PHILADELPHIA ELECTRIC COMPANY

2301 MARKET STREET

P.O. BOX 8699

PHILADELPHIA, PA. 19101

(215) 841-4502

JOHN S. KEMPER
VICE-PRESIDENT
ENGINEERING AND RESEARCH

AUG 14 1984

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket Nos.: 50-352
50-353

Subject: Limerick Generating Station, Units 1 and 2 Information for Auxiliary Systems Branch (ASB) and Accident Evaluation Branch (AEB) Regarding Light Loads/Fuel Handling Accident

References: (1) Telecon between L. Bell/R. E. Martin (NRC) and J. H. Arhar (PECO) on 4/24/84.
(2) Letter, J. S. Kemper (PECO) to A. Schwencer (NRC), dated 8/13/84.

File: GOVT 1-1 (NRC)

Dear Mr. Schwencer:

Attached are draft changes to FSAR Section 15.7.4 and response to RAI 410.37. These changes evaluate a more conservative fuel-handling accident than that presently described in the FSAR. As discussed in Reference (1), the resulting post accident offsite doses remain a small fraction of 10CFR100 limits.

Pursuant to our commitment noted on SER page 9-9, the evaluation contained in the revised response to Q410.37 demonstrates that the maximum kinetic energy resulting from the drop of each object weighing less than a fuel bundle and grapple assembly that could be handled over spent fuel will not result in consequences in excess of the revised fuel-handling accident.

The revised fuel-handling accident evaluation is consistently applied to the "Limerick Overhead Handling Systems Final Report" (Revision 3), transmitted to you in Reference (2).

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1/1

The information contained on these draft FSAR changes will be incorporated into the FSAR, exactly as it appears on the attachments, in the revision scheduled for September 1984.

Sincerely,

JW Gallagher
for
JS Kenyon

JHA/gra/07268405

Attachment

cc: See Attached Service List

cc:	Judge Lawrence Brenner	(w/o enclosure)
	Judge Richard F. Cole	(w/o enclosure)
	Troy B. Conner, Jr., Esq.	(w/o enclosure)
	Ann P. Hodgdon, Esq.	(w/o enclosure)
	Mr. Frank R. Romano	(w/o enclosure)
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	Charles W. Elliot, Esq.	(w/o enclosure)
	Zori G. Ferkin, Esq.	(w/o enclosure)
	Mr. Thomas Gerusky	(w/o enclosure)
	Director, Penna. Emergency Management Agency	(w/o enclosure)
	Angus R. Love, Esq.	(w/o enclosure)
	David Wersan, Esq.	(w/o enclosure)
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	Atomic Safety & Licensing Appeal Board	(w/o enclosure)
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	Ms. Phyllis Zitzer	(w/o enclosure)
	Judge Peter A. Morris	(w/o enclosure)

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QUESTION 410.37 (Section 9.1.2, 9.1.4)

Verify that the maximum potential kinetic energy resulting from dropping each object of less weight than a spent fuel assembly and its handling tool, which will be handled over spent fuel, will not exceed the effects of the fuel handling accident described in Section 15.7.4 of the FSAR. Provide a list of all objects considered and a discussion of the analysis.

RESPONSE

AND FUEL GRAPPLE ASSEMBLY 45,900

As noted in the discussion of the design basis fuel handling accident in Section 15.7.4, the maximum kinetic energy of a dropped fuel bundle AND FUEL GRAPPLE ASSEMBLY is ~~47,000~~ 45,900 ft-lb. A review has been made to determine whether there are any potential drops of loads lighter than a fuel bundle AND FUEL GRAPPLE ASSEMBLY (I.E., 1200 lb) that could have a higher kinetic energy due to a higher carrying height. The following conclusions have been reached:

No load that weighs less than ~~200~~ 540 lb can develop a higher kinetic energy than a fuel bundle AND FUEL GRAPPLE ASSEMBLY if dropped over spent fuel. This value is based on a potential energy of ~~47,000 ft/lb~~ 45,900 FT-LB with the load at the maximum lift height of the reactor enclosure crane and relative to the reactor core (worst case). The majority of light loads carried over spent fuel weigh less than ~~200~~ 540 lb.

~~With the exception of those items~~ As listed in Table 410.37-1, the potential energy of the remaining few light loads which WILL BE HANDLED OVER SPENT FUEL AND WHICH weigh more than ~~200~~ 540 lb is less than ~~47,000~~ 45,900 ft-lb because their maximum drop heights are less than the worst case. INSERT A ~~These lower drop heights occur for one or more of the following reasons:~~

DELETE

a. The load is carried only by the refueling platform hoist.

b. The load is carried only over the spent fuel pool.

c. The load is very long (i.e. the bottom of the load is close to the top of the fuel).

The following light loads may develop a higher kinetic energy than a dropped fuel bundle. The approximate potential energies at normal and maximum load carrying heights are listed, relative to the elevation of the top of the spent fuel in the core or in the spent fuel pool as appropriate.

