



J. Phillip Bayne
Executive Vice President
Nuclear Generation

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IPN-83-97

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
NUREG-0737; Item II.B.3
Post-Accident Sampling System

Dear Sir:

This letter and its Attachments serve to supplement the information contained in our letter dated October 26, 1983, IPN-83-89, regarding the subject NUREG-0737 Item. Attachment I contains a copy of the interim core damage assessment (CDA) procedure. This procedure was recently approved for use at Indian Point 3. A final procedure will be implemented subsequent to Authority review of the Westinghouse Owner's Group generic procedure scheduled for issuance in the first quarter of 1984. Attachment 2 contains additional information requested in the Safety Evaluation Report appended to Mr. S. A. Varga's letter dated August 15, 1983 regarding post-accident sampling system accuracy, range, and sensitivity (Criterion 10) and the potential for plate-out in the sample lines (Criterion 11). This information is being submitted in fulfillment of the commitment made in IPN-83-89.

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

C. M. Welverding

J. P. Bayne
Executive Vice President
Nuclear Generation

cc: attached

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cc: Resident Inspector's Office
Indian Point Unit 3
U. S. Nuclear Regulatory Commission
P. O. Box 66
Buchanan, New York 10511

Attachment 1 to IPN-83-97
Interim Core Damage Assessment Procedure
Item II.B.3 of NUREG-0737

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

CORE DAMAGE ASSESSMENT PROCEDURE

1.0 PURPOSE

To provide a methodology to determine the extent of core damage following a postulated accident based on radionuclide concentrations in reactor coolant (RCS) and containment atmosphere (VC) samples as well as other plant indications and parameters.

2.0 DATA PREREQUISITES

The information described in the following paragraphs will be utilized in assessing the core damage condition. Where information (other than measured concentrations) is not fully available and cannot be obtained in a timely manner, attempts should be made to conservatively estimate those values, with the use of such estimates noted in all discussions and evaluations of results.

2.1 RCS and VC samples have been taken and analyzed (with appropriate isotopic breakdown) in accordance with established post accident sampling procedures. The appropriate source of the liquid sample will depend on the particular accident scenario.

2.2 The following additional information has been obtained:

2.2.1 The reactor coolant temperature at the time the sample is taken.

2.2.2 Temperature of the RCS sample.

2.2.3 Pressure and temperature of both the containment atmosphere (at time of sampling) and the corresponding sample.

2.2.4 The volume of emergency core cooling water injected into the primary system.

2.2.5 The amount of dilution performed by the chemist during the sampling process.

- 2.2.6 The elapsed time from reactor shutdown to sample analysis.
- 2.2.7 Power history data for current cycle and Effective Full Power Days (EFPD) for two previous cycles.

3.0 QUALITATIVE ASSESSMENT OF CORE DAMAGE

Where plant parameters indicate an abnormal plant shutdown has occurred with core cooling jeopardized or interrupted and possible core damage, a preliminary determination of the extent of core damage can be made by qualitatively reviewing the analyzed sample and evaluating it together with other plant parameters and operating data. Based on the following indicators, a general core damage assessment can be made:

3.1 Cladding Damage Indicators

- 3.1.1 The appearance of noble gases (xenons, kryptons), iodines and possibly small amounts of cesium in the reactor coolant without the presence of other fission products is a fair indication that damage is limited to clad failure and possibly, a limited degree of fuel overheating.

3.1.2 Additional indicators of clad damage include:

- 3.1.2.1 Core exit thermocouple temperature readings higher than 650° (for LOCA with normal Engineered Safeguards response, the maximum expected core exit temperatures with natural circulation should be about 620 - 650°F).

NOTE: Thermocouple readings as confirmatory information for core conditions beyond clad damage must be used with caution since coolant conditions (i.e., steaming) can significantly affect thermocouple accuracy. Those readings should therefore be evaluated carefully before use. A review of any available thermocouple trends (log, computer, etc.) will be useful in this evaluation.

