Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

October 30, 2000

L-2000-222
10 CFR 50.90

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

RE: St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Personnel Airlock Open During Core Alterations

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) proposes to revise the St. Lucie Unit 1 Technical Specification (TS) 3.9.4, Containment Penetrations. TS 3.9.4 requires a personnel airlock (PAL) door to be closed during core alterations or movement of irradiated fuel within containment. The proposed change would allow both containment PAL doors to be open during core alterations and movement of irradiated fuel in containment provided: (a) that at least one personnel airlock door is capable of being closed; (b) the plant is in Mode 6 with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

Attachment 1 is a description of the change and Safety Analysis in support of the proposed amendment. Attachment 2 is the Determination of No Significant Hazards Consideration. Attachment 3 is a marked up copy of the proposed Technical Specification change. Attachment 4 is a copy of the revised fuel handling accident analysis, F-FSA-C-000001, Revision 0, *Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 1 Tech Spec 3.9.4*, prepared by Westinghouse Nuclear Systems.

The proposed amendment has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

A001

Approval of this proposed license amendment is requested by February 28, 2001 to support planning for the spring 2001 Unit 1 refueling outage (SL1-17).

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. William A. Passetti, Florida Department of Health and Rehabilitative Services

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Rajiv S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 30 day of October, 2000
by Rajiv S. Kundalkar, who is personally known to me.

Leslie J. Whitwell
Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commissioned Name of Notary Public)

ATTACHMENT 1

SAFETY ANALYSIS

INTRODUCTION

Florida Power & Light Company (FPL) proposes to revise the St. Lucie Unit 1 Technical Specification 3.9.4, Containment Penetrations. Currently TS 3.9.4 requires a personnel airlock (PAL) door to be closed during core alterations or movement of irradiated fuel within containment. The proposed change would allow both containment PAL doors to be open during core alterations and movement of irradiated fuel in containment provided that: (a) at least one personnel airlock door is capable of being closed; (b) the plant is in MODE 6 with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

BACKGROUND

Technical Specification 3.9.4.b, Containment Penetrations, requires at least one containment airlock door to be closed during core alterations and fuel movements (MODE 6). The basis for this requirement is to limit the consequences outside containment of a fuel handling accident inside containment. The consequences of the fuel handling accident (FHA) for the reactor containment building are bounded by the effects of the accident occurring in the fuel handling building since this building does not have an isolation system like the one installed in the reactor containment building. A reanalysis of the FHA was required with the assumption that the PAL doors are open.

FPL recalculated the doses resulting from the original design basis fuel handling accident incorporating the assumptions of Regulatory Guide 1.25[1] and used more bounding values for source term inventories and site boundary atmospheric dispersion factors.

The original design basis fuel handling accident analysis included the following major assumptions:

- For the limiting case all the rods in one fuel assembly are damaged. The more realistic case has damage limited to one row of fuel rods (14 fuel rods) from a single fuel assembly.
- In calculating the dose consequence it is assumed that the incident occurs in the fuel handling building and that all the activity released from the spent fuel pool is discharged

1 Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, 1972.

to the atmosphere. This assumption is conservative since the ventilation system exhausts to atmosphere at an elevated point and no credit was taken for filtration.

- Limiting Site Boundary Dose
  18.2 rem - thyroid and 0.243 rem - whole body

REVISED DESIGN BASIS ANALYSIS

In support of this submittal, FPL is revising the design basis for the St. Lucie fuel handling accident analysis to include the effects of a fuel handling accident inside the reactor containment building. The revised dose calculations use the methodology of Regulatory Guide 1.25. In this reanalysis, the containment PAL doors are assumed open, the refueling canal filled with 23 feet of water. The consequences from this event bound those from a fuel handling accident in the fuel handling building. The methodology to be used in calculating the control room dose is derived from an expression provided in *Nuclear Power Plant Control Room Ventilating System Design for Meeting General Design Criteria 19*, 13th AEC Air Cleaning Conference, CONF740-807, Vol. 1, which determines the radiological doses based on an activity balance within the control room. Attachment 4 is a copy of the revised fuel handling accident analysis, F-FSA-C-000001, Revision 0, *Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 1 Tech Spec 3.9.4*, prepared by Westinghouse Nuclear Systems. Table 1 of Attachment 4 is the list of input parameters used in the fuel handling calculation.

Assumptions used in this calculation are:

1. One whole fuel assembly is conservatively assumed to be damaged and its gap activity is assumed to be released to the water either in the reactor vessel or the spent fuel pool. This assumption is consistent with the recommendation of Regulatory Guide 1.25.

2. The hottest fuel assembly with the highest radial peaking factor is assumed to be damaged. This assumption is consistent with the recommendation of Regulatory Guide 1.25.

3. The overall decontamination factor for the iodine isotopes in the spent fuel pool and the reactor vessel is 100. This assumption is consistent with regulatory position C.1.g of Regulatory Guide 1.25.

4. Minimum water depth between a damaged fuel assembly and fuel pool surface is 23 feet. This assumption is supported by St. Lucie Unit 1 Technical Specifications (Ref. 2) 3.9.10 and 3.9.11. These TS requirements satisfy the regulatory position in Section C.1.c of Regulatory Guide 1.25.

5. All of the gap activity in the damaged fuel rods is assumed to be released and consist of:

   - 10% of all noble gases except Kr-85
   - 30% of Kr-85
   - 10% of radioactive iodine, except I-131
   - 12% of I-131 in the rods at the time of the accident

   This assumption is consistent with regulatory position C.1.d of Regulatory Guide 1.25, except for item (d). Item (d) uses a higher gap activity for I-131 isotope which is consistent with the guidance provided in NUREG/CR-5009[2] for extended burnup fuel use.

6. Fission product inventories are calculated assuming full power operation at the end of core life just before shutdown. A radial peaking factor of 1.65 is assumed. These assumptions are consistent with regulatory position C.1.e of Regulatory Guide 1.25.

7. Iodine gas inventory is 99.75% inorganic and 0.25% organic. This assumption is consistent with regulatory position C.1.f of Regulatory Guide 1.25.

8. The retention of noble gases in the pool is assumed to be negligible and therefore a noble gas overall decontamination factor of one is used in the analysis. This assumption is consistent with regulatory position C.1.h of Regulatory Guide 1.25.

9. All radioactive material that escapes from the pool is released from the containment or the fuel handling building over a two hour period. This assumption is consistent with regulatory position C.1.h of Regulatory Guide 1.25.

10. Building exhaust system absorbers are not credited in the analysis. This is conservative in relation to regulatory position C.1.j of Regulatory Guide 1.25.

11. No mixing of activity with fuel handling building air is assumed. This assumption is consistent with regulatory position C.1.k of Regulatory Guide 1.25.

12. No credit is assumed for depletion of effluent plume due to deposition or decay. This assumption conforms to regulatory position 3.a. (2) of Regulatory Guide 1.25.

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2 NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Baker A. D., et al., February 1988.

13. Consistent with the guidance of Regulatory Guide 1.25, the following iodine isotopes would be considered in the calculation of inhalation thyroid doses: I-131, I-132, I-133, I-134 and I-135. Of these, the contribution due to I-134 isotope would be neglected due to the short half life (52.6 min[3]) for this isotope.

14. The reactor would be subcritical for at least 72 hours prior to fuel movement for commencing refueling operations. This assumption is consistent with St. Lucie Unit 1 Technical Specification 3.9.3.

15. The 450 cfm control room fresh air makeup is composed of an unfiltered intake of 100 cfm and a filtered flow rate of 350 cfm. This is considered to be conservative and is supported by the fact that a containment isolation signal on high containment radiation will shut off the unfiltered intake.

16. The location specific atmospheric dispersion factors provided in PSL-ENG-SENS-00-103[4], Revision 1, are assumed to be applicable for the exclusion area boundary (EAB), low population zone (LPZ), and the control room. The EAB and LPZ atmospheric dispersion factors are consistent with the more conservative values used for the limiting site boundary dose event large break loss of coolant accident (LBLOCA) and are specified in Table 1 of Appendix 6B of the St. Lucie Unit 1 Updated Final Safety Analysis Report (UFSAR).

17. It is assumed that the source term inventories provided in PSL-ENG-SENS-00-103, Revision 1, bound the inventories that are expected for a peak assembly burnup of 55 GWD/MTU.

18. Only control room filters for filtering out iodine isotopes are considered in the analysis; no filtering in the containment or the fuel building is assumed in the analysis.

19. The dose conversion factors used in the analysis are consistent with those recommended in ICRP Publication II[5]. These dose conversion factors are conservative relative to the Technical Specification 1.10 stipulated ICRP-30 thyroid dose conversion factors.

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3 *Table of Isotopes*, Edited by Lederer, C. M., and Shirley, V. S., 7th Edition, John Wiley & Sons, New York, 1978.

4 PSL-ENG-SENS-00-103, Revision 1, *Engineering Evaluation for Input Parameters to be Used in the Updated Fuel Handling Accident*, St. Lucie Nuclear Plant, Unit 1, Nuclear Engineering Department, Florida Power & Light Company, August 24, 2000.

5 ICRP Publication 2, *Recommendations of the International Commission on Radiological Protection*, A Report of Committee 11 on Permissible Dose for Internal Radiation (1959), Pergamon Press, New York, 1960.

The results of this reanalysis are as follows:

- Control Room Dose
  8.48 rem – thyroid and 0.013 rem – whole body

- Site Boundary Dose (EAB)
  59.1 rem – thyroid and 0.68 rem – whole body

- Low Population Zone (LPZ) Dose
  27.9 rem – thyroid and 0.319 rem – whole body

These values remain well within the criteria specified in NUREG-0800, *Standard Review Plan*, Section 15.7.4, *Radiological Consequences of Fuel Handling Accidents*. The EAB and LPZ inhalation thyroid doses are determined to be 59.1 rem and 27.9 rem, respectively. The EAB and LPZ whole body doses are calculated to be 0.68 rem and 0.319 rem, respectively. The NRC acceptance criteria on offsite doses are given in NUREG-0800 as 25% of 10 CFR 100 exposure guidelines, i.e., 75 rem for the thyroid dose and 6 rem for the whole body dose. Comparison of the results of the revised analysis against the acceptance criteria indicates that both of these criteria are met with more than adequate margin for both the EAB and the LPZ locations.