It is reasonable to assume that the consequences of a light load drop will be no worse than those of the design basis fuel bundle drop for the following reasons:

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INSERT (A), PG 410.37-1

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INFREQUENT OR UNEXPECTED MOVEMENT OF LIGHT LOADS (I.E., WEIGHING GREATER THAN 540 lb BUT LESS THAN 1200 lb) OVER SPENT FUEL WHICH ARE NOT IDENTIFIED IN TABLE 410.37-1 WILL BE ADMINISTRATIVELY CONTROLLED ON A CASE-BY-CASE BASIS. THESE INFREQUENT OR UNEXPECTED LOAD HANDLING SITUATIONS WILL USE THE LOAD HANDLING PREPARATIONS, INSTRUCTIONS, AND EQUIPMENT BASED ON NUREG-0612 GUIDELINES, TO ASSURE THAT THE PROBABILITY OF A LOAD DROP IS EXTREMELY SMALL OR THAT THE CONSEQUENCES ARE ACCEPTABLE.

BASED ON THE ABOVE REVIEW, THE MAXIMUM KINETIC ENERGY RESULTING FROM THE DROP OF EACH OBJECT WEIGHING LESS THAN A FUEL BUNDLE AND GRAPPLE ASSEMBLY THAT COULD BE HANDLED OVER SPENT FUEL WILL NOT EXCEED THE EFFECTS OF THE FUEL HANDLING ACCIDENT DESCRIBED IN SECTION 15.7.4.

a. The actual kinetic energy developed during the drop of the light loads above will be less than their maximum potential energy due to bouyancy and drag of the water over the fuel. While no calculations have been made, reductions in kinetic energy due to drag should be significant for steam line plug and the in-vessel storage rack due to their relatively large surface areas.

b. The loads will usually be carried at less than maximum height (i.e., the bottom of the load will normally be near the refueling floor or bottom of the reactor well, as applicable).

c. As discussed in Section 15.7.4, all of the fuel rods in the dropped spent fuel bundle are assumed to fail, representing 50% of the resulting fission product release. Because no fission products are released from a dropped light load (including an unirradiated new fuel bundle), all releases must come from the impacted spent fuel. Thus, for the case of a light load drop, the impacted fuel can absorb more energy without exceeding the releases calculated for the spent fuel bundle drop.

d. Some of the impact energy would be absorbed by components other than the spent fuel or the dropped load (e.g., the fuel storage rack, core top guide or other impacted items) that would further reduce the energy available to cause fuel failure.

The three light loads listed in Table 410.37-1 will be treated as heavy loads in accordance with the guidelines of NUREG-0612, until calculations are performed which demonstrate that the effects of dropping these objects will not exceed the effects of the fuel handling accident described in Section 15.7.4 of the FSAR.

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Table 410.37-1

Load	Approx. Combined Wt., Handling Tool Plus Load (lb)	Potential Energy, (ft-lb) Normal Height	Max. Height
1) New fuel bundle or dummy bundle [Reactor enclosure crane relative to spent fuel pool]	700	21,000	29,000
2) In-Vessel storage rack [Refueling platform hoist relative to core]	600	21,000	33,000
3) Steam line plug and installing tool [Reactor enclosure crane relative to core]	450	16,000	33,000

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15.7.4 FUEL-HANDLING ACCIDENT

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel-handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto other fuel bundles. A variety of events that qualify for the class of accidents termed "fuel-handling accidents" has been investigated. The accident that produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle AND THE FUEL GRAPPLE ASSEMBLY OF THE REFUELING PLATFORM into the reactor core when the reactor vessel head is off. THE FUEL GRAPPLE ASSEMBLY CONSISTS OF A TELESCOPIC MAST AND HEAD ASSEMBLY.

15.7.4.1.2 Frequency Classification

This accident is categorized as a limiting fault.

15.7.4.2 Sequence of Events and System Operation

15.7.4.2.1 Sequence of Events

The sequence of events following this failure is shown in Table 15.7-15.

15.7.4.2.2 Identification of Operator Actions

The operator actions are as follows:

a. Initiate the evacuation of the refueling area and the locking of the refueling area doors.

b. The supervisor in charge of fuel handling should instruct his employees to go immediately to the radiation protection personnel decontamination area.

c. The supervisor in charge of fuel handling will alert the control room operator to the accident.

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d. Determine if the normal ventilation system has isolated and the SGTS is in operation.

e. Initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the refueling area.

f. Appropriate radiological control methods should be implemented at the entrance of the refueling area.

g. Before entering the refueling area, a careful study of conditions, radiation levels, etc, will be performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the SGTS. Operation of other plant or RPS or ESF systems is not expected.

15.7.4.2.4 The Effect of Single Failures and Operator Errors

The automatic ventilation isolation system includes the

radiation monitoring detectors and isolation valves. The SGTS is designed to the single failure criterion and safety requirements. Refer to Sections 7.6, 9.4 and 15.10 for further details.

