Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

April 15, 1983
LIC-83-083

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

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

The Omaha Public Power District received the Commission's letter dated January 31, 1983 and the attached draft Safety Evaluation Report (SER) on the Fort Calhoun post accident sampling system. The SER identified three (3) open items (specifically Criteria 2, 5, and 10) for which additional information was requested.

The attachment to this letter, OPPD Response to NRC Safety Evaluation Report, serves to provide the requested information for the three (3) open items. Any requested information which the District has not yet obtained will be provided to the Commission as it becomes available.

The District also wishes to provide clarifications to statements made in the draft SER. These clarifications are also provided in the attached response.

Sincerely,

W. C. Jones
Division Manager
Production Operations

A046

WCJ/TLP:jmm

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

Mr. L. A. Yandell, NRC
Senior Resident Inspector

OPPD RESPONSE TO NRC SAFETY EVALUATION REPORT

POST ACCIDENT SAMPLING SYSTEM (NUREG-0737), II.B.3

Criteria 2, 5, and 10 With Additional
Clarification to Criteria 3 and 8

Criterion (2): Provide a core damage procedure.

Response (2): A core damage procedure (OI-PAP-6) is attached. Please note that this procedure will not be formally incorporated into the operating manual until the system is declared fully operational.

Criterion (5): Provide for obtaining an undiluted sample for chloride analysis within 30 days and establish the accuracy of the chloride analysis on the diluted sample.

Response (5): As reported in our response (Ref. 1, Pg. 9) the Post Accident Sampling System (PASS) has the capability to provide a pressurized undiluted grab sample for offsite chemical and radiological analysis. This grab sample can also be analyzed for chloride concentration after a 30-day decay period. The District does not currently have an agreement with any outside laboratory to perform the chemical or radiological analysis on the pressurized undiluted grab sample.

With regard to an on-line chloride analysis, the District is actively working on resolving problems with the Ion Chromatograph (IC) supplied by Dionex. The Ion Chromatograph provided for PASS is a Dionex Model 10, which is the same model that was tested by Exxon Nuclear Idaho Company, Inc. (ENICO), in their evaluation titled "An Evaluation of GE and SEC Chemical Procedures for Post Accident Analysis of Reactor Coolant Samples," Nov. 1981.

The ENICO evaluation has shown that an accuracy of ± 15% is achievable by using the Dionex Chromatograph. The District stated previously that the accuracy was ± 5%. We wish to clarify that the accuracy is ± 15%.

The accuracy of the chloride analysis on diluted samples is notably dependent on the accuracy of determining the dilution ratio. The PASS has the capability of determining the dilution ratio to within approximately 30%. The accuracy of the laboratory chloride analysis on a diluted sample is estimated to be ± 35%.

Criterion (10): Provide information demonstrating applicability of procedures and instrumentation in the post accident water chemistry and radiation environment, and technical specifications on calibration and testing of the PASS and training of PASS operators.

Response 10 (a):
A. Procedure & Instruments

The District has noted the NRC's recommendations to perform a chemical matrix and radiation test or. in lieu of a test, to provide evidence that the selected instruments have been used successfully in a similar environment. Evidence presented in Table 1 shows that the selected instruments have been either successfully used or are qualified to perform in post accident conditions.

TABLE 1

Analysis	Radiation and Chemical Interference Effects on Components
1. Gross Activity	To analyze samples for gross gamma activity, detectors were provided by the Technology for Energy Corporation (TEC), Model No. 714. In accordance with the vendor information, the equipment is capable of operating in a 2.5×10^4 Rads/hr. radiation environment(3), which exceeds the post accident levels for the area of equipment operation. This equipment is not affected by chemical interference since it is not in physical contact with the sample.
2. Gamma Spectrum	To perform an isotopic analysis, Canberra Model 7229P-1020 Intrinsic Germanium Detectors have been provided. This equipment is capable of operating in a 1×10^4 Rads/hr. environment(4), which is well in excess of the background radiation levels in the area where it is located. This equipment will not be affected by chemical interference since the detectors are not in physical contact with the sample.
3. Boron Analysis and 4. Chloride Analysis	A Dionex Model 10 Ion Chromatograph will be used to perform the boron and chloride analyses. This model is the same one that was tested by Exxon Nuclear in their evaluation on various procedures for analyzing post accident samples(2). In Reference 2 (Pg. 26), ENICO states that the ion chromatograph cation resins degrade at approximately 10^8 Rads, and the electronic components degrade well above 10^5 Rads. Both levels are well above those anticipated to be encountered by the IC during analysis of samples.

TABLE 1

	Analysis	Radiation and Chemical Interference Effects on Components
4.	Chloride Analysis (Con't)	The IC procedure for simultaneous chloride and boron analysis has been laboratory tested (Ref. 2, Pg. 26) using simulated fission products, caustic, cooling water impurities, and normal reactor coolant chemical additives. No chemical matrix effects were observed within the specified measurement range. ENICO's report expressed some reservations concerning the suitability of the IC procedure for boron analysis. For this reason the District will do further testing to establish the Ion Chromatograph's suitability for such analysis.
5.	Total Gas Analysis	A total gas analysis is performed by isolating a pressurized undiluted sample and expanding it. The quantity of dissolved gas in the sample is determined by Henry's law using pressure and temperature instrumentation. Two Foxboro (Model #N-E11GH) pressure transmitters (PT-2578 and PT-2580) are used in the analysis. In accordance with the published data from the vendor, both these transmitters are qualified for a 2.2×10^8 Rad total integrated dose radiation environment(5&6). This is well above the post accident radiation environment. The temperature sensor, TE-6748, is an Autoclave Model No. TS-J6 element. Radiation environment data is currently being obtained from the vendor. No adverse chemical interference effects are anticipated on either the pressure or temperature measuring instrumentation.
6.	Oxygen	No provisions were made for O_2 analysis.
7.	pH Analysis	For the pH analysis of the RC sample, an in-line monitor is used. The monitor is a TBI Model No. 567-200-1-0-8. No radiation qualification data is currently available on the monitor. However, as stated in ENICO's report (Ref. 2), in-line monitors have been used reliably for many years in the nuclear industry, and there are no anticipated radiological effects. The pH probe is encased in glass, and thus no chemical matrix effects will be seen.

Response 10(a) Con't:
B. Calibration and Testing

The proposed calibration and testing schedule for the PASS equipment is shown in Table 2. This schedule is consistent with the calibration and testing schedule of other instrumentation at the Fort Calhoun Station.

TABLE 2

	Description	Equipment	Minimum Calibration or Testing Frequency
1.	Gross Gamma Activity	TEC Model 714 Detector	Annually
2.	Gamma Spectrum Activity	Canberra Model 7229P-1020 Intrinsic Germanium Detector	Annually
3.	Boron Analysis	Dionex 10	Annually
4.	Chloride Analysis	Ion Chromatograph	Annually
5.	Total Gas	Pressure Transmitters PT-2578, PT-2580	Annually
6.	Oxygen	Not applicable. No provisions for analysis.	
7.	pH	TBI Model 567-200-1-0-8 pH probe	Once per refueling
8.	All other instruments required for proper functioning of PASS		Every 18 months

Response 10 (a) Con't:
C. Training

Operators of the PASS are chemists who will receive initial training on the sampling system and will be required to receive refresher training in post accident sampling, analysis, and transport. This training will be performed annually. The District believes this is consistent with the requirement that operators be trained annually for all other types of systems and procedures.

Response 10 (a) Con't:
D. Technical Specifications

The provisions listed in (B) and (C) above will be incorporated into the Technical Specifications when a model Technical Specification has been received from the NRC.

CLARIFICATIONS TO THE SER ISSUED BY NRC

OPPD wishes to provide clarification to the following paragraphs of the SER issued per NRC letter dated 1/31/83:

Criterion (3): Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system (RWCUS)) to be placed in operation in order to use the sampling system.

On page 4, paragraph 1, of the SER, the following statement was made. "Reactor coolant and containment atmosphere sampling during post accident conditions does not require an isolated auxiliary system to be placed in operation in order to perform the sampling function. The PASS provides the ability to obtain samples from each reactor coolant hot leg, each reactor coolant cold leg, the RHR system, the containment sump, and the containment atmosphere without using an isolated auxiliary system. The licensee's response to 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."

OPPD Clarification (3): The District would like to point out that the PASS does not provide the ability to obtain samples from each reactor coolant cold leg. The PASS can obtain samples from both reactor coolant hot legs, the low pressure safety injection system, the containment sump, the pressurizer steam space, the pressurizer surge line, the volume control tank, the RHR system, and the containment atmosphere without using an isolated auxiliary system. The District did not specify that the PASS had the capability to perform the other sampling in our licensing response dated 12/3/82, Criterion 3.

