AmerGen
AmerGen Energy Company, LLC
Three Mile Island Unit 1
Route 441 South, P.O. Box 480
Middletown, PA 17057
Telephone: 717-944-7621
An Exelon/British Energy Company

10 CFR 50.90

February 22, 2001

5928-01-20026

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
ADDITIONAL INFORMATION - LICENSE AMENDMENT
REQUEST NO. 294 - REVISED STEAM GENERATOR TUBE
FAILURE ACCIDENT ANALYSIS DOSE CONSEQUENCE

Dear Sir/Madam:

The following additional information is provided to support NRC staff's review of TMI Unit 1 License Amendment Request No. 294, submitted to NRC on August 9, 2000 (5928-00-20059).

The postulated accident reanalysis described in the above referenced license amendment request assumed that while the main steam safety valves (MSSVs) are open, 5145 lbm of reactor coolant system fluid is leaked to the secondary system as a result of the steam generator tube rupture. The activity contained in this fluid is then released via the MSSVs. The assumed activity in the reactor coolant system fluid postulates operation with 1% failed fuel and the specific activities are listed in the current TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Table 14.2-4.

In addition, the reanalysis assumes a maximum of 80,000 lbm of secondary system steam is released for the two-minute MSSV opening period. Assuming the reactor coolant system concentration in UFSAR Table 14.2-4 and steady-state operation with a postulated 1 gpm primary-to-secondary leak, this secondary steam would have the following concentrations:

A001

<u>Isotope</u>	<u>Steam Activity (µCi/g)</u>
KR-83M	4.58 E-05
KR-85M	2.10 E-04
KR 85	8.42 E-04
KR 87	1.11 E-04
KR 88	3.41 E-04
XE-131M	2.31 E-04
XE-133M	3.64 E-04
XE-133	3.38 E-02
XE- 135M	4.19 E-05
XE 135	7.23 E-04
XE 138	5.97 E-05
I-131	5.31 E-04
I-132	1.78 E-04
I-133	5.64 E-04
I-134	6.98 E-05
I-135	2.86 E-04

The activity released in this additional 80,000 lbm of secondary steam is less than 0.2% of the activity in the reactor coolant system fluid assumed to be released, and therefore, is not included in the revised dose calculation.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Sincerely yours,

Mark E. Warner
Vice President - TMI Unit 1

MEW/djd/vvg

Enclosures

cc: H. J. Miller, USNRC Regional Administrator, Region I
T. G. Colburn, USNRC Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC Senior Resident Inspector, TMI Unit 1
File No. 00050

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
Additional Information - License Amendment Request No. 294

COMMONWEALTH OF PENNSYLVANIA)
) **SS:**
COUNTY OF DAUPHIN)

This additional information is submitted in support of Licensee's request to change the Updated Final Safety Analysis Report (UFSAR) for Three Mile Island Nuclear Station, Unit 1. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AmerGen Energy Company, LLC

BY: ______________________
Vice President, TMI Unit 1

Sworn and Subscribed to before me

this 22nd day of February 2001.

____
Notary Public

Notarial Seal
Suzanne C. Miklosik, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Nov. 22, 2003
Member, Pennsylvania Association of Notaries

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CALCULATION COVER SHEET

Calculation Number: C-1101-900-E000-076	Rev. # 0	System Number(s) 900	Sheet 1 of 10

Calculation Subject: TMI-1 FSAR Chapter 14 Offsite Dose Analysis for OTSG Tube Rupture – Direct-to-Atmosphere Release

	Yes	No
Is this calculation within the OQA Plan Scope? (If Yes, a verification is required)	☒	☐
Does this calculation contain assumptions / design input that requires confirmation? (If Yes, provide EDTTS No.(s))	☐	☒
Is this calculation performed as a design basis calculation? (If Yes, identify design basis parameters (section 3.3)) Offsite Dose Consequences	☒	☐

Reference Source Documents (Calculations, Safety Evaluations) (Section 4.3.1.3)