15.7.4.3 <u>Core and System Performance</u>

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic yet conservative assessment of the consequences.

→ INSERT ATTACHED PAGE →

To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

INSERT, pg 15.7-11 DRAFT

CALCULATIONS WERE PERFORMED TO EVALUATE THE CONSEQUENCES OF A VARIETY OF DROP SCENARIOS. THE WORST-CASE SCENARIO WHICH PRODUCED THE GREATEST NUMBER OF FAILED FUEL RODS IS REPRESENTED BY THE FUEL BUNDLE AND GRAPPLE ASSEMBLY FALLING AS TWO SEPARATE AND INDEPENDENT UNITS FROM THEIR RESPECTIVE FULLY RAISED HEIGHTS.

THE CONSEQUENCES OF A DROP SCENARIO CONSISTING OF A FUEL BUNDLE AND GRAPPLE ASSEMBLY FALLING AS A SINGLE UNIT ARE BOUNDED BY THE ABOVE WORST-CASE SCENARIO, MAINLY BECAUSE TWICE AS MANY FUEL RODS ARE STRUCK (AND CONSEQUENTLY FAIL DUE TO BENDING) BY THE DROP OF TWO INDEPENDENT UNITS.

THE CALCULATED RESULTS OF THE WORST-CASE SCENARIO ARE VERY CONSERVATIVE BECAUSE BOTH OF THE FOLLOWING UNLIKELY EVENTS WOULD HAVE TO OCCUR:

a. THE FUEL BUNDLE BECOMES DETACHED FROM THE GRAPPLE BY EITHER A BREAK OF THE BAIL HANDLE OR GRAPPLE, AND
b. THE GRAPPLE ASSEMBLY BECOMES DETACHED FROM THE REFUELING PLATFORM BY EITHER A BREAK OF THE CABLES OR A BREAK OF THE CABLE SUPPORT EYE BRACKET.

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The fuel assembly ~~is~~ AND GRAPPLE ASSEMBLY ARE expected to impact on the reactor core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based upon 1% uniform plastic deformation of the rods). The energy of the dropped assembly ASSEMBLIES is conservatively assumed to be absorbed by only the cladding and other core structures. Because a fuel assembly consists of 72% fuel, 11% cladding, and 17% other structural material by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass energy calculations that follow.

The energy absorption on successive impacts is estimated by considering a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are listed below:

a. The fuel assembly ~~is~~ AND GRAPPLE ASSEMBLY ARE dropped from the maximum height allowed by the ~~fuel-handling equipment (less than 30 feet)~~ REFUELING PLATFORM (32 AND 47 FEET, RESPECTIVELY, WHEN HANDLING FUEL) AS TWO SEPARATE AND INDEPENDENT UNITS.

b. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly AND GRAPPLE ASSEMBLY in the water above the core. and requires the complete detachment of the assembly from the fuel-hoisting

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equipment. This is only possible if the fuel assembly handle, the fuel grapple, or the grapple cable break.

c. None of the energy associated with the dropped ~~fuel~~ assembl~~y~~IES is absorbed by the fuel material (uranium dioxide).

d. The minimum water depth between the top of the fuel rods and the fuel pool surface is 23 feet.

e. Maximum fuel rod pressurization is 2072 psia.

f. The peak linear power density for the highest power assembly discharged is 13.4 kW/ft and the corresponding maximum centerline operating fuel temperature is 3412°F.

g. BECAUSE THE WEIGHTS OF THE FUEL ASSEMBLY AND GRAPPLE ASSEMBLY ARE SIMILAR, THE FRACTIONAL ENERGY LOSSES ARE ASSUMED TO BE THE SAME FOR BOTH ASSEMBLIES.

15.7.4.3.3 Results

15.7.4.3.3.1 Energy Available

Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling PLATFORM ~~equipment~~ (~~less than 30~~ 32 feet) results in an impact velocity of ~~30~~ 45.4 ft/sec. DROPPING THE FUEL GRAPPLE ASSEMBLY ONTO THE REACTOR CORE FROM THE MAXIMUM HEIGHT ALLOWED BY THE REFUELING PLATFORM (47 FEET) RESULTS IN AN IMPACT VELOCITY OF 55.0 FT/SEC. The TOTAL kinetic energy acquired by the falling ~~fuel assembly is less than 17,000~~ ASSEMBLIES IS APPROXIMATELY 45,900 ft-lb and is dissipated in one or more impacts.

15.7.4.3.3.2 Energy Loss Per Impact

Based upon the fuel geometry EACH in the reactor core AND THE LONG NARROW SHAPE OF BOTH DROPPED ASSEMBLIES, four fuel assemblies are struck by ~~the~~ dropped assembly. The fractional energy loss on the first impact is approximately 80%.

