

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 applicant's slow responses to this item is their confusion on exactly what to prepare. The attachment is intended to provide informal guidance to each NTOL applicant so that their procedures, when prepared, will address the core damage estimation in a manner acceptable to us.

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

The primary purpose in preparing a procedure for relating radionuclide concentrations to core damage is to be able to provide a realistic estimate of core damage. We are primarily interested in being able to differentiate between four major fuel conditions; no damage, cladding failures, fuel overheating and core melt. Estimates of core damage should be as realistic as possible. If a core actually has one percent cladding failures, we do not want a prediction of fifty percent core melt or vice versa; extremes in either direction could significantly alter the actions taken to recover from an accident. Therefore, the procedure for estimating core damage should include not only the measurement of specific radionuclides but a weighted assessment of their meaning based on all 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 indicators 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 pertain to meeting normal off-site doses as a consequence of fuel failures.

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

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

#### Example

In a given accident 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 must be associated with the amount of fuel pellet melting which is indicated. Therefore, the best estimate of fuel damage would be that "intermediate fuel overheating had occurred, with major cladding damage and the possibility of minor fuel pellet melting in one or two fuel bundles out of 150 fuel bundles."

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

- A. The radionuclide/s monitored are at equal concentrations in all fuel rods.

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

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

Considering that ideal conditions will not exist, then procedure for estimating core damage should be prepared in a manner that the effects of variables such as time in core life and type of accident are accounted for. Therefore, the procedure for estimating core damage should include the determination of both short and long lived gaseous and non-volatile radionuclides along with ratios for appropriate species. Each separate radionuclide analyzed, along with predicted ratios of selected radionuclides would be used to estimate core damage. This process will result in four separate estimates of core damage, (short and long-lived, gaseous and non-volatile species) which can be weighed, based on power history, to determine the best estimate of core damage.



The post-accident sampling system locations for liquid and gaseous samples varies for each plant. To obtain the most accurate assessment of core damage, it is necessary to sample and analyze radionuclides from each of these locations (reactor coolant, containment atmosphere, containment sumps and suppression pool), then relate the measured concentrations 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 chamber vapor space would contain the majority of gaseous activity. In the 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 indications which can be used to verify that the sample locations selected are best for the specified accident condition.

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

## Part 2, Vol II - Rogovin Report

- at 2 hours 54 minutes, although additional slumping occurred at 3 hours 45 minutes.
7. All of the fuel rods in the core burst, during an approximately 30-minute (center bundle) to 40-minute (lowest power peripheral bundles) period after the top of the core was uncovered at depths ranging from 1½ feet (center bundle) to 2 feet (peripheral bundle) from the top of the fuel rods.
  8. Temperatures at which liquefied fuel ( $\text{UO}_2$  dissolved in the zirconium metal-zirconium dioxide liquid eutectic at about 3500 to 3600°F) could be formed were calculated to have first been reached at 6 inches from the top of the fuel in the fuel rods in the central fuel bundle about 33 minutes after the top of the core was uncovered and were reached as low as 36 inches from the top of the fuel. Such temperatures were calculated to have been reached in the peripheral bundles at a depth of about 14 inches from the top of the fuel in about 46 minutes after the core was uncovered and at a depth of about 41 inches in 57 minutes.
  9. The peak temperatures calculated for the fuel rods ranged from 4370°F in about 52 minutes for the highest powered bundle to a maximum of 4412°F for a medium powered bundle at 58 minutes to about 4358°F for a lower powered peripheral bundle at about 78 minutes.
  10. The amount of hydrogen formed by oxidation of solid Zircaloy cladding during the temperature excursion was calculated to be about 308 pounds, and that formed from all of the damaged Zircaloy, including that contained in the liquefied fuel present at 3 hours, was calculated to be about 720 pounds. This is the minimum amount of hydrogen estimated to have been formed. The maximum could be as high as 820 pounds.
  11. The major releases of hydrogen to the containment occurred before 4 hours accident time and during the long depressurization around 8 hours. No significant amount of hydrogen was produced after about 4 hours.
  12. The minimum water level occurring in the core up to 3 hours is estimated to have been  $4 \pm \frac{1}{2}$  ft from the bottom of the fuel in the fuel rods on the basis of the amount of hydrogen produced, the amount of radioactivity released, the time at which significant levels of radioactivity were detected, and the structural damage estimated in the core.
  13. The total amount of Zircaloy oxidized is calculated to be not less than 16 400 pounds and may have been as high as 18 700 pounds; i.e., between about 10% and 20% of the total Zircaloy in the core.
  14. The damage in the core extends from the top downward at least 7 feet, and probably 8 feet, over most of the core and consists of oxygen embrittled Zircaloy cladding topped by a bed of debris that probably consists of fuel pellet fragments, partially dissolved fuel pellets, shells of Zircaloy oxide, and segments of embrittled Zircaloy cladding with outer skins of Zircaloy oxide, all glued together with liquefied fuel into a relatively tight and compact mass extending entirely across the core from wall to wall and penetrated by only a few vertical passageways, at most. In addition, fingers of liquefied fuel extend downward from the debris bed in several continuous subchannels between fuel rods, encompassing the neighboring fuel rods, to a depth of about 1 foot above the bottom of the fuel stack in the fuel rods. Not less than 32% of the fuel assemblies have such fingers of liquefied fuel.
- ### c. Core Damage Estimates from Fission Product Release
- At shutdown the reactor core contained fission products, activation products, and actinides. Some of these, notably krypton and xenon, are gaseous and can diffuse through the fuel pellet to collect in the gap between the fuel and the cladding. To a lesser extent, the halogens (iodine and bromine) can also diffuse into the fuel-clad gap. Any perforation of the cladding can release these fission products into the reactor coolant.
- If the fuel temperatures are higher than operating temperatures, but well below melting, other radioactive materials are volatilized and can diffuse out. Also, diffusion of the noble gases and halogens increases so that a larger fraction of these can be released. The release of cesium is quite variable and could be caused by compound formation. Because of this variability and what is now known about cesium, it is not possible to determine precisely the temperature at which a reasonably large fraction of the cesium would be released; however, it is believed temperatures would not be lower than 1300°C (2370°F).<sup>187,188</sup>
- At higher temperatures that cause the liquefaction or melting of fuel, some fraction of other fission products such as tellurium can be released. Data reported show that the escape of tellurium depends on many factors other than temperature.<sup>189</sup> Under oxidizing conditions some ruthenium may be