DOCUMENT NO.	REV. NO.
GPUN Calculation C-1101-900-E610-041 Rev 0, "TMI-1 Load Rejection Accident at 2772 MWt"	0
Vendor Calculation G98TM-4L, "PLG Report dated July 23, 1998 ACCIDENT X/Q VALUES FOR TMI-1"	0

APPROVALS:

		Date
Originator B. A. Parfitt	BAParfitt	5/26/99
Verification Engineer / Reviewer	R.C.Po	6/2/99
Section Manager J. W. Schmidt	JWSchmidt	6/16/99

GPU NUCLEAR

DOCUMENT NO.
C-1101-900-E000-076

TITLE TMI-1 FSAR Chapter 14 Offsite Dose Analysis for OTSG Tube Rupture – Direct-to-Atmosphere Release

REV	SUMMARY OF CHANGE	APPROVAL	DATE

N0036 (6/98)

1.0 Problem Statement

To calculate the additional maximum offsite radiation doses at the EAB during the postulated TMI-1 FSAR OTSG tube rupture accident if it is assumed that one or more of the main steam safety valves lifted during the event. The current analysis in Section 14.1.2.10 of the FSAR does not account for any releases via this pathway. The doses calculated in this analysis would be in addition to the doses presented in the FSAR analysis.

2.0 Results Summary

The results of this calculation are summarized below:

Two Hour Whole Body Dose at the EAB (Rem)	2.21E-02
Two Hour Thyroid Dose at the EAB (Rem)	7.38E+00
Total Noble Gas Released (Ci)	9.96E+02
Total Iodine Released (Ci)	4.10E+01
Total Dose Equivalent Iodine Released (Ci)	1.80E+01

3.0 References

3.1. TMI-1 FSAR Section 14.2.10

3.2. Vendor Calculation G98TM-4L, "PLG Report dated July 23, 1998 ACCIDENT X/Q VALUES FOR TMI-1"

3.3. TDR 989 Rev 1, "TMI-1 Cycle 8 Reload"

3.4. GPU Memo No. 5412-87-0187, "TMI-1 Cycle 7 Rad. Analysis Parameter Update", October 26, 1987

3.5. GPUN Calculation C-1101-900-E610-041 Rev 0, "TMI-1 Load Rejection Accident at 2772 MWt "

3.6. Report by MPR Associates, Inc., "TMI-1 OTSG Primary to Secondary Leakage", Revised September 10, 1982.

3.7. Reg Guide 1.4 Rev 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"

3.8. Lotus Note from E. Mcelwain to B. Parfitt, "Re: Power Upgrade Parameters", 10/16/96 (Attached)

4.0 Assumptions

4.1. The total primary leakage to the secondary side during the accident is 87,550 lbs per Reference 3.4. This is equivalent to the initial primary-to-seconday leak rate of 435 gpm (hot RCS), sustained for 34 minutes as stated in Tables 14.1-20 and 14.1-21 of the TMI-1 FSAR. This assumes a hot RCS density of 0.7094 g/cc per Reference 3.4.

4.2. A main steam safety is assumed to be open for approximately 2 minutes following the turbine trip, after which the bypass system takes full control (Reference 3.5). During that period, all activity in the RCS entering the secondary side is assumed to be released through the open main steam safety, directly to the atmosphere. No partitioning in the OTSG is credited.

4.3. Following closure of the main steam safety valve, all primary activity released into the secondary side is assumed to be released through the condenser offgas. This is already accounted for in the existing FSAR analysis.

4.4. Prior to the tube rupture, the unit is assumed to be operating with one percent failed fuel as assumed in Reference 3.1. The equilibrium RCS isotopic activities for this operation are per Reference 3.3, the basis for Table 14.2-4 of the TMI-1 FSAR.

4.5. The release of the equilbrium activity in the secondary side due to the one percent failed fuel and a primary-to-secondary leakage of 1 gpm prior to the tube rupture is not accounted for in this calculation since it is negligible compared to the activity released following the tube rupture. This will be demonstrated later in the calculation.