For the control room location, the calculated inhalation thyroid dose is 8.48 rem and the whole body is 0.013 rem. The NRC acceptance criteria for control room habitability as provided in Section 6.4 in NUREG-0800 is 30 rem for inhalation thyroid dose and 5 rem for the whole body gamma dose. The results of the revised analysis for the control room doses indicate that these dose acceptance criteria are met with significant margins.

The UFSAR will be revised and updated following the approval of this proposed license amendment to include the new design basis in-containment fuel handling accident analysis

DESCRIPTION OF THE PROPOSED CHANGE

FPL proposes to change the following Technical Specification in support of the proposed amendment.

1. TS 3.9.4 - Containment Penetrations: revise the current TS 3.9.4.b. to read (with the proposed new requirements in **bold**):

   b. A minimum of one door in each airlock is closed
   or, both doors of the containment personnel airlock may be open if:

      1) at least one personnel airlock door is capable of being closed,

      2) the plant is in MODE 6 with at least 23 feet of water above the fuel in the reactor core, and

      3) a designated individual is available outside the personnel airlock to close the door.

Justification and Safety Analysis

The proposed change revises TS 3.9.4, Containment Penetrations, to allow the containment personnel airlock doors to be open during fuel movement and core alterations provided: (a) that at least one PAL door is capable of being closed; (b) the plant is in MODE 6 (REFUELING) with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

The regulatory basis for TS 3.9.4, Containment Penetrations, is to ensure that the primary containment is capable of containing fission product radioactivity that may be released from the reactor core following a fuel handling accident inside containment. This ensures that offsite radiation exposures are maintained well within the requirements of 10 CFR 100.

The purpose of the limiting condition for operation (LCO) is to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. Complying with the LCO assures that the assumptions reflected in the analysis for this accident as documented in the St. Lucie UFSAR, Chapter 15.4.3, Fuel Handling Accident, are not violated and the resulting doses are lower than calculated.

The original analysis of the fuel handling accident for St. Lucie Unit 1 assumed that the in-containment fuel handling accident was bounded by the fuel handling building accident. In

that event all the activity released from the spent fuel pool is assumed to escape and reach the site boundary and low population zone with no credit taken for the filtration. This reanalysis estimates the dose with the PAL doors open. In this reanalysis, it is also assumed that all the activity released from the reactor cavity escapes the reactor containment building through the PAL with no credit taken for filtration.

The proposed change contains restrictions on allowing both PAL doors to be open to ensure that at least one door will be available to perform its safety function. The restriction to be in MODE 6 with at least 23 feet of water above the fuel provides sufficient time to respond to a loss of shutdown cooling and ensures a minimum water level exists to provide sufficient shielding during fuel movement. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner. Requiring that a designated individual is available outside of containment to close the PAL doors following evacuation of the containment will minimize the release of radioactive material. Administrative guidelines will be established describing the responsibilities and appropriate actions of the designated individual, in the event of an in-containment fuel handling accident. These guidelines will include the responsibility to be able to communicate with the control room, responsibility to ensure the door is capable of being closed in the event of an in-containment fuel handling accident, door closure, and to implement single door open operations in the event of a fuel handling accident. These administrative controls will ensure refueling containment integrity is re-established following an in-containment fuel handling accident.

The revised calculations and analysis indicates that the bases for the Technical Specification requirements will be met with both PAL doors open during core alterations with the ability to close one door following an event.

Containment Integrity

Technical Specification 3.6.1, Containment Vessel, requires that containment integrity be maintained while in MODES 1 to 4. During MODES 1 to 4 the reactor coolant system contains significant energy which provides the motive force for the expulsion of radionuclides subsequent to a design basis accident (DBA). This Technical Specification allows the opening of containment vessel penetrations under administrative control. The allowance described in this evaluation is being sought for MODE 6 where the effects of a fuel handling accident inside containment would be much less and bounded by the DBA.

Containment Closure

Technical Specification 3.9.4, Containment Penetrations, requires that a minimum of one PAL door, as well as other containment penetrations (except as permitted under Administrative Controls), be closed during core alterations or movement of irradiated fuel within the

containment. This requirement is more conservative than the assumptions used in the St. Lucie Unit 1 Updated Final Safety Analysis Report (UFSAR), Chapter 15.4.3, Fuel Handling Accidents. The revised accident analysis assumes that in the event of a fuel handling accident in containment, all of the iodines and noble gases that become airborne within the containment are assumed to escape and reach the site boundary and low population zone with no credit taken for the containment building barrier or for decay or deposition taken. The fuel handling accident analysis also assumes a minimum water level of 23 feet above the top of the fuel and a minimum decay time of 72 hours prior to fuel movement.

During a refueling outage, other work inside containment does not stop during fuel movement or core alterations. Licensed operators moving the reactor fuel are in constant communications with the control room and would request that the containment evacuation alarm be sounded following the fuel handling accident. The personnel inside the reactor containment building will evacuate. This requires that personnel operate the PAL doors to enter and exit the containment. The analysis assumes that the refueling pool water does not delay the dispersion of the source term gases following the accident. This is a conservative assumption when considering the dose to plant personnel inside containment. The plant personnel inside the reactor containment building would have adequate time to evacuate prior to the source term gases dispersing inside the reactor containment building which has a free volume of 2.5 million cubic feet. In MODE 6, REFUELING, the reactor coolant system is depressurized and there is no system active to pressurize the reactor containment building during a FHA. Therefore, the effects of a radioactive release in MODES 1 to 4 from a pressurized reactor coolant system would have a greater effect since the reactor containment building would be pressurized.

From a practical standpoint, the current TS 3.9.4 will not prevent all radioactive releases from the containment following a fuel handling accident. There are a large number of people in containment during a refueling outage, even during fuel movement and core alterations. Should a fuel handling accident occur inside containment, the airlock doors would be cycled several times to evacuate personnel from containment. With each PAL cycle, more containment air would be released. Under the proposed change, the containment could be evacuated more quickly and then refueling containment integrity re-established. This would reduce dose to workers without increasing dose to the general public.

Control Room Ventilation

The UFSAR provided the NRC with information on St. Lucie Unit 1 compliance with 10 CFR 50 Appendix A, general design criterion (GDC) 19. The NRC Safety Evaluation of the St. Lucie Plant Unit 1, November 8, 1974 concurred that the proposed design of the control room and the ventilation system would meet GDC 19 criteria. The St. Lucie Unit 1 control room is designed with an emergency cleanup system, which is actuated by a containment isolation signal (CIS) from either unit or an outside air intake (OAI) high radiation signal. The filter trains filter a portion of the recirculated air. Outside air makeup and toilet and kitchen exhaust flows

are isolated by dampers actuated by a CIS (either Unit) or OAI high radiation signal. Later, a reduced outside air flow, filtered by the cleanup part of the system, is manually adjusted to maintain a positive pressure in the control room. This prevents the ingress of unfiltered (i.e., potentially contaminated) outside air.

The CIS was designed to control the radioactive release from the plant under accident conditions such as a loss of coolant accident (LOCA). Since the doses expected from a LOCA event are significantly higher than the doses expected from an in-containment fuel handling accident, the requirements of GDC 19 are satisfied. The control room dose is bounded by the large break LOCA. This analysis assumed that 100 cfm of the 450 cfm fresh air intake was unfiltered. The results of the revised analysis for the fuel handling accident indicate that the LOCA dose is still the bounding accident for the control room dose.

Applicable Regulatory Requirements/Criteria

NUREG-0800, *Standard Review Plan*, Section 15.7.4, Radiological Consequences of Fuel Handling Accidents, describes the acceptance criteria for this event as, "the calculated doses at the exclusion boundary are well within the exposure guidelines of 10 CFR Part 100. *Well within* shall mean 25% or less of 10 CFR Part 100, i.e., 75 Rem to the thyroid and 6 Rem for the whole-body doses." Neither the current nor the revised design basis fuel handling accident analysis take credit for the containment building barriers. The results of the calculations performed show that the offsite dose consequences of a fuel assembly dropped inside containment are well within the 10 CFR Part 100 limits. Therefore, the proposed change does not result in a significant hazard.

Regulatory Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors*, is NRC guidance which describes a method acceptable to the NRC staff for licensee evaluation of the potential radiological consequences of a fuel handling accident. The parameters of concern and the acceptance criteria applied are based on the requirements of 10 CFR 100 with respect to the calculated radiological consequences of a fuel handling accident (FHA) and GDC 61 with respect to appropriate containment, confinement, and filtering systems.

NUREG/CR-5009, *Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors*, relates to the expected release fraction for the radioactive iodine. According to this report, the calculated release fraction for extended burnup fuel may be up to 20% higher than that assumed in Regulatory Guide 1.25 for Iodine-131.

The methodology, assumptions, and results of the revised FHA with the proposed Technical Specification change demonstrate compliance with the applicable regulatory requirements, criteria, and guidance.

10 CFR Part 50, Appendix A, General Design Criteria

GDC 16, Containment Design, requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

GDC 19 – Control Room, A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided with: (1) a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

GDC 54, Piping Systems Penetrating Containment, requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC 56, Primary Containment Isolation, describes the isolation provisions that must be provided for lines that connect directly to the containment atmosphere and which penetrate primary reactor containment unless it can be demonstrated that the isolation provisions for a specific class of lines are acceptable on some other defined basis.

GDC 61, Fuel Storage and Handling and Radioactivity Control, requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

The assumptions and results of the revised FHA with the proposed Technical Specification change demonstrate compliance with applicable GDC.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of Proposed License Amendments

Florida Power and Light Company (FPL) proposes to revise Technical Specification 3.9.4, Containment Penetrations. Currently TS 3.9.4 requires a personnel airlock (PAL) door to be closed during core alterations and movement of irradiated fuel within containment. The proposed change would allow both containment PAL doors to be open during core alterations and movement of irradiated fuel in containment provided that: (a) at least one personnel airlock door is capable of being closed; (b) the plant is in MODE 6 with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

Introduction

The Nuclear Regulatory Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed below for the proposed amendment.