CLARIFICATIONS TO THE SER ISSUED BY NRC

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 of 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.

On page 7, paragraph 2, of the SER, it was noted:

"Both diluted and undiluted reactor coolant and containment atmosphere backup grab sample will be obtained for all required analysis. In case of an emergency, the licensee has an agreement with Cooper Nuclear Station for use of analytical equipment. In addition, an in line chemical analysis panel is provided for reactor coolant pH, boron, chloride, and total gas concentrations, as well as containment hydrogen concentrations. We find that these provisions meet Criterion (8) and are, therefore, acceptable."

Clarification (8): The District wishes to point out that we have no provisions for analysis of undiluted reactor coolant samples. The District's position remains unchanged from that presented in our response to Criterion (8) in our letter dated 12/3/82.

REFERENCES

1. "Response to NRC letter, Post Accident Sampling System," NUREG-0737, Item II.B.3. dated 12/3/82.

2. "An Evaluation of GE and SEC Chemical Procedures for Post Accident Analysis of Reactor Coolant Samples," Nov. 1981 by Exxon Nuclear Idaho Company, Inc.

3. Technology for Energy Corporation Specification for Model 714 Gross Gamma Detector.

4. Canberra Specification for Model 7229P-1020 Intrinsic Germanium Detector.

5. Foxboro Pressure Transmitter E-11GHM Series Specifications.

6. Foxboro Test Report, T3-1097 for an E11GHM Series Pressure Transmitter.

FORT CALHOUN STATION UNIT NO. 1
OPERATING INSTRUCTIONS
OI-PAP-6

Post-Accident Procedures
Procedure To Estimate Core Damage

I. PURPOSE

To estimate the degree of core damage for any plant condition in which the operator suspects fuel failure. In estimating the core damage, four major fuel conditions will be considered: no damage, cladding failures, fuel overheating, and fuel melt.

II. DISCUSSION

The core damage estimates will be based on measuring fission product concentrations from various sample locations and relating the measured concentrations to core inventory. The fission product concentrations will be measured by utilizing the post-accident sampling system (PASS) which provides data from various sample points for the purpose of estimating core damage.

The four categories of fuel condition (no damage, cladding failures, fuel overheating and fuel melt) can be differentiated by presence, absence or abundance of certain fission products and by other plant indicators (i.e. core exit thermocouples, containment or primary system hydrogen concentration, indications of core voiding or water level).

The core damage estimate will start by determining which of the four fuel conditions represents the actual core environment. The following sections describe the four fuel conditions and the fission product behavior for each condition:

A. No Damage

In the event of shutdown with adequate core cooling, the fission product concentrations for Xe-133, KR-88[1] and KR-85[2] will be measured and then related to the normal operating concentrations in the primary system. If the upper limit normal operating concentrations are not exceeded, then it could be concluded that no damage has occurred. This conclusion can be further verified by

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1. KR-88 concentration will be measured during the early stages of the accident up to 20 hours after the accident. This is due to its high abundance in the early stages of the accident.

2. KR-85 concentration will be measured later on during the accident, typically, 20 hours after the accident. This is due to its high abundance in the later stages of the accident.

II. DISCUSSION (Continued)

A. (Continued)

other plant indicators [i.e. all the core exit thermocouple readings were <700°F[1] (<~1200°F cladding temperature)[2] and no excess amount of hydrogen is found in containment or the primary system]. I-131 concentration will not be measured for the purpose of determining no-damage conditions. This is due to iodine spiking conditions which may occur after a shutdown, depressurization, or a power transient. Since the exact iodine spiking behavior is not known, this temporary change in iodine concentration may be misinterpreted as a cladding failure, therefore it is not recommended to utilize I-131 data for establishing no damage conditions[3].

B. Cladding Failures

In the event of shutdown and adequate core cooling with possible mechanical clad failures (i.e. core exit thermocouple readings are >700°F[1] (>~1200°F cladding temperature)[2] along with high radioactivity levels in the reactor coolant however, no excess amount of hydrogen is found in containment or the primary system), the fission product concentrations for Xe-133, KR-88, and KR-85 will be measured and then related to core gas gap inventory to determine percent cladding failure. The severity of the cladding failure will be divided into three subgroups:

- less than 10% cladding failure

- 10 to 50% cladding failure

- greater than 50% cladding failure

Fission Product Behavior

The fission products released into the primary system during a cladding failure (burst release) are those fission products in gaseous or vapor form which were accumulated in the fuel clad gap spaces during normal operation.

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1. Nominal hot channel outlet temperature is approximately 630°F.

2. Approximate cladding temperatures predicted are based on test data presented in Ref. 10.

3. Iodine spiking data from Fort Calhoun Plant indicates that maximum peak I-131 equivalent concentration will not exceed 16 μCi/gm when normalized to represent 2 μCi/gm normal operating concentration (Ref. 8). However, data from other power plants indicate that concentration for I-131 may increase by factor of 2 to 25 above the equilibrium levels from 4 to 8 hours after the change. Iodine spiking may last up to 4 or 5 days after change.

II. DISCUSSION (Continued)

B. (Continued)

The typical temperature range for cladding failure burst release is from 1200° to 2000°F. The fractions of fission products released into the primary system during a cladding failure are based on nominal gap release values shown on Table 1 (Attachment A) from Ref. (2). The fractions for the noble gases are modified to represent the values from Table 1 [Page II.9 from Ref. (2)]. Even though the recent data from Oak Ridge National Laboratory (ORNL) indicate lower gap release values than the ones shown on Table 1 (Attachment A), the nominal values from this table will be utilized for the purpose of estimating cladding failure. The nominal values from Table 1 are assumed to be more reliable than the recent ORNL results since, the nominal values represent the weighted average of several independent tests performed by different laboratory facilities.

In the event where the fission products released indicate greater than 100% cladding failure, there is a possibility that some fuel overheating and fuel melting has occurred (this is due to nonuniform temperature distribution within the core). Therefore, in this region it would be difficult to determine exactly whether the released fission products are due to cladding failure, fuel overheating, or fuel melting. The presence of excess hydrogen in containment or primary system with high core exit thermocouple readings >900°F (~>1500°F cladding temperature), detectable traces of some fission products (i.e., Te, Ru, Ba), and greater than normal abundance of noble gases, iodines and cesiums will be indicators of fuel overheating or fuel melting.

C. Fuel Overheating

In the event there is suspicion that the fuel has been partially uncovered for a period of time greater than few minutes [i.e. core exit thermocouple readings are >900°F (>~1500°F cladding temperature) followed by detection of excess hydrogen in the containment or primary system], the fission product concentrations for Xe-133, KR-88, KR-85, I-131 and CS-134 will be measured and then related to core inventory to determine percent fuel overheating. Absence of tellurium, ruthenium, and other low volatiles indicate that no fuel melting has occurred. The severity of the fuel overheating will be divided into two subgroups:

- less than 50% fuel overheating
- greater than 50% fuel overheating

II. DISCUSSION (Continued)

C. (Continued)

Fission Product Behavior

It would not be possible to determine accurately the amount of fuel overheated by utilizing fission product release data. During fuel overheating the diffusion of the noble gases and halogens increases as a function of fuel temperature. At fuel temperatures ~2460°F the noble gases, cesiums, and iodines previously accumulated at U_{02} grain boundaries are released. At fuel temperatures >2460° for a high burnup fuel rod, approximately 20% of the total initial fuel rod inventory of stable isotopes of the above elements would be released. Release from the grain boundaries of lower burnup fuel would probably be less, and temperatures as high as 3270°F may be required. At completion of burst release, diffusional escape, and grain boundary release, approximately 60 to 90% of the above elements remain in the UO_2 grains. Release from the UO_2 grains becomes significant from 2460°F up to 3630°F (Ref. 1) for the above elements (Xe, Kr, I, Cs) and in addition, some trace amounts of tellarium group fission products (Te, Se, Sb) may be released.

