



Long
Island
Power
Authority

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APR 15 1993

LSNRC-2045

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

ATTN: Mr. Robert Bernero, Director
Office of Nuclear Material
Safety and Safeguards

Additional Information and Clarification
as Requested by Staff Regarding the
Decommissioning Project Termination Survey Plan
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

- Ref: (1) Long Island Power Authority Letter LSNRC-2014 dated December 2, 1992, subject: Termination Survey Plan.
(2) Long Island Power Authority Letter LSNRC-2022 dated January 8, 1993, subject: Response to Staff Comments on Decommissioning Project Termination Survey Plan.
(3) U. S. Nuclear Regulatory Commission (NRC) Letter dated December 16, 1992 to Long Island Power Authority (L.M. Hill) from C. L. Pittiglio; subject: Review and Comments on Termination Survey Plan.

Gentlemen:

Reference (1) submitted Long Island Power Authority's (LIPA) Termination Survey Plan for your review and approval. Reference (2) was submitted by LIPA to answer the NRC Staff's reference (3) letter, that commented on LIPA's Termination Survey Plan.

Subsequent to the issuance of reference (2), additional information and clarification was requested in a telecon between your Mr. Dave Fauver and our Mr. L. Britt. Several discussions ensued regarding a number of issues, involving transmittals by telecopy of proposed resolutions.

The first of these telecopies, addressing the issue of accounting for iron-55 in post-decommissioning residual contamination at Shoreham, was sent by LIPA on February 5, 1993, and is provided as Enclosure 1 to this letter. The approach described therein was subsequently deemed not acceptable to the NRC. A second

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telecopy dated February 25, 1993 (see Enclosure 2 to this letter) again addressed this issue, and was subsequently deemed acceptable to the NRC in principle, with further discussion of specific details to be conducted. These discussions were held during several follow-up telecons, which addressed a number of other issues as well, and resulted in the generation and transmittal of a third telecopy on March 11, 1993. It is LIPA's understanding that the information provided in the March 11, 1993 telecopy constitutes an acceptable basis for final resolution of all NRC Staff comments on the Shoreham Termination Survey Plan. Enclosure 3 to this letter provides the formal transmittal of this information. (Please note that minor editorial clarifications were made in this enclosure to ensure consistent format and use of terminology throughout all responses; that a more conservative, i.e., shorter, decay time has been applied to the iron-55 adjustment factor resulting in a slightly higher adjustment value, and that one of the parameters in response No. 4 has been relabelled. These changes were acknowledged in discussions between LIPA personnel and Dave Fauver).

With regard to the first three comments and responses in Enclosure 3, however, LIPA believes that the relevant subject, i.e., accounting for the potential presence of iron-55 in post-decommissioning residual contamination at Shoreham, merits some additional perspective beyond the technical details contained in these responses. LIPA believes that the approach to be taken to account for iron-55 will conservatively overstate the amount of this isotope, as well as the total amount of all residual contamination, remaining at Shoreham following the completion of decommissioning.

Because of the limitations associated with currently available instrumentation, it would be wholly impractical to physically survey the remaining Shoreham systems and structures at the sensitivities needed to directly account for iron-55. Thus, the approach to be taken will be an algebraic approach in which an extra factor for iron-55 is added, on a sliding scale, to all residual contamination measurements where the presence of any level of the dominant isotope, cobalt-60, is indicated. This approach will be taken regardless of whether any iron-55 is actually known to be present along with the cobalt-60 at any location, and will thus be conservative in terms of the number of locations, where iron-55 is reported. LIPA further believes that the factor to be used for the iron-55 adjustment is itself conservative.

It is also well known that the potential biological effects per curie of iron-55, when compared with an equal amount of cobalt-60, is small. This is based on the fact that a curie of iron-55 decays with much lower levels of energy release than does a curie of cobalt-60. However, the residual contamination release

criteria and the reported residual contamination levels will not account for this difference in relative biological effects. Thus, not only will the amount of residual contamination remaining at Shoreham be overstated, the residual biological hazard, while acceptably small, will be even more overstated.

Notwithstanding the above, LIPA is prepared to address this matter in the manner described. This approach is workable for Shoreham due to its highly unique circumstances, i.e., the brief history of low-power plant operation and the consequent generation of a limited number of "difficult-to-measure" isotopes in small amounts.

