DEVELOPMENT OF THE COMPREHENSIVE PROCEDURE GUIDELINE FOR CORE DAMAGE ASSESSMENT

TASK 467

Prepared on Behalf of THE C-E OWNERS GROUP

JULY 1983

POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

EGAL NOTICE

IRED AS AN ACCOUNT OF WORK SPONSORED
, INC. NEITHER COMBUSTION ENGINEERING
TS BEHALF:

ANTY OR REPRESENTATION, EXPRESS OR
IRANTIES OF FITNESS FOR A PARTICULAR
UTY, WITH RESPECT TO THE ACCURACY,
SS OF THE INFORMATION CONTAINED IN THIS
F ANY INFORMATION, APPARATUS, METHOD,
HIS REPORT MAY NOT INFRINGE PRIVATELY

JTIES WITH RESPECT TO THE USE OF, OR FOR
HE USE OF, ANY INFORMATION, APPARATUS,
D IN THIS REPORT.

DEVELOPMENT OF THE COMPREHENSIVE PROCEDURE GUIDELINE FOR CORE DAMAGE ASSESSMENT

TASK 467

May 1983

Prepared on Behalf of

THE C-E OWNERS GROUP

by

C-E POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

ABSTRACT

The purpose of this task is to provide procedure guidelines which can be used under post accident plant conditions to determine the degree and type of reactor core damage from the measured fission product isotopes and from various chemistry and physical parameter measurements readily available to the plant operators. Implementation of this task assumes that project specific implementation of the NUREG-0737 Item II.B.3 requirements for Post Accident Sampling Systems have been met. The task is divided into a two phase program. The first phase of this program is the preparation of a guideline for core damage assessment to serve in the interim to the preparation of the comprehensive procedure. This first phase will determine core damage assessment based only on the radiological analysis of samples obtained from the reactor coolant, containment building sump, and the containment building atmosphere.

The second phase will determine core damage assessment based on a comprehensive evaluation of data on plant condition. The information available from all potential indications will be factored into the final estimate. These indications include the core exit thermocouple temperatures, reactor coolant and containment atmosphere hydrogen concentrations, and containment radiation dose rates. The implementation of both phases is required to comply with the NRC criteria. This report provides the results of both phases of effort on this task.

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

Page

List of Figures

Page

List of Figures (Cont'd)

Page

List of Figures (Cont'd)

TABLE OF CONTENTS (Cont'd)

1.0 INTRODUCTION

1.1 BACKGROUND

The NRC instituted the NUREG-0737 (Reference 7.1) requirements as implementation of the Post TMI Action Plan in November 1980. Among these was the requirement for a design and operational review of plant reactor coolant and containment atmosphere sampling system capabilities under accident conditions. The quantitative review criteria were, in general, beyond the capabilities of existing plants. The industry expended substantial efforts to develop the post-accident sampling systems and equipment necessary to meet the review criteria. The implementation date for operating plants was January 1, 1982 and for other plants was four months prior to achieving five percent power during preoperational tests.

In March 1982, the NRC issued a clarification (Reference 7.2) to NUREG-0737 providing guidance for preparation of a procedure to assess core damage, Appendix A. As stated in this clarification, none of the near term operating license applicants had been successful in providing an acceptable procedure. As a consequence, each near term operating license applicant has a condition which may restrict power operation. Additionally, the NRC stated that a final procedure for estimating core damage may take approximately 12 months. Therefore, the NRC stated its willingness to accept an interim procedure. The interim procedure in conjunction with a firm date for the final procedure would be used to remove the power restricting license condition. The

clarification of the NUREG-0737 requirements was stated with respect to near term operating license applicants. A similar licensing condition may be anticipated by operating licensees as the NRC begins scheduling their review with respect to NUREG-0737.

1.2 PLAN FOR CORE DAMAGE PROCEDURE

Combustion Engineering, in conjunction with the C-E Owners Group (CEOG), has implemented a two-phase program to provide procedure guidelines for assessing core damage following severe accidents. This report is the final product of that two phase effort. The first phase was the interim procedure guideline required by the NRC for assessing the extent of core damage by utilizing only radiological analysis of samples obtained from the Post Accident Sampling System (PASS). These samples are 1) coolant from the Reactor Coolant System (RCS), 2) coolant from the containment building sumps, and 3) gas from the containment building atmosphere. Such samples are available from a Post Accident Sampling System which has the functional capabilities required by Section II.B.3 of NUREG-0737.

The second phase of the CEOG program provides comprehensive procedure guidelines for utilizing chemistry data from the PASS and also from other commonly available instrumentation in addition to the radiological data employed in the interim procedure. The PASS chemistry data include hydrogen concentrations and total gas content in the samples. Other instrumentation includes RCS pressure, Core Exit Thermocouple (CET) temperatures, and containment radiation levels. The final report from this second phase is a comprehensive procedure guideline which utilizes all the PASS sample data and

other instrument indications to provide several complementary estimates of the extent of the core damage. The plant personnel will interpret these damage estimates in combination with their knowledge of the particular plant and accident scenario and their prior training to arrive at a judgement on the extent of core damage.

1.3 DEVELOPMENT OF THE INTERIM PROCEDURE OUTLINE

There are three factors considered in the interim procedure which are related to the specific activity of the samples obtained and are employed to assess the degree and type of core damage. These are the identity of those isotopes which are released, the respective ratios of the specific activities of those isotopes, and the percent of the source inventory of the time of the accident which is observed to be present in the samples.

The NRC guidelines for preparation of this procedure define ten categories of fuel damage intended to address fuel integrity for post accident sampling. These ten categories are characterized according to the anticipated mechanism of fission product release from the fuel. Each mechanism of fission product release is then characterized by the identity of characteristic fission products present in a given post accident sample. This identity may be used to make an initial categorization of the type of core damage. The selection of the representative fission products is described in the following sections.

There are two sources of the fission products released by the fuel. These are the fuel pellet and the fuel gas gap. The presence of a fission product in either source is a function of the fuel history, the diffusion properties of

the isotope and the half life. The relative ratios of the quantity present for an isotope of a given element will differ between the fuel pellet and the fuel gas gap. The type of fuel damage, determined initially by the identity of the characteristic fission product, is then confirmed by calculating the isotope ratios and comparing them to analytically determined standards for the pellet and gas gap. The source of the release is added identification to the type of core damage.

The degree of core damage is expressed in terms of the percentage of the total core inventory available for release. The specific activity of the measured samples is compared to analytically determined curves for the specific activity at the sample location as a function of the total core inventory available for release. The assumptions used to describe the progressive damage expected in fuel rods during core melt accidents and the distribution of the fission products within the fuel rods under normal operation is based upon the material prepared for EPRI through the IDCOR Program, Reference 7.3.

1.4 DEVELOPMENT OF THE COMPREHENSIVE PROCEDURE OUTLINE

There are three factors considered in the comprehensive procedure which are related to the chemistry and physical parameters of the samples obtained and are employed to assess the degree and type of core damage. These are the hydrogen gas content of both the reactor coolant and containment building atmosphere, the reactor coolant temperatures as measured by the core exit thermocouples, and the radiation dose rates measured in the containment building atmosphere.

The NRC guidelines, as previously discussed, define ten categories of fuel damage intended to address fuel integrity for post accident sampling. These ten categories are characterized by the temperatures achieved on the fuel cladding surface. Characterization of core damage in terms of fuel clad surface temperature permits the use in this procedure of two parameters which are measurable following an accident. These are the amount of hydrogen gas within the containment building and the core exit thermocouple temperature. The amount of hydrogen gas measured within the containment by the PASS is correlated to the extent of chemical oxidation of the fuel cladding which occurs at elevated surface temperatures. The maximum coolant core exit temperature measured by thermocouples is correlated to the percent of fuel rods with clad surface temperature above that which is considered to be a threshold for clad rupture due to gas gap overpressurization.

The information obtained from hydrogen measurements as an indication of fuel cladding oxidation is more applicable within the fuel overheat category of core damage. Within this category the clad surface temperatures are sufficiently high to result in the production by oxidation of measurable quantities of hydrogen but are below that which results in fuel clad material melting. The hydrogen gas measurements are obtained by the PASS and analytically corrected to account for the presence of hydrogen gas produced in sources other than fuel clad oxidation. The measurements are used to obtain the total amount of hydrogen produced from fuel clad oxidation. This value is used by procedure to estimate the extent of fuel overheating according to the percent of fuel rods which have been oxidized beyond the limits for continued structural integrity.

The information obtained from reactor coolant core exit temperatures is most applicable within that category of core damage which addresses cladding failure. The mechanism of core damage measurable by core exit temperatures is the rupture of the cladding as a result of high temperature overpressurization of the gas found within the gas gap region of the fuel. It will be shown that this mechanism results in the rupture of a significant fraction of the fuel rods in the core prior to temperature reaching fuel overheating and the onset of oxidation initiated clad failures. The reactor coolant core exit temperatures are measured by thermocouples located within the reactor vessel. Analytical determinations of the radial distribution of the core exit temperature under the most general cases of post accident conditions are described. The measurements obtained following an accident are surveyed to identify the maximum channel exit temperature. This value along with the analytically determined radial distributions of temperature are used by the procedure to estimate the extent of cladding failure. The estimated damage is the percent of the fuel rods in the core which have clad surface temperatures above the established value for rupture due to gas gap overpressurization.

The dose rate inside the containment building is a physical parameter which also may be used to characterize the ten categories of core damage. As previously discussed, with regard to the Interim Procedure, these categories were characterized according to the mechanism for release of fission products from the core. The identity of each of these mechanisms of release may be determined by the presence outside the core of specific characteristic fission products. The dose rates inside the containment building are dependent upon

the quantity, identity, and distribution of those fission products. Therefore characterization of core damage in terms of the mechanisms of release permits the use in this procedure of the measured area radiation dose rate to assess core damage.

The information obtained from area dose rates within the containment building as an indication of fission product release is most applicable within the cladding failure and fuel overheat categories of damage. The application of the fission product release mechanisms to core damage assessment is the same as that used in the Interim Procedure. The difference, when applied to the Comprehensive Procedure, is the physical parameter being measured. In the Comprehensive Procedure the mechanism of release is correlated to the measurement of the area radiation dose rate rather than to the measurement of sample specific activity. The use of two different measurements for the evaluation of the same physical parameter is a means to reduce the uncertainity in the assessment of core damage.

2.0 CATEGORIZATION OF THE EXTENT OF CORE DAMAGE

The task of applying post accident sampling system data to assess the condition of a reactor core following an accident requires some description of the relevant conditions. A wide range of accident types and sequences are possible. Therefore it is not appropriate to attempt to employ specific accident scenarios in the development of such a procedure for core damage assessment. However the end product statement concerning core condition should be capable of describing the thermal hydraulic and material properties of the degraded core to the extent practical for the implementation of that information in emergency decisions. The statement of core condition should be in terms of defined categories which are commonly understood but at the same time do not imply quantitative assessments which are beyond the accuracy of the data evaluation. The Rogovin Report, Reference 7.4, categorizes core damage into four major types as follows; no fuel damage, fuel cladding failures, fuel pellet overheating, and fuel pellet melting. Consistent with these categories the NRC guidelines further delineate each of the three later categories into initial, intermediate, and major thereby assessing the extent of each type of damage. A rationale is then required to describe the resulting ten categories in terms of those physical conditions of the core for which measurable data may be obtained.

Independent of the accident scenario, the start of a degraded core condition in a Pressurized Water Reactor is the result of a thermal inbalance between the heat generated in the fuel and the heat removed from the core cooling water. Core heat removal and coolant heat removal are two of the principal Safety Functions activities of the reactor operator following an accident.

The events following this initiating condition as they relate to the thermal and material state of the core have been the subject of a number of analytical and experimental evaluations. In order to define the physical parameters across the spectrum of core damage it is necessary to first assume that the accident is allowed to progress through that spectrum and then to select an analytical model to predict the resulting conditions.

Particular accident scenarios could be postulated for which changes in the system pressure, the time period of core uncovery, and the rate of uncovery would result in a final core condition anywhere within the range of spectrum of core damage. However as stated previously, this discussion does not assume any particular accident scenario. Accident progression from initial fuel damage through to the eventual condition of major fuel pellet melting is assumed only to allow correlation of the physical parameters anticipated through the progressive core degradation to the ten selected categories of core damage.

The model selected to describe the progressive material interactions and damage expected in fuel rods through the spectrum of degraded core conditions is that described by EPRI through the IDCOR Program, Reference 7.3. That report provides a model which is the result of a state of the art evaluation of a number of independent analytical and experimental works. It is recognized that a definitive model for progressive core degradation has not been developed. However, the results of the IDCOR Program are widely accepted and will therefore be employed as a basis for this procedure.

The progression of core damage, which begins with a loss of the equilibrium in the reactor core heat balance, for the purposes of this report is taken to be as follows. The centerline temperature of a fuel rod will depend upon its power density, the thermal conductivity of the fuel, the gap conductance between the fuel and the cladding of the rod and the conditions of the surrounding coolant. Centerline temperatures are in the range of 2200 to 3300°F for normal operating conditions. Following an accident the core may not be able to reject the stored energy plus the fission product decay heat from the cladding surface due to the initiating loss of heat balance between the rod and the coolant. The surface temperature of the cladding increases, possibly resulting in temporary film boiling of the reactor coolant. The fuel temperatures continue rising, following a loss of coolant accident which uncovers the top of the core, since steam cooling of the uncovered portion of the fuel is not sufficient to remove the decay heat unless there is a large temperature difference between the clad and steam. During depressurization accidents, a pressure differential exists between the gas present in the fuel rod gap and the reactor coolant pressure which may cause the cladding to burst. The cladding burst can be expected to occur in the temperature range of 1400 to 2000°F depending upon the amount of fission gas and prepressurizing helium in the fuel rod, the reactor vessel pressure, the rate of temperature rise, and the time at temperature, Reference 7.5. Clad burst may occur at temperatures as low as 1000°F when high differential pressure is combined with long duration at temperature. The clad burst results in the release of volatile fission products present within the gas gap and to a lesser extent within the fuel pellet surface. Clad rupture does not occur uniformly across the core because of the radial variation in fuel rod peak clad temperature.

As the core becomes uncovered the steam surrounding the fuel rod oxidizes the zirconium present in the exposed length of the cladding. A chemical byproduct of the reaction is the production of hydrogen gas. The oxidation is an exothermic reaction whose rate is dependent upon the surface temperature of the cladding. The exothermic reaction provides an additional heat source which serves as a catalyst to accelerate the rate of reaction. This reaction is the cause of the rise in fuel temperature above 2200°F. During the later stages of core uncovery the steam rising from the lower regions of the core can be consumed by reaction with the cladding in the upper regions.

Oxidation of the zirconium present in the cladding causes embrittlement of the material with subsequent degradation of structural integrity. At some time during the accident the core may be reflooded and cooled or the reactor coolant pumps may be started causing a pressure transient. The embrittled fuel cladding would fragment as a result of either thermal or pressure shock. This increase in fuel surface to volume ratio would increase the release rate of fission products.

Above the temperature range of 2000 to 2550°F, general lattice mobility exists in the fuel allowing fission products to diffuse to more stable thermodynamic states. Atoms which do not react with the UO_2 or any foreign material in the pellet will diffuse from the interstitial location to either a microbubble or metallic phase. At approximately 2450°F, the fission products including noble gas, cesium and iodine will be released from the UO_2 grain boundaries. At temperatures above 2450°F, the fission gas microbubbles include vaporized cesium and iodine.

Above 3250°F the Zircaloy cladding makes. Endothermic reactions occur between molten Zircaloy and ZrO_2 and the dissolution of UO_2 by molten Zircaloy. The release of fission products by diffusion from UO_2 grains begins to occur at a rapid rate. The diffusion process is continuous but the rate is not significant at lower temperatures. The liquid formed as a result of these endothermic reactions flows through the fuel rod gap and continues to dissolve the UO_2 fuel.

Those material interactions and damage expected in fuel rods accompanying prolonged core uncovery relevant to the NRC categories of core damage are summarized in Table 2-1. As described above, a temperature range is associated with each physical condition. The mechanisms of fission product release from a fuel rod which has been burst are related to the fission product volatility and diffusion transport properties. Both of these are temperature dependent. Therefore each of the ten categories of core damage can be characterized by the type of fuel damage, the corresponding temperature range, and the mechanism of fission product release. The characterization of the categories is summarized in Table 2-2.

Therefore the combination of Tables 2-1 and 2-2 provide the definition and physical conditions for each of the ten NRC Categories of core damage which are employed throughout the subject procedure. This provides the required definitions for common understanding of the end product statement of core damage assessment.

TABLE 2-1

Progressive Material Interactions and Damage
Expected in Fuel Rods During Core Melt Accidents

	Types of Fuel Damage	Temperature °F
1.	Ballooning of Zircaloy cladding	> 1300
2.	Burst of Zircaloy cladding	1300-2000
3.	Oxidation of cladding and hydrogen generation	> 1600
4.	Embrittlement of fuel rod cladding by oxidation	> 2200
5.	Fission Product fuel lattice mobility	2000-2350
6.	Grain boundary diffusion release of fission products	> 2450
7.	Melting of metallic Zircaloy	> 3250
8.	Fission Product Diffusion from UO_2 Grains	< 3450
9.	Dissolution of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
10.	Melting of UO_2	5080

Table 2-2

Characterization of NRC Categories of Fuel Damage

NRC Category of Fuel Damage	Mechanism of Release	Temperature Range °F
1. No fuel damage		~ 750
2. Cladding Failures	Clad burst and diffusional gap release	1300-2000
3. Intermediate Cladding Failures		
4. Major Cladding Failures		
5. Fuel Pellet Overheating	Grain boundary diffusion	> 2450
6. Intermediate Fuel Pellet Overheating		
7. Major Fuel Pellet Overheating	Diffusional Release from UO_2 grains	< 3450
8. Fuel Pellet Melt	Escape from molten fuel	> 3650
9. Intermediate Fuel Pellet Melt		
10. Major Fuel Pellet Melt		

3.0 ESTABLISHMENT OF THE BASES FOR CORE DAMAGE ASSESSMENT USING RADIOLOGICAL DATA

The purposes for performing core damage assessments are first to assess the effectiveness both of the reactor operator actions and the automatic engineered safety feature systems to mitigate the consequences of an accident and second to assess the potential for subsequent release of radioactive material to the environment. Section 2.0 of this document described core damage in terms of the material interactions and structural integrity expected in fuel rods experiencing uncovery and and the consequent progressively increasing fuel temperature. Based upon the stated purposes for core damage assessment it is appropriate to define the categories of core damage for use in this procedure in terms of those physical parameters relevant to the release of radioactive material. The postulated scope of core damage encompasses a broad spectrum of physical conditions. Therefore, it becomes necessary to measure as many parameters as possible in order to define the location of the core within that spectrum of damage. Additionally, to obtain a workable procedure it is necessary to limit the definition to those physical parameters for which measurable data may be obtained using the Post Accident Sampling System. This means that those parameters are selected for which specific conclusions may be drawn with respect to core condition and for which the variations in the accident scenario have a minimum influence on that conclusion. Wherever possible, the conditions which influence the measurement of a given parameter are identified.

Within these criteria the core damage categories are defined in terms of the source of fission product release, the mechanism of fission product release, and the quantitative release of characteristic fission products expressed as a percent of the theoretical source inventory. The mechanism of fission product release is identified through the presence of characteristic fission products in the sample medium. The source of fission product release is identified through the relative ratios of the isotopes of a given fission product. The quantitative release is determined by calculation using the concentration measured in the sample and tabulated theoretical source inventories. Each of these selected physical parameters are quantified in terms of measurable data. In each case however there are conditions which may influence the accuracy or limit the validity of the measurement. The following sections describe the technical basis for the selection and use of each of these parameters including the conditions which may influence the accuracy of their measurement.

The objective of the subject core damage assessment procedure is to achieve an evaluation of the radiological data within sufficient accuracy to determine the existing core condition in terms of the ten defined categories described in Section 2.0. The following table provides the criteria by which each category is evaluated with respect to the three physical parameters selected above. By procedure the plant personnel will use the measured radiological data to determine each physical parameter, locate the parameter within the table, and then use the table to state the core condition in terms of the corresponding defined categories.

Table 3-1

Radiological Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release	Source of Release	Characteristic Isotope	Release of Characteristic Isotope Expressed as a Percent of Source Inventory
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I 131, Cs 137 Rb 88	Less than 1
2.	Initial Cladding Failure		Gas Gap		Less than 10
3.	Intermediate Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Xe 131m, Xe 133 I 131, I 133	10 to 50
4.	Major Cladding Failure		Gas Gap		Greater than 50
5.	Initial Fuel Pellet Overheating		Fuel Pellet	Cs 134, Rb 88, Te 129, Te 132	Less than 10
6.	Intermediate Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet		10 to 50
7.	Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet		Greater than 50
8.	Fuel Pellet Melt		Fuel Pellet		Less than 10
9.	Intermediate Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet	Ba 140, La 140 La 142, Pr 144	10 to 50
10.	Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

Core damage will not take place uniformly among all the fuel rods. Uniform fuel rod damage throughout a given core would in fact be an unrealistic assumption due to the radial variations in fuel rod peak cladding temperatures. Therefore, when considering the total effect of the damage on all of the individual fuel rods, the core damage assessment procedure yields a combination of categories which may exist at the time a given sample was obtained. As an example, the analysis of a given sample may indicate the presence of both (1) fission product isotopes characteristic of grain boundary diffusion in a quantity equal to 25 percent of the fuel pellet inventory and (2) fission product isotopes characteristic of cladding burst release in a quantity equal to 100 percent of the fuel gas gap inventory. In this example the core damage assessment would be intermediate fuel pellet overheating with concurrent major fuel cladding failure.

3.1 BASIS FOR SELECTION OF CHARACTERISTIC FISSION PRODUCTS

The mechanism of fission product release from a damaged fuel rod is identified through the presence of characteristic fission products in the sample medium. A survey has been completed to determine the fission product isotopes which characterize a given mechanism of release. These isotopes are chosen to determine the degree and type of core damage. Specifically the isotopes are selected to differentiate between the three major types of core damage - cladding failure, fuel overheat, and fuel melt. The criteria for selection of the isotopes includes half life, the quantity present in the core, the rate at which they reach equilibrium in the core inventory with respect to fuel burnup, the degree to which their presence in a sample represents a specific type of core damage, detectability using standard semiconductor and

multichannel analyzer techniques within a postulated fission product mixture, and the amount of information available on their chemical behavior.

The fission products selected all have radioactive half lives of sufficient duration to ensure that they will be present in quantity and time period following an accident to allow detection and analysis. Another important related factor is the history of the fuel prior to cladding rupture. The physical properties of the isotope determine the rate at which a specific isotope inventory approaches equilibrium in the core as a function of core burnup. Implementation of the subject procedure under post accident conditions necessitates simplification of data analysis whenever possible. Therefore analytical correction of measured data to a standard core burnup is not desirable. Selection of monitored isotopes which reach radiological equilibrium quickly within the fuel cycle eliminates this concern. The physical parameters of influence to this selection are isotopic half life, fission product yield, cross section for loss due to neutron absorption, and decay chain branching fractions.

To implement the selection criteria, the isotopes selected are divided into two groups. The first group includes those isotopes with half lives between four hours and fifteen days. These isotopes are used to assess the damage condition for cores that have been operational in a given cycle for more than thirty days. These isotopes reach radiological equilibrium levels in the core after thirty days of operation. The second group includes those isotopes with half lives between one hour and twenty four hours. These isotopes are used to assess the damage condition for cores that have been operational in a given

cycle for fewer than thirty days. This group is used in determining core damage early in a given core cycle, but has the limitation that sampling and analysis must be completed within a few hours following the accident to avoid the loss of data by isotopic decay.

The selection of fission products by detectability is a very practical criteria in the implementation of the subject procedure. Numerous factors influence the ability to sample and detect specific isotopes. Reliability of the sampling is hampered by rapidly changing plant conditions, equipment limitation, and lack of operator familiarity with rarely used analytical procedures. Chemists are required to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources and may not result from a uniform distribution of the sampled medium. Cooling or reactions may take place in the long sampling lines. Therefore the results obtained may not be representative of the plant condition. Plant conditions, radiation exposure, and time requirements may prohibit multiple samples and reduce statistical reliability.

Specific criteria for detectability of a fission product in a given sample is based upon the capabilities of typical semiconductor detectors employing multichannel analysis of the fission product gamma energy spectrum. These criteria include the principal fission product decay energy, the presence of other isotopes with similar or masking decay energies, and the success of such measurements in experiments conducted by C-E and other reported measurements.

Selection of fission product isotopes as being representative of specific types of core damage and with respect to the availability of data on their chemical behavior is based upon a survey of the published literature. The reports which were of specific contribution are the IDCOR Draft Final Report, Reference 7.3, and the Rogovin Report, Reference 7.4. The specific criteria to select isotopes as indicators of the type of core damage is their respective volatility.

The category of no core damage is characterized by the release of fission products through the mechanisms of spiking and tramp uranium fission. Reactor coolant system pressure, temperature, and power transients may result in iodine spiking. Iodine spiking is identified by a rise in reactor coolant iodine concentrations during the period from 4 to 8 hours after the transient. The iodine concentrations can be bounced by a value of 500 times the equilibrium levels during faulted conditions such as a steam generator tube rupture without any fuel cladding failure. Spiking is identified by a decrease in reactor coolant concentration subsequent to the spike peak at a rate equal to the system purification half life.

The categories of core damage identified as cladding failure are characterized by the release of fission products through the mechanisms of burst and gas gap diffusion. The characteristic fission products are the noble gases and halogens, which, because they are volatile can migrate quickly through the fuel pellet and gas gap for release following cladding rupture. These isotopes are volatile in the temperature range (1300-1800°F) accepted as cladding burst temperatures. When the cladding ruptures the entire amount of noble fission gases previously accumulated in the plenum and open voids in the

fuel will be assumed to be released. This amount can range up to 25% of the long half life fission gas isotopes depending on power history. Cesium and iodine are also released when the cladding ruptures but the quantity carried out with the vented gas is considerably less than that for the noble fission gases. The initial release of cesium and iodine depends upon the fuel temperature, the volume of gas vented, and the amount of cesium and iodine initially in the fuel gap. The diffusion release of the remaining halogens in the gas gap is a slow process in the cladding burst failure temperature range.

The categories of core damage identified as fuel overheat are characterized by the release of radioactivity through grain boundary diffusion and by diffusion from within the UO_2 grains. Grain boundary diffusion begins above 2450°F. The moderately volatile isotopes of cesium, rubidium, and tellurium are characteristic of this type of damage. The IDCOR report estimates that 20% of the total initial fuel inventory of stable and long lived cesium would be released at temperatures consistent with grain boundary diffusion. Diffusional release of these isotopes from within the UO_2 grains increases rapidly beyond this temperature and the rate is a subsequent function of temperature.

At greater temperatures (2550-3450°F) reactions begin between the solid UO_2 and solid metallic zircaloy, melting of the control rods materials, and melting of the zirconium. This is the onset of the categories of core damage identified as core melt. At these temperatures, greater amounts of tellurium are released. Alkali metals such as barium volatize as well as rare earths and actinides such as lanthanum and protactinium. The amount and type of isotopes released is dependent on the extent of fuel fragmentation.

Table 3-2

Selected Isotopes for Core Damage Assessment

Category of Core Damage	Selected Isotope	Half Life	Fuel History Grouping	Principal Energy, Mev	Core Inventory Order of Magnitude
	Kr 87	76m	2	0.403	1(+7)
Clad	Xe 131m	12d	1	0.164	5(+5)
Failure	Xe 133	5.4d	1	0.081	1(+8)
	I 131	8d	1	0.364	7(+7)
	I 132	2h	2	0.955	1(+8)
	I 133	21h	1&2	0.53	1(+8)
	I 135	6.8h	2	1.14	1(+8)
Fuel	Cs 134	2yr	1	0.605	2(+7)
Overheat	Rb 88	2m	2	1.86	4(+7)
	Te 129	70m	2	0.445	2(+7)
	Te 132	78h	1	0.23	1(-8)
Fuel Melt	Sr 89	52.7d	1	0.91	1(+8)
	Ba 140	12.8d	1	0.537	1(+8)
	La 140	40h	1	1.596	1(-8)
	La 142	90m	2	0.65	2(+8)
	Pr 144	17.4m	2	0.695	9(+7)

Based on these criteria Table 3-2 provides a list of the isotopes selected for analysis in the subject procedure. The isotopes are grouped according to the type of core damage their presence represents and according to their use with respect to fuel history prior to the accident.

3.2 BASIS FOR IDENTIFICATION OF THE SOURCE OF THE RELEASE

The identification of the source of the fission product release is useful in determining the extent of damage which may exist in a core following a given accident scenario. For a particular accident the radial variation in peak fuel cladding temperature can be significant. Therefore accident scenarios can be postulated in which a limited number of fuel rods may experience fuel pellet overheating while the majority of the fuel may not reach the 1200°F temperature required for cladding burst. During such an accident the identity and quantity of fission products detected in reactor coolant samples is insufficient information to determine the type of damage which has occurred. The added information needed to evaluate the accident is the source of the detected fission products. Specifically it is necessary to determine whether the fission products have been released from the fuel rod gas gap or from the fuel pellet. This determination can be performed using the relative ratios of the isotopes of a given fission product.

During the fission process the relative ratios of the isotopes of a given fission product will remain constant. The value of the ratio is dependent upon the material being fissioned and the energy of the neutron which induces the fission. Each isotope has its own characteristic half life. Therefore

the ratios of the isotopes will vary as a function of time following their production. If it is assumed that the only loss term from the fuel rod is due to decay of the isotopes then an equilibrium condition is reached in which the production of the isotopes will equal their loss due to decay. Under equilibrium conditions a fixed inventory of the isotopes exists within the fuel rod. The assumed condition is practical for selected fission products which are products of a limited number of precursors and whose isotopes have small neutron absorption probabilities. For these fission products the relative ratio of their isotopes within the fuel pellet can be considered a constant when the reactor has operated for sufficient time for equilibrium to have been reached.

During power operation the central temperature of a fuel rod is significantly higher than that of the fuel rod gas gap or cladding surface. Thus a large temperature gradient exists across the fuel pellet. Such temperature gradients cause substantial migration of volatile fission products if they are unhampered by chemical reaction within the pellet. Those fission products which migrate along that gradient and reach the gas gap will consist of material which has existed in the pellet for sufficient time for this migration to take place and therefore may be considered to consist of the older collection of material. The relative ratios of the isotopes of fission products found in the gas gap is therefore different from that found in the fuel pellet because the ratio varies as a function of time following production. Thus, theoretical calculations may be employed to determine

typical ratios for isotopes of a fission product in a given region of the core. Comparison of the ratios obtained from sample data with these calculated values determines the source of the fission product release.

The fission products iodine and xenon were chosen for use in this procedure by employing the criteria for selection of elements for which the assumption of equilibrium conditions is practical. Table 3-3 provides the results of theoretical calculations of the relative ratios of the isotopes of these elements when found in the fuel pellet and in the gas gap. The calculation of the values found in the fuel pellet employed the ORIGEN computer code for analysis of fission product inventories. The calculation of the values found in the gas gap employed the ANS 5.4 Standard assumptions for the percent of the fuel pellet fission product inventory which enters the gas gap region of a rod in a fuel assembly with core average burnup. The values are stated as a range rather than a specific value. The range is employed to account for the inaccuracies inherent in the calculations and for the differences in core design among the C-E NSSS's.

TABLE 3-3

ISOTOPE RATIOS FOR FISSION PRODUCT SOURCE IDENTIFICATION

ISOTOPE	ACTIVITY RATIO IN FUEL PELLET INVENTORY	ACTIVITY RATIO IN GAS GAP INVENTORY
Kr 87	0.2	< 0.001
Xe 131m	0.003	0.001-0.003
Xe 133	1.0	1.0
I 131	1.0	1.0
I 132	1.4	0.01-0.05
I 133	2.0	0.5-1.0
I 135	1.8	0.1-0.5

* $$\text{Ratio} = \frac{\text{Noble Gas Isotope Inventory}}{\text{Xe 133 Inventory}}$$

$$= \frac{\text{Iodine Isotope Inventory}}{\text{I 131 Inventory}}$$

3.3 BASIS FOR THE DETERMINATION OF THE QUANTITATIVE RELEASE OF FISSION PRODUCTS

The quantitative release of characteristic fission products is expressed as the percent of the source inventory at the time of the accident which is observed to be present in the sampled media and therefore available for immediate release to the environment. The initial source inventory is theoretically calculated for equilibrium conditions. Prior to use, this value is corrected by procedure to describe the fission product inventory at the time of the accident. The value of this inventory is dependent upon the source of the fission product release which, as described in Section 3.2, may be either the fuel rod gas gap or the fuel pellet. The reason to define the quantity of released fission product as that which is observed to be present in the sampled media is a consequence of the limits on the present capability to predict fission product transport and of the use of this information. Analytical models for fission product transport following release from a degraded reactor core are not definitively developed. Realistic estimates and data from actual accident case studies indicate that a smaller percent of the fission products is released to the environment than is anticipated by the Regulatory models. This is explained by retention of otherwise volatile species within chemical reactions occurring in the degraded core, by oxidizing reactions occurring within the water inventory present in containment, and finally by the plateout of non volatile species on containment building surfaces with subsequent reevolution into volatile form. The information on core condition is required to make realistic assessments of the potential for radioactive releases at the time of an accident. These assessments should not

be based upon analytical models developed for worst case licensing evaluations. Therefore, the quantitative assessments are defined in terms of the amount of fission products measured in sample fluids which are available for transport to the environment.

This distinction is best explained by example. Consider the case in which measured samples of the containment building atmosphere and reactor coolant indicate that 20 percent of the I-131 isotope calculated to be in the gas gap is now found in the sampled fluids. This does not indicate that 20 percent of the fuel rods have been ruptured. A greater number may be anticipated to have failed. This number cannot be determined because the effects of oxidation within the core and plateout are not analytically known. Therefore, it can only be stated that 20 percent of the gas gap source inventory is available for release to the environment. Using the core damage characteristics defined in Table 3-1 this would indicate Intermediate Cladding Failure.

The analytical models used to determine the fission product source inventories are well defined for equilibrium normal operating conditions. The fuel pellet inventory for the selected characteristic isotopes are provided in Table 3-4. These values were calculated using the ORIGEN computer code, Reference 7.6, with the assumptions of 2 year core average burnup and 100 percent power operation. The corresponding fuel rod gas gap inventories are provided in Table 3-5. These values were calculated with the assumption of equilibrium diffusion rates based upon average values predicted by ANS 5.4 Standard Models. The values are expressed as the gas gap fission product inventory of all rods in the average fuel assembly times the number of fuel assemblies in the core.

TABLE 3-4

EQUILIBRIUM CORE INVENTORY OF CHARACTERISTIC FISSION PRODUCTS

	PLANT CLASS, MWT						
ISOTOPE	1500	2530	2560	2700	2815	3390	3800
Kr 87	1.8(7)	3.0(7)	3.1(7)	3.2(7)	3.4(7)	4.7(7)	5.4(7)
Xe-131M	2.9(5)	4.5(5)	4.6(5)	4.9(5)	5.2(5)	7.0(5)	8.2(5)
Xe-133	1.5(8)	1.4(8)	1.5(8)	1.5(8)	1.6(8)	2.0(8)	2.4(8)
I-131	4.2(7)	7.2(7)	7.3(7)	7.6(7)	8.0(7)	9.9(7)	1.2(8)
I-132	7.0(7)	1.0(8)	1.0(8)	1.1(8)	1.2(8)	1.4(8)	1.7(8)
I-133	1.5(8)	1.4(8)	1.5(8)	1.5(8)	1.6(8)	2.0(8)	2.4(8)
I-135	8.6(7)	1.3(8)	1.3(8)	1.4(8)	1.5(8)	1.9(8)	2.3(8)
Rb-88	2.9(7)	4.4(7)	4.5(7)	4.8(7)	5.0(7)	6.8(7)	7.9(7)
Sr-89	3.9(7)	6.1(7)	6.1(7)	6.6(7)	7.0(7)	9.4(7)	1.1(8)
Te-129	1.6(7)	2.3(7)	2.4(7)	2.5(7)	2.5(7)	3.1(7)	3.7(7)
Te-132	7.0(7)	1.0(8)	1.0(8)	1.2(8)	1.3(8)	1.4(8)	1.7(8)
Cs-134	6.1(6)	1.1(7)	1.9(7)	1.2(7)	1.3(7)	1.8(7)	2.4(7)
Ba-140	8.0(7)	1.3(8)	1.3(8)	1.4(8)	1.5(8)	1.7(8)	2.1(8)
La-140	8.4(7)	1.3(8)	1.3(8)	1.4(8)	1.5(8)	1.8(8)	2.1(8)
La-142	1.0(8)	1.5(8)	1.57(8)	1.6(8)	1.7(8)	2.2(8)	2.6(8)
Pr-144	6.5(7)	9.1(7)	9.1(7)	9.6)7)	1.0(8)	1.2(8)	1.4(8)

TABLE 3-5

EQUILIBRIUM GAS GAP INVENTORY OF CHARACTERISTIC FISSION PRODUCTS

	PLANT CLASS, MWT						
ISOTOPE	1500	2530	2560	2700	2815	3390	3800
Kr 87	3.6(0)	6.1(0)	6.3(0)	6.5(0)	7.0(0)	9.5(0)	1.1(1)
Xe-131M	2.7(4)	4.2(4)	4.3(4)	4.6(4)	4.9(4)	6.6(4)	7.7(4)
Xe-133	1.3(7)	1.2(7)	1.3(7)	1.3(7)	1.4(7)	1.8(7)	2.1(7)
I-131	4.4(6)	6.6(6)	6.7(6)	7.0(6)	7.3(6)	9.0(6)	1.1(7)
I-132	4.9(3)	7.0(3)	7.0(3)	7.7(3)	8.4(3)	9.9(3)	1.2(4)
I-133	4.4(6)	6.2(6)	6.7(6)	6.7(6)	7.1(6)	8.9(6)	1.1(7)
I-135	7.0(5)	1.1(6)	1.1(6)	1.1(6)	1.2(6)	1.6(6)	1.9(6)

The tabulated values of fission product source inventory are for equilibrium normal operating conditions. The required information is the source inventory at the time of the accident. Therefore, these values must be corrected to account for the history of the core up to that time. The specific parameters which must be accounted include the core power level and average fuel burnup. To account for variations in core power level under the condition in which the power has been maintained for sufficient time to allow the characteristic fission product to reach equilibrium requires only a simple power ratio. Within the accuracy of the subject procedure it is established that a time period of 4 half lives is sufficient to achieve equilibrium conditions. For those power histories in which equilibrium conditions do not exist a transient analytical correction is provided in the procedure. Derivation of the transient correction equation is provided in Appendix B.1.

Implementation of the subject procedure under post accident conditions necessitates simplification of data analysis whenever possible. This can be achieved through appropriate selection of the characteristic fission products as described in Section 3.1 thereby avoiding the need for use of the transient power correction equation. The characteristic fission products are divided into two groups based upon their respective half lives. Under those conditions in which core power level has been maintained constant for a period of time greater than 4 days but less than 30 days then the fission products in Group 2 should be employed for analysis. Under those conditions in which the

core power level has been maintained constant for a time period greater than 30 days then the fission products in Group 1 should be employed for analysis. Proper selection of the fission product Group results in equilibrium inventories which do not require the transient analytical correction.

Selection of the appropriate fission product group requires a determination of the period of constant core power. Within the accuracy of the subject procedure, the acceptance criteria for constant power is a variation of ±10 percent from the time average value.

The final analytical corrections which must be made to the fission product release determination are the correction of the sample measured value to account for decay from the time the sample was analyzed back to the time of reactor shutdown and the correction for the difference between the temperature and pressure of the analyzed sample and that of the fluid prior to removal from containment.

The Post Accident Sampling System locations for liquid and gaseous samples are anticipated to be different for each plant. To obtain the most accurate assessment of core damage, it is necessary to sample and analyze radionuclides from at least the principal locations which include the reactor coolant system, the containment building sump, and the containment building atmosphere. Other samples may be taken dependent upon system capabilities. The measured specific activity of each sample is related to the total quantity at each sample location. The sum of these quantities is then considered to be the total quantity available for release to the environment. Typically

several hours are required to recirculate, obtain, and analyze a sample from each location. Therefore, the sample location to be used during the initial phase of an accident should be selected based on the type of accident in progress. Knowledge of a specific accident scenario is not required. The initial sample location can be selected based upon known pressure, temperature and level indications obtained from the plant control room. A list of the appropriate initial sample location is provided in Table 3-6 for various accidents should the scenario be known and for various plant conditions should the scenario be unknown.

The measured values obtained from the Post Accident Sampling System are expressed as the specific activity of the sample fluid. To obtain the total quantity of the fission products at each location it is required to know the quantity of sample fluid at that location. This information is obtained from the control room and includes the reactor coolant system pressurizer and reactor vessel levels, the reactor coolant pressure and temperature, the containment building sump level, and the containment building pressure and temperature. This is the same information which is used to select the initial sample location.

3.4 GENERAL CONSIDERATIONS ON THE LIMITATIONS OF THE PROCEDURE

Considering that ideal conditions will not exist.the subject procedure is based upon the measurement of as many parameters as possible. The core damage assessment procedure is anticipated to yield a combination of categories which may exist at the time a given sample is taken. Individual measurements may appear to be contradictory. The user is required to exercise knowledgeable

judgement in the interpretation of the limitations of the procedures capability to evaluate a given piece of information. There are numerous sources of error in the interpretation of such information including the determination of fission product inventory, the models for fission product transport out of the core, the system capability to obtain representative samples, and the system capability to analyze the samples.

TABLE 3-6
SAMPLE LOCATIONS APPROPRIATE FOR CORE DAMAGE ASSESSMENT

ACCIDENT SCENARIO KNOWN	RCS HOT LEG	RCS PRESSURIZER	CONTAINMENT SUMP	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM	STEAM GENERATOR SECONDARY
Small Break LOCA, Reactor Power >1%	Yes	Yes	---	Yes	Yes	---
Small Break LOCA, Reactor Power <1%	Yes	Yes	---	---	Yes	---
Small Steam Line Break	Yes	Yes	---	---	---	---
Large Break LOCA, Reactor Power >1%	Yes	---	Yes	Yes	Yes	---
Large Break LOCA, Reactor Power <1%	---	---	Yes	Yes	Yes	---
Large Steam Line Break	Yes	---	---	Yes	---	---
Steam Generator Tube Rupture	Yes	---	---	Yes	---	Yes

TABLE 3-6 (Cont.)
SAMPLE LOCATIONS APPROPRIATE FOR CORE DAMAGE ASSESSMENT

ACCIDENT SCENARIO UNKNOWN	RCS HOT LEG	RCS PRESSURIZER	CONTAINMENT SUMP	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM	STEAM GENERATOR SECONDARY
SIS Actuated	Yes	Yes	---	---	Yes	---
Alarm on Containment Building Radiation Monitor	---	---	Yes	Yes	---	---
Alarm on CVCS Letdown Radiation Monitor	Yes	Yes	---	---	---	---
Alarm on Containment Building Sump Level	---	---	Yes	Yes	---	---

The interim procedure is based on the comparison between measured sample data obtained under post accident conditions and analytically determined values for source inventory at the time of the accident. Therefore, the principal consideration is the model of the characteristic fission products in the fuel prior to cladding rupture. The two significant factors are the fuel power history and the power density. The fuel power history determines the fission product inventory in the fuel pellet. The power density determines the fission product migration behavior within the fuel.

Calculations of the fuel pellet inventory under the equilibrium normal operating conditions using the ORIGEN computer code yield reliable data. Parametric evaluations of the acceptance criteria for determining if the power history satisfies equilibrium conditions based upon the half life of the characteristic fission product are accurate to within 10 percent. Therefore, this technique is consistent with the intended purpose.

Calculations of the gas gap inventory is less reliable. Fission product migration to the gas gap is dependent upon local power density, fuel burnup, fuel rod temperature gradient, and chemical reaction with other fission products or with the cladding. The gas gap inventory can differ greatly among the individual fuel rods in the core. Therefore the procedure does not attempt to predict a specific number of fuel rod failures but compares the quantity of fission products released against the entire core gas gap inventory. The core average gas gap inventory can be calculated with greater reliability.

A number of other factors influence the reliability of the chemistry samples upon which the procedure is based. Reliability is influenced by the ability to obtain representative samples due to incomplete mixing of the fission products in the large liquid and gas volumes, equipment limitations, and lack of operator familiarity with rarely used procedures. The accuracy achieved in the radiological analyses are also influenced by a number of factors. The equipment employed in the analysis may be subjected to high levels of radiation exposure over extended periods of time. Chemists are required to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources and may not result from a uniform distribution of the sample fluid. Cooling or reactions may take place in the long sample lines. Therefore, the results obtained may not be representative of plant conditions. To minimize these effects multiple sample analysis over an extended time period is employed. Additionally, upon completion of the second phase of this task, procedures will be available to assess core damage using the balance of plant indications which include core exit temperatures, the quantity of hydrogen released from zirconium degradation, and containment radiation monitors.

As a result of these considerations, the assessment of core damage is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the ten categories of core damage characterized in Table 3-1. However, this is sufficient accuracy to allow plant operators to make informed decisions under post accident plant conditions.

4.0 ESTABLISHMENT OF THE BASIS FOR CORE DAMAGE ASSESSMENT USING HYDROGEN MEASUREMENT

This section discusses the sources of hydrogen gas during severe accidents, the amount of hydrogen predicted in the coolant and containment and the relation between the measured amounts and the core damage. There are multiple sources of hydrogen during severe accidents, including oxidation of zirconium in the core, oxidation of various containment materials and radiolytic decomposition of water. The discussion evaluates each of these sources and presents methods for determination of the amount of hydrogen which is generated by core oxidation.

The amount of hydrogen generated is related to the category of core damage through analyses of selected accident scenarios. The analyses predict the progression of core heatup and oxidation throughout the core during core uncovery accidents. It is shown that the amount of local oxidation is related to the local temperature and therefore to the time of clad rupture. Also, when the amount of local oxidation exceeds the oxidation threshold for clad embrittlement, clad fragmentation occurs. By summing the local oxidation throughout the core, the total core oxidation and hydrogen generation is obtained. This total is related to the type and amount of clad damage and therefore to the ten categories of damage.

Clad damage determined from hydrogen is expressed in two ways - as the number of fuel rods which are ruptured and as the number which exceed the oxidation embrittlement threshold. The number of rods with ruptured clad places the

damage estimate in one of categories 2 through 4 of Table 4-1, which is the equivalent of Table 3-1 for radiological characteristics. The number of rods embrittled represents the number which structurally fail and which release fuel pellets into the coolant. The analyses indicate that the oxidation embrittlement threshold is reached at about the same time as the clad melt temperature of 3350°F, placing embrittled rods in the equivalent categories 5 through 7 for fuel pellet overheating. The measurement of hydrogen is not utilized to place the damage in categories 8 through 10 for pellet melt because definitive relationships are not available between the amount of hydrogen produced and such severe core degradation.

The placement of a given measurement of hydrogen generated into a specific damage category is dependent on the scenarios analyzed to relate damage to oxidation. Selection of base conditions for analysis is discussed in the following section.

4.1 BASES FOR SELECTION OF HYDROGEN TO ASSESS CORE DAMAGE

Hydrogen is produced in the core by the oxidation of zirconium in the fuel cladding and other fuel assembly components according to the reaction:

$$2\ H_2O + Zr \rightarrow ZrO_2 + 2\ H_2$$

Table 4-1

Clad Damage Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1.	No Fuel Damage	~750	None	--	--	Less Than 1
2.	Initial Cladding Failure	1200-1800	Rupture Due to Gas Gap	Maximum Core Exit	<1550°F*	Less Than 10
3.	Intermediate Cladding Failure		Overpressurization	Thermocouple Temperature	<1700°F*	10 to 50
4.	Major Cladding Failure				≲2300°F ≲2% Oxidation	Greater Than 50
5.	Initial Fuel Pellet Overheating	1800-3350	Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6.	Intermediate Fuel Pellet Overheating				<18%	
7.	Major Fuel Pellet Overheating				≲65%	Greater Than 50

* Depends on Reactor Pressure and Fuel Burnup. Values Given for Pressure ≤1200 psia and Burnup >0.

This reaction always exists in the presence of water, but at normal operating temperatures the rate of reaction is very small. Hydrogen is also generated radiolytically and the amount usually in the primary system masks the hydrogen generated by the oxidation of zirconium, so that clad oxidation is not normally discernible by measurement of the hydrogen concentration. Typically, the equivalent normal oxide thickness accumulated on the cladding over the entire life of the fuel is about 0.0004 inch or 1.5% of the original clad thickness.

The normal maximum temperature reached at the clad surface is limited to a few degrees above the saturation temperature by nucleate boiling. At 2250 psia, and the typical peak heat flux, the Jens-Lottes correlation predicts maximum surface temperatures of about 6 degrees above the saturation temperature of 653°F. Two abnormal situations may be hypothesized which cause higher temperature and higher clad oxidation: 1) Departure from Nucleate Boiling (DNB) might occur while the core is covered with coolant; and 2) the core may uncover and heat up because the resulting steam cooling is inadequate.

In the first situation, DNB may occur during transients which are initiated from outside the Limiting Conditions for Operation (LCO), transients for which multiple failures occur, or transients for which deviations from the assumptions in the Reactor Protection System setpoints occur. A summary of event types which can result in DNB is included in Reference 7-7. DNB is localized and is expected to be of short duration. Temperatures are below about 1100°F and the total hydrogen generated by oxidation is too small to be observed. A detailed discussion of fuel rod behavior during DNB is given by

Reference 7-8. The PASS is not intended for assessment of damage caused by such events, except that if clad rupture should accompany DNB, an increase in coolant activity might be observable.

The second situation which can cause high clad temperature is the main subject of this section. In this situation, the reactor is tripped and power is from fission product decay only. The fuel is adequately cooled as long as the core is covered with fluid, even without primary coolant pump flow. In order to uncover the top of the core, over 70% of the primary coolant mass must be lost. This can occur only with the category of events known as Loss of Coolant Accidents (LOCA). These events are divided among three groups: large breaks, small breaks, and complete loss of heat sink.

Large breaks are characterized by very rapid blowdown to containment pressure, resulting in total core uncovery, followed by rapid reflood by the Safety Injection Tank (SIT) flow, and continuation of heat removal by HPSI and LPSI flows. If reflood does not occur the core will heat up adiabatically. If reflood does occur, but continued cooling flow does not occur, the coolant progressively boils off and uncovers the core again, similar to boiloff during a small break, but at a lower pressure.

Small breaks are characterized by rapid blowdown to saturation pressure at the temperature of the secondary side of the steam generators. This is followed by continued reactor coolant flow out of the break and by decreasing pressure. Both are dependent on break size. If more than 70% of the coolant is lost, the top of the core uncovers. Design functioning of the HPSI pumps and, if

pressure falls low enough, of reflood by the SIT, would recover or prevent uncovery of the core. In the unlikely event that they do not function, the coolant boils off and the core progressively uncovers. Extensive discussion of various small break scenarios is given in CEN-114 and CEN-115, prepared for the CEOG (References 7-9 and 7-10).

A complete loss of heat sink results in heat up of the primary system, with consequent pressure increase until the Power Operated Relief Valve (PORV) and/or safety valves on the pressurizer open, releasing primary coolant. Thereafter, the scenario is similar to a small break, except that core uncovery occurs at a much higher pressure. The accident at TMI-2 was essentially the same as this scenario, combined with the effects of various operator actions.

The preceding discussion touches on the variety of conditions which might accompany core uncovery and the consequent heatup and core damage. Essentially all these conditions are more severe than those calculated for Design Basis Accidents and approach the conditions called Degraded Core. Substantial equipment malfunctions and operator errors are required to achieve these conditions. In order to establish a procedure for damage assessment, one set of core conditions is selected as a basis for quantifying the relationship between the amount of hydrogen generated and the core damage. Other possible scenarios are evaluated qualitatively relative to this base.

As stated previously, the core is adequately cooled following reactor trip as long as the core is covered. Therefore, the base core conditions for evaluation of damage are independent of that portion of the scenario which prececes core uncovery. All scenarios leading to damage differ mostly in the rate of core uncovery. For a small break, the core uncovers by boiloff and possibly the effects of misoperation of primary pumps. For larger breaks, the core uncovers by blowdown. Since the most general situation to occur is that equivalent to boiloff, the following assumptions are made for the base conditions:

1) The core uncovers by boiloff at constant pressure.

2) After boiloff down to a given coolant level below the top of the fuel, the event is terminated by rapid, complete core recovery, with relatively little additional clad oxidation after the minimum level is attained.

3) The procedure for estimating core damage is implemented after core recovery. The time it takes to obtain and evaluate the PASS samples is long compared to the likely duration of core uncovery. Nevertheless, the procedure may be used with hydrogen samples obtained earlier. However, these earlier samples would probably provide a lower limit estimate of current core damage, since the hydrogen contained within the voided primary system would not be sampled.

If the progression of core uncovery differs from these assumed conditions, then the amount of core damage inferred from a given measurement of hydrogen may be biased, depending on the uncovery rate. Two examples are:

1) A large break LOCA may cause rapid total core uncovery followed by almost adiabatic heatup of the uncovered core. In the absence of steam to oxidize the zirconium, hydrogen generation is limited, but fuel overheat and melting occurs if cooling is not restored. A subsequent measurement of hydrogen yields an underprediction of core damage for this case.

2) A small break LOCA uncovers more slowly when some liquid enters the reactor vessel from the Emergency Core Cooling System (ECCS) or from runback of condensed steam via the hot legs. The hydrogen is generated by more extensive oxidation along a smaller length of fuel clad near the top of the core. A measurement of total hydrogen generation underestimates the extent of local oxidation on the damaged rods.

The base conditions selected for analysis to prepare this damage assessment procedure are boiloff without inlet flow. For a measured total amount of core oxidation, the base conditions yield a lower limit estimate of the number of damaged fuel rods. Later sections of this presentation qualitatively relate the results of other scenarios to the hydrogen generated during boiloff and indicate what additional instrument indications may be utilized. In conclusion, the use of hydrogen provides a good indication of damage for boiloff conditions and a lower limit indication for other scenarios.

4.2 PREDICTION OF FUEL CLAD EMBRITTLEMENT BASED ON HYDROGEN PRODUCTION

In this section, the relationship is established between measured hydrogen and core damage for the base case condition of boiloff at constant pressure. Simplified analyses are used to demonstrate the relations among the parameters and the applicability of the analyses to all C-E designed reactors. Most previous analyses of core uncovery were done for specified Design Basis Events with an NRC approved Evaluation Model to verify that the limits of 10CFR50.46 are satisfied. Those regulatory design bases limits are a peak clad temperature of 2200°F, maximum local clad oxidation of 17% and total core oxidation of 1%. These limits and the capability of the analytical models are inadequate for prediction of the severe core conditions for which the PASS is designed. Hence, the relationships among clad damage, hydrogen generation and other parameters are obtained with new analyses. They are less detailed, less rigid analytical models intended to give most probable or best estimate results with more severe core conditions. The results are considered adequate for the scope of damage assessment needed during an accident in progress.

A detailed analysis of these "degraded core" conditions would require state-of-the-art computer codes which properly model all the interactive physical phenomena. Such analyses are beyond the scope of effort for preparation of the damage assessment procedures provided herein. The analyses employed in the development of this procedure are sufficient to provide a basis of decision for implementation of the site emergency action plan. A few detailed analyses are included herein as a qualitative overcheck on the general predictions of the simplified analyses which are the bases for the damage assessment procedure.

4.2.1 Hydrogen Generation Correlation

The fission product decay power produced in the exposed length of core above the two-phase level is partially removed by the flowing steam generated by the covered length of core. As the core uncovers, less steam is generated and at the same time a longer length of fuel is exposed requiring cooling. Consequently, the temperature of the exposed fuel rises as the core uncovers until it is high enough to cause significant oxidation heating.

As the fuel clad temperature rises the zirconium oxidation rate increases. The oxidation reaction is exothermic, and the combination of poor cooling, fission product decay power and an exothermic reaction causes an acceleration in the fuel rate of temperature rise as temperature increases. Hydrogen generation is therefore significant in the predictions of the rate of damage.