During fuel overheating (typically from 1600° to 3600°F cladding temperatures), the release fractions for noble gases, iodines, and cesiums can be anywhere from nominal gap release values up to nominal meltdown release values shown in Table 1. For example, the release fractions for Xe-133 during fuel overheating can be anywhere from 0.030 to 0.873. From the above example it can be concluded that utilizing fission product release data to estimate fuel overheating would not be accurate due to a wide range of release fractions. Similarly attempting to use relative ratio of fission products would not be very accurate (i.e. attempting to differentiate between cladding failure and fuel overheating by utilizing the ratio of the isotopes in the gas gap to the ratio of isotopes in the fuel pellet) due to the fact that during 100% cladding failure only 0.03 fraction of Xe-133 is released as compared to the wide range of releases (0.03 to 0.873) which may occur during fuel overheating. This wide range of release during fuel overheating along with theoretical uncertainties associated with the ratio method makes it impractical to use fission product release data as the basis for establishing fuel overheating conditions. In conclusion more reliable methods (i.e. core exit thermocouples, core water level, and containment or primary system hydrogen concentrations) should be used in conjunction with the fission product release data. Therefore, for purpose of determining the severity of the fuel overheating, the following conditions will be assumed:

- if less than 0.20 fraction of the core noble gases, iodines or cesiums are released, it would be concluded that there has been less than 50% fuel overheating. This conclusion must be verified by indication that some of the core exit thermocouple readings are >900°F followed by detection of excess hydrogen in primary system or containment.

II. DISCUSSION (Continued)

C. (Continued)

- if greater than 0.20 fraction of the core noble gases, iodines or cesiums are released, it would be concluded that there has been greater than 50% fuel overheating. This conclusion can be further verified by indication that approximately 50% of core exit thermocouple readings show temperatures >900°F.

D. Fuel Melting

In the event where the core has been uncovered for a long period of time (i.e., core exit thermocouple readings >900°F for a long period and some thermocouple readings indicating temperatures >1100°F followed by detection of excess hydrogen in the containment or primary system), the fission product concentrations for Xe-133, KR-88, KR-85, I-131, CS-134, Te-132, Ba-140, and Ru-103 will be measured and then related to the nominal meltdown release fraction from Table 1 (Attachment A) in order to determine the percent of fuel melted. The severity of the fuel melting will be divided into three sub-groups:

- less than 10% fuel melt
- 10 to 50% fuel melt
- greater than 50% fuel melt

Fission Product Behavior

As the water boils and core is uncovered, fuel heats and becomes molten in the range of 3800° to 5000°F and large fraction of core noble gases, iodines and cesiums will be released as shown in Table 1 (Attachment A). Also detectable amounts of tellurium group (Te, Se, Sb) and alkaline earths (Ba, Sr) fission products are expected to be released. The fuel melting estimates based on noble gases are expected to be more reliable than the estimates from other fission products. This is due to the fact that noble gases unlike other fission products do not stay confined in the vicinity of the core material. They settle or plate-out at the sampling locations or within the sampling system and are released in large quantities. However, the advantage of utilizing low-volatiles (i.e. Te, Ba or Ru) for estimating fuel melting is that low-volatiles, unlike noble gases, iodines, or cesiums are not released in significant quantities during fuel overheating. (Since noble gases, iodines and cesiums are released in significant quantities during fuel overheating, it would be difficult to determine whether the releases were due to fuel overheating or fuel melting). In conclusion the fuel melting estimates will be based on Te, Ba, or Ru. The noble gas, iodine, or cesium data will be utilized for verification of the estimates made by low-volatiles (Te, Ba, or R

II. DISCUSSION (Continued)

E. Assumptions and Conditions

1. The estimates of core damage are based on assuming that the core inventory of fission products are uniformly distributed in all fuel rods. Under actual conditions, the equilibrium radionuclide concentrations vary from rod to rod because the time to reach equilibrium for each radionuclide is different due to their variable production rates and due to different fuel rod power histories. Therefore, the optimum radionuclides for estimating core damage will vary as a function of time after refueling and power history.

   If the accident were to occur under core equilibrium conditions, isotopes (i.e. Xe-133, Kr-88, I-131, Te-132, Ba-140, Ru-103) can be utilized in estimating core damage. Under these conditions, the core inventories from attachment "C" can provide for a relatively good estimate of the actual core inventory. However, if the accident were to occur under non-equilibrium conditions (i.e. few days after refueling), the core inventory data from the attachment "C" can be adjusted by applying the correction factors in Section VII.

2. It must be noted that equilibrium samples are not available from all sample locations at the time of sampling. Under actual conditions the samples analyzed initially may be significantly different than those samples analyzed during equilibrium conditions. This is due to the fact that the fission products may not distribute uniformly at all locations and also due to the fact that maximum core degradation may not have occurred at the time of sampling. Therefore, the samples taken during the later stages of the accident may provide for a more accurate core damage estimate than those taken in the early stages of the accident.

3. Fission product inventories in the core were calculated by ORIGEN computer code assuming the following core conditions:

   - 2 fuel cycles at 100% power
   - continuous operation at 1500 MWt
   - equilibrium fuel cycle (equivalent full power days): 432 days

II. DISCUSSION (Continued)

E. Assumptions and Conditions (Continued)

4. The fission product release mechanisms described in this procedure are based on test data published in References (1) and (2). Under the actual conditions the release fractions maybe different. However, utilizing these data provides for a reasonable estimate of core environment for the subgroups discussed earlier (i.e. <10%, 10-50%, >50%).

5. The graphs on attachments "E" and "F" show the core exit thermocouple (CET) behavior under specific postulated accident conditions. CET's do not behave the same under all accident conditions. However, under a core exit superheat condition, which is common to most accidents (i.e. large or small LOCA's, loss of feed water, steam line break), the CET's can behave similarly such that the increases in CET temperatures can show increases in cladding temperatures. Additional test data or theoretical predictions are necessary for correlating CET readings to cladding temperatures during large break LOCA's.

6. The fission products utilized in this procedure are based on the test data shown on Table 1 (Attachment "A"). Under accident conditions, the isotopes chosen are released in large quantities, are detectable in both early and later stages of the accident, and are available in large inventories within the core.

F. REFERENCES

1. U.S. Nuclear Regulatory Commission - NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," June 1981.

2. U.S. Nuclear Regulatory Commission - NUREG/CR-1237 "Best Estimate Loca Radiation Signature," January 1980.

3. M. Rogovin Report, "Three Mile Island," Volume II. PP. 524-527.

4. NUREG-0737, Section II.B.3 "Post-Accident Sampling System," MR-FC-79-191.

II. DISCUSSION (Continued)

F. References (Continued)

5. Post-Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage-USNRC.

6. Fort Calhoun Operating Instructions, OI-SL-2 and OI-PAP-2.

7. CE Report, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces and System which may be used in Post-Accident Operations," December, 1979. Table 6.2.1, "Core Fission Product Inventories."

8. OPPD Letter to Mr. R. A. Clark, Reference: Docket No. 50-285 Attachment 1. June 26, 1981.

9. ICC proposals to OPPD for Fort Calhoun Station Combustion Engineering Proposals: 252995-1 (HJTCS), 252996-1, 252997-1 Nov. 17, 1981.

10. Performance evaluation of core exit thermocouples as inadequate core cooling instrumentation (draft copy) Combustion Engineering, Sept. 1982.

III. INITIAL CONDITIONS

One or more of the following conditions are present:

A. Reactor coolant system high activity has been detected Per Emergency Procedure EP-23 by any one or more of the following indicators:

1. Process radiation recorder RR-214 is in high alarm.
2. Process radiation monitor PRM-1 is in high alarm.
3. Process radiation monitor PRM-2 is in high alarm.
4. Reactor coolant isotopic gamma activity high.
5. Reactor coolant sample total activity analysis indicates high activity level.
6. Containment radiation monitor is in high alarm.

B. Any accident or other condition exists in which the operator suspects failed fuel or needs to estimate core damage.

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(Note 3)

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otopic specific activities are available per OI-SL-2 or OI-PAP-2
om the following locations (if applicable):

reactor coolant (RC) system
containment sump (CS)
containment atmosphere (CA)

btain samples in accordance with the following table:

Type of Accident	Optimum Sample Location	Identifiable Fission Products
Primary system is quickly depressurized or a large break.	RC*	Halogens, cesiums and all solids ***
	CS**	Halogens, cesiums and all solids
	CA	All noble gases
2)Primary system is slowly depressurizing or a small break	RC (early stages of the accident)	All fission products
	For later stages of the accident. Same as (1)	Same as (1)
(3)Primary system remains pressurized throughout the accident	RC	All fission products

RC sample can be obtained from one of the hot legs or if that is not possible from the pressurizer surge line.

* CS sample should be taken from sump sample line if this is not possible use recirculation line of the LPSI pumps through valves SI-159.

*** Anything other than Iodines or Noble Gases.

IV. PREREQUISITES (Continued)

Tabulate the specific activities measured for the following fission products as shown below:

	Specific Activity µCi/cc (Note 3)		
Fission Product	RC	CS	CA
Xe-133			
Kr-88 (Note 1)			
Kr-85 (Note 2)			
I-131			
Cs-134			
Te-132			
Ba-140			
Ru-103			

Note (1) Measure Kr-88 concentration during the early stages of the accident, typically up to 20 hours after shutdown.

