September 1, 1983

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Docket No. 50-206

Mr. R. Dietch
Vice President
Nuclear Engineering and Operations
Southern California Edison Company
2244 Walnut Grove Avenue
Post Office Box 800
Rosemead, California 91770

Dear Mr. Dietch:

SUBJECT TMI ACTION PLAN ITEM II.B.3, POST-ACCIDENT SAMPLING SYSTEM
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NO. I

Enclosed is the NRC staff's evaluation of your December 3, 1982 submittal regarding the post-accident sampling system for San Onofre Nuclear Generating Station, Unit No. 1. The evaluation found that you meet nine of the eleven criteria in Item II.B.3 of NUREG-0737. A discussion of the open items (criterion 2 and criterion 10) is contained in the enclosed evaluation.

We request that you address the open items within 45 days of the receipt of this letter.

Sincerely,

Original signed by

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
As stated
Attach 2 - 8204070694

cc w/enclosure
See next page

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

September 1, 1983

Docket No. 50-206
LS05-83-09-042

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Safety Evaluation by
the Office of Nuclear Reactor Regulation
Related to Operation of
San Onofre Nuclear Generating Station Unit No. 1
Southern California Edison
Docket No. 50-206

Post-Accident Sampling System (NUREG-0737), II.B.3

Introduction

The post-accident sampling system (PASS) is evaluated for compliance with the criteria in NUREG-0737, Item II.B.3. The licensee should provide information on the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g. noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737, II.B.3.

To satisfy the requirements, the licensee should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary to comply with NUREG-0737, Item II.B.3, and (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements are met.

Evaluation

By letter dated December 3, 1982, the licensee provided information on the PASS.

Criterion: (1)

> The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

The PASS has sampling and analysis capability to promptly obtain pressurized and unpressurized reactor coolant samples, and analyze these samples and containment atmosphere samples within three hours from the time a decision is made to take a sample. Samples will be taken and with the exception of chlorides analyzed at a shielded underground sample station 400 feet from the control room. The station provides the capability of remote analysis of the reactor coolant, with remote collection and dilution of the reactor coolant and containment atmosphere samples for subsequent radiological analysis by using Unit 1 laboratory equipment. In addition, the Unit 1 Diesel Generators will be the alternative backup power source for the PASS if loss of offsite power occurs. The NRC staff has determined that these provisions meet criterion (1) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (2)

> The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes);

b) hydrogen levels in the containment atmosphere;

c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

d) Alternatively, have in-line monitoring capabilities to perform all or part of the above analyses.

The PASS is capable of analyzing, by grab samples and remotely controlled in-line monitors, noble gases, iodines and cesiums, non-volatile isotopes, boron, chloride, hydrogen, pH, and dissolved gases in the primary coolant and the containment atmosphere.

The NRC staff finds that the licensee partially meets Criterion (2) by establishing an on-site radioligical and chemical analysis capability. However, the licensee should provide a procedure, consistent with our clarification of NUREG-0737, Item II.B.3, Post Accident Sampling System, transmitted to the licensee on June 30, 1982, to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location. Guidance for the procedure to estimate core damage is attached (Attachment 1). In addition, the licensee should describe the accuracy of the chloride analysis.

Criterion: (3)

Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system.

Reactor coolant and containment atmosphere sampling during post accident conditions does not require an isolated auxiliary sysem to be placed in operation in order to perform the sampling function. The PASS provides the ability to obtain samples from loop C of the reactor coolant system and the containment atmosphere without using an isolated auxiliary system. The licensee's proposal to meet Criterion (3) is acceptable since PASS sampling is performed without requiring operation of an isolated auxiliary system and PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they need to operate.