- 3.1.2.2 High Range Containment Monitors (when conditions indicate release to containment)

100% clad damage may indicate up to 2.7×10^4 rad/hr on R-25 and R-26

5.1 Application and Determination of Correction Factors

The "reference sample" assumes the following characteristics:

- o A core average burnup of 1050 effective full power days (EFPD) - EOL conditions.
- o Dilution of the liquid source term in Reactor Coolant System volume only (non-line break accident).
- o No additional dilution during the sampling and analysis process.
- o The RCS sample at RCS temperature and pressure (and therefore density)
- o The VC sample at containment temperature and pressure
- o Samples taken at time of shutdown

To correct for deviations from the above assumptions under specific sampling conditions, the following correction factors should be applied as appropriate to the measured activity. The correction factors should be used for both the RCS and VC samples unless otherwise indicated.

5.1.1 Operating Time Correction Factor (P)

Long Lived Isotopes (See Table 2)

This correction factor is applied to correct long lived nuclides for operation for less than EOL conditions assumed in the formulation of the reference "maximum concentration" (MX) values shown in Table 2.

$$P = \frac{1 - e^{-1050\lambda_i}}{(1 - e^{-\lambda_i T_0}) + N_1/193[e^{-\lambda_i t_1}(1 - e^{-\lambda_i T_1})] + N_2/193[e^{-\lambda_i t_2}(1 - e^{-\lambda_i T_2})]}$$

where:

- T_0 = operating time for current cycle (EFPD)
- t_1 = time since end of last cycle (days)
- T_1 = operating time for last cycle (EFPD)
- t_2 = time since end of cycle before last (days)
- T_2 = operating time for cycle before last (EFPD)
- λ_i = decay constant for isotope "i" (days⁻¹)
- N_1 = Number of assemblies from the last cycle currently in core
- N_2 = Number of assemblies from cycle before last cycle currently in core.

Short Lived Isotopes

For short lived isotopes (See Table 2), the operating time correction factor (P) must be calculated for each isotope i:

$$P = \frac{100}{P_j (1 - e^{-\lambda_i T_j}) e^{-\lambda_i t_j}}$$

where:

P_j = steady reactor power in period j (percent)

T_j = duration of period j (days)

t_j = time from end of period j to reactor shutdown (days)

NOTE: This calculation should be made over a period starting from at least five half-lives prior to the present shutdown and ending at the time of the present shutdown.

Table 3 is included as a worksheet, if necessary. One sheet should be used for each short-lived isotope.

5.1.2

Correction for Additional Dilution Provided by Emergency Core Cooling System (ECCS) Volume (ED)
For accidents involving injection of emergency core cooling water into the primary system, the assumed dilution volume for the RCS sample must be corrected. The volumes which may need to be added will depend on the specific accident scenario and may typically include the accumulators, boron injection tank and Refueling Water Storage Tank. The actual volumes can be determined from appropriate level and flow instrumentation.

$$ED = \frac{91,600 + \text{ECCS Volume Injected (gallons)}}{91,600}$$

3.2 Overtemperature Indicators

3.2.1 The presence of an appreciable cesium concentration will be indicative of at least a fuel overheat situation since no substantial quantity of cesiums should be found if core temperatures remain below 2370°F or if the core has not been at least partially uncovered for some extended period of time. In addition, strontium and barium may be present.

3.2.2 High Range Containment Monitors: (When conditions indicate release to containment) R25/R26 would read up to 3.0×10^6 rad/hr for 100% overheat/melt.

3.2.3 Reactor Vessel Level Indicating System (if available) indicates core uncover for an extended period of time.

3.2.4 Significant releases (e.g. greater than 100 uCi/cc) of tellurium, ruthenium and more refractory materials will occur only if the temperature approaches the fuel melting point ($\sim 4500 - 5000^\circ\text{F}$). The presence of ruthenium and tellurium does not "prove" melting, but their absence is a good indicator that melt has not occurred.

