Long
Island
Power
Authority

Shoreham Nuclear Power Station
P.O. Box 628
North Country Road
Wading River, N.Y. 11792

MAR 8 1994

LSNRC-2148

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

Supporting Information For Proposed
License Amendment No. 11 To LIPA's Possession Only
License No. NPF-82 And Appendices A And B
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Ref: 1) LIPA letter to USNRC, LSNRC-2115, dated November 4, 1993; subject: License Change Application No. 2

Ladies and Gentlemen:

Enclosed per the request of Mr. C. L. Pittiglio of your staff is information supporting the subject proposed license amendment which was submitted in Reference 1. The enclosure specifically supports the estimates and conclusions that were provided in Reference 1 regarding the small quantity of remaining radioactive material, and the low radiological significance of potential accident releases from SNPS, following completion of the last shipment of irradiated fuel from the site.

Should you require any additional information concerning this submittal, please do not hesitate to contact my office.

Very truly yours,

A. J. Bortz
Resident Manager

Enclosure

100146

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

9403150118 940308
PDR ADOCK 05000322
P PDR

NL101/1

Shoreham Nuclear Power Station
Wading River, New York 11792

516 929-8300

Long
Island
Power
Authority

Memorandum

February 23, 1994
LDD-NAS-94-0067

TO: S. Schoenwiesner

FROM: S. H. Moss

REF: 1) LDD-NED-93-0552 dated 10/15/93 from S. H. Moss to D.Y. Filipowicz, Back-up Information in Support of LSNRC-2115: License Change Amendment No. 2

2) LSNRC-2115 dated 11/04/93, License Change Application No. 2 NRC License NPF-82 Shoreham Nuclear Power Station - Unit 1 Docket No. 50-322

SUBJECT: Estimate of Radiological Source Term and Potential Accident Releases Remaining Post-Fuel Transfer

In response to your request, this memo summarizes and clarifies the subject information which was provided earlier via Reference 1, and which was used in support of the submittal of Reference 2 to the NRC.

This memo discusses the assumptions used to develop the post-fuel transfer source terms and the conclusions to be drawn from the resulting post-fuel transfer accident analyses. **The main conclusion proves to be: 1) that less than 8 mCi of radioactive material (other than sealed solid sources) will remain at SNPS once fuel disposition is complete, and, 2) that despite the extremely conservative assumptions used to develop the source terms remaining after fuel transfer, the resulting accident analyses yield doses that are below the 10CFR50 APP I limits for normal plant operations without taking any credit for HEPA filtration.**

After fuel transfer there is expected to be very little radiologically contaminated or activated material remaining on-site. Attachment 1 contains an estimate of the radiological material anticipated to remain on-site after the fuel has been shipped but before any additional decommissioning is done. The assumptions used to develop this estimate are stated in Attachment 1.

Attachment 2 provides the original Decommissioning Plan accident analyses in a tabular form for convenient comparison with the post-fuel transfer estimates. This information has been extracted from the NRC approved Shoreham Decommissioning Plan (DP). Attachment 2 should be reviewed in parallel with Attachment 3.

Attachment 3 provides the results of comparable accident analyses based on the generally smaller source terms available in the post-fuel transfer environment. The Attachment 3 values were developed with the same methodology used to generate the DP accident analyses results shown in Attachment 2. The following paragraphs describe the post-fuel accident analyses results on a section by section basis as itemized in Attachment 3.

While Appendix I to 10 CFR Part 50 applies to normal plant operation, it may be illustrative to review the dose limits associated with App.I. prior to the detailed discussion of the accident analyses for the sake of comparison. The following are Appendix I limits:

DOSE TYPE	APP. I. ANNUAL LIMIT
Gamma Air	10 mRad
Beta Air	20 mRad
Organ Dose	15 mRem

Gamma and Beta Air Doses are based on noble gas releases. The only source of noble gases presently at Shoreham during decommissioning, comes from the Kr-85 in the fuel assemblies. The only non-trivial leakage mechanism for this noble gas is release following an accident (Fuel Handling Accident). There will be no sources of radioactive noble gas present on-site after fuel transfer is complete.

Organ Doses may be obtained from Shoreham plant source term, if this was somehow released into the atmosphere outside the plant in the form of particulates. The following accident analysis of the Post Fuel Transfer Source Term indicates that it would require a release to the outside atmosphere of at least 2 orders of magnitude higher radioactivity than source terms assumed by the analysis, to attain Appendix I organ dose limits.