Criterion: (4)

> Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

The PASS is able to obtain pressurized reactor coolant samples and also can quantify the amount of dissolved gases. O_2 concentration measurement capabilities is being provided to meet the requirements of Regulatory Guide 1.97, Rev. 2. The oxygen analyzer will provide an auto-ranging remote readout with four ranges: 0 to 19.9 ppb, a 1.8 to 199 ppb, a 0.18 to 1.999 ppm and 1.8 to 19.990 ppm. The accuracy of the reading is $\pm$ 4% in the lowest range. The NRC staff has determined that these provisions meet Criterion (4) of Item II.B. 3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (5)

The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

An undiluted grab sample facility is provided which meets the 96-hour chloride limit for a fresh water plant. The undiluted grap samples will be analyzed offsite at General Atomic Company within four days. The NRC staff has determined that these provisions meet Criterion (5) and are, therefore, acceptable.

Criterion: (6)

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

The licensee has performed a shielding analysis to ensure that operator exposure while obtaining and analyzing a PASS sample is within acceptable limits. This operator exposure includes entering and exiting the sample panel area, operating sample panel manual valves, positioning the grab sample into the shielded transfer carts, and performing manual sample dilutions, if required, for isotopic analysis: PASS personnel radiation exposures from reactor coolant and containment atmosphere

sampling and analysis are within 5 rem whole body and 75 rem extremities which meet the requirements of GDC (19) and Criterion (6) and are, therefore, acceptable.

Criterion: (7)

The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Boron analysis of the reactor coolant will be performed by an in-line boron meter with a measurement capability from 100 ppm 4,400 ppm under accident conditions with an accuracy of ± 2%. We find that this provision meets the recommendations of Regulatory Guide 1.97, Rev. 2 and Criterion (7) and is, therefore, acceptable.

Criterion: (8)

If in-line monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

The PASS has diluted grab sample capability and undiluted backup grab sample capability for sampling within 30 days as required. Diluted grab samples will be analyzed at the Unit 1 site and undiluted samples will be sent offsite for analysis. Provisions to flush in-line monitors are provided in the PASS using demineralized

water and nitrogen. The NRC staff has determined that these provisions meet Criterion (8) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (9)

The licensee's radiological and chemical sample analysis capability shall include provisions to:

a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source term given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1m Ci/g to 10 Ci/g.

b) Restrict background levels or radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source term given in Regulatory Guide 1.4, Rev. 2 and 1.7. The activity detection range for reactor coolant

is 1 to 10^7 Ci/cc. Radiation background levels will be restricted by shielding and ventilation in the radiological and chemical underground sampling station analysis facilities such that analytical results can be obtained within an acceptably small error (approximately a factor of 2). The NRC staff finds that these provisions meet Criterion (9) and are, therefore, acceptable.

Criterion: (10)

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Analytical accuracies for gamma spectra, boron, hydrogen, pH, and oxygen analyses are estimated. The equipment and procedures used for post-accident sampling and analyses will be calibrated and chemistry personnel retrained in the use of the system on a frequency of at least once every six (6) months.

The NRC staff finds that the licensee partially meets Criterion (10). The licensee should verify that the accuracy is consistent with the guidelines in our letter dated June 30, 1982 for chloride analysis. The licensee should also provide information on the measurement ranges and sensitivity of the procedure to demonstrate on the standard test matrix that the selected procedures and instrumentation will achieve the accuracies.

A report on the evaluation of Sentry Equipment Corporation and General Electric Company analytical chemical procedures for post accident analysis is provided for information (attachment 2).

Criterion: (11)

In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

a) Provisions for purging sample lines, for reducing plateout in sample line, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-effciency particulate air (HEPA) filters.

The licensee has addressed provisions for purging to ensure samples are representative, size of sample line to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. Furthermore, information was also provided regarding containment atmosphere sample line heat tracing to limit iodine plateout.

The NRC staff has determined that the licensee meets Criterion (11) of Item II.B.3 or NUREG-0737.

Conclusion

The NRC staff concludes that the post- accident sampling system partially meets the criteria of Item II.B.3 of NUREG-0737. The licensee's responses to nine of the criteria are acceptable. The two criteria which have not been fully resolved are:

Criterion (2) provide a core damage estimate procedure.