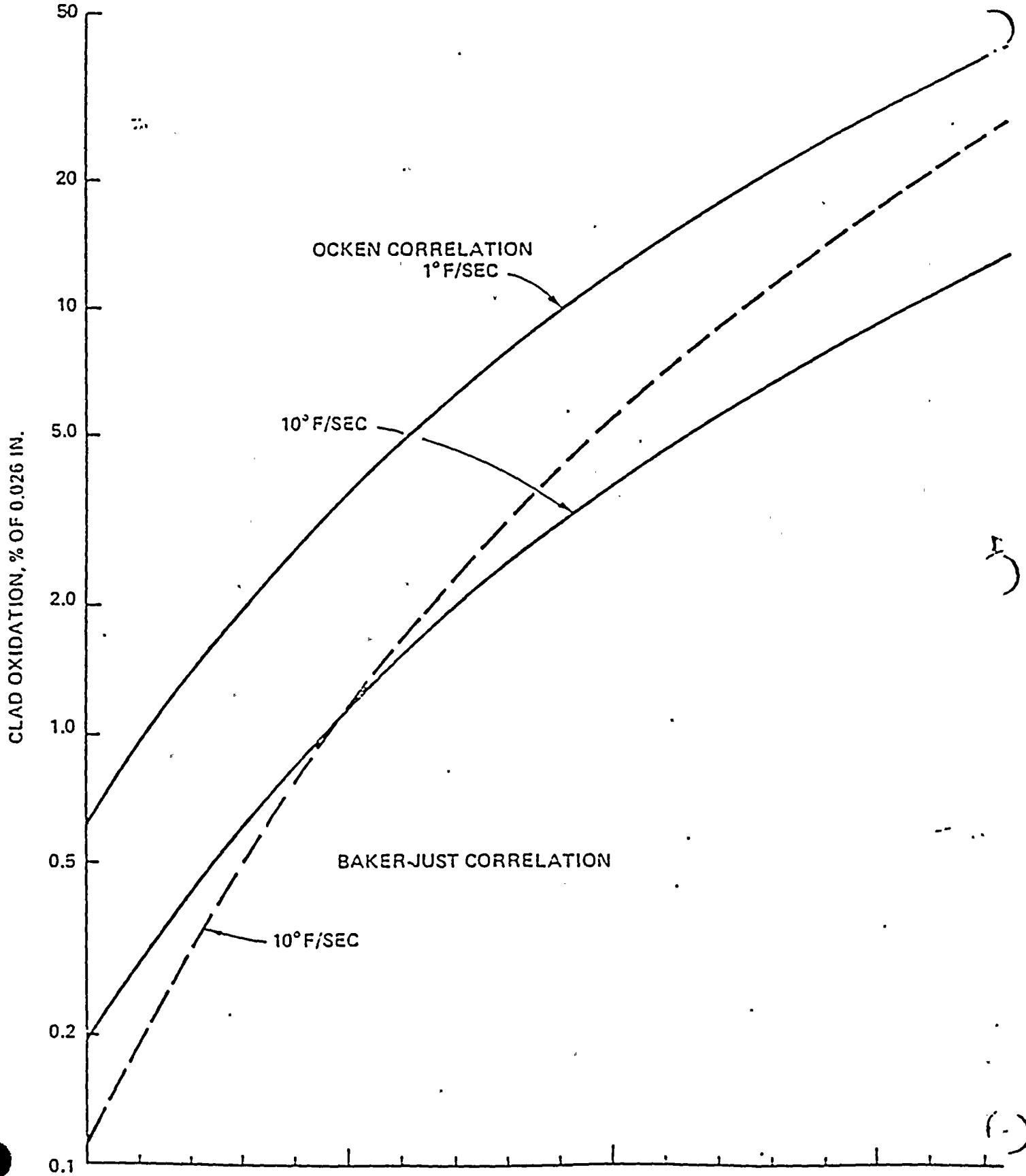
Essentially every calculation of clad oxidation performed with LOCA analyses at C-E is done using the Baker-Just correlation, which is the approved method in the licensing evaluation model.. It is based upon limited early experimental data and is deliberately biased to assure conservatively high predictions of clad oxidation at temperatures up to 2200°F. Oxidation rate data which are currently available with steady state and transient temperatures are reviewed in Reference 7-11. It is concluded that the Baker-Just equation yields substantially higher oxidation than actual at steady state, and still higher when compared with transient experimental data. Based on the steady data available, the referenced author Ocken, recommends a correlation to replace Baker-Just.

Considerable uncertainty remains, even in the Ocken correlation, because the experimental data do not represent reactor operating and accident conditions. A pressure enhancement effect, for example, is reported which increases the oxidation rate at high pressures for temperatures up to 2000°F. Also, ramp heating experiments yield lower oxidation than transient calculations which utilize correlations derived from steady state oxidation data. For PWR type transients and linear ramps, the experimentally measured oxidation may be 40% to 65% less than the equivalent calculated oxidation.

A comparison of the percent of equivalent clad thickness oxidized as a function of time is given in Figure 4-1 for the Baker-Just and the Ocken correlations. The original clad thickness is a typical value of 0.026 inch. At temperatures above about 2020°F, the Ocken correlation predicts progressively smaller oxidation. For example, when the clad temperature reaches 2500°F during a linear ramp temperature rise, Baker-Just predicts 40% more oxidation than Ocken.

Recall that the objective here is to relate the PASS measurement of the total amount of hydrogen generated throughout the course of an accident to an assessment of core damage. The amount of oxidation calculated at any instant is not so important as the relative distribution of oxidation amounts on cladding throughout the core because the total amount is available by the measurement of hydrogen. That relative distribution, for a given total amount, has less calculational uncertainty than the local value as a function of time. Also, the intent is to provide a realistic assessment to serve as a basis for emergency plan actions rather than a basis for plant design and licensing. The Baker-Just equation as discussed above is more appropriate for

FIGURE 4-1
PERCENT CLAD OXIDATION vs TEMPERATURE FOR
LINEAR RAMPS FROM 417°F



the conservative design and licensing activities. Therefore this procedure for damage assessment utilizes hydrogen measurements based upon the Ocken values of parameters in the usual equation for clad oxidation:

$$w^2 = A_{Zr}\, e^{-B/RT}\, t$$

where: w = Equivalent zirconium mass oxidized per unit area

$A_{Zr} = 3.33 \times 10^5$ (mg Zr/cm^2)^2/sec

$B = 140.6$ (kJ/mol)

T = Temperature (°k)

$R = 1.987$ cal/mol-°k

t = Time at temperature T (sec.)

The oxidation predicted by the previous equation can be expressed as a percentage of the original clad thickness. In English units the equivalent oxidation thickness is

$$x = 0.2851\, \frac{\sqrt{t}}{\Delta r}\, e^{-4696/T}$$

where: x = Percent oxidation of clad thickness

Δr = Clad thickness (ft.)

T = Temperature (°R)

t = Time at temperature (sec.)

Oxidation of Zircaloy causes embrittlement such that fracture may occur upon reflood and quench. The amount of oxidation in the above equation is expressed as the equivalent thickness of clad that would be converted to oxide if all the oxygen absorbed and reacted with the clad were converted to stoichiometric zirconium dioxide. As seen previously, the oxidation rate is strongly dependent on temperature. Test data (Reference 7-12) for specimens slowly cooled to 1520°F and then quenched, indicate the best estimate amount of oxidation to cause embrittlement is 28% for oxidation at about 2600°F and greater than 28% at lower temperatures. Below about 1800°F data indicate essentially no embrittlement for times up to 5 hours. These data are summarized as follows:

Temperature (°F)	Time at Temperature Required to Embrittle (Sec.)
1880	10,000
2060	2,000
2240	700
2420	200
2600	20

When the clad is rapidly quenched all the way from the oxidation temperature, embrittlement occurs at 20% equivalent oxidation for temperatures above 2500°F and at higher values for lower temperatures. For the analyses here, the value of 20% is selected as the equivalent oxidation required to embrittle the clad.

The NRC categories of fuel damage are characterized in Table 4-1 according to the extent of fuel clad embrittlement due to oxidation. The correlation developed by Ocken is employed to relate the equivalent mass of zirconium in the fuel rods which has been oxidized per unit surface area to the temperature of the clad surface and the time at that temperature. However, the core damage assessment procedure is based upon a measurement of the quantity of hydrogen produced during the accident. The quantity of hydrogen produced in the fuel by the oxidation of the zirconium found in the cladding and other assembly components is assumed to be stoichiometric according to the chemical reaction discussed in Section 4.1. Using this chemical reaction and the known values for the plant specific quantity of zirconium to be found in the core, the quantities of hydrogen shown in Table 4-2 are calculated as a function of the percent of total oxidation assumed. The purpose of this table is to demonstrate the significance of the quantity of hydrogen produced in degraded core accidents. The subject procedure is independent of accident scenario and therefore degraded core events are postulated in which substantial fractions of the total inventory of zirconium is oxidized. Because the reaction is assumed to be stoichiometric the quantity of hydrogen produced is linear with the percent of zirconium oxidized. Therefore a single plant specific value is employed from Table 4-2 to relate the quantity of hydrogen produced to the percent of zirconium oxidized. That value is the quantity of hydrogen produced per percent of zirconium oxidized listed in the first column of the table.

TABLE 4-2

QUANTITY OF HYDROGEN RESULTING FROM OXIDATION OF ZIRCONIUM IN THE CORE

(FT^3 AT STP)

PERCENT OF ZIRCONIUM OXIDIZED	1%	10%	50%	100%
Calvert Cliffs Units 1 & 2	4.23×10^3	4.23×10^4	2.11×10^5	4.23×10^5
Palo Verde Nuclear Generating Station	5.65×10^3	5.65×10^4	2.83×10^5	5.65×10^5
St. Lucie Unit 1	4.21×10^3	4.21×10^4	2.11×10^5	4.21×10^5
St. Lucie Unit 2	4.64×10^3	4.64×10^4	2.32×10^5	4.64×10^5
SONGS Units 2 & 3	5.07×10^3	5.07×10^4	2.53×10^5	5.07×10^5
WPPSS	5.65×10^3	5.65×10^4	2.83×10^5	5.65×10^5
Waterford Unit 3	5.04×10^3	5.04×10^4	2.52×10^5	5.04×10^5

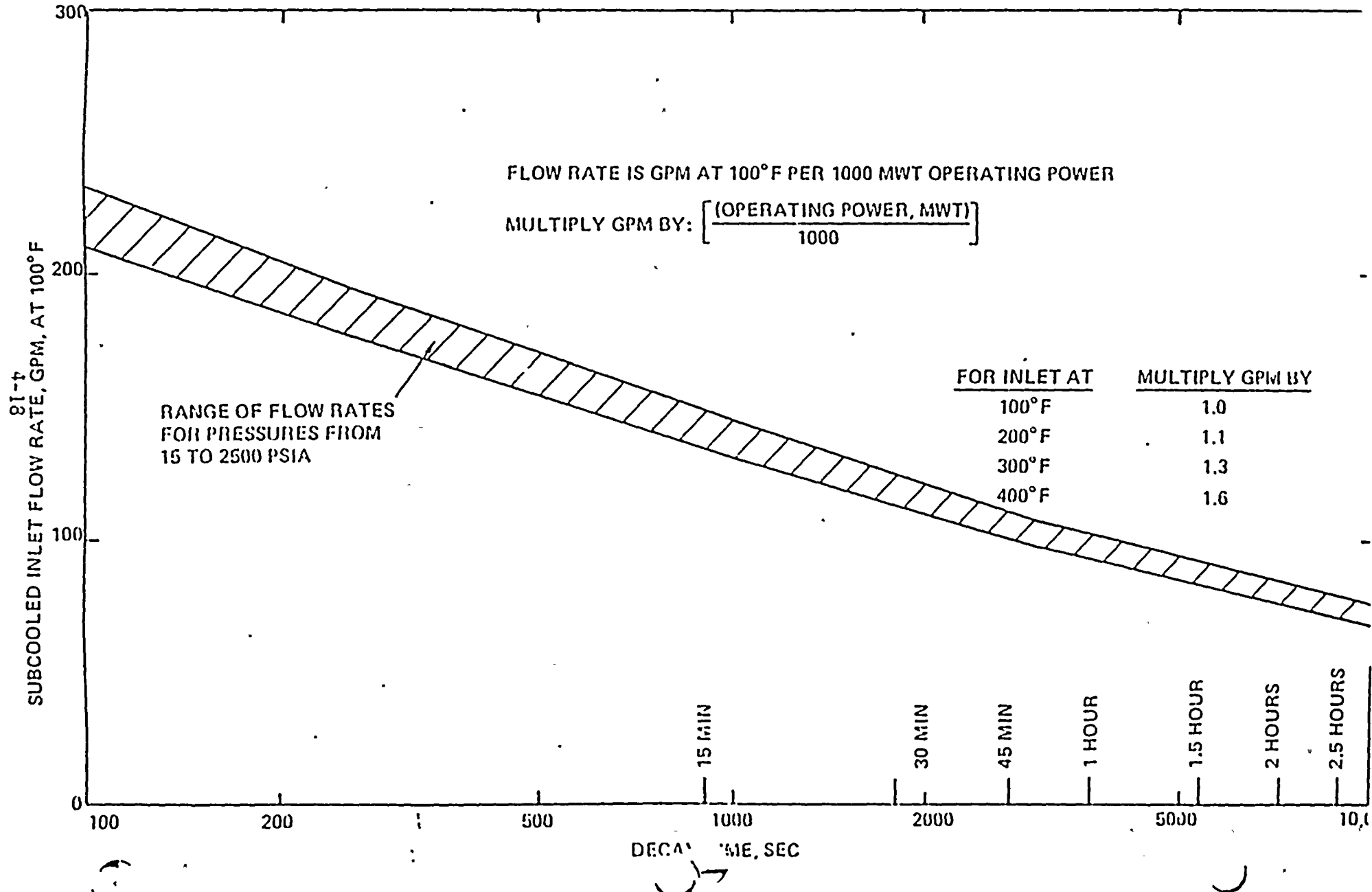
4.2.2 Analysis of Coolant Level Drop

The rate of level drop and the lowest level attained by the two-phase coolant in the core are the most significant parameters in determining the core heatup and subsequent oxidation of the fuel clad. It is shown in this section that the coolant level can be predicted as a function of time for the case of boiloff with no inlet flow. It is further shown that normalized results of this analysis are applicable to all C-E reactors.

After reactor trip, the core is adequately cooled to prevent oxidation until the level drops below the top of the active fuel. With the core just covered at steady state, the fission product decay power is equal to the sum of the power to raise any inlet water to saturation plus the power to vaporize liquid to saturated steam. The amount of core inlet flow required to maintain a covered core is dependent on the decay power level, the pressure, and the inlet temperature. Figure 4-2 gives the required flow rate to keep the core covered as a function of decay time, over a range of conditions for the C-E designed reactors.

The basis for the clad damage assessment procedure which postulates degraded core conditions assumes zero inlet cooling flow. This is equivalent to the unlikely events of inoperable ECCS and charging flow or the loss of all feedwater with a consequent high primary pressure above the HPSI shutoff head. Until the core uncovers, all decay power goes to heating and vaporizing reactor coolant. When the coolant level drops below the top of the active fuel, the portion of the active fuel below the coolant level is adequately

FIGURE 4-2
SUBCOOLED INLET FLOW RATE REQUIRED TO MAINTAIN
COOLANT LEVEL ABOVE CORE vs DECAY TIME



cooled while generating saturated steam. The fuel above the coolant level heats up and superheats the steam. Figure 4-3 is a schematic of the core and downcomer regions within the vessel showing the manometer effect as the core boils off. Steam formation in the covered portion of the core swells the volume of core coolant, producing a higher effective cooling level than would otherwise exist for the same mass of water with the vessel, and a higher level in the core than in the downcomer.

The time variation of level is obtained from a heat and mass balance on the covered length, z, of the active fuel length, L, as follows. An inlet flow is included for analytical completeness and later comparisons. A heat balance on the liquid in the core is:

Decay power below two-phase level	=	Power to vaporize liquid	+	Power to heat subcooled inlet to saturation

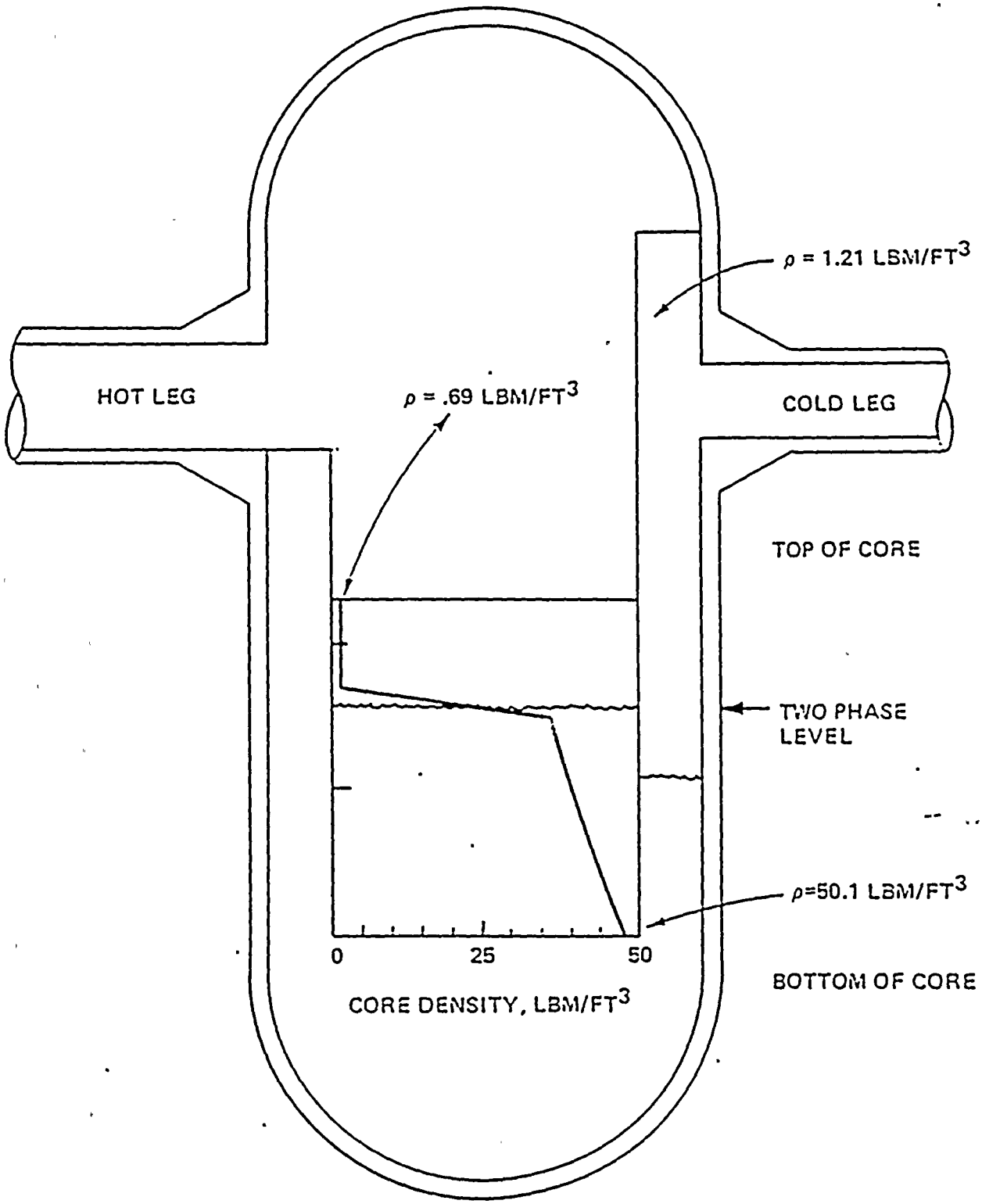
$$P_o \, DH \, \frac{z(t)}{L} = W_s(t) H_{fg} + W_{in} \, (H_f - H_{in})$$

yielding:

$$W_s(t) = \frac{P_o \, DH \, z(t)/L - W_{in} \, (H_f - H_{in})}{H_{fg}} . \qquad (4\text{-}1)$$

A mass balance on the fluid in the vessel is:

FIGURE 4-3
COOLANT DENSITY DISTRIBUTION DURING BOILOFF
(0.05 FT^2 LOCA)



Rate of change of liquid mass in vessel	=	Inlet mass flow rate	-	Steam exit flow rate

$$\rho A \frac{dz}{dt} = W_{in} - W_s(t)$$

Combining these equations yields:

$$\frac{dz}{dt} + \frac{P_o\ DH}{\rho A L h_{fg}} \cdot z = \frac{W_{in}\ (H_{fg} + H_f - H_{in})}{\rho A h_{fg}}$$

The solution for initial conditions $z(o) = L$ is:

$$\frac{z(t)}{L} = K_2\ (1 - e^{-\ t/K_1}) + e^{-\ t/K_1} \qquad (4\text{-}2)$$

where: $K_2 = \dfrac{W_{in}\ (H_{fg} + H_f - H_{in})}{P_o\ DH}$

$$K_1 = \frac{\rho A L H_{fg}}{P_o\ DH}$$

P_o = Operating power (B/hr)

DH = Decay heat fraction, assumed constant

W_{in} = Inlet flow rate (lbm/hr)

H_f = Saturated liquid enthalpy (B/lbm)

H_{fg} = Latent heat of vaporization (B/lbm)

H_{in} = Subcooled inlet enthalpy (B/lbm)

ρ = Liquid density (B/lbm)*

A = Cross sectional area of coolant in core and downcomer (ft^2)

$z(t)$ = Level in core* (ft) at time t

t = Time from start of uncovery (hr)

L = Active core height (ft)

When the inlet flow rate, W_{in}, is zero, the constant K_2 is zero. Equation (4-2) indicates that the fractional level, z/L, is then dependent on only the dimensionless ratio t/K_1. For all C-E designed reactors the value of K_1 is within ±8% of the average value. Hence, when the core uncovers by boiloff the level, as a fraction of core height, varies with time in the same way for all C-E designed reactors.

When the core is covered, the total decay power is converted to steam and the rate of level drop is fast. When the core is partially covered, only the covered length of fuel produces steam and the rate of level drop is slower. The fraction of the core uncovered (not the length in feet) at any time after the start of core uncovery is nearly the same for all the plants, at a given decay heat fraction or decay time. Hence, a longer core uncovers a longer length of fuel in a given time. This does not necessarily mean higher temperatures in the uncovered portion, since the covered portion is also longer and produces a larger steam flow rate to cool the uncovered length.

* Two-phase swell of the level in the core is neglected, resulting in a slower than actual prediction of core level drop.

Some important assumptions in the derivations are physically significant. The two-phase level within the core is uniform across the entire core, independent of fuel assembly power. This same assumption is made in the more detailed computer codes. It is equivalent to assuming some cross-flow and mixing of liquid below the two-phase level from the lower power assemblies to the higher power assemblies to accommodate their higher vaporization rate.

The axial power distribution is assumed uniform. Typical distributions with center peaks cause faster level drop for coolant levels above the elevation of the axial peak because a greater fraction of the decay power is produced below the two-phase level. Peaked distributions would cause slower level drop for coolant levels below the elevation of the peak. It is shown later, that substantial core damage occurs only after the level has dropped more than halfway in the core. Hence, for the purposes of damage assessment, the effect of axial power distribution on the coolant level is considered a second order effect.

The density of the two-phase fluid in the core is less than the density of the liquid in the downcomer. When a given amount of liquid in the core is vaporized, the core level drops more than the downcomer level. In the analysis above, these density differences are ignored. Consequently, the analyses might predict a slower than actual rate of core uncovery. The difference is dependent on the relative vapor/liquid density ratio and is therefore smaller at higher pressure. While these differences may affect the

level at any given instant, they are not considered significant in the establishment of the relationship between the amount of core damage and the amount of hydrogen generation.

Parameters for the 3400 Mwt class are used for some examples of coolant level drop as a function of time. Equation 4-2, with zero inlet flow, is plotted in Figure 4-4, at 1200 psia and for three values of decay power. For 1% decay power (about 2 hours decay time) and with no inlet flow, the 3400 Mwt class reactor uncovers 50% in 13 minutes. At 2% decay power (about 23 min. decay time) it takes half as long for 50% uncovery. Figure 4-5 shows the level vs. time at 1% decay power for three values of pressure. At 300 psia it takes 21 min. to uncover to 50% and at 2500 psia it takes 6 min. These results are typical for all C-E designed reactors within a time scale variation of +12% to -4%, which is the range of variation of the constant, K_1, from the value for the 3400 Mwt class. This range of error is considered small enough to permit definition of the extent of core damage within the ten categories defined in Table 4-1.

4.2.3 Core Heatup Analyses

The objective of the core heatup analyses is to predict the distribution of clad temperature and clad oxidation during an event which uncovers the core. The distribution of clad temperature is used to establish a relationship between the maximum core exit thermocouple temperature and the minimum number of rods which have ruptured due to gas gap overpressurization. This relationship will be used in Section 5 as a basis for the use of core exit

FIGURE 4-4
COOLANT LEVEL vs TIME DURING BOILOFF
3400 MWT SERIES, 1200 PSIA

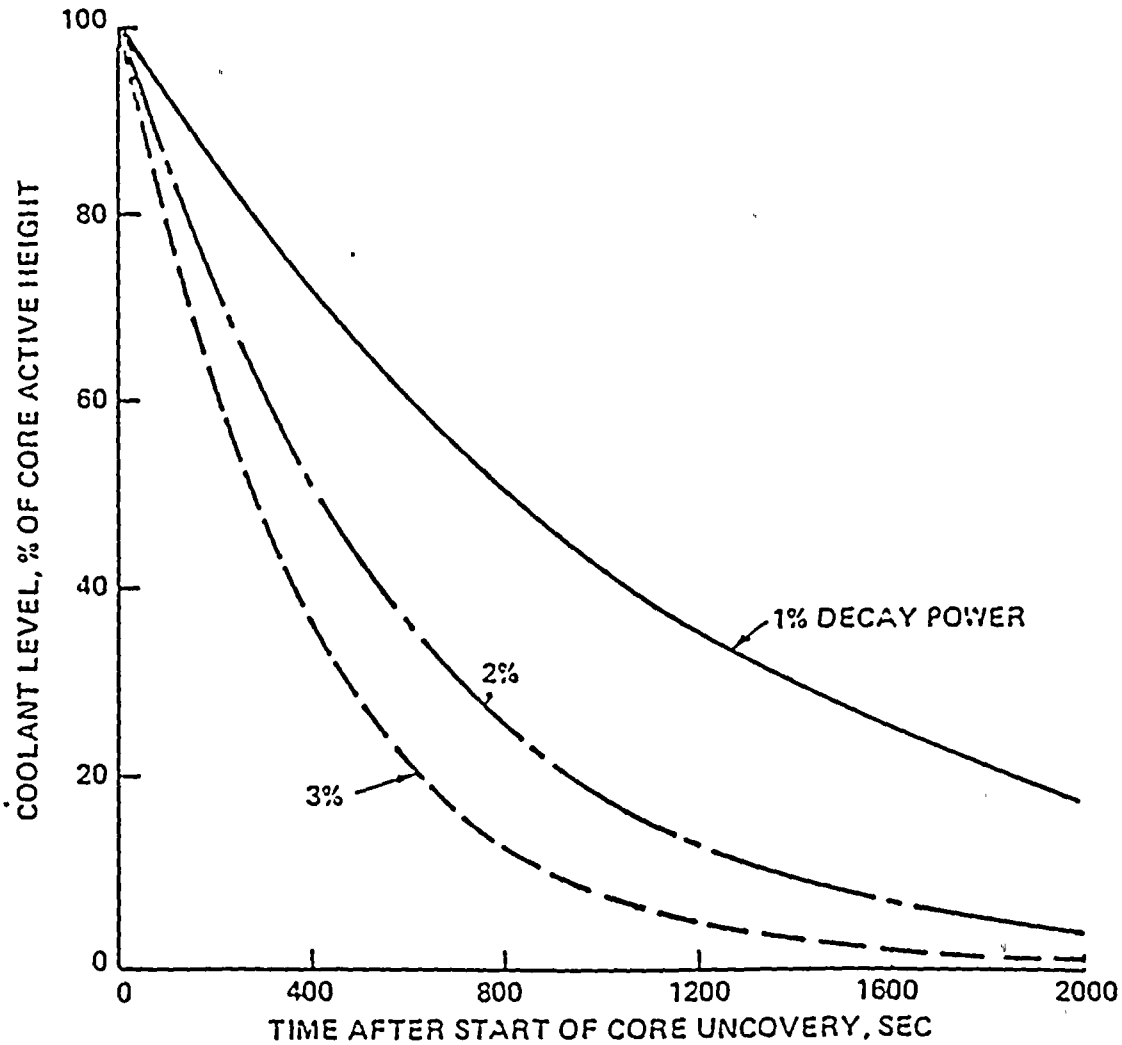
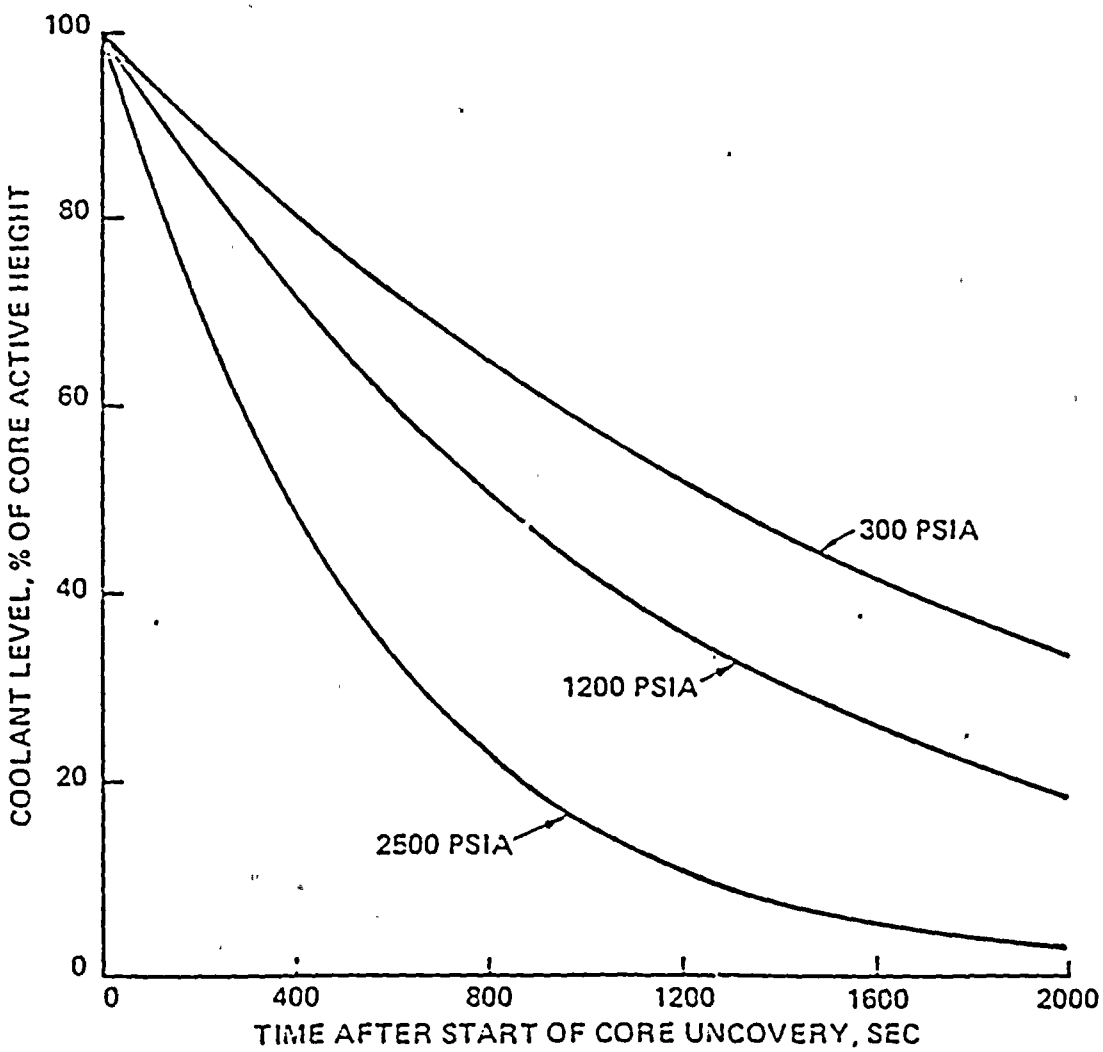


FIGURE 4-5
COOLANT LEVEL vs TIME DURING BOILOFF
3400 MWT SERIES, 1% DECAY POWER



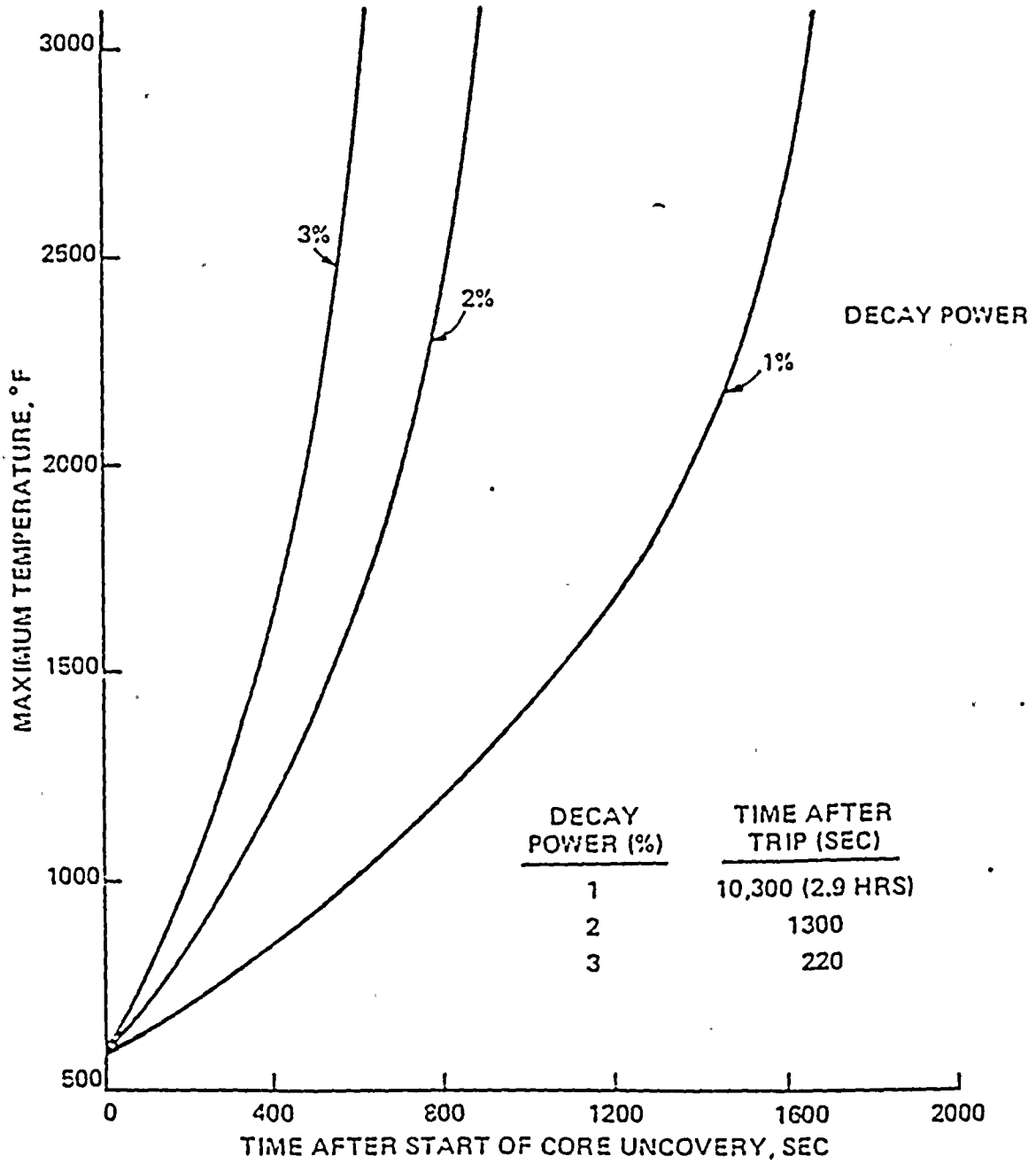
thermocouple data to assess core damage. The clad oxidation distribution is used to establish a relationship between the measurement of hydrogen generated by the oxidation and the minimum number of rods which have oxidized beyond the embrittlement threshold. Fuel rod structural integrity has been related to oxidation embrittlement through the discussion in Section 4.2.1.

An analytical derivation is used which includes the dominant physical phenomena to support the objective. This derivation is provided in Appendix B.2. Employment of an analytical derivation rather than a numerical computer code solution enables a convenient comparison of the significant parameters among all the C-E reactor designs. Detailed analyses on each different core configuration are shown to be unnecessary within the required accuracy of the overall PASS damage assessment procedures. As an overcheck on the analytical solutions, some analyses are done using the MAAP computer code. Here are some analytical results from the derivation in Appendix B.2.

The first result is that decay power determines the rate of coolant level drop when uncovery occurs by boiloff. Higher power causes faster level drop, as shown previously by Figure 4-4.

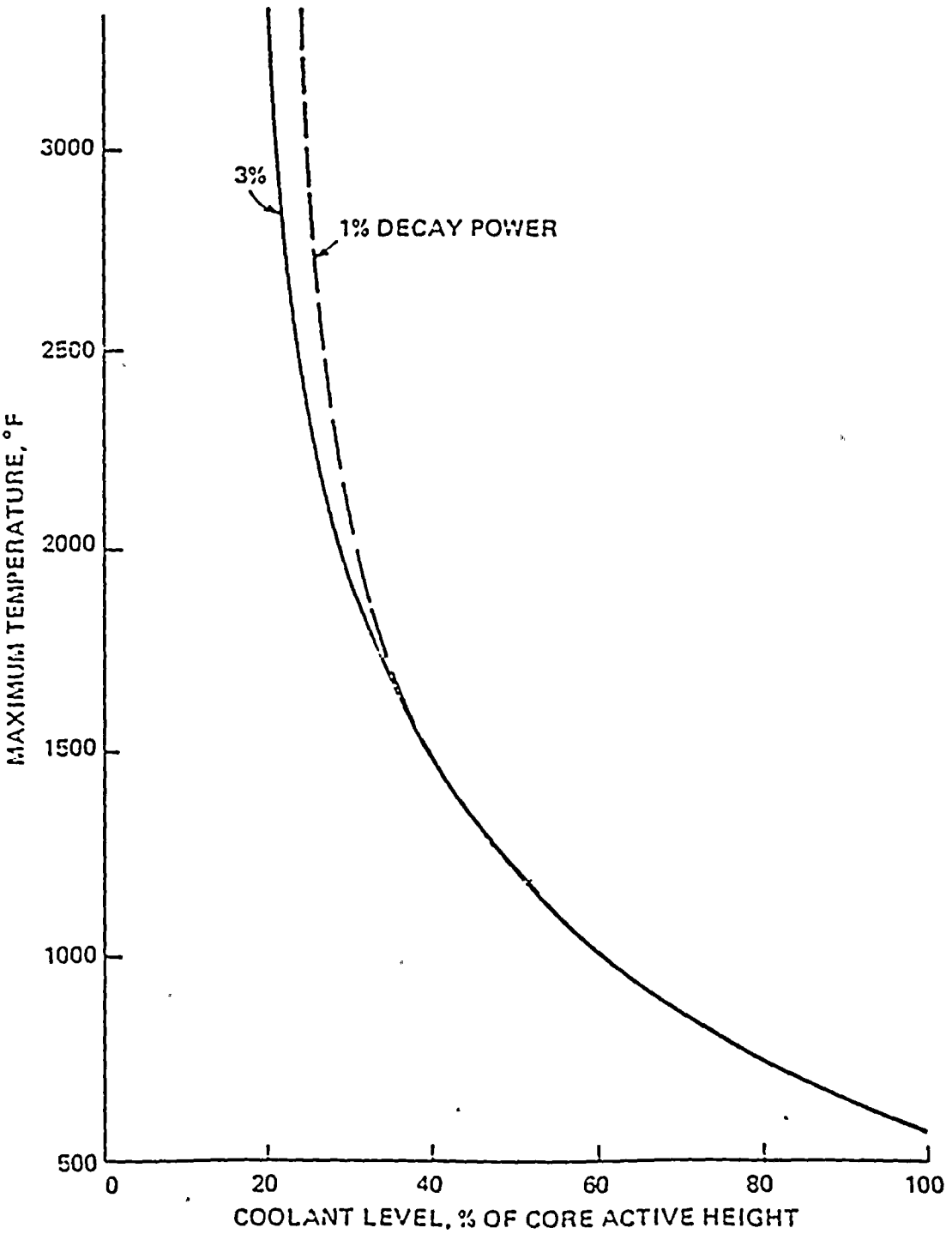
The second result is that decay power determines the rate of temperature rise with time. Figure 4-6 shows the peak clad temperature as a function of time after uncovery starts for three values of decay power. However, at a given level, the temperatures are almost the same for a range of 1% to 3% decay power. This is shown in Figure 4-7. The fact that temperatures at the lower

FIGURE 4-6
MAXIMUM CORE TEMPERATURE vs TIME AFTER
START OF CORE UNCOVERY DURING BOILOFF
3400 MWT SERIES, 1200 PSIA



DECAY POWER (%)	TIME AFTER TRIP (SEC)
1	10,300 (2.9 HRS)
2	1300
3	220

FIGURE 4-7
MAXIMUM CORE TEMPERATURE vs
COOLANT LEVEL DURING BOILOFF
3400 MWT SERIES, 1200 PSIA



power are slightly higher is probably caused by oxidation heating. Greater total reaction heat is added to the fuel rod during the longer period of time required to attain a given coolant level when the power is lower.

The conclusion is that coolant level leaves less uncertainty than time in determining the clad temperature for a given boiloff scenario. This conclusion might differ if other factors (such as HPSI flow) cause the rate of level drop to be less dependent on core power. A second conclusion is that without inlet cooling flow, the time after uncovery to reach high temperature is only minutes or a few tens of minutes. This second conclusion was also made evident in previous studies for the CEOG on the adequacy of the core exit thermocouples to provide an advance warning of the approach to inadequate core cooling (Reference 7-5). During typical small break LOCA events it was shown that the time interval is short from the first occurrence of steam superheat until the clad ruptured or exceeded 2200°F.

The amount of local oxidation is dependent on the magnitude of temperature and the length of time at temperature. For core uncovery by boiloff, Figure 4-8 shows the local clad oxidation as a function of time after uncovery starts, for three values of decay power. The oxidation rate is slow initially, until the temperature rises above about 1800°F. Thereafter, for a boiloff event, the rate of oxidation increases extremely fast. Within a few minutes, local oxidation increases from a few percent to well beyond the embrittlement threshold. Figure 4-9 shows that this rapid increase in oxidation occurs when the coolant level has dropped to about 25%.

FIGURE 4-8
MAXIMUM LOCAL CLAD OXIDATION vs
TIME AFTER START OF CORE UNCOVERY
DURING BOILOFF
3400 MWT SERIES, 1200 PSIA

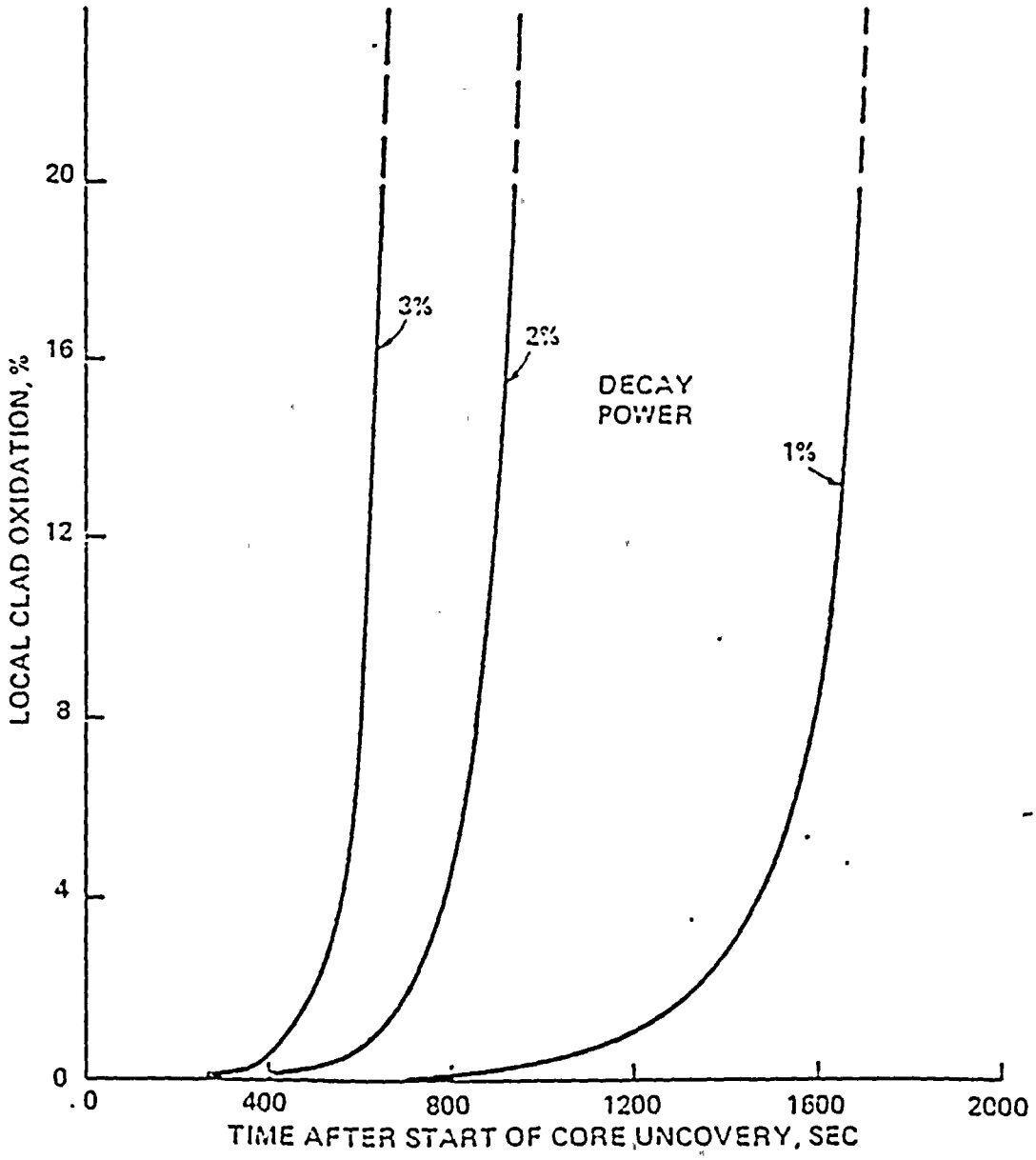
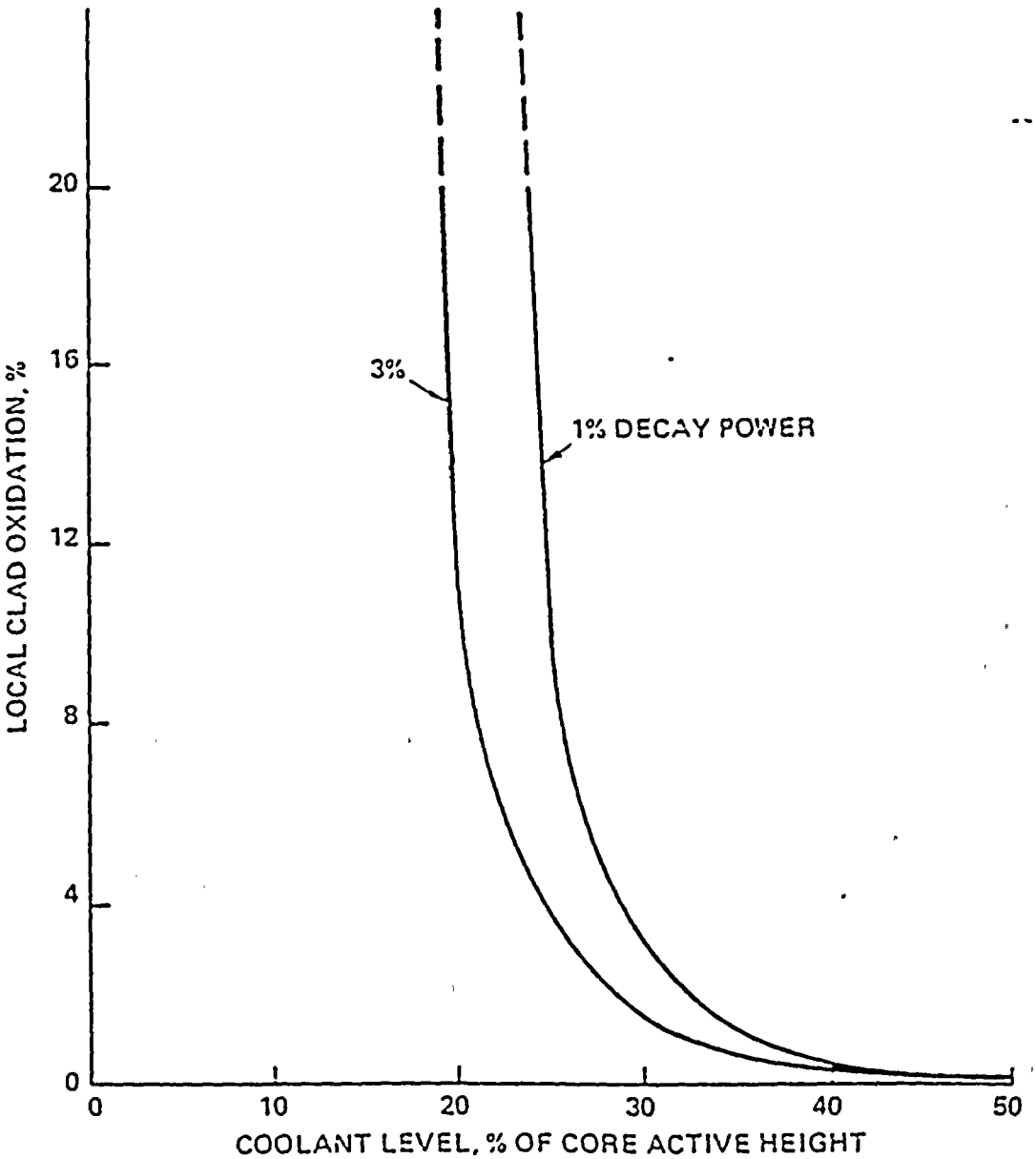


FIGURE 4-9
MAXIMUM LOCAL CLAD OXIDATION vs
COOLANT LEVEL DURING BOILOFF
3400 MWT SERIES, 1200 PSIA



These calculations assume sufficient steam production below the coolant level to oxidize the upper portion of the fuel clad. More comprehensive calculations suggest that the steam may be completely consumed along the lower length of exposed clad thereby limiting the oxidation along the top of the rod. Fuel will then heatup by decay power alone, with the clad and, later, the fuel being destroyed by melting. It has been reported (Reference 7-14) that for uncovery by boiloff the oxidation embrittlement of the clad will have already occurred prior to the buildup of hydrogen sufficient to limit the oxidation rate. Therefore, these calculations are adequate for predicting the fraction of fuel rods which have attained the local clad embrittlement threshold but not for predicting the axial distribution or extent of oxidation along the length of a rod.

Axial flow of steam and hydrogen tends to support the usual assumptions of a channel calculation which ignores coolant mass interchange among adjacent channels. Therefore, the limiting effects of steam consumption and hydrogen generation on clad oxidation in a high power channel would not significantly affect the oxidation in adjacent channels with lower radial peaks. This validates the calculation of the radial distribution of the rods in the core which may at least fragment upon quenching or later handling. A prediction of the total damage configuration requires additional modeling.

4.2.4 Effects of Radial Power Distribution

In the previous section it was discussed that decay power determined the rates of coolant level drop and peak clad temperature rise. The magnitude of temperature and the time at temperature determines the amount of local

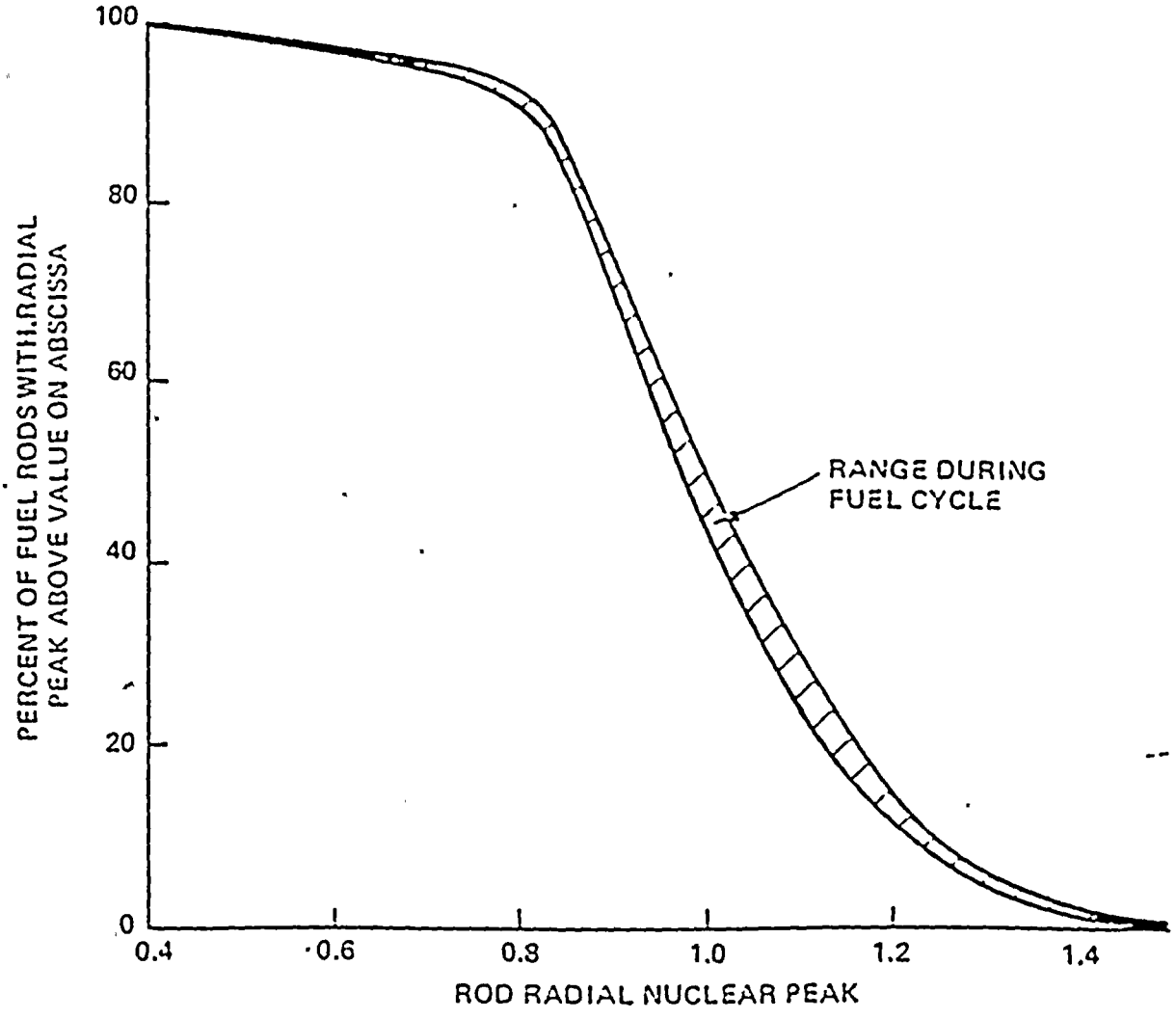
oxidation. This section provides the bases for the relationship between the radial power distribution and the amount of local oxidation. The relationship is given for several values of decay power and several values of pressure.

During an initial core uncovery and prior to substantial core structural damage, the coolant level is uniform across the core. Steam flow rate tends to be higher in higher power regions or channels, thereby tending to reduce the dependence of temperature on the radial power distribution. As the coolant level drops and the heat of reaction increases, convection removes a smaller fraction of power and the rods with higher radial peaks increase faster in temperature. At any instant there is a distribution in rod temperatures across the core above the coolant level. If reflood and core cooling should be accomplished at that time, there will result a radial distribution of fuel rod ruptures and clad embrittlement after core recovery.

A typical distribution of radial peaking factors is selected for steady power operation without CEA insertion. Figure 4-10 shows the cumulative fraction of fuel rods in the core with radial peaks above the value given on the abscissa. The band of the curve encompasses the variations, from burnup only, throughout a fuel cycle of length 14,000 MWD/T for the sixth cycle of a 2700 Mwt core. Five calculational intervals of radial peaks are selected with corresponding percentages of the core as follows:

FIGURE 4-10

DISTRIBUTION OF ROD RADIAL NUCLEAR PEAKS



Radial Peak Interval	Percentage of Core	Calculation Peak
>1.3	5	1.4
1.1 - 1.3	22	1.2
0.9 - 1.1	46	1.0
0.7 - 0.9	22	0.8
<0.7	5	0.6

This distribution is considered a best estimate for reactor conditions which would exist most of the time. It is adequate for generic calculations which support the procedure for damage assessment and which are necessarily performed prior to the occurrence of an accident.