Discussion

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 3.9.4 would allow the containment personnel airlock (PAL) doors to be open during fuel movement or core alterations. Currently, a single PAL door is closed during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. The PAL is not an initiator of an accident. Whether the PAL doors are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the PAL doors to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 100. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the PAL doors after the containment is evacuated. The proposed change should significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment.

The changes being proposed do not affect assumptions contained in plant safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Technical Specification 3.9.4, Containment Penetrations, affects a previously evaluated fuel handling accident. Both the current and the reanalyzed fuel handling accident analysis assume that all of the iodine and noble gases that become airborne within the containment escape and reach the site boundary and low population zone with no credit taken for filtration, the containment building barrier, or for decay or deposition taken. Since the proposed change does not involve the addition or modification of equipment, nor does it alter the design of plant systems and the revised analysis is consistent with the fuel handling accident analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The margin of safety as defined by 10 CFR Part 100 has not been reduced. The calculated dose is a well within of the limits given in 10 CFR Part 100 or NUREG-0800. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

Based on the above, FPL has determined that the proposed amendment does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

Environmental Impact Consideration Determination

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration and therefore, meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

St. Lucie Unit 1 Marked-Up Technical Specification Pages

3/4-9

REFUELING OPERATIONS

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, ~~and~~

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve except for valves that are open on an intermittent basis under administrative control, or

2. Be capable of being closed by an OPERABLE automatic containment isolation valve, or

3. Be capable of being closed by an OPERABLE containment vacuum relief valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing of containment isolation valves per the applicable portions of Specifications 4.6.3.1.1 and 4.6.3.1.2.

or, both doors of the containment personnel airlock may be open if:

1. at least one personnel airlock door is capable of being closed,
2. the plant is in MODE 6 with at least 23 feet of water above the fuel in the reactor core, and
3. a designated individual is available outside the personnel airlock to close the door.

St. Lucie Unit 1
Docket No. 50-335
L-2000-222 Attachment 4

ATTACHMENT 4

WESTINGHOUSE NUCLEAR SYSTEMS

Determination of Fuel Handling Accident Radiological Releases

in Support of Relaxation

of St. Lucie Unit 1 Tech Spec 3.9.4

Revision 00

October 2, 2000

Analysis Number F-FSA-C-000001

Westinghouse Nuclear Systems

DESIGN ANALYSIS

ANALYSIS NUMBER: F-FSA-C-000001	REVISION NUMBER: 00
TITLE: Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 1 Tech Spec 3.9.4	SUMMARY OF CONTENTS: Calculation 22 Pages Appendices 21 Pages Microfiche 0 Sheets

PURPOSE: Calculate the radiological releases for a postulated Fuel Handling Accident to show that the SRP acceptance criteria on doses are not exceeded for the case with the Containment Personnel Air Lock (PAL) doors open during refueling operations.

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DESIGN VERIFICATION

Description: The review process included review of equations and methodology and hand calculations of dominant releases and doses and validation of spreadsheet calculations. It was determined that the methodology and calculations are consistent with regulatory guidance and several conservatisms are embedded in the analysis. The results of the analysis are determined to be consistent with the methodology and the input data used.

VERIFICATION STATUS: COMPLETE

The design information contained in this document has been verified to be correct by means of:

__X__ Design Review
_____ Alternate Calculation –Copy Attached
_____ Qualification Testing -- Test Report No.

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RECORD OF REVISIONS

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1.0 INTRODUCTION

A relaxation of St. Lucie Unit 1 plant Technical Specification on the status of containment penetrations under refueling operations (Tech Spec 3.9.4) is sought by Florida Power and Light Company (FP&L). This relaxation would allow refueling operations to be done with an open personnel air lock (PAL) door. The only requirement would be that the air lock be capable of being closed under administrative control, when required.

An open PAL door has the potential to increase radiological releases beyond what is currently reported in the St Lucie Unit 1 FSAR (Reference 1) for a postulated Fuel Handling Accident. Therefore, this increase in radiological doses need to be quantified to support Tech Spec 3.9.4 relaxation effort by the Florida Power & Light Company (FP&L).

2.0 PURPOSE

The purpose of the analysis contained in this recorded calculation (RC) is to quantify the offsite and control room doses for a postulated Fuel Handling Accident (FHA) during refueling operations with both PAL doors open. For regulatory approval of the relaxation of Tech Spec 3.9.4, it is necessary to demonstrate that the offsite and control room doses are below the acceptance criteria set forth in Section 15.7.4 and Section 6.4, respectively, of the US NRC Standard Review Plan (Reference 2). The specific US NRC acceptance criteria for calculated doses for the FHA are shown in Table 2 in tabular form.

3.0 METHODOLOGY

The methodology to be used in the determination of the offsite doses (exclusion area boundary (EAB) and low population zone (LPZ)) is documented in Regulatory Guide 1.25 (Reference 3). The methodology to be used in calculating the control room doses is derived from an expression provided in Reference 4, which determines the radiological doses based on an activity balance within the control room.

3.1 Offsite Doses

Section 15.7.4 of Reference 2 requires the determination of the radiological releases at two locations, namely, the exclusion area boundary (EAB) and the low population zone (LPZ). Both the thyroid inhalation doses and the whole body doses (due to gamma and beta radiation) are required to be quantified for the two locations to demonstrate that the acceptance criteria on these doses are met.

3.1.1 Calculation of Offsite Inhalation Thyroid Dose

The equation provided in Reference 3 for calculating the inhalation thyroid doses due to exposure to each iodine isotope is as follows:

$$D_{th} = [F_g * I * F * P * B * R * (\chi/Q)] \div [DF_p * DF_f] \quad (1)$$

where,

D_{th} = Thyroid dose (rems) due to each iodine isotope over the time span of interest

F_g = Fraction of fuel rod iodine inventory in fuel rod void space

I = Core iodine isotope inventory at time of accident (curies)

F = Fraction of core damaged so as to release void space iodine

P = Fuel peaking factor

B = Breathing rate (m^3/sec) (value provided in table 1, from RG 1.25)

DF_p= Effective iodine decontamination factor for pool water

DF_f= Effective iodine decontamination factor for filters (if present)

χ/Q= Atmospheric diffusion factor at receptor location (sec/m^3)

R = Adult thyroid conversion factor for the iodine isotope of interest (rems/curie) (values provided in Table 1, from RG 1.25)

$*$ = Multiplication symbol

$\div$ = Division symbol

The total inhalation thyroid dose is obtained by summing the thyroid dose contribution due to all iodine isotopes of interest.

3.1.2 Calculation of Offsite Whole Body Dose

The equation provided in Reference 2 for calculating the whole body dose due to gamma radiation is as follows:

$D_{wb\text{-}\gamma} = 0.25\ \bar{E}_\gamma * \psi,$ (2)

where,

$D_{wb\text{-}\gamma}$= Whole body gamma dose due to each noble gas of interest (rems) over the time span of interest

0.25 is the dose conversion factor, per Mev, for internally absorbed radiation (Reference 5) [(rem – m^3 – disintegration)/ (Mev – curies – sec)]

$\bar{E}_\gamma$ = Average gamma energy per disintegration (Mev/dis) for each noble gas (values provided in Table 1, from Reference 14)

ψ = Concentration time integral for each noble gas in the cloud (curies – sec/m^3)
= $(\chi/Q) * Q_M$

Q_M = Total activity released to the environment from each noble gas of interest over the time span of interest (curies)
= Average core inventory for the noble gas of interest per the affected fuel assembly * Peaking Factor * Fraction of noble gas inventory in fuel rod void space

The total whole body dose due to gamma radiation is obtained by summing the whole body gamma dose contribution due to all noble gas isotopes of interest.

The equation provided in Reference 2 for calculating the whole body dose due to beta radiation is as follows:

$D_{wb\text{-}\beta} = 0.23\ \bar{E}_\beta * \psi,$ (3)

where,

$D_{wb\text{-}\beta}$= Whole body beta dose due to each noble gas of interest (rems) over the time span of interest

0.23 is the dose conversion factor, per Mev, for radiation at the surface of a receptor (Reference 5) [(rem – m^3 – disintegration)/ (Mev – curies – sec)]

$\bar{E}_\beta$ = Average beta energy per disintegration (Mev/dis) for each noble gas (values provided in Table 1, from Reference 14)

ψ = Concentration time integral for each noble gas in the cloud (curies – sec/m^3)
= $(\chi/Q) * Q_M$

Q_M = Total activity released to the environment from each noble gas of interest over the time span of interest (curies)
= Average core inventory for the noble gas of interest per the affected fuel assembly * Peaking Factor * Fraction of noble gas inventory in fuel rod void space

The total whole body dose due to beta radiation is obtained by summing the whole body beta dose contribution due to all noble gas isotopes of interest.

3.2 Control Room Doses

The control room doses are to be determined from the perspective of control room habitability as identified in SRP (Reference 2) Section 6.4. The inhalation thyroid, gamma whole body, and beta skin doses are required to be calculated to show that the US NRC acceptance criteria on these doses are met.

3.2.1 Calculation of Control Room Inhalation Thyroid Dose

The methodology for calculating the control room inhalation thyroid dose is documented in Reference 6 and is based on an expression in Reference 4. The equation in Reference 6 for calculating the control room thyroid dose due to each iodine isotope is:

$$D_{th} = (DCF_{th} * B * IQ_M * CRO * 3600) / V_{CR} \quad (4)$$

where,

D_{th} = Inhalation thyroid dose in the control room due to each iodine isotope of interest over the time span of interest (rems),

DCF_{th} = Thyroid dose conversion factor for each iodine isotope (rems/curies) (values provided in Table 1, from RG 1.25),

CRO = Control room occupancy factor (values provided in Table 1, from Reference 4), and

V_{CR} = Net free volume of control room (m^3) (values provided in Table 1, from FSAR, p. 15.4.1-15)

IQ_M = Integrated activity of each iodine isotope in the control room over the time span of interest (curies-hr).

