ATTACHMENT 1

INTERIM CORE DAMAGE PROCEDURE

THE INTERIM

PROCEDURE GUIDELINE FOR CORE DAMAGE ASSESSMENT

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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 the first phase of effort on this task.

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1.0 INTRODUCTION

1.1 BACKGROUND

The NRC instituted the NUREG-0737 (Reference 4.1) 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 4.2) to NUREG-0737 providing guidance for preparation of a procedure to assess core damage, Appendix 1. As stated in this clarification, none of the near term operating license applicant 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), is implementing a two-phase program to provide procedure guidelines for assessing core damage following severe accidents. This report is the product of the first phase. It is 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 procedure guidelines for utilizing additional data from the PASS and also from other commonly available instrumentation. The additional PASS 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 will be a 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 this 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 at 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 4.3.

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 4.4, 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 4.3. 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 4.5. 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.

TABLE 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	2000-2550
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

Above 3250°F the Zircaloy cladding melts. 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 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.

Therefore the combination of Tables 1 and 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

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

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

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 less 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 lives. 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 4.3, and the Rogovin Report, Reference 4.4. 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 bound 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) begin reactions between the solid UO_2 and solid metallic zircaloy, melting of the control rods materials, and melting of the zirionium. 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 4

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 4 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 1800°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 5 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 5.

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 this would indicated 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 6. These values were calculated using the ORIGEN computer code, Reference 4.6, 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 7. There 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.

TABLE 6

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	4.8(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 7

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 2.

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 8 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 8
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 8 (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. However, this is sufficient accuracy to allow plant operators to make informed decisions under post accident plant conditions.

4.0 REFERENCES

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

4.2 Post Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage, US NRC. (Included here as Appendix 1)

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

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

4.5 CEN-158-P Evaluation of Instrumentation for Detection of Inadequate Core Cooling in C-E NSSS. May 1981.

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

APPENDIX I

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 applicants 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 applicatn'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 available plant indicators. The following discussion is intended to provide general guidance 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. some 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.

3. What are the ratios of various radionuclide species (i.e. the gap activity ratio for various radionuclides may differ from the ratio in the pellet). The measurement of a specific ratio will then indicate whether the activity released came from the gap or 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 indicatros show fuel overheat and the radionuclide 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 pretain 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 condition, 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 indicators 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 judgment 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 mst 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 actuality, 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 availbe 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 histroy, 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 concentrationt 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 (to). 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 damge based on predicted radionuclide concentrations. Methodology fo this step is provided by letter of May 4, 1981, from McGuire Nuclear Station, Docket No. 50-369.

APPENDIX 2

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.

$$\frac{dN_\ell}{dt} = (F)(Y_\ell)(P) + (f_{\ell-1}\,\lambda_{\ell-1})\,N_{\ell-1} + \sigma_k\,\phi_k\,N_{\ell-1} - (\lambda_\ell + \nu_\ell + \sigma_\ell\phi)\,N_\ell$$

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

ℓ-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.

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

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.

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

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

APPENDIX 3

INTERIM PROCEDURE

FOR CORE DAMAGE ASSESSMENT

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 Interim Procedure Guidelines for Core Damage Assessment, C-E Owners Group Task 467, January 1982.

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 produce 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 multiplied by 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 $$\text{Power Correction Factor} = \frac{100}{\text{Steady State Power Level For Prior 30 Days}}$$

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

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.