The peak clad temperature as a function of radial peak during boiloff is plotted in Figure 4-11 for various coolant levels or times. At time zero the core is covered and the clad temperature is essentially uniform at saturation temperature. As the core uncovers, the temperature of rods with higher radial peaks increases faster than the temperature of lower peak rods. For example, when the core is half covered, the temperature is 1175°F on rods with a radial peak of 1.4 and is 960°F on rods with a peak of 0.6. These temperatures would increase proportionally on all rods, if a non-uniform axial distribution were used.

The same calculations yield the local percentage oxidation of clad thickness as a function of radial peak for various levels. Figure 4-12 shows that the maximum local oxidation on rods with 1.4 radial peak is 2% when the coolant level reaches 30%. On rods with a radial peak of 0.6, the local oxidation is

FIGURE 4-11
MAXIMUM ROD TEMPERATURE DURING SOILOFF vs
RADIAL NUCLEAR PEAK
3400 MWT SERIES; 1200 PSIA; 2% DECAY POWER

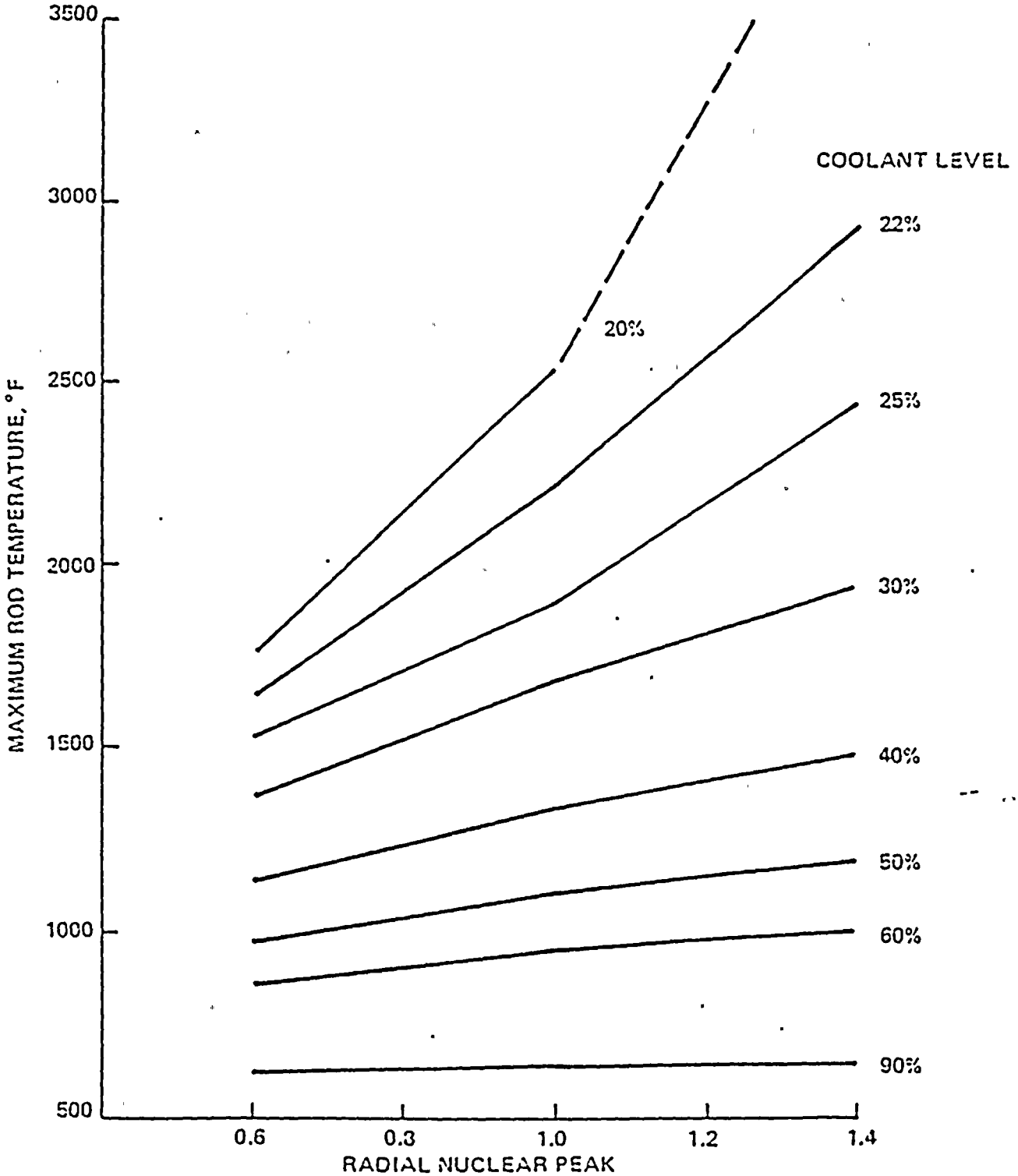
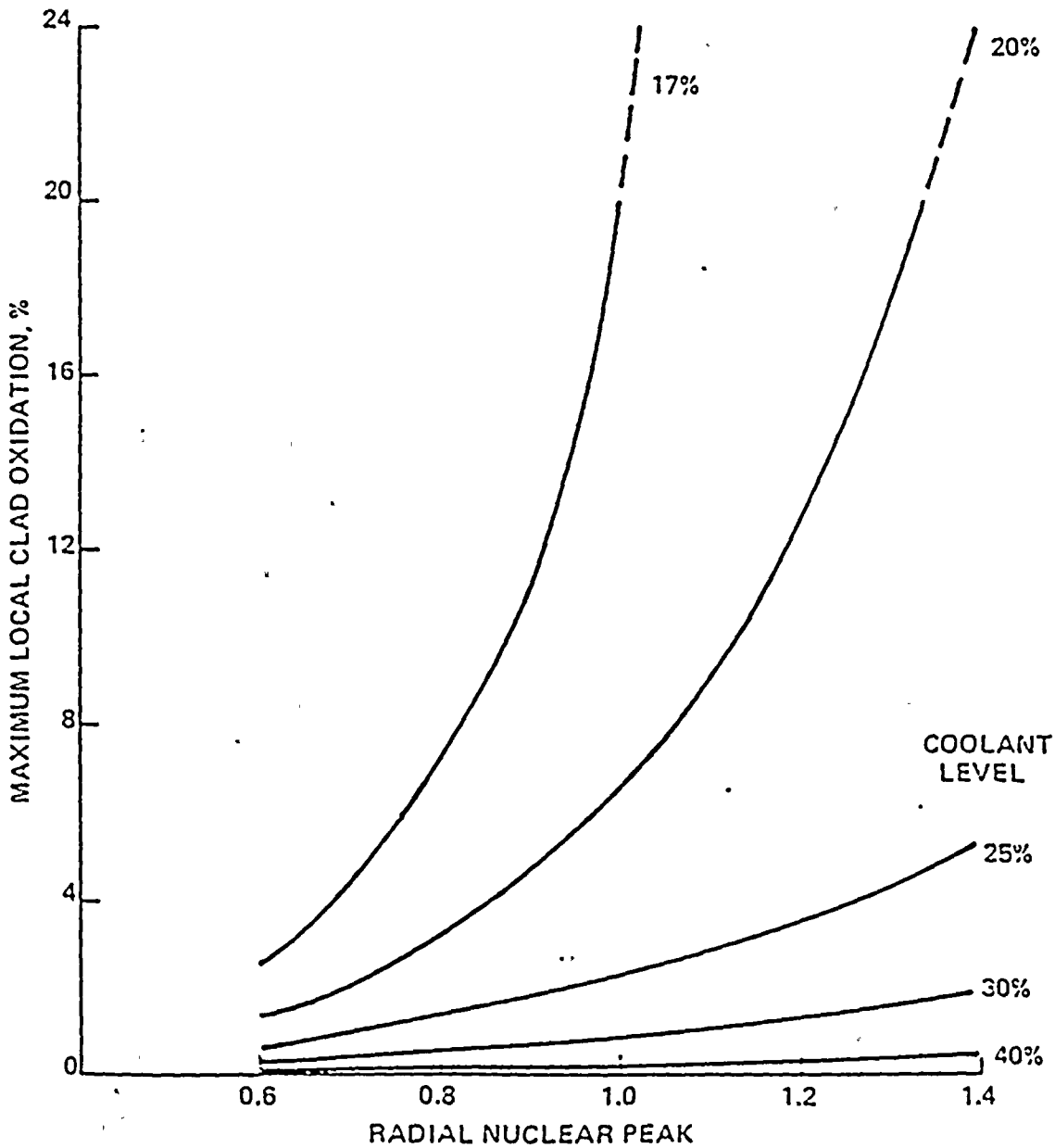


FIGURE 4-12
MAXIMUM LOCAL CLAD OXIDATION DURING
BOILOFF vs RADIAL NUCLEAR PEAK
3400 MWT SERIES; 1200 PSIA; 2% DECAY POWER



only 1/4%. At a coolant level of 20%, the local oxidation ranges from 24% on high peak rods to about 1 1/2% on low peak rods. There is a wide variation in oxidation at any instant and therefore in the potential for clad embrittlement. By comparison with Figure 4-4, there is only 4 minutes difference between these two coolant levels.

The conclusion is that once oxidation gets going, it proceeds rapidly, and at any instant there are large variations in the maximum local oxidation on the fuel rods in the core.

The total hydrogen released from the core is determined at each instant by summing the local oxidation along the exposed clad length for all the radial peaks. At the same instant, the number of rods which have local oxidation greater than the embrittlement threshold is determined. The results are given on Figure 4-13, as the percent of the number of rods in the core which have at least 20% local oxidation as a function of the total percent oxidation of all the clad in the active core length. Calculations are made for three values of decay power at 1200 psia. The figure indicates a relatively large increase in the number of rods embrittled compared to the increase in total core oxidation. The coarseness of the calculated valves limits the detailed conclusions which can be made. However, it can be concluded that a large fraction of the rods may be embrittled when a relatively small fraction of the zirconium in the core clad is oxidized.

Figure 4-14 shows, with the same coordinates, the variations in embrittlement and core oxidation when the pressure varies from 300 to 2500 psia. The

FIGURE 4-13
PERCENT OF RODS EMBRITTLED vs CORE OXIDATION
DURING BOILOFF
3400 MWT SERIES, 1200 PSIA

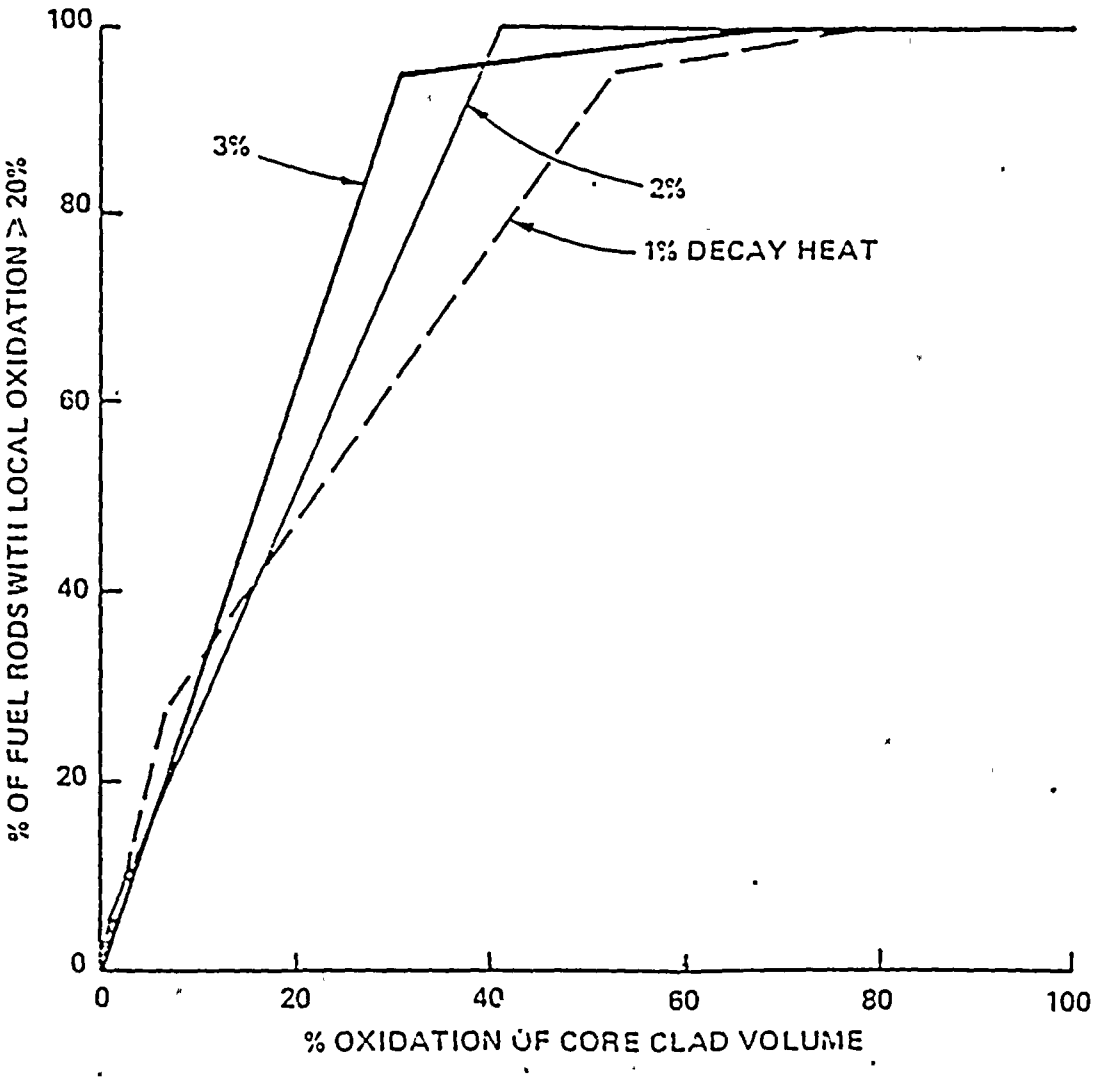
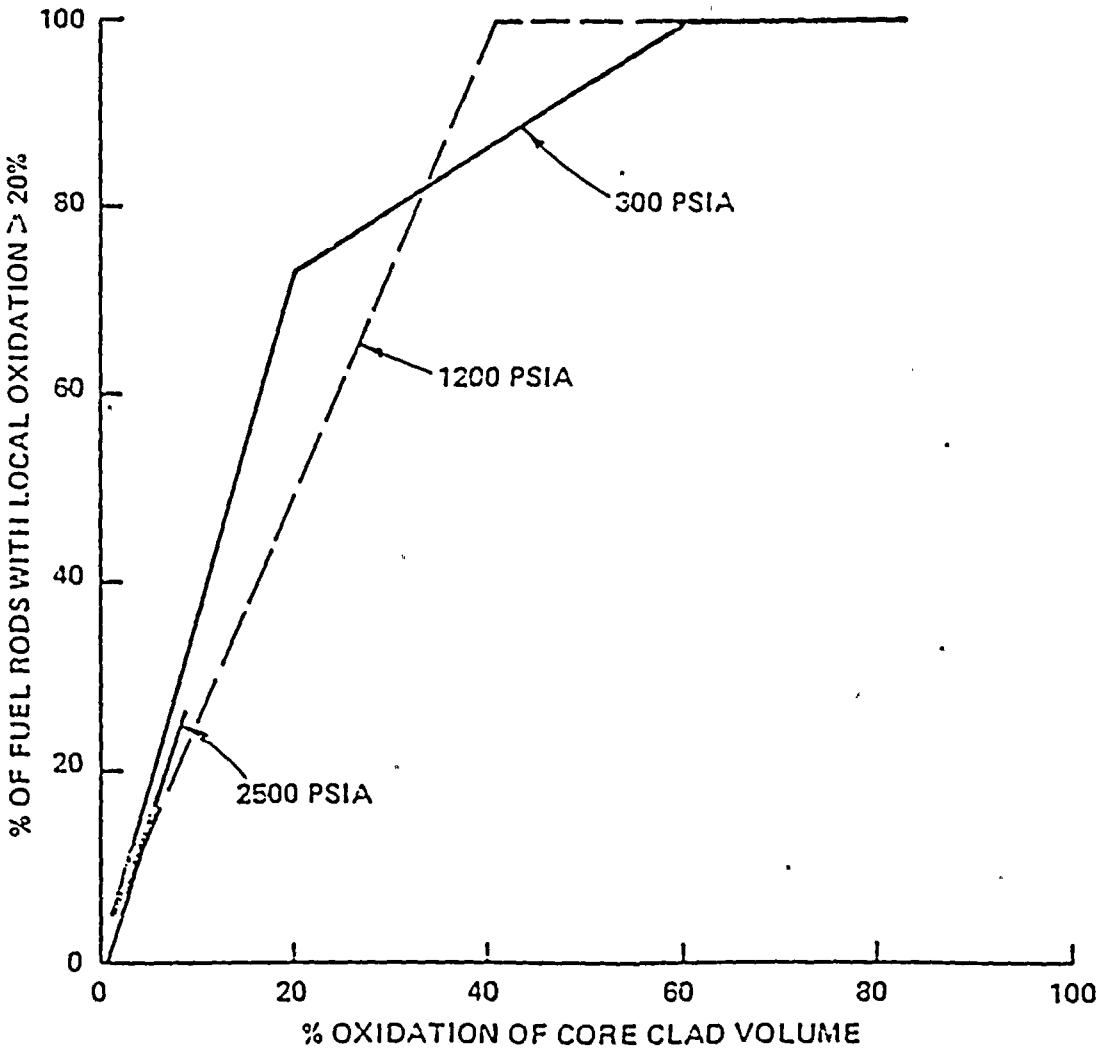


FIGURE 4-14
PERCENT OF RODS EMBRITTLED vs CORE OXIDATION
DURING BOILOFF
DECAY HEAT = 2%



results are similar to those for variations in decay power, in that a relatively small band encompasses a wide parameter range.

All the results obtained for the percent of the rods embrittled as a function of percent total core oxidation are plotted on Figure 4-15. This figure includes the range of 1% to 3% decay heat and the pressure range from 300 to 2500 psia. Given a PASS measurement of the amount of hydrogen released from the core, expressed as a percentage of the core clad volume which is oxidized to produce it, Figure 4-15 is utilized to estimate the percent of the fuel rods which might fragment upon quench and/or later handling. This figure is included in the procedure for damage assessment and is considered applicable to all C-E designed reactors.

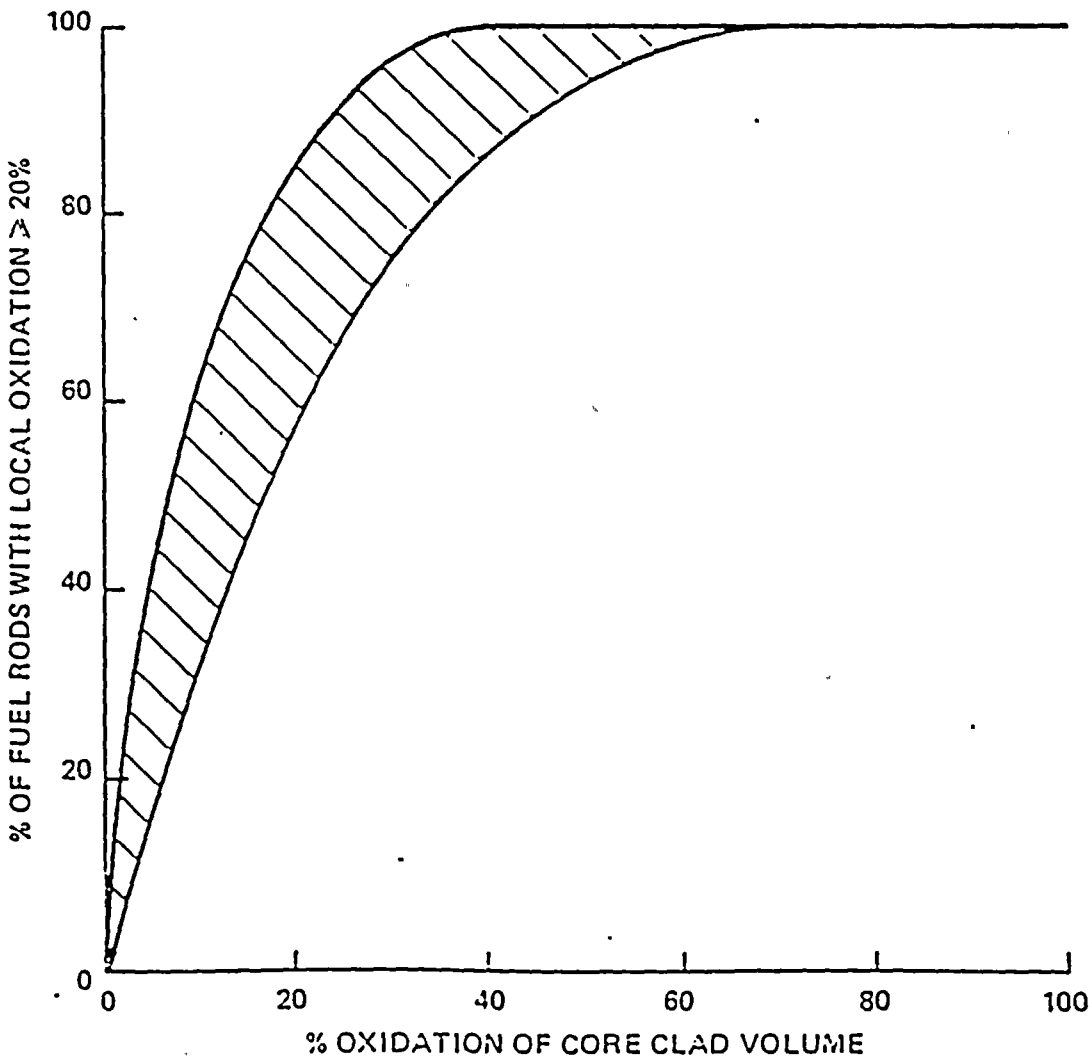
4.3 PREDICTION OF FUEL CLAD RUPTURE BASED ON HYDROGEN PRODUCTION

Previously, hydrogen production was related to clad temperature and to the radial distribution of clad temperatures in the core. In this section, the criteria are developed which relate clad temperature to the occurrence of clad rupture by overpressurization of the gas in the rod. Then the number of ruptured rods is related to the amount of hydrogen produced. Thus, the measurement of the total hydrogen produced may be used to infer the number of ruptured rods.

4.3.1 Clad Rupture Criteria

Fuel clad will balloon and rupture when the internal gas pressure is sufficiently greater than the external coolant pressure. Clad temperature and

FIGURE 4-15
% OF THE FUEL RODS WITH OXIDATION EMBRITTLEMENT vs
TOTAL CORE OXIDATION
FOR 1% TO 3% DECAY HEAT AND 300 PSIA TO 2500 PSIA
WHEN COOLANT LEVEL DROPS BY BOILOFF WITH
NO INLET FLOW UNTIL CORE IS RAPIDLY QUENCHED



duration at temperature are significant parameters in determining the rupture pressure differential. Normal values of these parameters and values expected during typical core uncovery events are discussed here. The temperature which causes clad failure during such events is determined. That rupture temperature is used in the prediction of the number of ruptured rods by this procedure.

C-E fuel is prepressurized, at room temperature, to 380 psig with helium. Increasing the temperature to normal operating conditions increases the internal gas pressure to about 800 psia. The normal external coolant pressure is about 1400 psi higher than the minimum internal pressure. Accumulation of fission gas increases the internal pressure, but does not cause it to exceed coolant pressure at the end of fuel life. The fuel does not rupture during a depressurization transient at normal temperatures.

Typical calculations for Design Basis small break LOCA events yield reactor coolant pressures below the secondary pressure when the core uncovers. Secondary pressures may range from about 850 to about 1100 psia depending on the plant. When the fission gas pressure is added to the helium gas pressure at elevated accident temperature, the internal pressure exceeds reactor coolant pressure. Whether or not the clad ruptures depends on the particular fuel rod burnup, on the event scenario and on the clad material properties.

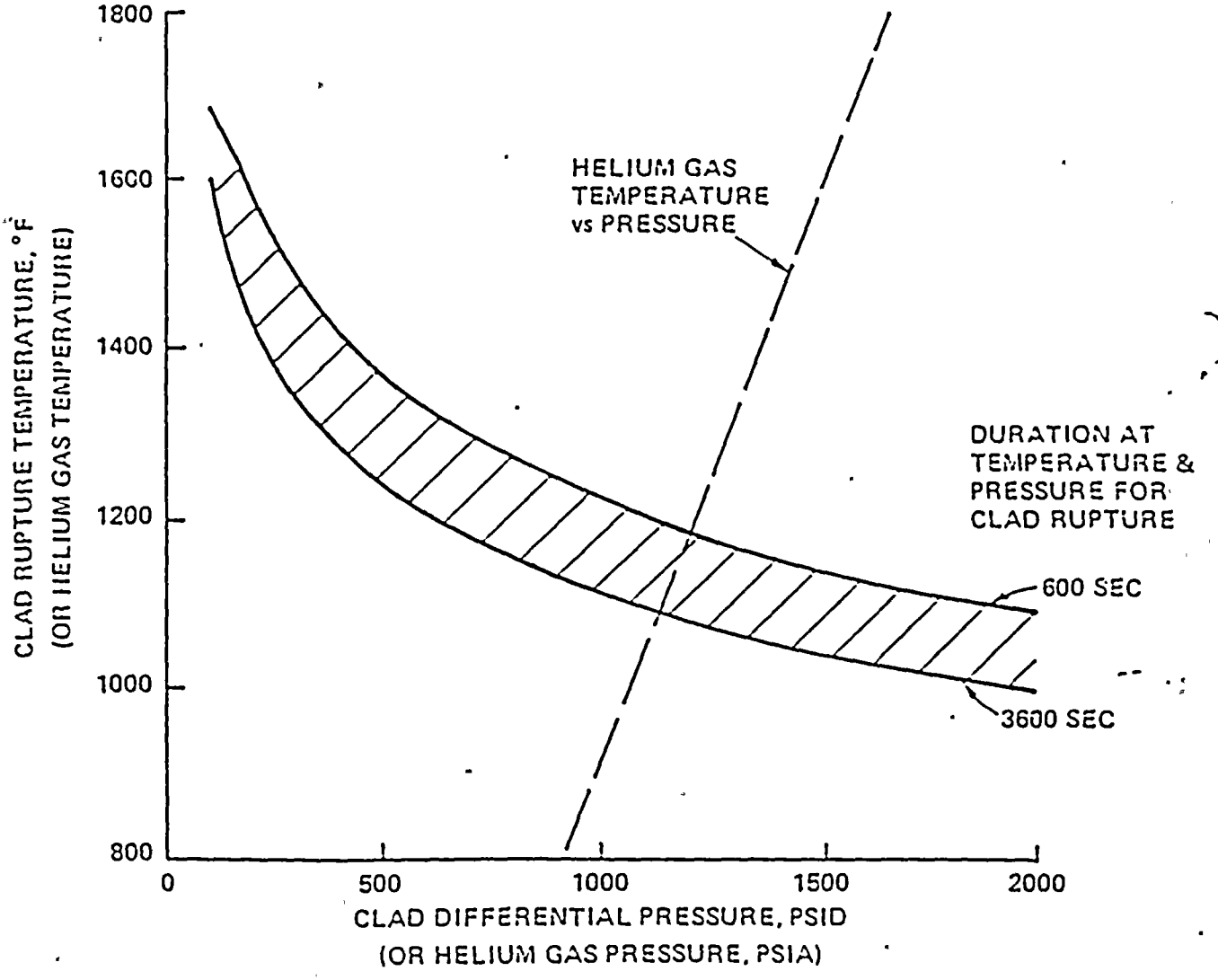
A survey of these factors and how they combine to determine if fuel ruptures is provided by the CEOG sponsored effort on Inadequate Core Cooling Instrumentation and appears in CEN-158 (Reference 7-5). Figure 4-16

summarizes those conclusions. It shows the local clad temperature at which rupture will occur as a function of the clad differential pressure for a range of the duration at temperature from 600 to 3600 sec. It also shows the temperature as a function of internal gas pressure for new fuel containing only helium fill gas. For example, at 1500°F the internal pressure is 1450 psia in new fuel and increases with fuel burnup. When the coolant pressure is 1100 psia or less the clad differential pressure is at least 350 psid. Figure 4-16 shows that clad rupture will occur in less than 600 sec.

Core uncovery is also predicted for complete loss of heat sink events where the external coolant pressure is higher than the internal gas pressure. Coolant pressure is governed by the primary safety valve setting which exceeds normal operating pressure. Clad rupture may occur later in uncovery or might occur by brittle fracture upon clad stress reversal when reflood and system depressurization occurs.

The conclusion is that for the most general uncovery events clad rupture will occur in the range of clad temperature from 1200°F to 1800°F. Three temperatures, 1200°F, 1500°F and 1800°F are selected for later use in evaluating the number of ruptured rods corresponding to measurements of hydrogen and core exit temperature. This range of rupture temperature is one source of inaccuracy in the use of the subject procedure. However, the results are adequate to locate the the extent of core damage within the defined categories characterized in Table 4-1.

FIGURE 4-16
CLAD RUPTURE TEMPERATURE vs CLAD
DIFFERENTIAL PRESSURE, AND HELIUM FILL
GAS TEMPERATURE vs PRESSURE
(2700 MWT SERIES)

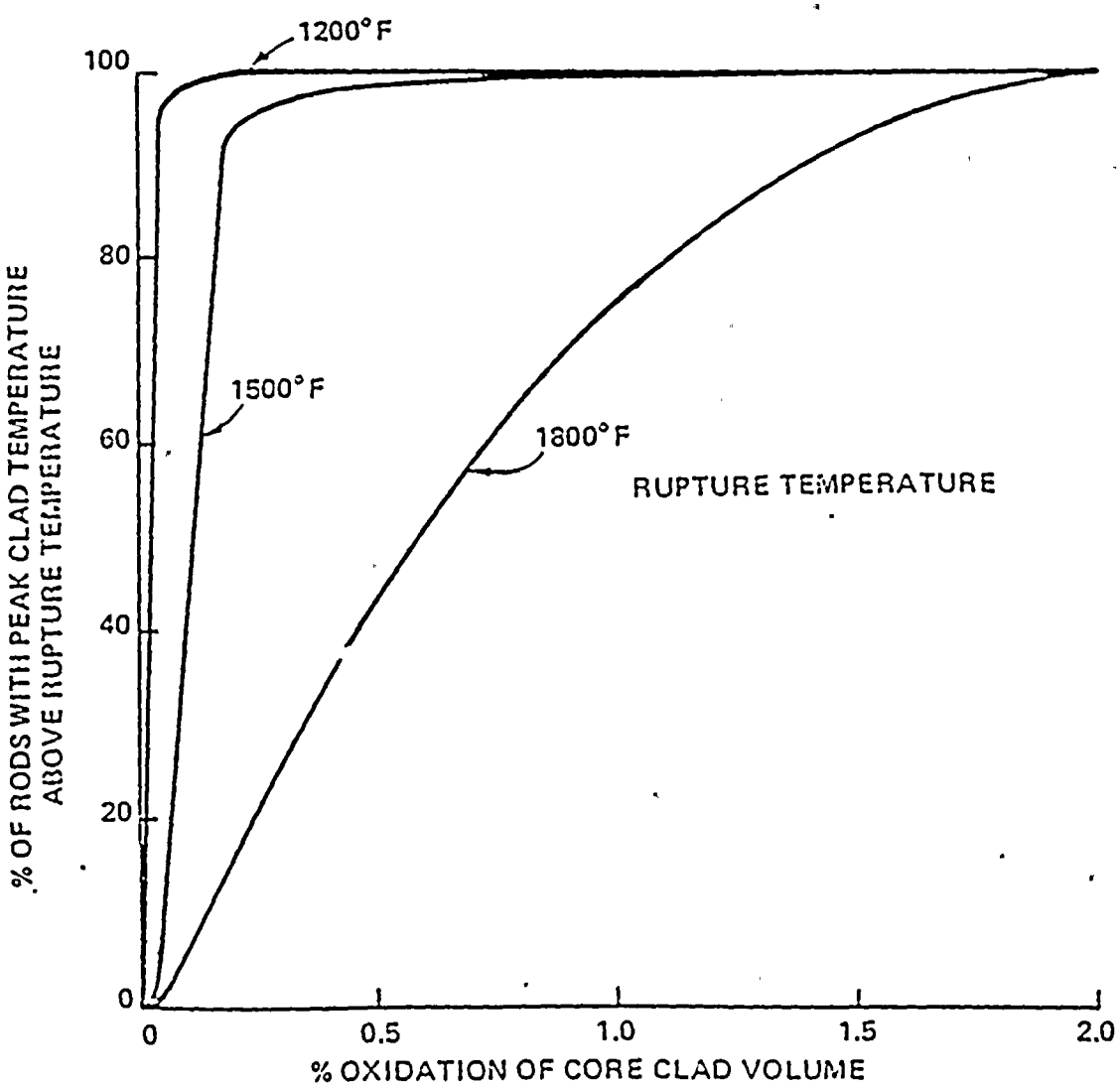


4.3.2 Effects of Radial Power Distribution

Clad rupture occurs at about 1500°F ± 300°F for core boiloff events. Rupture occurs earlier than the occurrence of 20% local clad oxidation. A relation is made between the number of rods which reach the rupture temperature as a function of the percent of the total core clad zirconium which is oxidized at any instant. Even though there is a wide variation in the rupture temperature with time, burnup, clad pressure differential etc., the uncertainty in the resultant relationship is probably not significant. The temperature rise on a rod is relatively fast compared to the total core wide oxidation so the temperature rapidly rises through the range of rupture temperatures. The time at which this occurs varies with radial peak. Figure 4-11 yields the radial peak for which the clad temperature exceeds 1500°F, at several times during core uncovery. This is combined with the core distribution of radial peaks in Figure 4-10 and with the percent of the core clad oxidized at each time. Results are plotted in Figure 4-17 as the percent of the total number of fuel rods which are ruptured vs. the percent of the core clad zirconium oxidized.

The earliest possible indication of clad rupture from a measurement of hydrogen depends on the sensitivity of the measurement. Typically, the minimum measurable concentration in the containment atmosphere is 0.1% by volume. This concentration is equivalent to a total amount of hydrogen which is produced by oxidation of about 0.5% of the core clad zirconium. Figure 4-17 shows that by the time 0.5% of the core clad is oxidized during boiloff, between 40% and 100% of the rods are ruptured, depending on whether the rupture temperature is 1200°F or 1200°F respectively.

FIGURE 4-17
PERCENT OF FUEL RODS ABOVE CLAD RUPTURE
TEMPERATURE vs CORE CLAD OXIDATION
3400 MWT SERIES, 1200 PSIA, 2% DECAY POWER



The conclusion is that hydrogen is not a sensitive parameter for assessment of small amounts of clad rupture. In fact the opposite is true. If any indication of hydrogen is obtained from the containment atmosphere which is attributable to core oxidation during a boiloff event, then a large percentage of the fuel rods are probably ruptured. Hydrogen measurement would be a backup to the more sensitive indication of radiation from the fission gas release from the ruptured fuel.

4.4 CONFIRMATION OF ANALYTICAL PREDICTION

In this section two methods are used to verify the previously stated conclusion that the analytical results from the boiloff analyses yield lower limits estimates of clad damage for all scenarios and that the simplified analyses are adequate for this damage assessment procedure. Analyses of slow uncovery with inlet flow are presented in Section 4.4.1 and are compared to the previous analyses of uncovery without inlet flow.

Analyses with a state-of-the-art computer code are presented in Section 4.4.2. Results are given for core uncovery by blowdown and by boiloff. The rapid blowdown results support the conclusion that the procedure yields low limit estimates of clad damage for such accident scenarios. The boiloff computer results are compared to previous analytical results to verify the applicability of the simplified analyses.

4.4.1 Comparisons to Predictions with Slow Uncovery

This section provides a comparison between the cases of boiloff with and without inlet flow. Because of the potential variability of inlet flow during an accident, it is necessary to know how the damage prediction is affected. Inlet flow causes a slower rate of uncovery. The rate is slower when liquid inlet flow replaces some of the steam flow from the core. Uncovery proceeds until a stable coolant level occurs for which the inlet mass flow rate is equal to the steam flow rate.

The height of the stable level is available from the previous derivations in Section 4.2.2. The level will rise as the decay power decreases and of course may vary if system conditions change.

Equation 4-2 gives the fractional elevation vs. time:

$$\frac{z(t)}{L} = K_2 \left(1 - e^{-t/K_1}\right) + e^{-t/K_1} \qquad (4\text{-}2)$$

For long times after uncovery, the fractional level becomes K_2, which is defined by:

$$\frac{z(t)}{L} \rightarrow K_2 = \frac{W_{in}\left(H_{fo} + H_f - H_{in}\right)}{P_o \, DH}$$

This shows that the decay power generated below the stable coolant level ($(z/L)(P_o\ DH)$) is equal to the power to heat the subcooled inlet to saturation and to vaporize it ($W_{in}(H_{fg} + H_f - H_{in})$). Figure 4-18 shows how the level approaches the stable level vs. the normalized time, which is t/K_1 in Equation (4-2).

Depending on the lowest level attained, the steam convection cooling might or might not be sufficient to keep the maximum temperature of the clad from rising because of the decay heat input plus the oxidation heating. It is estimated that even without oxidation heating the clad temperature is above 1800°F with the steady level at 60%. Therefore, the clad will continue to rise in temperature even if the level is held constant, when the level is lower than about 60%.

The oxidation below the coolant level is essentially zero. Above the coolant level, the local oxidation may be greater for a given total amount of hydrogen generated than if boiloff proceeded without inlet flow and the same amount of hydrogen were generated from a longer fraction of the core length. Figure 4-15, which predicts the number of embrittled rods for a measured total hydrogen generation, yields a lower limit damage estimate when there is some inlet flow to the vessel. This same argument may be extended to include the additional oxidation which occurs in scenarios where the refill proceeds slowly.

The effect of inlet flow on core temperature is exemplified by Figure 4-19, where inlet flow is sufficient to maintain a steady 60% coolant level. The average core exit steam temperature is plotted as a function of the subcooled

FIGURE 4-18
COOLANT LEVEL vs NORMALIZED TIME WITH
VARIOUS INLET FLOW RATES

$$K_2 = \frac{W_{INLET}\,(H_g - H_{in})}{DECAY\ POWER}$$

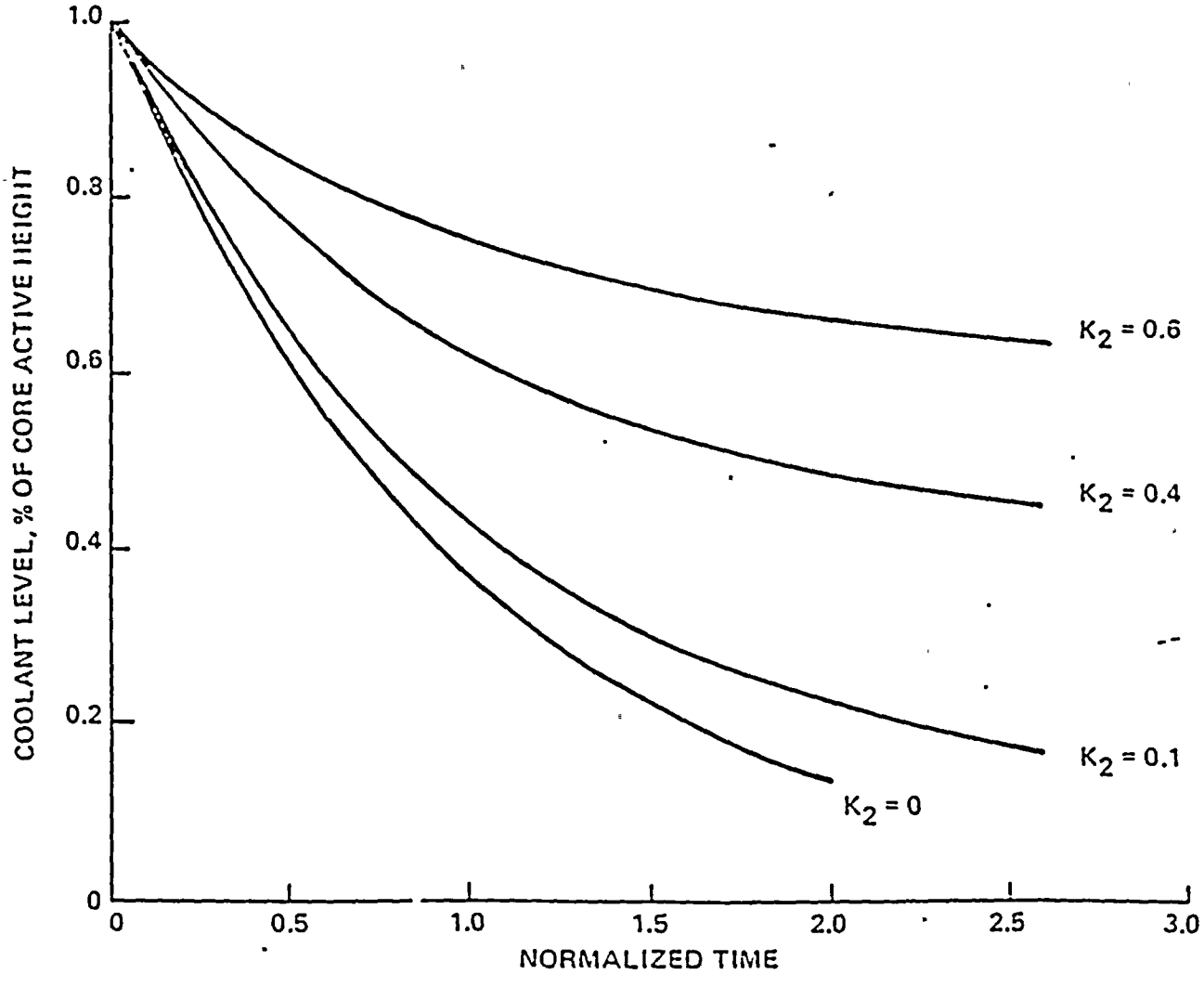
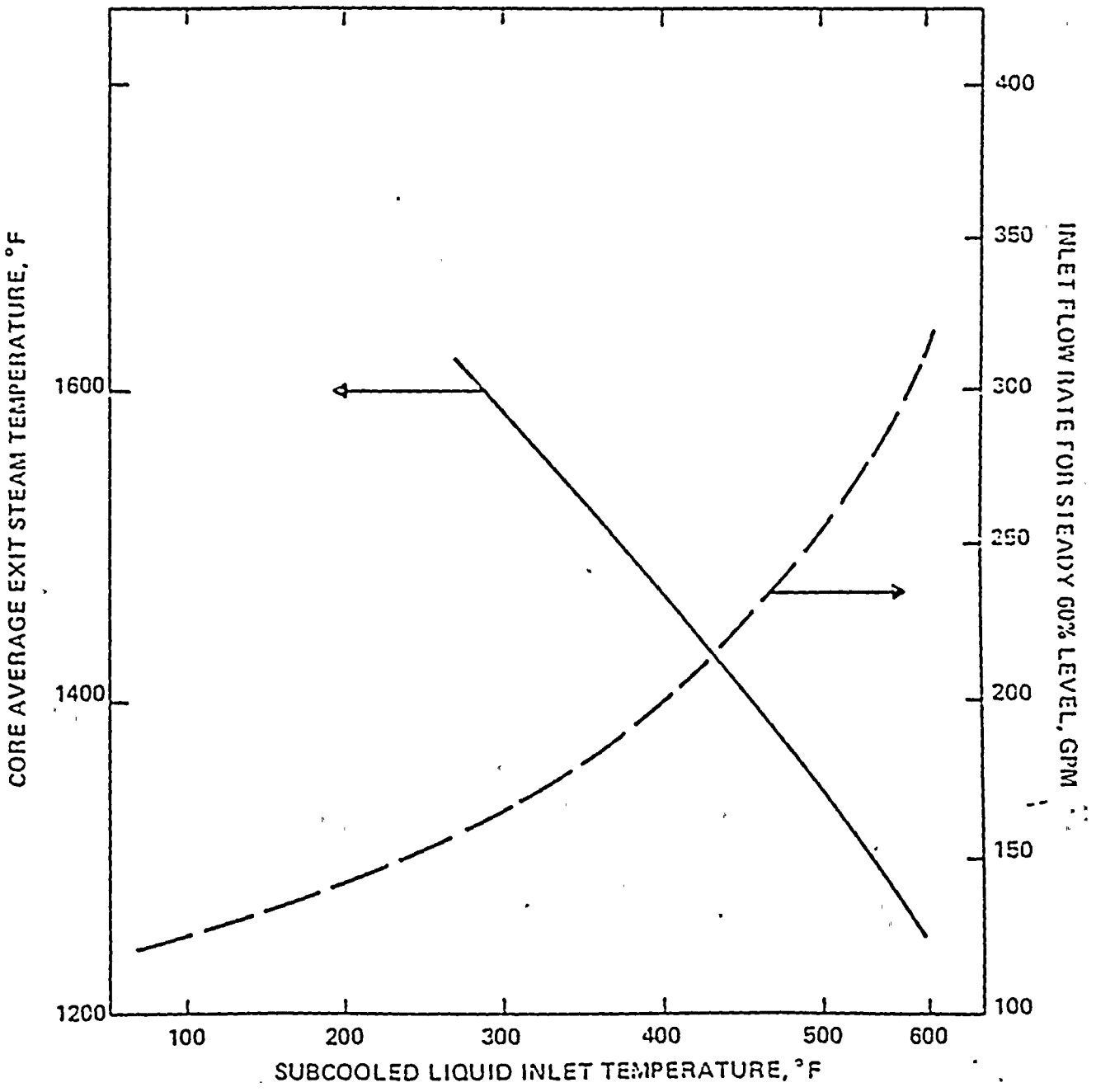


FIGURE 4-19
BOILOFF WITH INLET FLOW TO MAINTAIN STEADY 60%
COOLANT LEVEL - STEAM TEMPERATURE & INLET FLOW RATE vs
INLET TEMPERATURE. AT 1200 PSIA
3400 MWT CLASS, 1% DECAY POWER



inlet temperature. The decay power generated by the covered 60% of the core heats the subcooled inlet and vaporizes it. With high inlet temperature, a larger fraction of the decay power produces steam and the steam flow rate is high. With low inlet temperature, a larger fraction of the decay power heats the subcooled liquid to saturation and the steam flow rate is low. A lower steam flow rate yields higher steam superheat above the coolant level.

A saturated inlet flow of 315 GPM at 567°F and 1200 psia maintains a steady level at 60% and yields a core average exit temperature of 1250°F. A subcooled inlet flow of 125 GPM at 100°F and 1200 psia maintains the same steady level, but yields a core exit temperature of 1800°F. If boiloff proceeds with zero inlet flow, the transient temperature is 930°F as the level drops down past the 60% level. Hence, above a given coolant level, the temperatures and oxidation can vary depending on the inlet flow rate.

During the core uncovery period of the TMI-2 accident, there was some inlet flow. It probably caused the distribution of oxidation to be greater in the upper portion of the core than would have occurred without inlet flow and with the same total amount of core oxidation. Of course any inlet flow is better than none. The core damage at TMI-2 would have been greater for the same duration of uncovery if there were no inlet flow.

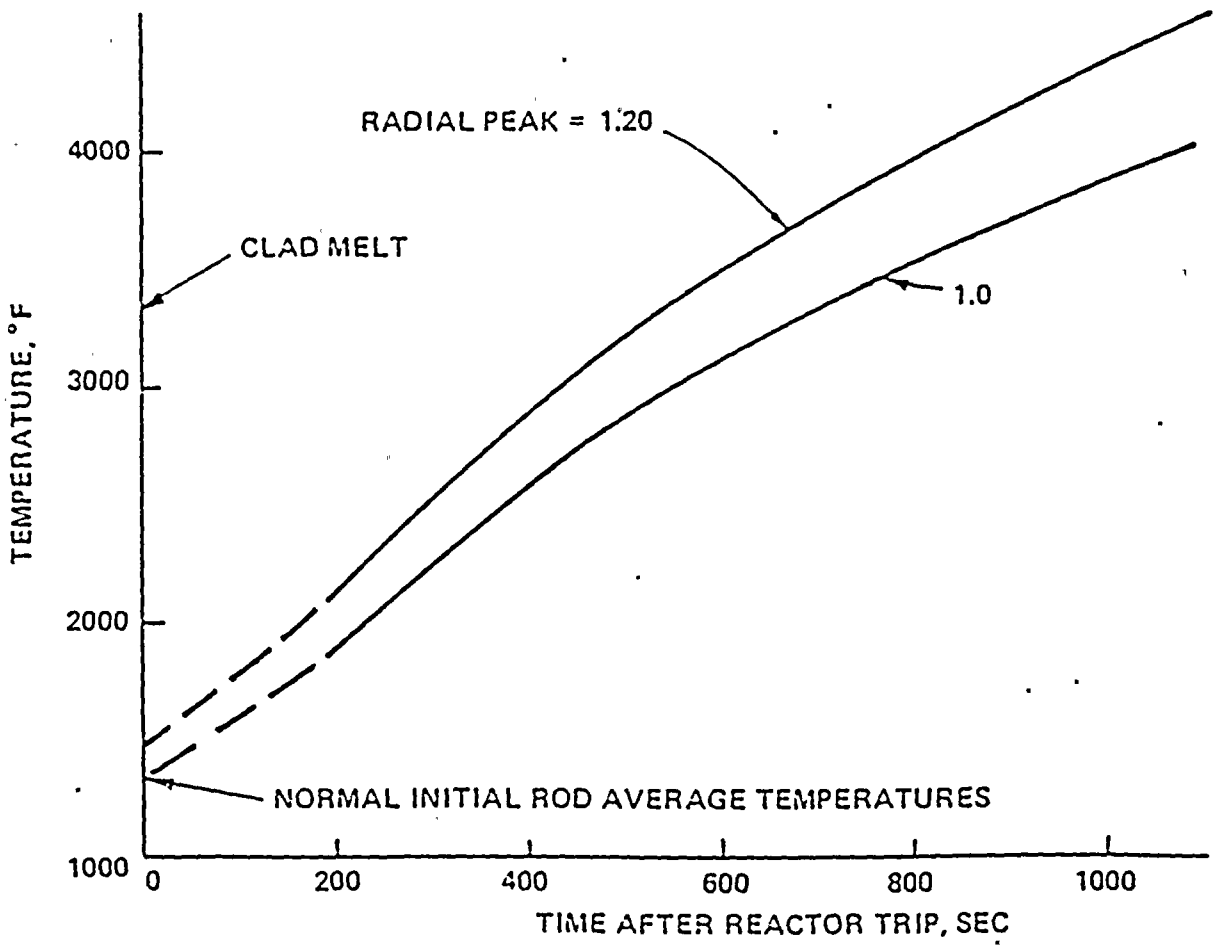
The conclusion is that a clad damage assessment utilizing Figure 4-15 may yield a low estimate of the number of rods embrittled if a) there is some inlet flow which slows uncovery and/or b) the refill is slow.

4.4.2 Comparisons with Alternate Computer Model

The MAAP computer code is a state-of-the-art tool for analyzing very severe postulated core meltdown scenarios. It is being developed under the IDCOR program sponsored by the nuclear industry. The early portions of several core melt scenarios are reviewed here. These examples are run on the 2700 Mwt core configuration. Only the portion of the accident starting from core uncovery and proceeding to 20% local oxidation is considered here. Assessment of more severe core damage would require much greater effort. The effort is beyond the scope of this PASS procedure and is of questionable value here since the measurement of the damage would require interpretive algorithms utilizing multiple sources of recorded instrument data. Such data is not recorded by the PASS and might not be available at all plants. Radiation measurements by the PASS can provide one estimate of the extent of severe core melt.

The first case analyzed with MAAP is a large break LOCA. It is included to show how the damage assessment procedure yields an underprediction of the extent of damage. Large break LOCA events depressurize to containment pressure within tens of seconds, exhausting the primary system and leaving the core empty of coolant. Figure 4-20 shows the adiabatic heatup which follows-- the blowdown assuming the ECCS does not function. Within about 10 minutes the temperature on the peak power rods exceeds the Zircaloy melting point of 3350°F. The fuel structure fails by melting rather than oxidation induced fragmentation.

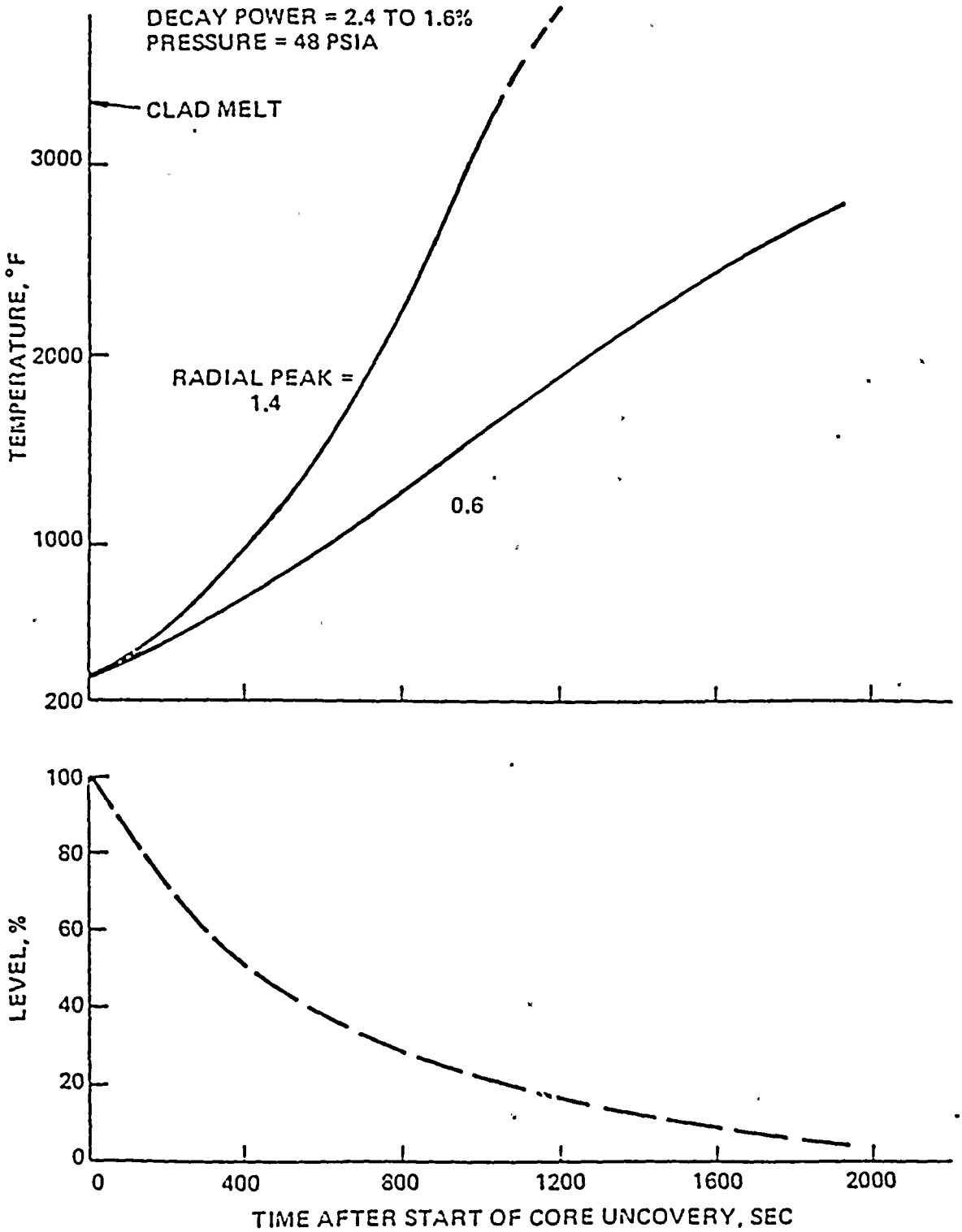
FIGURE 4-20
PEAK TEMPERATURE vs TIME
MAAP CODE RESULTS FOR HEATUP AFTER TOTAL CORE BLOWDOWN
PRESSURE = 50 TO 40 PSIA
LOCAL CLAD OXIDATION < 0.001%
OXIDATION LIMITED BY ABSENCE OF STEAM FLOW



The amount of hydrogen generation calculated for this example, prior to reflood, is much less than the measurement sensitivity for the containment atmosphere samples. During a subsequent reflood, a greater amount of hydrogen is generated, with the amount being dependent on the speed of the reflood. As the core floods, steam and gas pushed up and out of the core may cause a rapid rise in the Core Exit Thermocouple (CET) temperatures. Some indication of the speed of recovery is available from the recorded trace of thermocouple temperature. A rapid rise followed by quenching to saturation temperature indicates rapid reflood to above the CET elevation and minimum hydrogen generation. Figure 4-15 may then underestimate the number of rods damaged. A high (higher than saturation) valid CET temperature, which gradually changes over hundreds of seconds, indicates slower refill which is probably accompanied by oxidation and hydrogen generation. Figure 4-15 indicates that there is damaged fuel, but the relationship between core damage and hydrogen generated is uncertain. If the hydrogen measurement indicates a large fraction of the core zirconium is oxidized, say more than 20%, then substantial core damage is certain, regardless of the particular reflood scenario.

