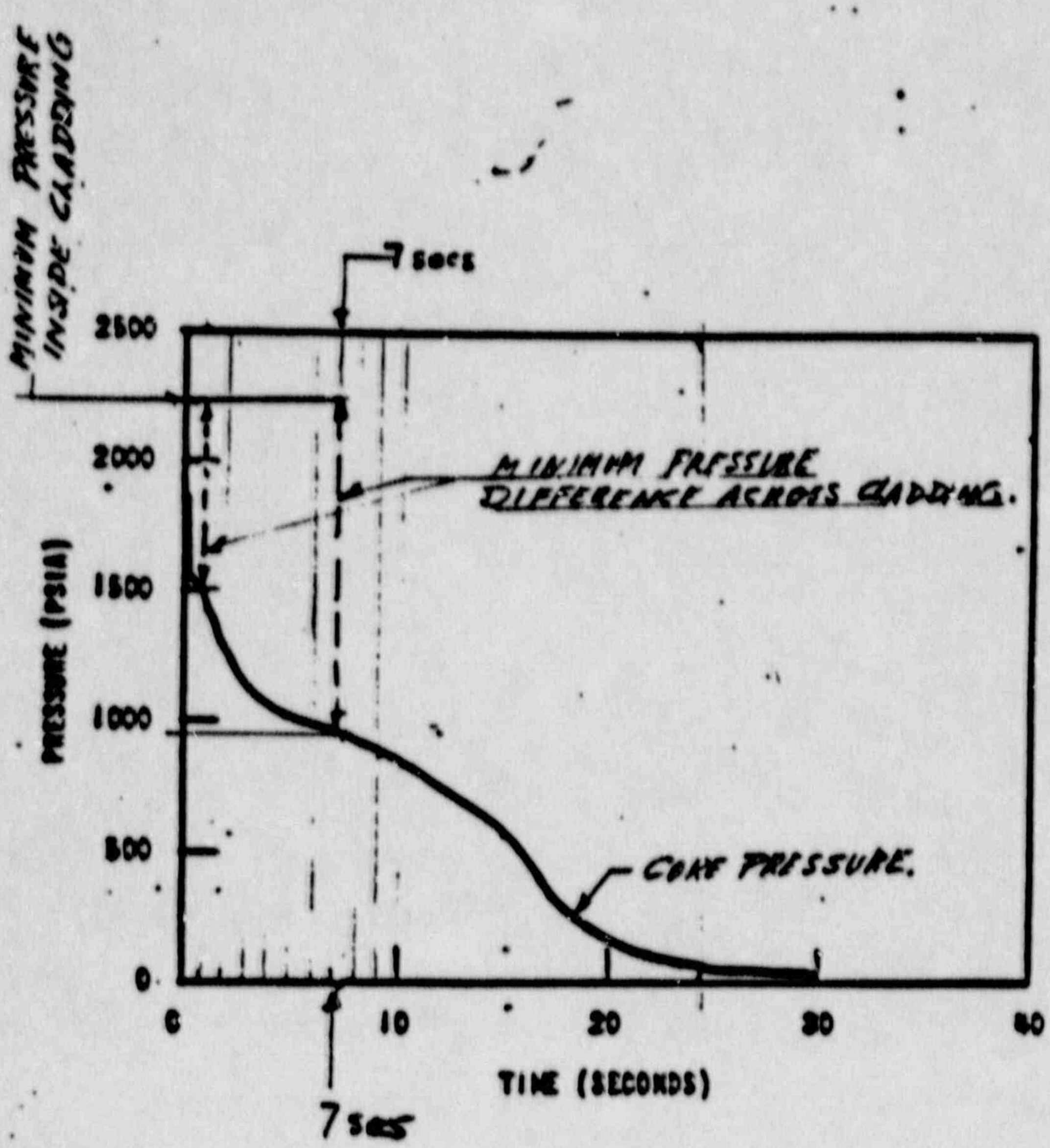


Figure 14 F.2-10a Core Pressure - DECLO (C_D=1.0)
(Unit 1)