4.6. The atmospheric dispersion at the exclusion area boundary is 8.00E-4 sec/m^3 per Reference 3.2. Since the duration of the accident is only 34 minutes per Reference 3.1, doses at the LPZ boundary must be bounded by those at the exclusion area boundary doses and do not need to be calculated.

4.7. The breathing rate of the maximum exposed individual at the exclusion area boundary is assumed to be 3.47E-4 m^3/sec for the duration of the accident, per Reference 3.7.

4.8. Dose calculations are performed using the methodologies of Reference 3.7, based on ICRP-2 methodologies.

5.0 Data and Calculations

5.1 Calculation of Offsite Doses Resulting Direct-to-Atmosphere Releases During the Tube Rupture

The activity of each isotope i released to the environment direct to atmosphere , A_{di}, is calculated as follows:

$$A_{di} = (C_i)(M)(454\ g/lb)/(PF)$$

Where

C_i = Reactor coolant concentration of isotope i for 1% failed fuel (uCi/g)

M = The mass of reactor coolant transferred to the secondary side while steaming direct to atmosphere (g)

PF = Direct to atmosphere partition factor (1 for all isotopes)

For this analysis, M = (87550 lb/34 min)(454 g/lb)(2 min) = 2.34E6 g

Thyroid doses are computed using the following general equation (Reference 3.7):

$$D_{THYi} = (X/Q)(BR)(A_{di})(DCF_i)$$

Where:

D_{THYi} = Thyroid dose for the distance being evaluated from iodine isotope i (Rem)

X/Q = atmospheric dispersion factor for the time period and distance being evaluated (sec/m^3)

BR = average breathing rate for the time period (m^3/sec)

A_{di} = total amount of iodine of isotope i released direct-to-atmosphere during the time period (Curies)

DCF_i = conversion factor for isotope i (Rem/Curie)

The general equation used in the whole body dose calculations (Reference 3.7) is:

$$D_{wb} = (0.25)(X/Q)(E_i)(A_{di})$$

Where:

D_{wb} = Whole body dose for the time period and distance being evaluated (Rad)

X/Q = atmospheric dispersion factor for the time period and distance being evaluated (sec/m^3)

E_i = Average Beta-gamma energy for isotope i (MeV/dis)

A_{di} = total amount of isotope i released direct-to-atmosphere during the time period (Curies)

The results of the calculation of offsite dose from the steam generator tube rupture are summarized in the following table:

OFFSITE DOSE ANALYSIS - OTSG TUBE BREAK

REACTOR COOLANT LEAKAGE TO SECONDARY	8.755E+04 lbs
REACTOR COOLANT LEAVING VIA CONDENSOR	8.240E+04 lbs
REACTOR COOLANT LEAVING DIRECT (MSRV)	5.145E+03 lbs
ATMOSPHERIC DISPERSION X/Q	8.00E-04 sec/m^3
CONDENSOR PARTITION FACTOR FOR IODINE	100
BREATHING RATE	3.47E-04 m^3/sec
REACTOR COOLANT DENSITY	0.7094 g/cc

