Long Island Power Authority	SAFETY EVALUATION	SE No. 93-587, Rev 0 SHEET 1 OF 15

SCOPE OF SAFETY EVALUATION

This safety evaluation addresses the removal of the activated portion of the Bioshield Wall which exceeds the free release limit of 5 μR/hr above background as measured at 1m. This safety evaluation determines if the removal of part of the Bioshield Wall by the method specified results in any unreviewed safety question or requires changes to the DP/DSAR/USAR/Defueled Technical Specifications, or yields environmental impacts different from or exceeding those set forth in the Supplement to the SNPS Environmental Report (Ref. 5)

REFERENCES

1. 10CFR50.59
2. SNPS Defueled Technical Specifications
3. SNPS USAR/DSAR, Rev. 4
4. SNPS Site Characterization Report dated May 1990 including Addendum 1 dated June 1990; Addendum 2 dated August 1990; and Addendum 3 dated June 1992

(Continued on Sheet 3)

DISCUSSION

See Sheet 3

CONCLUSION

The partial removal of the Bioshield Wall by the methods described does not result in an unreviewed safety question or require a change to the Defueled Technical Specifications nor result in an environmental impact different from or exceeding those set forth in the Supplement to the Environmental Report (Decommissioning). The Decommissioning activity does represent a change to the D-Plan and requires the revision of affected sections of the USAR/DSAR.

PREPARER SH Mon DATE 9/22/93	REVIEWER 16 Feitean DATE 9/22/93
APPROVALS	
RESPONSIBLE SECTION HEAD Charles W Addy DATE 9/22/93	OPS DIVISION MANAGER (Reg. Safety Rel.) N/R DATE
IMPLEMENTATION ORGANIZATION DIV. MGR. Ted Dod DATE 9/23/93	DATE
	RESPONSIBLE DIV. MANAGER (LAST) [signature] DATE 9/23/93

SAFETY EVALUATION CHECKLIST

SHOREHAM NUCLEAR POWER STATION

(1) CHECKLIST APPLICABLE TO: SE NO. 93-587 - BIOSHIELD WALL REMOVAL

(2) SAFETY EVALUATION - PART A

The item to which this evaluation is applicable: represents a change to the station or procedures as described in the DSAR/USAR, or, conducts tests or experiments not described in the DSAR/USAR. Yes X No ___

When the answer to the above is "Yes", attach a detailed description of the item being evaluated and the affected section(s) of the DSAR/USAR with the applicable changes indicated. See Attachment A - Marked-up DSAR pages

(3) SAFETY EVALUATION - PART B

Will this item require a change to the station Defueled Technical Specifications: Yes ___ No X

When the answer to the above is "Yes", identify the Specification(s) affected and/or attach the applicable page(s) with the change(s) indicated.

(4) SAFETY EVALUATION - PART C

The item to which this evaluation is applicable is a decommissioning activity and represents a change or deviation from the Decommissioning Plan (DP). Yes X No ___

When the answer to the above is "yes", attach a detailed description of the item being evaluated and identify the affected section(s) of the DP with the applicable changes or deviations indicated.

(5) SAFETY EVALUATION - PART D

As a result of the item to which this evaluation is applicable:

(A) Will the probability of an accident previously evaluated in the DP/DSAR/USAR be increased? Yes ___ No X

(B) Will the consequences of an accident previously evaluated in the DP/DSAR/USAR be increased? Yes ___ No X

(C) May the possibility of an accident which is different than any already evaluated in the DP/DSAR/USAR be created? Yes ___ No X

(D) Will the probability of a malfunction of equipment important to safety previously evaluated in the DP/DSAR/USAR be increased? Yes ___ No X

(E) Will the consequences of a malfunction of equipment important to safety previously evaluated in the DP/DSAR/USAR be increased? Yes ___ No X

(F) May the possibility of malfunction of equipment important to safety different than any already evaluated in the DP/DSAR/USAR be created? Yes ___ No X

(G) Will the margin of safety as defined in the bases to any Defueled Technical Specification be reduced? Yes ___ No X

If the answer to any of the preceding is "Yes", an Unreviewed Safety Question may be involved. Justify the conclusion that an Unreviewed Safety Question is or is not involved. Attach additional pages as necessary.

REFERENCES (Continued)

5. LIPA Supplement to SNPS Environmental Report (Decommissioning) dated December 1990.

6. Order Approving the Decommissioning Plan and Authorizing Decommissioning of Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322, dated June 11, 1992.

7. NRC Environmental Assessment and Finding of No Significant Impact Related to the Order Authorizing Decommissioning of SNPS, Unit 1 dated June 11, 1992.