released before melting. In general, rather large fractions of both tellurium and ruthenium are released in melting; but under some conditions, these materials can also be released before melt. The presence of ruthenium and tellurium does not prove that melt has occurred, but the absence of them is a good indicator that melt has not occurred. More recent experimental work,<sup>187,190</sup> while tending to confirm previous data, has not resolved all the questions regarding conditions—especially temperature conditions—under which fission products would be released.

Many of the fission products and most of the actinides occur as refractory oxides and are released only in relatively small amounts even at elevated temperatures. However, if damaged fuel pellets are rewetted, some of the more refractory radioactive material can be leached out. This process is slow and only small fractions of these materials find their way into the coolant by leaching. The longer damaged fuel is in contact with water, the more materials are released.

#### Categories of Fission Product Releases and Their Relation to TMI-2

Fission products and actinides can be divided into typical release groups, based on the ease with which they are volatilized. One such grouping (from Ref. 191) is in order of decreasing volatility.

- I Noble gases (Kr, Xe)
- II Halogens (I, Br)
- III Alkali metals (Cs, Rb)
- IV Tellurium (Te)
- V Alkaline earths (Sr, Ba)
- VI Noble metals (Ru, Rh, Pd, Mo, Tc)
- VII Rare earths and actinides
- VIII Refractory oxides of Zr and Nb

The fraction of gaseous and volatile fission products released depends on the temperature and the size of the fuel fragments. If the temperature is high or if the fuel is highly fragmented, nearly complete release of the volatile materials can be assumed.

Under the conditions that have been calculated for the accident at TMI-2,<sup>188</sup> nearly complete release of groups I and II can be assumed from all fuel that was severely damaged, plus some additional fraction from fuel rods whose cladding was perforated without damage to the fuel. This additional amount from perforated but otherwise undamaged rods is probably partly balanced by the amount not released from severely damaged fuel.

A major fraction of group III and a much smaller fraction of group IV could have been released from

the most severely damaged fuel. Small fractions, approximately 10% or less, could have been released from perforated but otherwise undamaged rods, but this cannot be well estimated.

#### Leaching from Irradiated Fuel

Very small fractions of the remaining groups may have been released from the very hottest fuel. The principal mechanism for release of these refractory materials is probably leaching. Leaching from irradiated  $UO_2$  has not been thoroughly studied. However, the work of Katayama<sup>192,193</sup> and of Forsyth and Eklund<sup>194</sup> has shown that the leaching rates are slow, comparable to those from glass. Quantitative data, especially for the temperatures and conditions existing in TMI-2, are too sparse for a reliable calculation of the rate of leaching, especially when one considers that the condition of the damaged fuel is completely unknown.

An additional complication is presented because the effective surface area of irradiated fuel presented to the water is almost impossible to estimate because of cracking and porosity. The most that can be done with the available data is to form an "educated guess" as to whether the fuel appears to be mainly in the form of very large pieces or in the form of very fine fragments. Without additional data it is not possible to estimate the actual size distribution of the fragments. However, a small fraction of the most refractory material can be expected to have found its way into the reactor coolant. An approximate leaching calculation is presented in Appendix II.7. On the basis of this approximate calculation, it is possible to state, with very low confidence, that a large fraction of the fuel can presently be fragmented and that the size of the fragments is more likely to be a few millimeters than dustlike. A similar calculation has been carried out by Powers.<sup>191</sup> His conclusions, although not identical with these, indicate that the observed activity may have been caused either by leaching from large-sized fragments or by distribution of particle sizes no more than a few percent smaller than 2 millimeters in diameter and none smaller than 0.6 millimeter in diameter.

#### Expected Dispersion of the Fission Products from the Reactor

Principal fuel damage probably started before 3 hours after turbine trip. There was probably only minor damage before 2 hours. The calculated total inventory<sup>195</sup> of fission products, activation products,



TABLE II-56. Activity in release groups\*

Group	Activity
I	$2.97 \times 10^8$ Ci
II	$4.47 \times 10^8$ Ci
III	$4.6 \times 10^7$ Ci
IV	$1.61 \times 10^8$ Ci
V	$3.85 \times 10^8$ Ci
VI	$6.34 \times 10^8$ Ci
VII	$2.69 \times 10^9$ Ci
VIII	$4.80 \times 10^8$ Ci
Total	$5.11 \times 10^9$ Ci**

\*A few elements of low total activity, notably Fe, Cu, As, and Sb, have been arbitrarily located on the basis of melting point.