Lastly, I would also like to take this opportunity to thank the members of your Staff and the Region I Staff, in particular Messrs. Fauver and Nimitz, for the extra efforts they put forth in accommodating the project schedule for the Main Turbine Termination Survey during the week of February 22, 1993, in spite of the uncooperative weather. We look forward to working closely with your Staff in the coming months to bring the first commercial decommissioning project to a successful license termination by mid-1994.

Please do not hesitate to contact me if you have any questions or require any additional information in this matter.



L. M. Hill
Resident Manager

LFB/ab

cc: L. Bell
C. L. Pittiglio
T. T. Martin
R. Nimitz

ENCLOSURE 1

TO

LSNRC-2045

Shoreham Decommissioning Surface Contamination Guidelines:
Adjustment to Account for Iron-55

The Long Island Power Authority has determined that Fe-55, which decays by electron capture and Co-60 are potentially present in post-decommissioning contaminants at Shoreham, (Ref. 1). The Draft NUREG/CR-5849 (Ref.2) indicates that when multiple radionuclides are present, development of site-specific guidelines may be required. Individual radionuclides are considered to be significant contaminants if, *at the time of license termination*, they contribute greater than 10 % of the total effective dose equivalent from all contaminants, or are present at concentrations which exceed 10 % of their respective guideline values, (NUREG/CR-5849 Appendix A).

LIPA has concluded that under a strict interpretation of the NUREG/CR-5849 criteria, Fe-55 is considered to be a significant contaminant. Therefore, adjustments to the guideline values for average total and removable surface contamination are made.

The adjustments are made using the method of NUREG/CR-5849 with the additional incorporation of a factor which includes the relative dose effectiveness of Fe-55 to Co-60. The resulting adjusted limits for Co-60 are:

4985 dpm/100 cm² for total surface contamination averaged over an area of 1 m² or less, and

997 dpm/100 cm² for removable surface contamination.

The details of the assumptions and the calculation are contained in the Attachment.

Attachment

Supporting Evaluation and Calculations for Adjustment of Guideline Values

1.0 DETERMINATION THAT Fe-55 IS A SIGNIFICANT CONTAMINANT

In determining if Fe-55 is a significant contaminant, the possibility that it could contribute greater than 10 % of the total dose at Shoreham is first considered. An upper bound estimate of the relative contribution of Fe-55 to the total dose from a mixture of Co-60 and Fe-55 can be obtained from an examination of dose factors for the two radionuclides. The following table is obtained from dose factors published in NUREG/CR-5512, Appendix H, (Ref.3).

Table 1

Dose Factors for Fe-55 and Co-60

Exposure Mode > Units >	Ingestion (Sv/Bq)	Inhalation (Sv/Bq)	Surface (Sv/d) per (Bq/m ²)	Soil (15cm) (Sv/d) per (Bq/m ²)
Fe-55	1.64E-10	7.26E-10	0.0	0.0
Co-60	7.28E-9	5.91E-8	2.03E-10	6.26E-12
Fe-55/Co-60*	0.02	0.01	0.0	0.0

* Ratio of Fe-55 and Co-60 dose factors, (unitless).

From the data presented in row four of the table, it is seen that the largest possible relative dose contribution from Fe-55 is 2 % of the total dose, (when Fe-55 and Co-60 are present in equal concentrations). This could occur when ingestion is the dominant exposure pathway. However, typical reactor decommissioning generic dose-pathway calculation results are dominated by the external exposure from surface and soil contamination. In such cases, the relative dose contribution of Fe-55 from all pathways combined is typically on the order of 10^{-4} or less for equal residual contamination activity concentrations of Fe-55 and Co-60.

Hence, it can be concluded that Fe-55 is clearly not a significant contaminant based upon its relative contribution to the total dose. It would not become significant, i.e., contribute more than 10 % of the total dose, even under the most conservative exposure scenario, unless its concentration in residual contamination were at least 5 times that of Co-60.