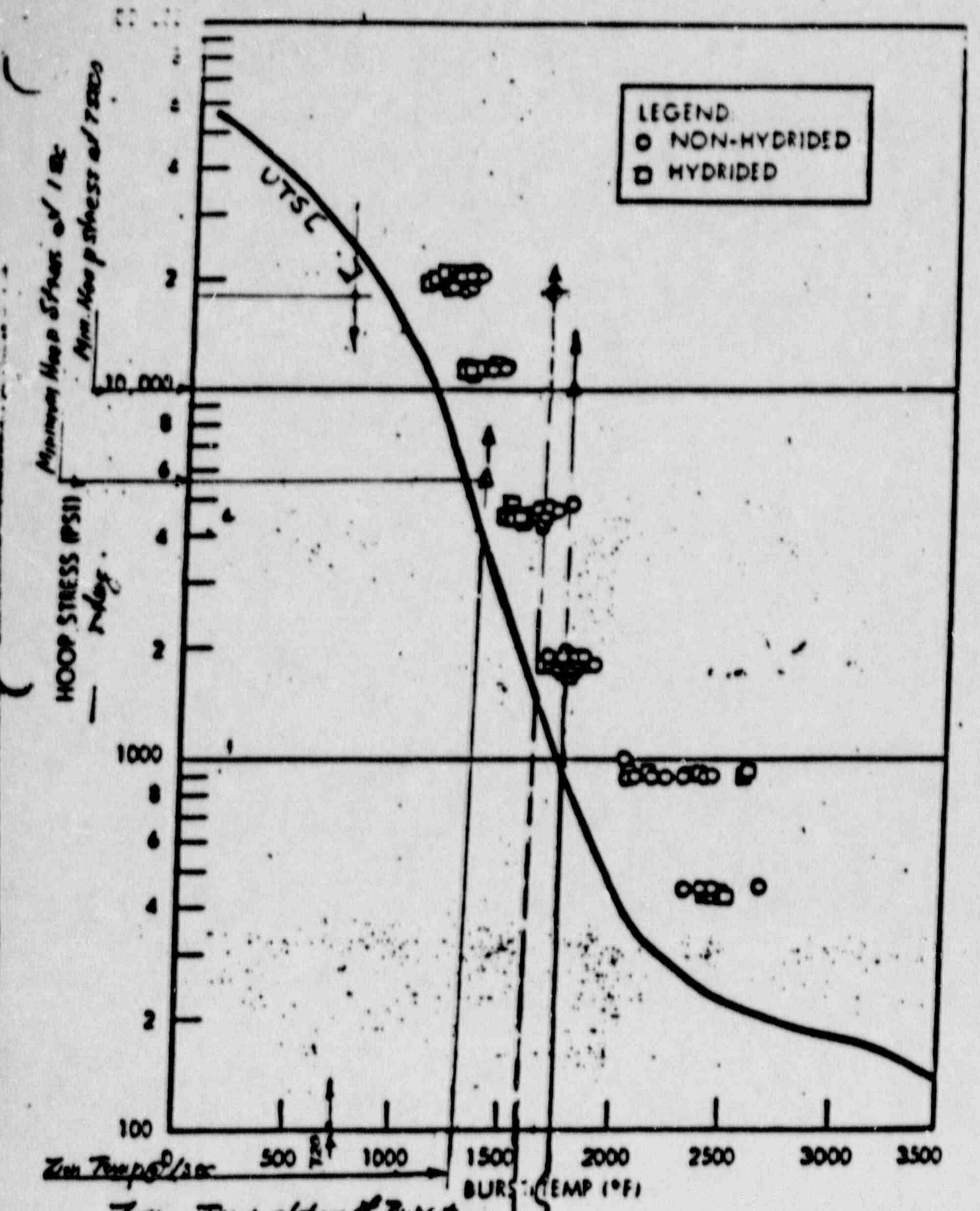
TABLE 1

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

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

* D. C. Cook, Unit 1

** Oyster Creek



Burst Temperature versus Stress at Burst

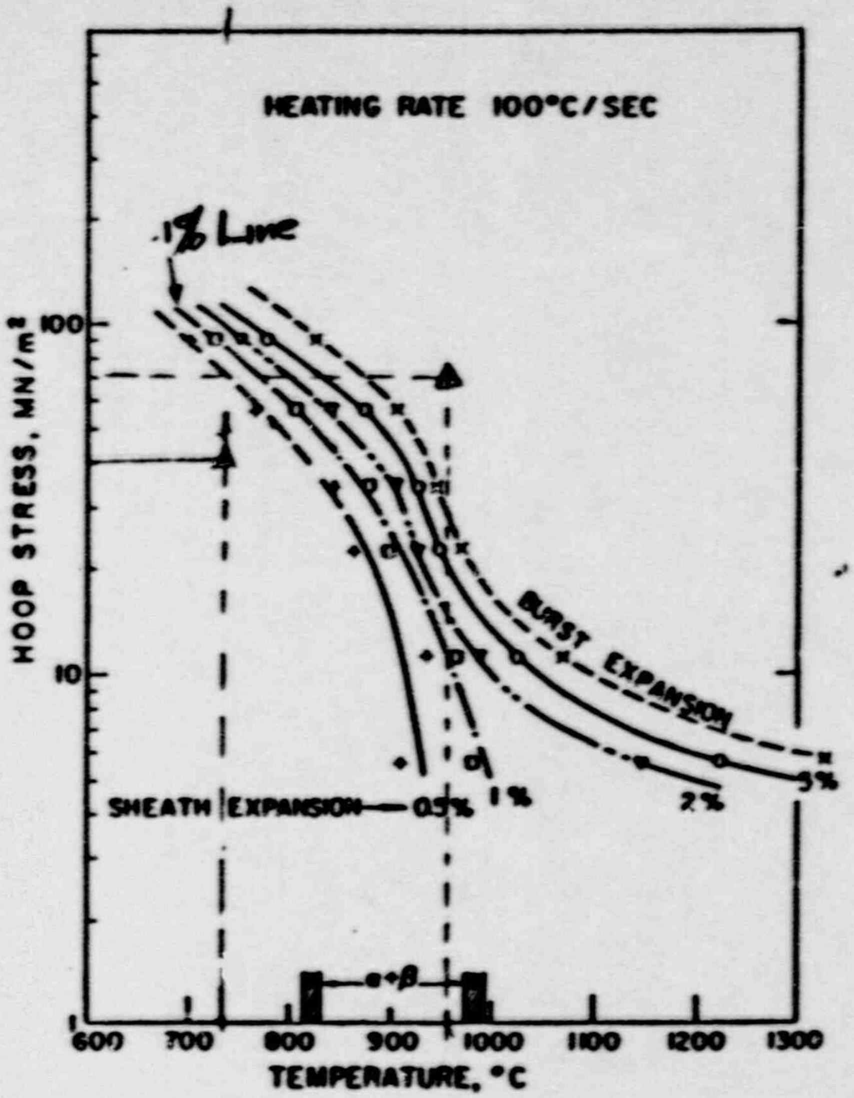


FIGURE 10 (HARDY)

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

263

1 MN/m² = 142.9 psi

1750°F = 954°C 1350°F = 732°C
10,000 psi = [illegible] MN/m² 5800 psi = 40 [illegible] MN/m²