The second case analyzed with MAAP is a boiloff at very low pressure. It is a variation of the previous case. Properly functioning SIT's rapidly recover the core following a large LOCA. Best estimate analyses indicate little or no core damage when all systems function. If subsequent action, both automatic and by the operator, fails to maintain continuous inlet flow from the HPSI and LPSI systems, the core uncovers again by boiloff at very low pressure. Figure 4-21 gives the results for a MAAP calculation of this scenario.

FIGURE 4-21
MAAP CODE RESULTS FOR BOILOFF FOLLOWING
BLOWDOWN AND REFILL
PEAK TEMPERATURE AND LEVEL vs TIME



The oxidation at the top of the peak rods reaches about 20% at about the same time the temperature reaches the clad melting point of 3350°F. The MAAP analytical model assumes no further local oxidation, but the temperature continues to rise until it reaches the fuel pellet melting point. Oxidation meanwhile continues at the lower, uncovered elevations on the maximum power rods and along the other lower power rods until each elevation on each rod reaches clad melting. Some core structural rearrangement may occur as melting progresses, which eventually invalidates the analytical model.

For this particular scenario, the local oxidation reaches the embrittlement threshold at about the same time that local clad melting starts. The total hydrogen generation from core wide oxidation yields a low estimate of the number of rods with embrittled clad which may spill fuel into the coolant. For example at 1400 sec. the coolant level is 9% and MAAP indicates 6% of the core clad is oxidized and/or melted, representing some structural damage to about 73% of the rods. At 6% core oxidation, Figure 4-15 indicates 20 to 50% of the rods may be embrittled.

The third MAAP case is for boiloff at high pressure. Boiloff at the safety valve pressure setting is postulated for a complete loss of heat sink event. The secondary boils dry and subsequently the primary temperature and pressure rise. Safety valves open at about 2500 psia and the primary boils off at the safety valve pressure. A MAAP code analysis of this scenario is available for the 2700 Mwt class reactor. Results are compared to simplified calculations done on the 3400 Mwt class reactor.

Figure 4-22 shows the peak temperature on an average power rod and the coolant level as a function of time. There is good agreement for temperature and for level. If the clad rupture temperature is 1800°F, the figure indicates that the average power rod ruptures within 16 to 18 minutes of uncovery, when the coolant level is down to 15%.

When the level is 10%, the simplified analysis predicts less than 2% of the core clad is oxidized. At the same time after uncovery starts MAAP predicts about 3% of the core clad is oxidized. Figure 4-15 indicates that between 5% and 22% of the rods exceed the embrittlement threshold when 2% of the total core clad is oxidized, and between 10% and 30% when 3% is oxidized. Both these results place the extent of damage in the same category and are adequate for damage assessment.

MAAP also predicts that clad melt occurs at the time when the local oxidation reaches about 20%. The code limits the local oxidation thereafter. Clad melt occurs at 3350°F and is assumed to block the flow channel, thereby limiting steam flow and preventing further oxidation at elevations above the blockage. To the extent that this analytical model correctly prevents continued local oxidation, the core wide total hydrogen production is limited and Figure 4-15 would yield a low estimate of the number of embrittled rods.

The last MAAP case is for boiloff at an intermediate pressure of 1200 psia, and is shown by Figures 4-23, 24 and 25. The maximum temperature vs. time is shown in Figure 4-23. Within less than a 200 sec. differential, the MAAP code yields the same temperature as the simplified analysis. Such a shift of 2 or

FIGURE 4-22
COMPARISON OF MAAP CODE WITH SIMPLIFIED
CALCULATION RESULTS
HIGH PRESSURE BOILOFF

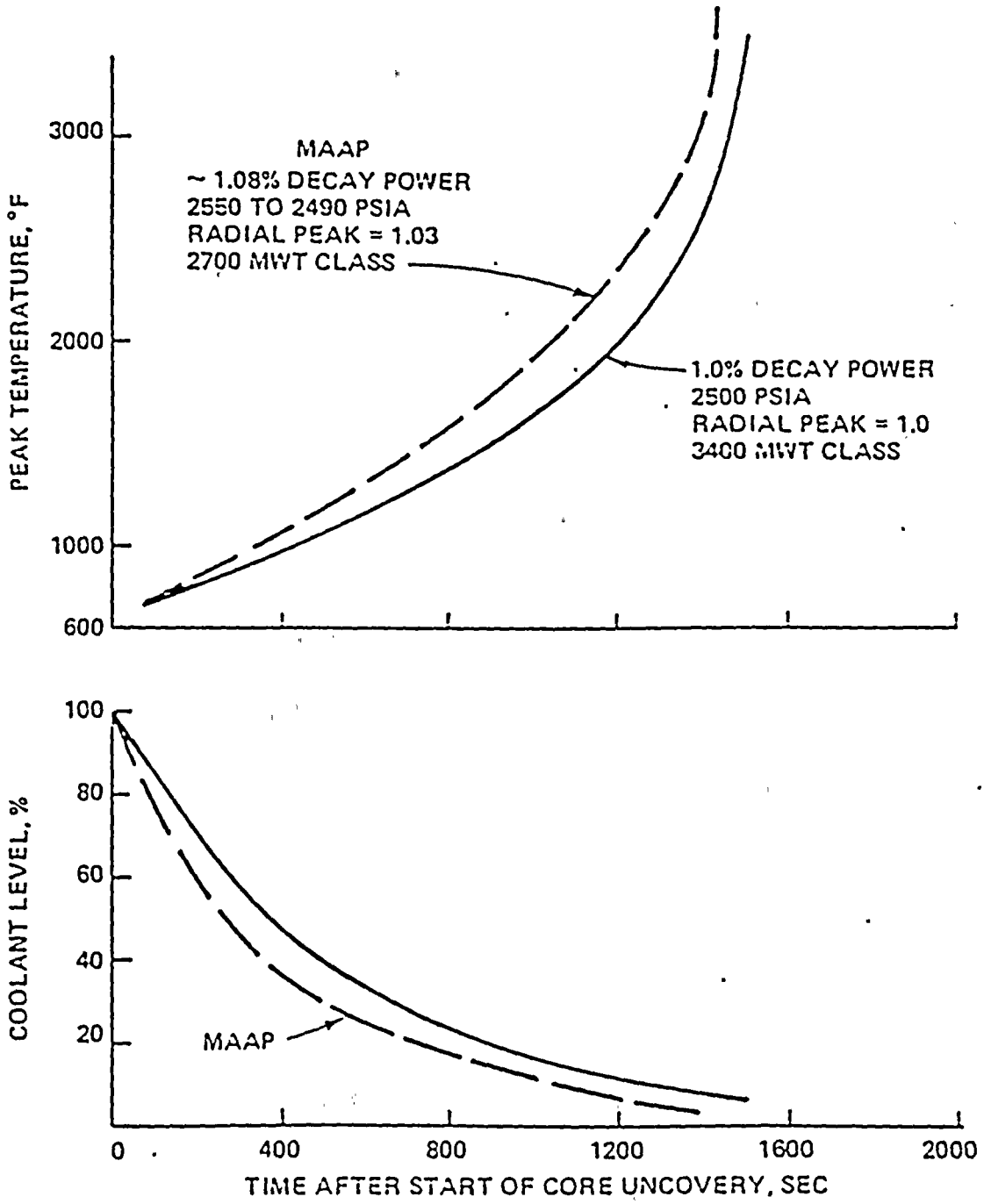


FIGURE 4-23
TEMPERATURE vs TIME
COMPARISON OF MAAP CODE AND SIMPLIFIED ANALYTICAL RESULTS

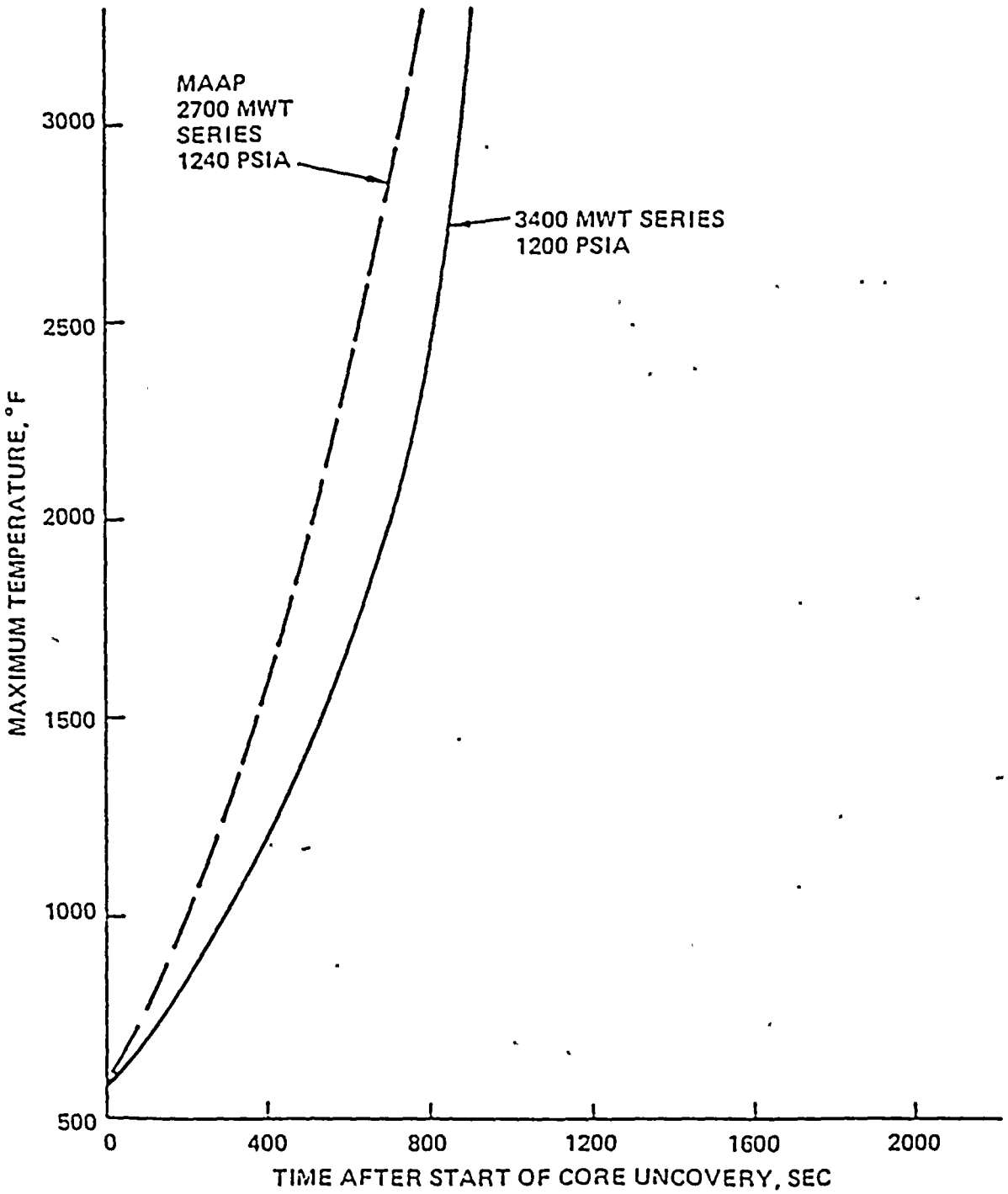


FIGURE 4-24
TEMPERATURE vs RADIAL NUCLEAR PEAK
COMPARISON OF MAAP CODE AND SIMPLIFIED ANALYTICAL RESULTS

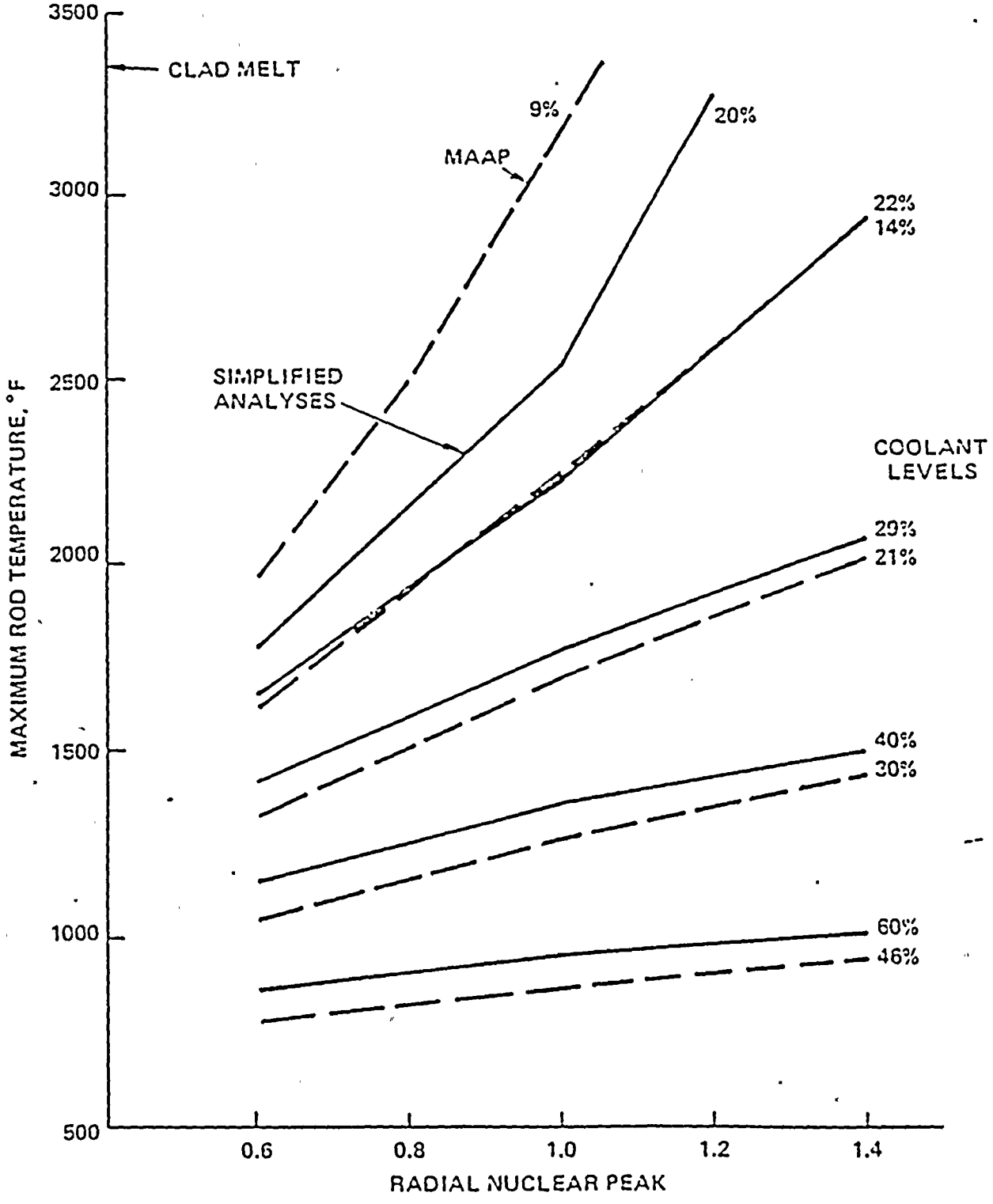
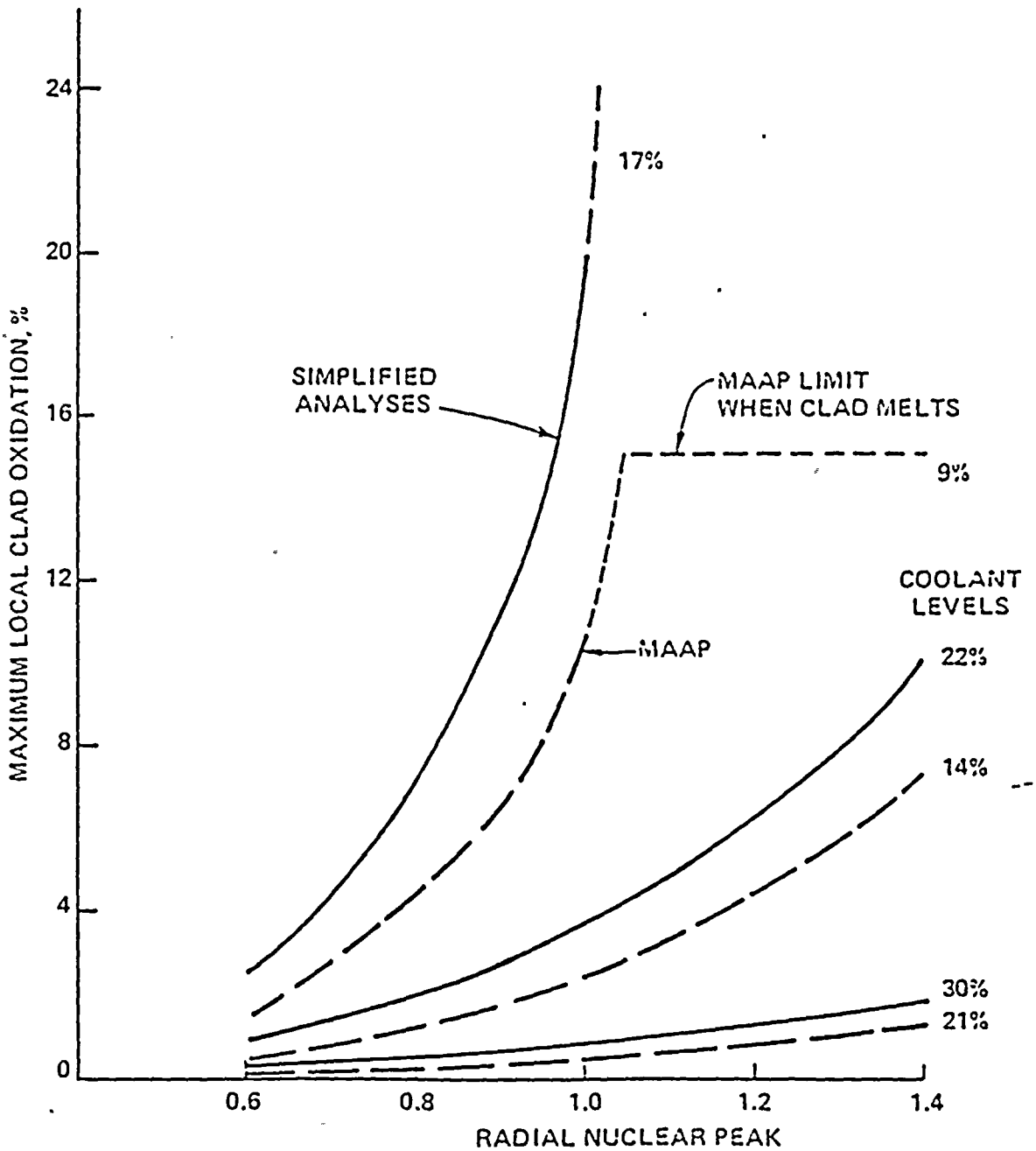


FIGURE 4-25
MAXIMUM LOCAL CLAD OXIDATION vs RADIAL NUCLEAR PEAK -
COMPARISON OF MAAP CODE AND SIMPLIFIED ANALYTICAL RESULTS



3 minutes to attain a given temperature is not so important to this procedure as the core wide distribution in temperature and oxidation at the time a given peak temperature occurs. Figure 4-24 shows the distribution of temperature with radial nuclear peak at several times or coolant levels. The MAAP code predicts a faster uncovery, but for a given peak core temperature, the core wide distributions are essentially identical. Similarly, Figure 4-25 shows that the radial distributions of local oxidation are identical.

The conclusion is that for a given total amount of oxidation as inferred from the PASS hydrogen measurements the simplified analytical model yields the correct core radial distribution of the oxidation and therefore of the number of rods which are ruptured or are embrittled. Similarly, for a given peak core temperature the simplified analytical model also yields a correct estimate of the number of rods with temperatures above a specified clad rupture temperature.

4.5 BASES FOR RELATIONSHIP BETWEEN AMOUNT OF HYDROGEN MEASURED AND AMOUNT OF CORE OXIDATION

There are multiple sources of hydrogen gas production inside the containment-building during postulated accidents in addition to the oxidation of the zirconium in the core. These other sources include: the hydrogen gas normally found in the reactor coolant for corrosion control, the oxidation of various metals used within the containment, and the radiolytic decomposition of water. The procedure for core damage assessment using hydrogen gas measurement employs correlations between the degree of cladding rupture or

oxidation and the amount of hydrogen produced by the initiating chemical reactions. The hydrogen measurement is performed by the PASS system on samples obtained from both the Reactor Coolant System hot leg piping and from the containment building atmosphere. These measurements of the hydrogen gas do not distinguish between the production source of that gas. Therefore, the procedure requires a means to determine that contribution of hydrogen to the sample measurement which is produced from sources other than core oxidation. This is accomplished through analytical estimates of the production rates. These production rates are shown to be source dependent upon either the containment building atmosphere temperature or the fission product distribution. This section describes in detail the analytical techniques and assumptions employed in the required determination. Each of the sources of hydrogen production are discussed.

The reactor coolant under normal plant operating conditions contains a dissolved hydrogen concentration which is present to scavange any oxygen which may be present and thereby reduce the potential for corrosion. This hydrogen is present as a result of the radiolytic decomposition of both the reactor coolant and in some plants the pH control additives and as a result of direct addition through the Chemical and Volume Control System. The normal operating range is between 10 and 50 cc@STP/kg. Therefore, the total quantity of hydrogen present in the coolant for any C-E NSSS is anticipated to be less than 500 SCF. This value is considered to be a negligible contribution to the measurement of total post accident hydrogen concentration for several reasons. First the presence of this quantity of gas is well known and would not be misinterpreted in a post accident measurement. Secondly, the contribution of this quantity to the concentration in the containment building atmosphere

under the assumed condition in which all the hydrogen is released is less than 0.2 volume percent. This value is close to the minimum sensitivity of the typical PASS measurement capability and is below that of the procedure which distinguishes only between the NRC categories of core damage. Also the assumed condition is unlikely because the solubility of hydrogen in water is such that a depressurization to below 100 psig would be required for a complete release.

Employed within the containment building are a variety of metals which constitute a potentially significant source of hydrogen production as a byproduct of the oxidation reactions which occur as corrosion. The specific metals which contribute to this source are principally aluminum and zinc. All other metals are known to be an insignificant contribution when compared to these two. Aluminum and zinc are found principally within the electrical components, paint, and galvanized steel structures. The corrosion reactions are a result of the chemical environment under accident conditions. Independent of the postulated accident scenario, these metals are exposed to a borated solution which has been pH adjusted to the alkaline range through the addition of chemical additives, which contains significant amounts of dissolved oxygen due to the exposure to the containment building atmosphere, and which undergoes transient temperatures that may range up to 300°F.

The rate of hydrogen production from oxidation of these metals is dependent upon many variables which include the surface to mass ratio of the metals, the quantity of the metals present, the use of protective surface coatings on the metals present, the surface temperature, the presence of pH additives, the

extent of metal emersion in the borate solution, and the experimentally determined reaction rate constant. These variables are plant specific. Therefore, implementation of this procedure requires the development of plant specific analytically determined hydrogen production rates. The production rates are required to be expressed as a function of the containment building atmosphere temperature. This information has been developed using input data available from the latest revision of each specific plant Final Safety Analysis Report (FSAR). The result is provided in the form of a graph, Figures 4-26 through 4-31, for each plant. A detailed description of the procedure used to calculate this information is provided. These curves may be used directly or each utility may choose to redevelop a given plant specific curve based upon more recent input data. The purpose of providing these graphs is to give examples of the analytical techniques and assumptions to be employed.

The subject procedure is a document intended to provide an actual assessment of core damage for the purpose of implementing emergency operational decisions following an accident. Therefore, care should be taken to employ results of realistic or best estimate dose rate analyses rather than conservative information which may have been developed for such purposes as licensing activities or the design bases for equipment. The use of hydrogen production analyses based on conservative assumptions could actually result in a lower than actual assessment of core damage. This is because the conservative assumptions dictate greater consequences for the production of hydrogen than may be actual for a given category of core damage. The measurement of the lower or realistic hydrogen quantity would then be correlated to a lower than actual category of core damage. The purpose of this section is to describe the analytical assumptions recommended to be employed in that realistic analytical development.

FIGURE 4-26
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND
ZINC vs TEMPERATURE FOR CALVERT CLIFFS UNITS 1 & 2

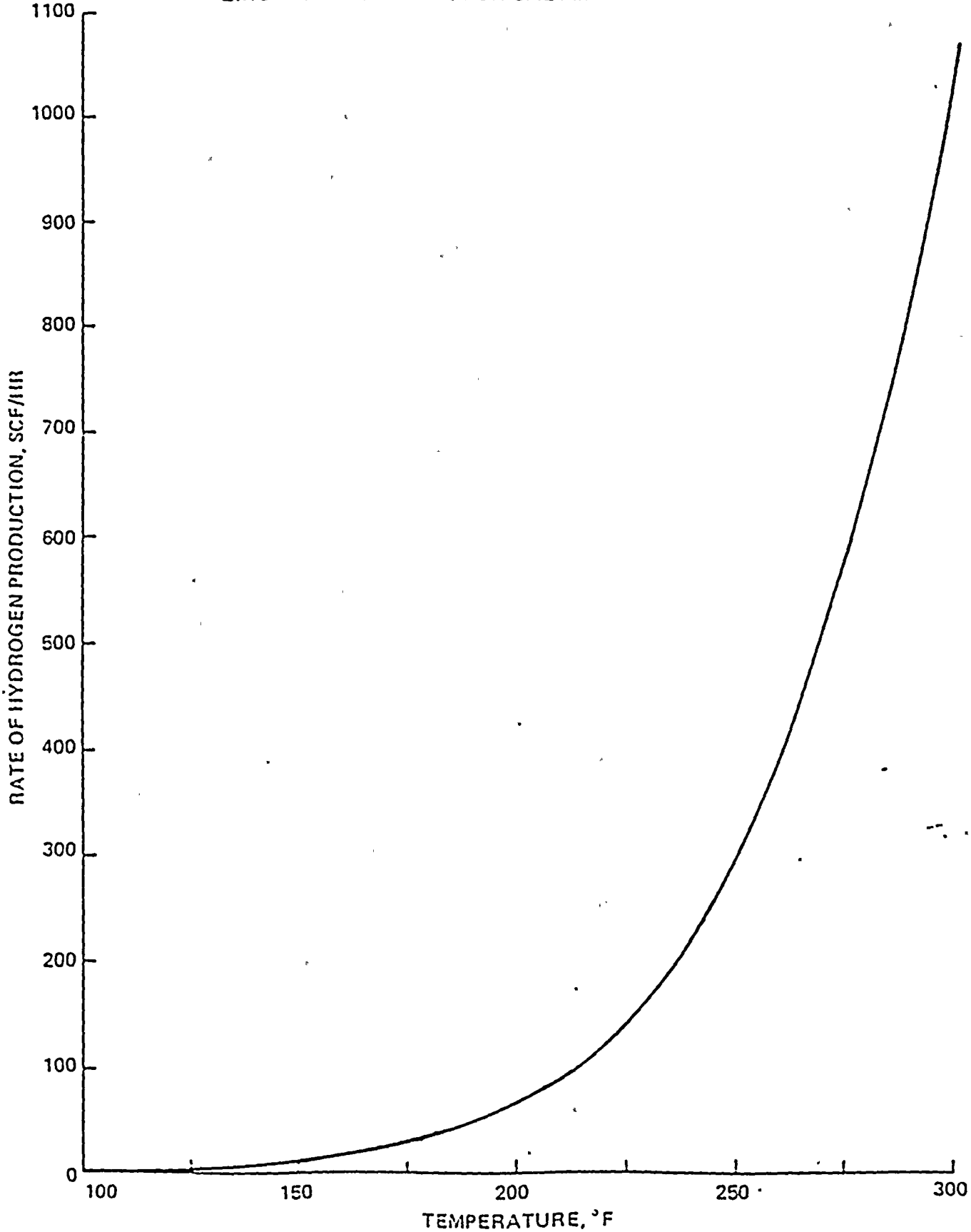


FIGURE 4-27
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND
ZINC vs TEMPERATURE FOR PALO VERDE NUCLEAR GENERATING STATION

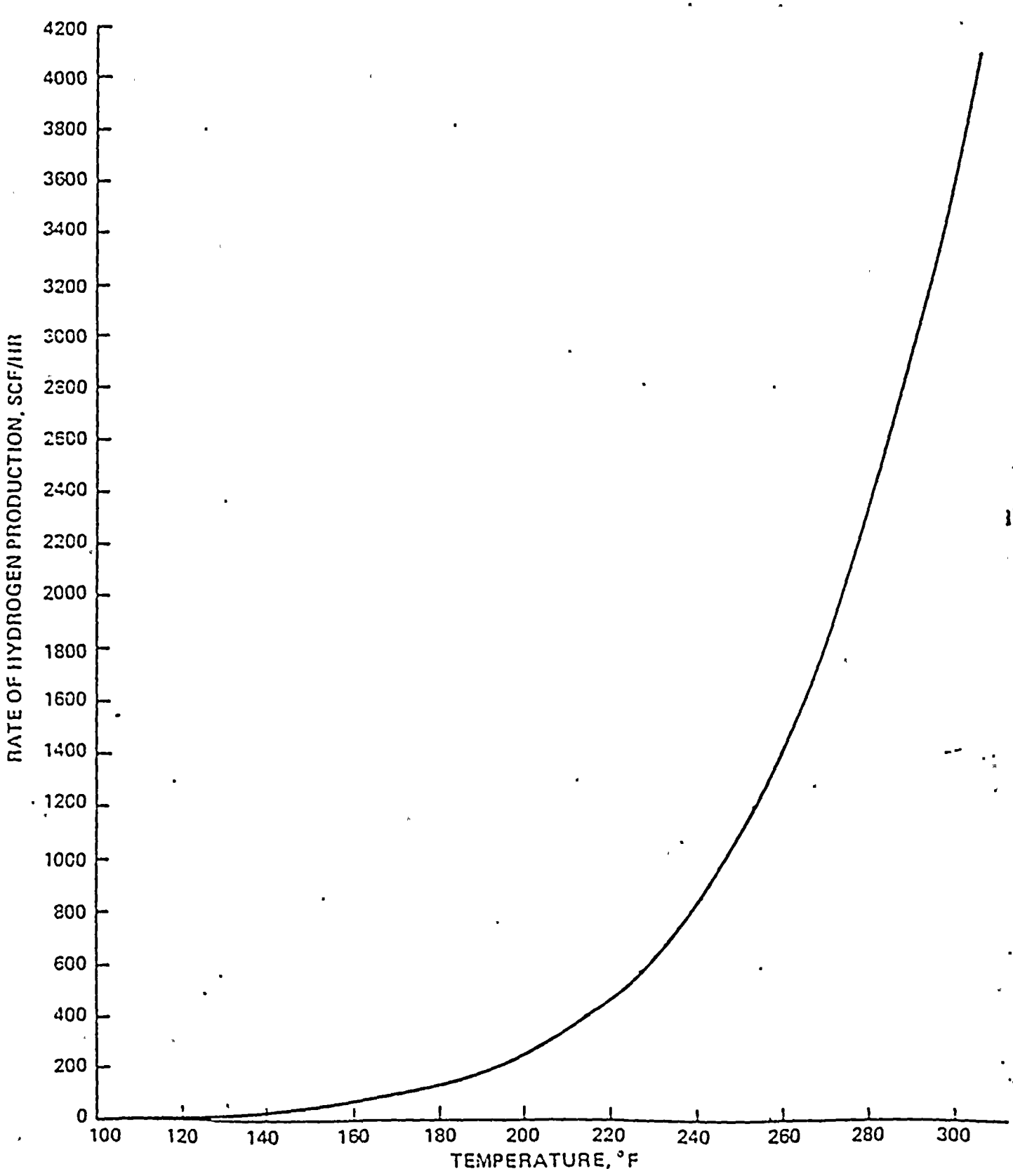


FIGURE 4-28
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND
ZINC vs TEMPERATURE FOR ST. LUCIE UNIT 2

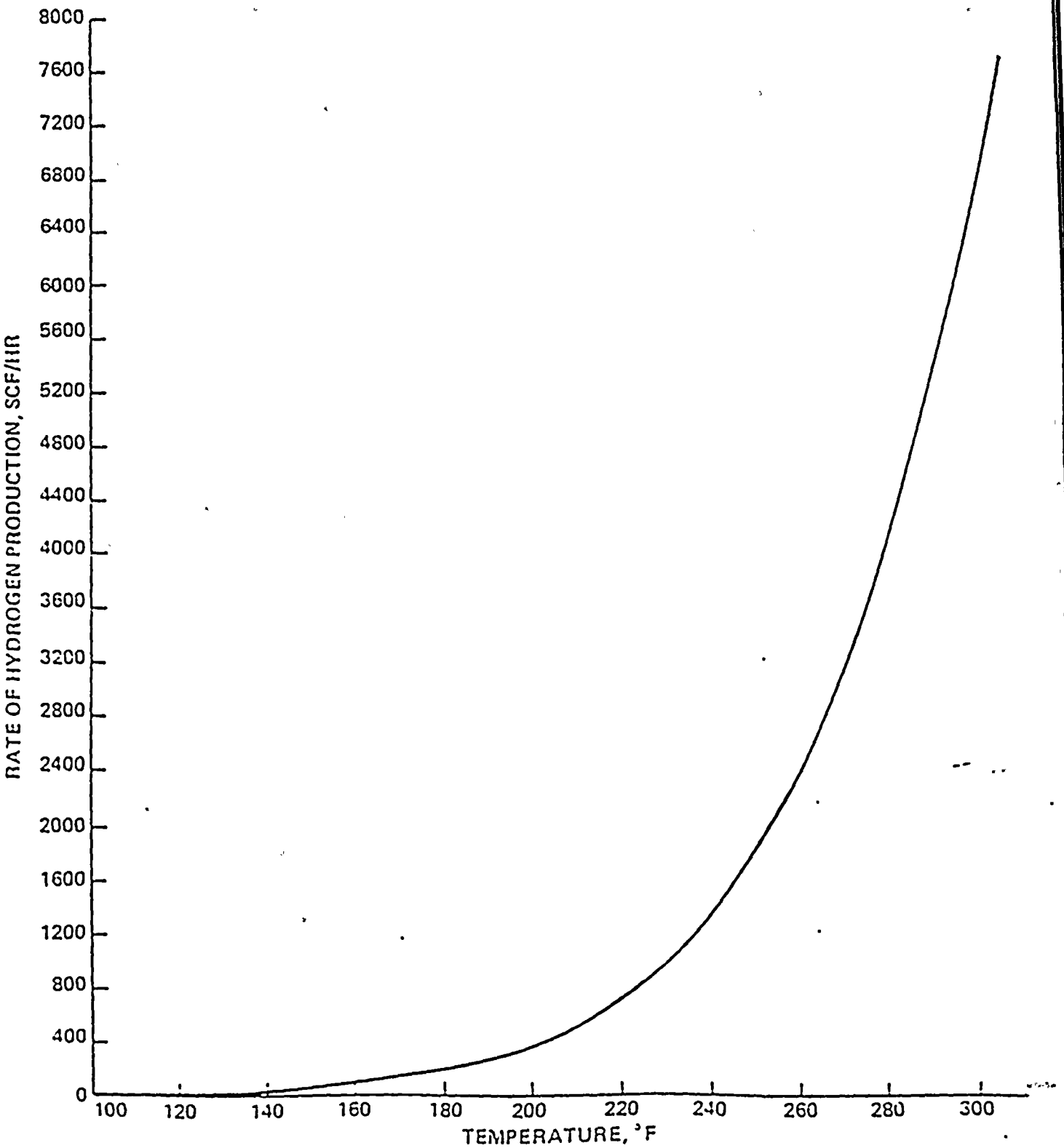


FIGURE 4-29
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE FOR SONGS UNITS 2 & 3

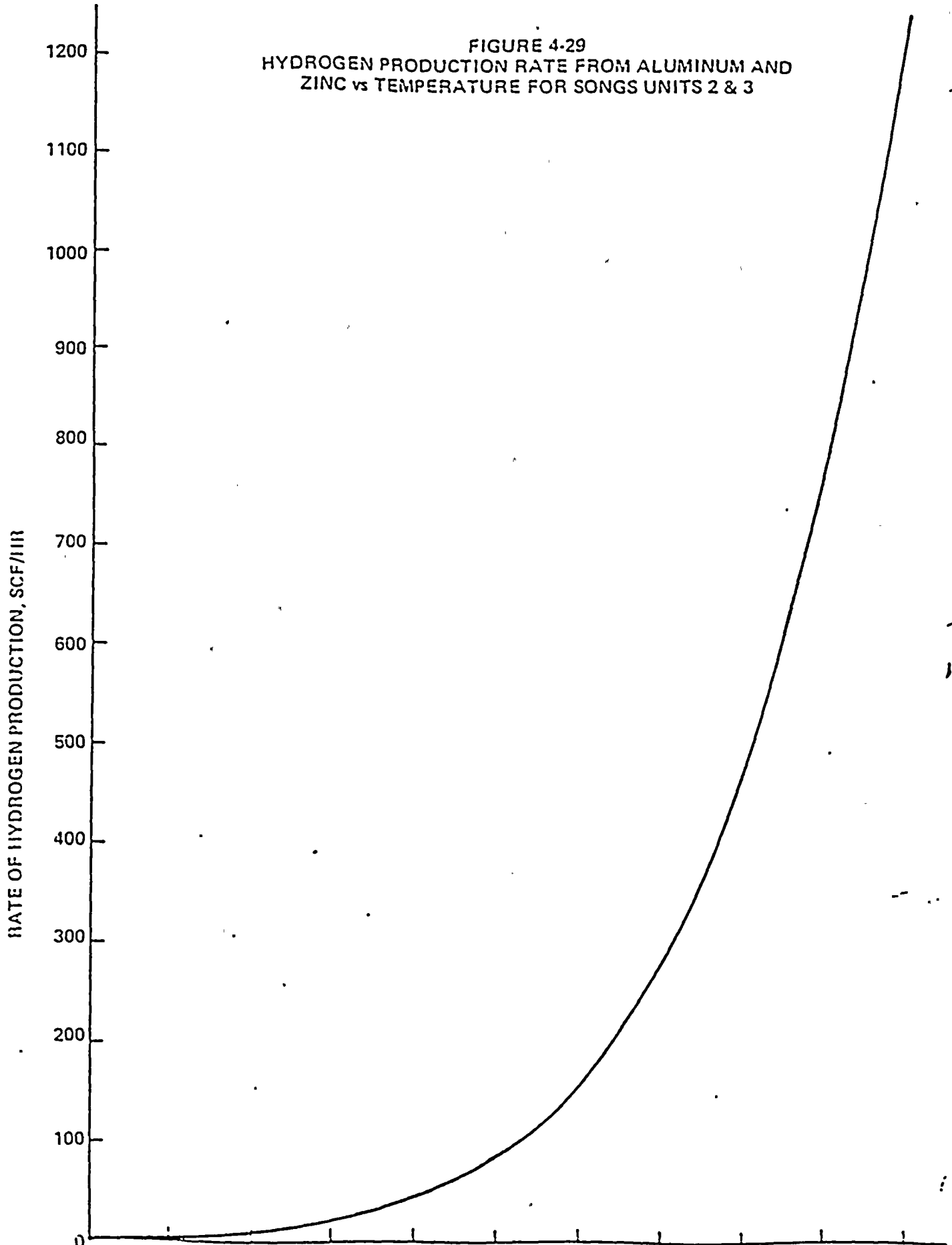
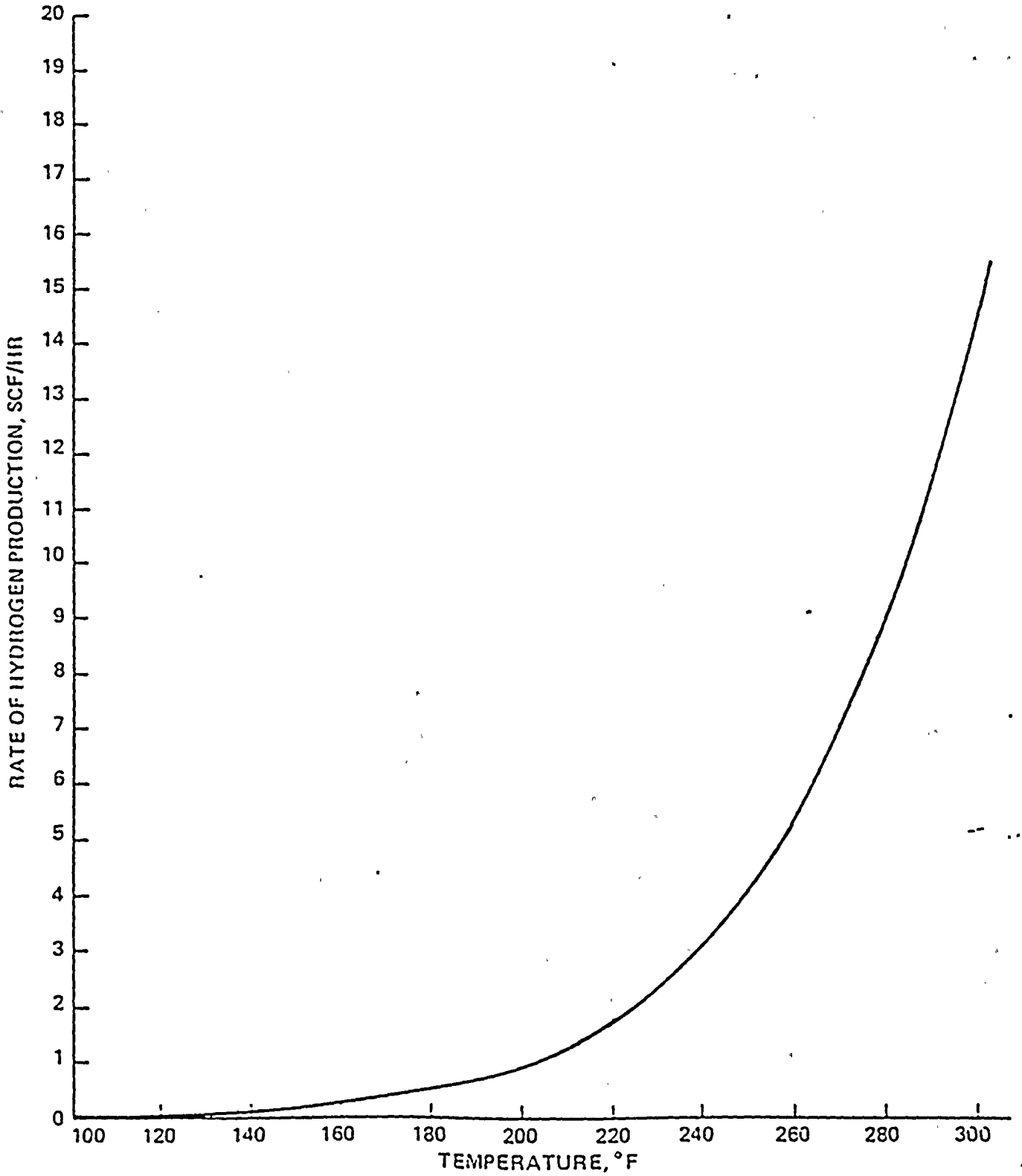


FIGURE 4-30
RATE OF HYDROGEN PRODUCTION FOR ALUMINUM AND ZINC vs TEMPERATURE FOR WPPSS



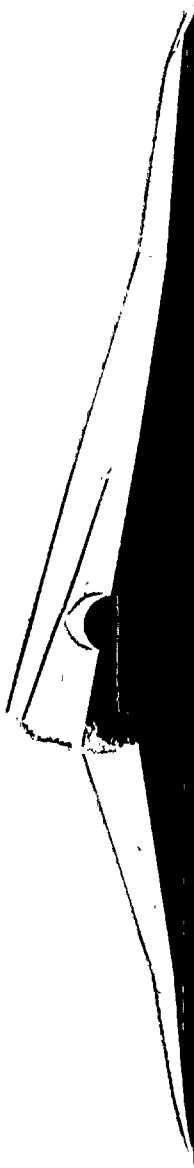
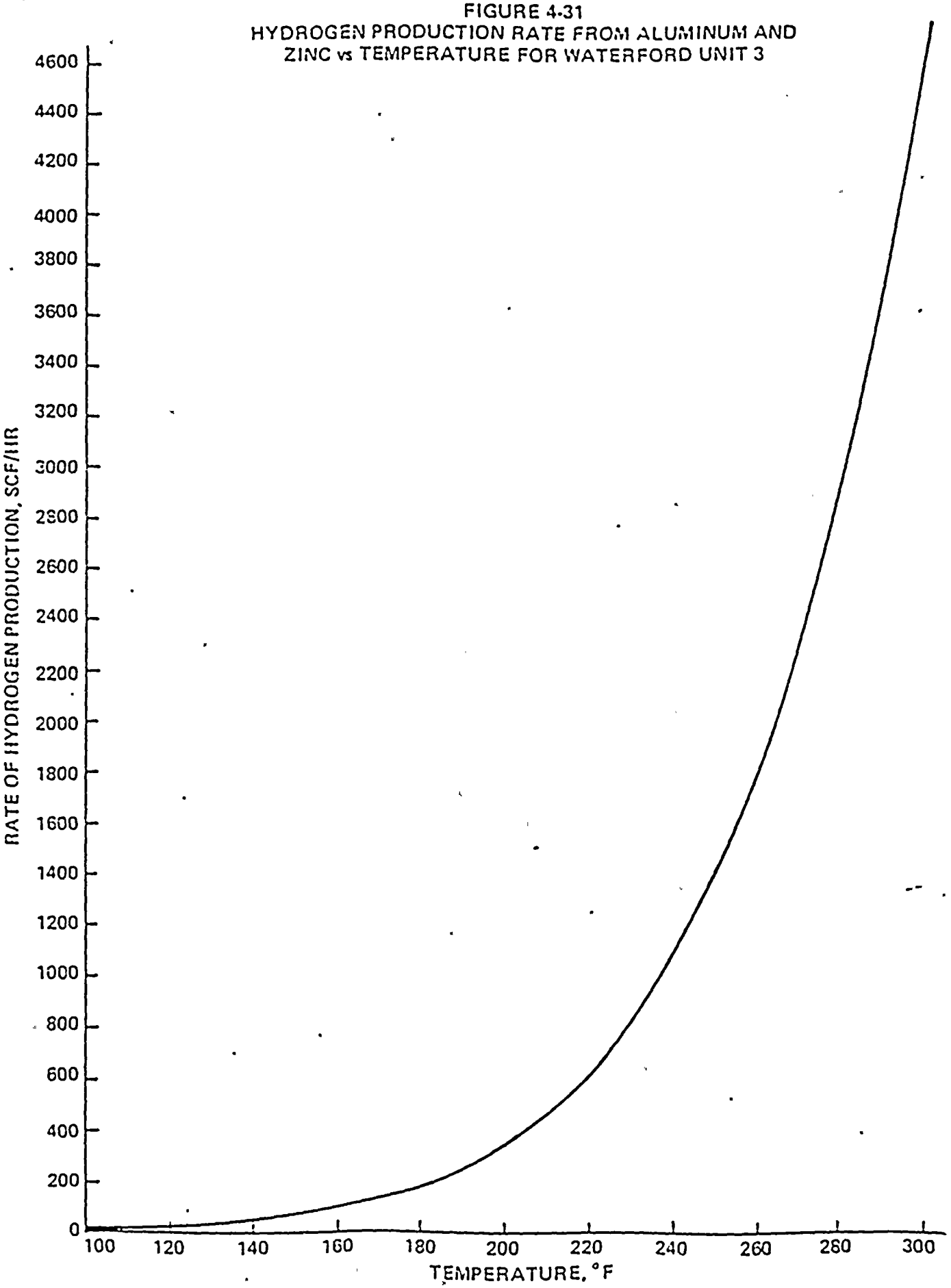


FIGURE 4-31
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE FOR WATERFORD UNIT 3



A number of the variables required in the analyses are well defined for a specified plant. These variables include the specifics of the metal inventory inside the containment building and the presence of additives to adjust the pH to an alkaline range. The information used in the analyses of the production rates provided in Figures 4-26 through 4-31 has been obtained from the plant specific FSAR. The inventory of aluminum and zinc expressed by weight and, when available, the surface to volume ratios are stated in the FSAR which describes the systems for hydrogen control within the containment building. Two assumptions were made in the analytical production rate determination with regard to the metal inventory. First it is assumed that those metals, such as paint, for which the surface to mass ratio is large, oxidize completely within one hour should the containment building atmosphere temperature exceed 200°F. This is assumed because accurate experimental rate constants are not available for large surface to mass ratios. The consequence of this assumption is that the procedure should not be applied to hydrogen sample measurements taken within the first hour following an accident. This is consistent with present design capabilities of typical PASS systems which require several hours to obtain and analyze a representative sample. A value is provided in Table 4-3 for hydrogen assumed to be produced in this manner for these plants with specific FSAR data. Second it is assumed that there is a limit to the total quantity or maximum yield of hydrogen produced based upon the depletion of the material present. This results from the assumption of complete reaction for those materials with large surface to mass ratio and an assumed surface depletion for those materials with a small ratio. A value is provided in Table 4-4 for the maximum hydrogen yield for those plants with specific FSAR data. As previously stated, these values may be used directly or each utility may choose to employ a plant specific value based upon more recent data.

TABLE 4-3

HYDROGEN RESULTING FROM INSTANTANEOUS REACTION OF ALUMINUM AND ZINC

PLANTS	HYDROGEN SCF
Calvert Cliffs Units 1 & 2	- -
Palo Verce Nuclear Generating Station	- -
St. Lucie Unit 1	- -
St. Lucie Unit 2	5235
SONGS Units 2 & 3	200
WPPSS	8398
Waterford Unit 3	- -

- - Indicates no instantaneous reaction was assumed, due to insufficient data on surface to mass ratio in the FSAR.

TABLE 4-4

MAXIMUM HYDROGEN YIELD FROM ALUMINUM AND ZINC

PLANTS	HYDROGEN SCF
Calvert Cliffs Units 1 & 2	Not Available*
Palo Verde Nuclear Generating Station	200,271
St. Lucie Unit 1	481,142
St. Lucie Unit 2	292,051
SCNGS Units 2 & 3	96,673
WPPSS	86,686
Waterford Unit 3	173,582

* FSAR does not provide the information needed to estimate Maximum Hydrogen Yield.

There are, however, several variables which require the use of analytical assumptions. These variables are the extent of metal emersion in the borate solution and the selection of the reaction rate constant based upon the available experimental data. These analytical assumptions are related because the reaction rate constant depends upon the extent of emersion in the borate solution. Each of these assumptions is discussed in detail.

The oxidation reaction rate is influenced by the availability of oxygen and the buildup of reaction byproducts on the surface of the metal. The availability of oxygen is assured by the presence of the containment building atmosphere. The buildup of reaction byproducts on a metal surface is dependent upon the presence or flow of the borate solution across that surface. A metal surface submerged in a stagnant pool of water will corrode at a slower rate than a surface which is undergoing a spray of water. This is because the spray flow will remove the soluable reaction byproducts and continually expose new metal to the corrosive environment. The procedure has been defined as applicable in the core damage category of fuel overheat. The category of fuel overheat cannot be reached unless some fraction of the core has been uncovered. Therefore, it must be concluded that a large mass of high enthalpy reactor coolant has been released to the containment building. This is sufficient to assume that the metal surfaces in question are emersed in steam with condensation resulting in a limited flow of water. Additionally, should it be activated, the Containment Spray System will further increase the surface flow rate with borated coolant and pH control additives. Independent of a specific accident scenario it is also concluded from the category of core damage that the contents of the Safety Injection Tanks and Refueling Water

Tank are introduced into the containment. This additional water results in a sufficient level in the building sump to submerge some of the metal inventory. However, a review of the FSAR data indicates that only a small fraction of the aluminum and zinc is anticipated to be completely submerged in the sump. Therefore, the recommendation is that the assumed reaction rate constants should be determined from experimental data which address a metal surface at elevated temperature exposed to a continuous flow of borated water with pH adjusted to the alkaline range.

The reaction rate equation to be employed in the oxidation of aluminum and zinc is similar in form to the equation used for zirconium oxidation described in Section 4.2.1. The form of this equation is expressed as:

$$W = A_{metal} e^{-B/RT}$$

where:

W = equivalent metal mass oxidized per unit area per unit time

A_{metal} = experimentally determined rate constant

B = experimentally determined activation energy

R = ideal gas constant

T = surface temperature

This equation can be used to yield the rate of hydrogen production by assuming the oxidation reaction to proceed in a stoichiometric manner. The equation is then adjusted by a simple unit conversion to yield the standard quantity of hydrogen produced per unit surface area per unit time.

The two values which must be determined from experimental data are the rate constant and the activation energy. These values will vary for each metal. A survey of the published experimental data was conducted to determine the values recommended for use in this procedure. This survey identified a number of temperature dependent influences on the applicability of an exponential rate equation. The observed effect of temperature on corrosion reactions is not explicitly exponential as it would be for most chemical reactions nor linear as it would be under physical change. Each are discussed as an indication of the limitations of the analytical prediction and therefore the accuracy of the procedure.

Temperature may affect the corrosion rate through its effect on oxygen diffusion in the borate solution at the metal surface. The corrosion rate may increase with temperature. The rate will decrease rapidly to a low value at the boiling point due to the decrease in oxygen solubility.

Temperature may affect the corrosion rate through its effect on pH. Because the dissociation of water increases with temperature the pH decreases with temperature. This is more significant in the early time periods of an accident prior to the introduction of the pH additives through the Containment Spray System.

Temperature may affect the corrosion rates through its effect on the surface films. The temperature dependence of the solubility of the protective corrosion byproducts will vary the corrosion rate with temperature unless an aggressive surface flow is present. A change in temperature also may bring about changes in the physical nature or the chemical composition of the protective byproducts.

Heat flux may affect the corrosion rate. The temperatures of the metal surface, the borated coolant, and the containment building atmosphere may all be different and each may be anticipated to vary with location inside the containment building.

Based upon the survey of the available experimental data the following correlations are recommended as being applicable to this procedure for the realistic analytical estimate of hydrogen production rate.

For the oxidation of aluminum, the experimental data provided in Reference 7-19 was fit to the exponential rate equation to yield

$$W = 5.9 \times 10^{9} \, e^{-\frac{1690}{T(^{\circ}K)}}$$

For the oxidation of zinc the data provided in Reference 7-20 states the rate equation to be:

$$W = 2.1 \times 10^{5} \, e^{-\frac{7233}{T(^{\circ}K)}}$$

where in each case the variable W is the production rate of hydrogen expressed as standard cubic foot produced per square foot of metal surface area per hour.

The validity of these selected correlations to provide a realistic estimate was verified by a series of parametric analyses in which the constants determined from a number of other published reports were employed to yield estimates of the hydrogen production rates. The recommend equations were

shown to yield results lower then those correlations known to have been developed for the conservative purpose of licensing activities or equipment design. Therefore, the above equations were employed in the calculation of the graphs provided in Figures 4-26 through 4-31.

The decomposition of water by radiolysis results in the production of a significant source of hydrogen gas. This is especially true following an accident in which fission products become dispersed in the reactor coolant. Under this condition, the radiation shielding provided by the structural materials of the core is no longer present and all of the radiation energy, both beta and gamma, is absorbed in the water. The rate of production of hydrogen from radiolysis is dependent upon many variables which include reactor power, fuel burnup, the extent of fission product release to the coolant, and the rate constant for hydrogen produced as a function of energy absorbed. Each of these variables can be normalized to yield a generic production rate expressed as a function of reactor power.

The hydrogen production from radiolysis is dependent directly upon the quantity of the fission products released from the core. The convention employed to express that quantity is the same as that described in Section 3.3 for use in the Interim Procedure. The quantitative release of fission products is expressed as the percent of the source inventory at the time of the accident which is observed to be present in the sample media and therefore available for immediate release to the environment. The reason for this convention is the limit on the present capability to predict fission product transport out of the core following an accident. Defining the quantitative

release in this way does not imply a quantitative knowledge of the mechanism or transport phenomena of that release. The radiological characteristics of the NRC Categories of Fuel Damage presented in Table 3-1 remain applicable to the Comprehensive Procedure. The radiological characteristics are expanded upon in Table 6-1 to include the distribution of the fission products inside the containment building. This distribution is an estimate based upon NRC Regulatory Guide assumptions.

The core damage assessment procedure using hydrogen measurements is applicable in the NRC category of fuel overheat. Table 3-1 identifies the source of fission product release for this category to be the fuel pellet. The source inventory is calculated using the techniques described in Section 3.3. The source inventory includes all fission products and is not limited to the list provided in Table 3-4. The source inventory is expressed as an energy source term in Table 4-5 for hydrogen production following major fuel overheat. Values are provided for energy deposited in the coolant from sources mixed with the coolant and from sources in the fuel rods. The fraction of the total gamma energy deposited in the coolant from sources in the fuel rods is analytically determined using a Monte Carlo radiation transport model using the geometry of C-E designed fuel assemblies.