ISOTOPE	RCS ACTIVITY (uCi/g)	ACTIVITY RELEASED DIRECT (Ci)	AVERAGE GAMMA (MeV/dis)	THYRIOD DCF (REM/CI)	X/Q (sec/m^3)	WHOLE BODY DOSE (REM)	THYRIOD DOSE (REM)
KR-83M	5.30E-01	1.24E+00	2.60E-03		8.00E-04	6.44E-07	
KR-85M	2.43E+00	5.68E+00	1.58E-01		8.00E-04	1.79E-04	
KR-85	9.75E+00	2.28E+01	2.23E-03		8.00E-04	1.02E-05	
KR-87	1.28E+00	2.99E+00	7.93E-01		8.00E-04	4.74E-04	
KR-88	3.95E+00	9.23E+00	1.95E+00		8.00E-04	3.61E-03	
XE-131M	2.68E+00	6.26E+00	2.01E-02		8.00E-04	2.52E-05	
XE-133M	4.22E+00	9.86E+00	4.15E-02		8.00E-04	8.17E-05	
XE-133	3.92E+02	9.16E+02	4.53E-02		8.00E-04	8.29E-03	
XE-135M	4.85E-01	1.13E+00	4.31E-01		8.00E-04	9.76E-05	
XE-135	8.37E+00	1.96E+01	2.50E-01		8.00E-04	9.77E-04	
XE-138	6.92E-01	1.62E+00	1.13E+00		8.00E-04	3.64E-04	
I-131	5.71E+00	1.33E+01	3.81E-01	1.48E+06	8.00E-04	1.02E-03	5.48E+00
I-132	1.92E+00	4.49E+00	2.29E+00	5.35E+04	8.00E-04	2.06E-03	6.66E-02
I-133	6.07E+00	1.42E+01	6.07E-01	4.00E+05	8.00E-04	1.72E-03	1.57E+00
I-134	7.57E-01	1.77E+00	2.63E+00	2.50E+04	8.00E-04	9.29E-04	1.23E-02
I-135	3.08E+00	7.19E+00	1.58E+00	1.24E+05	8.00E-04	2.27E-03	2.48E-01
		1.04E+03				2.21E-02	7.38E+00

TOTAL NOBLE GAS RELEASED (Ci)	9.96E+02
TOTAL IODINE RELEASED (Ci)	4.10E+01
TOTAL DOSE EQUIVALENT IODINE RELEASED (Ci)	1.80E+01

5.2 Determination of Impact of Release of Equilibrium Secondary Side Activity

For releases of secondary steam directly to the atmosphere, the calculation of equilibrium isotopic activities in the secondary system must be calculated. In the TMI-1 FSAR, it is assumed that the plant is operating with 1% failed fuel and a primary-to-secondary leak rate of 1 gpm. The methodology for calculating secondary side activities is described in detail in Reference 3.6 and is summarized below.

Calculation of main steam activity is performed as follows:

$$N_{si} = \frac{(2Ln_{pi}/M_s)+(N_{sumpi})(M_{cond}/M_s)}{[1-(M_s-M_{cond})/M_s]}$$

Where:

N_{si} = Activity of isotope i in main steam (uCi/g)

L = Primary-to secondary leak rate (g/hr)

n_{pi} = RCS concentration of isotope i (uCi/g)

M_s = Total steam mass flow rate (g/hr)

N_{sumpi} = Concentration of isotope i in the condensate flow downstream of the powdex filters (uCi/g)

M_{cond} = Flow rate of condensate from the powdex filters (g/hr)

(M_s-M_{cond})= flow rate from high pressure heater drains which is pumped forward to the final feedwater stream (g/hr)

N_{sumpi} is a function of the equilibrium hotwell concentration. It is only significant for the iodine isotopes since essentially all noble gases are removed via the air ejectors:

$$N_{sumpi} = \frac{Ln_{pi}/[V(\Sigma_{Ri} + F\{1-1/DF\}/V)]}{DF}$$

Where:

Σ_{Ri} = Decay constant for isotope i (day^{-1})

F = Flow rate from the hotwell (cc/day)

V = Hotwell volume (cc)

DF = Decontamination factor for Powdex

Other terms are as previously defined

Calculation of feedwater activity (N_f) is performed as follows, all terms being previously defined:

$$N_f = (N_{sumpi})(M_{cond}/M_s) + (N_{si})[(M_s-M_{cond})/M_s]$$

In performing these calculations the isotopic concentrations in the primary coolant were assumed to be those developed in Section 5.1 for 1% failed fuel. Other assumptions in the calculation are summarized in the following table, with hotwell, mass steam, and condensate flows provided in Reference 3.8:

PSLR (gph)	60
PRIMARY COOLANT DENSITY (g/cc hot)	0.7094
HOTWELL FLOW (gpm)	1.66E+04
POWDEX DF	10
HOTWELL VOLUME (gal)	1.65E+05
TOTAL MASS STEAM FLOW RATE (lb/hr)	1.09E+07
CONDENSATE FROM POWDEX (lb/hr)	8.22E+06

The calculated secondary side activities are as follows:

SECONDARY SIDE ACTIVITIES

ISOTOPE	RCS ACTIVITY (uCi/g)	λ_R (day^{-1})	HOTWELL* ACTIVITY (uCi/g)	SUMP* ACTIVITY (uCi/g)	STEAM ACTIVITY (uCi/g)	FW ACTIVITY (uCi/g)
KR-83M	0.53	9.088525	0	0	4.58E-05	1.11E-05
KR-85M	2.43	3.78	0	0	2.10E-04	5.09E-05
KR-85	9.75	0.000176	0	0	8.42E-04	2.04E-04
KR-87	1.28	13.09606	0	0	1.11E-04	2.68E-05
KR-88	3.95	5.94	0	0	3.41E-04	8.27E-05
XE-131M	2.68	0.058729	0	0	2.31E-04	5.61E-05
XE-133M	4.22	0.306863	0	0	3.64E-04	8.83E-05
XE-133	392	0.131478	0	0	3.38E-02	8.20E-03
XE-135M	0.485	63.96923	0	0	4.19E-05	1.02E-05
XE-135	8.37	1.819694	0	0	7.23E-04	1.75E-04
XE-138	0.692	69.3	0	0	5.97E-05	1.45E-05
I-131	5.71	0.086087	3.82E-04	3.819E-05	5.31E-04	1.58E-04
I-132	1.92	7.359292	1.22E-04	1.216E-05	1.78E-04	5.23E-05
I-133	6.07	0.81931	4.04E-04	4.038E-05	5.64E-04	1.67E-04
I-134	0.757	18.9	4.43E-05	4.425E-06	6.98E-05	2.03E-05
I-135	3.08	2.48982	2.02E-04	2.023E-05	2.86E-04	8.47E-05

*NOBLE GAS ACTIVITIES IN HOTWELL AND SUMP NEGLIGIBLE AS THEY ESSENTIALLY COMPLETELY REMOVED VIA CONDENSOR

Per Reference 3.5, a maximum of 80,000 lbs of steam would be released during the two minute lift of the main steam safeties. Assuming this steam is in addition to the RCS mass previously assumed to be released, the activity released form the initial activity in the secondary side is as follows:

ISOTOPE	STEAM ACTIVITY (uCi/g)	SECONDARY ACTIVITY RELEASED (Ci)
KR-83M	4.58E-05	1.66E-03
KR-85M	2.10E-04	7.62E-03
KR-85	8.42E-04	3.06E-02
KR-87	1.11E-04	4.01E-03
KR-88	3.41E-04	1.24E-02
XE-131M	2.31E-04	8.40E-03
XE-133M	3.64E-04	1.32E-02
XE-133	3.38E-02	1.23E+00
XE-135M	4.19E-05	1.52E-03
XE-135	7.23E-04	2.62E-02
XE-138	5.97E-05	2.17E-03
I-131	5.31E-04	1.93E-02
I-132	1.78E-04	6.46E-03
I-133	5.64E-04	2.05E-02
I-134	6.98E-05	2.53E-03
I-135	2.86E-04	1.04E-02

Comparison of this activity with that released directly to the atmoshphere from the RCS leak, shows that the activity released via this source is less than 0.2% of the activity released from the RCS leak and can be negleglected in the offsite dose analysis.