15

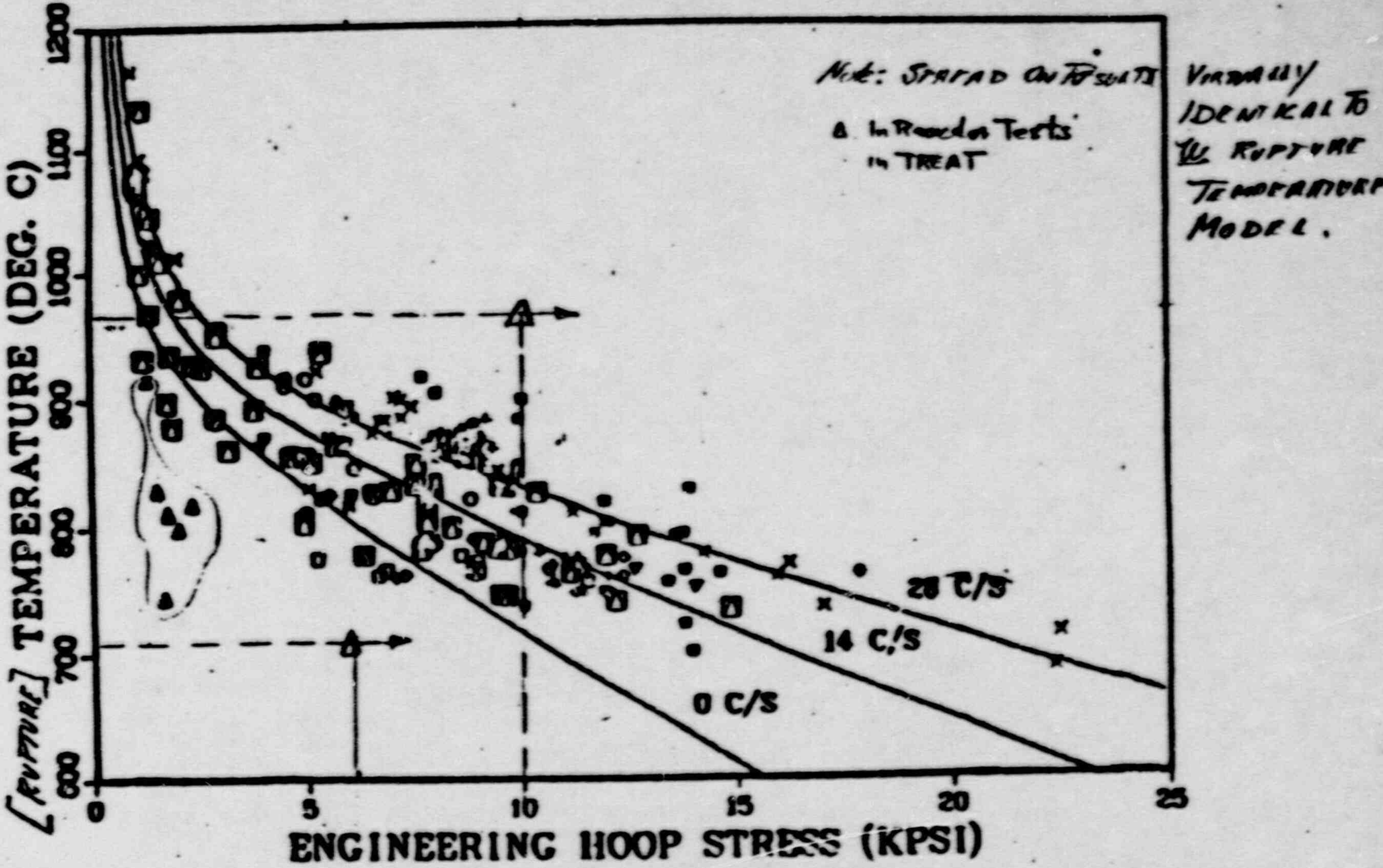


Fig. 3 ORNL correlation of rupture temperature as a function of engineering hoop stress and temperature-ramp rate with data from internally heated Zircaloy cladding in aqueous atmospheres.