Next, the ratio of Fe-55 to total activity concentrations (Fe-55 + Co-60) is examined against the NUREG/CR-5849 10 % criterion. An evaluation has been performed based upon radiochemical analysis of samples obtained from Shoreham piping corrosion product deposits in early 1991. This evaluation resulted in an estimated Fe-55:Co-60 ratio of 0.2 (Ref 1). This ratio corresponds to an Fe-55 contribution of approximately 17 % of the total activity. If the relative activity ratio is adjusted for radioactive decay to correspond to the time of license termination (assumed to be mid-to-late 1994), the Fe-55:Co-60 ratio is reduced to 0.14 and the Fe-55 activity is reduced to 12 % of the total, (see Section 2.0 of this Attachment for a summary of the calculation). The Fe-55 activity fraction of 12 % is only slightly above the 10 % criterion contained in NUREG/CR-5849. However, under a strict interpretation of the activity fraction criterion, Fe-55 is determined to be a significant contaminant.

2.0 RADIOACTIVE DECAY CORRECTION OF Fe-55/Co-60 RATIO

The estimated ratio of Fe-55/Co-60 based upon samples collected from the Shoreham Reactor Water Cleanup, (RWCU) system in early 1991 is 0.2 to 1, (Ref. 1). License termination is projected to occur in mid 1994. Decay of Fe-55, ($T_{1/2} = 2.7$ yr) and Co-60, ($T_{1/2} = 5.27$ yr) by three years reduces the terms in the ratio to 0.09 and 0.67 respectively. This translates to a Fe-55/Co-60 ratio of 0.14 with Fe-55 contributing 12 % of the total Fe-55 plus Co-60 activity.

3.0 ADJUSTMENT OF GUIDELINE VALUES

The method for adjusting the guideline value for a specific nuclide present in a mixture is based on Equation (A-1) of NUREG/CR-5849, Appendix A. It states that the sum of the ratios of the concentration of each radionuclide to its respective guideline must not exceed 1.

Applying the "sum of fractions" method to a mixture of Fe-55 and Co-60, the equation becomes:

$$\frac{C_{Fe}}{G} + \frac{C_{Co}}{G} \leq 1 \quad (1)$$

where: C_{Fe} = surface activity concentration of Fe-55,

C_{Co} = surface activity concentration of Co-60, and

G = the guideline value for surface contamination activity concentration. Note that G is equal for Fe-55 and Co-60, as they are both in the same group in Regulatory Guide 1.86 Table 1. In this discussion, all the variables are in units of dpm/100 cm^2 .

The adjusted guideline value for Co-60 which corresponds to a total surface contamination level of G dpm/100 cm^2 is obtained by substituting $C_{Fe} = fC_{Co}$ into equation (1) and solving for C_{Co} . The term $f = C_{Fe}/C_{Co}$, the ratio of Fe-55 to Co-60 activity. This results in the following expression:

$$C_{Co} = \frac{G}{1+f} \quad (2)$$

The method used by LIPA to obtain the adjusted guideline value for Co-60 incorporates an adjustment of f , the ratio of Fe-55/Co-60, to account for the relative biological significance of Fe-55. Using this approach, f in equation (2), is defined as the product: $f = f_a f_r$, where f_a = the activity ratio as defined above, and f_r = the risk ratio. The value assigned to f_r is the ratio of dose ingestion dose factors, 0.02 from Table 1. Hence $f = (0.14)(0.02)$, or 0.003.

The adjusted guideline values for Co-60 are now obtained using equation (2):

for total surface contamination:

$$C_{Co} = 5000/1.003 = 4985 \text{ dpm/100 } cm^2, \text{ and}$$

for removable surface contamination, ($G = 1000$ dpm/100 cm^2):

$$C_{Co} = 1000/1.003 = 997 \text{ dpm/100 } cm^2.$$

4.0 REFERENCES

1. Long Island Power Authority, Letter to U. S. Nuclear Regulatory Commission, Response to Staff Comments on Decommissioning Project Termination Survey Plan, Shoreham Nuclear Power Station - Unit 1 Docket No. 50-322, LSNRC-2022, January 8, 1993.
2. J. D. Berger, "Manual for Conducting Radiological Surveys in Support of License Termination", NUREG/CR-5849, Draft, June, 1992.
3. W. E. Kennedy, et. al., "Residual Radioactive Contamination from Decommissioning", NUREG/CR-5512, Vol. 1, Final Report, October, 1992.