The second EACH impact is expected to be less direct. The broadside of ~~the~~ dropped assembly impacts approximately 24 more fuel assemblies, so that after the second impact ~~only 136~~ LESS THAN 240 ft-lb (approximately 1% of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact.

THE TOTAL KINETIC ENERGY EQUALS THE SUM OF THE KINETIC ENERGY OF THE DROPPED FUEL ASSEMBLY (32 FEET x 700 LB = 22,400 FT-LB) AND THE DROPPED FUEL GRAPPLE ASSEMBLY (47 FEET x 500 LB = 23,500 FT-LB).

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If EACH ~~the~~ dropped ~~fuel~~ assembly strikes only one or two fuel assemblies on ITS ~~the~~ first impact, the energy absorption by the core support structure results in approximately the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

First impact	80%
Second impact	19%
Third impact	1% (no cladding failures).

15.7.4.3.3.3 Fuel Rod Failures

15.7.4.3.3 3.1 First Impact Failures

The first impacts dissipate 0.80 x ~~17,000~~ 45,900 or ~~13,600~~ 36,720 ft-lb of energy. It is assumed that 50% of this energy is absorbed by the dropped fuel assembly and that the remaining 50% is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the eight tie rods of each struck fuel assembly are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus 4 x 8 = 32 2x4x8=64 tie rods (total in four assemblies STRUCK BY EACH OF THE TWO ASSEMBLIES) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of ~~the~~ EACH GROUP OF four struck assemblies, 250 x 54 x 4 or 54,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

ADDITIONALLY, IT IS CONSERVATIVELY ASSUMED THAT 0% OF THIS ENERGY IS ABSORBED BY THE DROPPED GRAPPLE ASSEMBLY AND 100% IS ABSORBED BY THE STRUCK FUEL ASSEMBLIES.

DROPPED FUEL ASSEMBLY : $\frac{0.5 \times \cancel{43,600}\ (0.8 \times 22,400)}{250} \times \frac{11}{11 + 17}$ = ~~11~~ 14

DROPPED FUEL GRAPPLE ASSEMBLY : $\frac{1.0 \times 0.8 \times 23,500}{250} \times \frac{11}{11+17} = \frac{30}{44}$

Thus, during the first impact, fuel rod failures are as follows:

Dropped fuel assembly	62	rods (bending)
Struck assemblies	64 ~~32~~	tie rods (bending)
Struck assemblies	44 ~~11~~	rods (compression)
	~~105~~ 170	failed rods

15.7.4.3.3.3.2 Second Impact Failures

Because of the less severe nature of the second impact and the distorted shape of ~~the~~ EACH dropped ~~fuel~~ assembly, it is assumed that in only two of the 24 struck assemblies are the tie rods subjected to bending failure. ~~Thus 2 x 8 = 16~~ BECAUSE EACH DROPPED ASSEMBLY STRIKES 24 FUEL ASSEMBLIES, 2 x 2 x 8 = 32 tie rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

DROPPED FUEL ASSEMBLY : $\frac{\frac{0.19}{2} \times \cancel{17,000}\ 22,400}{250} \times \frac{11}{11 + 17} = 3$

DROPPED FUEL GRAPPLE ASSEMBLY : $\frac{1.0 \times 0.19 \times 23,500}{250} \times \frac{11}{11+17} = 7 / 10$

Thus, during the second impact, the fuel rod failures are as follows:

Struck assemblies	32 ~~16~~	tie rods (bending)
Struck assemblies	10 ~~3~~	rods (compression)
	42 ~~19~~	failed rods

15.7.4.3.3.3.3 Total Failures

The total number of failed rods resulting from the accident is as follows:

First impact	170 ~~105~~	rods
Second impact	42 ~~19~~	rods
Third impact	0	rods
	~~124~~ 212	total failed rods

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15.7.4.4 Barrier Performance

The reactor coolant pressure boundary and primary containment are assumed to be open. The transport of fission products from the refueling area is discussed in Sections 15.7.4.5.2.1 and 15.7.4.5.2.2.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. The first is based upon conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR, Part 100. This analysis is referred to as the "design basis analysis."

b. The second analysis is based upon assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis."

For both analyses, the fission product inventory in the fuel rods assumed to be damaged is based upon 1000 days of continuous operation at 3458 MW. A 24-hour period for decay from the above power condition is assumed, because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown. Figure 15.7-1 indicates the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based upon NRC Standard Review Plan 15.7.4 and NRC Regulatory Guide 1.25. The specific models and assumptions and the program used for computer evaluation are described in Section 15.10. Specific values of parameters used in the evaluation are presented in Table 15.7-16.

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15.7.4.5.1.1 Fission Product Release from Fuel

The fission product inventory of a core average rod is adjusted by a peaking factor of 1.5 to establish the inventory of each damaged rod. Ten percent of the noble gases inventory (30% for Kr-85) and 10% of the iodine inventory are assumed to be released to the reactor water and water in the reactor well. The activity airborne in the refueling area is presented in Table 15.7-17.