8. LIPA SNPS Decommissioning Plan as supplemented.

9. Calculation No. CCI-039242, Rev. 0 "Documenting the Range of Activated Steel and Concrete in SNPS Biological Shield and Segmented Reactor Pressure Vessel" dated 2/5/92

10. NUREG/CR-0672,"Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station." dated June 1980.

DISCUSSION

The Shoreham Site Characterization Program begun in 1990, was established to evaluate the Shoreham site from a radiological standpoint and to assemble and assess information pertinent to the eventual decommissioning of the Shoreham site. Its initial report (dated May 1990) noted that "... the biological shield wall has an external dose rate of less than 5 μR/hr, but small portions of the inner face may exceed 5 μR/hr. Detailed surveys will be required during decommissioning to determine what, if any, portions of the shield wall's inner steel liner will require removal." Subsequently the Decommissioning Plan as supplemented (Ref. 8) indicated that the steel-lined concrete bioshield wall which surrounded the Reactor Pressure Vessel (RPV) was activated to some extent. Analysis of radiological survey data at that time had indicated that the extent of activation above the 5 μR/hr at one meter gamma dose rate criterion was confined to portions of the wall's inner steel liner. Concrete and interior support steel was believed to be below the free release limit. It was also noted, however, that background contributions of the immediately adjacent RPV and limited accessibility during the surveys caused the analysis of the data to be complex and necessarily uncertain. Consequently, it was noted that a more accurate determination of the radiological conditions of the bioshield wall would be necessary.

With the removal of the bulk of the RPV and Internals, extraneous background radiation contributions have been significantly reduced. Recent and more accurate radiological data to characterize the condition of the bioshield has been obtained. This data shows that a band of roughly 17 feet in height of the bioshield liner extending from approximately elevation 109' to elevation 126' is activated above the gamma dose rate criterion for free release and that some interior concrete and support steel within this band is also activated above the limit. Therefore, the activated material must be removed as part of decommissioning.

Because the activated band of material is located approximately midway along the height of the bioshield, it will also be necessary to first remove the non-activated upper section of the wall, as well as the wall extension, interfering piping, components and structural platforms which are mounted to or near the wall. Removal of the non-contaminated/non-activated upper section interferences as well as the slightly contaminated main steam line segment interferences are not part of the scope of this safety evaluation.

The bioshield removal methodology will involve the use of remotely operated diamond wire saw cutting equipment to section the bioshield wall into manageable blocks. The blocks will be lifted from their locations and lowered to EL. 78'-0" using a new crane to be mounted on top of the lower head of the RPV. The blocks will then be moved through the drywell equipment hatch using specially designed and constructed rolling carts capable of handling well over the estimated 16,000 lb. weight of the average bioshield wall removable block.

The diamond wire saw operation will use a water slurry for both cooling of the blade and control of contamination. Water will be processed for reuse by collection of slurry, the syphoning of the water and the drying out of the residual material. Appropriate contamination control measures, such as use of radiation work permits, portable airborne radiation monitors, Herculite or comparable materials to cover open areas, contamination control tents with HEPA-filtered ventilation, and periodic air sampling will be employed as deemed necessary by H.P.

Respirators will be used if air sample results indicate the need for personnel protection against airborne radionuclides.

The activated Bioshield Wall Blocks cut and removed from the Reactor Building which do not meet the criteria for free release will be shipped offsite for disposal (burial) as low level radioactive waste.

Removal of a portion of the Bioshield Wall has no impact on fuel storage, fuel handling, or security but could impact RB structural integrity. In fact, DSAR Table 3.2-1, "Equipment Classification - Spent Fuel Storage Entry for Biological Shielding" refers to a Note 5 which states that the structure was originally constructed as Seismic Category I; though currently classified as QA Category N/A, Bioshield Wall modifications have been analyzed for DBE to ensure integrity of Reactor Building. (See PART G - Seismic/Structural Analysis of Bioshield Wall Removal on Reactor Building Integrity)

With the activated portions of the RPV now removed from the Reactor Building, the original shielding function of the Bioshield Wall is no longer required. Thus, removal of the Bioshield Wall will not result in a loss of protection to workers and equipment from high radiation levels.

PART A - REVIEW OF THE USAR/DSAR

Table 3.2-1 (Equipment Classification - Spent Fuel Storage) page 5 of 7, in the DSAR lists the Biological Shielding as LIPA QA Category II and Seismic Category N/A. Note 5 associated with that entry indicates that the structure was originally constructed as Seismic Category I; and that modifications will be analyzed for DBE to ensure the integrity of the Reactor Building. DSAR Section 12.1 refers to the original physical design for radiation protection as provided in USAR Sections 12.1 and 12.3. A change to the DSAR is required as a result of the scope of work covered by this safety evaluation.