$= \int A_{CR}\, dt$

A_{CR} = Activity in the control room as a function of time (curies)

t = Time (sec)

The activity in the control room of each isotope as a function of time is determined from an activity balance in the control room. This activity balance considers the buildup of activity within the control room, leakage from the containment into the control room, and discharge of activity from the control room. Figure 1 depicts this activity transport for the control room. With reference to this figure, the following differential equation describes the activity transport within the control room

$$dA_{CR}/dt + [\,(L_u/V_{CR}) + (L_f/V_{CR}) + f_R * R_c + \lambda_d\,]\, A_{CR} = (\chi/Q)_{CR}\,[\,L_u + F_{CR} * L_f\,]\, L_{21} * A_{CT} \qquad (5)$$

where,

L_u = Unfiltered leakage into the control room (m^3/sec),

L_f = Filtered Leakage into the control room (m^3/sec),

f_R = Recirculation filter efficiency in the control room for particular chemical form an individual iodine isotope,

R_c = Recirculation flow rate through the control room filters (fraction/sec),

λ_d = Radioactive decay constant for isotope of interest (sec^{-1}),

$(\chi/Q)_{CR}$ = Atmospheric dispersion factor at the control room (sec/m^3),

F_{CR} = $(1 - f_{CR})$,

f_{CR} = Intake filter efficiency in the control room for a particular chemical form of an individual iodine isotope,

L_{21} = Leakage rate from containment region to atmosphere (fraction/sec), and

A_{CT} = Activity in the containment region as a function of time (curies).

The above equation may be solved in closed form (see for example CRC Tables, Reference 13, Section IX). Integrating Equation (5) under the assumption of constant containment activity (A_{CT}) yields:

$$A_{CR}(t) = [(C_2 * L_{21} * A_{CT}) \div C_1] * [1 - EXP(-C_1 * t)], \quad (6)$$

where,

$$C_1 = [(L_u/V_{CR}) + (L_f/V_{CR}) + f_R * R_c + \lambda_d], \text{ and}$$

$$C_2 = (\chi/Q)_{CR} * [L_u + (F_{CR} * L_f)].$$

(Note: R_C=0 for time frame I defined below)

For the fuel handling accident analysis, all iodine activity released from the pool is assumed to be discharged to the atmosphere over the time period of interest. Thus $L_{21} * A_{CT}$ will be calculated as a rate using the total activity released over this period and the time duration.

For the control room, the maximum unfiltered leakage is 100 cfm and the filtered leakage is 350 cfm. At the beginning of the Fuel Handling Accident, when the radiation level becomes high in the containment, a containment isolation signal (CIS) would occur on high radiation. This is conservatively assumed to occur at about 30 minutes. The CIS would cause the control room intake valves to close terminating the filtered in-leakage to the control room. The unfiltered in-leakage is assumed to continue at the 100 cfm value.

To model this scenario appropriately, Equation (5) is solved for two time frames: time frame I between 0 and 30 mins, and time frame II between 30 mins and 8 hours. Note that Equation (6) applies to time frame I.

For time frame II, Eq. (5) is solved with new constants C_1' and C_2' and the initial condition which states that at time = 30 mins, control room activity at 30 mins for time frame II should equal the acitvity calculated using Eq. (6). The solution of Eq. (5) using this constraint leads to the control room activity for time frame II as:

$$A_{CR}(t) = A_{CR}(30) * EXP[-C_1' * (t-30)] + [(C_2' * L_{21} * A_{CT}) \div C_1'] * [1 - EXP(-C_1' * (t-30))] \quad (7)$$

where, $C_1' = [(L_u/V_{CR}) + f_R * R_c + \lambda_d]$, and

$$C_2' = (\chi/Q)_{CR} * L_u.$$

Equations (6) and (7) are numerically integrated to calculate the thyroid dose in the control room due to each iodine isotope. The total inhalation thyroid dose is obtained by summing the thyroid dose contribution due to all iodine isotopes of interest.

3.2.2 Calculation of Control Room Whole Body Dose

The methodology for calculating the control room whole body dose is documented in Reference 6 and is based on an expression in Reference 4. The equation in Reference 6 for calculating the control room whole body dose is:

$$D_{wb} = [\,(V_{CR} \div 0.02832)^{0.338} * DCF_{wb} * CRO * IQ_M)] \div [\,1173 * V_{CR}\,] \quad (8)$$

where,

D_{wb} = Whole body dose from gamma radiation from each isotope within the control room,

DCF_{wb} = Whole body gamma dose conversion factor for each isotope [(rem-m^3)/(curies-sec)] (values provided in Table 1, from RG 1.109)

IQ_M is calculated using Eq. (5). To simplify the calculation, the term, dA_{CR}/dt, is set equal to zero in Eq. (5) and an expression for $A_{CR}(t)$ is obtained as follows:

$$A_{CR}(t) = [(C_2 * L_{21} * A_{CT}(t)) \div C_1] \quad (9)$$

Equation (9) is integrated over the time period of interest to obtain IQ_M, the integrated activity for each isotope. The total whole body dose in the control room due to gamma radiation is obtained by summing the whole body gamma dose contribution due to all noble gas isotopes of interest.

3.2.3 Calculation of Control Room Skin Dose

The methodology for calculating the control room skin dose is documented in Reference 6 and is based on an expression in Reference 4. The equation in Reference 6 for calculating the control room whole body dose is:

$$D_{skin} = [3600 * DCF_{skin} * CRO * IQ_M)] \div V_{CR} \quad (10)$$

where,

D_{skin} = Whole body (skin) dose from beta radiation from each isotope within the control room (rem),

DCF_{skin} = Whole body beta (skin) dose conversion factor for each isotope ([(rem-m3)/(curies-sec)] (values provided in Table 1, from RG 1.109)

IQ_M is calculated using Eq. (5). To simplify the calculation, the term, dA_{CR}/dt, is set equal to zero in Eq. (5) and an expression for $A_{CR}(t)$ is obtained as follows:

$$A_{CR}(t) = [(C_2 * L_{21} * A_{CT}(t)) \div C_1] \quad (11)$$

Equation (11) is integrated over the time period of interest to obtain IQ_M, the integrated activity for each isotope. The total skin dose in the control room due to beta radiation is obtained by summing the skin dose contribution due to all noble gas isotopes of interest.

4.0 ASSUMPTIONS & JUSTIFICATION

The following assumptions and justifications are employed in this analysis to determine the offsite and control room doses.

1. One whole fuel assembly is conservatively assumed to be damaged and its gap activity is assumed to be released to the water either in the reactor vessel or the spent fuel pool. This assumption is consistent with the recommendation of Reg. Guide 1.25 (Ref. 3).

2. The hottest fuel assembly with the highest radial peaking factor is assumed to be damaged. This assumption is consistent with the recommendation of Reg. Guide 1.25 (Ref. 3).

3. The overall decontamination factor for the iodine isotopes in the spent fuel pool and the reactor vessel is 100. This assumption is consistent with regulatory position C.1.g of Reg. Guide 1.25 (Ref. 3).

4. Minimum water depth between damaged fuel assembly and fuel pool surface is 23 feet. This assumption is supported by St. Lucie Unit 1 plant Technical Specifications (Ref. 11) 3.9.10 and 3.9.11. These Tech Spec requirements satisfy the regulatory position in Section C.1.c of Reg Guide 1.25 (Ref. 3).

5. All of the gap activity in the damaged fuel rods is assumed to be released and consist of:
   (a) 10% of all noble gases except Kr-85
   (b) 30% of Kr-85
   (c) 10% of radioactive iodine, except I-131
   (d) 12% of I-131 in the rods at the time of the accident.

   This assumption is consistent with regulatory position C.1.d of Reg Guide 1.25 (Ref. 3), except for item (d). Item(d) uses a higher gap activity for I-131 isotope which is consistent with the guidance provided in NUREG/CR-5009 (Ref. 8) for extended burnup fuel use.

6. Fission product inventories are calculated assuming full power operation at the end of core life just before shutdown. A radial peaking factor of 1.65 is assumed. These assumptions are consistent with regulatory position C.1.e of Reg Guide 1.25 (Ref. 3).

7. Iodine gas inventory is 99.75% inorganic and 0.25% organic. This assumption is consistent with regulatory position C.1.f of Reg Guide 1.25 (Ref. 3).

8. The retention of noble gases in the pool is assumed to be negligible and therefore a noble gas overall decontamination factor of 1 is used in the analysis. This assumption is consistent with regulatory position C.1.h of Reg Guide 1.25 (Ref. 3).

9. deleted.

10. Building exhaust system adsorbers are not credited in the analysis. This is conservative in relation to regulatory position C.1.j of Reg Guide 1.25 (Ref.3).

11. No mixing of activity with fuel handling building air is assumed. This assumption is consistent with regulatory position C.1.k of Reg Guide 1.25 (Ref.3).

12. No credit is assumed for depletion of effluent plume due to deposition or decay. This assumption conforms to regulatory position 3.a. (2) of Reg Guide 1.25 (Ref. 3).

13. Consistent with the guidance of Reg Guide 1.25 (Ref. 3), the following iodine isotopes would be considered in the calculation of inhalation thyroid doses: I-131, I-132, I-133, I-134 and I-135. Of these, the contribution due to I-134 isotope would be neglected due to the short half life (52.6 min, from Ref. 9) for this isotope.

14. The decontamination factor for the noble gases in the spent fuel pool and the reactor vessel is 1. This assumption is consistent with the recommendation of Reg. Guide 1.25 (Ref. 3).

15. The reactor would be subcritical for at least 72 hours prior to fuel movement for commencing refueling operations. This assumption is consistent with St. Lucie Unit 1 plant Tech Spec 3.9.3 (Ref. 11).