\*\*Total does not quite agree with calculated total activity because of rounding.

and actinides is given in Table II-56 for 3 hours after shutdown.

A detailed discussion of the fission product-release pathways begins in Section II.B of this report where a short summary is included. Radioactive material released to the reactor coolant may have been partially flushed to the containment through the open PORV (RC-R2). Some of the material may have been flushed to the containment prior to the containment isolation and then pumped to the auxiliary building. However, the coolant may have contained only a minute fraction of the total activity at this time; it is highly improbable that a significant fraction of the coolant was released before the reactor building sump pumps were shutdown. There is an unsubstantiated possibility<sup>196</sup> that more water leaked to the auxiliary building after pump shutdown. This leakage would have terminated when the reactor building was isolated after 3 hours 56 minutes.

Most of the material flushed out of the RCS probably remained in the reactor building. Some additional material may have volatilized from the makeup tank. Aside from these losses, which are not expected to be very large, estimates of the total activity released from the fuel can be made by analyzing the reactor building air and water samples, the reactor coolant, and the auxiliary building tanks.

Iodine is quite volatile, and it may be supposed that a significant fraction is found in the air. Howev-

er, the very high solubility of iodine in water and the strong tendency of atmospheric iodine to plate out on surface quickly reduces the amount of iodine in the air. Cesium, less volatile, is not expected to be present in the air in a significant quantity. On the other hand, the solubility of xenon and krypton is very low; these gases will be found almost entirely in the air.

To summarize, nearly complete release of noble gases, iodine, and cesium from damaged fuel is expected, even if the temperature is below the melting point. Significant releases of tellurium, ruthenium, and more refractory materials will occur only if the temperature approaches the melting point. Most of the noble gases will be found in air, and most of the other fission products will be found in water.

#### Distribution of Fission Products at the TMI Site

Analyses of samples of containment air, reactor coolant water, and auxiliary building tank water are summarized in Ref. 197. Reactor coolant analyses show between 7% and 15% of the calculated inventory of iodine and cesium isotopes to be in the coolant. If these measurements are corrected for dilution by water from the borated water storage tank, the fractions will be a factor of 3 higher. Results for refractory materials show great variation. A sample taken on April 10 was analyzed by four laboratories. There was a large variation from laboratory to laboratory, indicating low confidence in the results. Analyses of krypton and xenon isotopes in the containment atmosphere also showed considerable variation. However, based on the most abundant isotopes (<sup>85</sup>Kr and <sup>133</sup>Xe), there seemed to be 29% to 62% of the core inventory of noble gases in the containment air. Only 2% to 3% of the iodine and cesium was found in the auxiliary building tanks.

On August 28, 1979, a hole was drilled into the reactor building and samples of sump water were removed. Analyses of these samples showed 22% to 48% of the core inventory of iodine and cesium to be in the reactor building sump water.<sup>198</sup> In addition to iodine and cesium, very small amounts of Ru, Zr, Nb, Sb, La, and Ag were found. As expected, little <sup>90</sup>Sr was found. At most, the amounts corresponded to a few millionths of the core inventory. About 0.02% of the core inventory of <sup>129m</sup>Te was found.

All of these sample analyses were corrected for decay of the radionuclides to the time of analysis. This correction process is certainly more accurate than the analyses themselves; i.e., the accuracy of the estimates does not depend on the accuracy of



the debris distribution. Table II-57 is a reconstitution of the release of volatiles.

#### Findings

From these results, one can cautiously conclude that between 40% and 60% of the core inventory of release groups I-III was released to the coolant; that only a small fraction of group IV was released; and that only minute amounts of the remaining groups were released. The amount of refractory isotopes released is consistent with leaching (see Appendix II.7).

These data tend to confirm other analyses of core damage. The data on radioactivity released are too sparse and variable for a precise conclusion to be made on the amount of core damage; however, the following conclusions appear to be supported.

1. About 50% of the reactor core was damaged sufficiently to release the most volatile fission products.
2. The low fractions of tellurium, ruthenium, and strontium indicate that no significant quantity of fuel reached the melting point of  $\text{UO}_2$  (5200°F).
3. The amount of refractory isotopes in the reactor coolant is consistent with leaching.

#### d. Hydrogen Production, Removal, and Hazard

##### Introduction

One of the surprises of TMI-2 was the formation of large amounts of hydrogen from the reaction of

the cladding material with the steam generated by the boiling water. In this section several aspects of the hydrogen "problem" are discussed. The following subjects are treated in this section:

1. hydrogen production,
2. hydrogen accounting,
3. calculation of bubble size,
4. removal of the hydrogen bubble, and
5. the hazard from the hydrogen bubble.

##### Hydrogen Production

Two possible sources of hydrogen are considered: metal-water reactions and radiolysis. Other conceivable sources include oxidation of  $\text{UO}_2$  which has not been investigated. The production of hydrogen from metal-water reactions is known to have been large; therefore any hydrogen from other mechanisms is expected to be small in comparison. Radiolysis is not expected to produce large amounts of hydrogen. It is investigated because the possibility of oxygen production was considered at the time of the accident. If oxygen had been released, the hydrogen that was trapped in the reactor coolant system could have become flammable.