Section 3.4.1.1 of the D-Plan originally addressed the Waste Container Drop accident. The source term was 300 μCi of activated concrete rubble containing mixed nuclides and was all presumed to be released from the Reactor Building without any HEPA credit (source term reduction). The closest corresponding event in the post-fuel transfer environment, in the absence of activated concrete rubble, would be a Spent Fuel Rack Drop accident.Here the same percentage of the total available source term is assumed to go airborne (10%). Based on the size of the largest single Spent Fuel Rack (96 Bundle Capacity) and contamination levels given by Attachment 1 (taken from the Site Characterization Report) the airborne source term would be 29.6 μCi Co-60 equivalent. Scaling from the doses generated by the airborne release associated with the Vacuum Filter Bag Rupture accident contained in Attachment 2 yields the comparable doses for the Spent Fuel Rack Drop accident. It should be noted that whereas Attachment 2 shows doses for the general public at the Exclusion Area Boundary (EAB) as well as for workers within the Reactor Building, Attachment 3 gives only the doses for the general public at the EAB but both with and without credit being taken for HEPA filtration.

While it appears that the source term associated with the Spent Fuel Rack Drop accident is much smaller than the source used for the Waste Container Drop accident, that is not strictly true. The Waste Container Drop source term consists of mixed nuclides, 97% of which are Fe-55 or H-3 with only 2.6% being Co-60. The source term for the Spent Fuel Rack Drop is all Co-60 equivalent which results in a larger more conservative dose impact.

Section 3.4.1.2 in the D-Plan addressed the Combustible Waste Fire accident. The source term used was based on a fire in a trash barrel containing 1% of the total system contamination (66.2 μCi of Co-60), creating a Reactor Building airborne activity of 9.93E-03 μCi with an atmospheric release of 4.96E-06 μCi assuming credit for HEPA filters. A corresponding post-fuel transfer event would be a similar fire using as barrel inventory 1% of the combined contamination of the Spent Fuel Storage Pool Walls, Floor and Fuel Racks (77.1 μCi of Co-60 equivalent). This yields an airborne Reactor Building activity of 1.16E-02 μCi (listed in Attachment 3 as the 'NO HEPA Source Term' and is equivalent to the Reactor Building Source Term). The equivalent atmospheric source term would be 5.78E-06 μCi (listed on Attachment 3 as the 'HEPA Source Term'). The comparable doses, both Whole Body (WB) and Maximum Organ (MO) are generated by scaling the respective source terms as was done above for the Spent Fuel Rack Drop accident. Though the post-fuel transfer source term is slightly higher than the original D-Plan source term for this accident it is still smaller than that for other accidents in the D-Plan and is not the limiting post-fuel transfer accident.

Section 3.4.1.3 in the D-Plan addressed the Contaminated Sweeping Compound Fire accident. The source term used assumes 10% of the total system contamination after piping system dismantlement operations is on the floor and collected by the sweeping compound, deposited in a drum with the drum catching fire. The activity in the drum was estimated to be 420 μCi of Co-60 equivalent. This results in a Reactor Building airborne activity of 6.3E-02 μCi and an atmospheric release of 3.15E-05 μCi assuming credit for HEPA filtration. The comparable post-fuel transfer drum activity is estimated as 10% of the remaining G41 piping contamination activity equal to 17.875 μCi. This value yields corresponding Reactor Building activity of 2.68E-03 μCi and atmospheric activity of 1.34E-06 μCi, assuming HEPA filtration credit for the atmospheric release. Scaling the doses to the source terms yields doses which are also lower for the post-fuel transfer environment than for the original D-Plan.

Section 3.4.1.4 in the D-Plan addressed the Vacuum Filter-Bag Rupture accident. The source term starts out with the same 66.2 μCi of Co-60 as the Combustible Waste Fire but does not burn it to make it airborne. Rather, the scenario assumes the vacuum filter-bag rupture occurs while the vacuum is on and that drives the activity airborne in the Reactor Building. The corresponding source term developed for the post-fuel transfer environment in the case of the Combustible Waste Fire was 77.1 μCi of Co-60 and that was also used here. These two values (66.2 μCi and 77.1 μCi) represent the Reactor Building airborne activity for the original D-Plan and the post-fuel transfer case respectively. Similarly, the corresponding atmospheric release activities, assuming credit for HEPA filtration, would be 3.31E-02 μCi and 3.86E-02 μCi respectively. As previously shown, the doses estimated are a reflection of the source terms used and have been scaled directly from the source terms.