ENCLOSURE 2

TO

LSNRC-2045

LIPA proposes to account for the potential presence of iron-55 in post-decommissioning residual contamination at Shoreham as described in the following discussion:

Rather than reduce the release criteria limits as indicated in draft NUREG/CR-5849 by an amount based on the fractional proportion of iron-55 to cobalt-60, LIPA would perform an upward adjustment of beta-gamma direct and beta-gamma smear measurement results by a factor based on the iron-55/cobalt-60 ratio. This approach would meet the intent of the draft NUREG by providing a numerical method of accounting for iron-55 that is not based on any pathway/dose analysis rationale, while eliminating a major practical dilemma with the NUREG approach. This dilemma relates to the established scan survey sensitivity requirements which are based on percentages of the release criteria. A reduction of the release criteria would necessarily result in a reduction in the minimum scan sensitivity requirements; a reduction which cannot readily be achieved. For Shoreham, such a reduction would bring the scan sensitivity requirement to a level which would effectively eliminate the ability to scan at any reasonable speed. Adjusting the measurement data by mathematical treatment as proposed herein avoids this problem, but accomplishes the same end.

The factor by which beta-gamma measurements would be adjusted is based on the ratio of iron-55 to cobalt-60 derived from samples taken from the Reactor Water Clean Up system, as corrected for decay between the time the samples were taken and the approximate time of expected license termination (assumed to be mid-1994). The derivation of this ratio and decay correction is attached. The data adjustment would be performed using equation (1) below. Scan survey instrument sensitivities and action levels as specified in Table 4.2 and Appendix A, respectively, of the Termination Survey Plan will remain numerically the same, and will be assessed with instruments as stated in Table 4.1.

Surface beta-gamma contamination measurements (direct and removable) are first converted to dpm/100 cm² in accordance with the method described in the Shoreham Termination Survey Plan. After the measurements are converted to dpm/100 cm², a further adjustment is made to account for the possible presence of Fe-55. The following equation is used:

$$dpm_{adj} = dpm + |dpm| \cdot f \quad (1)$$

where:

dpm^{adj} = surface contamination in dpm/100 cm² adjusted to account for Fe-55, i.e., Co-60 + Fe-55 activity,

dpm = uncorrected dpm/100 cm²,

f = adjustment factor, the ratio of Fe-55:Co-60

$|dpm|$ = the absolute value of the uncorrected dpm/100 cm².

The adjusted values dpm_{adj} are the reporting units used for comparison with release criteria.

11-Feb-93

Shoreham RWCU Smear Sample Study

Filter #	Co-60		Fe-55		Fe/Co	
	$\mu\text{Ci/smear}$	std dev	$\mu\text{Ci/smear}$	std dev	ratio	std dev
1	2.08E-02	$\pm 3.17\text{E-04}$	1.12E-02	$\pm 5.64\text{E-04}$	0.54	± 0.028
2	2.46E-02	$\pm 3.46\text{E-04}$	4.33E-03	$\pm 2.16\text{E-04}$	0.18	± 0.009
3	2.10E-02	$\pm 4.83\text{E-04}$	4.52E-03	$\pm 2.26\text{E-04}$	0.22	± 0.012
4	4.30E-03	$\pm 1.43\text{E-04}$	1.96E-03	$\pm 9.80\text{E-05}$	0.46	± 0.027
5	8.29E-03	$\pm 1.96\text{E-04}$	3.64E-03	$\pm 1.82\text{E-04}$	0.44	± 0.024
6	2.33E-02	$\pm 5.07\text{E-04}$	7.02E-03	$\pm 3.50\text{E-04}$	0.30	± 0.016
7	1.42E-02	$\pm 3.95\text{E-04}$	2.53E-03	$\pm 1.26\text{E-04}$	0.18	± 0.010
8	2.26E-02	$\pm 5.01\text{E-04}$	5.52E-03	$\pm 2.76\text{E-04}$	0.24	± 0.013
9	1.98E-02	$\pm 4.69\text{E-04}$	2.62E-03	$\pm 1.31\text{E-04}$	0.13	± 0.007

Regression Output: nine smear samples

Constant 0.0012

Std Err of Y Est 0.0026

R Squared 0.2617

No. of Observations 9

Degrees of Freedom 7

X Coefficient(s) 0.2057

Std Err of Coef. 0.1306

Decay of Fe-55/Co-60 ratio from Jan 1991 to June 1994

	Jan 1990 ratio	decayed to June 1994	June 1994 ratio	percent Co + Fe
Fe-55	0.2057	0.084	0.133	11.7
Co-60	1	0.631		

ENCLOSURE 3

TO

LSNRC-2045

Comment No. 1

NRC Comment: LIPA should use the unweighted mean of the Fe-55/Co-60 ratios from the nine (9) Reactor Water Cleanup (RWCU) system samples to determine the "best" estimate of the Fe-55/Co-60 ratio.