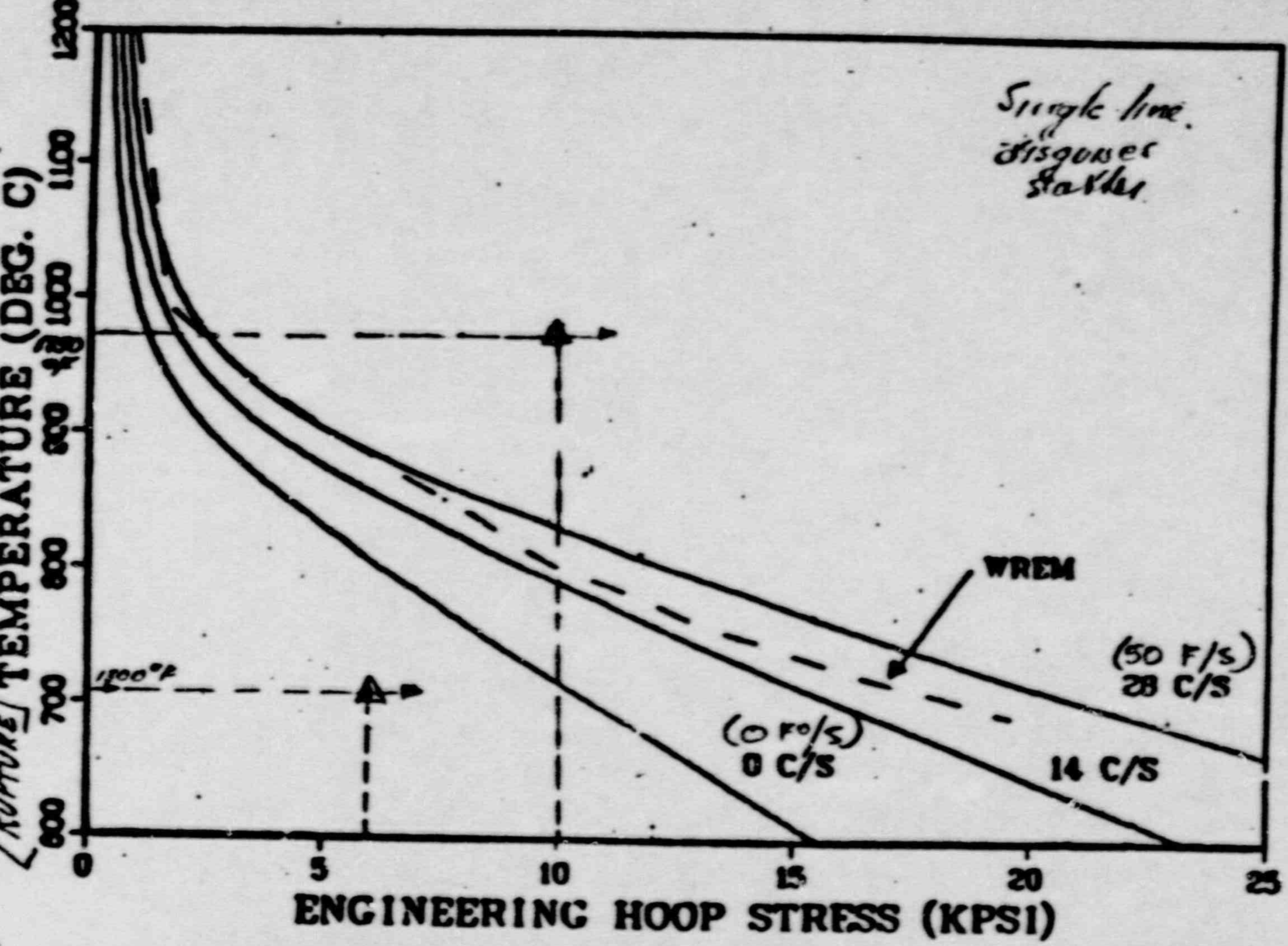


Fig. 17 WREM model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

1974 DATA 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 bas[illegible] the swelling (a,c)
model described above or if the clad temperature on the burst node reaches
the burst temperature. Fig 3-1 ⌈Burst temperature is calculated as a function of
hoop stress based on correlation of the Westinghouse single rod burst test [27]
data shown in Figure 3-1. The best estimate curve from figure 3-1 is used and
pressure is converted to hoop stress by the relationship described in
Equation 3-69 using original test specimen geometry. This best estimate
curve is described by the equation

(a,b,c)

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

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)