The rate constant for hydrogen produced as a function of energy absorbed is an experimentally determined value. Based upon a survey of the published experimental data, the value reported in Reference 7-21 of 0.3 molecules of hydrogen produced per 100 ev of energy absorbed is used in the realistic hydrogen production rate analysis. The radiolysis of water in a closed system

TABLE 4-5

ENERGY SOURCE TERMS FOR HYDROGEN PRODUCTION IN CONTAINMENT FOLLOWING MAJOR FUEL OVERHEAT

Energy Deposited in Coolant from Sources Mixed with the Coolant

Time After Shutdown (hr)	Beta Energy (MeV/Watt-sec)	Gamma Energy (Mev/Watt-sec)
1	2.26 (+9)*	6.55 (+9)
10	1.17 (+9)	3.20 (+9)
100	3.61 (+8)	1.10 (+9)
1000	4.55 (+7)	4.53 (+7)

Energy Deposited in Coolant from Sources in the Fuel Rods

Time After Shutdown (hrs)	Energy Deposited in Coolant (MeV/Watt-sec)	Fraction of the Total Gamma Energy in Fuel Rod Deposited in Coolant
1	2.91 (+9)*	0.076
10	9.25 (+8)	0.054
100	7.72 (+8)	0.072
1000	2.48 (+8)	0.066

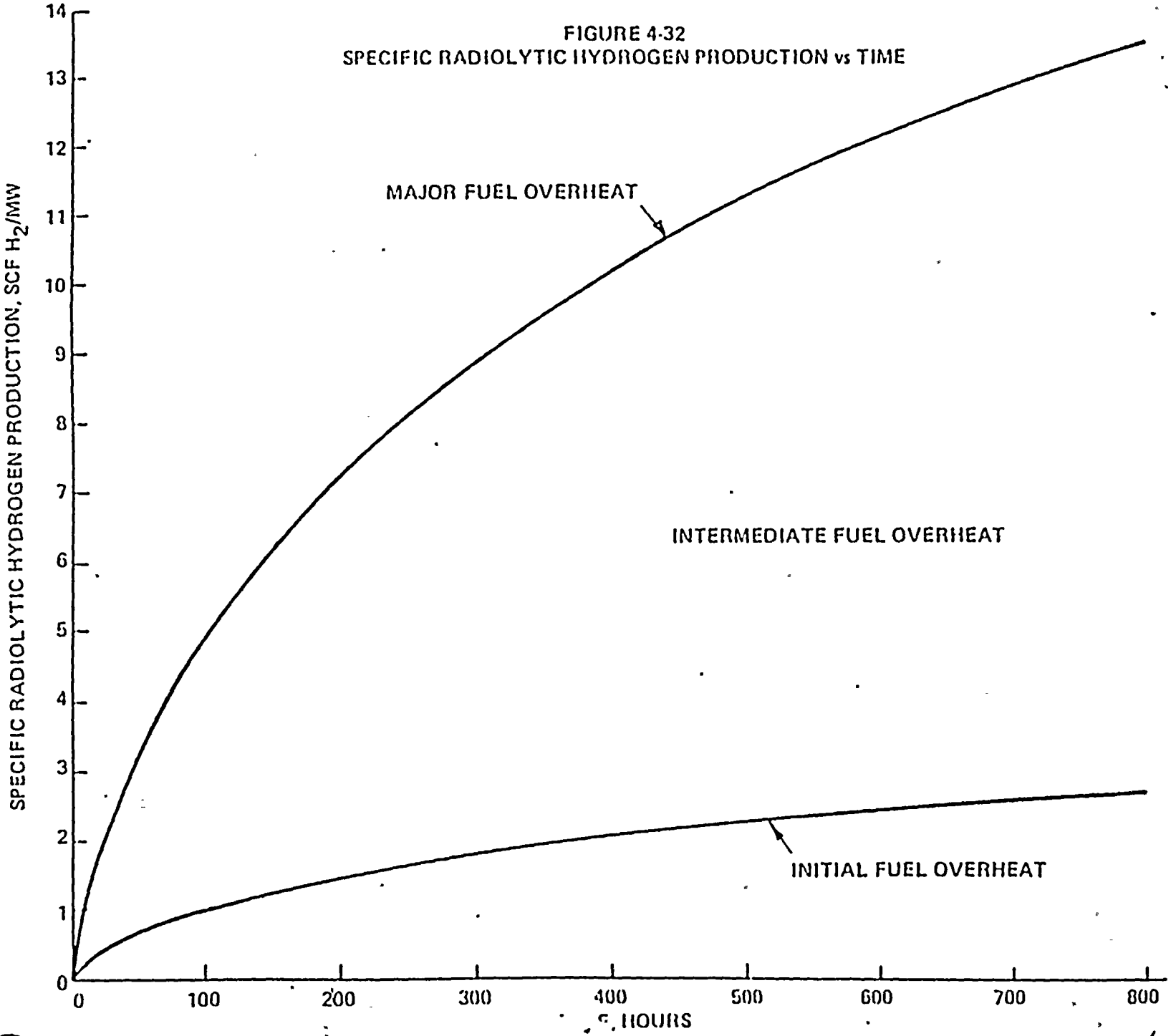
* Indicated powers of ten.

includes both the production and recombination of the byproduct gas. Therefore, the rate constant is known to exhibit some temperature dependence because of the indirect effect it has on the solubility of the gas byproducts. As temperature increases, the gas solubility will decrease and therefore, the rate of recombination will decrease. The value will be assumed to be a constant within the accuracy of this procedure.

The specific radiolytic hydrogen production rate calculated with the assumptions described above are provided in Figure 4-32 in the form of a graph. Two curves are shown in this figure because the hydrogen production is a function of the fission product release which is dependent upon the degree of core damage. The two curves represent the results of the calculation for major and initial fuel overheat using the radiological characteristics of these categories described in Table 6-1. The user of this procedure is required to rely upon an estimate of core damage obtained from one of the other procedures provided in this document to select the appropriate curve. The curves in Figure 4-32 are applicable to all plants because the values presented are normalized to power level.

Correction for variation in reactor power history prior to the accident is required because of the result upon the source fission product inventory. The correction is used to adjust the measured source inventory to the corresponding value had the plant been operating at full power. The analytically determined hydrogen production rates which are used to assess core damage are calculated assuming full power equilibrium source inventories. The techniques employed in this correction are the same as those discussed in Section 3.3. They are

FIGURE 4-32
SPECIFIC RADIOLYTIC HYDROGEN PRODUCTION vs TIME



the conditions that: equilibrium source inventory is reached after 30 days of constant power operation; the equilibrium value is directly proportional to the production rate expressed as reactor power; and that constant power operation means no change greater than ± 10 percent. The result is that a simple ratio of the power may be employed to obtain the full power equilibrium correction.

Constant reactor power operation is not anticipated. Therefore, engineering judgement is required to determine the reactor power which is most representative of the fission product inventory prior to the accident. Explicit determination of the representative value would require detailed computer code analysis. However, engineering judgement employed with the aid of specific guidelines is sufficient to yield a result within the accuracy of this procedure. These guidelines are as follows:

(1) The average power during the 30 day time period is not necessarily the most representative value for correction to equilibrium conditions.

(2) The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

(3) Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

5.0 ESTABLISHMENT OF THE BASES FOR CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES AND OTHER INSTRUMENTS

The objective in utilizing the Core Exit Thermocouples (CET)is to obtain another redundant indication of core damage. The CET can supplement the radiologic and hydrogen meaurements and can provide a damage assessment for events which are less severe than those which generate measurable amount of hydrogen. Two additional instruments indications, the pressure and the coolant level above the core are used to supplement the use of the CET in the determination of core damage.

The CET provide the primary indication of core heatup following the start of core uncovery. They function best for slow uncovery by boiloff. The pressure indication supplements the CET in providing information on the probable rate of core uncovery and therefore on whether the CET temperature is a valid representation of core temperature. Also, the pressure indication is used to reduce the uncertainty in the prediction of clad rupture temperature. The rupture temperature is a function of the differential between the internal gas pressure and the reactor coolant pressure during the period of core uncovery. The time period of core uncovery might be questionable when only the CET temperature is utilized to determine it. By using the reactor vessel level indication, another estimate of the uncovery period is obtained. The level provides the time period when the pressure should be observed and also can help in interpreting the transient behavior of the CET temperature. These three indications, pressure, level and CET temperature are discussed in the following three sections. The end result is a prediction of the number of

ruptured fuel rods, some interpretation on the probable accuracy of that prediction, and an assessment of clad damage expressed as one of the NRC categories of damage given in Table 4-1.

5.1 PRESSURE INDICATION

The rate of pressure decrease indicates the relative size of break for LOCA events. A large break LOCA causes the pressure to fall from the normal operating value to near containment pressure within about 100 sec. or less. This indicates complete core uncovery by blowdown, and is an indication that the CET temperature rise is probably much less than the core temperature rise. The damage assessment procedure using the CET could greatly underestimate core damage.

A small break LOCA causes the pressure to fall to within 100 pst above the secondary pressure, remain constant for a period of time which is proportional to break size and then to continue falling (See Reference 7-9). This indicates relatively slower core uncovery by boiloff, although functioning of the ECCS may prevent core uncovery. The CET respond best to such slower uncovery events and the CET temperature may be given greater weight in the core damage assessment.

The prediction of clad rupture using the CET temperature includes an assumption on the rupture temperature. This rupture temperature varies substantially with individual fuel rod burnup and system pressure. In order to remove some of the uncertainty, predictions are made for three postulated

rupture temperatures which are related to the system pressures. The pressure indication during the uncovery period enables the choice of the rupture temperature which yields the best damage prediction.

5.2 LEVEL INDICATION

The coolant level in the reactor vessel above the top of the core is used to estimate the period of core uncovery. Knowledge of when the core is uncovered is used in two ways in the procedure. First, it determines the time period during which the pressure should be observed as dicussed in the preceding section. Second, when the core is uncovered the CET temperature should rise. If it does not rise and fall according to the core uncovery period then some interpretation of the validity of the CET temperature as an indicator of core temperature is required. Hence, the level indication can help to reduce the possible uncertainty in the assessment of the core damage obtained with the CET temperature or at least can suggest that there is a potentially large underestimate of the predicated damage. The relationships between CET temperature and level are discussed in the following Section 5.3.

There are two commercially available reactor vessel level systems available. Neither indicates all they way down to the top of the active fuel, so some procedure for extrapolating the recorded traces of level may be needed in the plant specific procedure for core damage assessment in order to obtain a good estimate of the core uncovery period. Such plant specific details are out of

scope for this generic procedure guideline. Owners of C-E designed reactors with the C-E level monitoring system can obtain a performance evaluation during accidents from Reference 7-15, sponsored by the CEOG.

5.3 PREDICTION OF FUEL CLAD RUPTURE BASED ON CORE EXIT THERMOCOUPLE TEMPERATURES

A performance evaluation of CET's during core uncovery is provided for C-E designed reactors in CE-NPSD-212 (Reference 7-16) prepared for the C-E Owners Group. That report should be used to supplement the presentation here. It is assumed here that a Core Exit Thermocouple instrumentation system as required in Regulatory Guide 1.97 is available. In addition, the system capability must allow the user of the procedure to obtain the maximum CET temperature and superheat as a function of time during the uncovery period. Plant specific implementation may require additional hardware and/or instruction on how to obtain these data from the instrumentation system. Given a system, several considerations on the performance of the CET during core uncovery are presented and are related to the interpretation of CET temperature as a measure of core clad temperature.

The CET's are located depending on reactor design from 10 in. to 26 in. above the top of the active fuel. They do not monitor fuel temperature. They do monitor the temperature of fluid above the exit of the core which is liquid normally and steam and/or gas during uncovery accidents. In order to infer core temperature from the CET indication, there must be a flow of heated fluid from the core past the CET elevation. During complete core uncovery or

temporarily following any rapid partial uncovery, there is no flow to couple the core and CET temperatures, so the core heats up without a CET indication. During slower uncovery by boiloff, the superheated steam flowing up from the core provides effective thermal coupling and core heatup is indicated by the rising CET temperature. The information which is obtained from the CET record for both of these two cases is discussed in the following.

In the first case, while the core is uncovered, there is no thermal coupling but some useful information may be obtained from the CET temperature record during the recovery period.

During a rapid recovery, the liquid entering the bottom of the core pushes the steam and gas up and out of the core, past the CET. This may cause a sudden, short duration peak in the CET temperature when it follows a deep uncovery. It is an indicator of more severe prior core heatup than would be inferred from the earlier CET temperature record. Tese data from LOFT and Semiscale LOCA tests which are summarized in Reference 7-16 confirm this behavior. The conclusion is that such a scenario yields a CET temperature record which has some characteristic behavior that indicates the potential for higher core temperatures and therefore greater core damage than this procedure would predict.

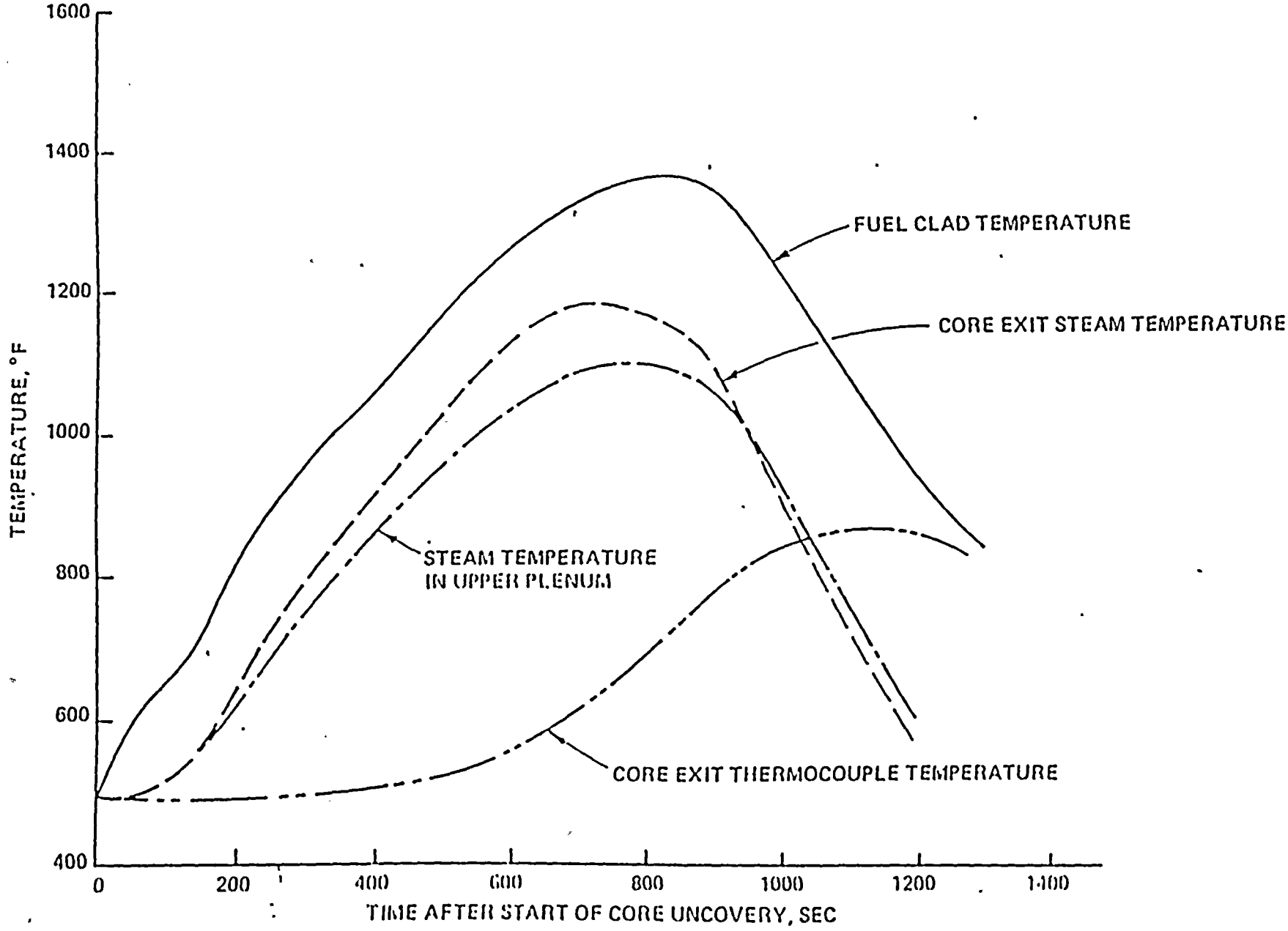
In the second case, when there is thermal coupling between core temperature and CET temperature, the effectiveness depends on the particular CET installation configuration and on the time rate of steam temperature changes. Both of these contribute to the amount of time lag in the CET response. Some

analytical calculations of the time response of CET in C-E designed reactors are given in Reference 7-16. This thermal lag is also evident in the tests at the LOFT and Semiscale facilities, which are summarized in Reference 7-11. For the C-E designed reactors, this lag varies with reactor design, up to about 6 minutes. It has the effect of reducing the peak temperature recorded from the CET and possibly of shortening the apparent duration of core uncovery. For example, Figure 5-1 shows the calculated steam temperature as a function of time for a 0.1 ft^2 small break LOCA and the delayed CET response temperature. The conclusion is that the CET recorded temperature represents a lower limit for the peak steam temperature and could be substantially lower than the peak core temperature. Therefore, this procedure will yield a lower limit prediction of core damage.

Typical analyses of Design Basis small break LOCA predict continuously decreasing pressure as the core uncovers. This can cause a delay in the apparent time of uncovery as indicated by the CET temperature. The reason is that the saturation temperature is also decreasing along with the pressure decrease. Therefore, the level could drop into the core and cause superheated steam at the top of the core but the temperature of the steam increases more slowly. The conclusion is that it is desirable to also trend the superheat derived from CET temperature and reactor coolant pressure.

The superheat of the CET also has a disadvantage for indicating the time of core uncovery. This disadvantage comes from the location of the CET above the top of the core combined with the thermal lag in the temperature response of the CET in a steam environment. The CET is at saturation temperature until

FIGURE 5-1
TEMPERATURE vs TIME AFTER START OF CORE UNCOVERY DURING
0.10 FT^2 LOCA, AVERAGE ASSEMBLY RESPONSE IN 2700 MWT SERIES



the level drops below the CET location. Thereafter, the CET temperature remains about the same or changes slowly whereas the saturation temperature is falling with pressure. Hence, the CET indicates superheat and a false indication of core uncovery. Fortunately, these effects are usually small and last only several minutes in the design basis LOCA analyses, until the core uncovers and temperatures rise substantially. It would probably be important only in the less severe scenarios where there is some question about whether or not the core started to uncover.

During recovery, the CET rapidly quenches when the coolant level rises above the CET elevation. The CET indication drops to saturation temperature or lower, depending on the rate of coolant flow. The conclusion is that if the CET indicate superheat, they are certainly uncovered and the core is either uncovering or is about to start uncovering. If the CET are subcooled and a level is indicated or if they suddenly go from superheat to subcooling, they and the core are covered.

The CET are generally designed to function for accidents which are less severe than the TMI-2 accident. They have good accuracy up to readings about 1650°F and can trend up to about 2300°F. The thermal-hydraulic functional design objectives for the CET and other inadequate core cooling instrumentation are provided in CE-NPSD-199 (Reference 7-17) for the CEOG. A comparison with the bottom entry CET systems on the S-80 design and on Maine Yankee is provided in CE-NPSD-171 (Reference 7-18). Their functional capability is essentially the same as for top mounted CET up to core temperatures of 2300°F.

In order to utilize the CET recorded temperature, three relations are necessary. First, the peak recorded CET temperature must be related to the peak core temperature. Second, the peak core temperature must be related to the distribution of temperature in the core. Third, the core temperature distribution must be related to an assessment of core damage. These three relations are discussed in order.

The relationship between the peak core temperature and the CET temperature is indicated by an example. Analyses of Design Basis small break LOCA events on the 2700 Mwt class indicate that when the core peak clad temperature during an event is in the range 1500 to 1700°F, the maximum subchannel exit steam temperature is about 250°F to 100°F lower, respectively. The magnitude of the CET temperature would be lower than the subchannel steam temperature because of the thermal lag and steam cooling above the active fuel and because the CET sees steam from an array of subchannels. The average temperature from the array is lower than the peak subchannel steam temperature.

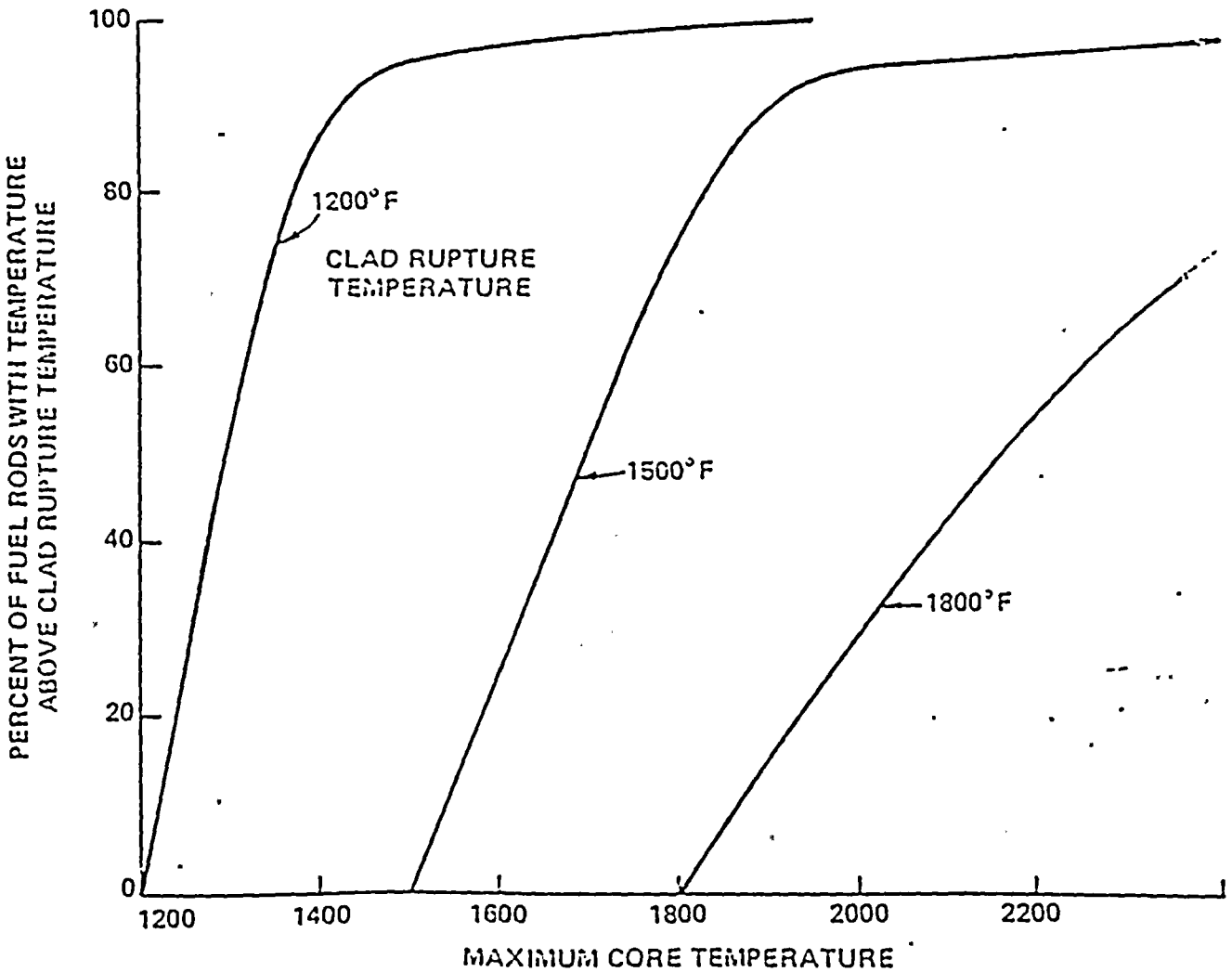
The CET temperature lags the steam temperature by up to about 6 min. When the duration of uncovery is long compared to the lag time and the temperatures vary smoothly, the CET temperature is closer to the steam temperature. At about 1500°F steam temperature, the peak core temperature attained during Design Basis Events could be within about 100°F above the peak steam temperature. The conclusion is that for smooth CET temperature records which peak near the upper limit of the CET range of good accuracy at 1650°F, and for uncovery durations of 20 min or longer, the peak core temperature attained is within a few hundred degrees above the peak recorded CET temperature.

Secondly, the peak core temperature is related to the distribution of temperature on all the rods in the core by the analyses described previously in Section 4. An example of how the desired relationship is developed is given as follows. Assume the peak core temperature is 1950°F, as shown in Figure 4-11 when the coolant level is 30%. The figure shows that all rods with radial peaks above 0.75 have temperatures above 1500°F. From Figure 4-10, about 94% of the rods in the core have radial peaks above 0.75. If clad rupture occurred at 1500°F, then a peak core temperature of 1950°F is an indicator that 94% of the rods are ruptured. In this manner, the third relation between peak temperature and core damage is established. Successive values of peak core temperature are used and yield various percentages of ruptured rods. If the rupture temperature is 1800°F, another set of values is obtained.

The results of this calculational routine are shown on Figure 5-2. Three values of rupture temperature are selected for plotting, 1200°F, 1500°F and 1800°F. As discussed in Section 4.3.1 on clad rupture criteria, the clad will rupture when held at temperatures in this range for about ten minutes, depending on the clad differential pressure. For new fuel, these rupture temperatures correspond to coolant pressures of 100 psia, 1100 psia and all pressures respectively.

In utilizing these curves in the procedure, the first step is to determine the coolant pressure during core uncovery, as discussed in Sections 5.1 and 5.2. The pressure determines which of the three curves best represents the clad rupture conditions. Use the curve corresponding to the measured pressure or

FIGURE 5-2
PERCENT OF FUEL RODS WITH RUPTURED
CLAD vs MAXIMUM CORE TEMPERATURE



higher. Observe the maximum CET temperature during the uncovery period. The peak core temperature is at least this high, and could be considerably higher depending on the scenario as discussed perviously. Enter Figure 5-2 with the peak temperature and read from the appropriate rupture curve the percentage of rods which are ruptured. This is considered a lower limit estimate of the percent of clad ruptures. Table 4-1 then yields the NRC category of core damage corresponding to the amount of clad failure.

It is evident from Figure 5-2 that the number of ruptured rods is very sensitive to the core peak temperature, and is even more sensitive to the assumed rupture temperature. The uncertainties in the damage assessment can be reduced with plant specific analyses. At any time during a given fuel cycle, the burnup and the consequent fission gas pressure vary among the fuel rods. Burnup is somewhat related to radial peaking factor. For example, third cycle fuel has the maximum internal pressure at a given temperature, but it usually does not have maximum radial peaking factors. Plant specific calculations for a given burnup cycle are feasible which could reduce the three curves for three rupture temperatures to one curve at a given core burnup and system pressure. Several curves could be derived for several values of burnup during the fuel cycles and for a set of system pressures. Such curves would reduce the uncertainty in the percent of rods which rupture at a given core peak temperature attained during a core uncovery accident.

The curves in Figure 5-2 provide a lower limit estimate of the number of ruptured rods for a given peak CET temperature during uncovery by boiloff. For example, at an observed peak CET temperature of 1600°F and a rupture

temperature of 1500°F, the figure predicts 25% ruptured rods. If the actual peak core temperature were 200° above the CET temperature, the correct answer would be 75% ruptured rods. As the temperature increases the uncertainty decreases. At a peak CET temperature of 2200°F, over 50% of the rods are predicted ruptured with high confidence, regardless of burnup or system pressure.

If the maximum temperature CET exceeds the upper limit of indication at 2300°F, no more specific statement about core damage is available from the maximum CET recorded trace. A core map of all CET temperatures is needed to indicate the number of CET above 2300°F, their radial locations and the racial nuclear peaks at those locations. With these data, an estimate of the number of rods which are above the clad melt temperature or which have exceeded the oxidation embrittlement threshold can be made. For example, if the CET in locations with core average radial peaks exceed 2300°F, at least 5% of the rods in the core have melted clad. The damage assessment procedure presentec in the appendix does not include the utilization of such core maps of CET temperature and radial peak. Plant specific information on data acquisition capabilities and more detailed analyses of more severe core damage scenarios are required to extend this procedure to include the measured distribution of core exit temperature.

6.0 ESTABLISHMENT OF THE BASIS FOR CORE DAMAGE ASSESSMENT USING DOSE RATE MEASUREMENT INSIDE THE CONTAINMENT BUILDING

The categories of core damage are characterized in Section 3.0 of this document in terms of those physical parameters relevant to the release of radioactive material from the core into the containment building. These parameters are the source of the fission product release, the mechanism of fission product release, and the quantitative release of characteristic fission products expressed as a percent of the theoretical source inventory. That information is used in the development of the Interim Procedure for Core Damage Assessment. The measurement employed for the purpose of the interim procedure was the specific activity of the samples obtained from the reactor coolant and from the containment building atmosphere. A discussion of the uncertainty in these measurements and the limitations of their use in core damage assessments is provided in Section 3.4.

The same physical parameters may be evaluated using the measurements of the dose rate inside the containment building. The use of two different measurements for the evaluation of the same physical parameters is employed as a means to reduce the uncertainty. As with all of the physical and chemical -- parameters discussed in this procedure guideline, the assessment of core damage based on one parameter serves to compliment the assessment based on the others reducing the overall uncertainty. This section describes the basis employed in the Comprehensive Procedure for use of the dose rate measurement inside the containment building to assess core damage.

The instrument employed in the measurement is the wide range area radiation monitor. There may be several of these instruments inside the containment building. They are typically located in the higher elevations of the building in order to measure the volatile fission products as they disperse through the building atmosphere. The dispersion of volatile fission products through the building atmosphere is increased by air flow resulting from operation of the HVAC system and by thermal gradients or steam flow. Chemical sprays are used to remove non-volatile and halogen fission products from the atmosphere.

As a consequence of the identity of the fission products to which these instruments are exposed, dose rate measurements are most applicable to core damage assessment within the categories of fuel cladding failure and fuel overheat. The fission products of high volatility, specifically noble gas and halogens, are characteristic of cladding failures. The fission products of intermediate volatility, cesium, telurium, and rubidium, are characteristic of fuel overheat. Dose rate measurements are not considered applicable to the categories of fuel melt because the fission product transport under this condition is poorly understood and therefore realistic dose rate correlations cannot be performed. The measurement of sample specific activity is more indicative of the fuel melt condition. For these reasons the core damage assessment using dose rate measurements will be limited to within the upper range of fuel overheat.

The convention of this document is to express the release of fission products as the percent of the source inventory at the time of the accident which is

observed to be present in the sampled or measured media. The radiologic characteristics of the NRC categories of fuel damage are defined in Tables 3-1 and 6-1.

6.1 ANALYSIS OF IN-CONTAINMENT DOSE RATE

The procedure for core damage assessment is based upon the comparison between dose rates measured following an accident and analytically determined values of the realistic or best estimate dose rates that would correspond to the specific categories of core damage. The radiation dose rates inside the containment building following an accident are dependent upon many variables which include reactor power, fuel burnup, containment building geometry, the identity and quantity of the fission products released from the core, the use of chemical sprays for the removal of airborne fission products, and the location within the building at which they are measured. Each of these variables are plant specific. Therefore, implementation of this procedure requires the development of plant specific analytically determined dose rates which is beyond the scope of this document. Such plant specific dose rates may already be developed on a case basis for other purposes. However, the subject procedure is a document intended to provide an actual assessment of core damage for the purpose of implementing emergency operational decisions following an accident. Therefore, care should be taken to employ the results of realistic or best estimate dose rate analyses rather than conservative information which may have been developed for such purposes as licensing activities or the design bases for equipment environmental qualification. The use of dose rate analyses based on conservative assumptions could actually

Table 6-1

Radiologic Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release From Core	Source of Release	Percent of Source Inventory Released to Containment	Distribution of Fission Products in Containment
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	Less than 1	Airborne
2.	Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Less than 10	Airborne
3.	Intermediate Cladding Failure		Gas Gap	10 to 50	Airborne
4.	Major Cladding Failure		Gas Gap	Greater than 50	Airborne
5.	Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Less than 10	Airborne: 100% Noble Gas 25% Halogen Plated Out 25% Halogen 1% Solids
6.	Intermediate Fuel Pellet Overheating		Fuel Pellet	10 to 50	
7.	Major Fuel Pellet Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet	Greater than 50	

result in a lower than actual assessment of core damage. This is because the conservative assumptions dictate greater consequences for the release of fission products than may be actual for a given category of core damage. The measurement of the lower or realistic dose rate would then be correlated to a lower than actual category of core damage. The purpose of this section is to describe the analytical assumptions recommended to be employed in that realistic analytical development and to provide an example of the results of such an analysis. Plant specific implementation may require the use of assumptions different from those employed in the example. Therefore, all of the assumptions used and the basis for each will be defined. The specific example shown is as generic as possible. Plant specific implementation may be performed by analytical correction of the example dose rates rather than an independent calculation.

A number of the variables required in the analyses are well defined. These variables are the design reactor power, equilibrium core inventory of fission products, containment building geometry, and the location of the wide range area radiation monitors used to measure the dose rate. There are however several variables which require the use of analytical assumptions. These variables are the quantitative release of fission products from the core under the defined conditions of fuel cladding failure and fuel overheat; their geometric distribution inside the containment building, and the consequences of the chemical sprays for the reduction of airborne fission products. These analytical assumptions are discussed in detail below. Additionally, a method is required to correct the measured dose rates for power history prior to the accident because of the result upon the source fission product inventory.

The dose rate measurement is dependent directly upon the quantity of the fission products released from the core. The convention employed to express that quantity is the same as that described in Section 3.3 for use in the Interim Procedure. The quantitative release of fission products is expressed as the percent of the source inventory at the time of the accident which is observed to be present in the sampled media and therefore available for immediate release to the environment. The reason for this convention is the limit on the present capability to predict fission product transport out of the core following an accident. Defining the quantitative release in this way does not imply a quantitative knowledge of the mechanism or transport phenomena of that release. The radiological characteristics of the NRC Categories of Fuel Damage presented in Table 3-1 remain applicable to the Comprehensive Procedure. The radiological characteristics are expanded upon in Table 6-1 to include the distribution of the fission products inside the containment building. This distribution is an estimate based upon NRC Regulatory Guide assumptions.

The categories of core damage identified as cladding failure are characterized in Table 6-1 by the release of fission products through the mechanisms of clad burst and gas gap diffusion. The characteristic fission products are the noble gases and halogens. These fission products are released in highly volatile chemical species. The dose rate analysis for the categories of cladding failures will assume that the fission products remain airborne. The quantitative release criteria to distinguish the extent of fuel cladding damage are also provided in Table 6-1 expressed as a percent of the source inventory. These values are the recommended assumptions for use in the

analytical determination. The source inventory in this case is the equilibrium gas gap inventory. This inventory includes all noble gas and halogen fission products and is not limited to the list provided in Table 3-5 in which the characteristic isotopes are identified for use in sample analysis. The implementation of these defined criteria in the form of a procedure require the calculation of the dose rate as a function of time for the two cases of 50 percent and 10 percent assumed quantitative release from the source into the containment building atmosphere.

The categories of core damage identified as fuel overheat are characterized in Table 6-1 by the release of fission products through grain boundary diffusion and by diffusion from within the UO_2 grains. The characteristic fission products are cesium, rubidium, and tellurium. The fission products released in this category include the less volatile chemical species driven off by the high temperatures in addition to increased quantities of the highly volatile species discussed with regard to cladding failure. Therefore, the dose rate analysis will assume that the distribution includes both airborne dispersion and surface plateout inside the containment building. The recommend assumptions are that: the airborne dispersion will include 100 percent of the noble gas and 50 percent of the halogen fission products which have been assumed to be released; and the plateout will include 25 percent of the halogen and 1 percent of the solid fission products which have been assumed to be released. These values represent the percentage of that which has been released and are not the percentage of the source inventory. The percentage of the source inventory which is released for each category is a characteristic defined in Table 6-1.

The airborne dispersion is assumed to be homogeneous and the plateout is assumed to be on the walls of the containment building. The quantitative release criteria to distinguish the extent of fuel cladding are also provided in Table 6-1 expressed as a percent of the source inventory. These values are the recommended assumptions for use in the analytical determination. The source inventory in this case is the equilibrium core inventory. This inventory includes all fission products and is not limited to the list provided in Table 3-4 in which the characteristic isotopes are identified for use in sample analysis. The implementation of these defined criteria in the form of a procedure require the calculation of the dose rate as a function of time for the two cases of 50 percent and 10 percent assumed quantitative release from the source into the containment building.

Operation of the Containment Spray System has the effect of reducing the airborne concentration of the halogen fission products. Evaluation of this effect requires assumptions concerning the distribution of chemical species among the halogen fission products and the efficiency of the spray to remove each from the atmosphere. NRC Regulatory Guide 1.4 assumptions are recommended for the distribution of chemical species. The efficiency of the chemical spray is expressed as the decontamination factor (DF) after two hours of operation. It is assumed that those fission products remaining airborne after two hours will be unaffected by continued operation of the spray. These assumptions are explicitly stated as follows: 91 percent of the halogens are of the elemental species which has a 2 hour DF of 20; 4 percent of the halogens are of the organic species which has a 2 hour DF of 1; and 5 percent are of the particulate species which has a 2 hour DF of 1. The combined

effect is that after two hours of containment spray operation 13.5 percent of the original quantity of halogen remains airborne. Expressed differently the halogens are assumed to be reduced by a factor of 7 after two hours of containment spray.

The validity of these assumptions were evaluated by a series of parametric dose rate calculations performed with the halogen removal factor varied between 1 and 20. The result indicated that increasing the removal factor to 20 (assuming all halogens to be elemental species) would lower the dose rate measured five hours after the accident to one-third of the value calculated with a removal factor of 7. Therefore, the assumptions used for the efficiency of the containment spray is a significant factor in the accuracy of the procedure to estimate core damage. Additionally, it must be assumed that the distribution of chemical species is homogeneous throughout the containment atmosphere and that the radiation monitors are located in a homogeneous region which is subjected to the spray.

Correction of the measured dose rates for variation in reactor power history prior to the accident is required because of the result upon the source fission product inventory. The correction is used to adjust the measured dose rate to the corresponding value had the plant been operating at full power. The analytically determined dose rates which are used for comparison to assess core damage are calculated assuming full power equilibrium source inventories. The techniques employed in this correction are the same as those discussed in Section 3.3. They are the conditions that: equilibrium source inventory is reached after 30 days of constant power operation; the equilibrium value is

directly proportion to the production rate expressed as reactor power; and that constant power operation means no change greater than ±10 percent. The result is that a simple ratio of the power may be employed to obtain the full power equilibrium correction.

Constant reactor power operation is not anticipated. Therefore, engineering judgement is required to determine the reactor power which is most representative of the fission product inventory prior to the accident. Explicit determination of the representative value would require detailed computer code analysis. However, engineering judgement employed with the aid of specific guidelines is sufficient to yield a result within the accuracy of this procedure. These guidelines are as follows:

(1) The average power during the 30 day time period is not necessarily the most representative value for correction to equilibrium conditions.

(2) The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

(3) Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

The case considered for the example is a 3800 Mwt plant. The dose rates are provided for two containment building geometries: the spherical containment has a volume of 3.3×10^6 cubic feet and a surface area of 1.2×10^5 square feet; the cylindrical containment has a volume of 2.7×10^6 cubic feet and a

surface area of 1.2×10^5 square feet. The assumptions include an equilibrium two year burnup at full reactor power, the removal of airborne iodine by the containment building spray system results in an average reduction by a factor of 7 two hours after spray initiation, and location of the dose rate measurement is top centerline of the containment building. The results of the example analyses are provided in Figure 6-1 and 6-2. These calculations were performed using the point kernal technique to determine the gamma flux. Several computer codes are available to perform this analysis.

6.2 GENERAL DISCUSSIONS ON THE LIMITATIONS OF THE PROCEDURE

This procedure is limited in applicability to the NRC categories of fuel cladding and fuel overheat. The procedure is limited in accuracy to that of the assumptions made in the analytical dose rate determination. These considerations have been discussed above. Additionally, there are limitations in the radiation monitoring system capability to obtain representative measurements which should be considered in the determination of core damage using measured dose rates.

This procedure relies upon radiation dose rate measurements taken from one or more monitors located inside the containment building to determine the total quantity of fission products released from the core and therefore available for release to the environment. The amount of fission products present at the location of the monitors may be changing rapidly due to transient plant conditions. Therefore multiple measurements should be obtained within a minimum time period and when possible under stabilized plant conditions.

FIGURE 6-1
TYPICAL ANALYSIS FOR POST ACCIDENT
DOSE RATE INSIDE A SPHERICAL CONTAINMENT

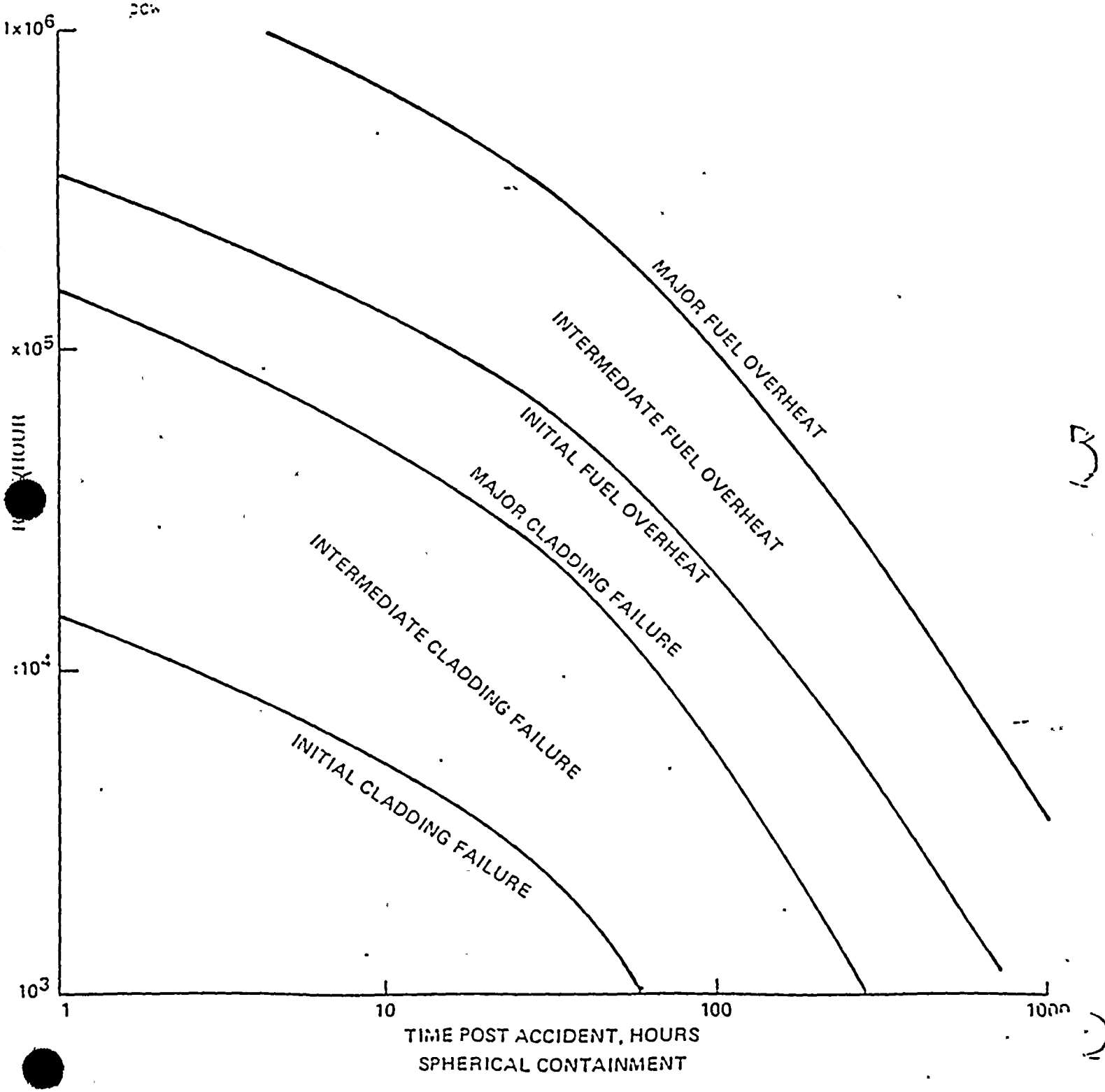
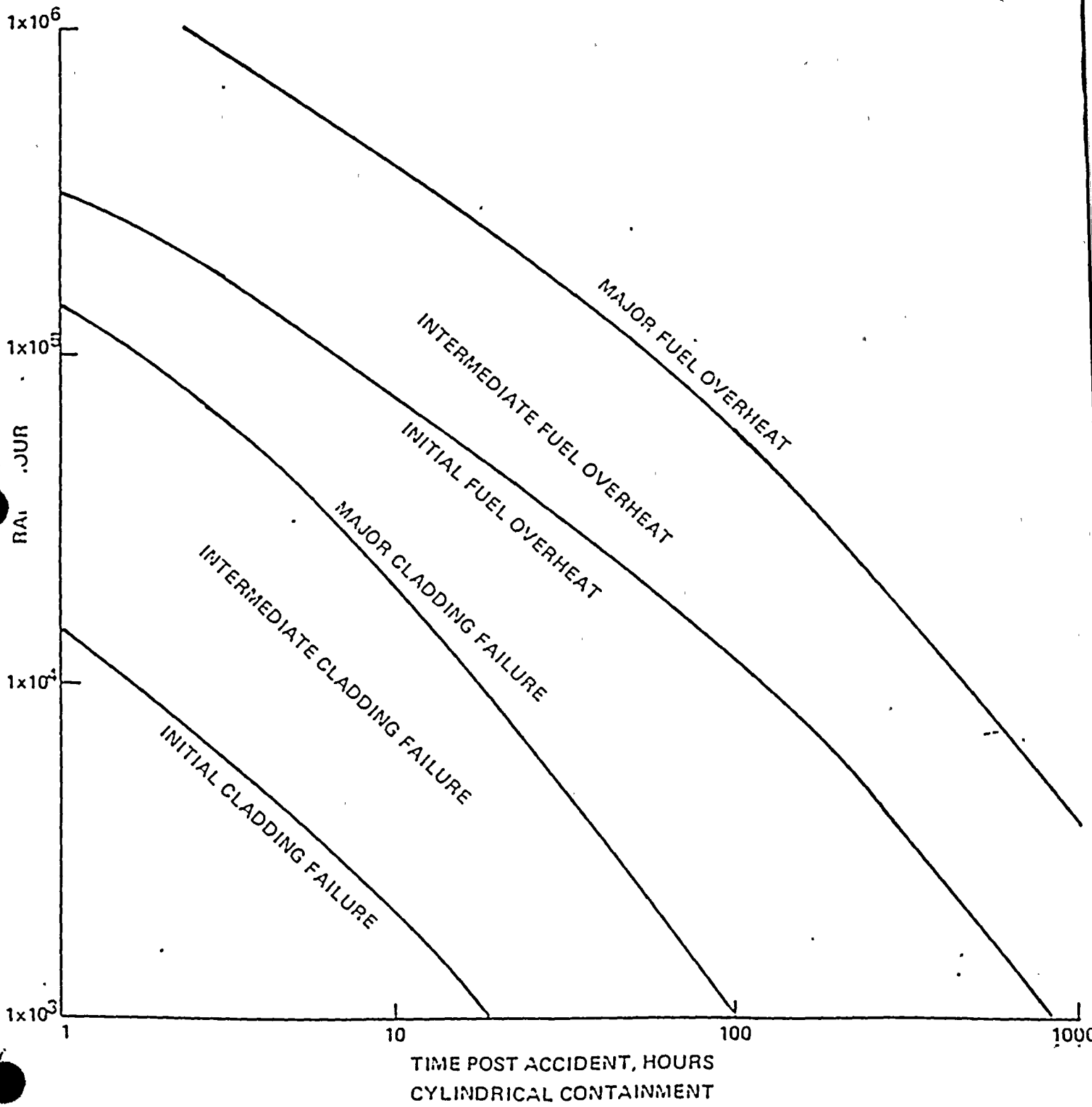


FIGURE 6-2
TYPICAL ANALYSIS FOR POST ACCIDENT
DOSE RATE INSIDE A CYLINDRICAL CONTAINMENT



Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

The reliability of the measured dose rates is influenced by a number of factors which include: the ability to obtain representative measurements due to incomplete mixing of the measured media; equipment operation in a harsh environment; and operator familiarity with rarely used procedures.

7.0 REFERENCES

7.1 Clarification of TMI Action Plan Requirements NUREG-0737 dated November, 1980.

7.2 Post Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage, US NRC. (Included here as Appendix A)

7.3 Release of Fission Products From Fuel in Postulated Degraded Core Accidents IDCOR Subtask 11.1 Draft Final Report dated July, 1982.

7.4 A Report to the Commission and to the Public, NRC Special Inquire Group. Mitchell Rogovin Director

7.5 CEN-153-P Evaluation of Instrumentation for Detection of Inadequate Core Cooling in C-E NSSS. May 1981.

7.6 ORIGEN Isotope Generation and Depletion Code Oak Ridge National Laboratory CCC-217.

7.7 EPRI NP-1022, "Requirements for Analysis of Transient Fuel Rod Behavior During Design Basis Events," Prepared by Combustion Engineering, Inc., for EPRI, March, 1979.

7.8 EPRI NP-1999, "Factors Affecting Post-DNB Operation for Light Water Reactors," Volume 1, Prepared by Combustion Engineering, Inc., for EPRI, August, 1981.

7.9 CEN-114-P, Amendment 1-P, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, July, 1979.

7.10 CEN-115-P, "Response to NRC IE Bulletin 79-06C, Items 2 and 3, for Combustion Engineering Nuclear Steam Supply Systems," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, August, 1979.

7.11 H. Ocken, "An Improved Evaluation Model for Zircaloy Oxidation," Nuclear Technology, Vol. 47, February, 1980, pp. 343-357.

7.12 H. M. Chung and T. F. Kassner, "Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light-Water Reactors: Summary Report," NUREG/CR-1344, ANL:-79-48, January, 1980.

7.13 R. K. Cole, Jr., "Generation of Hydrogen During the First Three Hours of the Three Mile Island Accident," NUREG/CR-0913, SAND79-1357, July, 1979.

7.14 NSAC-29, "The Retarding Effect of Hydrogen on Zircaloy Oxidation," July, 1981.

7.15 "Analysis of HJTC/RVLMS Performance During Accident Conditions," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, Draft Issued April, 1983.

7.16 C-E NPSD-212, "Performance Evaluation of Core Exit Thermocouples as Inadequate Core Cooling Instrumentation," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, March, 1983.

7.17 C-E NPSD-199, "Generic Thermal-Hydraulic Functional Design Objectives for Inadequate Core Cooling Instrumentation," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, Draft Issued, March, 1983.

7.18 C-E NPSD-171, "Comparison of Bottom-Mounted Core Exit and In-Core Thermocouples to Detect Inadequate Core Cooling," Prepared by Combustion Engineering, Inc., for the C-E Owners Group, March, 1983.

7.19 "The Corrosion of Materials in Reactor Containment Spray Solutions" Griess and Bacarella Nuclear Technology, Vol. 10, April 1971.

7.20 BNL-NUREG-24532 "Hydrogen Release Rates from Corrosion of Zinc and Aluminum", May 1978.

7.21 Fletcher, W. D., Bell, M. J., Marchese, R. T. and Gallagher, J. L., "Post LOCA Hydrogen Generation in PWR Containments," Nuclear Technology, Volume 10, April, 1971.

APPENDIX A

NRC POST ACCIDENT SAMPLING GUIDE FOR PREPARATION OF A PROCEDURE TO ESTIMATE CORE DAMAGE

POST-ACCIDENT SAMPLING

GUIDE FOR PREPARATION OF A PROCEDURE

TO ESTIMATE CORE DAMAGE

The major issue remaining to complete our evaluation of NTOL's for compliance with the post-accident sampling criteria of NUREG-0737 is preparation of procedures for relating radionuclide concentrations to core damage. To date, none of the applicant's has been successful in providing an acceptable procedure. As a consequence, each NTOL has a license condition which may restrict power operations. One of the contributing factors in the applicant's slow responses to this item is their confusion on exactly what to prepare. The attachment is intended to provide informal guidance to each NTOL applicant so that their procedures, when prepared, will address the core damage estimation in a manner acceptable to us.

We anticipate that preparation of a final procedure for estimating core damage may take approximately 12 months. Therefore, we are willing to accept an interim procedure which focuses on fewer radionuclides than are indicated in the attachment. The interim procedure in conjunction with a firm date for the final procedure would be used to remove the power restricting license condition.

The primary purpose in preparing a procedure for relating radionuclide concentrations to core damage is to be able to provide a realistic estimate of core damage. We are primarily interested in being able to differentiate between four major fuel conditions; no damage, cladding failures, fuel overheating and core melt. Estimates of core damage should be as realistic as

possible. If a core actually has one percent cladding failures, we do not want a prediction of fifty percent core melt or vice versa; extremes in either direction could significantly alter the actions taken to recover from an accident. Therefore, the procedure for estimating core damage should include not only the measurement of specific radionuclides but a weighted assessment of their meaning based on all variable plant indicators. The following discussion is intended to provide general quidance pertaining to the factors which should be considered in preparing a procedure for estimating core damage but is not intended to provide an all inclusive plant specific list.

The rationale for selecting specific radionuclides to perform "core damage estimates from fission product release" is included in the Rogovin Report (page 524 through 527, attached). Basically, the Rogovin Report states that three major factors must be considered when attempting to estimate core damage based on radionuclide concentrations.

1. For the measured radionuclides, what percent of the total available activity is released (i.e. is only gap activity released, is sufficient activity released to predict fuel overheating or is the quantity of activity released, only available through core melt?

2. What radionuclides are not present (i.e. radionuclides will, in all probability, not be released unless fuel overheating or melt occurs). The absence of these species bounds the maximum extent of fuel damage.

What are the ratios of various radionuclides species (i.e. the gap activity ratio for the various radionuclides may differ from the ratio in the pellet). The measurement of a specific ration will then indicate whether activity released came from the gap of fuel overheating/melt.

In addition to the radionuclide measurements, other plant indicators may be available which can aid in estimating core damage. These include incore temperature indicators, total quantity of hydrogen released from zirconium degradation and containment radiation monitors. When providing an estimate of core damage the information available from all indications should be factored into the final estimate (i.e. if the incore temperature indicators show fuel overheat and the radionuclides concentrations indicate no damage, then a recheck of both indications should be performed).

Consistent with the categorization of fuel damage in the Rogovin Report, the four major categories of fuel damage can be further broken down, similar to the following list, consistent with state-of-the-art technology. The suggested categories of fuel damage are intended solely to address fuel integrity for post-accident sampling and do not pertain to meeting normal off-site doses as a consequence of fuel failures.

1. No fuel damage
2. Cladding failures (<10%)
3. Intermediate cladding failures (10%-50%).
4. Major cladding failures (>50%)
5. Fuel pellet overheating (<10%)
6. Intermediate fuel pellet overheating (10%-50%).

7. Major fuel pellet overheating (>50%)
8. Fuel pellet melting (<10%)
9. Intermediate fuel pellet melting (10%-50%).
10. Major fuel pellet melting (>50%).

Because core degradation will in all probability not take place uniformly, the final categories will not be clear cut, as are the ten listed above. Therefore, the preparation of a core damage estimate should be an iterative process where the first determination is to find which of the four major categories is indicated (for illustrative purposes, only radionuclide concentrations will be considered in the following example, but as indicated above, the plant specific procedure should include input from other plant indicators). Then proceed to narrow down the estimate based on all available data and knowledge of how the plant systems function.

Example

In a given accident operation, there is 70% clad failure, significant fuel overheating and one fuel bundle melted. Utilizing the iterative process.

First

Calculate the maximum fuel melted by arbitrarily attributing all activity to fuel melt (under these conditions, five to ten melted bundles may be predicted). Therefore, the worst possible condition is fuel pellet melting.

Second

Calculate the maximum fuel overheated, by arbitrarily attributing all activity to fuel pellet overheating (under these conditions, major fuel pellet overheating is predicted).

Third

Calculate the maximum cladding failures, by arbitrarily attributing all activity to cladding failures (under these conditions, greater than 100% fuel cladding damage is predicted).

At this point it is obvious that major cladding damage is present and that a large amount of fuel pellet overheating has occurred with the potential for some minor fuel pellet melting.

Fourth

Check for the presence of radionuclides which are indicators of fuel pellet melting and overheating. In this instance, obvious of overheating will exist along with trace indicators of potential pellet melt.