15.7.4.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of mixing in the reactor well, migration from the flooded well to the refueling area atmosphere, and release to the environment through the SGTS. All of the noble gas and 1% of the iodines in the flooded well are assumed to become airborne in the refueling area.

The airborne activity is released to the environment over a 2-hour period after filtration by the standby gas treatment system (SGTS) (99% removal efficiency for iodine).

The release of activity to the environment is presented in Table 15.7-18.

15.7.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-21 and are well within the guidelines of 10 CFR, Part 100.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based upon a realistic but still conservative assessment of this accident. The specific models and assumptions and the program used for computer evaluation are described in Ref 15.7-1. Specific values of parameters used in the evaluation are presented in Table 15.7-16.

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15.7.4.5.2.1 Fission Product Release from Fuel

Fission product release estimates for the fuel-handling accident are based on the following assumptions:

a. The reactor fuel has an average irradiation time of 1000 days at 105% nuclear boiler rated (NBR) up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hour decay period allows time to shut down the reactor, depressurize the nuclear system, remove the reactor vessel head, and remove the reactor vessel upper internals. It is not expected that these operations could be accomplished in less than 24 hours and probably will require at least 48 hours.

b. An average of 1.8% of the noble gas activity and 0.32% of the halogen activity is in the fuel rod plena and available for release. This assumption is based upon fission product release data from defective fuel experiments (Ref 15.7-2).

c. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

d. It is assumed that ~~124~~ 212 fuel rods fail. This is considered to be conservative, because it is expected that fewer than ~~124~~ 212 rods would be damaged.

15.7.4.5.2.2 Fission Product Transport to the Environment

The following assumptions and conditions are used in calculating the release of activity to the environment.

a. All of the noble gases released to the fuel pool become airborne in the refueling area.

b. The iodine activity airborne is in proportion to the partition factor and the ratio of the volume of the refueling area (va) to the volume of fuel pool water

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above the core (Vw). It is assumed that a partition factor of 100 and Va/Vw of 10 is applicable for this event. It should be noted that the volume assumed for Va is not equal to the total volume of air in the refueling area, but is a conservative estimate of the volume of air that may form an equilibrium condition with the activity in the refueling pool.

c. The ventilation rate from the refueling area to the environment through the SGTS is 0.5 volume change per day (99% removal efficiency for iodine), assuming 764 scfm inleakage to the refueling area.

Based upon these assumptions, the activity airborne in the refueling area is as shown in Table 15.7-19.

The release rate of activity under normal ventilation conditions is sufficient to cause a trip of the refueling area discharge plenum radiation monitors, which results in refueling area isolation and SGTS startup.

The activity released to the environment is presented in Table 15.7-20.

15.7.4.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-21 and demonstrate the margin of conservatism in the design basis analysis.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

The spent fuel cask will be equipped with redundant sets of lifting lugs and yokes compatible with the single-failure-proof reactor enclosure crane and main hook, thus precluding a cask drop due to a single failure. Therefore, an analysis of the spent fuel cask drop is not required. Refer to Section 9.1.5 for a description of the reactor enclosure crane and the interlocks that prevent moving the spent fuel cask over the fuel pool.

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15.7.6 REFERENCES

15.7-1 Nguyen, D., "Realistic Accident Analysis - RELAC Code," October 1977 (NEDO-21143).

15.7-2 Horton, N.R., W.A. Williams, and J.W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," March 1969 (APED 5756).

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TABLE 15.7-15

SEQUENCE OF EVENTS FOR FUEL-HANDLING ACCIDENT

TIME-MIN	EVENT
0	Fuel assembly is being handled by refueling equipment. The FUEL assembly AND FUEL GRAPPLE ASSEMBLY ~~drops~~ drop onto the top of the core.
0	Some of the fuel rods in both the dropped assembly and reactor core are damaged, resulting in the release of gaseous fission products to the reactor coolant and eventually to the refueling area atmosphere.
< 1	The refueling area ventilation radiation monitoring system alarms to alert plant personnel, isolates the ventilation system and starts operation of the SGTS.
< 5	Operator actions begin.

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TABLE 15.7-16 (Page 1 of 2)