LIPA Response: LIPA agrees to use the unweighted mean of the Fe-55/Co-60 ratios from the nine (9) RWCU samples to represent the "best" estimate of the Fe-55/Co-60 ratio in the Shoreham facility residual contamination. This mean value is 0.30. LIPA will take into account radioactive decay from January, 1991, which is the date of sample analysis, to March, 1994, the earliest estimated date of license termination, in order to establish the actual ratio to be applied to termination survey measurements. Thus, when the ratio is corrected for isotopic decay, the mean value becomes 0.20.

Comment No. 2

NRC Comment: Provide the method to be used to adjust individual measurements to account for the undetected presence of Fe-55.

LIPA Response: LIPA will account for the presence of Fe-55 by adjusting individual surface contamination measurements. Measurements which are statistically above background levels will be adjusted. Adjustments for the presence of Fe-55 are not made for measurements which are indistinguishable from background, (see response to Comment No. 4 for explanation of the method used to distinguish measurements from background). When a measurement is not statistically different from background there is no confidence that contaminating Co-60 is present, therefore there is no justification for making the Fe-55 adjustment. Measurements above background are assumed to be due to the presence of Co-60, and thus the adjustment for Fe-55 is appropriate.

For individual measurements which indicate the presence of Co-60, adjustment for the undetected presence of Fe-55 is made by application of the following equation.

$$dpm_{adj} = \frac{f (gcpm - bcpm)}{E(\frac{A}{100})}$$

where:

- dpm_{adj} = surface activity concentration adjusted to include Fe-55 [dpm/100 cm²],
- $gcpm$ = detector reading in gross counts per minute, (assumed to be entirely due to Co-60 activity),
- f = numerical factor; $1 + \text{Fe-55/Co-60 ratio}$ [1.20],
- $bcpm$ = detector background in counts per minute,
- E = detector efficiency in counts per disintegration, and
- A = detector sensitive area in Cm².

Comment No. 3

NRC Comment: Provide justification for the conclusion that the 9 samples from the SNPS Reactor Water Cleanup (RWCU) system provide a representative ratio to indicate the Fe-55/Co-60 ratio in the Shoreham facility residual contamination.

LIPA Response: As described in Section 3.2 of the LIPA Termination Survey Plan, the Shoreham Characterization Study estimated an Fe-55/Co-60 ratio of approximately 2.5 based on an activation analysis of the materials of construction in the reactor pressure vessel and vessel internals. However, since the majority of these materials have been removed in accordance with the Shoreham Decommissioning Plan, the ratio provided by the activation analysis does not accurately reflect the ratio of Fe-55/Co-60 which might actually exist in residual contamination at the facility. This is primarily justified by the physical measurements which were performed and whose results are presented in the RWCU sample analyses (see Comment No. 1). The intent of these sample analyses was to measure the ratio of Fe-55/Co-60 in an environment closer to what could be expected in residual contamination at the facility. The results of this analysis are explained by considering the following:

1) The calculated Fe-55/Co-60 ratio considers a stainless steel sample irradiated by "out of the core" neutrons, in regions where the thermal neutron flux is greater than the epithermal flux. While this method is reasonable for activation profiles of the core shroud and reactor vessel inner clad, it fails to recognize that most activated crud is formed in the region of the reactor core (i.e., on fuel pin surfaces) before it is dispersed elsewhere. In the core region, the epithermal flux may be higher than the thermal flux. If such differences in neutron energy were accounted for the calculated Fe-55/Co-60 ratio for activated materials internal to the reactor vessel could be as low as 0.39.

2) The calculated Fe-55/Co-60 ratio is largely controlled by the amount of cobalt in the ferrous materials which comprise the reactor pressure vessel or vessel internal components. However, it must be recognized that the predominant source of residual contamination in the Shoreham facility is out-of-core corrosion product deposits which have been distributed as surface contamination within piping systems. This corrosion product deposit activity composition is largely determined by:

a) the composition of materials which deposit on the fuel,

b) the subsequent release of activated corrosion products from fuel surfaces, transport by the coolant, and deposition on out-of-core surfaces.