##### Metal-Water Reaction

Many metals are oxidized by water. The reaction is very slow at low temperatures for most metals. Both steel and zirconium are oxidized at an increasing rate as the temperature rises. The oxidation of zirconium, the major constituent of the cladding, oc-

TABLE II-57. Total volatile isotopes released from core

Released To	Isotope (fraction of core inventory)			
	$^{133}\text{Xe}$	$^{131}\text{I}$	$^{137}\text{Cs}$	$^{134}\text{Cs}$
Environment	0.01 <sup>1</sup>	— <sup>2</sup>	—	—
RB Atmosphere	0.46 <sup>3</sup>	—	—	—
RB Water	—	0.22 <sup>4</sup>	0.48 <sup>4</sup>	0.34 <sup>4</sup>
RC Water	—	0.14 <sup>4</sup>	0.12 <sup>4</sup>	0.08 <sup>4</sup>
Aux. Bldg. Tanks	—	0.03	0.03	0.02
Totals	0.46	0.39	0.63	0.44

<sup>1</sup>See Ref. 199

<sup>2</sup>Dashes indicate low values (generally less than 1%)

<sup>3</sup>Best estimate from data in Ref. 197.

<sup>4</sup>Average of observations.

Handwritten notes:  $^{35}\text{K}$ ,  $^{140}\text{Ba}$ ,  $^{138}\text{Kr}$

# DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.  
VICE PRESIDENT  
STEAM PRODUCTION

May 4, 1981

TELEPHONE: AREA 704  
373-4083

Mr. Harold R. Denton, Director  
Office of Nuclear Reactor Regulation  
U.S. Nuclear Regulatory Commission  
Washington, D. C. 20555

Attention: Ms. E. Adensam, Chief  
Licensing Branch No. 4

Re: McGuire Nuclear Station  
Docket Nos. 50-369, 50-370



Dear Mr. Denton:

Attached are ten copies of McGuire procedure, AP/O/A/5500, "Estimate of Failed Fuel Based on I-131 Concentration". This is one of the implementing procedures for the McGuire Emergency Plan and as such should be included with the other implementing procedures previously submitted on February 13, 1981.

By copy of this letter, three copies of this implementing procedure are being provided to NRC, Region II.

Very truly yours,

*William O. Parker, Jr.*  
William O. Parker, Jr.

GAC:pw  
Attachment

cc: M. J. Graham (w/o attach.)  
Resident Inspector  
McGuire Nuclear Station

Mr. J. P. O'Reilly, Director (w/3 cys.)  
U.S. Nuclear Regulatory Commission  
Region II  
101 Marietta Street, Suite 3100  
Atlanta, Georgia 30303

METHODOLOGY LOOKS GOOD  
BUT SHOULD BE APPLIED TO  
THE ISOTOPES IN THE ROGMAN  
REPORT (INCLUDING GASES) PLUS  
A FEW MORE WITH GOOD YIELD  
AND ABUNDANCE.

Boo!  
3  
1/1

F 1063080114

DUKE POWER COMPANY  
PROCEDURE PREPARATION  
PROCESS RECORD

(1) ID NO.: AP/0/A/5506  
Change(s) 0  
0 Incorporat

FOR INFORMATION ONLY

STATION: McGuire

PROCEDURE TITLE: Estimate of Failed Fuel Based on I-131 Concentration

PREPARED BY: Michael S. Kitten Jr DATE: 3/19/81

REVIEWED BY: Larry N. A. Smith DATE: 4-2-81

Cross-Disciplinary Review By: M. Sample SRC # 81-032 N/R:

TEMPORARY APPROVAL (IF NECESSARY)

BY:  (SRO) DATE:

BY:  DATE:

APPROVED BY: W. W. Felt DATE: 4/16/81

MISCELLANEOUS:

Reviewed/Approved By: Scott W. Brown DATE: 3/19/81

Reviewed/Approved By:  DATE:

8105080118



DUKE POWER COMPANY  
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) STATION: McGuire UNIT: 1 ☒ INTERVIEW ONLY  
OTHER: FSAR

(2) CHECK LIST APPLICABLE TO: AP/3/A/5500/33

## (3) SAFETY EVALUATION - PART A

The item to which this evaluation is applicable represents:

Yes ☐ No ☒ A change to the station or procedures as described in the FSAR or a test or experiment not described in the FSAR?

If the answer to the above is "Yes", attach a detailed description of the item being evaluated and an identification of the affected section(s) of the FSAR.

## (4) SAFETY EVALUATION - PART B

Yes ☐ No ☒ Will this item require a change to the station Technical Specifications?

If the answer to the above is "Yes," identify the specification(s) affected and/or attach the applicable pages(s) with the change(s) indicated.

## (5) SAFETY EVALUATION - PART C

As a result of the item to which this evaluation is applicable:

Yes ☐ No ☒ Will the probability of an accident previously evaluated in the FSAR be increased?

Yes ☐ No ☒ Will the consequences of an accident previously evaluated in the FSAR be increased?