FUEL-HANDLING ACCIDENT: PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	DESIGN BASIS ASSUMPTIONS	REALISTIC BASIS ASSUMPTIONS
I. Data and Assumptions used to Estimate Radioactive Source from Postulated Accidents		
A. Power Level	3458	3458
B. Radial Peaking Factor	1.5	1.0
C. Fission Products Released From Fuel (fuel damaged)	~~124~~ 212 rods	~~124~~ 212 rods
D. Release of Activity by Nuclide	Table 15.7-17	Table 15.7-19
E. Iodine Fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor Coolant Activity Before the Accident	NA	NA
II. Data and Assumptions Used to Estimate Activity Released		
A. Primary Containment Leak Rate (%/day)	NA	NA
B. Secondary Containment (REFUELING AREA) Release Rate (%/day)	100% for 2 hours	50%/day
C. Valve Movement Times	NA	NA
D. Adsorption and Filtration		
(1) Organic iodine	99%	99%
(2) Elemental iodine	99%	99%
(3) Particulate iodine	99%	99%
(4) Particulate fission products	NA%	NA
E. Containment Spray Parameters (flow rate, drop size, etc)	NA	NA
F. Refueling Area Containment Volumes (ft^3)	2.2×10^6	2.2×10^6
G. All Other Pertinent Data and Assumptions	None	None
III. Dispersion Data		
A. EAB/LPZ Distances (m)	731/2043	731/2043

NO CHANGE

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TABLE 15.7-16 (Cont'd) (Page 2 of 2)

	DESIGN BASIS ASSUMPTIONS	REALISTIC BASIS ASSUMPTIONS
B. X/Qs for Time Intervals of		
(1) 0-2 hrs - EAB	2.9×10^{-4}	1.2×10^{-4}
(2) 0-8 hrs - LPZ	4.0×10^{-3}	2.0×10^{-5}
(3) 8-24 hrs - LPZ	NA	1.6×10^{-5}
(4) 1-4 days - LPZ	NA	9.0×10^{-6}
(5) 4-30 days - LPZ	NA	4.2×10^{-6}
IV. Dose Data		
A. Method of Dose Calculation	Section 15.10	Ref 15.7-1
B. Dose Conversion Assumptions	Section 15.10	Ref 15.7-1
C. Peak Activity Concentrations in Containment	NA	NA
D. Doses	Table 15.7-21	Table 15.7-21

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TABLE 15.7-17

FUEL-HANDLING ACCIDENT:
ACTIVITY AIRBORNE IN REFUELING AREA

DESIGN BASIS ANALYSIS

ISOTOPE	ACTIVITY (Ci)
I-131	5.25 ~~2.07~~ x 10+2
I-132	5.47 ~~3.20~~ x 10-1
I-133	5.97 ~~3.49~~ x 10+2
I-134	-
I-135	1.01 ~~5.95~~ x 10+~~9~~2
Kr-83m	1.33 ~~2.80 x 10-1~~
Kr-85m	5.85 ~~3.42~~ x 10+2
Kr-85	2.60 ~~1.52~~ x 10+3
Kr-87	9.75 ~~5.70~~ x 10-2
Kr-88	1.81 ~~1.06~~ x 10+2
Kr-89	-
Xe-131m	4.09 ~~2.39~~ x 10+2
Xe-133m	2.46 ~~1.44~~ x 10+3
Xe-133	1.15 ~~6.72~~ x 10+~~4~~5
Xe-135m	-
Xe-135	2.02 ~~1.18~~ x 10+4
Xe-137	-
Xe-138	-

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TABLE 15.7-18

FUEL-HANDLING ACCIDENT:
ACTIVITY RELEASED TO THE ENVIRONMENT

DESIGN BASIS ANALYSIS

ISOTOPE	ACTIVITY (Ci)
I-131	5.25 ~~2.07~~
I-132	5.47 ~~3.20~~ x 10^{-3}
I-133	5.97 ~~3.49~~
I-134	-
I-135	1.01 ~~5.93 x 10^{-1}~~
Kr-83m	1.33 ~~7.80 x 10^{-1}~~
Kr-85m	5.85 ~~3.42~~ x 10^{+2}
Kr-85	2.60 ~~1.52~~ x 10^{+3}
Kr-87	9.75 ~~5.70~~ x 10^{-2}
Kr-88	1.81 ~~1.06~~ x 10^{+2}
Kr-89	-
Xe-131m	4.09 ~~2.39~~ x 10^{+2}
Xe-133m	2.46 ~~1.44~~ x 10^{+3}
Xe-133	1.15 ~~6.72~~ x $10^{+\cancel{4}5}$
Xe-135m	-
Xe-135	2.02 ~~1.18~~ x 10^{+4}
Xe-138	-

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TABLE 15.7-19

FUEL-HANDLING ACCIDENT:
ACTIVITY AIRBORNE IN REFUELING AREA(1)