Section 3.4.1.5 in the D-Plan addressed the Oxyacetylene Explosion accident.The source term was 7.04 μCi of Co-60. While no alternate source term could be suggested for the post-fuel transfer situation neither could the accident be precluded. Consequently, the same source term is assumed for both cases. This results in the same dose rate for the corresponding situation of EAB dose to the general public assuming credit for HEPA filtration. Whereas, the original D-Plan analysis also calculated a worker dose inside the Reactor Building based upon a very limited exposure time, it is more illustrative to calculate instead the dose to the general public at the EAB assuming no credit for HEPA filtration of the atmospheric release. It may be possible to show that the results in the case of the post-fuel transfer source terms are low enough to forego the credit for HEPA filtration and hence the need for HEPA filtration. The difference in the dose to the general public at the EAB based on the presence or lack of HEPA filtration credit is a factor of 2000 due to source term being strictly Co-60.

Section 3.4.1.6 in the D-Plan addressed the Explosion of Liquid Propane Gas (LPG) Leaked from a Front-End Loader (Forklift) both with and without Waste Container Rubble. The source terms used were 684 μCi of Mixed Nuclides for the case with Waste Container Rubble and 84 μCi of Co-60 without Waste Container Rubble. The waste container rubble in question was all activated concrete which would not be present in the post-fuel transfer environment. (Assuming all BioShield Wall blocks to be removed are also off-site.) Consequently, this version of the accident analysis is not appropriate for consideration on Attachment 3. The case without Waste Container Rubble cannot be argued away and so appears with the same source term as the original D-Plan analysis. The doses associated with the general public at the EAB are likewise the same. There can be no HEPA filtration credit taken in this scenario as the explosion is assumed to have occurred in the HEPA prefilters damaging them to the point of causing the release of the radiological inventory they have accumulated. Therefore, no HEPA filtered dose numbers are calculated.

Section 3.4.1.7 in the D-Plan addressed the Contamination Control Envelope Rupture accident. The source term used was 45,290 μCi of Mixed Nuclides. For a bounding source term from the post-fuel transfer timeframe, it was decided to sum all the remaining source terms available not including sealed calibration or instrumentation sources. This results in a source term of 7707 μCi of Co-60 equivalent. The doses associated with this source term correspond very closely with the original D-Plan doses for the general public at the EAB with credit taken for HEPA filtration. The Whole Body dose for the post-fuel transfer source term is about one-third of the dose for the original D-Plan source term, while their respective Maximum Organ doses are essentially equal. The reason for this apparent discrepancy is the fact that the mixed nuclide nature of the original D-Plan source term is not as organ specific as the pure Co-60 source term of the post-fuel transfer case. For a maximum bounding case, the unfiltered release dose is also calculated and that is clearly below the 10CFR50 APP I limits of 10 mRad Gamma - Air, 20 mRad Beta - Air, and 15 mRem Organ Dose.

Section 3.4.1.8 in the D-Plan addressed the Fuel Damage Accident which had always been the most limiting case of all the accidents. In the post-fuel transfer situation there is no possibility of any fuel accident as there is no longer any fuel left on-site.

Conclusion: In the post-fuel transfer environment there is no possibility of any conceivable accident based on any expected source term from exceeding 10CFR50 APP I limits even without taking any credit for HEPA filtration.

cc: T. Garvey	w/attach
T. Cardile	" "
C. Adey	" "
R. Pauly	" "
RM/DC	" "

ATTACHMENT 1

RADIOLOGICAL SOURCES REMAINING PAST FUEL TRANSFER				
TYPE	AMOUNT	LOCATION	DISPOSITION	COMMENTS / REFERENCE
GASEOUS				
(NONE)	N/A	N/A	N/A	Though 24.3 uCi of gaseous sources present currently, all expected to be disposed of by end of fuel xfer
LIQUID*				
SFSP-H2O	340,000 gal 51.5 uCi	R.B. Elev.175	Drain through RB-SWDT per ODCM	Estimated by OPS personnel. Based on recent Rad Chem survey.
SOLID*				
FUEL RACKS	26743 dpm 61000 ft^2 6710 uCi	R.B. Elev.175	Ship offsite for volume reduction or decontaminate on site and free release	From Site Characterization Report (SCR) for G41 system Estimated from dwg measurements
SFSP WALLS	26743 dpm 5386 ft^2 592.5 uCi	R.B. Elev.175	Hydrolaze or hand decontaminate Leave clean and free release	From Site Characterization Report (SCR) for G41 system Estimated from dwg measurements
SFSP FLOOR & SILT	26743 dpm 1113 ft^2 122.43 uCi	R.B. Elev.175	Hydrolaze or hand decontaminate Leave clean and free release	From Site Characterization Report (SCR) for G41 system Estimated from dwg measurements
SFSP DEMIN SKID&FILTERS	51.5 uCi	R.B. Elev.175	Remove for offsite shipment	Estimate assumes inventory comparable to SFSP
FUEL POOL CLG & CLNG PIPE & EQPT	26743 dpm 1625 ft^2 178.75 uCi	R.B. Elev 175 150' & 1?2'	Remove for offsite shipment	From Site Characterization Report (SCR) for G41 system Estimated from ECR T-208

* All sealed instrumentation and calibration sources will be considered as disposed of or as non-credible release contributions.