DSAR Sections 3.8, 12.1 and 12.3 need to be revised to reflect partial removal of the Bioshield Wall. (See Attachment 'A' for marked up DSAR pages.)

PART B - REVIEW OF THE DEFUELED TECHNICAL SPECIFICATIONS

The latest version of the Defueled Technical Specifications including Amendment No. 10, dated July 14, 1993, was reviewed and no change to the Defueled Technical Specifications is required to allow for the activity covered by this safety evaluation (removal of a portion of the Bioshield Wall). The Bioshield Wall was not associated with any of the controls/specifications, and is not required by any current technical specification.

PART C - REVIEW OF THE DECOMMISSIONING PLAN (DP)

A review of LIPA-SNPS Decommissioning Plan as Supplemented (Ref. 8) shows that the Bioshield Wall was considered as part of the Decommissioning Plan, but not with such extensive deconstruction anticipated.

Among the major considerations potentially impacted by the remediation of the Bioshield Wall are removal methods, total worker dose estimate, LLRW volume and curie content, project manhours and cost estimate. Impact on these considerations is addressed below. No impact on or revisions to the Accident Analyses provided in the DP are necessary for the removal of portions of the Bioshield for the reasons discussed under PART F - REVIEW OF RADIOLOGICAL IMPACT.

METHODOLOGY

The methodology to be used to segment the Bioshield Wall has previously been used in the course of the present decommissioning effort to cut Shield Blocks on Elev. 175'. The use of Diamond Wire Saw to mechanically cut through materials, to segment them, is among the approved mechanical techniques considered by the DP. In particular, DP Table 2.2-4 refers to both shield blocks and RPV ring sections as being considered for segmentation by Diamond Wire Saw (DWS). There is nothing unique about the Bioshield Wall that would preclude the suitability of DWS methodology; considering that DWS was deemed a suitable method for RPV ring sections and shield blocks.

TOTAL WORKER DOSE ESTIMATE

The occupational radiation exposure estimated to be incurred by the removal of the Bioshield Wall is 0.34 man-rem, which is a small fraction of the original decommissioning project exposure estimate of 189 man rem. The decommissioning radiation exposure incurred to date indicates that the additional exposure due to bioshield removal will not cause the initial estimate to be exceeded. See Table 1 for a breakdown of the dose estimate.

TABLE 1
BIOSHIELD WALL REMOVAL DOSE ESTIMATE

ITEM NO.	TASK	MANPOWER [men]	DURATION [days]	EXPOSURE	MANHOURS [mn-hrs]
1	Install waste water collection system	4	10	yes	320
2	Support wastewater collection	2	80	yes	1280
3	Install water services	2	20	yes	320
4	Install electrical services	2	20	yes	320
5	Fabricate m't'l. move. cart mech.	4	10	no	0
6	Install m't'l move. cart mech.	2	5	no	0
7	Fabricate m't'l. move lifting rig	2	10	no	0
8	Install tent across reactor cavity	8	10	yes	640
*9	Prepare for wall extension removal	2	10	no	0
*10	Welders to cut wall extension	7	20	no	0
*11	Rigging for wall and extension	5	80	yes	3200
12	Mount support attachments	2	80	yes	1280
13	Drilling of 91 holes	2	91	yes	1456
14	Prepare foundation of crane	4	25	yes	800
15	Install crane	4	5	yes	160
16	Crane operators	1.5	80	no	0
17	Radwaste support personnel	6	120	yes	5760
18	Cutting contractors	3	55	yes	1320
				SUM	16856
	Assuming all exposure at maximum Bioshield exposure rate [20 μRem/hr @ 1m] yields			337.12 milli-man-rem or 0.34 man-rem	

* These items, although included in this table to provide an upper limit on dose, address removal of Bioshield Wall Extension, which is not strictly within the scope of this SE. The wall extension removal activity is specifically addressed in ECR T-00296 and SE-93-593.

LLRW VOLUME AND CURIE CONTENT

All radioactive waste associated with the Bioshield Wall removal will be Class A. A total volume of approximately 3500 ft^3. of steel and concrete will be disposed of as low level radioactive waste, based on the complete removal of the activated portion of the Bioshield Wall from GE Vessel El 188" to El. 452" (which corresponds to Reactor Building elevation of 107' - 9.5" to 129'-9 1/2") with Inner-Radius of 135" and Outer-Radius of 159". Portions of the Bioshield Wall above 129'-9 1/2" which are to be removed as part of the remediation process are not activated and are expected to be free releasable. Therefore, they are not included in estimates of the LLRW.