Yes ☐ No ☒ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

Yes ☐ No ☒ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ☐ No ☒ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

Yes ☐ No ☒ May the possibility of malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

Yes ☐ No ☒ Will the margin of safety as defined in the bases to any Technical Specification be reduced?

If the answer to any of the preceding is "Yes", an unreviewed safety question is involved. Justify the conclusion that an unreviewed safety question is or is not involved. Attach additional pages as necessary.

(6) PREPARED BY: Michael S. Kline DATE: 4/10/81

(7) REVIEWED BY: McGuire SRC DATE: 4/10/81

DUKE POWER COMPANY  
McGUIRE NUCLEAR STATION  
ESTIMATE OF FAILED FUEL BASED ON I-131 CONCENTRATION.

1.0 Symptoms

- 1.1 LMF-48 reactor coolant radiation monitor has alarmed.
- 1.2 LMF-18 reactor coolant filter 1A radiation monitor has alarmed.
- 1.3 LMF-19 reactor coolant filter 1B radiation monitor has alarmed.
- 1.4 Any plant condition in which the operator would suspect failed fuel or want an estimate of the amount of failed fuel.

2.0 Immediate Action

2.1 Automatic

None

2.2 Manual

- 2.2.1 Obtain a chemistry sample of the reactor coolant in order to determine the I-131 concentration of the coolant.
- 2.2.2 Once the I-131 concentration is known for the reactor coolant determine which of the following four cases best describes the present fuel conditions.

NOTE:

- A. The numbers obtained by using this procedure are at best, estimates only.
- B. All formulas quoted are based upon equilibrium full power core iodine. If fuel damage is suspected to have occurred during times of reduced power or near the time of significant power change, the core iodine inventory must be compensated accordingly by using Enclosure 4.2. This is the correction factor Y.
- C. All values given are normalized to volumes of coolant at normal reactor coolant system pressure and temperature. To correct for other NC system temperatures or reduced NC sample temperatures, use Enclosure 4.1. This is the correction factor X.
- D. The decay of  $T_e^{131}$  to I-131 has been neglected as insignificant in this analysis.

- E. Iodine spiking may occur after a shutdown or significant power change. Data from other Westinghouse plants has shown that the iodine spiking process has been observed to occur during a period of 1 to 3 days after the change or shutdown. However, the spike seems to peak during the period from 4 to 8 hours after the change. I-131 concentrations can increase by a factor of 2 to 25 above the equilibrium levels during these times, although an increase over a factor of 10 is unusual and would only be seen at a shutdown. Increases by a factor of 2 to 3 are typical for a significant power decrease (i.e., 100% to 50% power). Do not misinterpret this temporary change for fuel failure if there is no other evidence of fuel damage. Other evidence of fuel damage can be constituted by any indication of inadequate core cooling, loose parts indication, high incore thermocouple indication, etc.
- F. If estimates for fuel failure are needed for fuel conditions other than those covered by the four cases described below, or if more accurate fuel failure data is needed, see Section 2.2.7 of this procedure.
- G. The following four cases cover a very broad range of core conditions. Choose the one that best suits the existing conditions.
- H. Chemistry samples should be taken as soon as damage is suspected.

### 2.2.3 Case I - Normal Operation

- 2.2.3.1 The conditions which pertain to Case I - Normal operation are as follows:

- 2.2.3.1.1 Normal reactor operation at any power or shutdown with no unusual conditions prior to shutdown. Adequate core cooling has been maintained.



2.2.3.2 If the above best describes the core conditions, use the following formulas to calculate the range of failed fuel values. Evaluate correction factors X and Y by using Enclosures 4.1 and 4.2.

2.2.3.2.1 (Measured I-131 concentration  $\mu\text{CI/ml}$   
 $+ 3.5 \times 10^{-3} \mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
 = Number of failed pins (Max. expected and best estimate)

2.2.3.2.2 (Measured I-131 concentration  $\mu\text{CI/ml}$   
 $+ 4.9 \times 10^{-3} \mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
 = Number of failed pins (Min. expected)

2.2.3.2.3 (Measured I-131 concentration  $\mu\text{CI/ml}$   
 $+ 1.8 \mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
 = Percent failed fuel (Max. expected and best estimate)

2.2.3.2.4 (Measured I-131 concentration  $\mu\text{CI/ml}$   
 $+ 2.5 \mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
 = Percent failed fuel (Min. expected)

NOTE: Typical values for I-131 concentration in  $\mu\text{CI/ml}$  for a normally operating plant are between  $1.0 \times 10^{-2}$  and  $4.0 \times 10^{-2} \mu\text{CI/ml}$ . These values are based on the reactor coolant I-131 activities experienced by the Zion and Trojan Plants.

#### 2.2.4 Case II - Macroscopic Clad Damage

2.2.4.1 The conditions which pertain to Case II - Macroscopic clad damage are as follows:

2.2.4.1.1 Normal reactor operation at any power, or shutdown where some mechanical clad failure (i.e., a loose part monitor indication) or a flow induced failure is suspected. The core has adequate cooling and no significant fuel overtemperature is observed.

2.2.4.2 If the above best describes the core conditions, use the following formulas to calculate the range of failed fuel values. Evaluate correction factors X and Y by using Enclosure 4.1 and 4.2.