16. The control room intake and exhaust flow rates are assumed to be equal. The total in-leakage is assumed to be 450 cfm (Flow rate provided by Reference 7).

17. The location specific atmospheric dispersion factors provided in Reference 7 are assumed to be applicable for the exclusion area boundary (EAB), low population zone (LPZ), and the control room.

18. It is assumed that the source term inventories provided in Reference 7 bound the inventories that are expected for a peak assembly burnup of 55 GWD/MTU.

19. Only control room filters for filtering out iodine isotopes are considered in the analysis; no filtering in the containment or the fuel building is assumed in the analysis.

20. The dose conversion factors used in the analysis are consistent with those recommended in ICRP Publication 2 (Reference 10). These dose conversion factors are conservative relative to the Technical Specification 1.10 stipulated ICRP-30 thyroid dose conversion factors.

21. Part of the control room in-leakage (450 cfm) is assumed to be unfiltered (100cfm) with the remainder (350 cfm) being filtered leakage. At the time of containment isolation on CIS (conservatively assumed to be 30 minutes after initiation of the event), the filtered in-leakage is assumed to be 0 cfm since the CIS would close the control room outside intake valves and start the control room booster fans . The booster fans recirculate the control room air through HEPA and charcoal filters at a rate of 2000 cfm in a closed loop. For control room gamma whole body and beta skin dose calculations, the unfiltered leakage is conservatively assumed to be the total in-leakage of 450 cfm. No filtering occur for noble gases.

5.0 INPUT DATA

The bulk of the data used in this analysis were obtained from FP&L in response to a data request. These data which were formally transmitted via Reference 7 are provided in Table 1. In addition, data from Regulatory Guides, NRC reports, Technical Specifications and the St. Lucie Unit 1 FSAR were used. These data are also listed in Table 1.

6.0 ANALYSIS

Equations (1), (2), (3), (4), (6), (7), (8), (9) (10) and (11) of Section 3.0 were entered into Microsoft Excel spread sheets to calculate the offsite and control room doses. These spreadsheets including the formula spreadsheets are provided in Appendix A for the offsite dose calculations and in Appendix B for the control room dose calculations. The analysis employed the assumptions listed in Section 4.0 and the input data provided in Table 1.

7.0 RESULTS

The results for the offsite and control room doses for the revised Fuel Handling Accident Analysis are provided in Table 2. The EAB and LPZ inhalation thyroid doses are determined to be 59.1 rems

and 27.9 rems, respectively. The EAB and LPZ whole body doses are calculated to be 0.68 rem (including the dose due to beta radiation) and 0.319 rem (including the dose due to beta radiation), respectively. The US NRC acceptance criteria on offsite doses are given in Ref. 2 as 25% of 10 CFR 100 exposure guidelines, i.e., 75 rems for the thyroid dose and 6 rems for the whole body dose. Comparison of the results of the revised analysis documented herein against the acceptance criteria indicates that both of these criteria are met with more than adequate margin for both the EAB and the LPZ locations.

For the control room location, the calculated inhalation thyroid dose is 8.48 rems, the whole body gamma dose is 0.013 rem, and the beta skin dose is 0.47. The US NRC acceptance criteria for control room habitability as provided in Section 6.4 of Ref. 2 is 30 rems for inhalation thyroid dose, 5 rems for the whole body gamma dose, and 30 rems (without protective clothing) for the beta skin dose. The results of the revised analysis for the control room doses indicate that these dose acceptance criteria are met with significant margins.

8.0 CONCLUSIONS

The results of the revised analysis indicate that more than adequate margins to the acceptance criteria for offsite and control room doses are maintained even with the containment PAL doors fully open. Therefore, the relaxation of Tech Spec 3.9.4 to carry out refueling operations in Mode 6 with the PAL doors open is justified.

9.0 REFERENCES

1. St. Lucie Unit 1, Updated Final Safety Analysis Report, Amendment 17.

2. NUREG-0800, US NRC Standard Review Plan, July 1981.

3. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", 1972.

4. Murphy, K. G. and Campe, K. M., "Nuclear Power Plant Control Room Ventilating System Design for Meeting General Design Criteria 19", 13th AEC Air Cleaning Conference, CONF740-807, Vol. 1, pp 401-430.

5. NUREG/CR-3011, "Dose Projection Considerations for Emergency Conditions at Nuclear Power Plants", Stoetzel, G.A., et al, May 1983.

6. CD-TML-083, "Methodology for LOCA Dose Analysis", Rosen, S., ABB Combustion Engineering Nuclear Power, Engineering Services Recorded Calculation, August 21, 1990.

7. PSL-ENG-SENS-00-103, Rev. 1, "Engineering Evaluation for Input Parameters to be Used in the Updated Fuel Handling Accident", St. Lucie Nuclear Plant, Unit 1, Nuclear Engineering Department, Florida Power & Light Company, August 24, 2000.

8. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," D. A. Baker, et. al., February 1988.

9. "Table of Isotopes", Edited by Lederer, C. M., and Shirley, V. S., 7th Edition, John Wiley & Sons, New York, 1978.

10. ICRP Publication 2, "Recommendations of the International Commission on Radiological Protection", A Report of Committee 11 on Permissible Dose for Internal Radiation (1959), Pergamon Press, New York, 1960.

11. St. Lucie Unit 1, Technical Specifications, Amendment 108.

12. Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, October 1977.

13. CRC Standard Mathematical Tables, 25th Edition, CRC Press, 1979.

14. ORNL-5114, "Nuclear Decay Data for selected Radionuclides", Martin, M. J., March 1976.

TABLE 1

Input Data for Analysis of Fuel Handling Accident with the PAL Doors Open

Parameter	Value	Source
Plant Power Level (MWt):	2700	TS
Radial Peaking Factor:	1.65	RG 1.25
Burnup (GWD/MTU):	55	Bounding Target
Decay Time (hours):	72	TS
Number of fuel rods in one assembly:	176	FSAR
Number of fuel assemblies in the core:	217	FSAR
Fraction of fission product gases contained in the gap region of fuel rods (%):		
Kr-85	30	RG 1.25
All other noble gases	10	RG 1.25
I-131	12	NUREG/CR-5009
All other iodines	10	RG 1.25
Activity Release Data:		
Percentage of gap activity released to pool (%)	100	RG 1.25
Core inventory source term for the iodine and noble gas isotopes of interest (curies):		
I-131	8.66E7	
I-132	1.23E8	
I-133	1.61E8	
I-135	1.43E8	
Xe-131m	7.05E5	
Xe-133	1.61E8	
Xe-133m	3.92E6	

Xe-135	3.58E7
Xe-135m	4.33E7
Kr-85	1.30E6
Kr-85m	1.88E7

Core inventory source term for the iodine and noble gas isotopes of interest after 72 hours decay(curies) :

I-131	6.68E7
I-132	3.84E-2
I-133	1.46E7
I-135	7.31E4
Xe-131m	5.92E5
Xe-133	1.08E8
Xe-133m	1.52E6
Xe-135	1.48E5
Xe-135m	4.36E-78
Kr-85	1.30E6
Kr-85m	2.73E2

Decontamination Factors:

Pool decontamination factor for noble gases:	1	RG 1.25
Effective pool decontamination factor for iodine:	100	RG 1.25

Filter efficiency for iodine removal –
Fuel Building Exhaust Filter:

Elemental (%)	0	Not credited
Organic (%)	0	Not credited
Containment Purge Filter Efficiency	0	Not credited

Atmospheric Dispersion Factors (sec/m^3):

0-2 hr Exclusion Area Boundary	1.8E-4	FSAR
0-8 hr Low Population Zone	8.5E-5	FSAR

Control Room		FSAR
0-8 hrs	4.86E-4	
8-24 hrs	4.17E-4	
1-4 days	1.68E-4	
4-30 days	6.36E-5	
Control Room in-leakage rate (cfm):		TS
Total	450	
Unfiltered	100	
Filtered	350	
Control Room Charcoal adsorber: Conservative with respect to TS (Actual 99.9%)	90%	
Control Room Recirculation Flow Rate (cfm)	2000	TS
Breathing Rate, B (m^3/sec)	3.47×10^{-4}	RG 1.25
Average Gamma Disintegration Energies ($\bar{E}_\gamma$. Mev/dis)		Ref. 14
Xe-131m	0.020	
Xe-133	0.045	
Xe-133m	0.0416	
Xe-135	0.247	
Xe-135m	0.432	
Kr-85	0.002	
Kr-85m	0.159	
Average Beta Disintegration Energies ($\bar{E}_\beta$. Mev/dis)		Ref. 14
Xe-131m	0.143	
Xe-133	0.135	
Xe-133m	0.190	
Xe-135	0.316	
Xe-135m	0.095	
Kr-85	0.251	
Kr-85m	0.253	

Inhalation Thyroid Dose Dose Conversion Factors (R or DCF_{th}, rems/curies)		RG 1.25*
I-131	1.48E+6	
I-132	5.35E+4	
I-133	4.00E+4	
I-135	1.24E+5	
Whole Body Dose Conversion Factors for Gamma Radiation in Control Room [DCF_{wb},(rem-m3)/(curies-sec)]		RG 1.109
Xe-131m	2.90E-3	
Xe-133	9.32E-3	
Xe-133m	7.96E-3	
Xe-135	5.74E-2	
Xe-135m	9.89E-2	
Kr-85	5.10E-4	
Kr-85m	3.71E-2	
Skin Dose Conversion Factors for Beta Radiation in Control Room [DCF_{skin},(rem-m3)/(curies-sec)]		RG 1.109
Xe-131m	1.51E-2	
Xe-133	9.70E-3	
Xe-133m	3.15E-2	
Xe-135	5.90E-2	
Xe-135m	2.25E-2	
Kr-85	4.24E-2	
Kr-85m	4.63E-2	
Half Lives of Isotopes of Interest		Ref. 9
I-131	8.04 days	
I-132	2.20 hours	
I-133	20.9 hours	
I-135	6.61 hours	
Xe-131m	11.77 days	
Xe-133	5.25 days	
Xe-133m	2.19 days	
Xe-135	9.10 hours	