Brad Parfitt 05/28/99 12:01 PM

To: Ardesar Irani
cc:

Subject: Re: POWER UPGRADE PARAMETERS

Reference 3.8

--------------------- Forwarded by Brad Parfitt on 05/28/99 12:00 PM ---------------------------

To: Brad Parfitt
cc:
From: Erik B Mcelwain
Date: 10/16/96 08:56:57 AM
Subject: Re: POWER UPGRADE PARAMETERS

Brad,
Rather than giving you existing FSAR numbers, I gave you existing ACTUAL numbers. Also, for the anticipated values, we assumed no MOP's back to the condenser (this is the configuration we expect). See table below....

SECONDARY SYSTEM PARAMETERS

	EXISTING FSAR VALUE	VALUE FOR 2772 MWt
HOTWELL FLOW (gpm)	1.66E+04	1.87E+04
HOTWELL VOLUME (gal)	1.65E+05	1.65E+05
TOTAL MASS STEAM FLOW RATE (lb/hr)	1.12E+07	1.19E+07
CONDENSATE FROM POWDEX (lb/hr)	7.80E+06	9.27E+06

	Existing Actual	@2772
HOTWELL FLOW	1.66 E+4	1.81 E+4
HOTWELL VOLUME		
TOTAL MASS STEAM FLOW RATE	1.085E+7	1.196E+7
CO FROM POWDEX (same as hotwell flow)	8.22 E+6	8.94 E+6

If you have any questions, please call or E-mail.
Erik

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VERIFICATION PLAN/SUMMARY SHEET (EP-006)

PLAN

Scope of Verification

Verify the assumptions, calculational methodology, and conclusions of Calculation C-1101-900-E000-076

Item No.	Method/Depth of Verification Required	Req'd Compl. Date
1	Verify the assumptions and calculational methodology used to calculate doses resulting from direct-to-atmosphere releases during and OTSG tube repture	6/7/99

Assigned Verification Engineer L. C. Po Qualified per 4.4.1.3.b ☒ Yes ☐ Waived

Justification for Waiver

Section Manager (SM) (sign) JWSchmidt J. W. Schmidt Date 6/16/99

SUMMARY

Summary of verification scope, methods, results and conclusions.

The assumptions, methodology and results are reviewed and verified. In addition, an alternate method by Scaling from the original release from the condenser is provided below:

FSAR Table 14.1.21 Condenser offgas 2 hr thyroid dose = 1.07 Rem, whole body dose = 0.31 Rem based on Iodine partition factor of 1/00 (Note, a typo of 1E-4 or 10-4 in the current FSAR) and X/Q = 6.8E-4 sec/M3.

Since direct release through the MSSV lasts 2 out of the total 34 minute release time, and the partition factor is 1, With the new X/Q = 8E-4 sec/M3, the direct dose should be:

Thyroid = 1.07 x 2/34 x 100 x 8E-4 / 6.8E-4 = 7.41 Rem

Whole body = 0.35 x 2/ 34 x 8E-4 / 6.8E-4 = 0.021 Rem

These are almost identical to the results of C-1101-900-E000-076 (7.38 Rem for thyroid and 0.021 Rem for whole Body).

Based on this evaluation, the calculation is verified to be acceptable.

Verification Engineer (print) L. C. Po (sign) L. C. Po Date 6/7/99

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CALCULATION VERIFICATION CHECKLIST
(Ref. EP-006)

Subject	Calculation No.	Rev. No.	System Nos.	Sheet:
Subject: TMI-1 FSAR Chapter 14 Offsite Dose Analysis for OTSG Tube Rupture – Direct-to-atmosphere release	C-1101-900-E000-076	0	900	10 of 10

Place an "X" in the applicable box (Yes, No, N/A) for each item.

A **"NO"** response may indicate that the design or verification is incomplete and may require an ETTS No. to be assigned by the responsible Section Manager. The Section Manager shall review each **"NO"** response to determine if the **"NO"** response requires further investigation.

A "N/A" (Not Applicable) response does **not** require any further action by the Verification Engineer.

The Verification Summary (Exhibit 7A) may be used to outline the Verification Engineer's work or to document comments that are deemed appropriate by the Verification Engineer.