3) Numerous chemistry effects must be considered in evaluating the material composition of activated crud buildup. These include:

a) Iron ions are more tenaciously held on ion exchange resins than cobalt ions.

b) Cobalt oxide is more soluble than iron oxide.

c) The concentration of iron in the reactor system is such that, when in solution, it will probably be in particulate form.

d) Crud buildup on fuel surfaces at power is primarily iron oxide with cobalt ions adsorbed onto the crud surface. With the reactor shut down, cobalt tends to leave in soluble form, in contrast with the insoluble iron crud or particulates on the fuel.

e) As the initially amorphous iron particulate ages, its surface loses the adsorptive capacity for ions such as cobalt.

f) The RWCU system filter/demineralizers remove cobalt less effectively than iron. These filter/demineralizers are most effective in removing crud and particulate which is predominantly iron. Furthermore, as previously stated, the resin removes iron ions more effectively than cobalt ions.

With the above considerations, it is reasonable to conclude the Fe-55/Co-60 ratio in out-of-core deposits should be smaller than predicted by a purely stoichiometric model.

Comment No. 4

NRC Comment: Provide an explanation of the method used to identify surface contamination measurement results which are "positive", i.e., above background. Also, it is requested that these measurements be "flagged" or otherwise identified in the data report for each survey unit.

LIPA Response: LIPA will use the following decision rule for determining whether an individual measurement is considered positive evidence of contamination:

if the measured value of contamination (net dpm minute counts), is greater than L_c , the measurement is "flagged" to identify it as above background. The decision level L_c , sometimes called the critical level, is defined as:

$$L_c = \frac{1.96\sqrt{s_s^2 + s_b^2}}{E(\frac{A}{100})}$$

where:

s_s = counting error in sample measurement,
or $s_s = c/t^2$, where c = measurement total counts (sample + bkg), and t = measurement count time,

s_b = counting error in background measurement, or $s_b = B/t_b^2$,
where B = total background counts, and t_b = background count time.

E = detector efficiency in counts per disintegration,

A = detector sensitive area in cm^2 , and

1.96 = 97.5th percentile value of a one-tailed normal distribution.

Each measurement will be converted to net dpm/100 cm^2 and screened against L_c . Those above L_c are reported (in the data report) as "above" background, and adjusted to account for the presence of Fe-55 using the method described in Response No. 2.

Comment No. 5

NRC Comment: The NRC Staff does not agree with the LIPA proposal to use 50 % of the 5,000 dpm/100 cm² guideline value for direct beta-gamma surface contamination measurements as the action level for investigation (and potential reclassification) in unaffected areas. An action level has been suggested which is similar to the critical level concept, but at a higher probability level, perhaps three or four sigma. Please provide a definition of the proposed action level and explain how it will be implemented.

LIPA Response: LIPA will calculate an action level, L_R , for direct beta-gamma measurements and use it to screen each measurement which has been flagged as being above background as described in response No. 4 above. This action level is defined as:

$$L_R = \frac{3\sqrt{s_s^2 + s_b^2}}{E(\frac{A}{100})}$$

where:

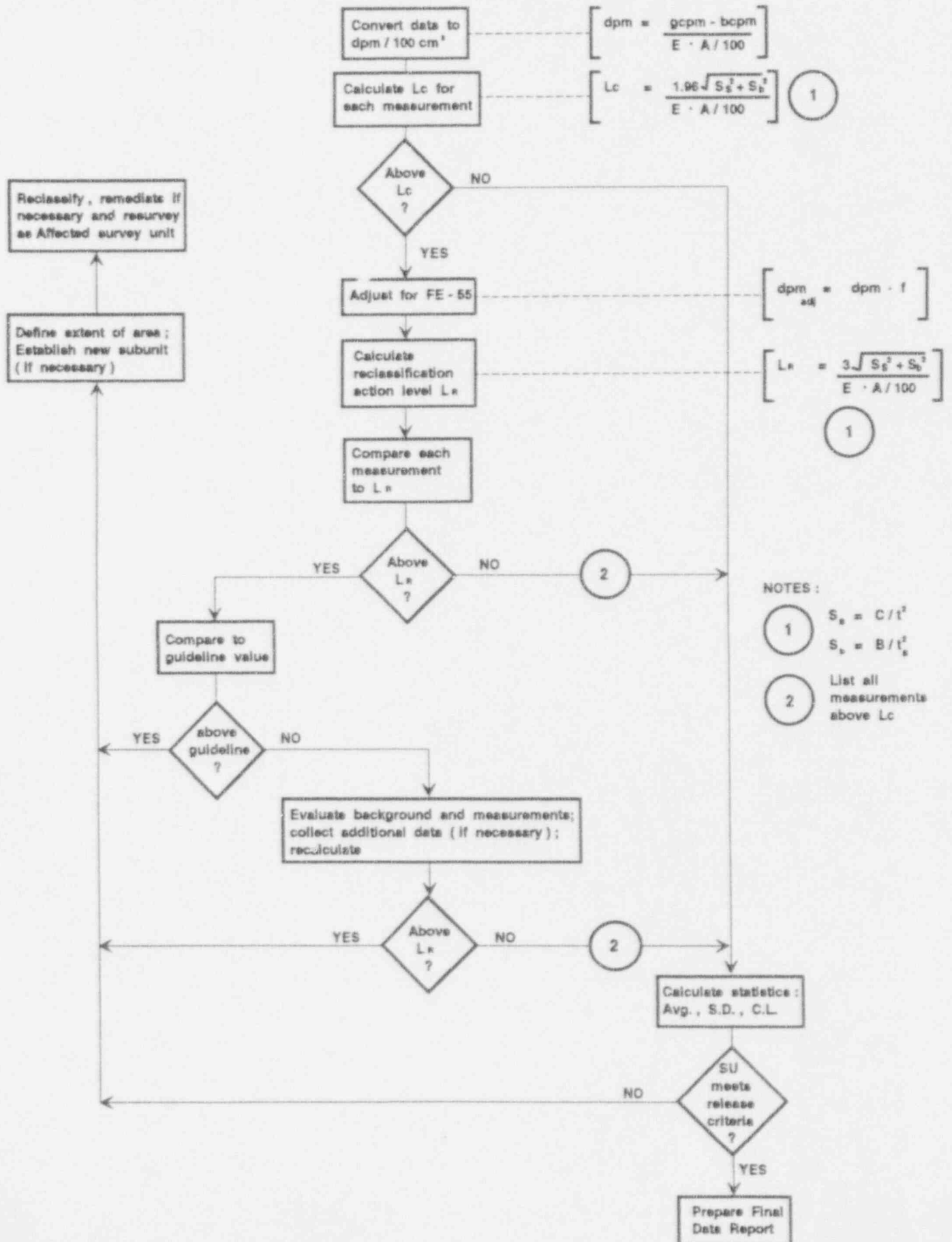
the parameters: s_s , s_b , E and A are defined in the response to comment No. 4, and

3 = the 99.7 th percentile value of the standard normal distribution ².

The attached flow chart summarizes the data evaluation process for unaffected survey units. It shows how the data are evaluated and screened against the critical level and the action level. It identifies the actions taken when measurements exceed the action level, including reclassification and resurvey if required. It should be noted that LIPA has previously agreed to the use of an action level of 25% of the guideline values for all other measurements of surface contamination, i.e., removable beta-gamma, removable alpha and direct alpha, and intend to retain this aspect of the survey program.

². The three sigma coefficient represents the 99.7th percentile of the standard normal distribution (two tailed). From NUREG/CR-2082, "Monitoring for Compliance with Decommissioning Termination Survey Criteria", p. 132.