2.2.4.2.1 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $5.5 \times 10^{-2} \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Max. expected)

2.2.4.2.2 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $16.5 \times 10^{-2} \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Best estimate)

2.2.4.2.3 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $27.4 \times 10^{-2} \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Min. expected)

2.2.4.2.4 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $27.9 \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Max. expected)

2.2.4.2.5 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $33.7 \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Best estimate)

2.2.4.2.6 (Measured I-131 concentration  $\mu\text{Ci/ml}$   
+  $119.5 \mu\text{Ci/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Min. expected)

## 2.2.5 Case III - Severe Fuel Overtemperature

2.2.5.1 The conditions which pertain to Case III - Severe Fuel Overtemperature are as follows:

2.2.5.1.1 TMI type accident where there has been an abnormal shutdown and it is suspected that the fuel has been at least partially uncovered for a period of time greater than a few minutes. Voiding in the core is detected by high incore thermocouple readings and loss of margin to saturation. Fuel clad oxidation is detected by excess hydrogen in the containment or in the reactor coolant sample; however, no fuel melting is suspected.

2.2.5.2 If the above best describes the core conditions, use the following formulas to calculate the range of failed fuel values. Evaluate correction factors X and Y by using Enclosures 4.1 and 4.2.

2.2.5.2.1 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 2.4  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Max. expected)

2.2.5.2.2 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 2.9  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Best estimate)

2.2.5.2.3 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 3.2  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Min. expected)

2.2.5.2.4 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 1255  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Max. expected)

2.2.5.2.5 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 1535  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Best estimate)

2.2.5.2.6 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 1675  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Percent failed fuel (Min. expected)

## 2.2.6 Case IV - Fuel Melting

2.2.6.1 The conditions which pertain to Case IV - Fuel Melting, are as follows:

2.2.6.1.1 Severe accident where there has been an abnormal shutdown and the core is uncovered for a long period of time. Incore thermocouple temperature readings are above 2300°F for a long period of time. Fuel melting is suspected (i.e., fuel temperature exceeds 3000°F) and is verified by the instrument. Operate the incore instrumentation system properly.



2.2.6.2 If the above best describes the core conditions, use the following formulas to calculate the failed fuel values. Evaluate correction factors X and Y by using Enclosures 4.1 and 4.2.

2.2.6.2.1 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 5.5  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Number of failed pins (Best estimate)

2.2.6.2.2 (Measured I-131 concentration  $\mu\text{CI/ml}$   
+ 2790  $\mu\text{CI/ml}$ )  $\cdot X \cdot Y$   
= Percent of failed fuel (Best estimate)

2.2.7 If fuel conditions other than those described above exist, or if a more detailed failed fuel estimation is desired for either emergency or normal operation, contact the appropriate Westinghouse people below in the order listed until contact is made.

2.2.7.1 Emergency Plant Conditions - Emergency Response  
Team Westinghouse, Pittsburgh, Pennsylvania

2.2.7.1.1 Director: Hank Ruppel 412/256-3611 Work  
412/366-8781 Home

2.2.7.1.2 Deputy Director: Ron Lehr 412/256-3401 Work

2.2.7.1.3 Technical Support Manager: Tom Anderson  
412/373-5766 Work; 412/327-8289 Home

2.2.7.1.4 Materials Design: Wally Chubb 412/373-4364 Work

2.2.7.2 Normal Plant Conditions

2.2.7.2.1 Southern Regional Manager - Steve Longdon -  
404/885-5900, Work

2.2.7.2.2 Westinghouse - Duke Representative - Mike  
Waller - 412/373-3160, Work

2.2.7.2.3 Materials Design - Wally Chubb - 412/373-4364,  
Work

### 3.0 Subsequent Actions

3.1 Follow up as necessary with Westinghouse - Pittsburgh, Pennsylvania depending on the plant situation.

4.0 Enclosures

- 4.1 Density Correction Factor,  $X$ , for NC Temperature Changes
- 4.2 Iodine 131 Inventory Correction Factor,  $Y$ , for reduced power operation or for times of power change
- 4.3 Examples

AP/O/A/5500/33

Enclosure 4.1

Density Correction Factor, X, for NC Temperature Changes

Find the appropriate NC System temperature at the time of accident. Find the approximate temperature at which the NC samples are taken. The intersection of both numbers is the density correction factor, X.

NOTE: Normal NC System sample temperature is approximately 90°F. Use this temperature if no other information is available.

NCS Sample Temperature °F

Reactor Coolant System Temperature at time of accident °F	NCS Sample Temperature °F		
	80	90	100
100	.996	.998	1
150	.983	.985	.987
200	.966	.968	.970
250	.945	.947	.949
300	.921	.923	.924
350	.894	.895	.897
400	.862	.864	.865
450	.827	.828	.830
500	.787	.788	.790
550	.739	.740	.741
560	.723	.729	.731
570	.717	.718	.719
580	.706	.708	.708
590	.693	.694	.695
600	.680	.681	.683

AP/O/A/5500/33

Enclosure 4.2

Iodine 131 Inventory Correction, Y, for Reduced  
Power Operation or for Times of Power Change

Situation 1

To correct for core Iodine Inventory if fuel damage is suspected to have occurred during times of any power level except 0% where the power level has not changed greater than +10% within the last 22 days, use the following equation.

$$Y = \frac{100}{\% \text{ Full Power at time of failure}}$$

where Y is the correction factor to be used in Section 2.0.