Note (2) Measure Kr-85 concentration during the later stages of the accident, typically 20 hours after shutdown.

Note (3) For limitations see Section II.E.2.

B. Reactor water levels are available from plant computer from HJTCS.
NOTE: Heated junction thermocouple system (HJTCS) which provides an indication of liquid inventory in the reactor vessel is now under construction.

C. Hydrogen concentration in the primary coolant sample and containment atmosphere has been determined per OI-PAP-3. Containment H2 concentration can be obtained by utilizing VA-81A and VA-81B (control room panels AI-65A and AI-65B per OI-VA-6).

NOTE: Hydrogen concentrations in the primary coolant sample can be determined during normal operation by obtaining and analyzing the grab sample. However, during post-accident conditions when the reactor coolant activity is very high, use of manual methods for determination of hydrogen is not possible. The PASS is capable of analyzing the total dissolved gas. Considering that a large fraction of the total gas will be hydrogen, the increase in total gas is indicative of increase in hydrogen levels.

D. Core exit thermocouple readings are available from plant computer.

E. Data from the level indicators at the following locations is available (if SI tanks and SIRW tank have been utilized during the accident):

- SI tanks (LIA-2904, LIA-2924, LIA-2944, LIA-2964 Level Indicators)
- SIRW tank (LIC-381 or LIC-382 Level Indicators)

F. Containment pressure and temperature conditions are known.

G. Sample pressure and temperature conditions are known.

V. PRECAUTIONS AND LIMITATIONS

A. The core damage estimates assume equilibrium full power core operation as discussed in Section II.E.3 of this procedure. If the accident were to occur at non-equilibrium conditions (i.e. few days after refueling), the core damage estimate can be determined more accurately in accordance with section VII.

B. The core damage estimates calculated at different times after accident may not agree due to non-equilibrium sample conditions discussed in Section II.E.2. The samples taken during the later stages of the accident may provide for a more accurate estimate than those taken in the early stages of the accident.

VI. PROCEDURE

A. Determine the total activity released from the core (A°) for each of the fission products tabulated in section IV.A. in accordance with the following equations:

1. In the event the primary system remained pressurized:

$$A^\circ = A_{RC} \times F_T \times E_i \quad \text{For Isotope (i)}$$

2. In the event the primary system is depressurized:

$$A^\circ = (A_{RC} + A_{CS} + A_{CA})\, E_i \quad \text{For Isotope (i)}$$

$$A_{RC} = S_{RC} \times V_{RC}$$

$$A_{CS} = S_{CS} \times V_{CS}$$

$$A_{CA} = S_{CA} \times V_{CA} \times F_{TP}$$

Where:
A° = total activity released from the core for Isotope (i) - Ci
A_{RC} = total activity released to the reactor coolant - Ci
A_{CS} = total activity released to the containment sump - Ci
A_{CA} = total activity released to the containment atmosphere - Ci
S_{RC} = specific activity measured from reactor coolant (μCi/cc)* $\times 10^{-6}$ = Ci/cc
S_{CS} = specific activity measured from containment sump (μCi/cc)* $\times 10^{-6}$ = Ci/cc
S_{CA} = specific activity measured from containment atmosphere (μCi/cc)* $\times 10^{-6}$ = Ci/cc

*Obtain specific activities from table in section IV.A.

V_{RC} = reactor coolant volume = 1.873×10^8 cc
V_{CA} = containment free volume = 2.973×10^{10} cc
V_{CS} = containment sump volume - cc (See Note)

VI. PROCEDURE (Continued)

NOTE: Containment sump water volume can be approximated from the level indicators in the SI Tanks and SIRWT Tank as follows:

$V_{CS} = V_{SI} + V_{SIRWT}$
V_{SI} = Total volume released from all SI Tanks - CC

Volume released from any of the SI Tanks $= \frac{(A-B) \times 3.68 \times 10^7 CC}{100}$

A = % level before discharge, B = % level after discharge

SI TANKS	LEVEL INDICATOR	A	B	VOLUME RELEASED(CC)
SI-6A	LIA-2904			
SI-6B	LIA-2924			
SI-6C	LIA-2944			
SI-6D	LIA-2964			
Total Volume released from the SI Tanks (V_{SI}) =				

V_{SIRWT} = total volume released from the SIRW Tank
$= (L_1 - L_2)\ 6.8 \times 10^6 CC/in$

L_1 = level of WATER in SIRWT before safety injection - INCH
L_2 = level of WATER in SIRWT after safety injection - INCH
L_1 and L_2 can be obtained from LIC-381 or LIC-382.

F_{TP} = temperature pressure correction factor = $\frac{P_2(T_1+460)}{P_1(T_2+460)}$

Where: P_1, T_1 = gas dilution loop pressure and temperature
P_2, T_2 = containment atmosphere pressure and temperature

F_T = temperature density correction factor from the table below:

RCS Temperature at the Time of Sample (°F)	Primary Sample Temperature (100°F)*
100	1
150	0.987
200	0.970
250	0.949
300	0.924
350	0.897
400	0.865
450	0.830
500	0.790
550	0.741
560	0.731
580	0.709
600	0.683

*Assume 100°F sample temperature - little error will be introduced if the sample temperatures are ±50°F from actual temperatures.

VI.A. PROCEDURE (Continued)

$$E_i = e^{0.693\ \Delta t / T\text{-}1/2}$$

Where:
Ei = decay correction factor* for isotope (i)
Δt = elapsed time between the time the reactor was shutdown and the time the sample was measured - days
e = 2.718
T-1/2 = half life of a particular fission product - days
* Decay corrected values can also be obtained directly from canberra isotopic analysis equipment by inputting the decay time (the time between reactor shutdown and analysis)

Fission Product	T (Days)
Xe-133	5.27
Kr-88	0.117
Kr-85	3927
I-131	8.05
Cs-134	750
Te-132	3.24
Ba-140	12.8
Ru-103	39.5

3. Calculate A°

Calculate A° in accordance with section VI.A.1 or VI.A.2 and tabulate the results for the fission products which were detected:

TABLE (A)

Fission Product	A° (Curies)
Xe-133	
Kr-88 (Note 1)	
Kr-85 (Note 2)	
I-131	
Cs-134	
TE-132	
Ba-140	
Ru-103	

Note (1) Utilize Kr-88 data up to 20 hours after shutdown.
Note (2) Utilize Kr-85 data 20 hours after shutdown.

VI.A. PROCEDURE (Continued)

B. Interpretation of Data to Determine the Extent of Core Damage

The core damage estimate will start by determing which of the four fuel conditions (no damage, cladding failure, fuel overheating or fuel melting) best describes the core environment. A flow chart (Attachment "D") has been prepared to show the logic for estimating core damage.

1. No Damage

The core conditions which pertain to no damage or normal operation are as follows:

1.1 The reactor was shutdown with no abnormal conditions and other plant indicators verify that there has been adequate core cooling such as:

- All of the core exit thermocouple (CET) readings show temperatures <700°F (~<1200°F cladding temperature) during shutdown
  NOTE: Nominal hot channel outlet temperature is approximately 630°F.

- No excess amount of hydrogen is found in the primary system and no detectable hydrogen is found in the containment atmosphere.

1.2 If the above best describes the core environment then divide the total activity A° as recorded on Table (A) by the upper limit normal operating activities for the following fission products:

Fission Product	Reactor Coolant** Normal Operating Total Activity (Ci)	Reactor Coolant* Upper Limit Normal Operating Total Activity (Ci)
Xe-133	135	1350
Kr-88	13.7	137
Kr-85	2.38	23.8

*Reactor coolant normal operating total activities are 10% of the upper limit activities.

**Normal reactor coolant total activities were obtained by multiplying the normal operating concentrations by the reactor coolant volume (1.873×10^8cc). The normal operating concentrations are listed in the attachment "B". If desired, last available pre-accident normal concentrations may be used.

VI.B. PROCEDURE (Continued)

1.3 If $\frac{A^{\circ}}{\text{total upper limit activity}} < 1$ for all of the above fission products

then it would be concluded that no fuel damage has occurred.

1.4 If $\frac{A^{\circ}}{\text{total upper limit activity}} > 1$ for any of the above fission products

then it is possible that some cladding failure has occurred. In this case recheck the CET readings to see if any one of them indicated temperatures >700°F (>~1200°F cladding temperature) and refer to section VI.B.2 for determining the percent cladding failure.