Based on calculation No. CCI-039242 Rev. 0, 'Documenting the Range of Activated Steel and Concrete in SNPS Biological Shield and Segmented Reactor Pressure Vessel' (Ref. 9) and SNPS Site Characterization Report Addendum 3 dated June 1992 (Ref. 4), the maximum curie content of the activated Bioshield Wall segments are estimated as follows:

H-3	@	8.36E+00 mCi
Ni-63	@	5.88E-01 mCi
Co-60	@	2.08E+02 mCi
Fe-55	@	1.75E+02 mCi
Mn-54	@	8.64E+00 mCi
Eu-152	@	5.64E+01 mCi
Total	@	4.57E+02 mCi

The above values reflect decay through July 1, 1993.

The total waste associated with the remediation of the Bioshield Wall is small in terms of volume and radioactivity content relative to the decommissioning project estimate of 81,132 ft^3 and 1972 curies of radioactive material. (These totals include the original DP estimate plus the Control Rod Blades, LPRMs, antimony pins and beryllium sleeves addressed in response to NRC questions.) Volume reduction efforts to date on other material removed from Shoreham are such that the waste material added by removal of the Bioshield will not cause the estimated burial volume to exceed the previous project estimates.

PROJECT MAN-HOURS AND BUDGET

See discussion under Total Worker Dose Estimate for man-hr breakdown of physical work associated with Bioshield Wall removal. Engineering and supervision man-hrs are currently estimated at 8333 man-hrs resulting in a total of 16,856+8,333 = 25,189 man-hrs. The cost of this work is estimated at 2 million dollars. Because of projected under runs in the balance of the decommissioning project, however, this additional cost is not expected to result in exceeding the original decommissioning cost estimate of 186 million dollars.

CONCLUSION FOR DECOMMISSIONING PLAN REVIEW

In conclusion, the work associated with Bioshield Wall removal will not result in environmental impacts different from or exceeding those set forth in the LIPA Supplement to the SNPS Environmental Report (Decommissioning) (Ref. 5) for the following reasons:

1) There is negligible, if any, additional radioactive effluents released to the environment during decommissioning;

2) There is negligible additional radiation exposure to workers beyond that previously identified for decommissioning;

3) There is no significant amount of radioactive waste generated beyond the total originally identified for decommissioning;

4) Removal methods do not introduce any new or increased hazards which could result in exceeding the previously postulated accident impacts;

5) Non-radiological impacts such as air quality, land use, noise and dust will not be increased due to confinement of the work within the Reactor Building. Water use will be minimized through processing and recirculation of the water used to cool and lubricate the DWS blades.

PART D - REVIEW OF SAFETY QUESTIONS

A) The probability of any accident previously evaluated in the DP/DSAR/USAR will not be increased by the removal of portions of the Bioshield Wall for the following reasons:

The removal of portions of the Bioshield Wall, while not specifically discussed in any detail in the text of the DP, was considered in the accident analyses based upon similar considerations in NUREG/CR-0672, 'Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station.' (Ref. 10) Section 3.4.1.1 in the DP Accident Analysis specifically discusses the Waste Container Drop accident which is predicated upon the removal of activated concrete rubble from the Biological Shield Wall. Furthermore, the methodology to be used to segment the Bioshield Wall is also called out and discussed in the DP (Diamond Wire Saw - DWS) and hence was considered during the development of the accident analyses.

In addition, although the Bioshield Wall removal activity will be occurring during a period when fuel storage, handling and shipping activities are occurring at SNPS, the Bioshield Wall removal activities will not impact the fuel-related activities. All Bioshield Wall removal activities are physically independent of fuel-related activities. A dedicated crane will be installed to handle Bioshield Wall segment handling. Bioshield Wall removal activities will occur in the Reactor Cavity, whereas fuel-related activities are concentrated in the Spent Fuel Storage Pool, on the Refuel Floor (el.

175'-9"), and in the Reactor Building Main Equipment Hatch. The Bioshield Wall is structurally independent of the Refuel Floor and the Spent Fuel Storage Pool.

Therefore, the activity contemplated here will not increase the probability of any accident previously evaluated in the DP/DSAR/USAR.

(B) The consequences of an accident previously evaluated in the DP/DSAR/USAR will not be increased by the removal of the Bioshield Wall because the methods and techniques for the removal of portions of the Bioshield Wall are consistent with the postulated accident analyses in the DP. Additionally, the source term associated with the methodology to be used for removing the Bioshield Wall (i.e. Diamond Wire Saw Cutting) is much less than the source terms used for the various accidents considered. The radiological impact of accidents associated with the removal of the Bioshield Wall are detailed in PART F - REVIEW OF RADIOLOGICAL IMPACT and does not exceed the impact on the offsite general public already described in the DP.

(C) The possibility of an accident which is different than any already evaluated in the DP/DSAR/USAR will not be created for the reasons specified in (A) & (B) above. The accident scenario of dropping a segmented block of the Bioshield Wall is considered to be an example and subset of the already considered Waste Container Drop Accident.