Xe-135m	15.60 mins	
Kr-85	10.70 years	
Kr-85m	4.58 hours	
Control Room Occupancy Factor (CRO)		Ref. 4
0 to 8 hrs	1.0	
8 to 24 hrs	1.0	
1 to 4 days	0.6	
4 to 30 days	0.4	
Volume of Control Room (V_{CR}, m^3)	1594	FSAR Page 15.4.1-15

* Derived from "standard man" parameters recommended in ICRP Publication 2.
TS: Technical Specifications
RG: Regulatory Guide
FSAR: Final Safety Analysis Report

TABLE 2

SUMMARY OF RESULTS*** VS. ACCEPTANCE CRITERIA
Radiological Releases for the Fuel Handling Accident with the PAL Doors Open

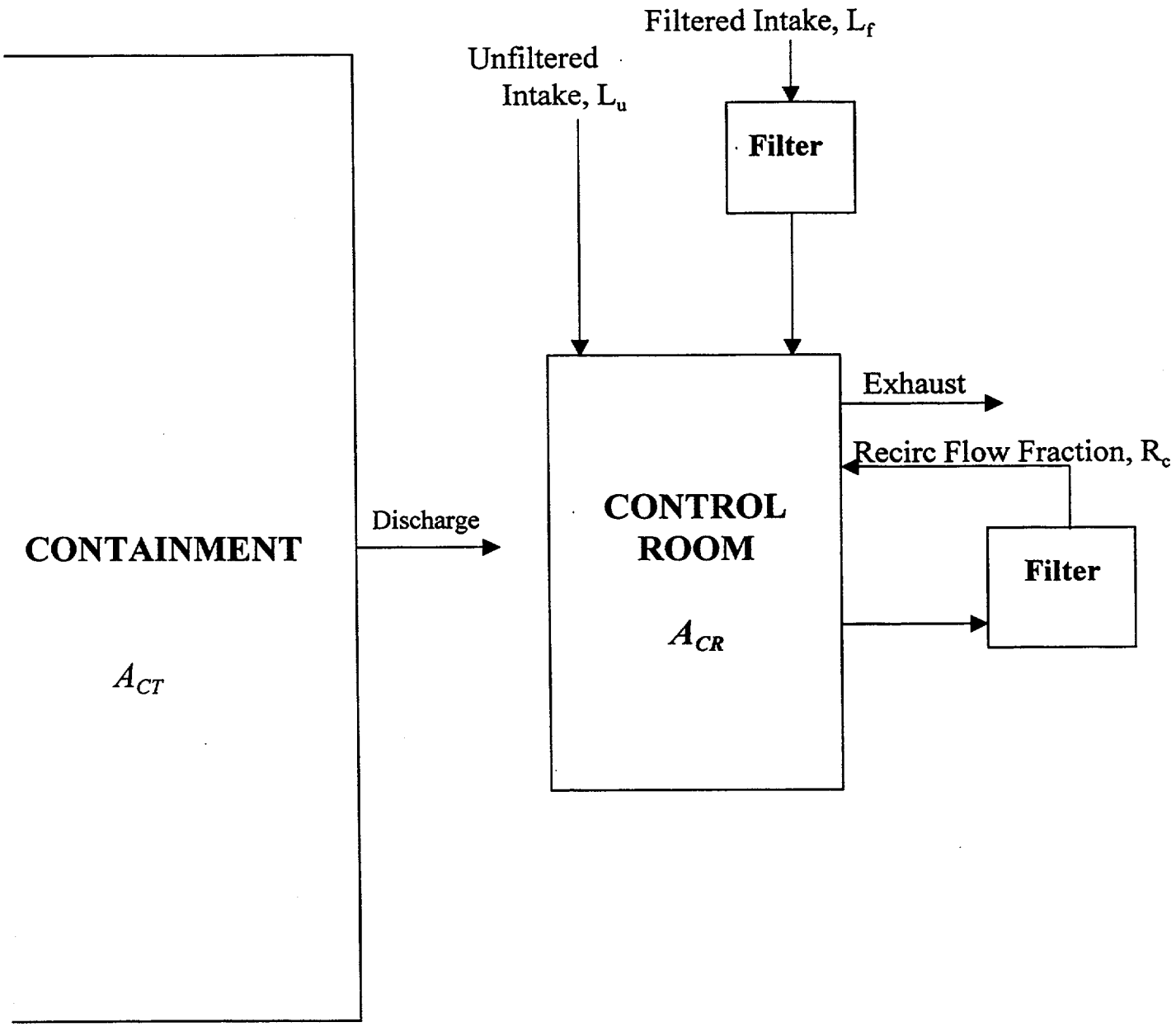
	Inhalation Thyroid Dose		Whole Body Dose		Skin Dose	
Location	**Analysis Results**	**US NRC Acceptance Criteria**	**Analysis Results**	**US NRC Acceptance Criteria**	**Analysis Results**	**US NRC Acceptance Criteria**
EAB Doses (rems)	59.1	**75**	0.68	**6**	*	**
LPZ Doses (rems)	27.9	**75**	0.319	**6**	*	**
Control Room Doses (rems)	8.48	**30**	0.013	**5**	0.47	**30**

* Included in the value provided for whole body dose (i.e., whole body dose = gamma dose + beta dose).

** SRP does not provide a separate acceptance criteria for skin dose for offsite locations.

*** Values of calculated doses were rounded up.

FIGURE 1
ACTIVITY TRANSPORT FOR THE CONTROL ROOM



APPENDIX A

EXCEL SPREAD SHEETS FOR EAB & LPZ INHALATION THYROID & WHOLE BODY DOSE CALCULATIONS

St. Lucie 1 Fuel Handling Accident - EAB LPZ Thyrod Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K
1		EAB & LPZ INHALATION	THYROID DOSES								
2	F=	1									
3	P=	1.65									
4	B=	3.47E-04									
5	X/Q EAB=	1.80E-04									
6	X/Q LPZ=	8.50E-05									
7	DFp=	100									
8	DFf=	1									
9	NFA=	217									
10											
11											
12											
13											
14											
15	Period/Isotope	Fg	TCI	TCI/NFA	R	D					
16	0-2 hr/I-131	0.12	6.68E+07	3.08E+05	1.48E+06	5.63E+01					
17	0-2 hr/I-132	0.1	3.84E-02	1.77E-04	5.35E+04	9.76E-10					
18	0-2 hr/I-133	0.1	1.46E+07	6.73E+04	4.00E+05	2.77E+00					
19	0-2 hr/I-135	0.1	7.31E+04	3.37E+02	1.24E+05	4.30E-03					
20	Total					5.91E+01					
21											
22	0-8 hr/I-131	0.12	6.68E+07	3.08E+05	1.48E+06	2.66E+01					
23	0-8 hr/I-132	0.1	3.84E-02	1.77E-04	5.35E+04	4.61E-10					
24	0-8 hr/I-133	0.1	1.46E+07	6.73E+04	4.00E+05	1.31E+00					
25	0-8 hr/I-135	0.1	7.31E+04	3.37E+02	1.24E+05	2.03E-03					
26	Total					2.79E+01					
27											
28											
29											
30											
31											
32											
33											
34											
35											

St. Lucie 1 Fuel Handling Accident - EAB LPZ Thyrod Dose Calculation with Formulas

	A	B	C	D	E	F
1		EAB & LPZ INHALATION	THYROID DOSES			
2	F=	1				
3	P=	1.65				
4	B=	0.000347				
5	X/Q EAB=	0.00018				
6	X/Q LPZ=	0.000085				
7	DFp=	100				
8	DFf=	1				
9	NFA=	217				
10						
11						
12						
13						
14						
15	Period/Isotope	Fg	TCI	TCI/NFA	R	D
16	0-2 hr/I-131	0.12	66800000	=C16/B9	1480000	=(D16*B2*B3*B16*B4*E16*B5)/(B7*B8)
17	0-2 hr/I-132	0.1	0.0384	=C17/B9	53500	=(D17*B2*B3*B17*B4*E17*B5)/(B7*B8)
18	0-2 hr/I-133	0.1	14600000	=C18/B9	400000	=(D18*B2*B3*B18*B4*E18*B5)/(B7*B8)
19	0-2 hr/I-135	0.1	73100	=C19/B9	124000	=(D19*B2*B3*B19*B4*E19*B5)/(B7*B8)
20	Total					=SUM(F16:F19)
21						
22	0-8 hr/I-131	0.12	66800000	=C22/B9	1480000	=(D22*B2*B3*B22*B4*E22*B6)/(B7*B8)
23	0-8 hr/I-132	0.1	0.0384	=C23/B9	53500	=(D23*B2*B3*B23*B4*E23*B6)/(B7*B8)
24	0-8 hr/I-133	0.1	14600000	=C24/B9	400000	=(D24*B2*B3*B24*B4*E24*B6)/(B7*B8)
25	0-8 hr/I-135	0.1	73100	=C25/B9	124000	=(D25*B2*B3*B25*B4*E25*B6)/(B7*B8)
26	Total					=SUM(F22:F25)
27						
28						
29						
30						
31						
32						
33						
34						
35						