DATA EVALUATION - UNAFFECTED AREAS

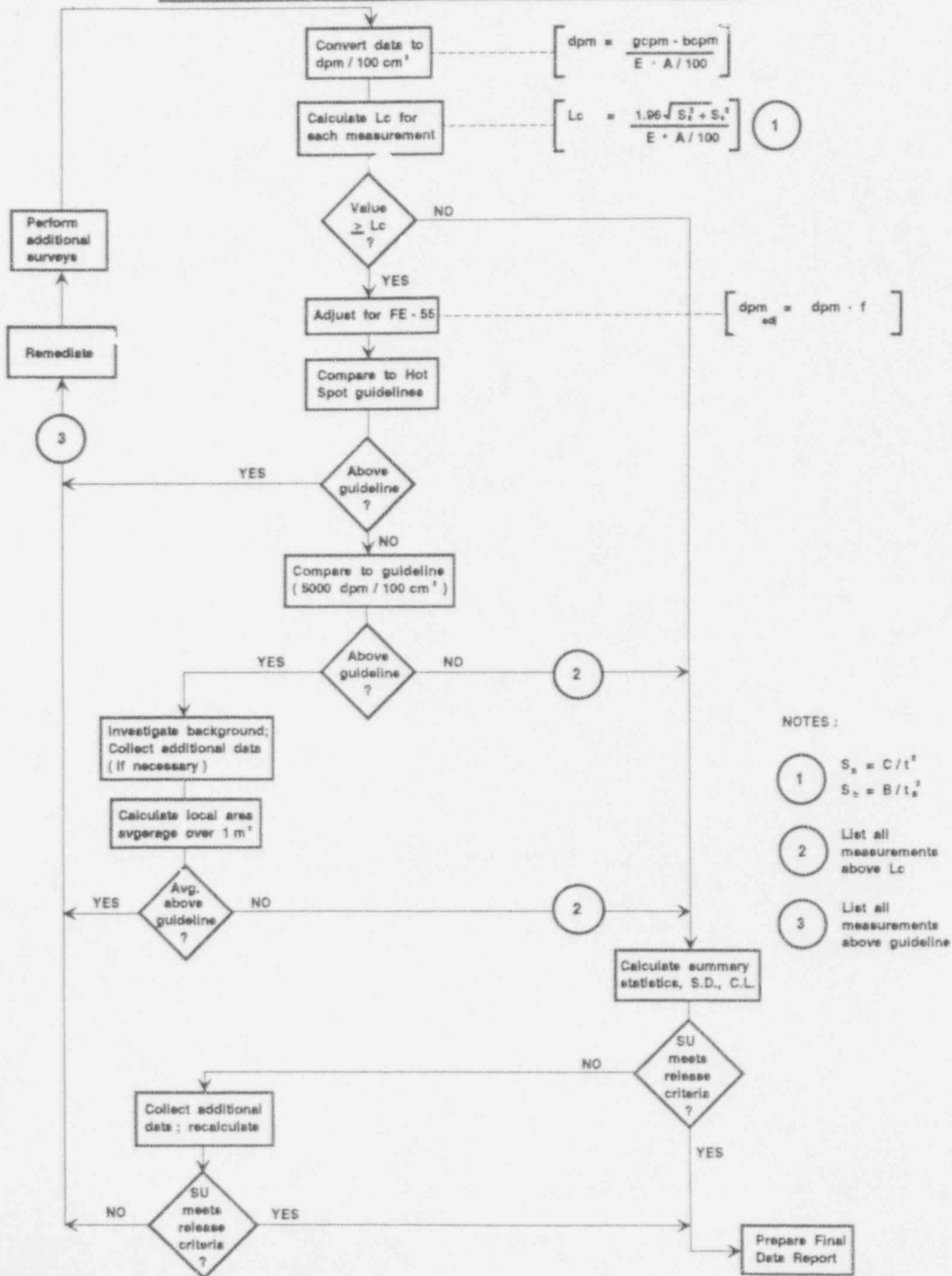


Comment No. 6

NRC Comment: Please identify in the Release Record data report for each survey unit, all instances where individual surface contamination measurements are above guideline values in affected areas. Summarize the investigation and show the evaluation to determine if the local area average exceeds the guideline value. For any cases where the guideline value is exceeded, describe the remediation and present the results of the resurvey which demonstrates that the release criteria have been met.

LIPA Response: LIPA will identify in the Release Record data report for each survey unit, all instances where individual surface contamination measurements are above guideline values in affected areas. The Release Record will summarize the investigation and the evaluation to determine if the local area averages exceed the guideline value. For any cases where the guideline value is exceeded, the remediation will be described. Results of the resurvey will be presented to demonstrate that the release criteria have been met. The attached flow chart summarizes the data evaluation process for affected areas. It shows the identification of all measurements above the critical level. It includes the screening of each measurement against the elevated level guideline and the average guideline values, and the response when measurements fail the test.

DATA EVALUATION - AFFECTED AREAS



Comment No. 7

NRC Comment: In reviewing the Release Record for the Shoreham main Turbine, several measurement results were identified which LIPA has indicated were due to data entry typographical errors. Please provide a corrected data report.

LIPA Response: LIPA will correct the errors and revise the Turbine Release record data report. The revised report will be submitted as part of the Decommissioning Phase One Final Report in June, 1993.