3.3 Fuel Melt Indicators

3.3.1 Presence of cerium and lanthanum in fluid samples are generally indicative of fuel melt. Ruthenium and tellurium must also be present (although as stated in 3.2.4, their presence does not prove melt).

3.3.2 High Range Containment Monitors: (when conditions indicate release to containment) R25/R26 would read up to 3.0×10^6 rad/hr for 100% fuel overheat/melt.

4.0 ASSESSMENT OF CLAD DAMAGE FROM HYDROGEN GENERATION FROM ZIRCONIUM WATER REACTION

The percent of the zirconium cladding reacting with water can be estimated using the graph in Figure 1. This assumes that there has been no substantial loss of Hydrogen due to Hydrogen recombiner use.

5.0 ASSESSMENT OF THE CORE DAMAGE FROM SAMPLE ANALYSIS

The following procedural steps address more detailed characterization of core damage based on results of sampling and analysis of RCS and VC samples. The calculations are based on characteristics of a reference sample corrected for actual sampling conditions. Containment atmosphere sampling is of importance only for "line-break" type accidents.

5.1.3 Correction for Dilution During Sampling Process (SD)

The "reference sample" is an undiluted sample. A Sampling Dilution (SD) correction factor must be applied if the chemist further dilutes the drawn sample.

SD = Dilution Ratio
(e.g., SD = 1000 for a 1000: 1 dilution)

5.1.4 Density Correction Factor (R)

This a straightforward correction if the density of the sample differs from that of the RCS or Containment. The correction factor (R) is taken from Table 1 for the RCS sample. Temperature of the RCS Sample is assumed to be 80° F.

For the Containment atmosphere sample this factor can be determined by:

$$R = \frac{(P_{vc}) T_s}{P_s (T_{vc})}$$

where: P_{vc} = Containment Pressure (psia)
P_s = Sample pressure (psia) during analysis
T_{vc} = Containment temperature (°R)
T_s = Sample temperature (°R) during analysis

Sample temperature is assumed to be equal to the VC temperature. Sample pressure is assumed to be atmospheric.

5.1.5 Decay Correction Factor

This corrects the measured fission product concentration (MC) to account for decay from reactor shutdown to the time of analysis. This calculation must be performed for each appropriate isotope.

$$MC_o = MC (e^{.0417 \lambda t})$$

where: MC_o = decay corrected measured concentration
MC = uncorrected measured concentration
t = time (hrs) from shutdown to analysis
λ = decay constant (See Table 2)(days⁻¹)
.0417 = conversion factor (hours to days)

NOTE: The gamma spectrometry computer may be used to back calculate these values.

5.2 Determination of "Adjusted" Measured Concentration

Having determined and calculated all appropriate correction factors, the "adjusted" measured concentration AC is given by:

$$AC_i = (MC_o \cdot P \cdot ED \cdot SD \cdot R)_{RCS} + (MC_o \cdot P \cdot SD \cdot R)_{VC} \frac{(VC \text{ volume})}{(RCS \text{ volume})}$$

where: AC_i = Adjusted total measured concentration for isotope i. This includes both RCS and VC concentration and will be compared to the "reference" sample concentrations (MX's)

RCS = Adjusted Reactor Coolant System concentration

VC = Adjusted Containment atmosphere concentration

$\frac{(VC \text{ volume})}{(RCS \text{ volume})}$ = This term adjusts the containment atmosphere concentration to allow it to be directly added to the RCS concentration.

This concentration should be determined for each of the isotopes Table 2 (or as many as possible). These adjusted concentrations should then be compared to the reference concentrations (MX values) given in Table 2 as follows:

NOTE: A different calculation must be performed for each isotope, and it is expected that different fractions will be calculated. Engineering judgement must be used on the aggregate results to resolve any discrepancies.

5.3 Clad Damage Estimate

To estimate the fraction of clad failure damage:

- 5.3.1 Determine the maximum clad failure concentration, MX_{CFi} from Table 2, Column I for each selected isotope "i".