St. Lucie 1 Fuel Handling Accident - EAB LPZ Whole Body Gamma Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
1			EAB & LPZ	WHOLE	BODY GAMMA	DOSES									
2															
3	Period/Isotope	DCF per Mev	E bar gamma	TCI	Fg	Qm	Dwb								
4	0-2 hr/Xe-131m	0.25	2.00E-02	5.92E+05	1.00E-01	4.50E+02	4.05E-04								
5	0-2 hr/Xe-133	0.25	4.50E-02	1.08E+08	1.00E-01	8.21E+04	1.66E-01								
6	0-2 hr/Xe-133m	0.25	4.16E-02	1.52E+06	1.00E-01	1.16E+03	2.16E-03								
7	0-2 hr/Xe-135	0.25	2.47E-01	1.48E+05	1.00E-01	1.13E+02	1.25E-03								
8	0-2 hr/Xe-135m	0.25	4.32E-01	4.36E-78	1.00E-01	3.32E-81	6.44E-86								
9	0-2 hr/Kr-85	0.25	2.00E-03	1.30E+06	3.00E-01	2.97E+03	2.67E-04								
10	0-2 hr/Kr-85m	0.25	1.59E-01	2.73E+02	1.00E-01	2.08E-01	1.49E-06								
11															
12	Total						1.70E-01								
13															
14	0-8 hr/Xe-131	0.25	2.00E-02	5.92E+05	1.00E-01	4.50E+02	1.91E-04								
15	0-8 hr/Xe-133	0.25	4.50E-02	1.08E+06	1.00E-01	8.21E+04	7.85E-02								
16	0-8 hr/Xe-133m	0.25	4.16E-02	1.52E+06	1.00E-01	1.16E+03	1.02E-03								
17	0-8 hr/Xe-135	0.25	2.47E-01	1.48E+05	1.00E-01	1.13E+02	5.91E-04								
18	0-8 hrXe-135m	0.25	4.32E-01	4.36E-78	1.00E-01	3.32E-81	3.04E-86								
19	0-8 hr/Kr-85	0.25	2.00E-03	1.30E+06	3.00E-01	2.97E+03	1.26E-04								
20	0-8 hr/Kr-85m	0.25	1.59E-01	2.73E+02	1.00E-01	2.08E-01	7.01E-07								
21															
22	Total						8.05E-02								
23															
24	DCF per Mev=	0.25													
25	PF=	1.65													
26	X/Q 2 hr =	1.80E-04													
27	X/Q 8 hr =	8.50E-05													
28	NFA=	217													
29															
30															
31															
32															
33															
34															
35															
36															
37															
38															
39															
40															
41															
42															
43															
44															

St. Lucie 1 Fuel Handling Accident - EAB LPZ Whole Body Gamma Dose Calculation with Formulas

	A	B	C	D	E	F	G
1			EAB & LPZ	WHOLE	BODY GAMMA	DOSES	
2							
3	Period/Isotope	DCF per Mev	E bar gamma	TCI	Fg	Qm	Dwb
4	0-2 hr/Xe-131m	0.25	0.02	592000	0.1	=(D4/B$28)*B$25*E4	=B4*C4*B$26*F4
5	0-2 hr/Xe-133	0.25	0.045	108000000	0.1	=(D5/B$28)*B$25*E5	=B5*C5*B$26*F5
6	0-2 hr/Xe-133m	0.25	0.0416	1520000	0.1	=(D6/B$28)*B$25*E6	=B6*C6*B$26*F6
7	0-2 hr/Xe-135	0.25	0.247	148000	0.1	=(D7/B$28)*B$25*E7	=B7*C7*B$26*F7
8	0-2 hr/Xe-135m	0.25	0.432	4.36E-78	0.1	=(D8/B$28)*B$25*E8	=B8*C8*B$26*F8
9	0-2 hr/Kr-85	0.25	0.002	1300000	0.3	=(D9/B$28)*B$25*E9	=B9*C9*B$26*F9
10	0-2 hr/Kr-85m	0.25	0.159	273	0.1	=(D10/B$28)*B$25*E10	=B10*C10*B$26*F10
11							
12	Total						=SUM(G4:G10)
13							
14	0-8 hr/Xe-131	0.25	0.02	592000	0.1	=(D14/B$28)*B$25*E14	=B14*C14*B$27*F14
15	0-8 hr/Xe-133	0.25	0.045	108000000	0.1	=(D15/B$28)*B$25*E15	=B15*C15*B$27*F15
16	0-8 hr/Xe-133m	0.25	0.0416	1520000	0.1	=(D16/B$28)*B$25*E16	=B16*C16*B$27*F16
17	0-8 hr/Xe-135	0.25	0.247	148000	0.1	=(D17/B$28)*B$25*E17	=B17*C17*B$27*F17
18	0-8 hr/Xe-135m	0.25	0.432	4.36E-78	0.1	=(D18/B$28)*B$25*E18	=B18*C18*B$27*F18
19	0-8 hr/Kr-85	0.25	0.002	1300000	0.3	=(D19/B$28)*B$25*E19	=B19*C19*B$27*F19
20	0-8 hr/Kr-85m	0.25	0.159	273	0.1	=(D20/B$28)*B$25*E20	=B20*C20*B$27*F20
21							
22	Total						=SUM(G14:G20)
23							
24	DCF per Mev=	0.25					
25	PF=	1.65					
26	X/Q 2 hr =	0.00018					
27	X/Q 8 hr =	0.000085					
28	NFA=	217					
29							
30							
31							
32							
33							
34							
35							
36							
37							
38							
39							
40							
41							
42							
43							
44							

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
1		EAB,	LPZ	WHOLE	BODY	SKIN DOSE									
2															
3	Period/Isotope	DCF per Mev	E bar beta	TCI	Fg	Qm	Dwb								
4	0-2 hr/Xe-131m	0.23	1.43E-01	5.92E+05	1.00E-01	4.50E+02	2.66E-03								
5	0-2 hr/Xe-133	0.23	1.35E-01	1.08E+08	1.00E-01	8.21E+04	4.59E-01								
6	0-2 hr/Xe-133m	0.23	1.90E-01	1.52E+06	1.00E-01	1.16E+03	9.09E-03								
7	0-2 hr/Xe-135	0.23	3.16E-01	1.48E+05	1.00E-01	1.13E+02	1.47E-03								
8	0-2 hr/Xe-135m	0.23	9.50E-02	4.36E-78	1.00E-01	3.32E-81	1.30E-86								
9	0-2 hr/Kr-85	0.23	2.51E-01	1.30E+06	3.00E-01	2.97E+03	3.08E-02								
10	0-2 hr/Kr-85m	0.23	2.53E-01	2.73E+02	1.00E-01	2.08E-01	2.17E-06								
11															
12	Total						5.03E-01								
13															
14	0-8 hr/Xe-131	0.23	1.43E-01	5.92E+05	1.00E-01	4.50E+02	1.26E-03								
15	0-8 hr/Xe-133	0.23	1.35E-01	1.08E+08	1.00E-01	8.21E+04	2.17E-01								
16	0-8 hr/Xe-133m	0.23	1.90E-01	1.52E+06	1.00E-01	1.16E+03	4.29E-03								
17	0-8 hr/Xe-135	0.23	3.16E-01	1.48E+05	1.00E-01	1.13E+02	6.95E-04								
18	0-8 hr/Xe-135m	0.23	9.50E-02	4.36E-78	1.00E-01	3.32E-81	6.16E-87								
19	0-8 hr/Kr-85	0.23	2.51E-01	1.30E+06	3.00E-01	2.97E+03	1.46E-02								
20	0-8 hr/Kr-85m	0.23	2.53E-01	2.73E+02	1.00E-01	2.08E-01	1.03E-06								
21															
22	Total						2.38E-01								
23															
24	DCF per4 Mev=	0.23													
25	PF=	1.65													
26	X/Q 2 hr =	1.80E-04													
27	X/Q 8 hr =	8.50E-05													
28	NFA=	217													
29															
30															
31															
32															
33															
34															
35															
36															
37															
38															
39															
40															
41															
42															
43															
44															

St. Lucie 1 Fuel Handling Accident - EAB LPZ Whole Body Beta Dose Calculation with Formulas

	A	B	C	D	E	F	G	H
1		EAB,	LPZ	WHOLE	BODY	SKIN DOSE		
2								
3	Period/Isotope	DCF per Mev	E bar beta	TCI	Fg	Qm	Dwb	
4	0-2 hr/Xe-131m	0.23	0.143	592000	0.1	=(D4/B$28)*B$25*E4	=B4*C4*B$26*F4	
5	0-2 hr/Xe-133	0.23	0.135	108000000	0.1	=(D5/B$28)*B$25*E5	=B5*C5*B$26*F5	
6	0-2 hr/Xe-133m	0.23	0.19	1520000	0.1	=(D6/B$28)*B$25*E6	=B6*C6*B$26*F6	
7	0-2 hr/Xe-135	0.23	0.316	148000	0.1	=(D7/B$28)*B$25*E7	=B7*C7*B$26*F7	
8	0-2 hr/Xe-135m	0.23	0.095	4.36E-78	0.1	=(D8/B$28)*B$25*E8	=B8*C8*B$26*F8	
9	0-2 hr/Kr-85	0.23	0.251	1300000	0.3	=(D9/B$28)*B$25*E9	=B9*C9*B$26*F9	
10	0-2 hr/Kr-85m	0.23	0.253	273	0.1	=(D10/B$28)*B$25*E10	=B10*C10*B$26*F10	
11								
12	Total						=SUM(G4:G10)	
13								
14	0-8 hr/Xe-131	0.23	0.143	592000	0.1	=(D14/B$28)*B$25*E14	=B14*C14*B$27*F14	
15	0-8 hr/Xe-133	0.23	0.135	108000000	0.1	=(D15/B$28)*B$25*E15	=B15*C15*B$27*F15	
16	0-8 hr/Xe-133m	0.23	0.19	1520000	0.1	=(D16/B$28)*B$25*E16	=B16*C16*B$27*F16	
17	0-8 hr/Xe-135	0.23	0.316	148000	0.1	=(D17/B$28)*B$25*E17	=B17*C17*B$27*F17	
18	0-8 hrXe-135m	0.23	0.095	4.36E-78	0.1	=(D18/B$28)*B$25*E18	=B18*C18*B$27*F18	
19	0-8 hr/Kr-85	0.23	0.251	1300000	0.3	=(D19/B$28)*B$25*E19	=B19*C19*B$27*F19	
20	0-8 hr/Kr-85m	0.23	0.253	273	0.1	=(D20/B$28)*B$25*E20	=B20*C20*B$27*F20	
21								
22	Total						=SUM(G14:G20)	
23								
24	DCF per4 Mev=	0.23						
25	PF=	1.65						
26	X/Q 2 hr =	0.00018						
27	X/Q 8 hr =	0.000085						
28	NFA=	217						
29								
30								
31								
32								
33								
34								
35								
36								
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APPENDIX B