Example: The plant has been at 35% full power for the last 30 days when fuel damage is suspected. Therefore:

$$Y = \frac{100}{35} = 2.86$$

Situation 2

To correct for core iodine if fuel damage is suspected to have occurred at times other than fit Situation 1 above, use the following equation.

$$Y = \frac{100}{\text{old power level in \%} (e^{-\lambda_1 t}) + \text{new power level in \%} (1 - e^{-\lambda_1 t})}$$

where:

Y = correction factor to be used in Section 2.0

old power level in % = the % full power before the power change

new power level in % = the % full power after the power change at which time the fuel failure has occurred

$\lambda_1$  = is the decay constant for  $I_{131}$  which equals .0864 day<sup>-1</sup>

t = is the median time to make a power change plus the time after the power change until damage is suspected to have occurred, in days.

Example: If it took 2 hours to make a power change and damage was suspected 10 hours after the power change.

$$t = \frac{2}{24} + 10 = 11 \text{ hours}$$



AP/O/A/5300/33  
Enclosure 4.3  
Examples

Problem 1

- a. Power level has been decreased from 85% to 50%.
- b. This power change took four hours and occurred between 1200 and 1600.  
 $T_{AVG}$  at 50% is  $570^{\circ}\text{F}$ .
- c. At 1800 a loose part monitor alarm goes off indicating a loose object in the core. The reactor is not tripped.
- d. A Chemistry team is immediately dispatched to take a sample NC System as failed fuel is suspected.
- e. Chemistry sample indicates I-131 concentration is  $10.0 \mu\text{Ci/ml}$ .

Part 1. Determine the best estimate of the number of failed pins.

Part 2. Determine the best estimate of percent failed fuel.

Solution

This is Case II, Step 2.2.4

Use equation 2.2.4.2.2 for Part 1

Use equation 2.2.4.2.5 for Part 2

$$\text{Part 1. } \left( \frac{\text{Measured I-131 concentration } \mu\text{Ci/ml}}{16.5 \times 10^{-2} \mu\text{Ci/ml}} \right) \cdot X \cdot Y = \text{Number of failed pins}$$

Determine X: Enclosure 4.1

$T_{AVG}$  is  $570^{\circ}\text{F}$  at 50%.

Assume NCS Sample Temperature is  $90^{\circ}\text{F}$

Therefore,  $X = .713$

Determine Y: Enclosure 4.2

$$\lambda_I = .0864 \text{ day}^{-1}$$

$$t = \frac{(4) - (2)}{2} = 4 \text{ hours}$$

Remember,  $t$  is the median time to make a power change plus the difference between the time when the damage is suspected and the time the new power level is reached.

$$\text{Convert } t \text{ to days } t = 4 \text{ hours} \times \frac{1 \text{ day}}{24 \text{ hrs.}} = .167 \text{ days}$$

AP/O/A/5500/33  
 Enclosure 4.3  
 Examples

$$Y = \frac{100}{85 - [(.0864 \text{ day}^{-1})(.167 \text{ day})] + 50 \left[ 1 - e^{-[ (.0864 \text{ day}^{-1})(.167 \text{ day})]} \right]}$$

$$Y = \frac{100}{85(.9857) + 50 (.0143)} = 1.183$$

Part 1.  $\frac{10 \text{ } \mu\text{CI/ml}}{16.5 \times 10^{-2} \text{ } \mu\text{CI/ml}} (.718) (1.183) = 51.5 = 52 \text{ failed pins}$

Answer

Part 2.  $\frac{\text{Measured I-131 Concentration } \mu\text{CI/ml}}{83.7 \text{ } \mu\text{CI/ml}} \cdot X \cdot Y = X \text{ failed fuel}$

$$\frac{10 \text{ } \mu\text{CI/ml}}{83.7 \text{ } \mu\text{CI/ml}} (.718) (1.183) = 0.1\% \text{ failed fuel}$$

Answer

AP/Q/A/5500/33  
Enclosure 4.3  
Examples

Problem 2

- a. The reactor has just tripped instantly from 100% power due to a malfunctioning instrument. There were no unusual conditions prior to the trip.
  - b.  $T_{AVG}$  is now 557°F at 0% power.
  - c. The operator, while having no reason to suspect failed fuel, is curious about the amount of failed fuel present now following the trip.
  - d. A Chemistry team is sent to take an NC sample 12 hours after the trip.
  - e. The Chemistry sample gives an I-131 concentration of  $3.0 \times 10^{-3}$   $\mu\text{Ci/ml}$ . (A typical value for a normally operating plant. See Note under Case I, Step 2.2.3.2.)
  - f. Chemistry personnel also indicate that NC sample temperature is 100°F.
- Part 1. Determine the maximum expected number of failed fuel pins.
- Part 2. Determine the maximum expected percent failed fuel in the core.