Decommissioning Plan Onsite Accident Analyses – Original

D–Plan Section	Accident Type	Rx. Bldg. Source Term [uCi]	Atmospheric Source Term [uCi]	(1) Worker WB–Dose [mRem]	(2) Worker MO–Dose [mRem]	(1),(3) EAB WB–Dose [mRem]	(2),(3) EAB MO–Dose [mRem]
3.4.1.1	Waste Container Drop	300 Mixed Nuclides	300 Mixed Nuclides	5.36E–04	2.91E–02	6.48E–05	3.36E–03
3.4.1.2	Combustible Waste Fire	9.93E–03 Co–60	4.96E–06 Co–60	1.74E–07	2.92E–05	8.04E–12	1.63E–09
3.4.1.3	Contaminated Sweeping Compound Fire	6.30E–02 Co–60	3.15E–05 Co–60	1.10E–06	1.85E–04	5.10E–11	1.03E–08
3.4.1.4	Vacuum Filter Bag Rupture	66.2 Co–60	3.31E–02 Co–60	1.16E–03	1.95E–01	5.36E–08	1.09E–05
3.4.1.5	Oxyacetylene Explosion	7.04 Co–60	3.52E–03 Co–60	1.23E–04	2.07E–02	5.70E–09	1.16E–06
3.4.1.6	LPG Explosion with WCR	684 Mixed Nuclides	684 Mixed Nuclides	1.22E–03	6.64E–02	1.48E–04	7.76E–03
	LPG Explosion without WCR	84 Co–60	84 Co–60	1.47E–03	2.47E–01	1.36E–04	2.76E–02
3.4.1.7	Contamination Control Envelope Rupture	45,290 Mixed Nuclides	30.1 Mixed Nuclides	9.08E–01	2.20E–02	1.79E–05	1.25E–03
3.4.1.8	Fuel Damage Accident	1.50E+09	1.50E+09			1.08E+00	9.39E+01

(1) WB–Dose = Whole Body Dose [10CFR50 APP. I. Dose Limit = 5 mREM]

(2) MO–Dose = Maximum Organ Dose = Lung Dose except for Fuel Damage Accident where it is the Skin Dose [10CFR50 APP. I. Dose Limit = 15 mREM]

(3) EAB = Exclusion Area Boundary

Decommissioning Plan Onsite Accident Analyses – Post Fuel Transfer

D-Plan Section	Accident Type	NO HEPA Source Term [uCi]	HEPA Source Term [uCi]	(1) NO HEPA WB-Dose [mRem]	(2) NO HEPA MO-Dose [mRem]	(1) HEPA WB-Dose [mRem]	(2) HEPA MO-Dose [mRem]
3.4.1.1	Spent Fuel Rack Drop	2.96E+01 Co-60	1.48E-02 Co-60	4.79E-05	9.75E-03	2.40E-08	4.87E-06
3.4.1.2	Combustible Waste Fire	1.16E-02 Co-60	5.78E-06 Co-60	1.87E-08	3.80E-06	9.37E-12	1.90E-09
3.4.1.3	Contaminated Sweeping Compound Fire	2.68E-03 Co-60	1.34E-06 Co-60	4.34E-09	8.77E-07	2.17E-12	4.38E-10
3.4.1.4	Vacuum Filter Bag Rupture	7.71E+01 Co-60	3.86E-02 Co-60	1.25E-04	2.54E-02	6.24E-08	1.27E-05
3.4.1.5	Oxyacetylene Explosion	7.04E+00 Co-60	3.52E-03 Co-60	1.14E-05	2.32E-03	5.70E-09	1.16E-06
3.4.1.6	LPG Explosion with WCR	Not Credible W/O Concrete Rubble					
	LPG Explosion without WCR	8.40E+01 Co-60	No HEPA Credit Possible	1.36E-04	2.76E-02		
3.4.1.7	Contamination Control Envelope Rupture	7707 Co-60	3.85E+00 Co-60	1.25E-02	2.54E+00	6.24E-06	1.27E-03
3.4.1.8	Fuel Damage Accident	Not Credible W/O Fuel					

(1) WB-Dose = Whole Body Dose [10CFR50 APP. I. Dose Limit = 5 mRem]

(2) MO-Dose = Maximum Organ Dose = Lung Dose [10CFR50 APP. I. Dose Limit = 15 mRem]

All doses at Exclusion Area Boundary.

'HEPA' means 'with credit taken for HEPA filtration prior to release'.

'No HEPA' means 'no credit taken for HEPA filtration prior to release'.