NOTE: Do not utilize I-131 data to establish no damage conditions since the temporary change in iodine concentrations (iodine spiking) may be misinterpreted as cladding failure.

2. Cladding Failure

The core conditions which pertain to cladding failure are as follows:

2.1 The reactor was shutdown with adequate core cooling and other plant indicators verify that there has been a possibility for cladding failure, such as:

- one or more of the CET readings show temperatures >700°F (>~1200°F cladding temperature) during shutdown.
  NOTE: Nominal hot channel outlet temperature is approximately 630°F.

- no excess amount of hydrogen is found in the primary system and no detectable hydrogen is found in the containment atomosphere.

- $\frac{A^{\circ}}{\text{total upper limit activity}} > 1$ from section VI.B.1

VI.B. PROCEDURE (Continued)

2.2 If the above best describes the core environment, then divide the total activity A° recorded on Table (A) by the total gap release activity for the following fission products:

Fission Product	Gap Release** Fractions	Total Gap Release* Activity (Ci)
Xe-133	0.03	2.529×10^6
Kr-88	0.03	8.40×10^5
Kr-85	0.08	2.704×10^4

*Total gap release activities are obtained by multiplying the core inventory by the gap release fractions. The core inventories are listed in Attachment C.

**Gap release fractions are obtained from Table 1 (Attachment "A").

then $\frac{A° \times 100}{\text{total gap release activity}}$ = percent cladding failure

2.3 If $\frac{A° \times 100}{\text{total gap release activity}}$ >100% for any of the above fission products

then there is a possibility that some fuel overheating or fuel melting has occurred. Recheck the following plant indicators to determine if fuel overheating or fuel melting has occurred:

- one or more core exit thermocouples show readings >900°F (>~1500°F cladding temperatures) followed by detection of excess hydrogen in the containment atmosphere or the primary system. Refer to section VI.B.3 to estimate fuel overheating.

- the presence of any low-volatile fission products (i.e. Te, Ba, or Ru) indicates that some fuel melting has occurred. If so, refer to section VI.B.4 to estimate fuel melting.

NOTE: Absence of low-volatiles (i.e., Te, Ba, Ru) is a good indication that no fuel melting has occurred.

3. Fuel Overheating

The core conditions which pertain to fuel overheating are as follows:

3.1 There has been an abnormal shutdown and there is a possibility that the fuel has been partially uncovered for a period of time greater than a few minutes. Other plant indicators verify that there has been fuel overheating such as:

VI.B. PROCEDURE (Continued)

- Heated junction thermocouple system (HJTCS) verifies that the core has been uncovered for longer than a few minutes (HJTCS is now under construction).

- one or more core exit thermocouples show readings >900°F (>~1500°F cladding temperature).

- excess amount of hydrogen is found in the primary system or a detectable amount of hydrogen is found in the containment atmosphere.

- $\frac{A^{\circ} \times 100}{\text{total gap release activity}}$ >100% from section VI.B.2.2

3.2 If the above best describes the core environment, then divide the total activity A° recorded on Table (A) by the total core inventory for the following fission products:

Fission Product	Total Core Inventory (Ci)*
Xe-133	8.43×10^7
Kr-88	2.8×10^7
Kr-85	3.38×10^5
I-131	4.15×10^7
Cs-134	7.46×10^6

*From Attachment "C".

3.3 If $\frac{A^{\circ}}{\text{total core inventory}}$ <0.2 for all of the noble gases**

then it could be concluded that there has been <50% fuel overheating. This conclusion can be further verified by indication that some of the core exit thermocouple readings are >900°F followed by detection of excess hydrogen in primary system or containment. In case low volatiles (i.e. Te-132, Ru-103 or Ba-140) were detected, refer to section VI.B.4. to determine the percent fuel melting.

**The fuel overheating estimates which are based on noble gases are considered to be more reliable than the estimates based on iodine or cesium. In case of a great disagreement between the estimates, rely on the noble gas estimates. Use the data for iodine or cesium only if the data for noble gases are not available.

VI.B. PROCEDURE (Continued)

3.4 If $\frac{A^{\circ}}{\text{total core inventory}} > 0.2$ for any of the noble gases**

then it could be concluded that there has been >50% fuel overheating. This conclusion can be further verified by indication that >50% of core exit thermocouple readings show temperatures >900°F. In case low-volatiles (i.e. Te-132, Ru-103 or Ba-140) were detected refer to section VI.B.4 to determine the percent fuel melting.

4. Fuel Melting

The core conditions which pertain to fuel melting are as follows:

4.1 There has been a severe accident and the core has been uncovered for a long period of time. The following plant indicators verify that there has been fuel melting:

- HJTCS readings verify that the core has been uncovered for a long period of time.
- more than one core exit thermocouple show readings >900°F for a long period of time.
- one or more core exit thermocouple readings show readings >1100°F
- excess amount of hydrogen is found in the primary system or a detectable amount of hydrogen is found in the containment atmosphere.

- $\frac{A^{\circ}}{\text{total core inventory}} < 0.2$ with presence of low-volatiles (i.e. Te-132, Ru-103, or Ba-140) from section VI.B.3.3.

- $\frac{A^{\circ}}{\text{total core inventory}} > 0.2$ with presence of low-volatiles (i.e. Te-132, Ru-103, or Ba-140) from section VI.B.3.4.

- $\frac{A^{\circ} \times 100}{\text{total gap release activity}} > 100$ with presence of low-volatiles (i.e. Te-132, Ru-103 or Ba-140) from section VI.B.2.2.

**The fuel overheating estimates which are based on noble gases are considered to be more reliable than the estimates based on iodine or cesium. In case of a great disagreement between the estimates, rely on the noble gas estimates. Use the data for iodine or cesium only if the data for noble gases are not available.

VIB. PROCEDURE (Continued

4.2 If these conditions best describes the core environment, then divide the total activity A° recorded in Table (A) by the total meltdown release activity for the following fission products:

	Fission Products	Nominal Meltdown* Release Fraction	Total Meltdown** Release Activity (Ci)
	Te-132	0.150	8.91×10^6
	Ba-140	0.100	7.33×10^6
	Ru-103	0.030	2.03×10^6
SEE NOTE	Xe-133	0.873	7.36×10^7
	Kr-88	0.873	2.44×10^7
	Kr-85	0.873	2.95×10^5
	I-131	0.885	3.67×10^7
	Cs-134	0.760	5.67×10^6

*Nominal meltdown release fractions are obtained from Table 1. (Attachment A)

**Total meltdown release activities are obtained by multiplying the core inventories (from Attachment C) by the nominal meltdown release fractions.

then $$\frac{A^\circ \times 100}{\text{total meltdown release activity}} = \text{percent fuel melted}$$

NOTE: The fuel melting estimates will be primarily based on the data from low-volatiles (i.e. Te, Ba, or Ru) since the data from other fission products (i.e. Xe, Kr, I, Cs) will not provide for a good estimate of fuel melting in the lower sub-group (i.e. lower than 10% fuel melting). This is due to the fact that Xe, Kr, I and Cs fission products are released in significant quantities during fuel overheating. Therefore, it would be difficult to determine whether the releases were due to fuel overheating or fuel melting. However, estimates based on Xe, Kr, I and Cs can provide verification of the estimates which are based on Te, Ba, or Ru. In the upper sub-groups (i.e. 10-50% or >50% fuel melting) the fuel melting estimates based on noble gases, iodines and cesiums will become more realistic.

VI.B. PROCEDURE (Continued)

5. Interpretation of the Results

Tabulate the results from sections VI.B.2., VI.B.3, and VI.B.4, in the appropriate blocks on the following table:

		Core Damage Estimate * Sub-Groups - Percent		
Fuel Condition		<10%	10-50%	>50%
Cladding(a) Failure	Xe-133			
	Kr-88 or Kr-85			
		<50%		>50%
Fuel(a) Overheating	Xe-133			
	Kr-88 or Kr-85			
		<10%(c)	10-50%	>50%
Fuel(b) Melting	Te-132			
	Ba-140			
	Ru-103			
	Xe-133			
	Kr-88 or Kr-85			
	I-131			
	Cs-134			

The letters in ( ) refers to the following notes:

* Core damage estimates can be any combination of fuel conditions tabulated above.

VI.B. PROCEDURE (Continued)

5. (Continued)

The following notes should be considered in interpretation of the core damage estimates:

(a) If the cladding failure or fuel overheating estimates based on Xe and Kr are significantly different (i.e. Xe and Kr estimates are in different sub-groups), recheck the estimates and if necessary obtain new samples to determine new concentrations. If the results remain the same, rely on Xe-133 estimates.