REALISTIC ANALYSIS

ISOTOPE	2 HRS	8 HRS	1 DAY	4 DAYS	30 DAYS
I-131	4.07 ~~2.38~~ x 10^{1}	3.11 ~~1.82~~ x 10^{1}	1.51 ~~8.87~~ x 10^{1}	5.803 ~~3.35~~ x 10^{-1}	3.15 ~~1.84~~ x 10^{-13}
I-132	2.94 ~~1.72~~	3.83 ~~2.24~~ x 10^{-1}	1.69 ~~9.88~~ x $10^{-\cancel{4}3}$	4.22 ~~2.47~~ x 10^{-14}	0.0
I-133	7.32 ~~4.28~~	4.67 ~~2.73~~	1.41 ~~8.26 x 10^{-1}~~	6.48 ~~3.79~~ x 10^{-3}	0.0
I-134	1.19 ~~6.95~~ x $10^{-\cancel{8}7}$	7.98 ~~4.67~~ x 10^{-10}	1.29 ~~7.52~~ x $10^{-1\cancel{6}5}$	0.0	0.0
I-135	6.65 ~~3.89~~ x 10^{-1}	2.97 ~~1.65~~ x 10^{-1}	2.75 ~~1.61~~ x 10^{-2}	8.24 ~~4.82~~ x 10^{-7}	0.0
Kr-83m	3.35 ~~1.96~~ x 10^{-1}	2.94 ~~1.72~~ x 10^{-2}	4.50 ~~2.63~~ x 10^{-5}	9.56 ~~5.59~~ x 10^{-18}	0.0
Kr-85	8.29 ~~4.85~~ x 10^{2}	6.46 ~~3.78~~ x 10^{2}	3.32 ~~1.94~~ x 10^{2}	1.65 ~~9.64~~ x 10^{1}	8.34 ~~4.88~~ x 10^{-11}
Kr-85m	2.21 ~~1.29~~ x 10^{1}	6.62 ~~3.87~~	2.67 ~~1.56~~ x 10^{-1}	1.41 ~~8.22~~ x $10^{-\cancel{8}7}$	0.0
Kr-87	7.544 ~~4.41~~ x 10^{-4}	2.39 ~~1.40~~ x 10^{-5}	2.44 ~~1.43~~ x 10^{-9}	0.0	0.0
Kr-88	3.13 ~~1.83~~	5.44 ~~3.18~~ x 10^{-1}	5.09 ~~2.98~~ x 10^{-3}	3.80 ~~2.22~~ x 10^{-12}	0.0
Kr-89	1.29 ~~7.58~~ x $10^{-1\cancel{7}6}$	0.0	0.0	0.0	0.0
Xe-131m	9.63 ~~5.63~~ x 10^{1}	7.39 ~~4.32~~ x 10^{1}	3.66 ~~2.14~~ x 10^{1}	1.53 ~~8.94 x 10^{0}~~	1.73 ~~1.01~~ x 10^{-12}
Xe-133m	1.27 ~~7.44~~ x $10^{\cancel{2}3}$	9.18 ~~5.37~~ x 10^{2}	3.86 ~~2.26~~ x 10^{2}	7.76 ~~4.54~~	1.56 ~~9.13~~ x $10^{-1\cancel{3}4}$
Xe-133	1.12 ~~6.54~~ x $10^{\cancel{3}4}$	8.43 ~~4.93~~ x 10^{3}	3.97 ~~2.32~~ x 10^{3}	1.33 ~~7.78~~ x $10^{\cancel{1}2}$	2.22 ~~1.30~~ x 10^{-11}
Xe-135m	7.71 ~~4.51~~ x 10^{-3}	6.87 ~~4.02~~ x 10^{-10}	0.0	0.0	0.0
Xe-135	2.97 ~~1.74~~ x 10^{3}	1.47 ~~8.59~~ x $10^{\cancel{2}3}$	2.24 ~~1.31~~ x 10^{2}	4.68 ~~2.74~~ x 10^{-2}	0.0
Xe-137	1.08 ~~6.35~~ x $10^{-1\cancel{5}4}$	0.0	0.0	0.0	0.0
Xe-138	3.56 ~~2.08~~ x 10^{-7}	1.17 ~~6.85~~ x $10^{-1\cancel{4}3}$	0.0	0.0	0.0
TOTAL	1.64 ~~9.61~~ x $10^{\cancel{3}4}$	1.16 ~~6.77~~ x $10^{\cancel{3}4}$	4.96 2.90 x 10^{3}	1.60 ~~9.35~~ x $10^{\cancel{1}2}$	1.08 ~~6.31~~ x $10^{-1\cancel{1}0}$

(1) Units for activities are in curies.

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TABLE 15.7-20

FUEL HANDLING ACCIDENT:
ACTIVITY RELEASED TO THE ENVIRONMENT(1)