$$\frac{\Delta d^{*}}{d_o} \Big[\qquad \Big] \qquad (3\text{-}71B)$$

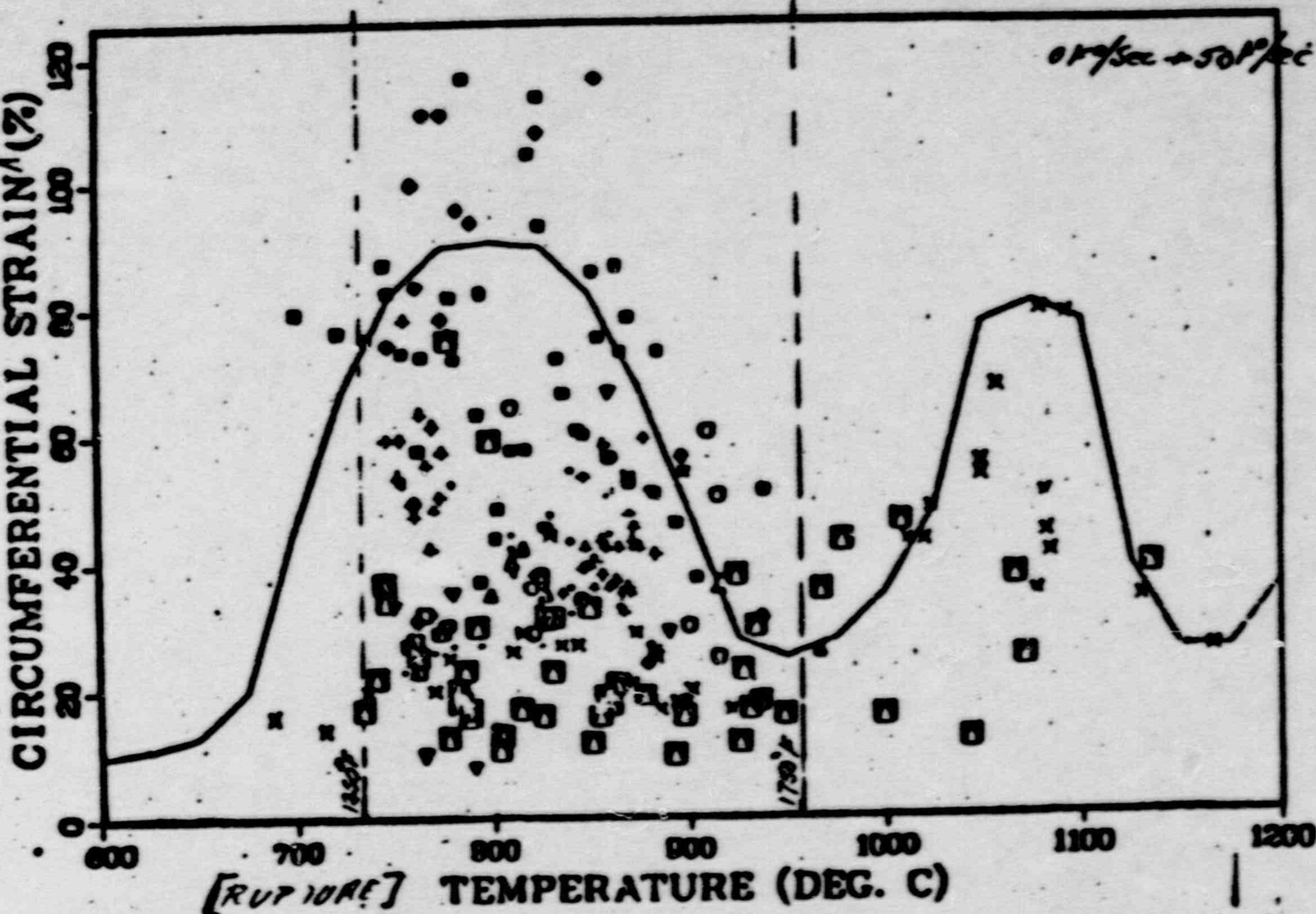


Fig. 9 ~~Maximum circumferential strain as~~ a function of ~~rupture temperature~~ for internally heated Zircaloy cladding in aqueous atmospheres for all heating rates.