(b) If the core environment is as described in section VI.B.4.1 and the estimates based on noble gases, iodines, and cesiums indicate significantly higher percentage of fuel melting than the estimates based on Te, Ru, or Ba (i.e. the estimates are in a different sub-groups), the low-volatile obtained estimates must be rechecked and if necessary new samples to determine new concentrations. If the results remain the same as before, the fuel melting estimates can be based on other fission products (preferably Xe-133).

(c) The fuel melting estimates based on Xe, Kr, I, or Cs will not provide for a good estimate in the <10% sub-group. This is due to the fact that these fission products are released in significant quantities during fuel overheating and it would be difficult to determine whether the releases were due to fuel overheating or fuel melting. In the upper sub-groups (i.e. 10-50% or >50% fuel melting), the fuel melting estimates based on Xe, Kr, I or Cs will become more realistic.

(d) In the event the core damage estimate is determined after a containment purge operation, correct the noble gas concentrations to reflect the initial concentrations prior to the purge operation.

(e) The samples taken during the later stages of the accident may provide for a more accurate core damage estimate, than those taken in the early stages of the accident (refer to section II.E.2 for more details).

(f) If the accident has occurred under non-equilibrium core conditions (i.e. few days after refueling), refer to Section VII of this procedure for a more accurate estimate of core damage.

VI.B. PROCEDURE (Continued)

Summary and Conclusions

In conclusion Xe-133 alone can provide for reasonably accurate data for estimating cladding failure and fuel overheating. Xe-133 is abundant both in early and later stages of the accident. Te, Ba, or Ru can provide for estimating fuel melting and the estimates can be further verified by the estimates obtained from Xe-133 and other fission products.

VII. Determination Of Core Damage Under Non-Equilibrium Core Conditions

The estimates of core damage in the previous section were based on assuming a uniform core inventory distribution under equilibrium conditions. Therefore, if accident were to occur at non-equilibrium conditions (i.e. few days after refueling up to 30 days), the core inventory data from Attachment "C" cannot serve as a good approximation of the core inventory. Under non-equilibrium core conditions the following fission products with relatively good half-lives will be utilized for estimating core damage:

Fission Product *	Half-Life (Days)
Xe-133	5.27
I-131	8.05
Te-132	3.24
Ba-140	12.8

* All of these fission products, if released, are abundant and detectable both in early and later stages of the accident.

The 100% core equilibrium inventories from attachment "C" can be corrected to represent non-equilibrium inventories by applying the following inventory correction factor:[1]

$$I_i = \frac{100}{\Sigma_j P_j \left(1-e^{\lambda_i T_j}\right) e^{-\lambda_i T_j^0}}$$

1. The above equation was obtained from reference (5) "Post Accident Sampling guide for Preparation of a Procedure to Estimate Core Damage". USNRC.

VII. Determination of Core Damage Under Non-Equilibrium Core Conditions (Continued)

Where:

I_i = Inventory corection factor for isotopes (i)

P_j = Steady reactor power operated in period j (MWT)
NOTE: In each period, the variation of steady power should be limited to ±10%.

T_j = Duration of operating period j (days)

T_j^o = Time between the end of operating period j and time of reactor shutdown (days)

λ_i = Decay constant of isotope i (days) = $\frac{0.693}{T}$ (days)

The calculated inventory correction factors can then be multiplied by the inventories on Attachment "C" to obtain the new core inventories for estimating core damage under non-equilibrium conditions. The corrected core inventories for Xe-133, I-131, Te-132, or Ba-140 must be applied to the following Sections of this procedure in order to obtain the core damage estimates under non-equilibrium conditions:

- Section VI.B.2.2: Multiply the corrected core inventories by the gap release fractions to obtain a new total gap release activity.

- Section VI.B.3.2: Use the corrected core inventories to obtain percent fuel overheating.

- Section VI.B.4.2: Multiply the corrected core inventories by the nominal meltdown release fractions to obtain a new "total meltdown release activities"

VII. Determination of Core Damage Under Non-Equilibrium Core Conditions (Continued)

The inventory correction factors calculated by the method in the previous page are an approximation and are based on the following assumptions:

- Fission product production and loss by neutron activation are assumed to be negligible.

- The production is assumed to be a linear function of core power.

Therefore, if a more accurate estimate of core inventory is desired; GSE Nuclear Department can be contacted and a new core inventory will be calculated by utilizing ORIGEN2 computer code.[1]

The following parameters will be utilized as input data to ORIGEN2:

- Number of irradiation intervals - Power levels or corrected flux input
- Radius of fuel - Length of fuel - Density of fuel - Enrichment of fuel
- Number of assemblies - number of fuel rods per assembly.

The calculated core inventories for Xe-133, I-131, Te-132, or Ba-140 can be applied to the following Sections VII.B.2.2, VI.B.3.2, or VI.B.4.2 to obtain a more accurate core damage estimate.

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1. ORIGEN2 is an isotope generation and depletion code and it can be accessed by utilizing the "NUCLIB" System available through BOEING computer services.

REFERENCE(2) - NUREG/CR-1237 "BEST ESTIMATE LOCA RADIATION SIGNATURE"

"Table 1. Best-Estimate Fission Product Release Fractions

	Gap Release[a]			Meltdown Release			Oxidation Release			Vaporization Release		
	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit
Noble Gases (Xe, Kr) — Xe^{133}, Kr^{88}, Kr^{85}	0.030 0.030 0.080	0.010 0.020 0.060	0.12 0.04 0.10	0.873	0.485	0.970	0.087	0.078	0.097	0.010	0.010	0.010
Halogens (I, Br)	0.017	0.001	0.20	0.885	0.492	0.983	0.088	0.078	0.098	0.010	0.010	0.010
Alkali Metals (Cs, Rb)	0.050	0.004	0.30	0.760	0.380	0.855	---	---	---	0.190	0.190	0.190
Tellurium Group (Te, Se, Sb)	0.0001	3×10^{-7}	0.04	0.150	0.05	0.250	0.510	0.340	0.680	0.340	0.340	0.340
Noble Metals (Ru,Rh,Pd,Mo,Tc)	---	---	---	0.030	0.01	0.10	0.873	0.776	0.970	0.005	0.001	0.024
Alkaline Earths (Sr, Ba)	1×10^{-6}	3×10^{-9}	0.0004	0.100	0.02	0.20	---	---	---	0.009	0.002	0.045
Rare Earths (Y,La,Ce,Nd,Pr, Eu,Pm,Sm,Np,Pu)	---	---	---	0.003	0.001	0.01	---	---	---	0.010	0.002	0.050
Refractories (Zr, Nb)	---	---	---	0.003	0.001	0.01	---	---	---	---	---	---

FROM TABLE I PAGE II-9 OF THIS REPORT

[a] Note: Recent values of the gap release measured at Oak Ridge National Laboratory (Ref 11) are significantly lower. For the stable and long-lived members of the chemical groups they report 0.0127 for the noble gases, 0.00053 for the halogens, and 0.00025 for the alkali metals.

Attachment "B"

TABLE IV

NORMAL REACTOR COOLANT CONCENTRATIONS ($\frac{\mu Ci}{cc}$)

Isotope	(ρ=1 g/cc) μCi/cc	Isotope	(ρ=1 g/cc) μCi/cc
I-131	1.56 (-2)*	~~Sr-91~~	
I-132	6.97 (-2)	~~Sr-92~~	
I-133	5.77 (-2)	Y-90	$\leq$2.97 (-7)
I-134	9.29 (-2)	~~Y-91~~	
I-135	8.56 (-2)	Zr-95	5.93 (-4)
Ar-41	4.96 (-2)	Nb-95	7.96 (-4)
Kr-85	1.27 (-2)	Zr-97	2.19 (-4)
Kr-85m	5.52 (-2)	Nb-97	2.35 (-4)
Kr-87	5.58 (-2)	Mo-99	4.40 (-4)
Kr-88	7.31 (-2)	Tc-99m	2.15 (-4)
Xe-133	7.19 (-1)	Ru-103	5.35 (-5)
Xe-133m	1.89 (-2)	Sn-117m	2.58 (-5)
Xe-135	2.69 (-1)	Sb-124	1.48 (-5)
H-3	2.64 (-2)	Cs-134	1.45 (-2)
F-18	1.07 (-1)	Cs-136	1.70 (-4)
Na-24	9.81 (-3)	Cs-137	1.88 (-2)
Cr-51	5.91 (-3)	Cs-138	1.14 (-1)
Mn-54	4.26 (-4)	Ba-139	3.75 (-3)
Mn-56	5.86 (-4)	Ba-140	2.21 (-4)
Co-57	3.36 (-5)	La-140	2.54 (-4)
Co-58	1.42 (-2)	Ce-141	2.42 (-5)
Fe-59	3.89 (-4)	Ce-144	2.47 (-5)
Co-60	3.92 (-4)	Pr-144	2.47 (-5)
~~Cu-64~~		Ag-110m	2.64 (-5)
Ni-65	3.70 (-4)	W-187	2.69 (-3)
~~Zn-65~~		Np-239	4.25 (-4)
Sr-89	2.01 (-2)	~~Te-134~~	
Sr-90	$\leq$2.97 (-7)	~~Sm-156~~	