Fifth

Based on the radionuclide indicators of fuel pellet overheating damage (confirmed by incore temperature) make an estimate of how much fuel

overheated. This result will in all probability indicate major fuel pellet overheating.

Sixth

Subtract the activity estimated from fuel pellet overheating, plus the activity attributable to 100% gap release from the total activity found. This will result in a negative number because the contributions from overestimating cladding damage (100% versus 70%) and fuel overheating (major versus intermediate) will exceed the activity contribution from one melted bundle.

At this point, knowledgeable judgement must be employed to establish the best estimate of core damage. Although all damage could be attributable to cladding damage and fuel pellet overheating, the trace of radionuclide indicators of fuel pellet melt indicate the possibility of some fuel melting. Based on knowledge of core temperature variations, it is highly unlikely that 100% cladding damage would exist without significant fuel melting. Also, some of the activity attributed to fuel pellet overheating must be associated with the amount of fuel pellet melting which is indicated. Therefore, the best estimate of fuel damage would be that "intermediate fuel overheating had occurred, with major cladding damage and the possibility of minor fuel pellet melting in one or two fuel bundles out of 150 fuel bundles."

The above example is obviously ideal and makes the major assumptions that:

A. The radionuclide/s monitored are at equal concentrations in all fuel rods. In actually, at no time will all radionuclides be at equal concentrations in all fuel rods. Because the time to reach equilibrium for each radionuclide is different, due to their highly variable production and different decay rates. Some isotopes will approach equilibrium quickly, while others never reach equilibrium. Therefore, it is necessary to factor in reactor power history when determining which radionuclide is optimum for monitoring in a given accident condition. Probably the optimum radionuclides for estimating core damage will vary as a function of time after refueling and based on power history.

B. Equilibrated samples are readily available from all sample locations at the instant of sampling. Considering the large volumes of liquid and vapor spaces that a leakage source migrates to and mixes with, for other than very large leaks, it will take many hours or even days to approach equilibrium conditions at all sample locations.

C. Maximum core degradation occurred prior to initiation of sampling. Unless total cooling is lost, core degradation can be anticipated to progress over a period of hours. Thus, there is not a given instant when sampling can be conducted with positive assurance that maximum degradation has occurred.

Considering that ideal conditions will not exist, then procedure for estimating core damage should be prepared in a manner that the effects of variables such as time in core life and type of accident are accounted for.

Therefore, the procedure for estimating core damage should include the determination of both short and long lived gaseous and non-volatile radionuclides along with ratios for appropriate species. Each separate radionuclide analyzed, along with predicted ratios of selected radionuclides would be used to estimate core damage. This process will result in four separate estimates of core damage, (short and long-lived, gaseous and non-volatile species) which can be weighed, based on power history, to determine the best estimate of core damage.

The post-accident sampling system locations for liquid and gaseous samples varies for each plant. To obtain the most accurate assessment of core damage, it is necessary to sample and analyze radionuclides from each of these locations (reactor coolant, containment atmosphere, containment sumps and suppression pool), then relate the measured concentration to the total curies for each radionuclide at each sample location. These measured radionuclide concentrations need to be decay corrected to the estimated time of core damage. Their relationship to core damage can be obtained by comparing the total quantity and ratios of the radionuclides released with the predetermined radionuclide concentrations and ratios which are available in the core based on power history. Assuming one hour per sample location to recirculate, obtain and analyze a sample from each location it would take ____ hours to perform each of those analyses.

Based on the above rationale, the final procedure for estimating core damage using measured radionuclide concentrations will probably rely only on one or two sample locations during the initial phases of an accident. The optimum

radionuclides for estimating core damage will also, in the short term, be based on recent power history. When equilibrium conditions are established at all sample locations, radionuclide analysis can be performed to obtain a better estimate of core damage. The specific radionuclides to be analyzed under equilibrium conditions may be different than those initially analyzed because of initial abundances and different decay rates.

The specific sample locations to be used during the initial phases of an accident should be selected based on the type of accident in progress (i.e. for a BWR, a small liquid line break in the primary containment would release only small quantities of volatile species to the dry well. Therefore sampling the dry well first would not indicate the true magnitude of core damage). For the same small break accident, if pressure is reduced by venting safety valves to the suppression pool, then the suppression case of a small steam line break, without venting safety valves to the suppression pool, the dry well may be the best sample location.

To account for the variations in prime sample locations, based on type of accident, the procedure should include a list of primary sample locations. This list should include both a prime liquid and gaseous location and state -- the reasoning used to determine that these locations are best. Additionally, the procedure should address other plant locations which can be used to verify that the sample locations selected are best for the specified accident condition.

Finally, the procedure should incorporate plant specific examples which show estimates of core damage based on predicted radionuclide concentrations. Methodology of this step is provided by letter of May 4, 1981, from McGuire Nuclear Station, Docket No. 50-369.

APPENDIX B.0

ANALYTICAL DERIVATIONS

APPENDIX B.1

DERIVATION OF THE TRANSIENT POWER CORRECTION EQUATION FOR SOURCE INVENTORY

DERIVATION OF THE TRANSIENT POWER CORRECTION EQUATION FOR SOURCE INVENTORY

For those plant power histories in which equilibrium conditions do not exist an analytical correction is provided in the procedure. The mathematical model used to calculate the quantity of fission products in the core fuel pellets as a function of time involves a group of linear, first order differential equations. These equations are obtained by applying a mass balance for production and removal. The terms for fission product production include direct fission yield, parent fission product decay, and neutron activation. The terms for fission product loss include decay, neutron activation and escape to the coolant. Each equation in the group is expressed as follows.

$$\frac{dN_i}{dt} = (F)(Y_i)(P) + (f_{i-1}\,\lambda_{i-1})\,N_{i-1} + \sigma_k\,\phi_k\,N_{i-1} - (\lambda_i + \nu_i + \sigma_i\phi)\,N_i$$

where the variables are defined as follows.

N = Fuel pellet fission product inventory, atoms

F = Average fission rate, fission/Mwt-sec

Y = Fission product yield, fraction

P = Core power, Mwt

λ = Decay constant, sec^{-1}

σ = Microscopic cross section, cm^2

ν = Escape rate coefficient, sec^{-1}

f = Branching fraction

t = Time, sec

and where the subscripts are defined as follows.

ℓ = Isotope

$\ell-1$ = Precursor to isotope ℓ for decay

k = Precursor to isotope ℓ for neutron activation

Within the accuracy of this procedure, the terms for fission product production by neutron activation and for fission product loss by neutron activation and escape to the coolant are insignificant. The equation then becomes as follows:

$$\frac{dN_\ell}{dt} = (F)(Y_\ell)(P) + (f_{\ell-1}\ \lambda_{\ell-1})\ N_{\ell-1} - \lambda_\ell\ N_\ell$$

Additionally, it can be assumed that the terms for production are both linear with respect to plant power. Therefore, the equation becomes as follows.

$$\frac{dN_\ell}{dt} = (G)(P) - \lambda_\ell\ N_\ell$$

where (G)(P) is the production term which is linear with respect to power. The solution of these equations are of the following form.

$$N_\ell(t) = \frac{(G)(P)}{\lambda_\ell}\ (1-e^{-\lambda_\ell t})$$

This represents the quantity of fission product isotope, ℓ, produced during time, t, while the reactor is at power, P. At some time after the reactor is shutdown, the fission products which remain are as follows.

$$N_\ell(t) = \frac{(G)(P)}{\lambda_\ell} (1-e^{-\lambda_\ell t_j}) e^{-\lambda_\ell t_j^o}$$

where t_j^o = the time between the end of period j and the time of reactor shutdown.

The equation which expresses the total fission products which remain after multiple time periods of different power levels is as follows.

$$N_\ell(t) = \Sigma_j \frac{(G)(P_j)}{\lambda_\ell} (1-e^{-\lambda_\ell t_j}) e^{-\lambda_\ell t_j^o}$$

The power correction factor then becomes as follows.

$$\frac{N(t) \text{ @ Power } P}{N_\ell(t) \text{ @ 100\% Power}} = \frac{\Sigma_j P_j (1-e^{-\lambda_\ell t_j}) e^{-\lambda_\ell t_j^o}}{100 (1-e^{-\lambda_\ell \Sigma t_j})}$$

Within the accuracy of this procedure and under the condition in which the total period of operation is greater than four radioactive half lives the power correction is as follows.

$$\Sigma_j t_j \geq 4 \times \frac{0.693}{\lambda_\ell}$$

$$\text{Power Correction Factor} = \frac{\Sigma_j P_j (1-e^{-\lambda_i t_j}) e^{-\lambda t_j^o}}{100}$$

APPENDIX B.2

DERIVATION OF THE CORE HEATUP AND OXIDATION RELATIONSHIPS

DERIVATION OF THE CORE HEATUP AND OXIDATION RELATIONSHIPS

This Appendix presents the mathematical derivation of the heatup and oxidation relations which are used to obtain the final results which appear in the damage assessment procedure. Knowledge of the derivation in this Appendix is not essential to proper implementation of the procedure, but it may help to understand the physical phenomena and thereby enable a better judgement on the amount and configuration of core damage for an event which deviates substantially from the usual predictions for core uncovery events. The derivation presented here is similar to that presented in Reference 7-13.

The derivation applies to the uncovered portion of a fuel channel during boiloff at constant pressure and core decay power. A channel may be considered to be a fuel assembly or a single fuel rod and associated flow subchannel. The only difference among channels (prior to clad ballooning and rupture) is the power level, which is directly proportional to the radial nuclear peaking factor. It is assumed that the normalized radial peak distribution is the same for decay power and for normal operation, and is equivalent to typical core wide distributions with all CEA's withdrawn.

The steam generated below the coolant level is assumed to remain in the same channel above the coolant level. The mass flow rate is directly proportional to the radial peak. If the total core steam flow is W_S for N channels, the flow in a channel with radial peak F_R is $W_S F_R/N$. Decay power per unit length

of channel is the core average value times the radial peak, $\bar{q}_d F_R$. An energy balance on a unit length of channel, above the coolant level is written, assuming a homogenized rod of fuel and clad at uniform temperature.

Time rate of change of sensible heat in fuel and steam	=	Heat convected in minus heat convected out	+	Decay power

$$\frac{\partial}{\partial t}\left((M_R C_R)\,T_R + \rho_g A_c C_g T_g\right) = \frac{W_S F_R}{N} C_g T_g - \left[\frac{W_S F_R}{N} C_g T_g + \frac{\partial}{\partial Z}\left(\frac{W_S F_R C_g}{N} T_g\right)\right] + F_R \bar{q}_d \quad (1)$$

where: $(M_R C_R)$ = Total fuel rod specific heat (B/ft. - °F)

T_R = Rod temperature (°F)

t = time after start of core uncovery (hr)

ρ_g = Steam density (lbm/ft^3)

A_c = Channel flow area (ft^2)

C_g = Steam specific heat (B/lbm - °F)

T_g = Steam temperature (°F)

W_s = Core total steam flow rate (lbm/hr)

F_R = Radial peaking factor

N = Number of rods or channels in core

Z = Height above bottom of active length (ft)

$\bar{q}_d$ = Core average decay linear heat rate (B/hr - ft)

Several physically based assumptions are made. In the first term, the heat capacity of the steam is very much less than that of the fuel, so the second term in the parenthesis is neglected by comparison with the first. Physical

properties of fuel and steam are assumed constant. The partial derivative with height becomes $\partial(W_s T_g)/\partial Z$ at any instant of time, and because density changes are neglected this reduces to $W_s\ \partial T_g/\partial Z$. From the previous equations for coolant level, substituting Equation (4-2) into (4-1) yields an expression for the steam flow, W_s, as a function of time after uncovery starts:

$$W_s = W_{in} + \rho AL \frac{(1 - K_2)}{K_1} e^{- t/K_1} \qquad (2)$$

Combining (1) and (2) yields:

$$\frac{\partial T_R}{\partial t} + \frac{K_3}{\rho A} \left[W_{in} + \rho AL \left(\frac{1 - K_2}{K_1}\right) e^{- t/K_1}\right] \frac{\partial T_c}{\partial Z} = \frac{F_R \bar{q}_d}{M_R C_R} \qquad (3)$$

where:

$$K_3 = \frac{\rho A C_c F_R}{N M_R C_R}$$

Assume that the steam attains the rod temperature instantly, which is equivalent to an infinitely large equivalent surface heat transfer coefficient. The rod actually rises above the steam temperature, so Equation (3) results would be low for temperatures below the rapid oxidation temperature range. The heat input from the exothermic oxidation reaction is added later.

The partial differential equation is reduced to an ordinary differential equation by substituting for time after uncovery, t, the time interval, t*, from the time the coolant level drops past a given elevation, Z, to the current time when the level is Z. From Equation (4-2), t* is:

$$t^* = t + K_1 \ln \left(\frac{Z/L - K_2}{1 - K_2}\right) \qquad (4)$$

Using:

$$\frac{\partial T}{\partial Z} = \frac{\partial T}{\partial t^*} \frac{\partial t^*}{\partial Z}$$

and taking account of the assumption that $T_R = T_g$, Equation (3) becomes:

$$\frac{dT}{dt^*} = \frac{F_R \bar{q}_d / M_R C_R}{1 + \frac{W_{in} K_3 K_1}{\rho A L (Z/L - K_2)} + K_3 e^{-t^*/K_1}} \qquad (5)$$

This represents the time variation of the rod temperature at a given elevation as a function of time after the level drops past that elevation. It accounts for convection of some of the decay power by steam and for the increase in sensible heat of the rod by the remainder.

Assume that all heat of reaction by zirconium oxidation goes to raising the rod temperature. Oxidation heating becomes significant above about 1800°F, which is attained only for low coolant levels when the steam flow rate is

relatively low. With low steam flow, cooling by convection and thermal radiation to steam is less effective, and a smaller fraction of the total decay power and reaction heat is transferred to the steam. Therefore, the addition of reaction heat as an adiabatic temperature rise is a fair assumption when the core uncovers by boiloff. Reaction heating is expressed by q_{react}, the equivalent linear heat rate caused by oxidation. The rate of temperature rise is $q_{react}/M_R C_R$, and is added to Equation (5).

The reaction heat rate is expressed as the time rate of conversion of the mass of zirconium into oxide per unit length of clad times the heat of reaction:

$$q_{react} = - \frac{d}{dt} \left(\rho_{Zr} \frac{\pi}{4} (D_o^2 - D_i^2)\right) \Delta H_{react}$$

Assume oxidation on only the outer clad surface, D_o. This yields an appropriate estimate for total core oxidation prior to substantial core damage, but may underestimate the local oxidation on both sides of thinned clad at the location of clad rupture. Let x be the equivalent percentage of the original clad thickness, Δr, which is oxidized. Then the preceding equation becomes:

$$q_{react} = \pi \rho_{Zr} D_o \frac{\Delta r}{100} \Delta H_{react} \frac{dx}{dt} \qquad (6)$$

The equivalent oxide thickness, x, is given as a function of time and temperature - by the equation in Section 4.2.1. The resulting equations for clad temperature and oxidation are:

$$\frac{dT}{dt^*} = \frac{F_R \bar{q}_d / M_R C_R}{1 + \frac{W_{in} K_3 K_1}{\rho A L (Z/L - K_2)} + K_3 e^{- t^*/K_1}} + \frac{\pi \rho D_o (\Delta r/100) \Delta H_{react}}{M_R C_R} \frac{dx}{dt}$$

$$x^2 = \left(\frac{100}{\Delta r}\right)^2 \frac{A_{Zr}\, t}{\rho_{Zr}^2} e^{- B/R(T + 460)}$$

These equations are normalized by substituting:

$$t' = t^*/K_1$$

$$T' = (T - T(o)) \left(\frac{K_4 M_R C_R}{K_1 \bar{q}_d F_R}\right) \qquad (7)$$

Yielding:

$$\frac{dT'}{dt'} = \frac{1}{1 + \frac{K_3}{K_4} e^{- t'}} + K_4 K_5 \frac{dx}{dt'} \qquad (8)$$

$$x^2 = K_6\, t'\, e^{- 1/(T' K_7/K_4 + K_8)} \qquad (9)$$

These equations give the clad temperature and local oxidation vs. time for core uncovery by boiloff with some inlet flow. When the inlet flow is zero,

K_4 is 1.0. The equations are solved numerically with zero inlet flow and for values of the constants, K, evaluated for the 3400 Mwt class.

In order to compare the results for various reactors, the constants are defined in Table 1 and numerical valves are given in Table 2 for a selected decay heat fraction of 1% and for two pressures. The table shows that the variation of the values among the C-E designed reactors is relatively small and is much less than the difference in values at 300 psia and 2500 psia. In other words, the differences in the fuel heatup behavior among the C-E designed plants for a boiloff event are much less than the differences to be expected just because of the effect of possible pressure differences during two alternate accident scenarios. Therefore, the 3400 Mwt class is evaluated over the range of 300 to 2500 psia and from 1% to 3% decay heat, and the relative distribution of oxidation is used in the procedure for damage assessment on all reactors. The range of error introduced by this decision is considered sufficiently small to allow identification of the extent of core damages within the ten categories defined in Table 4-1.

Table 1

Definitions of Constants in Heatup Equations

Constant	Definition
K_1	$\frac{\rho A L H_{fg}}{P_o \Delta H}$
K_2	$\frac{W_{in}(H_{fg} + H_f - H_{in})}{P_o \Delta H}$
K_3	$\frac{\rho A c_a F_R}{\pi M_R C_R}$
K_4	$1 + \frac{F_R K_3 K_2 H_{fg}}{(Z/L - K_2)(H_{fg} - H_f - H_{in})}$
K_5	$\frac{\pi D_o \rho_{Zr} \Delta H_{react}}{K_1 F_R \bar{q}_d} (\frac{\Delta r}{100})$
K_6	$(\frac{100}{\Delta r})^2 K_1 \frac{A_{Zr}}{\rho_{Zr}^2}$
K_7	$\frac{F_R K_1 \bar{q}_d}{M_R C_R \delta/R}$
K_8	$\frac{T(o) + 460}{\delta/R}$

Table 2

Constants In Heatup Equations-Comparison of Values for C-E Designed Reactors

Variable	Ft. Calhoun	Calvert Cliffs	SONGS-2	S-80
L (ft)	10.67	11.4	12.5	12.5
A (ft^2)	60.1	107.3	109.7	118.5
$\bar{q}_d$ @ 1% (B/hr-ft)	205	212	181	182
P_oDH @ 1% (B/hr)	5.12×10^7	9.21×10^7	1.16×10^8	1.30×10^8
N -	23408	38192	51212	56876
D_o (ft)	0.442/12 = 0.0368	0.440/12 = 0.0367	0.382/12 = 0.0318	0.0318
Δr (ft)	0.032/12 = 0.00267	0.026/12 = 0.00217	0.025/12 = 0.00208	0.00208
$M_R C_R$ (B/hr-ft)	0.0474	0.0479	0.0356	0.0356
ρ_{Zr} (lbm/ft^3)	407	407	407	407
ΔH_{react} (B/lbm)	2770	2770	2770	2770
A_{Zr} (lbm^2/ft^4-hr)	5020	5020	5020	5020
B/R (°R)	30429	30429	30429	30429

Table 2 (Cont'd)

Constants In Heatup Equations-Comparison of Values for C-E Designed Reactors

Variable	Ft. Calhoun		Calvert Cliffs		SONGS-2		S-80	
Pressure (psia)	300	2500	300	2500	300	2500	300	2500
ρ (lbm/ft^3)	52.9	35.0	52.9	35.0	52.9	35.0	52.9	35.0
H_{fg} (B/lbm)	808.9	361.6	808.9	361.6	808.9	361.6	808.9	361.6
C_g (B/lbm-°F)	0.565	0.612	0.565	0.612	0.565	0.612	0.565	0.612
K_1	0.536	0.159	0.561	0.168	0.506	0.150	0.488	0.144
K_2	V A R I A B L E W I T H I N L E T F L O W R A T E							
K_3/F_R	1.62	1.16	1.75	1.26	1.80	1.29	1.75	1.25
K_4	V A R I A B L E W I T H I N L E T F L O W R A T E							
K_5 F_R	0.0294	0.0993	0.0218	0.0737	0.0240	0.0811	0.0247	0.0836
K_6	2.28×10^7	6.75×10^6	3.67×10^7	1.08×10^7	3.53×10^7	1.05×10^7	3.41×10^7	1.01×10^7
K_7/F_R	0.0762	0.0226	0.0825	0.0244	0.0845	0.0250	0.0819	0.0242
K_8	0.0288	0.0371	0.0288	0.0371	0.0288	0.0371	0.0288	0.0371

APPENDIX C.0

PROCEDURE GUIDELINE FOR ASSESSMENT OF CORE DAMAGE USING RADIOLOGICAL ANALYSIS OF SAMPLES

TABLE OF CONTENTS

LIST OF ENCLOSURES

LIST OF ENCLOSURES (Cont'd.)

1.0 PURPOSE

This procedure is to be followed under post accident plant conditions to determine the type and degree of reactor core damage which may have occurred by using fission product isotopes measured in samples obtained from the Post Accident Sampling System (PASS). There are three factors considered in this procedure which are related to the specific activity of the samples. These are the identity of those isotopes which are released from the core, the respective ratios of the specific activity of those isotopes, and the percent of the source inventory at the time of the accident which is observed to be present in the samples. The resulting observation of core damage is described by one or more of the ten categories of core damage in Enclosure 1.

2.0 REFERENCES

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, C-E Owners Group Task 467, May 1983.

2.2 Post Accident Sampling System Operating Procedures. (Plant specific document).

3.0 DEFINITIONS

3.1 Fuel Damage: For the purpose of this procedure fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the reactor coolant starting with a penetration in the zircaloy cladding. The type of fuel damage as determined by this procedure is reported in terms of four major categories which are: no damage, cladding failure, fuel overheat, and fuel melt. Each of these categories are characterized by the identity of the fission products released, the mechanism by which they are released, and the source inventory within the fuel rod from which they are released. The degree of fuel damage is measured by the percent of the fission product source inventory which has been released into fluid media and therefore available for immediate release to the environment. The degree of fuel damage as determined by this procedure is reported in terms of three levels -- which are: initial, intermediate, and major. This results in a total of ten possible categories as characterized in Enclosure 1.

3.2 Source Inventory: The source inventory is the total quantity of fission products expressed in curies of each isotope present in either source; the fuel pellets or the fuel rod gas gap.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the 10 categories of core damage described in Enclosure 1. The procedure is based on radiological data. Other plant indications may be available which can improve upon estimation of core damage. These include incore temperature indicators, the total quantity of hydrogen released from zirconium degradation and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 This procedure relies upon samples taken from multiple locations inside the containment building to determine the total quantity of fission products available for release to the environment. The amount of fission products present at each sample location may be changing rapidly due to transient plant conditions. Therefore, it is required that the samples should be obtained within a minimum time period and if possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

4.3 A number of factors influence the reliability of the chemistry samples upon which this procedure is based. Reliability is influenced by the ability to obtain representative samples due to incomplete mixing of the fluids, equipment limitations, and lack of operator familiarity with rarely used analytical procedures. The accuracy achieved in the radiological analyses are also influenced by a number of factors. The equipment employed in the analysis may be subjected to high levels of radiation exposure over extended periods of time. Chemists are required to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources and they may not result from a uniform distribution of the sample fluid. Cooling or reactions may take place in the long sample lines. Therefore, the results obtained may not be representative of plant conditions. To minimize these effects multiple samples should be obtained over an extended time period from each location.

5.0 INITIAL PLANT CONDITIONS AND SYMPTOMS

This procedure is to be employed for analysis of radiochemistry sample data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. This list is not a complete representation of all events which may cause core damage. One or more of these symptoms may exist at or before the time the sample is obtained. Under these conditions, sampling should be performed using the Post Accident Sampling System.

5.1 High alarm on the containment radiation monitor.

5.2 High alarm on the CVCS letdown radiation monitor.

5.3 High alarm on the main condenser air ejector exhaust radiation monitor.

5.4 Pressurizer level low.

5.5 Safety Injection System may have automatically actuated.

5.6 Possible high quench tank level, temperature, or pressure.

5.8 Possible noise indicative of a high energy line break.

5.9 Decrease in volume control tank level.

5.10 Standby charging pumps energized.

5.11 Unbalanced charging and letdown flow.

5.12 Reactor Coolant System subcooling low or zero.

6.0 PREREQUISITES

An operational Post Accident Sampling System with the capability to obtain and analyze the identity and concentration of fission product isotopes in fluid samples which have the potential to be highly radioactive. The system should meet the requirements of NUREG-0737 Item II.B.3, Reference 3.

7.0 PROCEDURE

7.1 Record the following plant indications. Because of transient conditions the values should be recorded as close as possible to the time at which the radiological samples are obtained from the Post Accident Sampling System.

7.1.1 Reactor Coolant System:

Pressure	_____	PSIG
Temperature	_____	°F
Reactor Vessel Level	_____	%
Pressurizer Level	_____	%

7.1.2 Containment Building:

Atmosphere Pressure	_____	PSIG
Atmosphere Temperature	_____	°F
Sump Level	_____	%

7.1.3 Prior 30 days Power History

Power, Percent	Duration, Days
_____	_____
_____	_____
_____	_____
_____	_____

7.1.4 Time of Reactor Shutdown Date_____ Time_____

7.2 Select the most appropriate sample locations required for core damage assessment using the guidelines provided in Enclosure 2.

7.3 Obtain and analyze the selected samples for fission product specific activity using the procedures for Post Accident Sample System operation described in Reference 2. Record the required sample data for each selected sample. Enclosure 3 is provided as a worksheet. All of the isotopes listed in the enclosure may not be observed in the sample.

7.4 Correct the measured sample specific activity to standard temperature and pressure.

NOTE: This step is required only if it is not included in the procedures for Post Accident Sample System Operation, Reference 2.

7.4.1 Reactor coolant liquid samples are corrected for system temperature and pressure using the factor for water density provided in Enclosure 4. The correction factor obtained from the enclosure is divided into the measured value to obtain the density corrected value.

7.4.2 Containment building sump samples do not require correction for temperature and pressure within the accuracy of this procedure.

7.4.3 Containment building atmosphere gas samples are corrected using the following equation.

$$\text{Specific Activity(STP)} = \text{Specific Activity} \times \left(\frac{P_2}{P_1 + P_2}\right) \times \left(\frac{T_1 + 460}{T_2 + 460}\right)$$

where:

T_1, P_1 = Measured Sample temperature and Pressure recorded in step 7.3.

T_2, P_2 = Standard temperature, 32°F and Standard Pressure 14.7 psia.

7.4.4 Enclosure 5 is provided as a worksheet.

7.5 Correct the sample specific activity at STP for decay back to the time of reactor shutdown which is recorded in step 7.1.4 using the following equation. Enclosure 6 is provided as a worksheet.

$$A_o = \frac{A}{e^{-\lambda t}}$$

where:

A_o = the specific activity of the sample corrected back to the time of reactor shutdown, $\mu ci/cc$.

A = the measured specific activity, $\mu ci/cc$.

λ = the radioactive decay constant, 1/sec.

t = the time period from reactor shutdown to sample analysis, sec.

7.6 Identification of the Fission Product Release Source.

7.6.1 Calculate the following ratios for each noble gas and iodine isotope only using the specific activities obtained in step 7.5. Enclosure 7 is provided as a worksheet.

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Specific Activity}}{\text{Xe 133 Specific Activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Isotope Specific Activity}}{\text{I-131 Specific Activity}}$$

7.6.2 Determine the source of release by comparing the results obtained to the predicted ratios provided in Enclosure 7. An accurate comparison is not anticipated. Within the accuracy of this procedure it is appropriate to select as the source that ratio which is closest to the value obtained in step 7.6.1.

7.7 Calculate the total quantity of fission products available for release to the environment. Enclosure 8 is provided as a worksheet.

7.7.1.1 If the water level in the reactor vessel recorded in step 7.1.1 indicates that the vessel is full, the quantity of fission products found in the reactor coolant is calculated by the following equation.

Total Activity (Ci) = A_o(μci/cc) x RCS Volume

where:

A_o = the specific activity of the reactor coolant sample corrected to time of reactor shutdown obtained in step 7.5, μci/cc.

RCS Volume = the full reactor coolant system water volume corrected to standard temperature and pressure using Enclosure 4.

7.7.1.2 If the water levels in the reactor vessel and pressurizer recorded in step 7.1.1 indicates that a steam void is present in the reactor vessel, then the quantity of fission products found in the reactor coolant is again calculated by step 7.7.1.1. However, it must be recognized that the value obtained will overestimate the actual quantity released. Therefore, this sample should be repeated at such time when the plant operators have removed the void from the reactor vessel.

7.7.1.3 If the water level in the reactor vessel recorded in step 7.1.1 is below the low end capability of the indicator, it is not possible to determine the quantity of fission products from this sample because the volume of water in the reactor coolant system is unknown. Under this condition, assessment of core damage is obtained using the containment sump sample.

7.7.2 The quantity of fission products found in the containment building sump is determined as follows.

7.7.2.1 The water volume in the containment building sump is determined from the sump level recorded in step 7.1.2 and the curve provided in Enclosure 9.

7.7.2.2 The quantity of fission products in the sump is calculated by the following equation.

Total Activity, Ci = A_o(μci/cc) x Sump Volume

where:

A_o = the specific activity of the containment sump sample corrected to the time of reactor shutdown obtained in step 7.5., μci/cc.

7.7.3 The quantity of fission products found in the containment building atmosphere is determined as follows.

7.7.3.1 The volume of gas in the containment building, at the time of the accident, is corrected to standard temperature and pressure using the following equation.

$$\text{Gas Volume (STP)} = \text{Gas Volume} \times \frac{(P_2 + P_1)}{P_2} \times \frac{(T_2 + 460)}{(T_1 + 460)}$$

where:

T_1, P_1 = Containment Atmosphere temperature and pressure recorded in step 7.1.2.

T_2, P_2 = Standard temperature, 32°F and Standard Pressure 14.7 psia.

7.7.4 The total quantity of fission products available for release to the environment is equal to the sum of the values obtained from each sample location.

7.8 Plant Power Correction

The quantitative release of the fission products is expressed as the percent of the source inventory at the time of the accident. The equilibrium source inventories are to be corrected for plant power history.

7.8.1 To correct the source inventory for the case in which plant power level has remained constant for a period greater than four radioactive half lives the following procedure is employed. Enclosure 10 is provided as a worksheet.

7.8.1.1 The fission products are divided into two groups based upon the radioactive half lives. Group 1 isotope are to be employed in the case where core power had not changed greater than ±10 percent within the last 30 days prior to the reactor shutdown. Group 2 isotopes are to be employed in the case where core power had not changed greater than ±10 percent within the last 4 days prior to the reactor shutdown.

7.8.1.2 The following equation may be applied to the fission product Group which meets the criteria stated in 7.8.1.1 only.

Group 1 $$\text{Power Correction Factor} = \frac{\text{Steady State Power Level for Prior 30 Days}}{100}$$

Group 2 $$\text{Power Correction Factor} = \frac{\text{Steady State Power Level for Prior 4 Days}}{100}$$

7.8.2 To correct the source inventory for the case in which plant power level has not remained constant prior to reactor shutdown, the following equation is employed. The entire 30 days power history should be employed. Enclosure 11 is provided as a worksheet.

$$\text{Power Correction Factor} = \frac{\Sigma_j P_j (1 - e^{-\lambda t_j}) e^{-\lambda t_j^o}}{100}$$

where:

P_j = steady reactor power in period j

t_j = duration of period j

t_j^o = time from end of period j to reactor shutdown

7.9 Comparison of Measured Data with Source Inventory

The total quantity of fission products available for release to the environment obtained in step 7.7.4 is compared to the source inventory corrected for plant power history obtained in step 7.8.2. This comparison is made by dividing the two values for each isotope and calculating the percent of the corrected source inventory that is now in the sampled fluid and therefore available for release to the environment. Enclosure 12 is provided as a worksheet.

7.9 CORE DAMAGE ASSESSMENT

The conclusion on core damage is made using the three parameters developed above. These are:

1. Identification of the fission product isotopes which most characterize a given sample, step 7.3.

2. Identification of the source of the release, step 7.6.

3. Quantity of the fission produce available for release to the environment expressed as a percent of source inventory, step 7.9.

Knowledgeable judgement is used to compare the above three parameters to the definitions of the 10 NRC categories of fuel damage found in Enclosure 1. Core damage is not anticipated to take place uniformly. Therefore when evaluating the three parameters listed above the procedure is anticipated to yield a combination of one or more of the 10 categories defined in Enclosure 1. These categories will exist simultaneously.

The type of core damage is described in terms of the 10 NRC categories defined in Enclosure 1. The degree of core damage is described as the percent of the fission products in the source inventory at the time of the accident which is now in the sampled fluid and therefore available for release to the environment.

ENCLOSURE 1

Radiological Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release	Source of Release	Characteristic Isotope	Release of Characteristic Isotope Expressed as a Percent of Source Inventory
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I 131, Cs 137 Rb 88	Less than 1
2.	Initial Cladding Failure		Gas Gap		Less than 10
3.	Intermediate Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Xe 131m, Xe 133 I 131, I 133	10 to 50
4.	Major Cladding Failure		Gas Gap		Greater than 50
5.	Initial Fuel Pellet Overheating		Fuel Pellet	Cs 134, Rb 88, Te 129, Te 132	Less than 10
6.	Intermediate Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet		10 to 50
7.	Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet		Greater than 50
8.	Fuel Pellet Melt		Fuel Pellet		Less than 10
9.	Intermediate Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet	Ba 140, La 140 La 142, Pr 144	10 to 50
10.	Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

where:

P_j = steady reactor power in period j

t_j = duration of period j

t_j^o = time from end of period j to reactor shutdown

7.9 Comparison of Measured Data with Source Inventory

The total quantity of fission products available for release to the environment obtained in step 7.7.4 is compared to the source inventory corrected for plant power history obtained in step 7.8.2. This comparison is made by dividing the two values for each isotope and calculating the percent of the corrected source inventory that is now in the sampled fluid and therefore available for release to the environment. Enclosure 12 is provided as a worksheet.

7.9 CORE DAMAGE ASSESSMENT

The conclusion on core damage is made using the three parameters developed above. These are:

1. Identification of the fission product isotopes which most characterize a given sample, step 7.3.

2. Identification of the source of the release, step 7.6.

3. Quantity of the fission produce available for release to the environment expressed as a percent of source inventory, step 7.9.

Knowledgeable judgement is used to compare the above three parameters to the definitions of the 10 NRC categories of fuel damage found in Enclosure 1. Core damage is not anticipated to take place uniformly. Therefore when evaluating the three parameters listed above the procedure is anticipated to yield a combination of one or more of the 10 categories defined in Enclosure 1. These categories will exist simultaneously.

The type of core damage is described in terms of the 10 NRC categories defined in Enclosure 1. The degree of core damage is described as the percent of the fission products in the source inventory at the time of the accident which is now in the sampled fluid and therefore available for release to the environment.

ENCLOSURE 1

Radiological Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release	Source of Release	Characteristic Isotope	Release of Characteristic Isotope Expressed as a Percent of Source Inventory
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I 131, Cs 137 Rb 88	Less than 1
2.	Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap		Less than 10
3.	Intermediate Cladding Failure		Gas Gap	Xe 131m, Xe 133 I 131, I 133	10 to 50
4.	Major Cladding Failure		Gas Gap		Greater than 50
5.	Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Cs 134, Rb 88, Te 129, Te 132	Less than 10
6.	Intermediate Fuel Pellet Overheating		Fuel Pellet		10 to 50
7.	Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet		Greater than 50
8.	Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet		Less than 10
9.	Intermediate Fuel Pellet Melt		Fuel Pellet	Ba 140, La 140 La 142, Pr 144	10 to 50
10.	Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

ENCLOSURE 2
SAMPLE LOCATIONS APPROPRIATE FOR CORE DAMAGE ASSESSMENT

ACCIDENT SCENARIO KNOWN	RCS HOT LEG	CONTAINMENT SUMP	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM
Small Break LOCA, Reactor Power >1%	Yes	---	Yes	Yes
Small Break LOCA, Reactor Power <1%	Yes	---	---	Yes
Small Steam Line Break	Yes	---	---	---
Large Break LOCA, Reactor Power >1%	Yes	Yes	Yes.	Yes
Large Break LOCA, Reactor Power <1%	---	Yes	Yes	Yes
Large Steam Line Break	Yes	---	Yes	---
Steam Generator Tube Rupture	Yes	---	Yes	---

ENCLOSURE 3

RECORD OF SAMPLE SPECIFIC ACTIVITY

Sample Number:

Location:

Time of Analysis:

Temperature, °F:

Pressure, PSIG:

Sample Activity, $\mu Ci/cc$:

Kr 87

Xe 131m

Xe 133

I 131

I 132

I 133

I 135

Cs 134

Rb 88

Te 129

Te 132

Sr 89

Ba 140

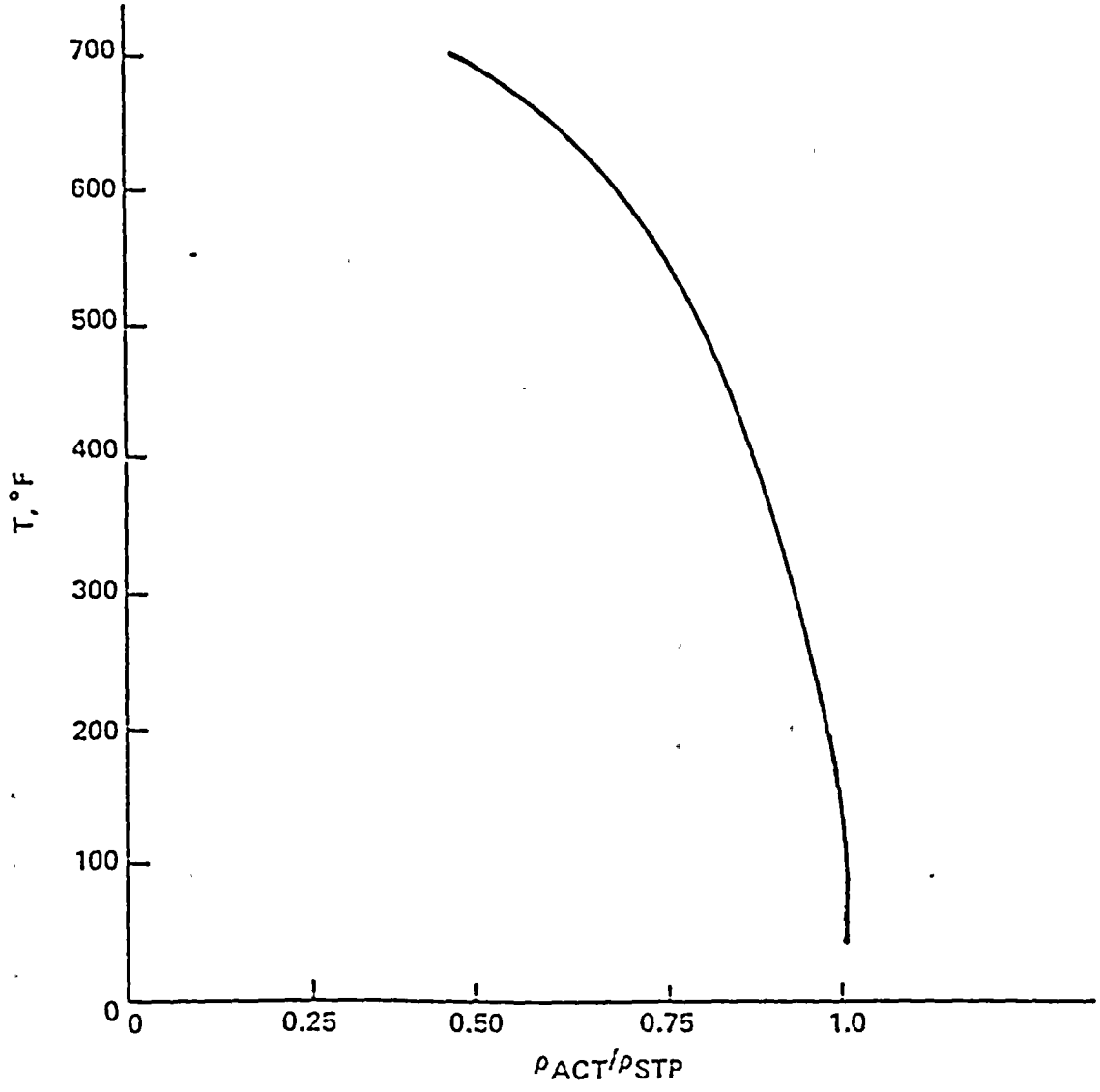
La 140

La 142

Pr 144

ENCLOSURE 4

RATIO OF H_2O DENSITY TO H_2O DENSITY AT STP vs TEMPERATURE



ENCLOSURE 5

RECORD OF SAMPLE TEMPERATURE CORRECTION

Sample Number:

Location:

Time of Analysis:

Temperature, °F:

Pressure, PSIG:

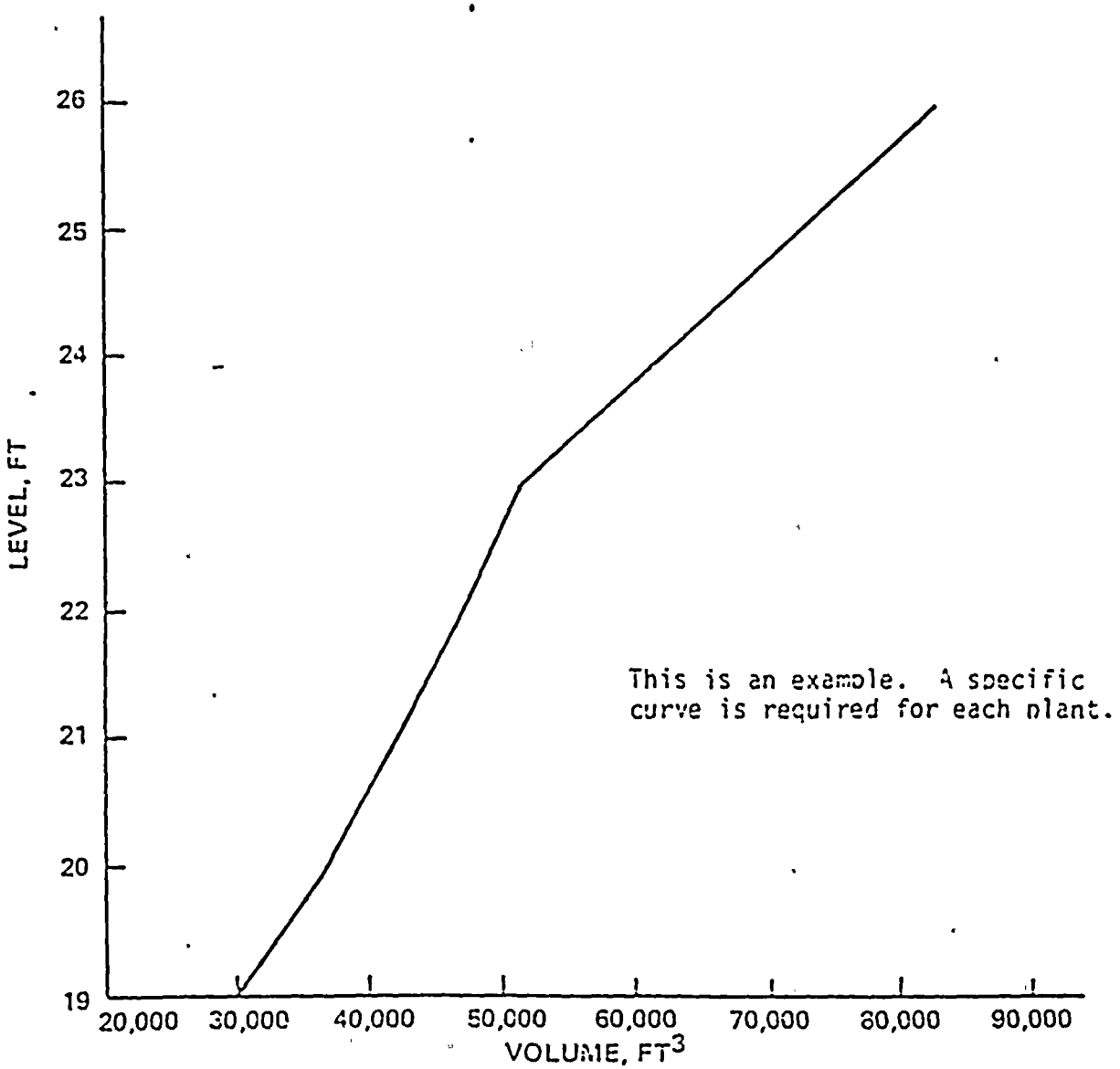
Isotope	Measured Specific Activity (Enclosure 3), $\mu ci/cc$	Correction Factor	Specific Activity @ STP, $\mu ci/cc$
Kr 87			
Xe 131m			
Xe 133			
I 131			
I 132			
I 133			
I 135			
Cs 134			
Rb 88			
Te 129			
Te 132			
Sr 89			
Ba 140			
La 140			
La 142			
Pr 144			

ENCLOSURE 8

RECORD OF RELEASE QUANTITY

Isotope	Reactor Coolant Sample Number, Ci	Containment Sump Sample Number, Ci	Containment Atmosphere Sample Number, Ci	Total Quantity Ci
Kr 87				
Xe 131m				
Xe 133				
I 131				
I 132				
I 133				
I 135				
Cs 134				
Rb 88				
Te 129				
Te 132				
Sr 89				
Ba 140				
La 140				
La 142				
Pr 144				

ENCLOSURE 9
CONTAINMENT BUILDING WATER
LEVEL vs VOLUME



ENCLOSURE 10

RECORD OF STEADY STATE POWER CORRECTION

Sample Number:
Location:
Steady State 30 Days Power Level:
Steady State 4 Day Power Level:

Isotope	Fuel History Grouping	Power Correction Factor	x	Equilibrium Source Inventory*	=	Corrected Source Inventory
Gas Gap Inventory						
Kr 87	2			6.3(0)		
Xe 131m	1			4.3(4)		
Xe 133	1			1.3(7)		
I 131	1			6.7(6)		
I 132	2			7.0(3)		
I 133	2			6.7(6)		
I 135	2			1.1(6)		
Fuel Pellet Inventory						
Kr 87	2			3.1(7)		
Xe 131m	1			4.6(5)		
Xe 133	1			1.5(8)		
I 131	1			7.3(7)		
I 132	2			1.0(8)		
I 133	2			1.5(8)		
I 135	2			1.3(8)		
Cs 134	1			1.9(7)		
Rb 88	2			4.5(7)		
Te 129	2			2.4(7)		
Te 132	1			1.0(8)		
Sr 89	1			6.1(7)		
Ba 140	1			1.3(8)		
La 140	1			1.3(8)		
La 142	2			1.6(8)		
Pr 144	2			9.1(7)		

* Plant specific values should be substituted here from Tables 3-4 and 3-5 of Section 3.3. Example here is for 2560 Mwt class.

ENCLOSURE 11

RECORD OF TRANSIENT POWER CORRECTION

Sample Number:
Location:
Prior 30 Day Power History:

Power %	Duration, Days

Isotope	Power Correction Factor	x	Equilibrium Source Inventory*	=	Corrected Source Inventory
Gas Gap Inventory					
Kr 87			6.3(0)		
Xe 131			4.3(4)		
Xe 133			1.3(7)		
I 131			6.7(6)		
I 132			7.0(3)		
I 133			6.7(6)		
I 135			1.1(6)		
Fuel Pellet Inventory					
Kr 87			3.1(7)		
Xe 131m			4.6(5)		
Xe 133			1.5(8)		
I 131			7.3(7)		
I 132			1.0(8)		
I 133			1.5(8)		
I 135			1.3(8)		
Cs 134			1.9(7)		
Rb 88			4.5(7)		
Te 129			2.4(7)		
Te 132			1.0(8)		
Sr 89			6.1(7)		
Ba 140			1.3(8)		
La 140			1.3(8)		
La 142			1.6(8)		
Pr 144			9.1(7)		

* See footnote on Enclosure 10.

ENCLOSURE 12

RECORD OF PERCENT RELEASE

Isotope	Total Quantity Available For Release (Enclosure 8), Ci	÷	Power Corrected Source Inventory, x 100 = Ci (Enclosure 10 or 11)	Percent
Gas Gap Inventory				
Kr 87				
Xe 131				
Xe 133				
I 131				
I 132				
I 133				
I 135				
Fuel Pellet Inventory				
Kr 87				
Xe 131m				
Xe 133				
I 131				
I 132				
I 133				
I 135				
Cs 134				
Rb 88				
Te 129				
Te 132				
Sr 89				
Ba 140				
La 140				
La 142				
Pr 144				

APPENDIX C.1

EXAMPLE USE OF THE PROCEDURE

The following is an example of the use of this procedure for assessment of core damage. The specific case sited is for an NSSS of the 2560 Mwt class. The data recorded on plant condition at the time of the sample analysis is as follows:

Reactor Coolant System:	Pressure	1600 PSIG
	Temperature	300 °F
	Reactor Vessel Level	100 %
	Pressurizer Level	80 %
Containment Building:	Pressure	0.5 PSIG
	Temperature	220 °F
	Sump Level	21 feet

Prior 30 Day Power History	Power, Percent	Duration, Days
	75	22
	50	17
	100	2

Time of reactor shutdown 0100 on 12/25/82

ENCLOSURE 3

RECORD OF SAMPLE SPECIFIC ACTIVITY

Sample Number: 1

Location: RCS Hot Leg

Time of Analysis: 12/25/82 0400

Temperature, °F: 300

Pressure, PSIG: 1600

Sample Activity, $\mu ci/cc$:

Kr 87	
Xe 131m	1(÷2)
Xe 133	
I 131	1(÷4)
I 132	
I 133	1(÷2)
I 135	
Cs 134	
Rb 88	
Te 129	1(÷3)
Te 132	
Sr 89	
Ba 140	
La 140	
La 142	1(÷1)
Pr 144	

ENCLOSURE 3

RECORD OF SAMPLE SPECIFIC ACTIVITY

Sample Number: 2

Location: Containment Sump

Time of Analysis: 0500 12/25/82

Temperature, °F: 150

Pressure, PSIG: 0.5

Sample Activity, $\mu ci/cc$:

Kr 87	
Xe 131m	1(-5)
Xe 133	
I 131	1(+2)
I 132	
I 133	1(0)
I 135	
Cs 134	
Rb 88	
Te 129	1(+1)
Te 132	
Sr 89	
Ba 140	
La 140	
La 142	1(-1)
Pr 144	

ENCLOSURE 3

RECORD OF SAMPLE SPECIFIC ACTIVITY

Sample Number: 3

Location: Containment Atmosphere

Time of Analysis: 0600 12/25/82

Temperature, °F: 220

Pressure, PSIG: 0.5

Sample Activity, $\mu ci/cc$:

Kr 87	
Xe 131m	
Xe 133	1(-1)
I 131	1(-1)
I 132	
I 133	1(-3)
I 135	
Cs 134	
Rb 88	
Te 129	
Te 132	
Sr 89	
Ba 140	
La 140	
La 142	
Pr 144	

ENCLOSURE 5

RECORD OF SAMPLE TEMPERATURE CORRECTION

Sample Number: 1

Location: RCS Hot Leg

Time of Analysis: 12/25/82 0400

Temperature, °F: 300

Pressure, PSIG: 1600

Isotope	Measured Specific Activity (Enclosure 3), $\mu ci/cc$	Correction Factor	Specific Activity @ STP, $\mu ci/cc$
Kr 87			
Xe 131m			
Xe 133	1(÷2)	1/0.9	1.1(÷2)
I 131	1(÷4)	1/0.9	1.1(÷4)
I 132			
I 133	1(÷2)	1/0.9	1.1(÷2)
I 135			
Cs 134			
Rb 88			
Te 129	1(÷3)	1/0.9	1.1(÷3)
Te 132			
Sr 89			
Ba 140			
La 140			
La 142	1(÷1)	1/0.9	1.1(÷1)
Pr 144			

ENCLOSURE 5

RECORD OF SAMPLE TEMPERATURE CORRECTION

Sample Number: 2

Location: Containment Sump

Time of Analysis: 0500 12/25/82

Temperature, °F: 150

Pressure, PSIG: 0.5

Isotope	Measured Specific Activity (Enclosure 3), $\mu ci/cc$	Correction Factor	Specific Activity @ STP, $\mu ci/cc$
Kr 87			
Xe 131m			
Xe 133	1(-5)	N/A	1(-5)
I 131	1(-2)	N/A	1(-2)
I 132			
I 133	1(0)	N/A	1(0)
I 135			
Cs 134			
Rb 88			
Te 129	1(+1)	N/A	1(+1)
Te 132			
Sr 89			
Ba 140			
La 140			
La 142	1(-1)	N/A	1(-1)
Pr 144			

ENCLOSURE 5

RECORD OF SAMPLE TEMPERATURE CORRECTION

Sample Number: 3

Location: Containment Atmosphere

Time of Analysis: 0600 12/25/82

Temperature, °F: 220

Pressure, PSIG: 0.5

Isotope	Measured Specific Activity (Enclosure 3), $\mu ci/cc$	Correction Factor	Specific Activity @ STP, $\mu ci/cc$
Kr 87			
Xe 131m			
Xe 133	1(-1)	1.3	1.3(-1)
I 131	1(-1)	1.3	1.3(-1)
I 132			
I 133	1(-3)	1.3	1.3(-3)
I 135			
Cs 134			
Rb 88			
Te 129	1(+1)	N/A	1(+1)
Te 132			
Sr 89			
Ba 140			
La 140			
La 142	1(-1)	N/A	1(-1)
Pr 144			

The containment atmosphere specific activities must be corrected for temperature and pressure.

The correction factor calculation is performed as follows:

$$\left(\frac{14.7}{14.7 + P_1}\right)\left(\frac{T_1}{T_2}\right) = \left(\frac{14.7}{14.7 + 0.5}\right)\left(\frac{680°R}{492°R}\right) = 1.3$$

This value is recorded on Enclosure 5.

ENCLOSURE 6

RECORD OF DECAY CORRECTION

Time of Reactor Shutdown, Step 7.1.4: 12/25/82 0100

Sample Number: 1

Location: RCS Hot Leg

Time of Analysis: 12/25/82 0400

Isotope	Decay Constant, 1/sec	Specific Activity @ STP (Enclosure 5), $\mu ci/cc$	Decay Corrected Specific Activity, $\mu ci/cc$
Kr 87	1.5 (-4)		
Xe 131m	6.7 (-7)		
Xe 133	1.5 (-6)	1.1(÷2)	1.1(÷2)
I 131	9.9 (-7)	1.1(÷4)	1.1(÷4)
I 132	8.4 (-5)		
I 133	9.3 (-6)	1.1(÷2)	1.2(÷2)
I 135	2.9 (-5)		
Cs 134	1.1 (-8)		
Rb 88	6.5 (-4)		
Te 129	1.7 (-4)	1.1(÷3)	6.9(÷3)
Te 132	2.5 (-6)		
Sr 89	1.6 (-7)		
Ba 140	6.3 (-7)		
La 140	4.8 (-6)		
La 142	1.2 (-4)	1.1(÷1)	4.0(÷1)
Pr 144	6.7 (-4)		

ENCLOSURE 6

RECORD OF DECAY CORRECTION

Time of Reactor Shutdown, Step 7.1.4: 12/25/82 0100

Sample Number: 2

Location: Containment Sump

Time of Analysis: 0500 12/25/82

Isotope	Decay Constant, 1/sec	Specific Activity @ STP (Enclosure 5), $\mu ci/cc$	Decay Corrected Specific Activity, $\mu ci/cc$
Kr 87	1.5 (-4)		
Xe 131m	6.7 (-7)		
Xe 133	1.5 (-6)	1(-5)	1(-5)
I 131	9.9 (-7)	1(+2)	1(+2)
I 132	8.4 (-5)		
I 133	9.3 (-6)	1(0)	1(0)
I 135	2.9 (-5)		
Cs 134	1.1 (-8)		
Rb 88	6.5 (-4)		
Te 129	1.7 (-4)	1(+1)	I.2(+2)
Te 132	2.5 (-6)		
Sr 89	1.6 (-7)		
Ba 140	6.3 (-7)		
La 140	4.8 (-6)		
La 142	1.2 (-4)	1(-1)	5.6(-1)
Pr 144	6.7 (-4)		

ENCLOSURE 6

RECORD OF DECAY CORRECTION

Time of Reactor Shutdown, Step 7.1.4: 12/25/82 0100

Sample Number: 3

Location: Containment Atmosphere

Time of Analysis: 0600 12/25/82

Isotope	Decay Constant, 1/sec	Specific Activity @ STP (Enclosure 5), $\mu ci/cc$	Decay Corrected Specific Activity, $\mu ci/cc$
Kr 87	1.5 (-4)		
Xe 131m	6.7 (-7)		
Xe 133	1.5 (-6)	1.3(-1)	1.3(-1)
I 131	9.9 (-7)	1.3(-1)	1.3(-1)
I 132	8.4 (-5)		
I 133	9.3 (-6)	1.3(-3)	1.5(-3)
I 135	2.9 (-5)		
Cs 134	1.1 (-8)		
Rb 88	6.5 (-4)		
Te 129	1.7 (-4)		
Te 132	2.5 (-6)		
Sr 89	1.6 (-7)		
Ba 140	6.3 (-7)		
La 140	4.8 (-6)		
La 142	1.2 (-4)		
Pr 144	6.7 (-4)		

Decay Corrections

RCS

Xe_{133}: $1.1(+2) \div e^{-[1.5(-6)](3)3600} = 1.1(+2)$

I_{131}: $1.1(+4) \div e^{-[9.9(-7)](3)3600} = 1.1(+4)$

I_{133}: $1.1(+2) \div e^{-[9.3(-6)](3)3600} = 1.2(+2)$

Te_{129}: $1.1(+3) \div e^{-[1.7(-4)](3)3600} = 6.9(+3)$

La_{142}: $1.1(+1) \div e^{-[1.2(-4)](3)3600} = 4.0(+1)$

Containment Sump

Xe_{133}: $1(-5) \div e^{-[1.5(-6)](4)3600} = 1(-5)$

I_{131}: $1(+2) \div e^{-[9.9(-7)](4)3600} = 1(+2)$

I_{133}: $1(0) \div e^{-[9.3(-6)](4)3600} = 1(0)$

Te_{129}: $1(+1) \div e^{-[1.7(-9)](4)3600} = 1.2(+2)$

La_{142}: $1(-1) \div e^{-[1.2(-4)](4)3600} = 5.6(-1)$

Containment Atmosphere

Xe_{133}: $1.3(-1) \div e^{-[1.5(-6)](5)3600} = 1.3(-1)$

I_{131}: $1.3(-1) \div e^{-[9.9(-7)](5)3600} = 1.3(-1)$

I_{133}: $1.3(-3) \div e^{-[9.3(-6)](5)3600} = 1.5(-3)$

These values are recorded on Enclosure 6.