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

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

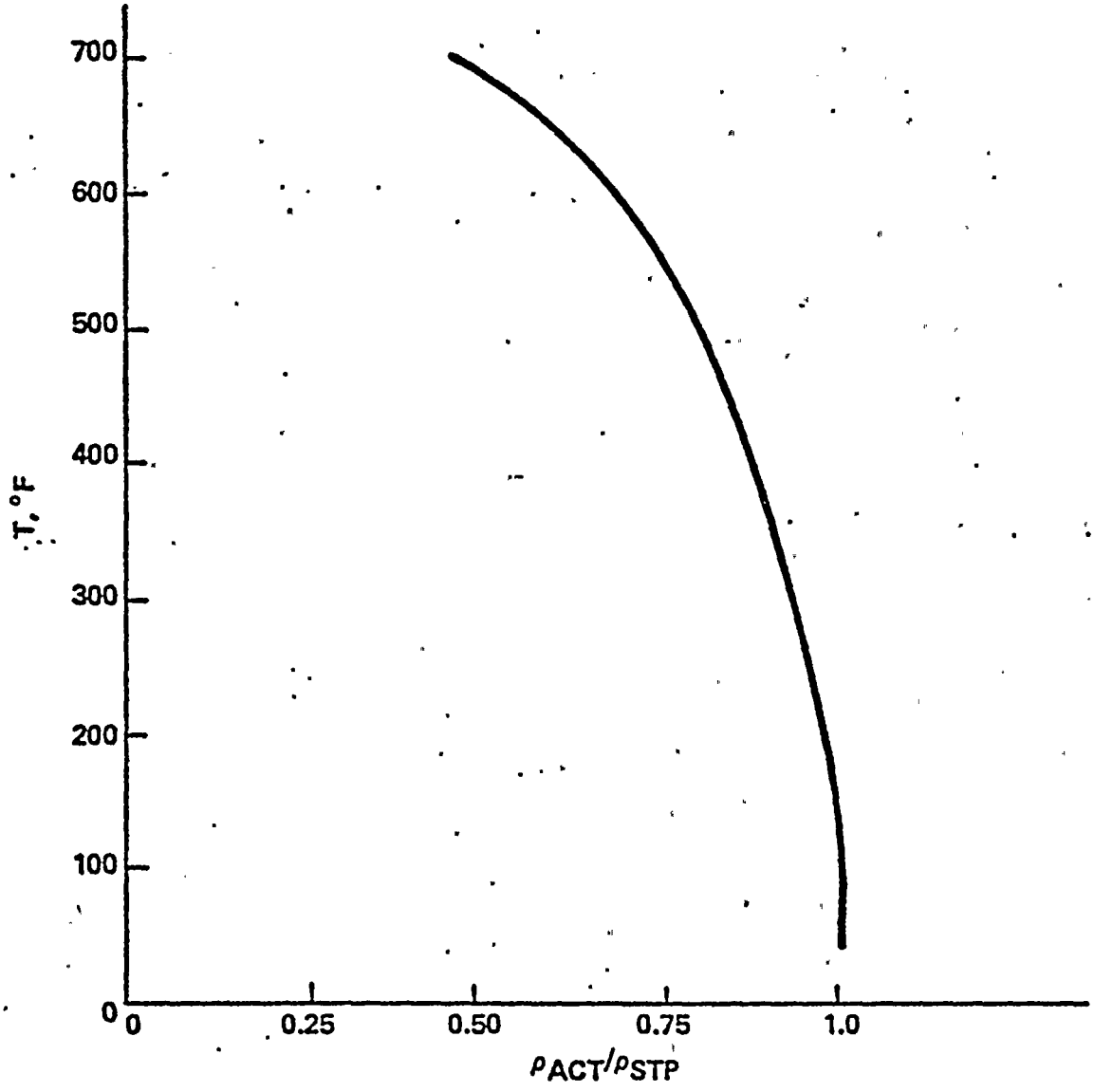
La 140

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 6

RECORD OF DECAY CORRECTION

Time of Reactor Shutdown, Step 7.1.4:

Sample Number:

Location:

Time of Analysis:

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)		
I 131	9.9 (-7)		
I 132	8.4 (-5)		
I 133	9.3 (-6)		
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)		

ENCLOSURE 7

RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

Sample Number:

Location:

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			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	