(D) The possibility of a malfunction of equipment important to safety previously evaluated in the DP/DSAR/USAR will not be increased because the removal of portions of the Bioshield Wall will not affect any fuel storage or handling equipment.

The removal of portions of the Bioshield Wall will not adversely affect any safety related systems, structures, or components. (See PART G - 'SEISMIC/STRUCTURAL ANALYSIS OF BIOSHIELD WALL REMOVAL ON REACTOR BUILDING INTEGRITY' for a complete discussion of the impact). No dismantlement will be performed on elevation 175' level of the Reactor Building.

(E) The consequences of a malfunction of equipment important to safety previously evaluated in the DP/DSAR/USAR will not be increased for the reasons specified in Item (D) above.

(F) The possibility of a malfunction of equipment important to safety different than any already evaluated in the DP/DSAR/USAR will not be increased for the reasons specified in Item (D) above.

(G) The margin of safety as defined in the bases to any technical specification will not be reduced because the Bioshield Wall is not required by the Defueled Technical Specifications. The removal of portions of the Bioshield Wall by the methodology planned will not affect any fuel, radwaste storage or fuel handling equipment. Administrative controls will be in place to permit surveillances and actions to occur during implementation, to prevent the Bioshield Wall removal work from causing

violation of the ODCM, FHAR, Defueled Technical Specifications or any other technical requirements.

PART E - REVIEW OF SAFETY SIGNIFICANCE

The Bioshield Wall is classified as QA Category II and Seismic N/A in accordance with DSAR Table 3.2-1, Item XLII - Civil Structures subheading (11) - Biological Shielding. The classification is modified by a footnote which states that modifications to structures originally constructed as Seismic Category I (which the Bioshield Wall was) must be analyzed for the Design Basis Earthquake to ensure the integrity of the Reactor Building. This analysis can be found in PART G- 'SEISMIC/STRUCTURAL ANALYSIS OF BIOSHIELD WALL REMOVAL ON REACTOR BUILDING INTEGRITY'. The impact of the removal of portions of the Bioshield Wall on equipment, systems, and structures which remain installed in the plant has been evaluated and it has been determined that there will be no safety significance in the implementation of this change.

PART F - REVIEW OF RADIOLOGICAL IMPACT

The removal of the Bioshield Wall by Diamond Wire Saw cutting will not have any impact on the offsite general public beyond that already defined in the DP either through normal evolutions or accidental events occurring during the removal process based on the following analysis.

The source term to be developed for the normal evolution or for accidental mishandling of the cutting debris must include all the radioactive material to be made available for release by the direct result of the cutting process. According to NUREG/CR-0672 (Ref. 10) the airborne release associated with cutting neutron-activated piping or equipment is calculated by

$$Q'c = LKTCm$$

Where:

Q'c	=	the airborne radioactivity from cutting neutron-activated material in air [Ci]
L	=	the length of cut [m]
K	=	the width of the cut (kerf) [m]
T	=	the thickness of the material being cut [m]
Cm	=	the concentration of radioactivity in the material being cut [Ci/m^3]

SWARF CO-60 ACTIVITY ESTIMATE

LINER	
1) Kerf Dimension [m], assuming 10 mm wire = *	9.5E-03 m
2) 13 vertical cuts from el 130.25' to 107.92' =	2.2E+00 m^2
3) 4 horizontal cuts @various elevations =	2.1E+00 m^2
4) Total swarf material [cc] =	4.1E+04 cc
5) Co-60 Activity in Liner @ 25pCi/cc = **	1.0E+00 uCi
6) Total Co-60 equivalent activity @27.8pCi/cc = **	1.1E+00 uCi
CONCRETE	
1) Kerf dimension [m], assuming 10 mm wire = *	9.5E-03 m
2) 13 vertical cuts from el 130.25' to 107.92' =	5.2E+01 m^2
3) 4 horizontal cuts @various elevations =	5.5E+01 m^2
4) Total swarf material [cc] =	1.0E+06 cc
5) Co-60 activity in concrete @1.74pCi/cc = **	1.8E+00 uCi
6) Total Co-60 equivalent activity @2.05 pCi/cc = **	2.1E+00 uCi
Avg Co-60 Activity in liner = 25 pCi/cc. [90% total dose rate] **	
Avg Co-60 Activity in conc. = 1.74 pCi/cc. [85% total dose rate] **	
Total Co-60 equivalent activity in Liner and Concrete swarf =	3.2E+00 uCi
* Kerf width for 10 mm Diamond Wire Saw	
** Taken from calculation CCI-039242 Rev. 0, "Documenting Range of Activated Steel and Concrete in RPV and B-Shield" (Ref. 9)	

Under normal evolutions any airborne material would be trapped within the Reactor Building: initially by the water used as the coolant, lubricant and cleaning agent for the saw blade; secondly by the splash guard/water dam; thirdly by the local HEPA-filtered contamination control envelope or tent.