ENCLOSURE 7

RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

Sample Number: 1

Location: RCS Hot Leg

Isotope	Decay Corrected Specific Activity (Enclosure 6), $\mu ci/cc$	Calculated Isotope Ratio*	Fuel Pellet Inventory	Activity Ratio In Gas Gap	Identified Source
Kr 87					
Xe 131m	--	--	0.003	0.003	--
Xe 133	1.1(+2)	1	1.0	1.0	NA
I 131	1.1(+4)	1	1.0	1.0	NA
I 132	--	--	1.4	0.01	--
I 133	1.2(+2)	1.1(-2)	2.0	0.5	Gas Gap
I 135	--	--	1.8	0.17	--

* Noble Gas Ratio = $\frac{\text{Decay Corrected Noble Gas Specific Activity}}{\text{Decay Corrected Xe 133 Specific Activity}}$

Iodine Ratio = $\frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I-131 Specific Activity}}$

ENCLOSURE 7

RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

Sample Number: 2

Location: Containment Sump

Isotope	Decay Corrected Specific Activity (Enclosure 6), $\mu ci/cc$	Calculated Isotope Ratio*	Fuel Pellet Inventory	Activity Ratio In Gas Gap	Identified Source
Kr 87					
Xe 131m	--	--	0.003	0.003	--
Xe 133	1(-5)	1	1.0	1.0	NA
I 131	1(+2)	1	1.0	1.0	NA
I 132	--	--	1.4	0.01	--
I 133	1(0)	1(-2)	2.0	0.5	Gas Gap
I 135	--	--	1.8	0.17	--

* Noble Gas Ratio = $\frac{\text{Decay Corrected Noble Gas Specific Activity}}{\text{Decay Corrected Xe 133 Specific Activity}}$

Iodine Ratio = $\frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I-131 Specific Activity}}$

ENCLOSURE 7

RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

Sample Number: 3

Location: Containment Atmosphere

Isotope	Decay Corrected Specific Activity (Enclosure 6), $\mu ci/cc$	Calculated Isotope Ratio*	Fuel Pellet Inventory	Activity Ratio In Gas Gap	Identified Source
Kr 87					
Xe 131m	--	--	0.003	0.003	--
Xe 133	1.3(-1)	1	1.0	1.0	NA
I 131	1.3(-1)	1	1.0	1.0	NA
I 132	--	--	1.4	0.01	--
I 133	1.5(-3)	1.2(-2)	2.0	0.5	Gas Gap
I 135	--	--	1.8	0.17	--

* Noble Gas Ratio = $\frac{\text{Decay Corrected Noble Gas Specific Activity}}{\text{Decay Corrected Xe 133 Specific Activity}}$

Iodine Ratio = $\frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I-131 Specific Activity}}$

ENCLOSURE 8

RECORD OF RELEASE QUANTITY

Isotope	Reactor Coolant Sample Number, Ci	Containment Sump Sample Number, Ci	Containment Atmosphere Sample Number , Ci	Total Quantity Ci
Kr 87				
Xe 131m	--	--	--	--
Xe 133	2.5(+4)	1.1(-2)	6.9(+3)	3.2(+4)
I 131	2.5(+6)	1.1(+5)	6.9(+3)	2.6(+6)
I 132	--	--	--	--
I 133	2.8(+4)	1.1(+3)	8.0(+1)	2.9(+4)
I 135	--	--	--	--
Cs 134	--	--	--	--
Rb 88	--	--	--	--
Te 129	1.6(+6)	1.3(+3)	--	1.6(+6)
Te 132	--	--	--	--
Sr 89	--	--	--	--
Ba 140	--	--	--	--
La 140	--	--	--	--
La 142	9.2(+3)	5.8(+2)	--	9.8(+3)
Pr 144	--	--	--	--

Volume corrections to STP:

RCS [RCS volume$_{STP}$ = 9400 ft^3 x .87 = 8178 ft^3 = 2.3(8) cc]

Xe_{133}: 1.1(+2) $\mu c/cc$ x 2.3(8)cc x 1(-6) $ci/\mu ci$ = 2.5(+4)

I_{131}: 1.1(+4) $\mu c/cc$ x 2.3(8)cc x 1(-6) $ci/\mu ci$ = 2.5(+6)

I_{133}: 1.2(+2) $\mu c/cc$ x 2.3(8)cc x 1(-6) $ci/\mu ci$ = 2.8(+4)

Te_{129}: 6.9(+3) $\mu c/cc$ x 2.3(8)cc x 1(-6) $ci/\mu ci$ = 1.6(+6)

La_{142}: 4.0(+1) $\mu c/cc$ x 2.3(8)cc x 1(-6) $ci/\mu ci$ = 9.2(+3)

Containment Sump [(@ 21' 40,000 ft^3 = 1.1(9) cc) See Enclosure 9]

Xe_{133}: 1(-5) x 1.1(9) 1(-6) = 1.1(-2)

I_{131}: 1(+2) x 1.1(9) 1(-6) = 1.1(+5)

I_{133}: 1(0) x 1.1(9) 1(-6) = 1.1(+3)

Te_{129}: 1.2(+2) x 1.1(9) 1(-6) = 1.3(-3)

La_{142}: 5.6(-1) x 1.1(9) 1(-6) = 5.8(+2)

Containment Atmosphere [volume @ STP 7.1 x 10^{10} cc x $\frac{14.7 + 0.5}{14.7}\frac{492}{680}$ = 5.3 x 10^{10} cc

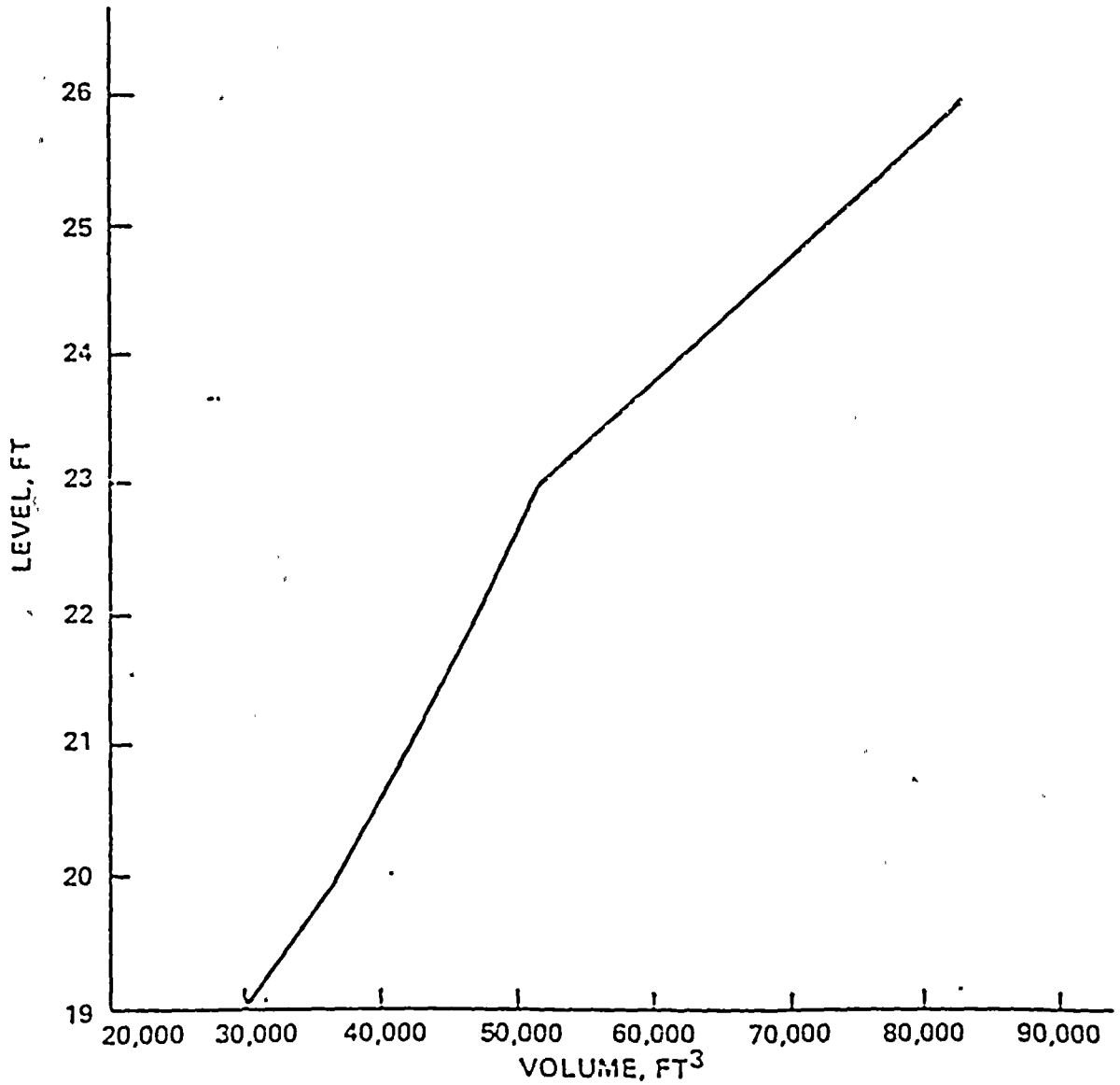
Xe_{133}: 1.3(-1) x 5.3(10) x 1(-6) = 6.9(+3)

I_{131}: 1.3(-1) x 5.3(10) x 1(-6) = 6.9(+3)

I_{133}: 1.5(-3) x 5.3(10) x 1(-6) = 8.0(+1)

These values are recorded on Enclosure 8.

ENCLOSURE 9
CONTAINMENT BUILDING WATER
LEVEL vs VOLUME



ENCLOSURE 11

RECORD OF TRANSIENT POWER CORRECTION

Sample Number: 1, 2, 3
Location:
Prior 30 Day Power History:

Power %	Duration, Days
75	22
50	17
100	2

Isotope	Power Correction Factor	x	Equilibrium Source Inventory	=	Power Corrected Source Inventory
Gas Gap Inventory					
Kr 87					
Xe 131m	--		--		--
Xe 133	0.63		1.3(7)		8.1(6)
I 131	0.63		6.7(6)		4.2(6)
I 132	--		--		--
I 133	.91		6.7(6)		6.1(6)
I 135	--		--		--
Fuel Pellet Inventory					
Kr 87					
Xe 131m	--		--		--
Xe 133	0.63		1.5(8)		9.4(9)
I 131	0.63		7.3(7)		4.6(7)
I 132	--		--		--
I 133	.91		1.5(8)		1.4(8)
I 135	--		--		--
Cs 134	--		--		--
Rb 88	--		--		--
Te 129	1.0		2.4(7)		2.4(7)
Te 132	--		--		--
Sr 89	--		--		--
Ba 140	--		--		--
La 140	--		--		--
La 142	1.0		1.6(8)		1.6(8)
Pr 144	--		--		--

POWER CORRECTION FACTORS

$$\text{Power Correction Factor} = \frac{\Sigma\, P_j\, (1-e^{-\lambda t_j})\, e^{-\lambda t_j^0}}{100}$$

For Xe_{133} the Power Correction Factor is calculated as follows:

$$\Sigma P_j(1-e^{-\lambda t_j})e^{-\lambda t_j^0} = 75(1-e^{-[1.5(-6)][1.9(+6)]})(e^{-[1.5(-6)][1.5(+6)]}) + 50(1-e^{-[1.5(-6)][1.5(+6)]})$$

$$(e^{-[1.5(-6)][1.7(+5)]}) + 100(1-e^{-[1.5(-6)][1.7(+5)]})(e^{\circ}) = 63.6$$

$$\text{Power Correction Factor} = \frac{63.6}{100} = 0.636$$

The remaining isotopes are calculated in the same manner; the results are recorded in Enclosure 11.

ENCLOSURE 12

RECORD OF PERCENT RELEASE

Isotope	Total Quantity Available For Release (Enclosure 8), Ci	÷	Power Corrected Source Inventory, Ci (Enclosure 10 or 11)	x 100 =	Percent
Gas Gap Inventory					
Kr 87					
Xe 131	--		--		--
Xe 133	3.2(+4)		8.1(6)		0.39
I 131	2.6(+6)		4.2(6)		62
I 132	--		--		--
I 133	2.9(+4)		6.1(6)		0.48
I 135	--		--		--
Fuel Pellet Inventory					
Kr 87					
Xe 131m	--		--		--
Xe 133	2.8(+4)		9.4(9)		0.0003
I 131	2.6(+6)		4.6(7)		5.7
I 132	--		--		--
I 133	2.9(+4)		1.4(8)		0.021
I 135	--		--		--
Cs 134	--		--		--
Rb 88	--		--		--
Te 129	1.6(+6)		2.4(7)		6.7
Te 132	--		--		--
Sr 89	--		--		--
Ba 140	--		--		--
La 140	--		--		--
La 142	9.8(+3)		1.6(8)		0.01
Pr 144	--		--		--

The following results are concluded:

(1) The characteristic fission products are I 131 and Te 129.

(2) The source of iodine release is principally from the fuel rod gas gap.

(3) 62 percent of the fuel rod gas gap I-131 inventory is available for release to the environment.

6.7 percent of the fuel pellet Te-129 inventory is available for release to the environment.

Based on these three pieces of information and the characteristics of the ten categories of core damage described in Enclosure 1 the following conclusion is drawn.

Conclusion:

The core damage is estimated to be Major Fuel Cladding Failure with concurrent Initial Fuel Pellet Overheating.

APPENDIX D.0

PROCEDURE GUIDELINE FOR ASSESSMENT OF CORE DAMAGE USING HYDROGEN

TABLE OF CONTENTS

LIST OF ENCLOSURES

TABLE OF CONTENTS (Cont'd.)

1.0 PURPOSE

This procedure is to be followed under post accident plant conditions to determine the extent of fuel clad damage which may have occurred. It utilizes hydrogen measured in samples obtained with the Post Accident Sampling System (PASS). The measured hydrogen is related to the amount of fuel clad oxidation. Clad oxidation is in turn related to clad damage which is expressed in terms of the percent of fuel rods which are ruptured and the percent which are embrittled. The resulting observation of damage is described by one or more of the ten categories of core damage in Enclosure 1.

2.0 REFERENCES

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, C-E Owners Group Task 467, May, 1983.

2.2 Post Accident Sampling System Operating Procedures. (Plant Specific Document)

2.3 NUREG 0737 Item II.B.3.

3.0 DEFINITIONS

Clad Rupture: The fuel clad ruptures when the internal gas pressure exceeds the external coolant pressure and the clad yield strength is reduced because of elevated temperatures. Clad rupture results in release of gaseous fission products from the gas gap and possibly some fragments of fuel pellets but does not otherwise destroy the structure of the fuel assembly.

Clad Embrittlement: At temperatures above the rupture temperature significant oxidation of the clad occurs. If the oxidation exceeds the embrittlement threshold, fragmentation of embrittled clad may subsequently occur from thermal shock, hydraulic pressure forces or handling such that the structure of the fuel assembly is destroyed and substantial fuel pellet fragments are released to the coolant.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the 10 categories of core damage in Enclosure 1. The procedure is based on hydrogen data. Other plant indications may be available which can improve upon estimation of core damage. These include radiologic sample characteristics, incore temperature indicators, and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

1.0 PURPOSE

This procedure is to be followed under post accident plant conditions to determine the extent of fuel clad damage which may have occurred. It utilizes hydrogen measured in samples obtained with the Post Accident Sampling System (PASS). The measured hydrogen is related to the amount of fuel clad oxidation. Clad oxidation is in turn related to clad damage which is expressed in terms of the percent of fuel rods which are ruptured and the percent which are embrittled. The resulting observation of damage is described by one or more of the ten categories of core damage in Enclosure 1.

2.0 REFERENCES

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, C-E Owners Group Task 467, May, 1983.

2.2 Post Accident Sampling System Operating Procedures. (Plant Specific Document)

2.3 NUREG 0737 Item II.B.3.

3.0 DEFINITIONS

Clad Rupture: The fuel clad ruptures when the internal gas pressure exceeds the external coolant pressure and the clad yield strength is reduced because of elevated temperatures. Clad rupture results in release of gaseous fission products from the gas gap and possibly some fragments of fuel pellets but does not otherwise destroy the structure of the fuel assembly.

Clad Embrittlement: At temperatures above the rupture temperature significant oxidation of the clad occurs. If the oxidation exceeds the embrittlement threshold, fragmentation of embrittled clad may subsequently occur from thermal shock, hydraulic pressure forces or handling such that the structure of the fuel assembly is destroyed and substantial fuel pellet fragments are released to the coolant.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the 10 categories of core damage in Enclosure 1. The procedure is based on hydrogen data. Other plant indications may be available which can improve upon estimation of core damage. These include radiologic sample characteristics, incore temperature indicators, and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 This procedure relies upon hydrogen samples taken from the containment atmosphere and the reactor coolant system hot leg. Those samples may contain a mixture of hydrogen generated within the core by clad oxidation and also hydrogen from radiolytic dissociation of water and oxidation of aluminum and zinc in the containment. The estimate of clad damage is influenced by the amount of hydrogen generated by ex-core sources and by the ability to identify plant conditions conducive to such hydrogen generation. Therefore, a hydrogen measurement is not a unique indicator of the amount of core clad oxidation.

4.3 There are large areas of aluminum components in the containment building of some plants. This aluminum would oxidize rapidly at temperatures about 200°F and would be consumed within about two hours. The remainder of the aluminum and other oxidizing material react at a rate determined by temperature and over a longer time. In the procedure all of the short term transient hydrogen is generated within the first two hours and is added to the slower accumulation as a function of time. Hence, in containments with large areas of rapidly reacting aluminum, the procedure is valid for hydrogen samples taken after about two hours with temperatures about 200°F, or after the short term oxidation is complete.

4.4 This procedure yields estimates of the percentages of fuel rods with ruptured clad and embrittled clad. Simultaneous with embrittling of the clad, there may be clad melting and pellet overheating occurring. This procedure provides an estimate of only the percentage of rods which have progressed to at least clad rupture or clad embrittlement, and does not attempt to predict the physical configuration of those rods which have progressed beyond local clad fragmentation.

4.5 Depending on the accident scenario, a given total amount of hydrogen produced by oxidation of fuel clad can represent varying local amounts and distributions of clad damage. This procedure attempts to bias the damage estimates such that the results represent lower limit estimates of clad damage. Actual damage could be greater, depending on plant specific details and on the accident scenario.

4.6 This procedure is applicable under conditions for which there are no voids measurable by the Reactor Vessel Level Monitoring System. It is assumed that if such voids had been found, their removal would be accomplished by using the Reactor Vessel Vent System as prescribed elsewhere in the actions to mitigate the consequences of accidents. However, if the hydrogen samples are taken under conditions in which measurable void does exist, a guideline for analysis is provided in the addendum attached to this procedure to estimate the contribution of that source to be added to the total hydrogen measured.

5.0 INITIAL PLANT CONDITIONS AND SYMPTOMS

This procedure is to be employed for analysis of hydrogen sample data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. This list is not a complete representation of all events which may cause core damage. One or more of these symptoms may exist at or before the time the sample is obtained. Under these conditions, sampling should be performed using the Post Accident Sampling System.

5.1 High alarm on the containment radiation monitor.

5.2 High alarm on the CVCS letdown radiation monitor.

5.3 High alarm on the main condenser air ejector exhaust radiation monitor.

5.4 Pressurizer level low.

5.5 Safety Injection System may have automatically actuated.

5.6 Possible high quench tank level, temperature, or pressure.

5.8 Possible noise indicative of a high energy line break.

5.9 Decrease in volume control tank level.

5.10 Standby charging pumps energized.

5.11 Unbalanced charging and letdown flow.

5.12 Reactor Coolant System subcooling low or zero or superheated.

6.0 PREREQUISITES

An operational Post Accident Sampling System with the capability to obtain and analyze the concentration of hydrogen in fluid samples which have the potential to be highly radioactive. The system should meet the requirements of NUREG-0737 Item II.B.3,.Reference 2.3.

7.0 PROCEDURE

7.1 Record the Following Plant Indicators

7.1.1 Core damage can occur following reactor trip only when the coolant level within the reactor vessel drops below the top of the active fuel. Several instrument records are available from which an estimate of the core uncovery and recovery times might be made. The instruments are:

Reactor Vessel Level Monitoring System
Core Exit Thermocouple Temperature
Core Exit Thermocouple Saturation Margin

Record data from these instruments according to the instructions on the worksheet of Enclosure 2.

7.1.2 The magnitude of Reactor Coolant System (RCS) pressure during the core uncovery period can influence the number of early clad ruptures. Interpret the data from Step 7.1.1 to determine the best estimate for the time period of core uncovery and determine the range of RCS pressure during this time period. Record on the Enclosure 2 worksheet.

7.1.3 The presence of some subcooled inlet flow while the core is uncovering can slow the uncovery and cause greater local clad oxidation for a given total amount of core oxidation, thereby leading to a greater underestimate of the number of damaged rods predicted by this procedure. Observe available instrument records to determine if there was some reactor vessel inlet flow during the rising temperature portion of the core uncovery period. Include net flow from charging and letdown systems, HPSI, LPSI, spray, etc. Record the data on the Enclosure 2 worksheet.

7.1.4 Record the conditions in the containment and the reactor coolant system at the time the hydrogen samples are obtained in Step 7.2 following. Enter on the worksheet of Enclosure 3.

7.2 Obtain a liquid sample from the RCS hot leg and a gas sample from the containment atmosphere and analyze them for hydrogen concentration using the procedures for Post Accident Sample System operation described in Reference 2.2. Record the results on the worksheet of Enclosure 3. Follow the instructions on Enclosure 3 to obtain the total amount of hydrogen measured in units of cubic feet of hydrogen at standard temperature and pressure.

7.3 The total measured hydrogen in Step 7.2 includes the hydrogen generated by three processes: 1) core clad oxidation, 2) radiolysis of water and 3) oxidation of containment materials such as aluminum and zinc. The amount of hydrogen generated by the last two processes is calculated and then subtracted from the total measured to yield the amount generated by core clad oxidation.

Enclosure 5 is a worksheet for calculating the amount of hydrogen generated by oxidation of materials within the containment. It utilizes measured data for the containment temperature as a function of time up to the sampling time and a plant specific curve of the rate of production as a function of containment temperature in Enclosure 6. Record the data required on Enclosure 5 and complete the indicated calculations to obtain the cubic feet of hydrogen at STP generated by containment materials oxidation.

7.4 The hydrogen generated by radiolysis is a function of operating power and decay time. Record the data required on the worksheet of Enclosure 7, and utilize the curve of Enclosure 8 to obtain the cubic feet of hydrogen at STP generated by radiolysis. The appropriate power is determined as follows:

7.4.1 For the case in which the operating power is constant or has not changed by more than ±10 percent for a period greater than 30 days, that power is used.

7.4.2 For the case in which the power has not remained constant during the 30 days prior to the reactor shutdown engineering judgement is used to determine the most representative power level. The following guidelines should be considered in the determination.

7.4.2.1 The average power during the 30 day time period is not necessarily the most representative value for determining radiolysis by fission products.

7.4.2.2 The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

7.4.2.3 Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

7.4.3 For the case in which the reactor has produced power for less than 30 days, the procedure may be employed. However, the estimate of hydrogen from radiolysis will be too high and therefore the calculated hydrogen by core oxidation will be too low. Hence an underprediction of core damage may result.

7.5 Enter the amounts of hydrogen from Steps 7.2, 7.3 and 7.4 on the worksheet of Enclosure 9. Subtract the amounts in 7.3 and 7.4 from 7.2 as indicated on the worksheet to yield the cubic feet of hydrogen generated by core clad oxidation. Adjust with the plant specific constant as shown on the worksheet to obtain the estimated percent of the core clad which is oxidized.

7.6 Enter the abscissa of the curve on Enclosure 10 with the percent of core clad oxidized from Step 7.5. Use the curve labeled with the pressure closest to but greater than the RCS pressure during the core uncovery period as obtained in Step 7.1.2 and recorded on Enclosure 2. Read on the ordinate of Enclosure 10, the percent of fuel rods with ruptured clad. Record on the worksheet of Enclosure 9. Note that the sensitivity of measurement of hydrogen is comparable to the range of oxidation on Enclosure 10. Hence, small amounts of clad rupture are not reliably predicted by this procedure.

7.7 Enter the abscissa of the curve on Enclosure 11 with the percent of core clad oxidized from Step 7.5. Read on the ordinate the lower

and upper values of the range indicated by the curve for the percent of fuel rods which have embrittled clad. Record on the worksheet of Enclosure 9.

7.8 For a given percent oxidation of the core clad, the lower limit estimate of embrittled clad in Step 7.7 is, for most accident scenarios, the least amount of potential fuel structural failure. Actual values are probably greater. The upper limit of the range in Step 7.5 may be interpreted as follows:

7.8.1 When the pressure during uncovery, from Step 7.1.2 and recorded on Enclosure 2, is less than about 100 psia, a rapid core uncovery by blowdown is concluded. Heatup with minimum clad oxidation occurs. The extent of potential clad structural failure by melting may be greater than the upper limit of embrittlement from Step 7.7 as determined by oxidation. Hence, use the upper limit from Step 7.7.

7.8.2 When there is inlet flow while the core is uncovering, the rate of uncovery is slower than assumed in the derivation of the curves on Enclosures 10 and 11. For a measured total amount of oxidation, the local percentage oxidation is probably greater along a shorter length of the upper portions of the fuel. Hence, favor the upper limit from Step 7.7.

7.9 CORE DAMAGE ASSESSMENT

The conclusion on core damage is made using the two results from above. These are:

1. Percentage of fuel rods with ruptured clad, Step 7.6.

2. Percentage of fuel rods with embrittled or structurally damaged clad, Step 7.7.

Knowledgeable judgement is used to compare the above two results to the definitions of the 10 NRC categories of fuel damage found in Enclosure 1. Core damage does not take place uniformly. Therefore when evaluating damage using these results, Enclosure 1 may yield a combination of categories of damage which exist simultaneously.

ENCLOSURE 1

CLAD DAMAGE CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

	NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1.	No Fuel Damage	~750	None	--	--	Less Than 1
2.	Initial Cladding Failure	1200-1800	Rupture Due to Gas Gap Overpressurization	Maximum Core Exit Thermocouple Temperature	<1550°F*	Less Than 10
3.	Intermediate Cladding Failure				<1700°F*	10 to 50
4.	Major Cladding Failure				≲2300°F ≲2% Oxidation	Greater Than 50
5.	Initial Fuel Pellet Overheating	1800-3350	Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6.	Intermediate Fuel Pellet Overheating				<18%	10 to 50
7.	Major Fuel Pellet Overheating				≲65%	Greater Than 50

*Depends on Reactor Pressure and Fuel Burnup. Given for Pressure ≤1200 psia and Burnup ≥0.

ENCLOSURE 2

CORE UNCOVERY CONDITIONS

Step 7.1.1 Time period of core uncovery. Complete the following table using recorded instrument data.

Instrument	Estimated Core Uncovery Time	Estimated Core Recovery Time
Reactor Vessel Level Monitoring System	Lower Limit Elevation Uncovers. Time _____	Lower Limit Elevation Recovers. Time _____
Core Exit Thermocouple Temperature	Start of Continuous Rise or Exceed 660°F. Time _____ Temperature _____	Rapid Temperature Drop to Saturation. Time _____ Temperature _____
Core Exit Thermocouple Saturation Margin	Start of Superheat. Time _____	Return to Saturation or Subcooling. Time _____

Step 7.1.2 Interpret above data to obtain best estimate for time period of core uncovery and obtain pressurizer pressure range during that period. The superheat derived from the thermocouple temperature and corresponding system pressure is considered as the best indicator for core uncovery during boiloff and should be used, but should be compared with the other indicators to help identify possible anomalies. The pressure during uncovery is used later on Enclosure 8, Step 7.6, to determine the appropriate curve for assessment of the number of clad ruptures.

	Core Uncovery	Core Recovery
Time	_____	_____
Pressure	_____	_____

Step 7.1.3 Estimate vessel inlet flow rates during core uncovery heatup period, up to approximately the time of peak core exit thermocouple temperature. Net inlet flow indicates that procedure may have additional bias which underpredicts clad damage.

Charging Flow Rate _____

Letdown Flow Rate _____

HPSI Flow Rate _____

LPSI Flow Rate _____

Other Inlet Flows _____

ENCLOSURE 3

SAMPLING CONDITIONS AND MEASURED HYDROGEN

Step 7.1.4 Obtain the RCS and containment conditions at the time of sampling for hydrogen.

Reactor Coolant System		Containment	
Sampling Time	_____	Atmosphere Pressure	_____ psig
Pressure	_____ psig	Atmosphere Temperature	_____ °F
Temperature	_____ °F	Has Hydrogen Recombiner Operated	Yes/No
Reactor Vessel Coolant Level	_____ %	Does Pressure or Temperature History Indicate a Hydrogen Burn	Yes/No
Pressurizer Level	_____ %		

Step 7.2 Hydrogen Sample Data Reduction.

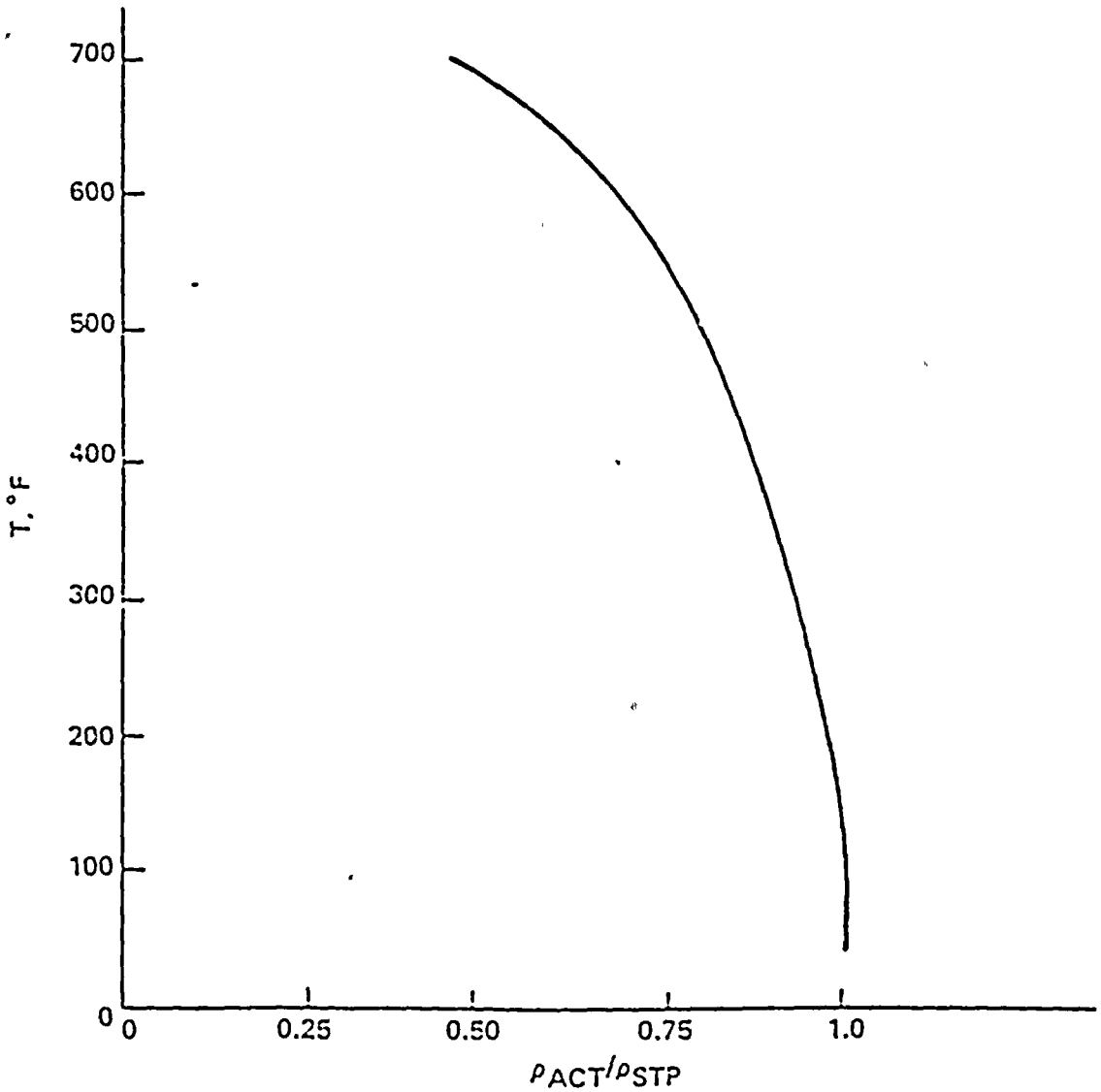
Cont. Sample (Vol. %/100)	x	Cont. Vol. (Ft^3)	x	(32 + 460)	÷	(Normal Temp. + 460)	=	Ft^3 Hydrogen at STP
_____	x	_____	x	492	÷	_____	=	_____

Hot Leg Sample (cc/kg @ STP)	x	RCS Vol. (Ft^3)	x	Density Ratio (Enclosure 4)	÷	1000	=	Ft^3 Hydrogen at STP
_____	x	_____	x	_____	÷	1000	=	_____
						Total	=	_____

Also record total on Enclosure 9.

ENCLOSURE 4

RATIO OF H_2O DENSITY TO H_2O DENSITY AT STP vs TEMPERATURE



ENCLOSURE 5

HYDROGEN GENERATED IN CONTAINMENT

Step 7.3 Record the containment temperature at selected time intervals and calculate the hydrogen generated by oxidation of containment materials utilizing the plant-specific production rates from Enclosure 6.

1	2	3	4	5
Time at Start of Intervals	Interval Duration (hr)	Avg. Containment Temp. During Interval (°F)	H_2 Prod. Rate (ft^3/hr, Enclosure 6)	H_2 Produced = 2 x 4
Accident Starts				

Sampling Time

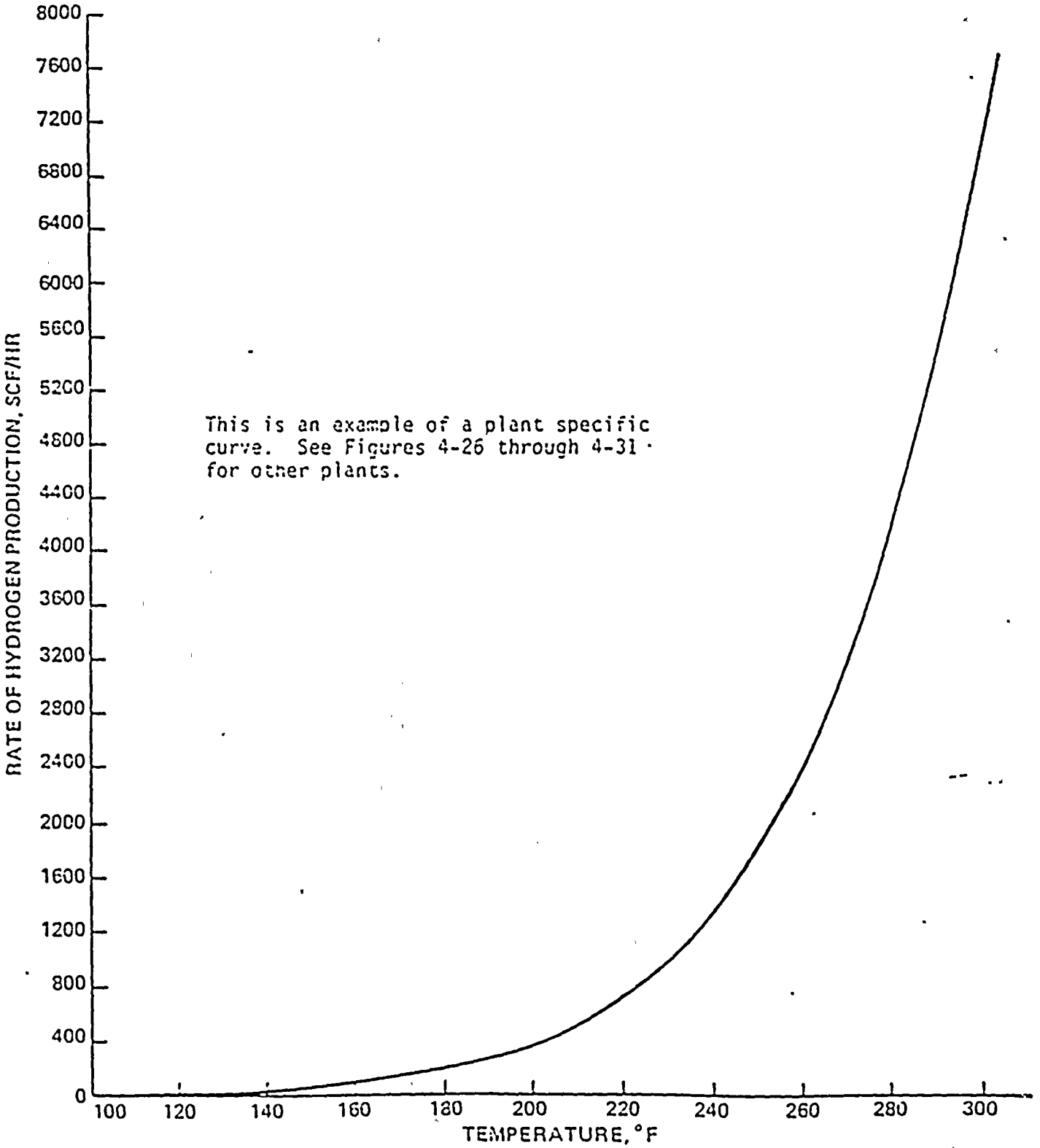
Long Term Hydrogen Production in Containment, Total = _____ ft^3 @ STP

Short term rapid hydrogen production by containment aluminum . (Table 4-3, Section 4.5) _____

Total Hydrogen Production in Containment _____ SCF

Record total on Enclosure 9 also.

ENCLOSURE 6
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE FOR ST. LUCIE UNIT 2



ENCLOSURE 7

HYDROGEN GENERATED BY RADICLYSIS

Step 7.4 Record the following data and utilize the curves of Enclosure 8 to determine the hydrogen generated by radiolysis.

Prior 30 day power history	Power, Percent	Duration, Days
	_____	_____
	_____	_____
	_____	_____
	_____	_____

Power to use in evaluating long term hydrogen production by radiolysis = (Full Power, Mwt) x _____.

Reactor Trip Time _____ hrs

Sampling Time (see Enclosure 3) _____ hrs

Decay Time (Sampling Time - Trip Time) _____ hrs

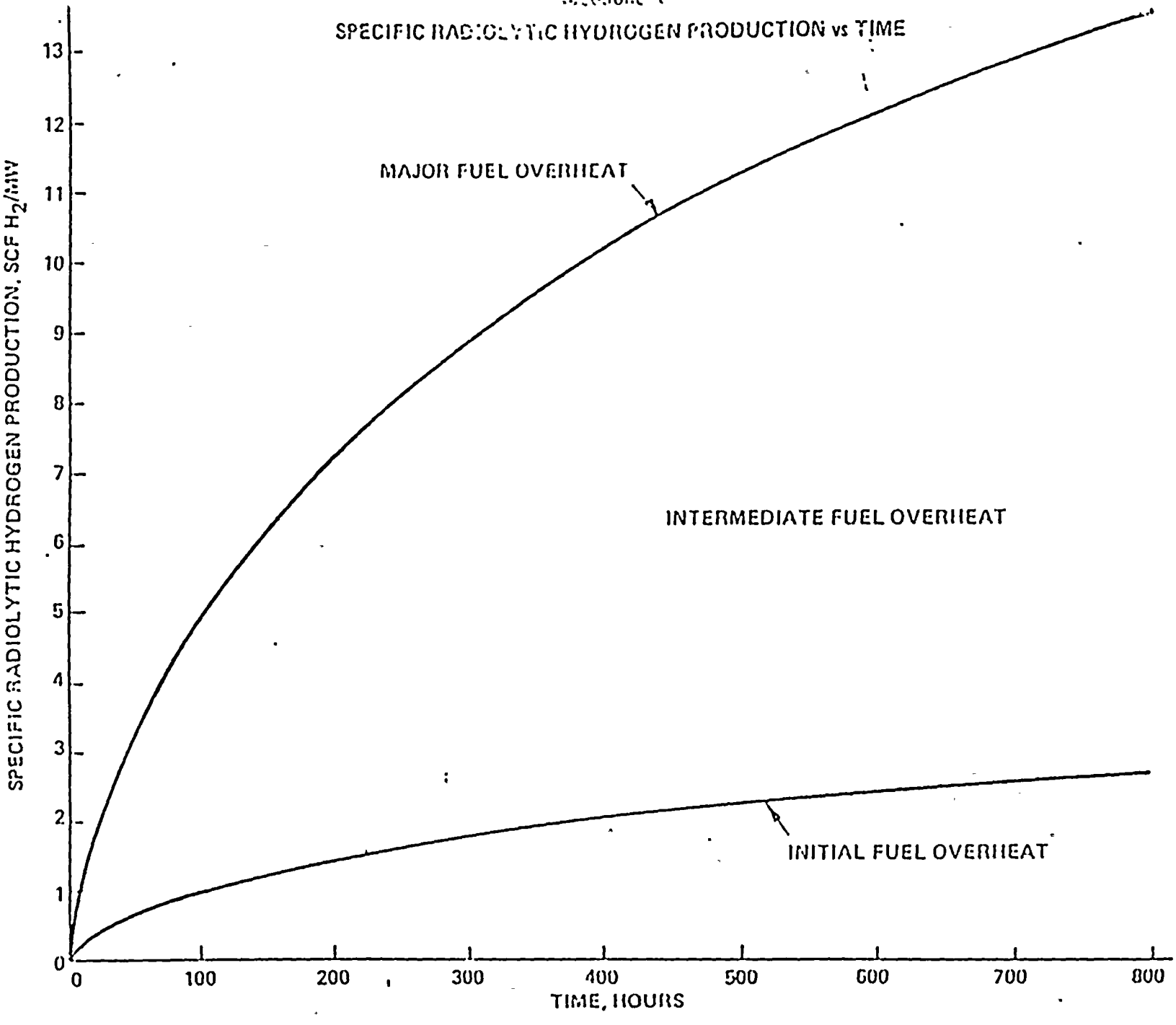
Enter abscissa on Enclosure 8 with above decay time and read two values of hydrogen produced by radiolysis, one from each curve, in cubic feet of hydrogen at STP per Mwt operating power. Multiply by above power and record as follows:

Limit Curve	Hydrogen Produced (SCF/Mwt, Enclosure 8)	x	Operating Power	=	Total Hydrogen Produced (SCF)
Upper	_____		_____		_____
Lower	_____		_____		_____

Using results from Radiological Damage Assessment Procedure estimate which results should be used; upper limit for major fuel overheat, lower limit for initial fuel overheat or appropriate estimate between the two curves for intermediate fuel overheat. Circle corresponding value of hydrogen above and also record on Enclosure 9.

FIGURE 9

SPECIFIC RADIOLYTIC HYDROGEN PRODUCTION vs TIME



ENCLOSURE 9

CORE DAMAGE ASSESSMENT FROM

HYDROGEN MEASUREMENT

Step 7.5 Hydrogen Measured, Step 7.2, Enclosure 3 _____ SCF

Hydrogen Produced in Containment, Step 7.3, Enclosure 5 _____ SCF

Hydrogen Produced by Radiolysis, Step 7.4, Enclosure 7, _____ -SCF

Subtract Step 7.3 and 7.4 from 7.2 to Get Hydrogen Procuced by Core Clad Oxidation _____ SCF

Divide by (* SCF/1% Clad Oxidized) = _____ %

= % Core Oxidized

Step 7.6 Enter abscissa on Enclosure 10 with "% Core Oxidized" and read ordinate from curve labelec with pressure during core uncovery as given on Enclosure 2, Step 7.1.2. Record here Percent of Fuel Rods with Ruptured Clad _____ %.

Step 7.7 Enter abscissa on Enclosure 11 with above "% Core Oxidized" and read range of values on ordinate. Record here

Percent of fuel rods embrittled

Range - Upper _____ %

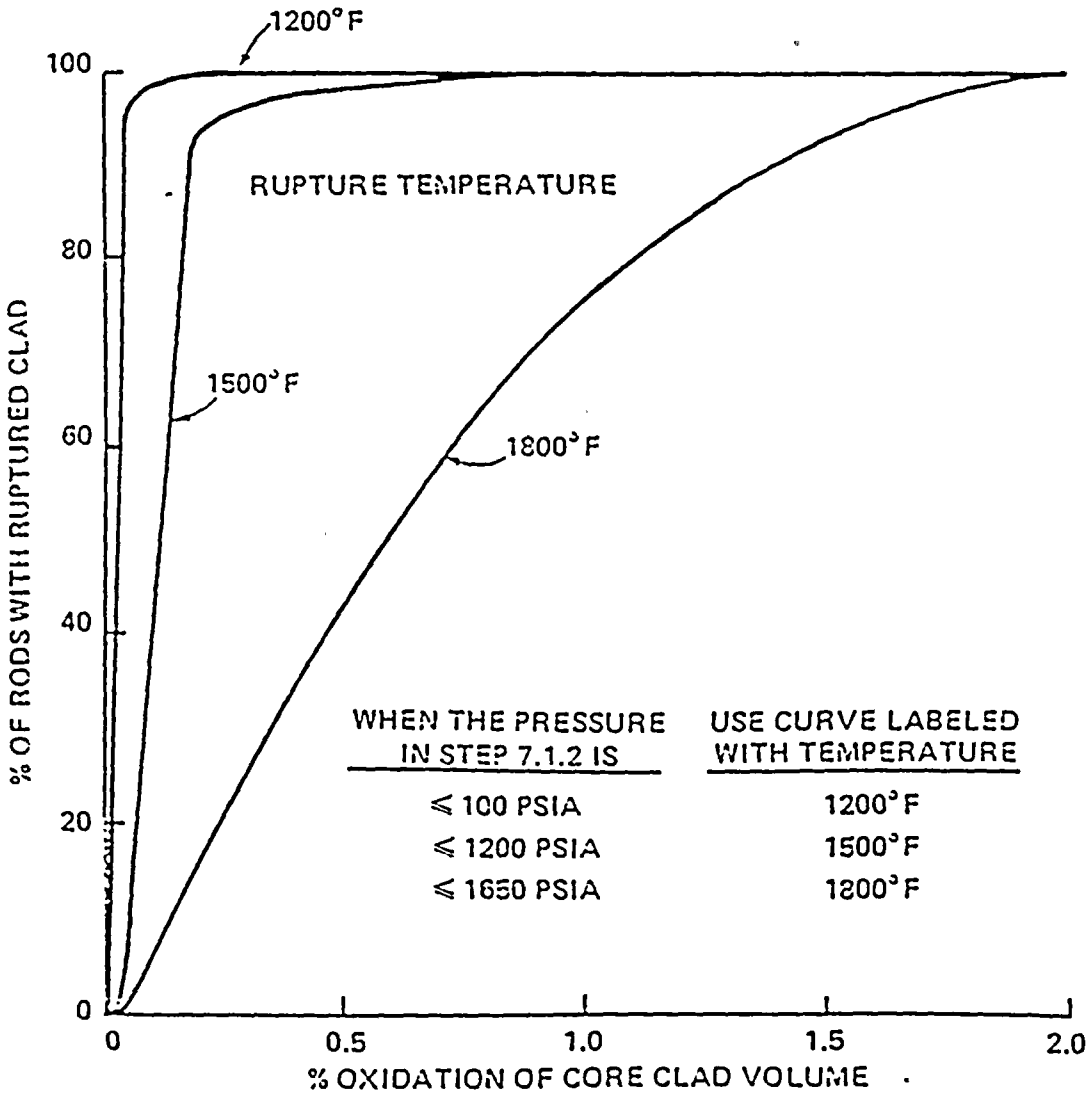
- Lower _____ %

Step 7.8 Review Step 7.1 and Bases sections to determine which of these limits is more likely to be representative of the core damage.

Step 7.9 From Enclosure 1 select the core clad damage categories based on the above percentages of rods ruptured and rods embrittled.

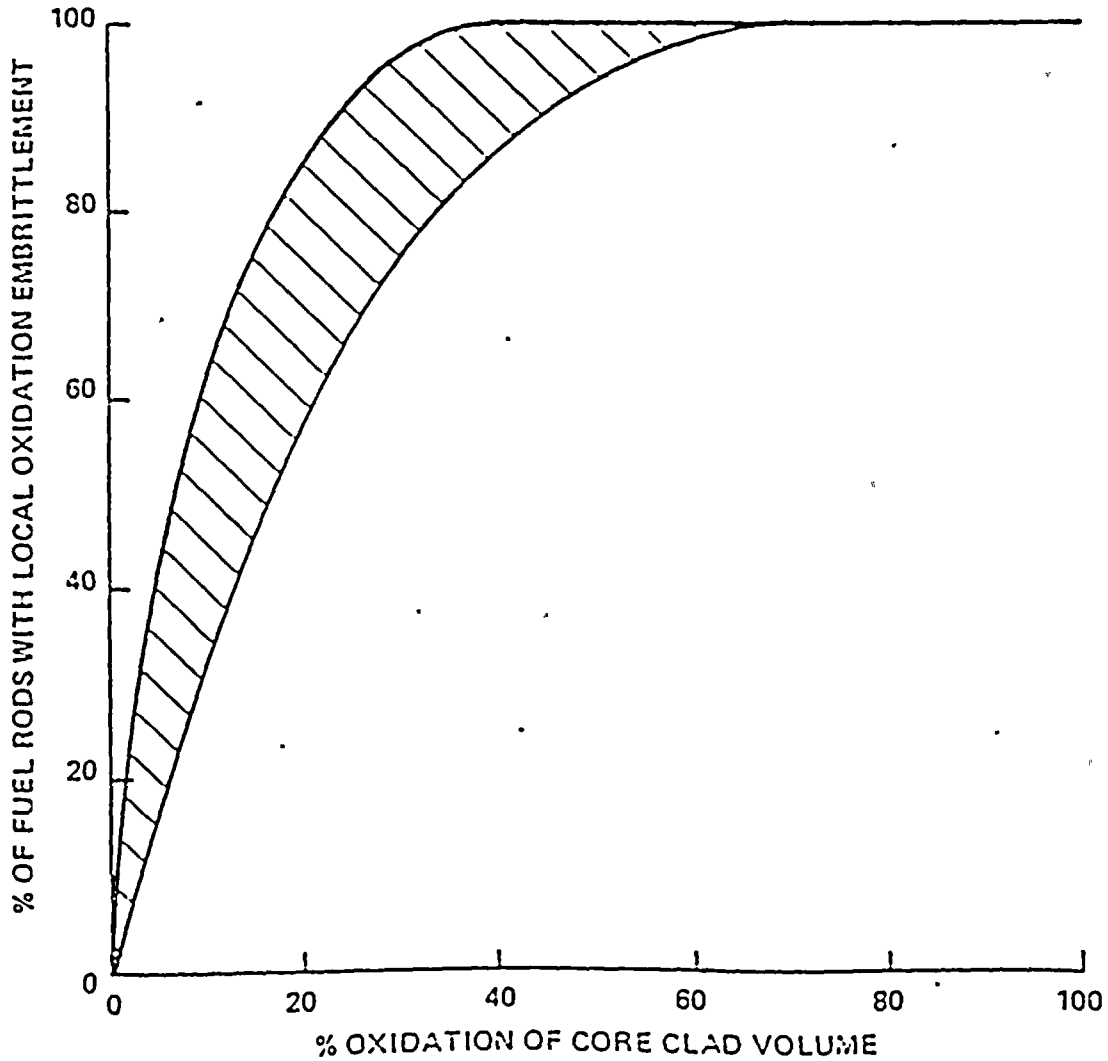
(*) Plant Specific Factor from column 1, Table 4-2 of Section 4.5.

ENCLOSURE 10
PERCENT OF FUEL RODS WITH RUPTURED
CLAD vs CORE CLAD OXIDATION



WHEN THE PRESSURE IN STEP 7.1.2 IS	USE CURVE LABELED WITH TEMPERATURE
≤ 100 PSIA	1200°F
≤ 1200 PSIA	1500°F
≤ 1650 PSIA	1200°F

ENCLOSURE 11
% OF THE FUEL RODS WITH OXIDATION EMBRITTLEMENT vs
TOTAL CORE OXIDATION
FOR 1% TO 3% DECAY HEAT AND 300 PSIA TO 2500 PSIA
WHEN COOLANT LEVEL DROPS BY BOILOFF WITH
NO INLET FLOW UNTIL CORE IS RAPIDLY QUENCHED



ADDENDUM TO APPENDIX D.0

ESTIMATION OF AMOUNT OF HYDROGEN

IN REACTOR VESSEL HEAD VOID

1.0 PURPOSE

The purpose of this addendum is to provide a guideline for an analytical procedure to calculate the amount of hydrogen gas contained in a void in the top of the reactor vessel. This hydrogen is added to the measured amount in Step 7.5 of the procedure to determine the total hydrogen generated by all sources.

2.0 LIMITATIONS

2.1 The preferred method of determining the amount of hydrogen in the primary system is to sample liquid from the hot leg when the system is full. However, if the system cannot be filled, a procedure based on this addendum could be used to estimate the hydrogen which is in the vessel void and which would not be evident from the hot leg liquid sample.

2.2 This guideline applies when the coolant level is above the hot leg and the remainder of the primary system is filled. Verification that the steam generator tubes are filled can be provided by the existence of natural convection flow in the primary system. If the coolant level is below the hot leg, the guidelines of this addendum do not apply.

2.3 A reactor vessel level monitoring system is required which can provide the coolant level. The volume of the void is obtained by relating the volume in the vessel above the coolant level to the value of level for each specific reactor vessel design.

2.4 This guideline provides the analytical means for only an estimate of the hydrogen contained in the void. The presence of other gases including helium, nitrogen and fission product gases will add uncertainty to the result.

3.0 PROCEDURE

The following is a guideline for an analytical procedure to be followed to estimate the amount of hydrogen contained in a void in the top of the reactor vessel. Calculational details and plant specific information must be included to implement this guideline.

3.1 Determine the conditions of the void as follows:

V = Void volume (Ft^3) derived from measurement of coolant level
T_L = Temperature of liquid at coolant surface (°F)
P_{sat} = Water saturation pressure at temperature T_L
P_{to} = Reactor coolant system pressure (psia)

3.2 A first approximation is made assuming the following:

3.2.1 The partial pressure of vapor in the void is assumed equal to saturation pressure at the liquid temperature, T_L. This implies no

heating of the void gas by the reactor vessel walls and head. They are normally at reactor outlet temperature and could remain above the temperature of the void causing the vapor to be superheated.

3.2.2 All the non-condensible gas in the void is hydrogen. This implies no helium or fission product gas from ruptured fuel rods and no nitrogen from Safety Injection Tanks. A second approximation which eliminates this assumption is given in 3.4.