- 5.3.2 Calculate a Clad Failure Fraction (CFF for each appropriate isotope)

$$CFF = \frac{AC_i}{MX_{CFi}}$$

where AC_i = adjusted concentration for isotope "i"

- 5.3.3 Criterion for Cladding Failure

If $CFF < .01$ there is little or no cladding damage. Verify that MC is within normal concentration measurements.

If $.01 \leq CFF \leq 1.0$ there is some degree of clad damage. To estimate percentage of damage, multiply CFF by 100.

If $CFF > 1.0$ proceed to next paragraph to estimate damage.

5.4 Fuel Overtemperature Estimate

To estimate the degree of fuel overtemperature:

- 5.4.1 Determine the maximum fuel overtemperature concentration, MX_{FOi} , from Table 2, Column II for each selected isotope "i".

- 5.4.2 Calculate a Fuel Overtemperature Fraction (FOF):

$$FOF = \frac{AC_i}{MX_{FOi}}$$

where AC_i = adjusted concentration for isotope "i"

- 5.4.3 To estimate the percentage of the core in an overtemperature conditions, multiply FOF by 100. If FOF is greater than 1.0, proceed to the next paragraph to evaluate for fuel melt.

5.5 Fuel Melt Estimate

To estimate the fraction of the core which may have experienced core melt:

- 5.5.1 Determine the maximum fuel melt concentration, MX_{FMi} from Table 2, Column III for each selected isotope "i".

5.5.2 Calculate f Fuel Melt Fraction (FMF):

$$FMF = \frac{AC_i}{\sum_{i=1}^n FM_i}$$

where AC_i adjusted concentration for isotope "i"

5.5.3 To estimate the percentage of fuel melt, multiply FMF by 100. Note the presence of ruthenium and tellurium as per 3.3.1.

6.0 CAUTIONS AND RECOMMENDATIONS

If conflicting data exists based on criteria guidance given in Sections 3 and 4, reanalyze all indications. For example; the analysis yields results which indicate core melt. The core thermocouple readings show the core is not hot enough to have melted. The sample analysis should therefore be re-examined to determine if the isotopics chosen are appropriate.

It should also be noted that the above evaluations address an assumed uniform distribution of core damage. Since the degree of damage is likely to vary within the core, calculations for CFF, FOF and FMF should all be performed to provide a better understanding and perspective with regard to the potential existence of a mixture of fuel damage conditions. Variations in core exit thermocouple readings may supply additional supporting information.

Finally it should also be noted that the results determined using this procedure are subject to inaccuracies related to physical processes occurring in the RCS and VC in the post accident condition. These processes, including among others, adsorption, sedimentation and plateout, may remove a significant amount of source term from both the atmosphere and fluid from which the sample is drawn. Although inclusion of these processes is beyond the scope of this evaluation, their potential effect should be borne in mind.

7.0 REFERENCES

- 7.1 Reactor Safety Study (WASH-1400)
- 7.2 Three Mile Island - Report to the Commissioners and to the Public ("Rogovin Report")
- 7.3 W Mitigating Core Damage Training Manual, Sections 6 & 8
- 7.4 Westinghouse Radiation Analysis Design Manual
- 7.5 ORIGEN (Isotope generation and depletion code)

TABLE 1

DENSITY CORRECTION FACTORS (R)

	<u>RCS Sample Temperature ($^{\circ}$F)</u>			
	70	80	90	100
100	.995	.996	.998	1.0
150	.982	.983	.985	.987
200	.965	.966	.968	.970
250	.944	.945	.947	.949
300	.920	.921	.923	.924
350	.892	.894	.895	.897
400	.861	.862	.864	.865
450	.826	.827	.828	.830
<u>RCS Tem-</u> <u>perature</u> <u>at time</u> <u>of sample</u> <u>($^{\circ}$F)</u> 500	.787	.788	.789	.791
550	.737	.738	.739	.742
560	.726	.727	.728	.730
570	.716	.717	.718	.720
580	.704	.705	.706	.707
590	.692	.693	.694	.695
600	.680	.681	.682	.683
620	.650	.651	.652	.653
640	.617	.618	.619	.620
660	.579	.580	.582	.582
680	.528	.529	.530	.531
700	.438	.438	.439	.441