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

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

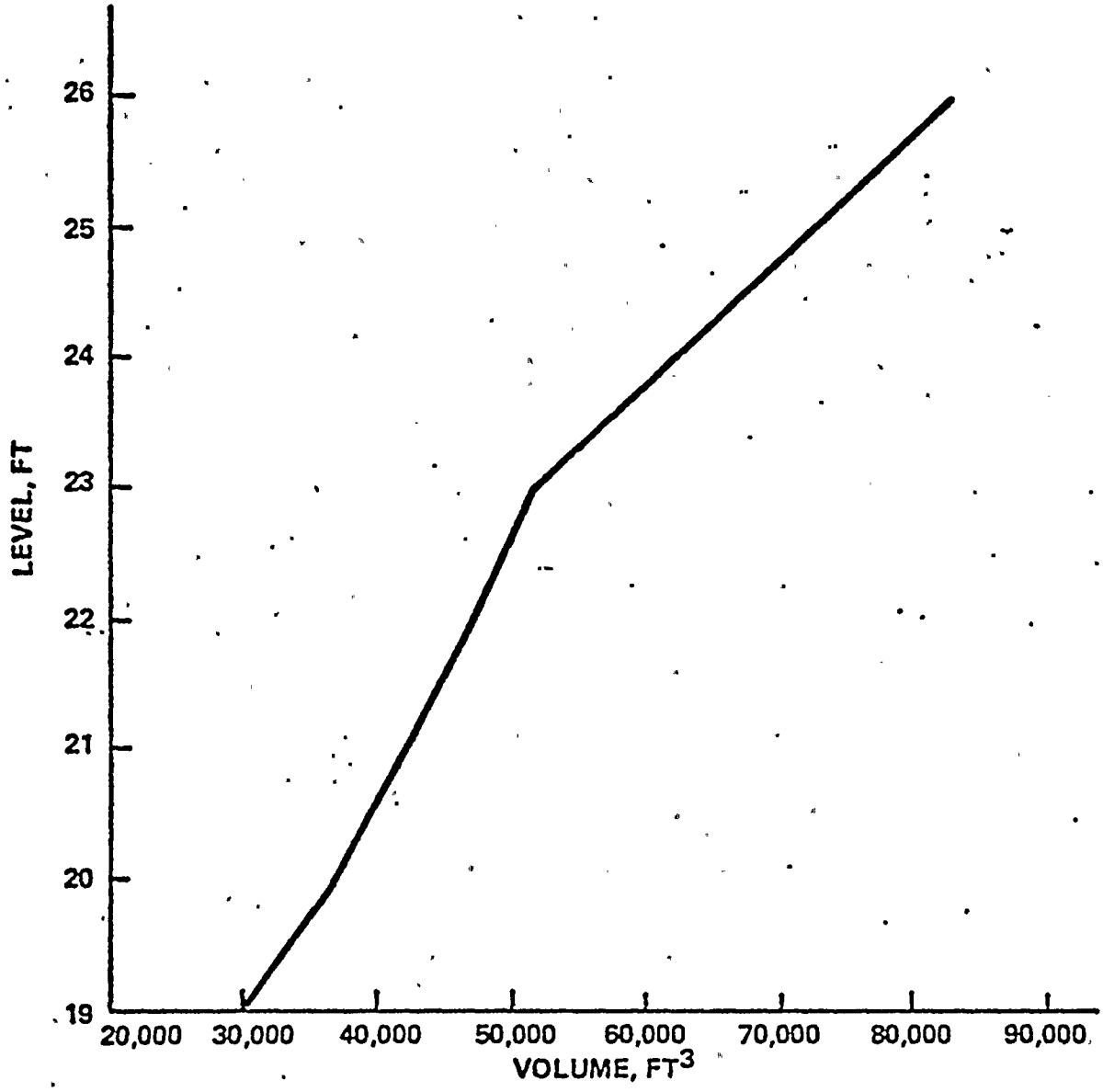
ENCLOSURE 8

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				
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	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)	

ENCLOSURE 11

RECORD OF TRANSIENT POWER CORRECTION

Sample Number:
Location:
Prior 30 Day Power History:

Power %	Duration, Days

Isotope	Power Correction Factor	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)	

ENCLOSURE 12

RECORD OF PERCENT RELEASE

Isotope	Total Quantity Available For Release (Enclosure 8), Ci	Power Corrected Source Inventory, 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 4

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	

EXAMPLE

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 0900

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)	1/0.87	1.1(+2)
I 131	1(+4)	1/0.87	1.1(+4)
I 132			
I 133	1(+2)	1/0.87	1.1(+2)
I 135			
Cs 134			
Rb 88			
Te 129	1(+3)	1/0.87	1.1(+3)
Te 132			
Sr 89			
Ba 140			
La 140			
La 142	1(+1)	1/0.87	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 0900

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)	1.2(+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 = $\frac{\text{Decay Corrected Iodine Isotope Specific Activity}}{\text{Decay Corrected I-131 Specific Activity}}$

EXAMPLE

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

EXAMPLE

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

EXAMPLE

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)	3.4(+3)	2.8(+4)
I 131	2.5(+6)	1.1(+5)	3.4(+3)	2.6(+6)
I 132	--	--	--	--
I 133	2.8(+4)	1.1(+3)	3.9(+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 $\frac{19.7 + .5}{19.7}\frac{49C}{690}$ = 2.6 x 10^{10} cc]

Xe_{133}: 1.3(-1) x 2.6(10) x 1(-6) = 3.4(+3)

I_{131}: 1.3(-1) x 2.6(10) x 1(-6) = 3.4(+3)

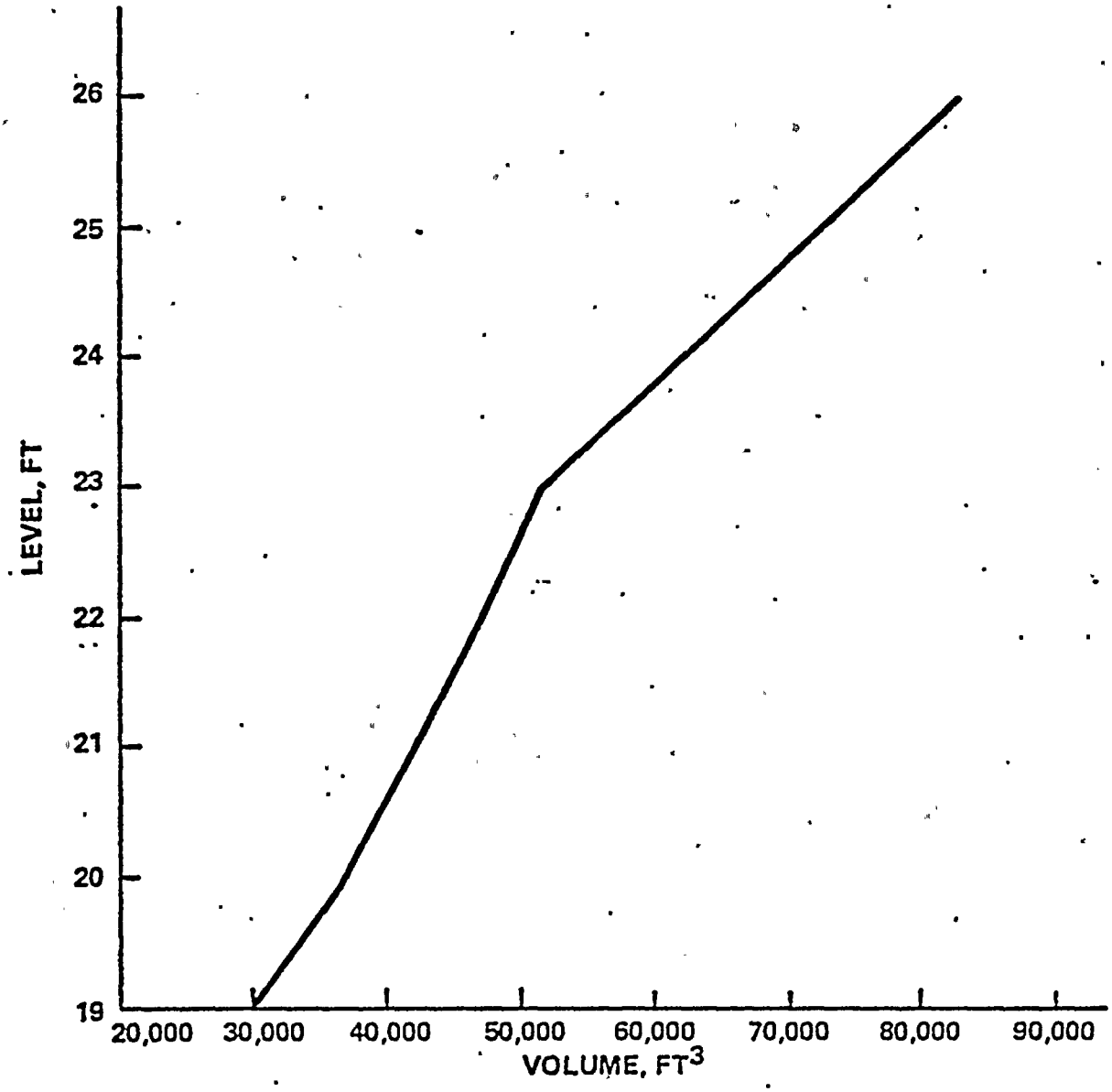
I_{133}: 1.5(-3) x 2.6(10) x 1(-6) = 3.9(+1)