3.3 Calculate the amount of hydrogen as follows:

$$P_{H_2} = P_{tot} - P_{sat};$$

$$Ft^3\ H_2\ @\ STP = (V)\left(\frac{P_{H_2}}{14.7}\right)\left(\frac{492}{T_L + 460}\right)$$

Add this amount to the total hydrogen in Step 7.5 of Appendix D.0.

3.4 A second approximation can be made in plants with a C-E designed PASS which measures both total gas and hydrogen which are dissolved in the hot leg liquid sample. This approximation includes the following assumptions regarding the relative solubilities of the non-condensible gases in the liquid.

3.4.1 The gases are assumed to have the same values of Henry's law constant which relates the partial pressure of gas to the amount of gas dissolved in the liquid sample at equilibrium.

3.4.2 When the dissolved gas is not in equilibrium with the gas in the void, the dissolved concentrations are in the same relative proportion as if equilibrium did exist.

3.5 The partial pressure of hydrogen is calculated from

$$P_{H_2} = (P_{tot} - P_{sat}) \cdot \frac{(cc/kg)_{H_2}}{(cc/kg)_{total}}$$

and the amount of hydrogen in the vessel head void is calculated using the equation above in 3.3.

3.6 This procedure can be extended to include specific values of Henry's law constants but the assumption of equilibrium at the gas liquid interface would still be questionable. Also, to utilize detailed values of the gas constants, the individual gases in the sample would have to be identified and measured. This would require additional measurement capability.

APPENDIX D.1

EXAMPLE USE OF THE PROCEDURE

The following is an example of the use of the procedure for assessment of clad damage by utilizing the total amount of hydrogen generation. The specific case cited is for the 2560 Mwt class of reactor. Quantities in the example are selected for an accident equivalent to that used in the example for core damage assessment based on radiologic characteristics and given in Appendix C.1. This example consists of a set of completed worksheets given in the procedure and the accompanying figures and plant specific information.

EXAMPLE

ENCLOSURE 1

CLAD DAMAGE CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

	NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1.	No Fuel Damage	~750	None	---	--	Less Than 1
2.	Initial Cladding Failure	1200-1800	Rupture Due to Gas Gap	Maximum Core Exit Thermocouple Temperature	<1550°F*	Less Than 10
3.	Intermediate Cladding Failure		Overpressurization		<1700°F*	10 to 50
4.	Major Cladding Failure				≲2300°F ≲2% Oxidation	Greater Than 50
5.	Initial Fuel Pellet Overheating	1800-3350	Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6.	Intermediate Fuel Pellet Overheating				<18%	
7.	Major Fuel Pellet Overheating				≲65%	Greater Than 50

*Depends on Reactor Pressure and Fuel Burnup. Value given for Pressure ≤1200 psia and Burnup ≥0.

ENCLOSURE 2

CORE UNCOVERY CONDITIONS

Step 7.1.1 Time period of core uncovery. Complete the following table using recorded instrument data.

Instrument	Estimated Core Uncovery Time	Estimated Core Recovery Time
Reactor Vessel Level Monitoring System	Lower Limit Elevation Uncovers. Time 0200 on 12/25/82	Lower Limit Elevation Recovers. Time 0235
Core Exit Thermocouple Temperature	Start of Continuous Rise or Exceed 660°F. Time 0210 Temperature 660°F	Rapid Temperature Drop to Saturation. Time 0235 Temperature 632°F
Core Exit Thermocouple Saturation Margin	Start of Superheat. Time 0205	Return to Saturation or Subcooling. Time 0235

Step 7.1.2 Interpret above data to obtain best estimate for time period of core uncovery and obtain pressurizer pressure range during that period. The superheat derived from the thermocouple temperature and corresponding system pressure is considered as the best indicator for core uncovery during boiloff and should be used, but should be compared with the other indicators to help identify possible anomalies. The pressure during uncovery is used later on Enclosure 8, Step 7.6, to determine the appropriate curve for assessment of the number of clad ruptures.

	Core Uncovery	Core Recovery
Time	0205	0235
Pressure	1000	900

Step 7.1.3 Estimate vessel inlet flow rates during core uncovery heatup period, up to approximately the time of peak core exit thermocouple temperature. Net inlet flow indicates that procedure may have additional bias which underpredicts clad damage.

Charging Flow Rate 0 GPM

Letdown Flow Rate 0

HPSI Flow Rate 0

LPSI Flow Rate 0

Other Inlet Flows 0

ENCLOSURE 3

SAMPLING CONDITIONS AND MEASURED HYDROGEN

Step 7.1.4 Obtain the RCS and containment conditions at the time of sampling for hydrogen.

Reactor Coolant System		Containment	
Sampling Time	0400, 12/25/82	Atmosphere Pressure	0.5 psig
Pressure	1600 psig	Atmosphere Temperature	220 °F
Temperature	300 °F	Has Hydrogen Recombiner Operated	No
Reactor Vessel Coolant Level	100 %	Does Pressure or Temperature History Indicate a Hydrogen Burn	No
Pressurizer Level	80 %		

Step 7.2 Hydrogen Sample Data Reduction.

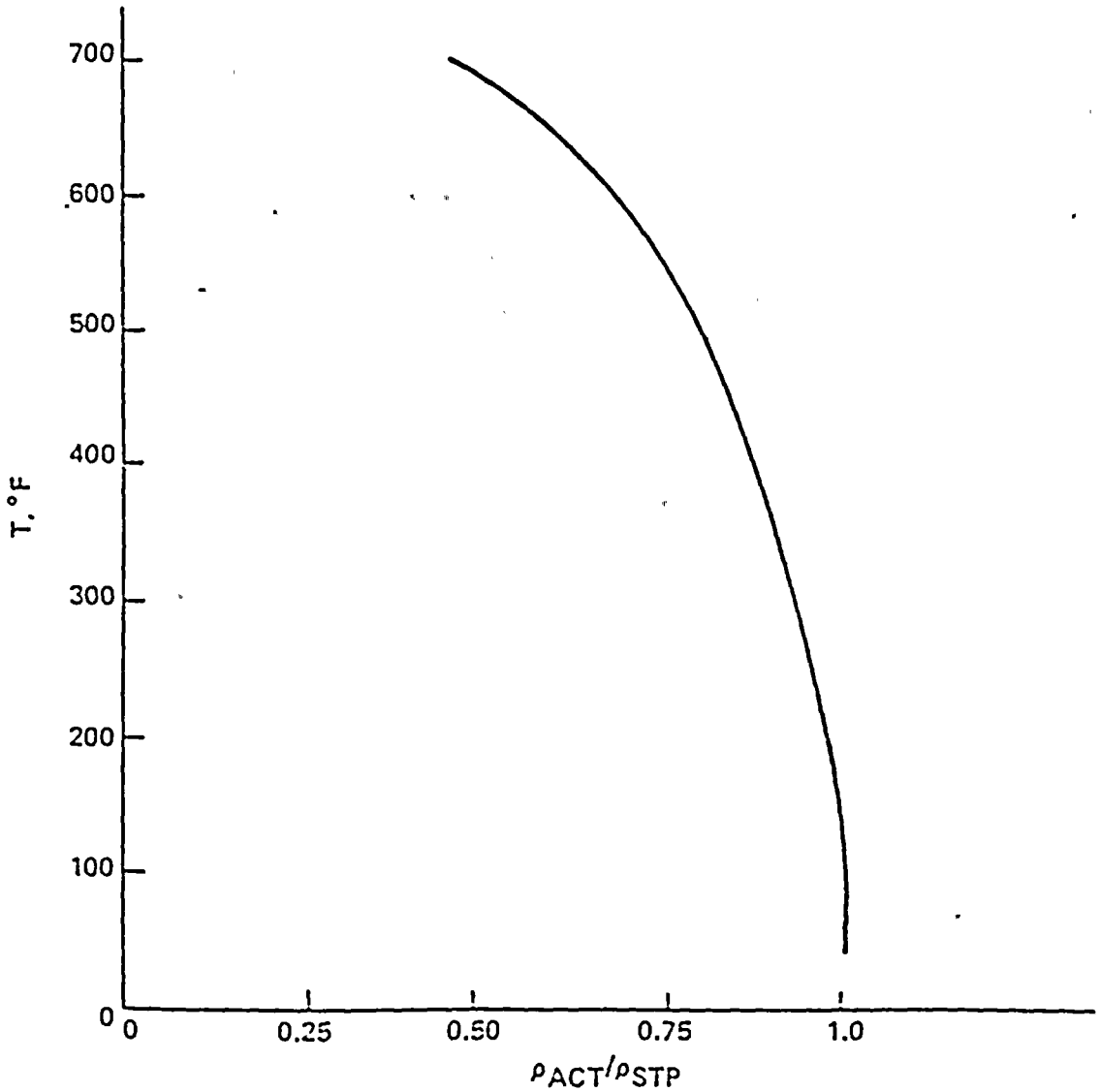
Cont. Sample (Vol. %/100)	x	Cont. Vol. (Ft^3)	x	(32 + 460)	÷	(Normal Temp. + 460)	=	Ft^3 Hydrogen at STP
0.005	x	2.5×10^6	x	492	÷	580	=	10,600

Hot Leg Sample (cc/kg @ STP)	x	RCS Vol. (Ft^3)	x	Density Ratio (Enclosure 4)	÷	1000	=	Ft^3 Hydrogen at STP
1200	x	9400	x	0.9	÷	1000	=	10,200
						Total	=	20,800

Also record total on Enclosure 9.

ENCLOSURE 4

RATIO OF H_2O DENSITY TO H_2O DENSITY AT STP vs TEMPERATURE



ENCLOSURE 5

HYDROGEN GENERATED IN CONTAINMENT

Step 7.3 Record the containment temperature at selected time intervals and calculate the hydrogen generated by oxidation of containment materials utilizing the plant-specific production rates from Enclosure 6.

1	2	3	4	5
Time at Start of Intervals	Interval Duration (hr)	Avg. Containment Temp. During Interval (°F)	H_2 Prod. Rate (ft^3/hr, Enclosure 6)	H_2 Produced = 2 x 4
Accident Starts				
0100	25/60	300	7000	2920
0125	20/60	260	2400	800
0145	15/60	250	1800	450
0200	1.0	240	1400	1400
0300	1.0	220	700	700

Sampling Time

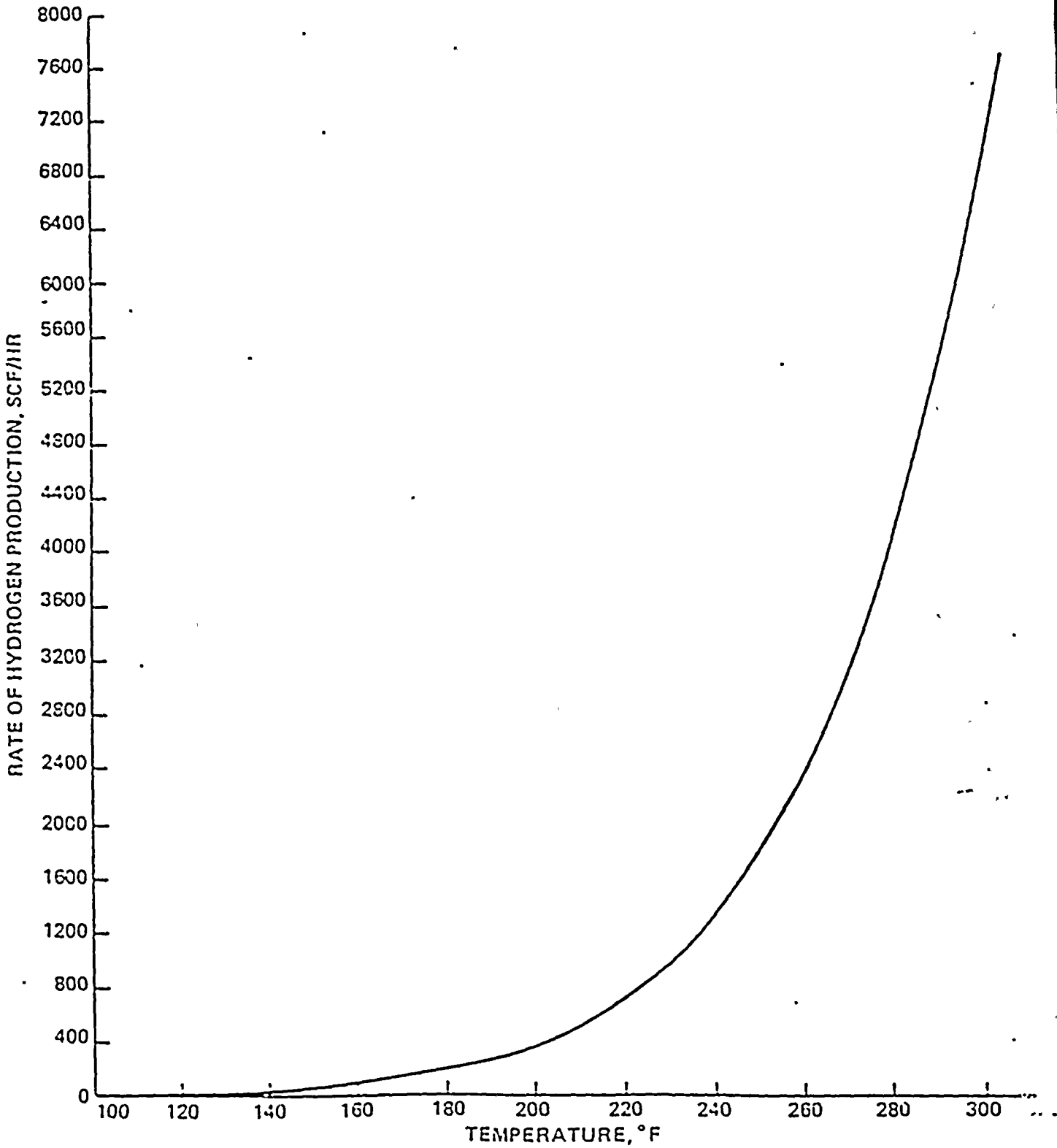
Long Term Hydrogen Production in Containment, Total = 6300

Short term rapid hydrogen production by containment aluminum . (Table 4-3, Section 4.5) 5300

Total Hydrogen Production in Containment 11,600 SCF

Record total on Enclosure 9 also.

FIGURE 4-28
HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE FOR ST. LUCIE UNIT 2



ENCLOSURE 7

HYDROGEN GENERATED BY RADIOLYSIS

Step 7.4 Record the following data and utilize the curves of Enclosure 8 to determine the hydrogen generated by radiolysis.

Prior 30 day power history	Power, Percent	Duration, Days
	75	22
	50	17
	100	2

Power to use in evaluating long term hydrogen production by radiolysis = (2560) x 0.5 .

Reactor Trip Time 0100 hrs

Sampling Time (see Enclosure 3) 0400 hrs

Decay Time (Sampling Time - Trip Time) 3 hrs

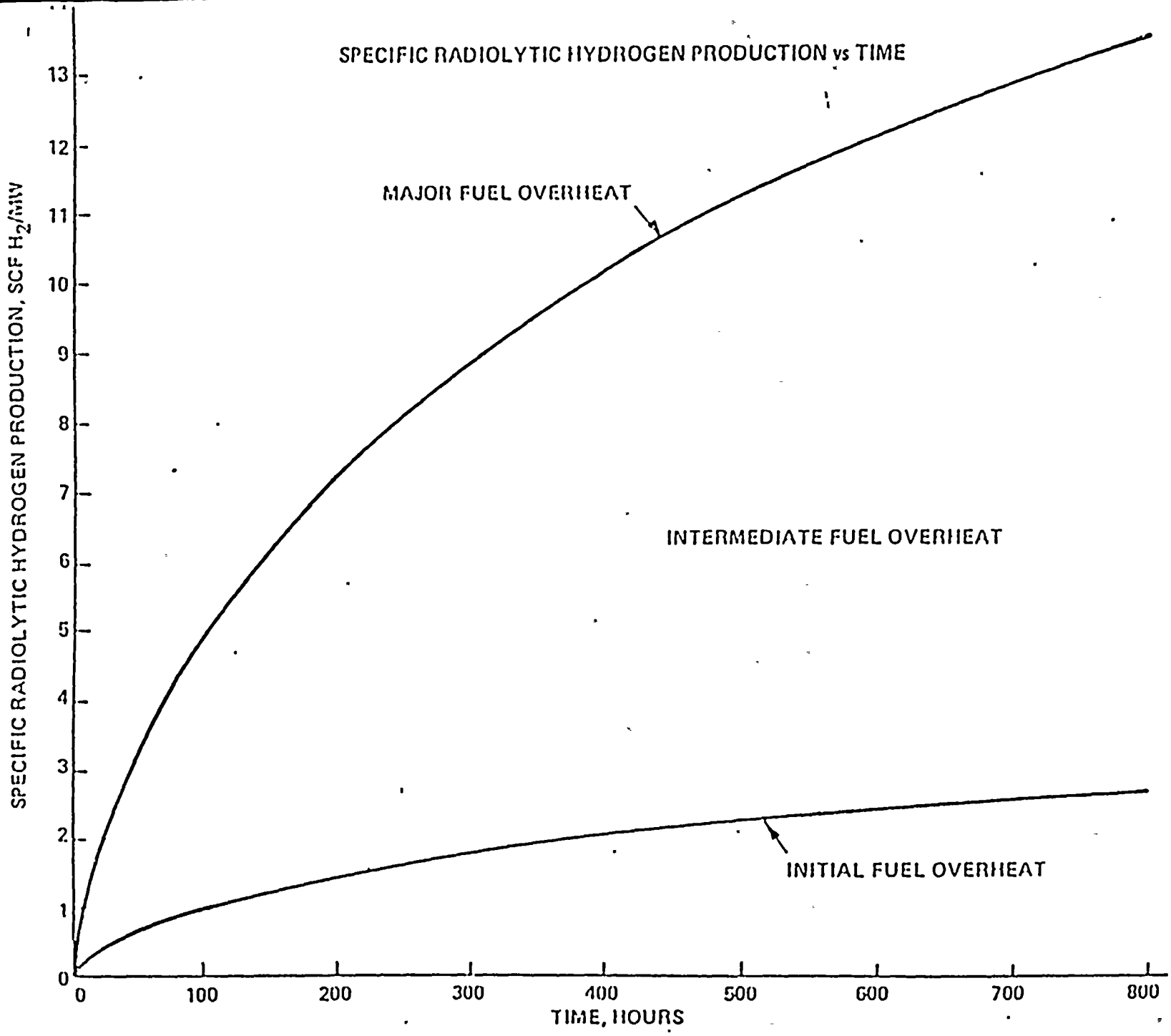
Enter abscissa on Enclosure 8 with above decay time and read two values of hydrogen produced by radiolysis, one from each curve, in cubic feet of hydrogen at STP per Mwt operating power. Multiply by above power and record as follows:

Limit Curve	Hydrogen Produced (SCF/Mwt, Enclosure 8)	x	Operating Power	=	Total Hydrogen Produced (SCF)
Upper	0.4		1280		512
Lower	0.1		1280		128

Using results from Radiological Damage Assessment Procedure estimate which results should be used; upper limit for major fuel overheat, lower limit for initial fuel overheat or appropriate estimate between the two curves for intermediate fuel overheat. Circle corresponding value of hydrogen above and also record on Enclosure 9.

EXAMPLE

Enclosure 8



ENCLOSURE 9

CORE DAMAGE ASSESSMENT FROM

HYDROGEN MEASUREMENT

Step 7.5 Hydrogen Measured, Step 7.2, Enclosure 3 ___20,800___ SCF

Hydrogen Produced in Containment, Step 7.3, Enclosure 5 ___11,600___ SCF

Hydrogen Produced by Radiolysis, Step 7.4, Enclosure 7, ___130___ SCF

Subtract Step 7.3 and 7.4 from 7.2 to Get Hydrogen Produced by Core Clad Oxidation ___9,100___ SCF

Divide by (4650* SCF/1% Clad Oxidized) = ___2___ %

= % Core Oxidized

Step 7.6 Enter abscissa on Enclosure 10 with "% Core Oxidized" and read ordinate from curve labeled with pressure during core uncovery as given on Enclosure 2, Step 7.1.2. Record here Percent of Fuel Rods with Ruptured Clad ___~100___ %.

Step 7.7 Enter abscissa on Enclosure 11 with above "% Core Oxidized" and read range of values on ordinate. Record here

Percent of fuel rods embrittled

Range - Upper ___22___ %

- Lower ___6___ %

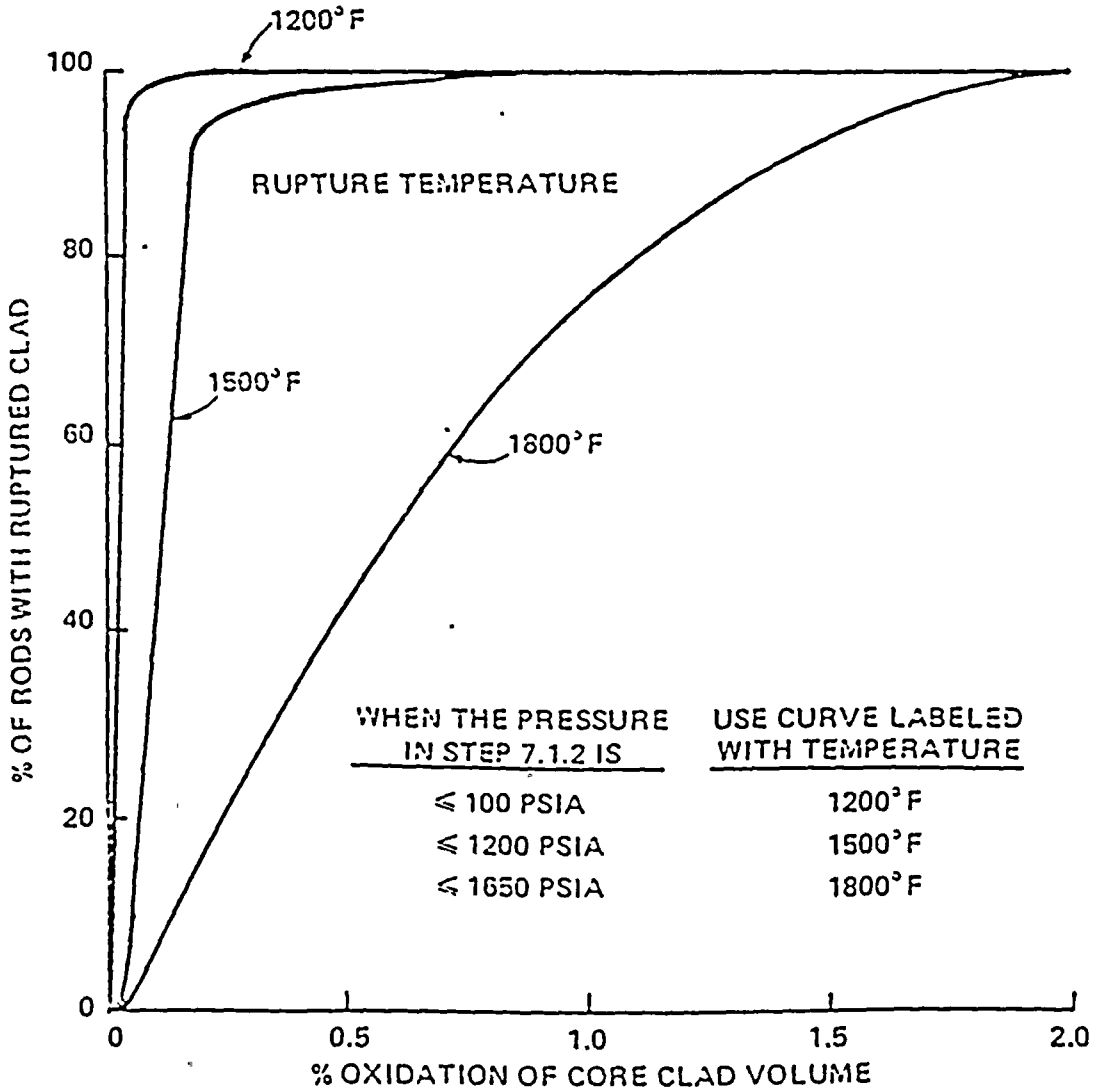
Step 7.8 Review Step 7.1 and Bases sections to determine which of these limits is more likely to be representative of the core damage.

Step 7.9 From Enclosure 1 select the core clad damage categories based on the above percentages of rods ruptured and rods embrittled.

The assessment yields category 4, Major Clad Failures with category 5, Initial Fuel Pellet Overheating. Because of slow core uncovery and moderate pressure, lower limit of % rods embrittled is selected.

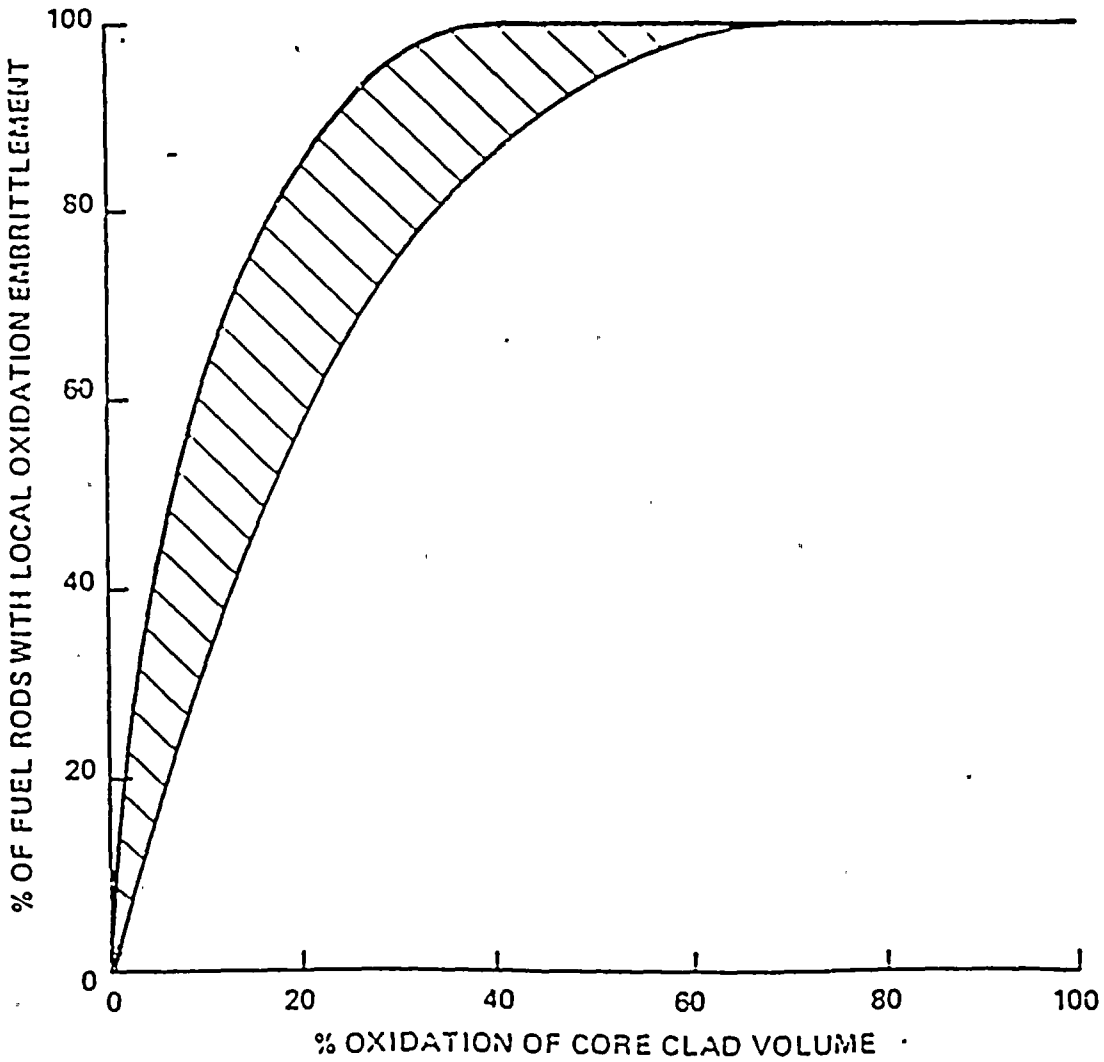
* From Column 1, Table 4-2 of Section 4.5.

ENCLOSURE 10
PERCENT OF FUEL RODS WITH RUPTURED
CLAD vs CORE CLAD OXIDATION



WHEN THE PRESSURE IN STEP 7.1.2 IS	USE CURVE LABELED WITH TEMPERATURE
≤ 100 PSIA	1200°F
≤ 1200 PSIA	1500°F
≤ 1650 PSIA	1800°F

ENCLOSURE 11
% OF THE FUEL RODS WITH OXIDATION EMBRITTLEMENT vs
TOTAL CORE OXIDATION
FOR 1% TO 3% DECAY HEAT AND 300 PSIA TO 2500 PSIA
WHEN COOLANT LEVEL DROPS BY BOILOFF WITH
NO INLET FLOW UNTIL CORE IS RAPIDLY QUENCHED



APPENDIX E.0

PROCEDURE GUIDELINE FOR ASSESSMENT OF CORE DAMAGE USING CORE EXIT THERMOCOUPLES

TABLE OF CONTENTS

LIST OF ENCLOSURES

1.0 PURPOSE

This procedure is to be followed under post accident plant conditions to determine the number of fuel rods with ruptured clad. It provides and estimate of damage up to about the time when the peak core temperature reaches about 2300°F. At that time most of the rods probably have ruptured clad but little other structural degradation has occurred. Therefore this procedure applies to the relatively less severe accidents although it may be used for other accidents to confirm that damage exceeds this minimum amount. The resulting observation of core damage is described by categories 1 through 4 of the ten NRC categories in Enclosure 1.

2.0 REFERENCES

2.1 Development of the Comprehensive Procedure Guidelines for Core Damage Assessment, C-E Owners Group Task 467, May, 1983.

2.2 (Appropriate plant specific document which describes capabilities and operation of Inadequate Core Cooling Instrumentation including Core Exit Thermocouples.)

2.3 Generic Thermal-Hydraulic Functional Design Objectives for Inadequate Core Cooling Instrumentation, CE-NPSD-199, prepared for the C-E Owners Group.

3.0 DEFINITIONS

3.1 Clad Rupture: Clad rupture is defined as a break in the fuel rod clad at least sufficient to release the internal gas pressure. Rupture may be preceded by ballooning of the clad if the internal gas pressure exceeds the external coolant pressure during an accident, and the temperature is higher than normal.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the 10 categories of core damage described in Enclosure 1. The procedure is based on core exit temperature data. Other plant indications may be available which can improve upon estimation of core damage. These include radiologic sample characteristics, the total quantity of hydrogen released from zirconium degradation and containment radiation monitors. Whenever possible these additional indicators should be factored into the assessment.

4.2 The assessment of damage provided by this procedure extends up to the time of clad rupture on most of the fuel rods. This time occurs early in very severe core uncovery accidents. More severe core damage cannot be quantified by this procedure.

4.3 The relationship between the core exit thermocouple temperature and the clad temperature varies with the core uncovery scenario. This procedure applies to slow core uncovery by boiloff of the coolant. For other more rapid uncovery scenarios this procedure could yield a very low estimate of the number of ruptured rods. In general, for core uncovery at pressures below about 1200 psia there is high confidence that at least the predicted estimate of rods are actually ruptured.

5.0 INITIAL PLANT CONDITIONS AND SYMPTOMS

This procedure is to be employed for analysis of core exit thermocouple data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. This list is not a complete representation of all events which may cause core damage. One or more of these symptoms will exist at or before the time the core exit thermocouple recorded temperature is utilized to estimate damage.

5.1 High alarm on the containment radiation monitor.

5.2 High alarm on the CVCS letdown radiation monitor.

5.3 High alarm on the main condenser air ejector exhaust radiation monitor.

5.4 Pressurizer level low.

5.5 Safety Injection System may have automatically actuated.

5.6 Possible high quench tank level, temperature, or pressure.

5.8 Possible noise indicative of a high energy line break.

5.9 Decrease in volume control tank level.

5.10 Standby charging pumps energized.

5.11 Unbalanced charging and letdown flow.

5.12 Reactor Coolant System subcooling low or zero.

6.0 PREREQUISITES

An operational inadequate core cooling instrumentation system which includes core exit thermocouples and which can select and permanently record the highest thermocouple temperature for convenient, later inspection. A system which satisfies the requirements for core exit thermocouples in Reference 2.3 is adequate.

7.0 PROCEDURE

7.1 Obtain the following from the instrument recordings:

7.1.1 From the recording of maximum core exit thermocouple temperature as a function of time, obtain and record on Enclosure 2 the maximum temperature and the time it occurs.

7.1.2 From the recording of reactor coolant system pressure as a function of time, obtain and record on Enclosure 2 the pressure during the period of maximum thermocouple temperature.

7.2 Select the curve on Enclosure 3 which is labeled with a pressure approximately equal to or greater than the pressure in Step 7.1.2. Enter the abscissa at the maximum temperature from Step 7.1.1 and read on the ordinate the percent of the fuel rods which have ruptured clad. Record on Enclosure 2.

7.3 The result in 7.2 is probably a lower limit estimate of damage. Some judgement on the bias is available as follows.

7.3.1 This procedure applies most directly for relatively slow core uncovery with a maximum temperature below the rapid oxidation temperatures at about 1800°F and above. A smooth core exit thermocouple recording and an uncovery duration of 20 minutes or longer are indicators for a good prediction of clad ruptures.

7.3.2 If the pressure in 7.1.2 drops to less than about 100 psia within less than about two minutes of accident initiation, a large break is indicated. This causes undetected core heatup followed by flashing during refill. Depending on the rate of refill, the thermocouple temperature may rise rapidly then quench when the core is recovered. This procedure could yield a very low estimate for the percent of rods ruptured.

7.3.3 If the pressure in Step 7.1.2 is above about 1650 psia, it could exceed the rod internal gas pressure depending on rod burnup, causing clad collapse onto the fuel pellet instead of outward clad ballooning. The clad rupture criteria are less well defined for such conditions, but at temperatures above 1200°F where the highest pressure curve applies on Enclosure 3, clad failure sufficient to release fission gas is likely and this procedure may be used to obtain estimates of damage.

7.4 CORE DAMAGE ASSESSMENT

Use the percent of rods ruptured from Step 7.2 and the clad damage characteristics of Enclosure 1 to determine the NRC category of cladding failure. This procedure yields damage estimates in categories 2, 3 or 4.

ENCLOSURE 1

CLAD DAMAGE CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

	NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1.	No Fuel Damage	~750	None	--	--	Less Than 1
2.	Initial Cladding Failure	1200-1800	Rupture Due to Gas Gap	Maximum Core Exit Thermocouple Temperature	<1550°F*	Less Than 10
3.	Intermediate Cladding Failure		Overpressurization		<1700°F*	10 to 50
4.	Major Cladding Failure				≲2300°F ≲2% Oxidation	Greater Than 50
5.	Initial Fuel Pellet Overheating	1800-3350	Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to % Oxidation of Core)	Equivalent Core Oxidation <3%	Less Than 10
6.	Intermediate Fuel Pellet Overheating				<18%	
7.	Major Fuel Pellet Overheating				≲65%	Greater Than 50

*Depends on Reactor Pressure and Fuel Burnup. Value given for Pressure ≤1200 psia and Burnup ≥0.

ENCLOSURE 2

RECORD OF TEMPERATURE PRESSURE AND

DAMAGE ESTIMATE

Step 7.1 Record the following data

Maximum Core Exit Thermocouple Temperature _____°F

Time of Maximum Temperature _____

Reactor Coolant System Pressure at Above Time _____psia

Step 7.2 From Enclosure 3, at maximum thermocouple temperature and at appropriate pressure

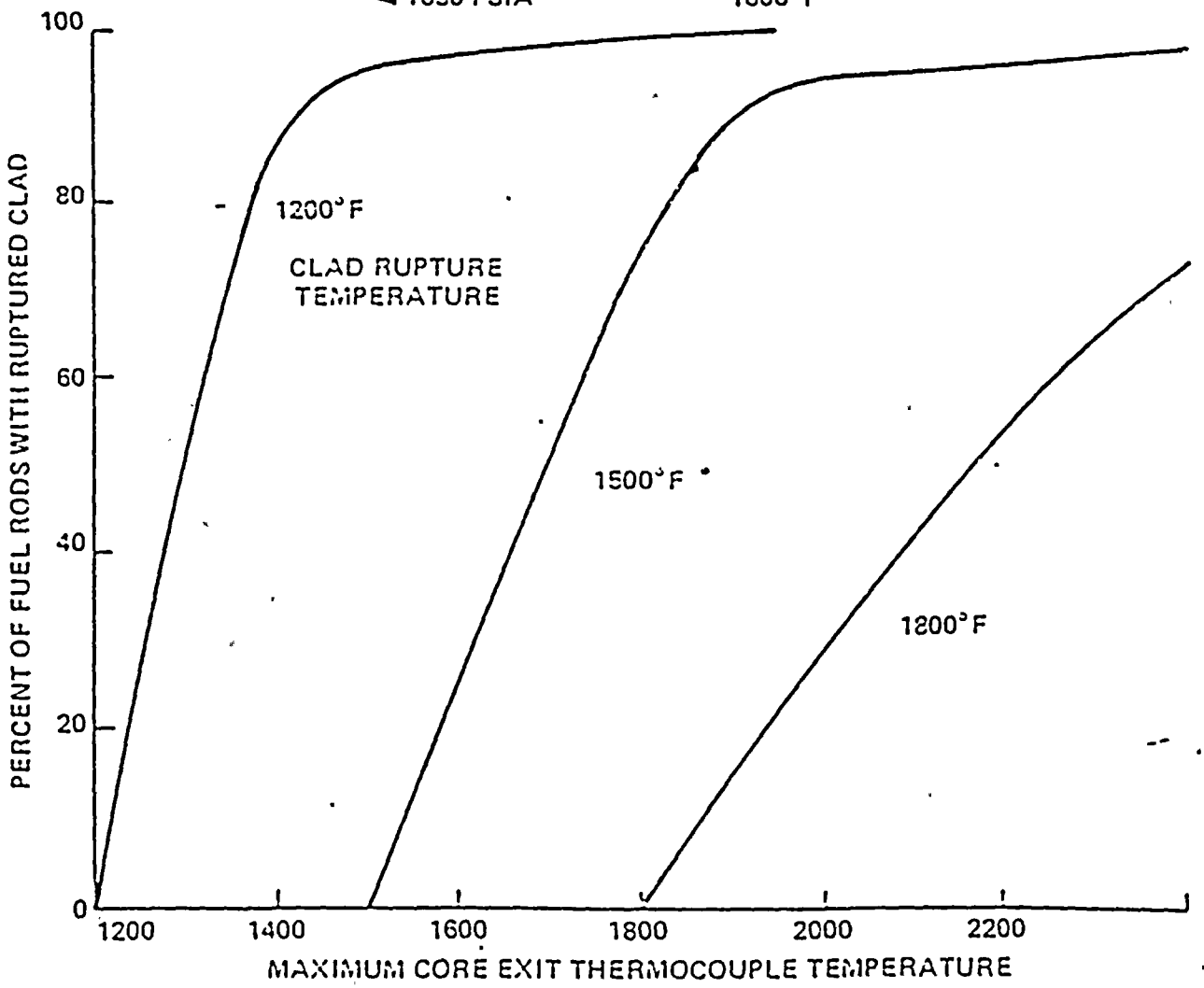
read percent of ruptured rods. _____%

Step 7.3 Comment on probable bias of result in 7.2 (see paragraph 7.3 in text).

Step 7.4 NRC category of cladding failure from Enclosure 1 _____

ENCLOSURE 3
PERCENT OF FUEL RODS WITH RUPTURED CLAD vs
MAXIMUM CORE EXIT THERMOCOUPLE TEMPERATURE

WHEN THE PRESSURE IN STEP 7.1.2 IS	USE CURVE LABELED WITH TEMPERATURE
≤ 100 PSIA	1200° F
≤ 1200 PSIA	1500° F
≤ 1650 PSIA	1800° F



APPENDIX E.1

EXAMPLE USE OF THE PROCEDURE

The following is an example of the use of the procedure for assessment of clad damage by utilizing the maximum temperature recorded from the core exit thermocouples. The specific case cited is for the 2560 Mwt class of reactor.

Quantities in the example are selected for an accident equivalent to that used in the example for core damage assessment based on radiological characteristics and given in Appendix C.1. This example consists of a completed worksheet given in the procedure.

ENCLOSURE 2

RECORD OF TEMPERATURE PRESSURE AND

DAMAGE ESTIMATE

Step 7.1 Record the following data

Maximum Core Exit Thermocouple Temperature	2000 °F
Time of Maximum Temperature	0230 on 12/25/82
Reactor Coolant System Pressure at Above Time	900 psia

Step 7.2 From Enclosure 3, at maximum thermocouple temperature and at appropriate pressure

read percent of ruptured rods. 95 %

Step 7.3 Comment on probable bias of result in 7.2 (see paragraph 7.3 in text). Coolant pressure <1100 psia for curve, so estimate is low. Uncovery period long compared to CET delay time so temperature represents steam closely. Rod temperature is actually higher than CET temperature, but the estimate would not change much with a 300° increase. Conclude that prediction is a good estimate of actual clad failures.

Step 7.4 NRC category of cladding failure from Enclosure 1, Category 4, Major clad failure.

APPENDIX F.0

PROCEDURE GUIDELINE FOR ASSESSMENT OF CORE DAMAGE USING RADIATION DOSE RATES

TABLE OF CONTENTS

LIST OF ENCLOSURES

1.0 PURPOSE

This procedure is to be followed under post accident plant conditions to determine the type and degree of core damage which may have occurred by using radiation dose rates measured inside the containment building using the wide range radiation monitor. The radiation dose rate is related to the quantitative release of fission products from the core expressed as the percent of the source inventory at the time of the accident. The resulting observation of core damage is described by one or more of the ten categories of core damage in Enclosure 1.

2.0 REFERENCES

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, C-E Owners Group Task 467, May 1983.

2.2 Wide Range Containment Radiation Dose Rate Monitor Operating Procedures (Plant Specific Document).

3.0 DEFINITIONS

3.1 Fuel Damage: For the purpose of this procedure fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the reactor coolant starting with a penetration in the zircaloy cladding. The type of fuel damage as determined by this procedure is reported in terms of four major categories which are: no damage, cladding failure, fuel overheat, and fuel melt. The first three categories are characterized by the resulting radiation dose rate inside the containment building. The degree of core damage is determined by making a comparison between dose rates measured following an accident and analytically determined values of the realistic or best estimate dose rates that would correspond to the specific categories of core damage. The degree of core damage as determined by this procedure is reported in terms of three levels which are: initial; intermediate; and major. This results in a total of ten possible categories as characterized in Enclosure 1.

3.2 Source Inventory: The source inventory is the total quantity of fission products expressed in curies of each isotope present in either source; the fuel pellets or the fuel rod gas gap.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 The assessment of core damage obtained by using this procedure is only an estimate. The techniques employed in this procedure are only accurate to locate the core condition within one or more of the 10 categories of core damage described in Enclosure 1. The procedure is based on radiation dose rate. Other plant indications may be available which can improve upon the estimation of core damage. These include sample radiological analysis, incore

temperature indicators, and the total quantity of hydrogen released from zirconium degradation. Whenever possible these additional indicators should be factored into the assessment.

4.2 This procedure relies upon radiation dose rate measurements taken from one or more monitors located inside the containment building to determine the total quantity of fission products released from the core and therefore available for release to the environment. The amount of fission products present at the location of the monitors may be changing rapidly due to transient plant conditions. Therefore, multiple measurements should be obtained within a minimum time period and when possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

4.3 A number of factors influence the reliability of the measured radiation dose rates upon which this procedure is based. Reliability is influenced by the ability to obtain representative measurements due to incomplete mixing of the measured media, equipment limitations, and lack of operator familiarity with rarely used procedures. Additionally the procedure relies upon analytically determined values of the best estimate dose rates that are anticipated to correspond to the specific categories of core damage. These analytical values are based upon assumptions made about the identity and relative proportions of the fission products released from the core and their transport within the containment building. Therefore the procedure is only accurate to within the validity of the assumptions.

4.4 This procedure is limited to the upper bound condition of fission product release from the core due to fuel overheat. Simultaneous with fuel overheat, there may be localized fuel pellet melting within the core. The transport of the non-volatile fission products released due to melting is not known. The dose rates measured under conditions of fuel pellet melting are anticipated to exceed those, shown in Enclosure 2, for major fuel overheat. However, this procedure does not attempt to identify the extent of any potential fuel melting.

4.5 This procedure is limited to the interpretation of the dose rate measurement resulting from a mix of fission products. The procedure cannot accurately distinguish between the conditions of fuel cladding failure and fuel overheat when the resulting dose rates are the same. The procedure does provide an upper limit estimate of the progressive core damage. Concurrent conditions of cladding failure and overheat should be anticipated due to the radial distribution of heat generation within the core. Distinction between the type of core damage requires the identification of the characteristic fission products. The procedure for core damage assessment using radiological analysis of fluid samples is required to explicitly distinguish between the categories.

4.6 This procedure is limited in applicability to those conditions in which the fission product inventory in the core has had sufficient time to reach equilibrium. Equilibrium fission product inventory is a function of reactor power and burnup. Based upon the fission products of concern equilibrium conditions are achieved after thirty days of operation at constant power. Constant power is considered to include changes of no greater than ±10 percent. The procedure may be used following non-constant periods of operation by using engineering judgement to select the most representative power level during the period. The procedure may also be used if the reactor has produced power for less than thirty days, however, the resulting assessment of core damage would be an underprediction of the actual conditions.

5.0 INITIAL PLANT CONDITIONS AND SYMPTOMS

This procedure is to be employed for analysis of dose rate data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. This list is not a complete representation of all events which may cause core damage. One ore more of these symptoms may exist at or before the time the sample is obtained. Under these conditions, sampling should be performed using the Post Accident Sampling System.

5.1 High alarm on the containment radiation monitor.

5.2 High alarm on the CVCS letdown radiation monitor.

5.3 High alarm on the main condenser air ejector exhaust radiation monitor.

5.4 Pressurizer level low.

5.5 Safety Injection System may have automatically actuated.

5.6 Possible high quench tank level, temperature, or pressure.

5.8 Possible noise indicative of a high energy line break.

5.9 Decrease in volume control tank level.

5.10 Standby charging pumps energized.

5.11 Unbalanced charging and letdown flow.

5.12 Reactor Coolant System subcooling low or zero.

6.0 PREREQUISITES

An operational Wide Range Radiation Dose Rate Monitor System with the capability to measure the area dose rates inside the containment

building resulting from fission products dispursed in the building atmosphere and plated out on building surfaces. The system should meet the requirements of Regulatory Guide 1.97.

7.0 PROCEDURE

7.1 Record the following plant indications.

7.1.1 Containment Building:

Radiation Dose Rate _____ Rads/hr.

Time of Measurement Date _____ Time _____

7.1.2 Prior 30 days power history:

Power, Percent	Duration, Days
_____	_____
_____	_____
_____	_____
_____	_____
_____	_____

7.1.3 Time of reactor shutdown Date _____ Time _____

7.2 Plant Power Correction

The measured radiation dose rate inside the containment building is to be corrected for the plant power history. A correction factor is used to adjust the measured dose rate to the corresponding value had the plant been operating at 100 percent power.

7.2.1 To correct the radiation dose rate for the case in which plant power level has remained constant for a period greater than 30 days a simple ratio of the power may be employed. The reactor power is considered to be constant if it has not changed by ±10 percent within the last thirty days prior to the reactor shutdown.

7.2.2 To correct the radiation dose rate for the case in which reactor power level has not remained constant during the 30 days prior to the reactor shutdown engineering, judgement is used to determine the most representative power level. The following guidelines should be considered in the determination.

7.2.2.1 The average power during the 30 day time period is not necessarily the most representative value for correction to equilibrium conditions.

7.2.2.2 The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

7.2.2.3 Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

7.2.3 In the case in which reactor has produced power for less than 30 days the procedure may be employed. However, the estimate of core damage obtained under this condition may be an under prediction of the actual condition.

7.2.4 The following equation is applied to determine the radiation dose rate corresponding to equilibrium full power source inventory conditions.

$$\text{Equilibrium Dose Rate} = \text{Measured Dose Rate} \times \frac{100}{\text{Reactor Power Level}}$$

The reactor power level and the resulting dose rate correction factor used above will be the same for all subsequent measurements of the dose rate. Record these values to reduce the work required to evaluate the subsequent measurements.

7.3 The decay correction for the radiation dose rate requires the determination of the time duration between the reactor shutdown and the measurement of the dose rate. This is done simply using the time of reactor shutdown recorded in Section 7.1.3.

7.4 The conclusion on the extent of core damage is made using the equilibrium dose rate, the duration of reactor shutdown, and the analytically determined dose rates provided in Enclosure 2. The equilibrium dose rate is plotted on Enclosure 2 as a function of time following reactor shutdown. Engineering judgement is used to determine which category of core damage shown on Enclosure 2 is most representative of the particular value that has been plotted. The following criteria should be considered in the determination.

7.4.1 Dose rate measurements may have been recorded during periods of transient conditions within the plant. Measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.

7.4.2 Dose rates significantly above the lower bound for the category of major fuel overheat may indicate concurrent fuel pellet melting. This procedure may not be employed to estimate the degree of fuel pellet melting.

7.4.3 Dose rates within any category of fuel overheating may be anticipated to include concurrent fuel cladding failure. This procedure may not be used to distinguish the relative contributions of the two categories to the total dose rate. The procedure does give the estimate of the highest category of damage.

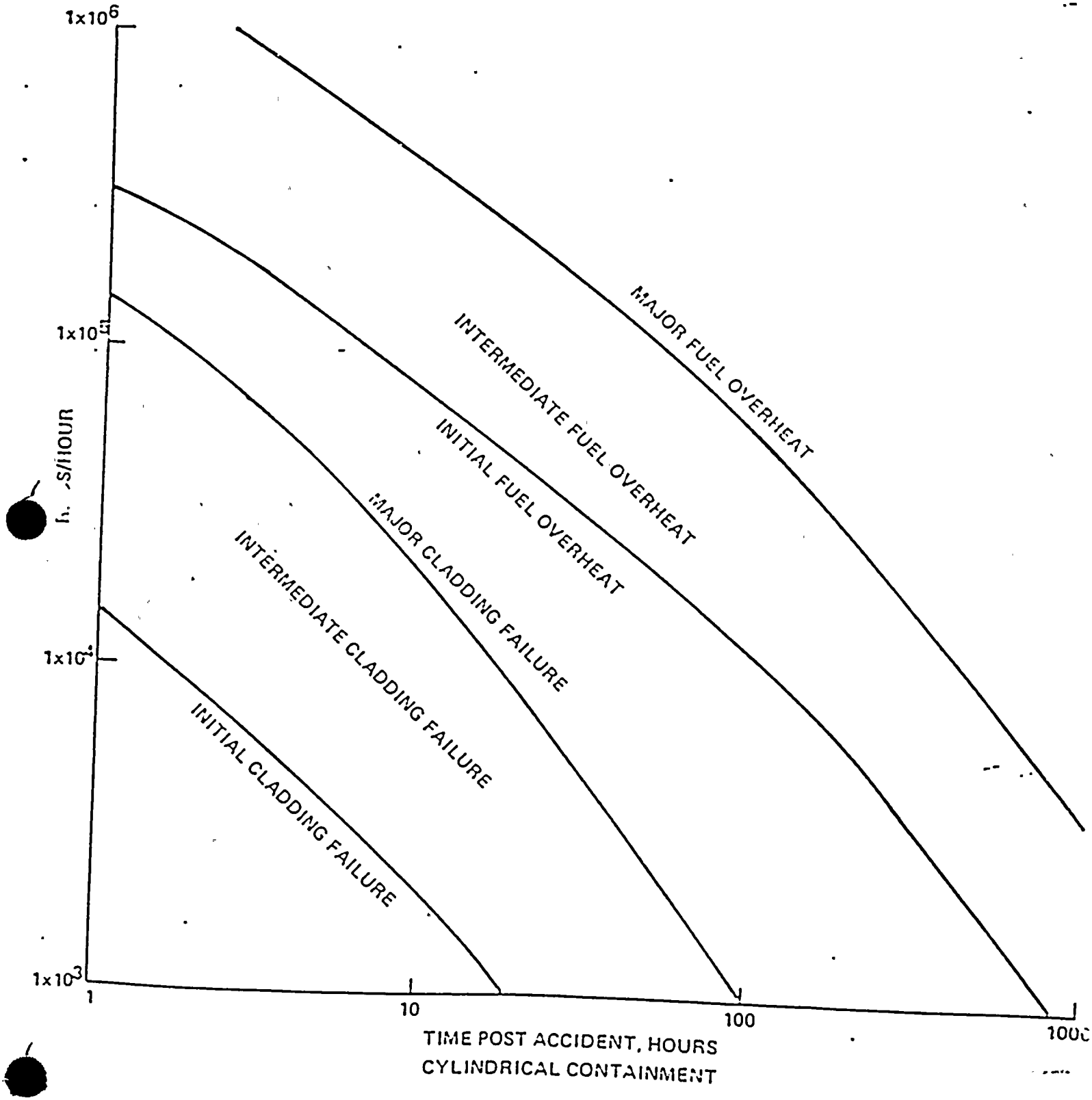
7.4.4 Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Enclosure 2. The evaluation of other plant parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

Enclosure 1

Radiologic Characteristics of NRC Categories of Fuel Damage

	NRC Category of Fuel Damage	Mechanism of Release From Core	Source of Release	Percent of Source Inventory Released to Containment	Distribution of Fission Products in Containment
1.	No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	Less than 1	Airborne
2.	Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Less than 10	Airborne
3.	Intermediate Cladding Failure		Gas Gap	10 to 50	Airborne
4.	Major Cladding Failure		Gas Gap	Greater than 50	Airborne
5.	Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Less than 10	Airborne: 100% Noble Gas 25% Halogen Plated Out 25% Halogen 1% Solids
6.	Intermediate Fuel Pellet Overheating		Fuel Pellet	10 to 50	
7.	Major Fuel Pellet Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet	Greater than 50	

ENCLOSURE 2
TYPICAL ANALYSIS FOR POST ACCIDENT
DOSE RATE INSIDE A CYLINDRICAL CONTAINMENT



APPENDIX F.1

EXAMPLE USE OF THE PROCEDURE

The following is an example of the use of this procedure for assessment of core damage. The data recorded on plant condition is as follows:

Containment Building:

Measurement Number 1 — Dose Rate = 1×10^5 Rads/hr
Time = 0300 on 12/25/82

Measurement Number 2 — Dose Rate = 5×10^4 Rads/hr
Time = 0600 on 12/25/82

Measurement Number 3 — Dose Rate = 1.5×10^4 Rads/hr
Time = 0100 on 12/26/82

Measurement Number 4 — Dose Rate = 4×10^3 Rads/hr
Time = 0100 on 12/31/82

Prior 30 day power history:

Power, Percent	Duration, Days
75	22
50	17
100	2

Time of reactor shutdown: 0100 on 12/25/82

Step 7.2 As shown in the recorded data, the reactor power has not remained constant for the 30 days prior to the accident. Therefore engineering judgement is used to determine the power level employed in the assessment of core damage. The criteria stated in Step 7.2.2.2 and 7.2.2.3 are used in the determination. The value selected is 50 percent. This value is selected because during the 17 days at that level many fission products reach equilibrium inventory in the core. During the 2 day time period at 100 percent power, the inventory of all fission products increased. The short lived fission products may even have increased to their equilibrium corresponding to 100 percent power. However, most of the isotopes with longer half lives, those greater than 1 day, remain at inventories closer to that corresponding to equilibrium at 50 percent power. Also using 50 percent power would somewhat underestimate the fission product source inventory and the resulting core damage assessment would therefore be conservative.

Step 7.2.4 Using 50 percent power, the full power equilibrium dose rate for measurement number 1 is as follows:

$$\text{Equilibrium Dose Rate} = 1 \times 10^5 \times \frac{100}{50} = 2 \times 10^5 \text{ Rad/hr}$$

the remaining full power equilibrium dose rates are:

Measurement Number 2 = 1×10^5 Rad/hr

Measurement Number 3 = 3×10^4 Rad/hr

Measurement Number 4 = 8×10^3 Rad/hr

Step 7.3 The time duration between reactor shutdown and the measurement of the dose rate for each case is:

Measurement Number 1 = 2 hours

Measurement Number 2 = 5 hours

Measurement Number 3 = 24 hours

Measurement Number 4 = 144 hours

Step 7.4 Assessment of Core Damage

The equilibrium full power dose rates and the duration of reactor shutdown are plotted on the following copy of Enclosure 2.

The conclusion is that core damage is in the category of Initial Fuel Overheat. Because of the extent of this damage, concurrent fuel cladding failure is anticipated although it is not expressly distinguished from the total damage by this procedure.

ENCLOSURE 2
TYPICAL ANALYSIS FOR POST ACCIDENT
DOSE RATE INSIDE A CYLINDRICAL CONTAINMENT