Thus, under normal circumstances there would be an insignificant release of any airborne material as a consequence of cutting and removing the Bioshield Wall and negligible exposure to the offsite general public.

Even if by some unforeseen combination of events there was an accident which caused the release of the entire potential airborne source associated with the Bioshield Wall cutting operation the Reactor Building source term would still be limited to a maximum of 3.2 μCi Co-60 equivalent. By comparison with the accident analyses in the DP any accident

associated with Bioshield Wall removal is bounded by both the existing Vacuum Filter Bag Rupture (VFBR) accident as well as the Waste Container Drop (WCD) accident (which specifically considers as a source term activated concrete rubble from the removal of a Bioshield Wall). Other accident analyses in the DP also exceed or envelope the maximum source term which could be associated with similar accidents during BioShield Wall Removal.

ACCIDENT DOSE ESTIMATE

PARAMETER	VFBR	BIOSHIELD WALL REMOVAL SCENARIO	WCD ACCIDENT
RX BLDG SOURCE TERM [uCI]	66.2	3.2	300
NUCLIDES	Co-60	Co-60 Equivalent	mixed (Co-60 @7.8 μCi)
ATMOSPHERE SOURCE TERM [uCi]	3.31E-02	1.60E-03	300
WORKER WHOLE BODY DOSE [mRem]*	1.16E-03	5.61E-05	5.36E-04
WORKER LUNG DOSE [mRem]*	1.95-01	9.43E-03	2.91E-02
SITE BOUND WHOLE BODY DOSE [mRem]**	5.36E-08	2.59E-09	6.48E-05
SITE BOUND LUNG DOSE [mRem]**	1.09E-05	5.27E-07	3.36E-03

* Due to the immediately apparent nature of these accidents, workers are expected to evacuate the affected areas within 15 minutes. Hence, these worker doses are based on 15 minute exposure times.

** Maximum individual = child (whole body) & teen (lung)

Therefore, the removal of portions of the Bioshield Wall by the methodology considered (Diamond Wire Saw Cutting) will not have any impact on the offsite general public beyond that already described in the DP.

One particular postulated scenario consists of the dropping of a single Bioshield Wall block resulting in activated concrete dust being created and made airborne in the

Reactor Building, as a direct consequence of the fall itself. It will now be shown that the source term associated with such an event is less than and enveloped by the previous evaluation presented above.

There is no surface contamination associated with any portion of the Bioshield Wall. All the radioactivity associated with the Bioshield Wall is due to activation by neutron leakage from the Reactor core. The segmented sections of the wall to be removed consist of an inner steel liner welded to an outer steel liner by connecting internal steel supports with voids filled in by cast concrete. There is no activity at all associated with the outer steel liner. The activity within the inner steel liner can not be released by any block dropping accident. The only activity which could be released by a block drop accident would be limited to the amount of activated concrete which broke free from the block itself and was converted to dust particles small enough to become airborne as result of the block drop impact itself. For the highest activated section of the Bioshield Wall the following estimate is made:

1) Largest single block will be 6′ X 7′ X 2′

2)

Inner steel liner	@	1"
concrete	@	22"
Outer steel liner	@	1"

3) Volume of concrete in block = 6′ X 7′ X 22/12′ = 77 cu. ft. = 2.18E+06cc

4) Activity of concrete due to Co-60 varies over length of interest from 10.4 to 7 pCi/cc within 1st inch of depth. At the location of interest, the first 7" of concrete shows no decrease in activity with the remaining 15" showing a fall off to zero. Co-60 activity represents 85% of the dose from concrete (rest due to Eu-152). (Based on Ref. 9).

Average Co-60 equivalent activity throughout most activated block therefore estimated at:

$$\left(\frac{100}{85}\right) x \left[\frac{7 \; x \left(\frac{10.4+7}{2}\right) \; + \; 15 \; x \left(\frac{10.4+7}{4}\right)}{22}\right] = 6.75 \; pCi/cc$$

5) Total activity of concrete in block =
2.18E+06cc X 6.75 pCi/cc = 1.47E+07 pCi = 14.70 μCi

6) Maximum postulated fraction of concrete to be converted to airborne dust as a result of block drop ≤ 10% (WCD accident uses 10% for already broken rubble).