NUREG-75/077

THE ROLE OF FISSION GAS RELEASE IN REACTOR LICENSING

CORE PERFORMANCE BRANCH

U. S. NUCLEAR REGULATORY COMMISSION

NOVEMBER 1975

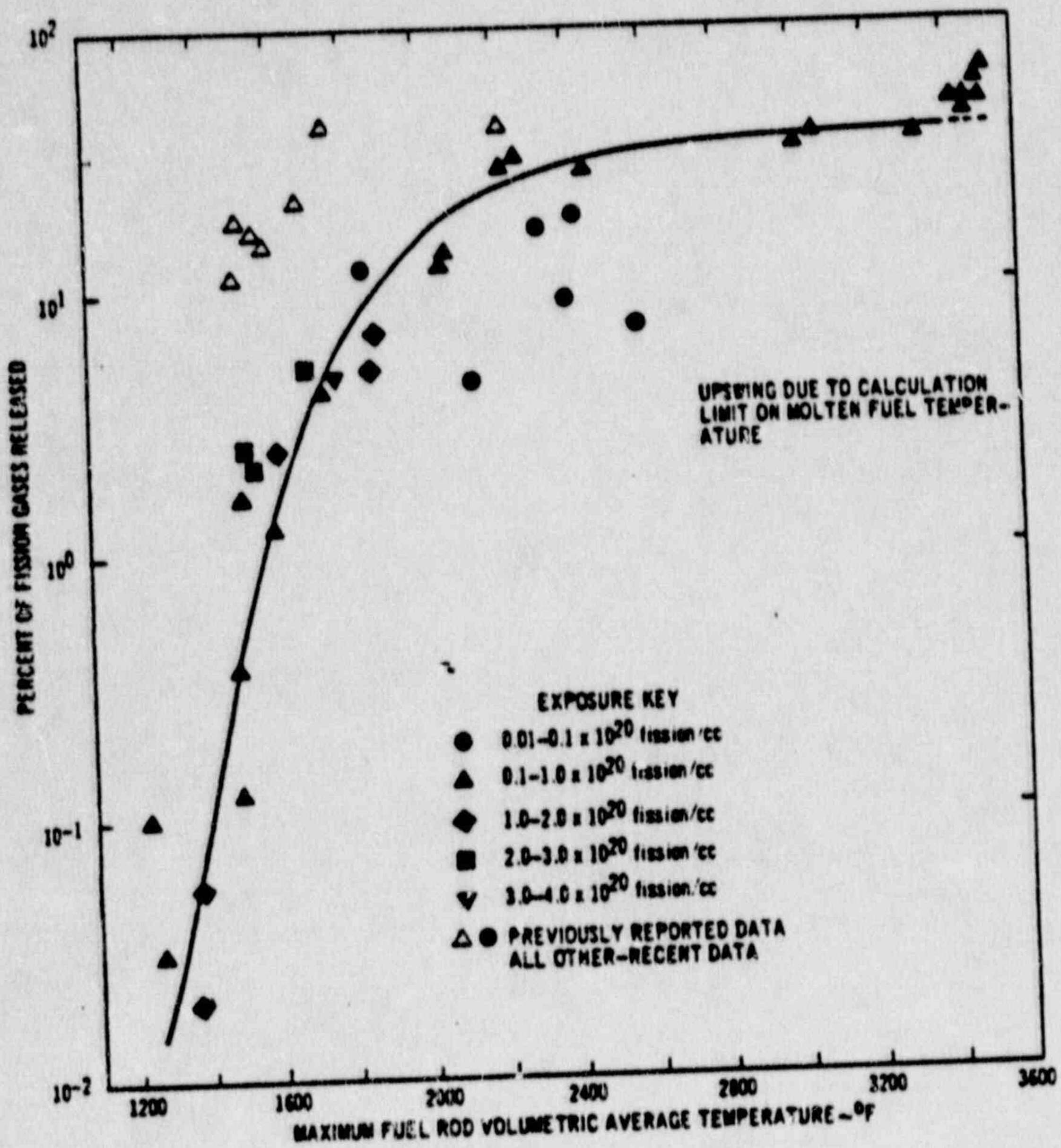


Fig. 2. The Hoffman & Coplin correlation of fission gas release as a function of temperature (from Ref. 35).

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

Ref. 2

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the lines associated with the containment purge and vent systems. It supplements the position taken in SRP Section 6.2.4.

While the containment purge and vent systems provide plant operational flexibility, their designs must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on their use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously. There is similar disparity in the need for, and use of, containment vent systems at operating plants.

Containment purge systems have been used in a variety of ways; for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. Containment vent systems are typically used to relieve the initial containment pressure buildup caused by the heat load imposed on the containment atmosphere during reactor power ascension, or to periodically relieve the pressure buildup due to the operation of pneumatic controllers. However, the purge and vent lines provide an open path from the contaiment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR Part 100 guidelines values.

The sizing of the purge lines in most plants have been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore, the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations and they must be sealed closed in all other operational modes.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the contaiment backpressure.

Purge system designs that are acceptable for use on a nonroutine basis during normal plant operation can be achieved by providing additional purge lines.

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

The size of the purge lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR Part 100 guideline values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR, or HTGR.

B. BRANCH TECHNICAL POSITION

The systems used to purge the containment for the reactor operational modes of power operation, startup, hot standby and hot shutdown; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of cold shutdown and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:

   a. General Design Criterion 54 requires that the reliability and performance capabilities of containment isolation valves reflect the importance of safety of isolating the systems penetrating the containment boundary. Therefore, the performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in Branch Technical Position MEB-2, "Pump and Valve Operability Assurance Program." (Also see SRP Section 3.10.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the supply line and exhaust line flows as a function of time up to and during valve closure.