TABLE 2

Short Lived Isotopes

<u>Isotope</u>	<u>Halflife</u>	<u>$\lambda(\text{day}^{-1})$</u>	<u>I MX-CF (Ci/cc)</u>	<u>II MX-FO (Ci/cc)</u>	<u>III MX-FM (Ci/cc)</u>
Mo-99	66.0h	.252	N/A	4.8E-3	1.4E-2
I-131	8.04d	.0862	4.7E-3	1.4E-1	2.5E-1
I-133	20.8h	.800	8.6E-3	2.5E-1	4.5E-1
Xe-133	5.25d	.132	1.5E-2	2.5E-1	4.5E-1
Te-132	78.2h	.213	N/A	1.9E-2	5.7E-2
Ru-105	4.44h	3.75	N/A	3.3E-5	1.0E-3
Ba-140	12.79d	.0542	N/A	9.0E-3	4.5E-2
La-140	40.22h	.414	N/A	N/A	1.4E-3

Long Lived Isotopes

Kr-85	10.72y	1.77E-4	9.3E-5	1.5E-3	2.8E-3
Cs-137	30.17y	6.09E-5	1.6E-3	1.3E-2	2.6E-2
Ce-141	32.5d	2.13E-2	N/A	N/A	1.2E-3
Sr-89	50.6d	1.37E-2	N/A	3.6E-3	1.8E-2
Sr-90	28.6y	6.64E-5	N/A	4.2E-4	2.1E-3
Ru-103	39.4d	1.76E-2	N/A	3.7E-5	1.1E-3
Ru-106	368d	1.88E-3	N/A	1.9E-5	5.5E-4
Ce-144	284d	2.44E-3	N/A	N/A	9.3E-4