7) Maximum airborne source term for Bioshield Wall block drop
$\leq 14.70 \div 10$
$\leq 1.47\ \mu Ci$
QED

PART G - SEISMIC/STRUCTURAL ANALYSIS OF BIOSHIELD WALL REMOVAL ON REACTOR BUILDING INTEGRITY*

During seismic event, the Bioshield Wall, which is supported on the pedestal, will transfer the earthquake loadings to the foundation. Based upon the structural analysis of Bioshield wall removal (Ref. Calculation No. CCI 039344 Rev. 0), the Reactor Building and its foundation mat will not be affected by these modifications.

[1]* This meets the intent of Note 5 on page 7 of 7 of Table 3.2-1 in SNPS DSAR, for modifications to structures originally constructed as Seismic Category I to ensure integrity of Reactor Building in a DBE.

ATTACHMENT A

MARKED-UP DSAR PAGES

(3 pages)

SE 93-587
ATTACHMENT A
Pg 1 of 3

3.7 SEISMIC DESIGN

Seismic design methods remain the same; however, hydrodynamic load effects resulting from safety relief valve discharge and loss-of-coolant-accidents are no longer applicable for a defueled reactor.

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

The design methods for seismic Category I structures such as the reactor building will remain as described in USAR Section 3.8 except that Safety Relief Valve (SRV) and LOCA hydrodynamic loads are no longer applicable to a defueled reactor. Additionally, the primary containment structure, and the containment internal structures which include primary shield wall (Bioshield wall) are no longer required to be maintained as Seismic Cat. I structures under current plant's condition. Portions of the Primary Shield Wall (Bioshield wall) have been removed.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section addresses methods and procedures used to qualify mechanical equipment. The information contained in this section is relevant only to reactor operating conditions and is, therefore, not applicable to the DSAR.

In the future, mechanical equipment will be accorded the safety significance demonstrated by the classification in Table 3.2-1 of the DSAR.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Seismic Category I equipment is identified in Table 3.2-1 and is limited to structures and equipment required to maintain the integrity of the fuel in the spent fuel pool. As discussed in Section 3.2, only the Reactor Building, fuel pool, fuel racks, and fuel handling equipment are required to be Seismic Category I. The instrumentation described in USAR Section 3.10 is no longer required to be seismically qualified.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Electrical Equipment Environmental Qualification

Purpose

The purpose of the Electrical Equipment Environmental Qualification Program for Shoreham is to provide assurance that electrical equipment important to safety as defined by 10CFR50.49 located in potentially harsh environments maintains functional operability when required to mitigate the consequences of a

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C) All visitors within the Protected Area are escorted by qualified personnel. Those visitors requiring access to the radiologically controlled area (RCA) are given an appropriately abbreviated indoctrination in protection against radiation, prior to their entry into the RCA.

(Reference USAR Section 12.1.3.1.1, Use of Individual Personnel Monitoring Devices. The basis of this change is a more stringent application of security requirements for visitors to Shoreham).

D) With the generally very low dose rates associated with the plant's defueled condition, there is no longer a requirement to have all personnel (permanent and temporary) equipped with approved dosimetry devices upon their entry to the radiologically controlled area (see Section 12.5.2.1, Access Control. Rather, only individuals working on a Radiation Work Permit (RWP) are required to use approved dosimetry devices. That the requirements of 10CFR20.202 are met by this approach will be assured by the ongoing station radiation surveillance program (as described in USAR Section 12.5.3.1), as well as the posting of thermoluminescent dosimeters (TLDs) in general access areas of the RCA.

(Reference USAR Section 12.1.3.1.1, Use of Individual Personnel Monitoring Devices. This change is justified by the low dose rates seen presently at Shoreham, and by the very low historical man-rem data in Section 12.5 of the DSAR.)

It should be noted that the Shoreham station's original physical design for radiation protection (e.g., shield walls, penetrations, sample stations, etc.) remains generally unchanged from that described in the USAR, Sections 12.1 and 12.3. Portions of the BioShield Wall have been removed. This is despite the fact that the actual source strengths and unshielded dose rates do not necessitate the degree of protection afforded. The physical design is based upon the presumption of plant operations, with the associated source terms and unshielded dose rates as described in the USAR. Although they will not generally be needed, operational considerations described in the USAR (e.g, the precautions for high dose rate jobs -- in excess of 100 mrem/hr) will be maintained.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

Fuel Sources

The Shoreham reactor core has undergone three periods of low power (0-5%) testing over the past four years. The low power tests are summarized below:

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12.3.2 Shielding

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged as it is used to develop the basic design criteria of the plant. Refer to the USAR for information on this subject. However, portions of the Primary Shield Wall (Bio Shield Wall) have been removed

12.3.3 Ventilation

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

12.3.4 Radiation Monitoring Instrumentation

In order to support the storage of the fuel in the fuel pool, SNPS will need process and effluent radiation monitoring instrumentation, and area and airborne radiation monitoring instrumentation.