*Numbers in parentheses denote powers of ten

Core Fission Product Inventories (Curies)

From Ref. (6)

Attachment "C"

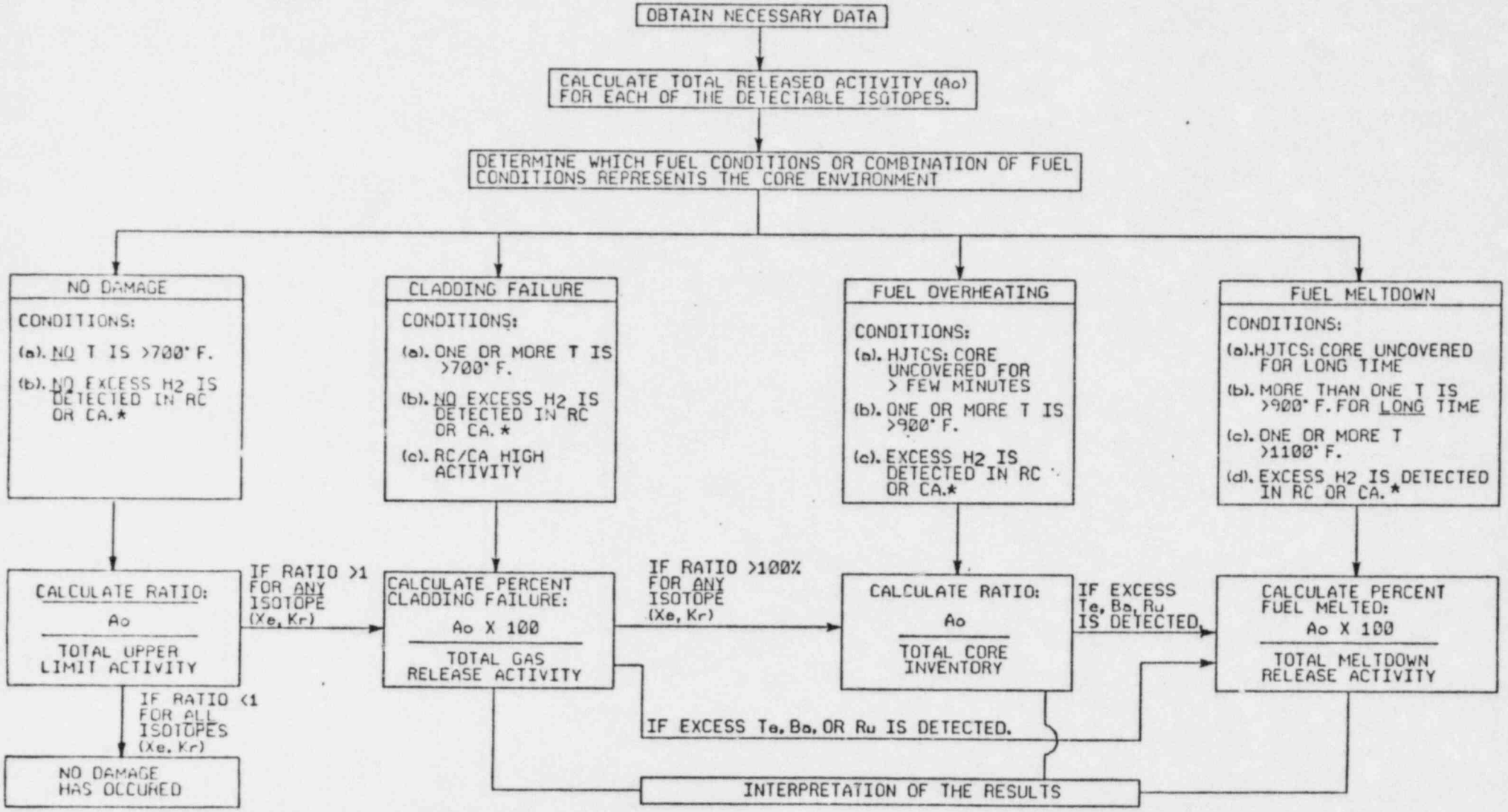
Nuclide	Curies	Nuclide	Curies	Nuclide	Curies
Se-84	7.84(+6)*	Mo-99	7.63(+7)	Sb-135	2.51(+6)
Br-84	8.22(+6)	Tc-99m	6.62(+7)	Te-135	3.34(+7)
As-85	1.37(+6)	Mo-103	6.66(+7)	I-135	7.84(+7)
Se-85	4.85(+6)	Tc-103	6.74(+7)	Xe-135m	1.69(+7)
Br-85	1.05(+7)	Ru-103	6.78(+7)	Xe-135	1.47(+7)
Kr-85m	1.07(+7)	Tc-106	2.76(+7)	Cs-135	1.04(+1)
Kr-85	3.38(+5)	Ru-106	1.89(+7)	Cs-136	2.09(+6)
Se-87	7.71(+6)	Sn-129	4.42(+6)	I-137	3.51(+7)
Br-87	1.70(+7)	Sb-129	1.37(+7)	Xe-137	7.42(+7)
Kr-87	1.96(+7)	Te-129m	3.54(+6)	Cs-137	3.64(+6)
Br-88	1.80(+7)	Te-129	1.29(+7)	Ba-137m	3.46(+2)
Kr-88	2.80(+7)	I-127	6.49(+24)**	I-138	1.77(+7)
Rb-88	2.84(+7)	I-129	1.04(00)	Xe-138	6.78(+7)
Br-89	1.24(+7)	Sn-131	1.23(+7)	Cs-138	7.21(+7)
Kr-89	3.45(+7)	Sb-131	3.38(+7)	Xe-140	3.48(+7)
Rb-89	3.71(+7)	Te-131m	6.28(+6)	Cs-140	6.53(+7)
Sr-89	3.96(+7)	Te-131	3.62(+7)	Ba-140	7.33(+7)
Br-90	7.92(+6)	I-131	4.15(+7)	La-140	7.54(+7)
Kr-90	3.41(+7)	Xe-131m	2.93(+5)	Xe-143	8.13(+5)
Rb-90	3.50(+7)	Sn-132	7.16(+6)	Cs-143	1.41(+7)
Sr-90	2.75(+6)	Sb-132	2.01(+7)	Ba-143	5.60(+7)
Y-90	2.89(+6)	Te-132	5.94(+7)	La-143	6.28(+7)
Kr-91	2.51(+7)	I-132	6.03(+7)	Ce-143	6.32(+7)
Rb-91	4.51(+7)	Sn-133	2.46(+6)	Pr-143	6.24(+7)
Sr-91	4.85(+7)	Sb-133	2.28(+7)	Xe-144	1.79(+5)
Y-91m	2.79(+7)	Te-133m	3.04(+7)	Cs-144	4.26(+6)
Y-91	5.14(+7)	Te-133	4.80(+7)	Ba-144	4.18(+7)
Sr-95	5.14(+7)	I-133	8.43(+7)	La-144	5.52(+7)
Y-95	6.74(+7)	Xe-133	8.43(+7)	Ce-144	5.01(+7)
Zr-95	7.08(+7)	Cs-134	7.46(+6)	Pr-144	5.06(+7)
Nb-95	7.16(+7)	Sb-134	3.96(+6)		
Zr-99	6.95(+7)	Te-134	6.40(+7)		
Nb-99	7.25(+7)	I-134	9.10(+7)		

* Values in parentheses denote powers of ten.
** I-127 is a stable isotope, value is expressed in atoms.