N/A = Not Applicable

TABLE 3
 CALCULATION OPERATING HISTORY CORRECTION FOR ISOTOPE i

Isotope: _____
 Half Life: _____
 $\lambda =$ _____ day^{-1}

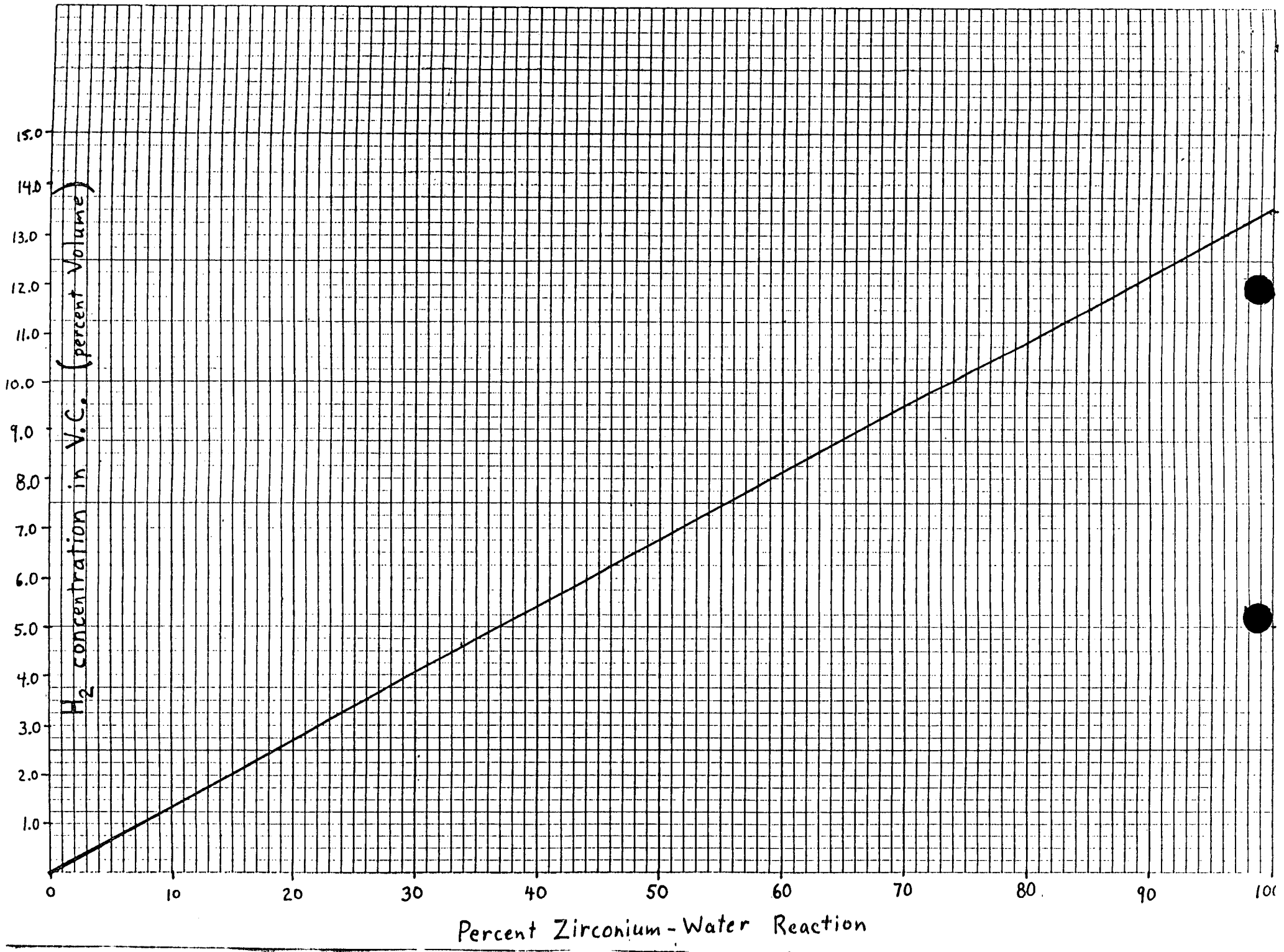
Period	Power Level (%)	Duration (days)	(Days) Decay Time	$(1 - e^{-\lambda_i T_j})$	$(e^{-\lambda t_j})$	$P_j (1 - e^{-\lambda_i T_j}) (e^{-\lambda t_j})$
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						

$\Sigma =$

P_j = steady reactor power level (percent)
 λ_i = decay constant for isotope i (day^{-1})
 T_j = time at power level P_j (days)
 t_j = time since end of T_j to reactor shutdown (days)

$$\frac{100\%}{P_j (1 - e^{-\lambda_i T_j}) (e^{-\lambda t_j})} =$$

Figure 1



Attachment 2 to IPN-83-97
Supplemental Information Regarding
Item II.B.3 of NUREG-0737

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

Supplemental Information Regarding NRC Criterion/Clarification
No. 10

Per the commitment made in the Authority's letter dated October 26, 1983, accuracy, range, and sensitivity information is being provided for the existing post-accident sampling system at Indian Point 3. As stated in our October 26, 1983 letter, the post-accident sampling system instrumentation will be reviewed as part of the Authority's review of Regulatory Guide 1.97, Rev. 2. A summary of existing instrumentation ranges and accuracies follows.

<u>Analysis</u>	<u>Accuracy</u>	<u>Range</u>
Gross Activity	Factor of 2	<10 uCi/ml to 10 Ci/ml
Boron	15%	130ppm to 10000ppm in RCS
Chloride	40%	40ppb to 20ppm in RCS
Hydrogen	21%	< 5cc/kg to 200cc/kg in RCS
Oxygen	10% (for >.5ppm) ±50ppb (for <.5ppm)	<1ppb to 20ppm

Supplemental Information Regarding NRC Criterion/Clarification
No. 11

The Authority will address the issue of plate-out in the containment air sample lines by conducting a test subsequent to the unit's return to service, scheduled for February 1984. A followup report will be submitted to the NRC by the end of the second quarter of 1984.