Process and Effluent Radiation Monitoring System

The process and effluent radiation monitoring system is designed in accordance with General Design Criterion 64. All normal paths for release of radioactive materials are monitored to ensure compliance with the requirements of 10CFR20, 10CFR50, and Regulatory Guide 1.21.

Table 12.3.4A lists the monitors in service, and Table 12.3.4B provides data for each monitor.

Normally, nonradioactive systems that may become significantly contaminated by leaks from radioactive systems are monitored continually to ensure that no condition hazardous to the operating personnel or to the general public develops. For effluent streams that discharge to the environs, sample points are located downstream of the last point of possible radioactive fluid addition to the effluent being monitored.

All monitors in the process and effluent radiation monitoring system detect gross activity levels and readout and alarm in the main control room. Alarms in the main control room are by annunciators and cathode ray tube (CRT) display.

There are three normal effluent release points from the station that require radiation monitors: the station ventilation exhaust, the liquid radwaste effluent, and the reactor building salt water drain tank.

MEETING MINUTES

Site Review Committee Meeting

Date Issued 10/8/93

Attending:

N. Nilsen - Chairman
S. Schoenwiesner - Member
T. Cardile - Member
F. Petschauer - Member
L. Britt - Member
T. Garvey - Member
J. DeFrancesco - Alt Member
D. Durand - Alt Member
R. Pauly - Alt Member
C. Adey - Guest
W. Brown - Guest
J. Cunningham - Guest
M. Bowden - Guest
C. Thayer - Guest
K.K. Lin - Guest
W.J. Sun - Guest
S. Moss - Guest
J. Wright - Guest
J. Hauptman - Guest
R. Finn - Guest

Meeting Called to Order by Chairman:

Date: September 29, 1993

Time: 1:10 P.M.

Meeting: 93-073

Approval of Previous Minutes:

93-066 - approved
93-067 - approved
93-068 - approved
93-069 - approved

New Items

93-073-01 EIP 93-044, "Removal of RCIC Suction Line from Suppression Pool," (ECR T-00302 and Safety Evaluation 93-534, Rev. 1). Based on a presentation by W. Brown, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-02 SP 23X425.01, Rev. 2, "Primary Containment Inerting System", SPCN 93X0598. Based on a presentation by J. Cunningham, O&M Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. SRC approved deactivation of this procedure, as revised.

Item Closed

93-073-03 SP 23X716.01, Rev. 1, "Placing LRW Sample Tanks in Service", SPCN 93X0581. Based on a presentation by J. Cunningham, O&M Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-04 SP 12X001.01, Rev. 0, "Index and Organization of Station Operation Manual", SPCN 93X0751. This item revised the QA review requirements listed on the Project Procedure Status Listing (PPSL). Based on a presentation by M. Bowden, QA Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved as revised.

Item Closed

93-073-05 EIP 93-040A, "Removal of Platform Upper Levels which are Attached to Bioshield Wall", ECR T-00299A. Based on a presentation by J. DeFrancesco, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-06 EIP 93-040B, "Partial Removal of Platform at Elevation 86' to Facilitate Removal of Bioshield Wall", ECR T-00299B. Based on a presentation by J. DeFrancesco, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-07 EIP 93-045, "Bioshield Wall Upper Extension Removal", ECR T-00296. Based on a presentation by J. DeFrancesco, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-08 EIP 93-046, "Removal of RPV Segment for Salvage", ECR T-00148D. Based on a presentation by J. Sun, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-09 Safety Evaluation 93-587, Rev. 0, "Bioshield Wall Removal". Based on a presentation by S. Moss, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-10 Associated with item 93-073-09, Decommissioning Dept. is to review industrial safety issues associated with bioshield wall removal, including but not limited to wire rope cutting hazards and overhead load handling.

Item Open

93-073-11 Voluntary Change - 1101-F to the FHAR and Safety Evaluation 93-584, Rev. 1 for ECR T-00294. Based on a presentation by J. Wright, Decommissioning Dept., and R. Pauly, LRCD, this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-12 SMD 93-031, "G11 - Liquid Radwaste Phase IID", Safety Evaluation 93-558, Rev. 1. Based on a presentation by J. Hauptman and J. DeFrancesco, Decommissioning Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

93-073-13 SP 53X001.02, Rev. 3, "Pac-Nuc IF-300 Irradiated Fuel Shipping Cask Handling", TPCN 93-23. Based on a presentation by R. Finn, O&M Dept., this item has been reviewed in accordance with 10CFR50.59 and SRC has determined that there are no unreviewed safety questions. This item does not result in environmental impacts different from and exceeding those set forth in the licensee's Supplement to Environmental Report December 1990. This item was approved.

Item Closed

Meeting Adjourned by Chairman:

Time: 3:52 P.M.

Chairman