The above core inventory is based on the following conditions:

- 2 fuel cycles at 100% power
- Continuous operation at 1500 MWt
- Equilibrium fuel cycle (equivalent full power days): 432 Days

OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION
UNIT *1
CORE DAMAGE ESTIMATE FLOWCHART



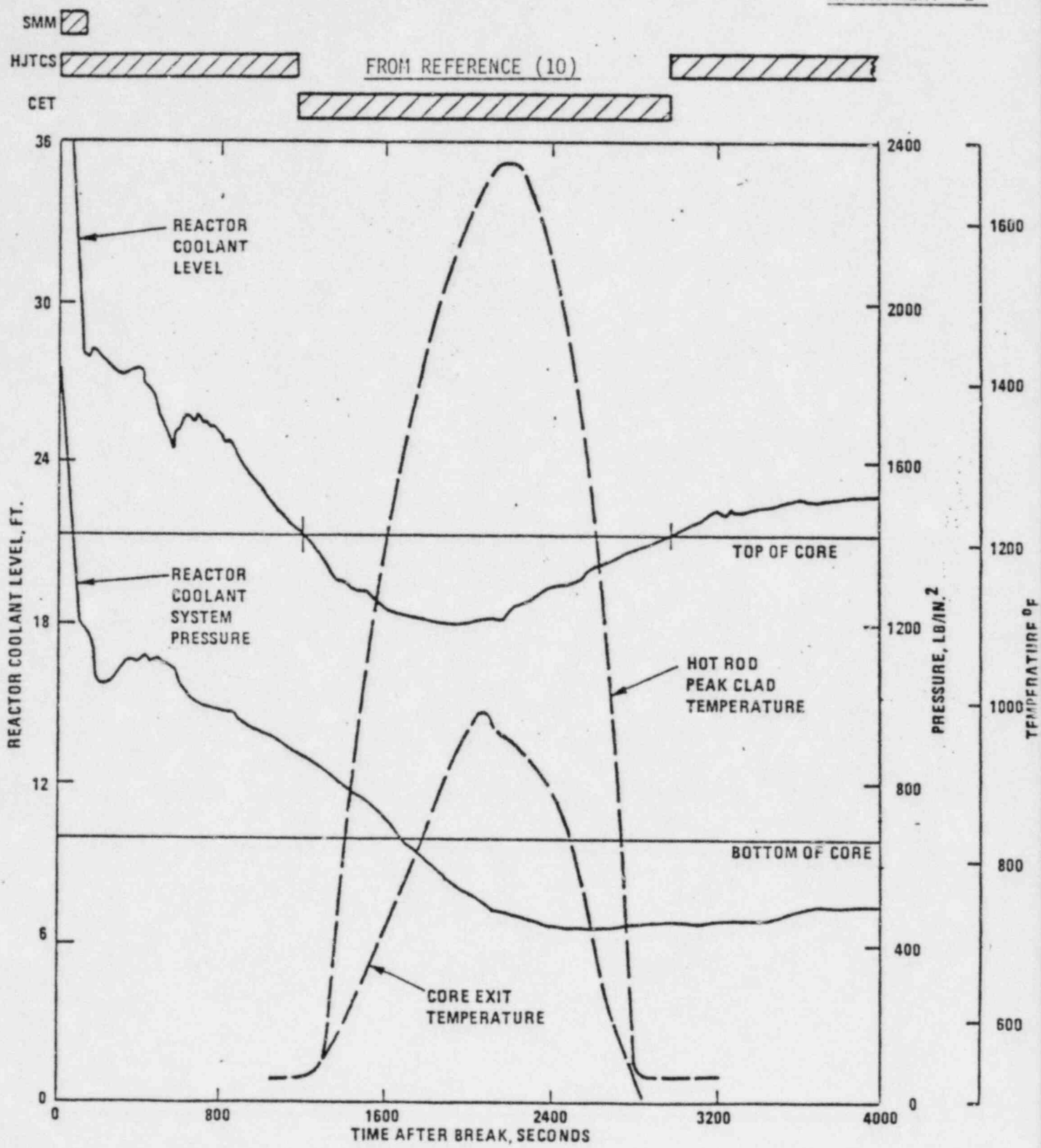
* = See Note at the Bottom of Page 10
T = CORE EXIT THERMOCOUPLE TEMPERATURE READING
CA = CONTAINMENT ATMOSPHERE
RC = REACTOR COOLANT
H_2 = HYDROGEN
HJTCS = HEATED JUNCTION THERMOCOUPLE SYSTEM (IS NOT OPERATIONAL AT PRESENT).

OPPD-GSE NUCLEAR-M. NIROOMAND-RAD 1/10/83

OI-PAP-6-C
ATTACHMENT

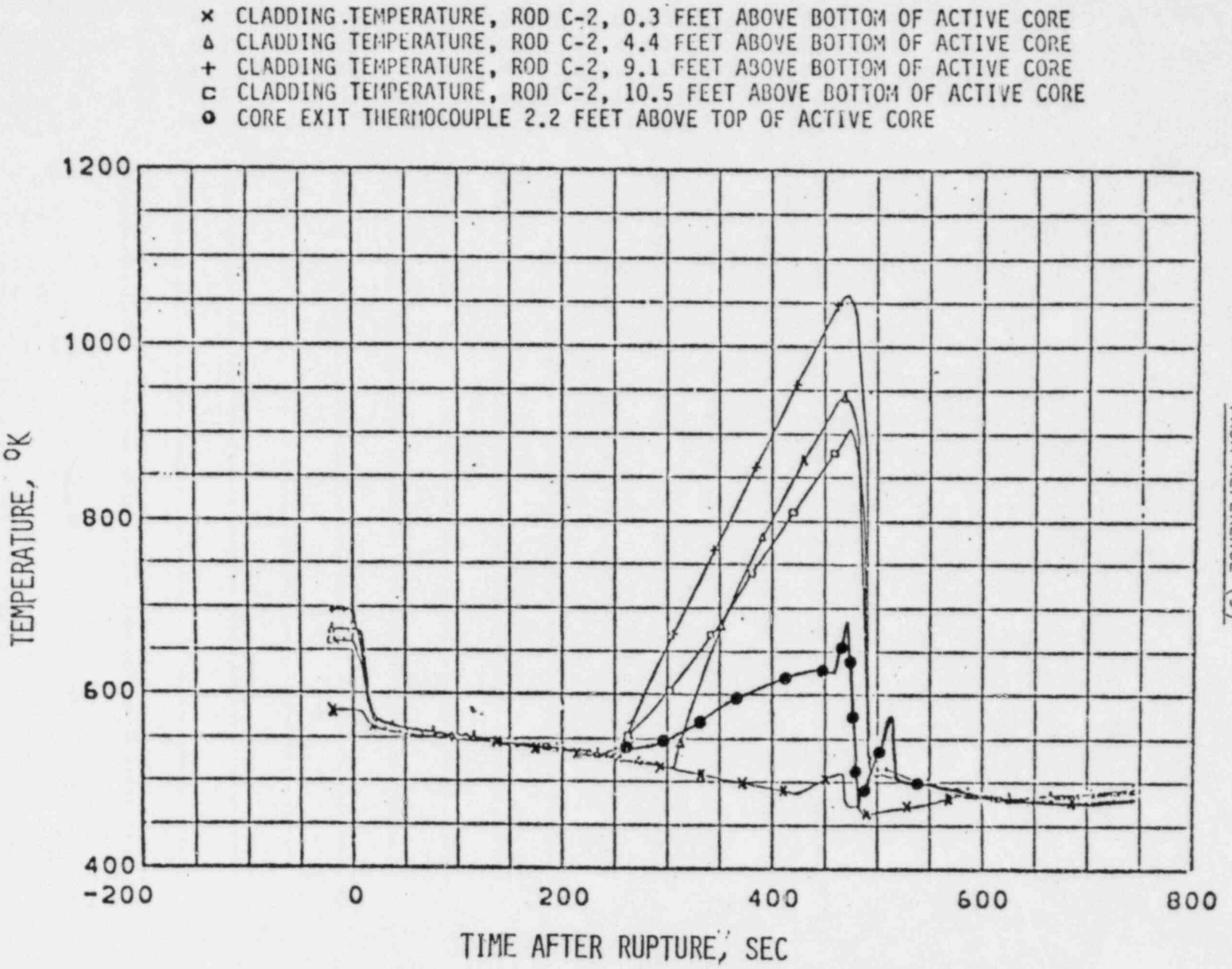
USEFUL OPERATING RANGE OF C-E ICC MONITORING INSTRUMENTS*

ATTACHMENT "E"



* SMM - SUBCOOLED MARGIN MONITOR
HJTCS - HEATED JUNCTION THERMOCOUPLE SYSTEM
CET - CORE EXIT THERMOCOUPLES

POWER SYSTEMS COMBUSTION ENGINEERING, INC	TYPICAL REACTOR COOLANT PARAMETERS FOLLOWING SMALL BREAK LOCA	Figure 1



FROM REFERENCE (9)

FIGURE 4-13

RESPONSE OF CORE EXIT THERMOCOUPLE IN SEMISCALE TEST S-07-10D

DRAFT COPY

ATTACHMENT "F"