   b. The number of supply and exhaust lines that may be used should be limited to one supply line and one exhaust line, to improve the reliability of the isolation function as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in the evaluation of the emergency core cooling system effectiveness and 10 CFR Part 100 regarding offsite radiological consequences.

   c. The size of the lines should not exceed about eight inches in diameter, unless detailed justification for larger line sizes is provided, to improve the reliability and performance capability of the isolation and containment functions as required by General Design Criterion 54, and to facilitate compliance with the requirements of Appendix K to 10 CFR Part 50 regarding the containment pressure used in evaluating the emergency core cooling system effectiveness and 10 CFR Part 100 regarding the offsite radiological consequences.

d. As required by General Design Criterion 54, the containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria, to reflect the importance to safety of isolating these lines. General Design Criterion 56 establishes explicit requirements for isolation barriers in purge system lines.

e. To improve the reliability of the isolation function, which is addressed in General Design Criterion 54, instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. Furthermore, if energy is required to close the valves, at least two diverse sources of energy shall be provided, either of which can effect the isolation function.

[On-Line]

f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds, to facilitate compliance with 10 CFR Part 100 regarding offsite radiological consequences.

g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves during reactor operation.

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

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

b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters, and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.

c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The maximum allowable leak rate of the purge isolation valves should be specified on a case-by-case basis giving appropriate consideration to valve size, maximum allowable leakage rate for the containment (as defined in Appendix J to 10 CFR Part 50), and where appropriate, the maximum allowable bypass leakage fraction for dual containments.

NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. The fuel system consists of arrays (assemblies or bundles) of fuel rods including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 (Ref. 1), and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 (Ref. 2) for postulated accidents. "Coolability," in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, §50.46 (Ref. 3).

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

The fuel system review covers the following specific areas.

A. Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that damage will be limited to acceptable levels. The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. Design Evaluation

The performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents is reviewed to determine if all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. Testing, Inspection, and Surveillance Plans

Testing and inspection of new fuel is performed by the licensee to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. On-line fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by CPB to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

Specific criteria necessary to meet the requirements of 10 CFR Part 50, §50.46; General Design Criteria 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 identified in subsection I of this SRP section are as follows:

A. <u>Design Bases</u>

The fuel system design bases must reflect the four objectives described in subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following:

1. <u>Fuel System Damage</u> -

This subsection applies to normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Such damage criteria should address the following to be complete.

(a) Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the ASME Code (Ref. 4) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles (Ref. 5). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the Safety Analysis Report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) Worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

2. Fuel Rod Failure

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

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

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

(a) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 7) for UO_2 fuel pellets state an equivalent limit of 2 ppm of hydrogen from all sources. For other materials clad in

Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown (Ref. 8) to be insufficient for primary hydride formation.

(b) Cladding Collapse: If axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an abnormal operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during transients.

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

N. [illegible] from [illegible]

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

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

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

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) Fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

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

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

3. Fuel Coolability

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addresses the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. Control rod insertability criteria are also addressed in this subsection. Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, §50.46, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 1800°F previously described in Reference 11 is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of Appendix K of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data (such as Refs. 10, 12, and 13) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 2200°F cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

B. <u>Description and Design Drawings</u>

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

Type and metallurgical state of the cladding
Cladding outside diameter
Cladding inside diameter
Cladding inside roughness
Pellet outside diameter
Pellet roughness
Pellet density
Pellet resintering data
Pellet length
Pellet dish dimensions
Burnable poison content
Insulator pellet parameters
Fuel column length
Overall rod length
Rod internal void volume
Fill gas type and pressure
Sorbed gas composition and content
Spring and plug dimensions
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawing have also been found necessary for an acceptable fuel system description:

Fuel assembly cross section
Fuel assembly outline
Fuel rod schematic
Spacer grid cross section
Guide tube and nozzle joint
Control rod assembly cross section
Control rod assembly outline
Control rod schematic
Burnable poison rod assembly cross section
Burnable poison rod assembly outline
Burnable poison rod schematic
Orifice and source assembly outline

C. Design Evaluation

The methods of demonstrating that the design bases are met must be reviewed. Those methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the Safety Analysis Report by reference.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed when practical to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested prior to irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

Spacer grid structural tests
Control rod structural and performance tests
Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
Fuel assembly growth
Fuel assembly bowing
Channel box wear and distortion
Fuel rod ridging (PCI)
Crud formation
Fuel rod integrity
Holddown spring relaxation
Spacer grid spring relaxation
Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished prior to operation of a full core of that design. This inability to perform in-reactor testing may result from an incompatability of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see subsection II.D below).

3. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Phenomenological models that should be reviewed include the following:

Radial power distribution
Fuel and cladding temperature distribution
Burnup distribution in the fuel
Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
Densification of the fuel
Thermal expansion of the fuel and cladding
Fission gas production and release
Solid and gaseous fission product swelling
Fuel restructuring and relocation
Fuel and cladding dimensional changes
Fuel-to-cladding heat transfer coefficient
Thermal conductivity of the gas mixture
Thermal conductivity in the Knudsen domain
Fuel-to-cladding contact pressure
Heat capacity of the fuel and cladding
Growth and creep of the cladding
Rod internal gas pressure and composition
Sorption of helium and other fill gases
Cladding oxide and crud layer thickness
Cladding-to-coolant heat transfer coefficient*

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (Refs. 14 and 15). Examples of previous fuel performance code reviews are given in References 16 through 20.

(b) Densification Effects: In addition to its effect on fuel temperatures (discussed above), densification affects (1) core

*Although needed in fuel performance codes, this model is reviewed as described in SRP Section 4.4.

power distributions (power spiking, see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR, see SRP Section 4.4), and (3) the potential for cladding collapse. Densification magnitudes for power spike and LHGR analyses are discussed in Reference 21 and in Regulatory Guide 1.126 (Ref. 22). To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References 23 and 24.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 25. Interim methods that may be used prior to compliance with this guidance are given in Reference 26. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by CPB. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References 10, 12, and 13.

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate, the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27). For non-LOCA applications, other correlations may be used if justified.

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

D. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

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

Reference 3

July 20, 1989

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

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

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

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

Differing Professional View Concerning

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

b) Methodology Used For Calculating Related Offsite Doses

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

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

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

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

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

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