REALISTIC ANALYSIS

ISOTOPE	0-2 HR	2-8 HR	8-24 HR	1-4 DAYS	4-30 DAYS
I-131	3.56 2.08×10^{-2}	8.92×10^{-2}	1.478×10^{-1}	1.332×10^{-1}	5.22×10^{-3}
I-132	3.50 2.05×10^{-3}	3.13×10^{-3}	4.702×10^{-4}	2.071×10^{-6}	5.18×10^{-17}
I-133	6.58 3.85×10^{-3}	1.47×10^{-2}	1.81×10^{-2}	7.834×10^{-3}	3.61×10^{-5}
I-134	2.55 1.49×10^{-10}	5.903×10^{-11}	4.002×10^{-13}	6.433×10^{-19}	0.0
I-135	6.43 3.76×10^{-4}	1.11×10^{-3}	7.254×10^{-4}	7.934×10^{-5}	2.381×10^{-9}
Kr-83m	4.31 2.52×10^{-2}	3.15×10^{-2}	3.031×10^{-3}	4.63×10^{-6}	9.835×10^{-19}
Kr-85	7.20 4.21×10^{1}	1.831×10^{2}	3.15×10^{2}	3.15×10^{2}	1.659×10^{1}
Kr-85m	2.27 1.33	3.21	1.32	5.542×10^{-2}	2.92×10^{-8}
Kr-87	1.18 6.89×10^{-4}	5.283×10^{-5}	1.74×10^{-6}	1.78×10^{-10}	0.0
Kr-88	3.54 2.07×10^{-1}	3.692×10^{-1}	7.684×10^{-2}	7.27×10^{-4}	5.44×10^{-13}
Kr-89	1.01 5.88×10^{-7}	4.10×10^{-19}	0.0	0.0	0.0
Xe-131m	8.39 4.91	2.12×10^{1}	3.54×10^{1}	3.30×10^{1}	1.44
Xe-138m	1.12 6.55×10^{2}	2.72×10^{2}	4.10×10^{2}	2.91×10^{2}	5.97
Xe-133	9.78 5.72×10^{2}	2.44×10^{3}	3.952×10^{3}	3.39×10^{3}	1.18×10^{2}
Xe-135m	2.65 1.55×10^{-2}	1.19×10^{-4}	1.06×10^{-11}	0.0	0.0
Xe-135	2.79 1.63×10^{2}	5.333×10^{2}	4.41×10^{2}	7.42×10^{1}	1.66×10^{-2}
Xe-137	8.33 4.87×10^{-8}	4.22×10^{-17}	0.0	0.0	0.0
Xe-138	8.57 5.01×10^{-7}	5.953×10^{-9}	1.97×10^{-15}	0.0	0.0
TOTAL	1.45 8.49×10^{3}	3.452×10^{3}	5.15×10^{3}	4.10×10^{3}	1.42×10^{2}

(1) Units for activities are in curies.

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TABLE 15.7-21

FUEL-HANDLING ACCIDENT: RADIOLOGICAL EFFECTS

DESIGN BASIS ANALYSIS

	WHOLE-BODY DOSE (rem)	INHALATION DOSE (rem)
Exclusion Area Boundary (731 meters - 2-hr dose)	7.18 ~~4.2~~ x 10^{-1}	~~5.38~~ 9.54 x 10^{-1}
Low Population Zone (2043 meters - 2-hr[1] dose)	9.93 ~~5.61~~ x 10^{-2}	~~3.12~~ 1.32 x $10^{-~~2~~ 1}$

REALISTIC ANALYSIS

	WHOLE-BODY DOSE (rem)	INHALATION DOSE (rem)
Exclusion Area Boundary (731 meters - 2-hr dose)	3.16 ~~1.85~~ x 10^{-3}	~~1.04~~ 2.29 x 10^{-3}
Low Population Zone (2043 meters - 30-day[1] dose)	3.04 ~~1.30~~ x 10^{-3}	~~1.40~~ 2.39 x 10^{-3}

[1] Section 15.7-4 gives a discussion of accident duration times

NO CHANGE

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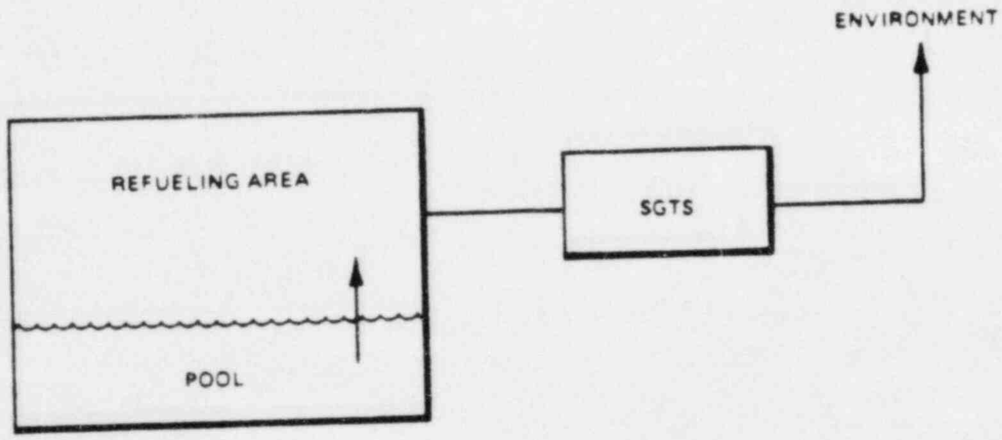
info only



LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

LEAKAGE PATH FOR FUEL-HANDLING ACCIDENT

FIGURE 15.7-1 REV. 15, 12/82