Solution

This is Case I, Step 2.2.3

Use equation 2.2.3.2.1 for Part 1

Use equation 2.2.3.2.3 for Part 2

$$\text{Part 1. } \frac{\text{Measured I-131 concentration } \mu\text{Ci/ml}}{3.5 \times 10^{-3} \mu\text{Ci/ml}} \cdot X \cdot Y = \text{Number of failed pins}$$

Determine X: Enclosure 4.1

NC Temperature is 557°F at 0%

NC sample temperature is 100°F

Therefore,  $X = .732$

AP/O/A/5500/33  
Enclosure 4.3  
Examples

Determine Y: Enclosure 4.2

Situation 2:  $t = 12 \text{ hours} \times \frac{1 \text{ day}}{24 \text{ hrs.}} = .5 \text{ days}$

$$Y = \frac{100}{100 - [(.0864)(.5)] + 0 \left[ 1 - e^{-[ (.0864)(.5) ]} \right]}$$

$$Y = 1.044$$

NOTE: If  $t = 0$  or a sample was taken immediately,  $Y = 1.0$ .

Part 1.  $\frac{2.0 \times 10^{-2} \text{ uCI/ml}}{3.5 \times 10^{-3} \text{ uCI/ml}} (.732)(1.044) = 4.4$

or =4 to 5 failed pins

Answer

Part 2.  $\frac{\text{Measured I-131 Concentration uCI/ml}}{1.8 \text{ uCI/ml}} \cdot X \cdot Y = \text{Z failed fuel}$

$$\frac{2.0 \times 10^{-2} \text{ uCI/ml}}{1.8 \text{ uCI/ml}} (.732)(1.044) = .0085 \text{ Z failed fuel}$$

The above numbers are indicative of normal operation.

Answer

NOTE: I-131 spiking may be a problem here. See Step 2.2.2, Note E.



AP/O/A/5500/33

Enclosure 4.3

Examples

Problem 3

- a. Power level has been between 50% and 65% for the last 30 days and is presently at 60% at 1800.
- b.  $T_{AVG}$  is  $575^{\circ}F$  at 60% power.
- c. It is desired to see if any significant failed fuel exists in the core even though no abnormal occurrences have taken place.
- d. At 2200 the same day, a chemistry sample is taken of the NC system.
- e. The chemistry sample indicates I-131 concentration is  $3.9 \times 10^{-2}$   $\mu CI/ml$ .

Part 1. Determine the best estimate of the number of failed pins.

Part 2. Determine the best estimate of the % failed fuel.

Solution

This is Case I. Step 2.2.3

Use equation 2.2.3.2.1 for Part 1

Use equation 2.2.3.2.3 for Part 2

Part 1.  $\frac{\text{Measured I-131 Concentration } \mu CI/ml}{3.5 \times 10^{-3} \mu CI/ml} \cdot X \cdot Y = \text{number of failed pins}$

Determine X: Enclosure 4.1 $T_{AVG}$  is  $575^{\circ}F$  at 60% powerAssure NCS sample temp. of  $90^{\circ}F$ Therefore  $X = .713$ Determine Y: Enclosure 4.2

Situation 1

$$Y = \frac{100}{60} = 1.67$$

AP/O/A/5500/33  
 Enclosure 4.3  
 Examples

Part 1.  $\frac{3.9 \times 10^{-2} \text{ } \mu\text{CI/ml}}{3.5 \times 10^{-3} \text{ } \mu\text{CI/ml}}$   $(.713)(1.67) = 13.27$   
 $= 14 \text{ failed pins}$  Answer

Part 1. Measured I-131 Concentration  $\mu\text{CI/ml}$   $\cdot X \cdot Y = Z \text{ failed fuel}$   
 $1.8 \text{ } \mu\text{CI/ml}$

$\frac{3.9 \times 10^{-2} \text{ } \mu\text{CI/ml}}{1.8 \text{ } \mu\text{CI/ml}}$   $(.713)(1.67) = .026\% \text{ failed fuel}$  Answer

The above numbers are acceptable for a normally operating plant.

AP/O/A/5500/33  
Enclosure 4.3  
Examples

Problem 4

- a. The unit has been at 97% power for a month when a depressurization of the NC system occurs.
- b. The reactor trips.
- c. Heavy vibration is observed in the NC pumps.
- d. Thermocouple temperatures over 1000°F are indicated in the core.
- e. LMF 48 and LMF 13 have gone off.
- f. Safety Injection was delayed and it is suspected the core was uncovered between 30 and 60 minutes before sufficient reactor vessel water level was regained.
- g. The incore instrumentation system is still operable.
- h. The NC sample indicates an I-131 concentration of 3800  $\mu\text{Ci/ml}$ .
- i. A Chemistry sample is taken immediately (within the hour) after the trip.

Part 1. Determine the maximum expected number of failed pins.

Part 2. Determine the maximum expected % of failed fuel.

Solution

This is Case III, Step 2.2.5

Use equation 2.2.5.2.1 for Part 1

Use equation 2.2.5.2.4 for Part 2

Determine X: Enclosure 4.1

NC Temp.  $T_{\text{AVG}}$  at 0% power is 357°F

Assume sample temperature of 90°F

Therefore,  $X = .730$

Determine Y:

$$Y = \frac{100}{97} = 1.03$$

AP/O/A/5300/33  
Enclosure 4.3  
Examples

Part 1.  $\frac{3800 \text{ uCi/ml}}{2.4 \text{ uCi/ml}} (.730)(1.03) = 1190.5$

=1191 number failed pins, max. expected Answer

Part 2.  $\frac{3800 \text{ uCi/ml}}{1255 \text{ uCi/ml}} (.730)(1.03) = 2.28\% \text{ failed fuel, max. expected}$

Answer