These values are recorded on Enclosure 8.

EXAMPLE

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	Equilibrium Source Inventory	Corrected Source Inventory
Gas Gap Inventory				
Kr 87				
Xe 131m	1	--	--	--
Xe 133	1	1.6	1.3(7)	2.1(7)
I 131	1	1.6	6.7(6)	1:1(7)
I 132	2	--	--	--
I 133	2	1.1	6.7(6)	7.4(6)
I 135	2	--	--	--
Fuel Pellet Inventory				
Kr 87				
Xe 131m	1	--	--	--
Xe 133	1	1.6	1.5(8)	2.4(8)
I 131	1	1.6	7.3(7)	1.2(8)
I 132	2	--	--	--
I 133	2	1.1	1.5(8)	1.7(8)
I 135	2	--	--	--
Cs 134	1	--	--	--
Rb 88	2	--	--	--
Te 129	2	1.0	2.4(7)	2.4(7)
Te 132	1	--	--	--
Sr 89	1	--	--	--
Ba 140	1	--	--	--
La 140	1	--	--	--
La 142	2	1.0	1.6(8)	1.6(8)
Pr 144	2	--	--	--

POWER CORRECTION FACTORS

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

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

$$\frac{100}{\left\{75(1-e^{-[1.5(-6)][1.9(+6)]})(e^{-[1.5(-6)][1.6(+6)]}) + 50(1-e^{-[1.5(-6)][1.5(+6)]})(e^{-[1.5(-6)][1.7(+5)]}) + 100(1-e^{-[1.5(-6)][1.7(+5)]})(e^{o})\right\}} = \frac{100}{63.6} = 1.6$$

The remaining isotopes are calculated in the same manners; 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)	Percent
Gas Gap Inventory			
Kr 87			
Xe 131	--	--	--
Xe 133	2.8(+4)	2.1(7)	.13
I 131	2.6(+6)	1.1(7)	24
I 132	--	--	--
I 133	2.9(+4)	7.4(6)	.39
I 135	--	--	--
Fuel Pellet Inventory			
Kr 87			
Xe 131m	--	--	--
Xe 133	2.8(+4)	2.4(8)	.01
I 131	2.6(+6)	1.2(8)	2.2
I 132	--	--	--
I 133	2.9(+4)	1.7(8)	.02
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)	.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) 24 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 Intermediate Fuel Cladding Failure with concurrent Initial Fuel Pellet Overheating.