ITEMS	Review Check / Design Compliance		
	Yes	No	N/A
1. **Design Input and Data** – Were the inputs correctly selected, referenced (latest revision) and incorporated into the calculation?	☒	☐	☐
2. **Assumptions** – Are assumptions necessary to perform the calculation adequately described and reasonable?	☒	☐	☐
3. **Regulatory Requirements** – Are the applicable codes and standards and regulatory requirements, including issue and addenda, properly identified and their requirements met?	☒	☐	☐
4. **Construction and Operating Experience** – Has applicable construction and operating experience been considered?	☐	☐	☒
5. **Interfaces** – Have the design interface requirements been satisfied?	☐	☐	☒
6. **Methods** – Were the inputs correctly selected, referenced (latest revision) and incorporated into the calculation?	☒	☐	☐
7. **Output** – Is the output reasonable compared to the inputs?	☒	☐	☐
8. **Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that the design requirements have been satisfactorily accomplished?	☒	☐	☐
9. **Radiation Exposure** – Has the calculation properly considered radiation exposure to the public and plant personnel?	☒	☐	☐

Comments:

APPROVALS ***(Print Name)***

Assigned Verification Engineer L. Po	Date 06/07/99
Other Verification Engineer	Date

InfoWire

AN INFORMATION SERVICE TO NEI MEMBERS

#01-06 February 22, 2001 - 1:20 p.m. (ET)

Four Groups Allege Nuclear Industry has "License to Kill" Marine Wildlife, Ocean Habitat

At a Washington, D.C. press conference today, four groups will release a report titled ""Licensed to Kill: How the Nuclear Power Industry Destroys Endangered Marine Wildlife and Ocean Habitat to Save Money." The report will be released by The Safe Energy Communication Council, Nuclear Information and Resource Service, Standing for Truth about Radiation (STAR), and The Humane Society of the United States.

NEI
NUCLEAR ENERGY INSTITUTE

SUITE 400
1776 I STREET, NW
WASHINGTON, DC
20006-3708
202.739.8000
www.nei.org

The groups summarize the report as follows:

"The routine operation of many atomic power plants unnecessarily kills marine wildlife and ocean habitat. This is documented in a major report released February 22, 2001... The 137-page full report and accompanying 29 minute video focus on the industry's evasive tactics used to avoid responsibility for the destruction of ocean habitat and marine species, with particular emphasis on endangered sea turtles, through the intake and discharge of as much as one million gallons of reactor coolant water per minute at 59 of the United States' 103 operating reactors."

According to the group's new release, the report also cites a "lack of oversight by government agencies, particularly, the Nuclear Regulatory Commission, National Marine Fisheries Service and Environmental Protection Agency." The report claims to document the nuclear industry's "use of the ecologically harmful, but relatively inexpensive once-through cooling technology responsible for devastating the marine ecosystem from New England to California."

The study specifically mentions a number of nuclear power plants, including Diablo Canyon, San Onofre, Millstone, Crystal River, St. Lucie, Calvert Cliffs and Seabrook.

The report will be available at the web sites of the participating groups: http://www.safeenergy.org, http://www.nirs.org or http://www.noradiation.org

– more –

A spokesman from the Nuclear Energy Institute is attending the press conference to distribute materials that include a statement submitted to the Nuclear Regulatory Commission in April 1999 by George Abbe, a senior scientist with the Academy of Natural Sciences Estuarine Research Center office in Maryland. Abbe, then-president of the National Shellfisheries Association, told the NRC that years of study of the Chesapeake Bay, dating back to 1968, showed "relatively little effect of power generation at the Calvert Cliffs Nuclear Power Plant on the aquatic organisms that live in the Bay immediately adjacent to the plant."

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For further information on this INFOWIRE, contact Patricia Bryant at 202.739.8020 or email pb@nei.org. To contact a Nuclear Energy Institute staffer outside of working hours, call 703.644.8805