Criterion (10) provide information demonstrating applicability of procedures and instrumentation in the post accident water chemistry and radiation environment.

ACKNOWLEDGEMENT

Paul Wu contributed to this evaluation.

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 conditons; 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 is 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 [illegible] 54 minutes, although accidental 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 (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 ± ½ 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., [illegible] caloy in the core.
14. The damage in the core [illegible] rods from t downward at least 7 feet, and probably over most of the core and consists of embrittled Zircaloy cladding topped by a debris that probably consists of fuel pelle ments, partially dissolved fuel pellets, s Zircaloy oxide, and segments of embrittl caloy cladding with outer skins of Zirca ide, all glued together with liquefied fuel relatively tight and compact mass ext entirely across the core from wall to w penetrated by only a few vertical passag at most. In addition, fingers of liquefied f tend downward from the debris bed in continuous subchannels between fuel ro compassing the neighboring fuel rods depth of about 1 foot above the bottom fuel stack in the fuel rods. Not less th of the fuel assemblies have such fin liquefied fuel.

c. Core Damage Estimates from Fission Product Release

At shutdown the reactor core contained products, activation products, and actinides. of these, notably krypton and xenon, are g and can diffuse through the fuel pellet to c the gap between the fuel and the cladding lesser extent, the halogens (iodine and brom also diffuse into the fuel-clad gap. Any per of the cladding can release these fission p into the reactor coolant.

If the fuel temperatures are higher than o temperatures, but well below melting, other r tive materials are volatilized and can diffu Also, diffusion of the noble gases and halo creases so that a larger fraction of these released. The release of cesium is quite and could be caused by compound formatic cause of this variability and what is now about cesium, it is not possible to determi cisely the temperature at which a reasonab fraction of the cesium would be released; h it is believed temperatures would not be low 1300°C (2370°F).[187,188]

At higher temperatures that cause the li tion or melting of fuel, some fraction of othe products such as tellurium can be release reported show that the escape of tellurium on many factors other than temperature.[189] oxidizing conditions some ruthenium m

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,[187,190] 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,[188] 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[192,193] and of Forsyth and Eklund[194] 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.[191] 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[195] of fission products, activation products,

TABLE II-56. Activity in volatile groups*

Group	Activity
I	2.97×10^8 Ci
II	4.47×10^5 Ci
III	4.6×10^7 Ci
IV	1.61×10^8 Ci
V	3.85×10^8 Ci
VI	6.34×10^8 Ci
VII	2.69×10^9 Ci
VIII	4.80×10^8 Ci
Total	5.11×10^5 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[196] 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. However, 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 (^{85}Kr and ^{133}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 27% to 48% of the core inventory of iodine and cesium to be in the reactor building sump water.[198] In addition to iodine and cesium, very small amounts of Ru, Nb, Sb, La, and Ag were found. As expected, ^{90}Sr was found. At most, the amounts corresponded to a few millionths of the core inventory. At 0.02% of the core inventory of ^{129m}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

[illegible] of the amount of volatiles.

Findings

From those 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 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 [illegible] steam generated by the [illegible] reaction 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 UO 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, o[illegible]

TABLE II-57. Total volatile isotopes released from core

Released To	Isotope (fraction of core inventory)				
	^{133}Xe	^{131}I (^{135}I)	^{137}Cs	^{134}Cs	
Environment	0.01[1]	—[2]	—	—	
RB Atmosphere	0.46[3]	—	—	—	X
RB Water	—	0.22[4]	0.48[4]	0.34[4]	X
RC Water	—	0.14[4]	0.12[4]	0.08[4]	X
Aux. Bldg. Tanks	—	0.03	0.03	0.02	
Totals	0.46	0.39	0.63	0.44	

[1] See Ref. 199
[2] Dashes indicate low values (generally less than 1%)
[3] Best estimate from data in Ref. 197.
[4] Average of observations.