EXCEL SPREAD SHEETS FOR CONTROL ROOM INHALATION THYROID, WHOLE BODY, AND BETA SKIN DOSE CALCULATIONS

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P
1			SL1	CONTROL ROOM		THYROID DOSE										
2	Lu=	0.047195			Luprime=	0.047195										
3	fR=	0.9			fRprime=	0.9										
4	RC=	0			Rcprime=	0.000592158										
5	VCR(ft3)=	56291.27379	VCR(m3)=	1594												
6	Lf=	0.1651825	fCR=	0.9	Lfprime=	0										
7	X/Q=	4.86E-04	Lf/VCR=	0.000103628	Lfprime/VCR=	0										
8	Lu/VCR(m3)=	2.96079E-05	FCR*Lf=	0.01651825	Luprime/VCR=	2.96079E-05										
9	fR*RC=	0	FCR*Lfprime=	0	fRprime*RCprime=	0.000532842										
10	P=	1.65	NFA=	217												
11	DFp=	100	DFf=	1												
12	C2=	3.10E-05	C2prime=	2.29368E-05												
13	B=	3.47E-04	CRO=	1												
14																
15	Period/Isotope	HL	Lambda	C1	C1Prime	TCI	TCI/NFA	Fg	A2	IQM	DCF	Dth				
16	0-8 hr CR/I-131	6.95E+05	9.98E-07	1.34E-04	5.64E-04	6.68E+07	3.08E+05	0.12	0.021163594	6.97E-03	1.48E+06	8.081906097				
17	0-8 hr CR/I-132	8.21E+03	8.44E-05	2.18E-04	6.47E-04	3.84E-02	1.77E-04	0.1	1.01382E-11	2.93E-12	5.35E+04	1.22685E-10				
18	0-8 hr CR/I-133	7.52E+04	9.21E-06	1.42E-04	5.72E-04	1.46E+07	6.73E+04	0.1	0.003854647	1.25E-03	4.00E+05	0.392409445				
19	0-8 hr CR/I-135	2.38E+04	2.91E-05	1.62E-04	5.92E-04	7.31E+04	3.37E+02	0.1	1.92996E-05	6.07E-06	1.24E+05	0.000589551				
20																
21																
22																
23										Total		8.474905093				
24																
25	I-131						I-132					I-133				
26	ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr			ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr		ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr	
27	0	0.00E+00	0.001888883	3.14814E-05			0	0.00E+00	8.82871E-13	1.47145E-14		0	0.00E+00	0.000343198	5.71997E-06	
28	10	3.78E-04	0.007409366	0.000123489			10	1.77E-13	3.42339E-12	5.70584E-14		10	6.86E-05	0.001344678	2.24113E-05	
29	20	7.26E-04	0.016280429	0.00027134			20	3.32E-13	7.41858E-12	1.23643E-13		20	1.32E-04	0.002950514	4.91752E-05	
30	30	1.05E-03	0.026491794	0.00044153			30	4.68E-13	1.19198E-11	1.98663E-13		30	1.90E-04	0.004794979	7.99163E-05	
31	40	9.94E-04	0.036244799	0.00060408			40	4.33E-13	1.61292E-11	2.6882E-13		40	1.79E-04	0.006552864	0.000109214	
32	50	9.56E-04	0.045670948	0.000761182			50	4.09E-13	2.01406E-11	3.35677E-13		50	1.72E-04	0.008249311	0.000137489	
33	60	9.29E-04	0.054864014	0.0009144			60	3.93E-13	2.40178E-11	4.00297E-13		60	1.67E-04	0.009902163	0.000165036	
34	70	9.10E-04	0.063890868	0.001064848			70	3.82E-13	2.78039E-11	4.63398E-13		70	1.63E-04	0.01152408	0.000192068	
66	390	8.61E-04	0.340556771	0.005675946			390	3.59E-13	1.4322E-10	2.38701E-12		390	1.55E-04	0.061191211	0.001019854	
67	400	8.61E-04	0.349170498	0.005819508			400	3.59E-13	1.46815E-10	2.44691E-12		400	1.55E-04	0.062737538	0.001045626	
68	410	8.61E-04	0.357784222	0.00596307			410	3.59E-13	1.50409E-10	2.50681E-12		410	1.55E-04	0.064283865	0.001071398	
69	420	8.61E-04	0.366397945	0.006106632			420	3.59E-13	1.54003E-10	2.56671E-12		420	1.55E-04	0.065830191	0.00109717	
70	430	8.61E-04	0.375011668	0.006250194			430	3.59E-13	1.57597E-10	2.62662E-12		430	1.55E-04	0.067376517	0.001122942	
71	440	8.61E-04	0.38362539	0.006393756			440	3.59E-13	1.61191E-10	2.68652E-12		440	1.55E-04	0.068922844	0.001148714	
72	450	8.61E-04	0.392239111	0.006537319			450	3.59E-13	1.64785E-10	2.74642E-12		450	1.55E-04	0.07046917	0.001174486	
73	460	8.61E-04	0.400852832	0.006680881			460	3.59E-13	1.68379E-10	2.80632E-12		460	1.55E-04	0.072015496	0.001200258	
74	470	8.61E-04	0.409466553	0.006824443			470	3.59E-13	1.71973E-10	2.86622E-12		470	1.55E-04	0.073561822	0.00122603	
75	480	8.61E-04	0.418080274	0.006968005			480	3.59E-13	1.75568E-10	2.92613E-12		480	1.55E-04	0.075108148	0.001251802	

St. Lucie 1 Fuel Handling Accident - Control Room Thyroid Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL1	CONTROL ROOM		HYROID DOSE						
2	Lu=	=(100*28317)/(60*1000000)			Luprime=	=(100*28317)/(						
3	fR=	0.9			fRprime=	0.9						
4	RC=	=(0*28317)/(60*1000000*1594)			Rcprime=	=(2000*28317)						
5	VCR(ft3)=	=(1594*1000000)/28317	VCR(m3)=	1594								
6	Lf=	=(350*28317)/(60*1000000)	fCR=	0.9	Lfprime=	=(0*28317)/(60						
7	X/Q=	0.000486	Lf/VCR=	=B6/D5	Lfprime/VCR=	=F6/D5						
8	Lu/VCR(m3)=	=B2/D5	FCR*Lf=	=(1-D6)*B6	Luprime/VCR=	=F2/D5						
9	fR*RC=	=B3*B4	FCR*Lfprime=	=(1-D6)*F6	fRprime*RCprime=	=F3*F4						
10	P=	1.65	NFA=	217								
11	DFp=	100	DFf=	1								
12	C2=	=B7*(B2+D8)	C2prime=	=B7*(F2+D9)								
13	B=	0.000347	CRO=	1								
14												
15	Period/Isotope	HL	Lambda	C1	C1Prime	TCI	TCI/NFA	Fg	A2	IQM	DCF	Dth
16	0-8 hr CR/I-131	694656	=LN(2)/B16	=B$8+B$9+D$7+C16	=F$8+F$7+F$9+C16	66800000	=F16/217	0.12	=(G16*H16*B10)/(B11*D11*8*3600)	=D$75	1480000	=(3600*K16*B$13*D$13*J16)/(D$5)
17	0-8 hr CR/I-132	8208	=LN(2)/B17	=B$8+B$9+D$7+C17	=F$8+F$7+F$9+C17	0.0384	=F17/217	0.1	=(G17*H17*B10)/(B11*D11*8*3600)	=J$75	53500	=(3600*K17*B$13*D$13*J17)/(D$5)
18	0-8 hr CR/I-133	75240	=LN(2)/B18	=B$8+B$9+D$7+C18	=F$8+F$7+F$9+C18	14600000	=F18/217	0.1	=(G18*H18*B10)/(B11*D11*8*3600)	=O$75	400000	=(3600*K18*B$13*D$13*J18)/(D$5)
19	0-8 hr CR/I-135	23796	=LN(2)/B19	=B$8+B$9+D$7+C19	=F$8+F$7+F$9+C19	73100	=F19/217	0.1	=(G19*H19*B10)/(B11*D11*8*3600)	=T$75	124000	=(3600*K19*B$13*D$13*J19)/(D$5)
20												
21												
22												
23										Total		=SUM(L16:L19)
24												
25	I-131						I-132					I-133
26	ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr			R Time, m	ACR	IQM Ci-min	IQM Ci-hr		ACR Time, min
27	0	=((B$12*I$16)/D$16)*(1-EXP(-D$16*60*A27))	=(B27+B28)*10/2+0	=C27/60			0	=((B$1	=(H27+H28)*10/2+0	=I27/60		0
28	10	=((B$12*I$16)/D$16)*(1-EXP(-D$16*60*A28))	=(B28+B29)*10/2+C27	=C28/60			10	=((B$1	=(H28+H29)*10/2+I27	=I28/60		10
29	20	=((B$12*I$16)/D$16)*(1-EXP(-D$16*60*A29))	=(B29+B30)*10/2+C28	=C29/60			20	=((B$1	=(H29+H30)*10/2+I28	=I29/60		20
30	30	=((B$12*I$16)/D$16)*(1-EXP(-D$16*60*A30))	=(B30+B31)*10/2+C29	=C30/60			30	=((B$1	=(H30+H31)*10/2+I29	=I30/60		30
31	40	=B$30*EXP(-E$16*60*(A31-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A31-30)))	=(B31+B32)*10/2+C30	=C31/60			40	=H$30	=(H31+H32)*10/2+I30	=I31/60		40
32	50	=B$30*EXP(-E$16*60*(A32-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A32-30)))	=(B32+B33)*10/2+C31	=C32/60			50	=H$30	=(H32+H33)*10/2+I31	=I32/60		50
33	60	=B$30*EXP(-E$16*60*(A33-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A33-30)))	=(B33+B34)*10/2+C32	=C33/60			60	=H$30	=(H33+H34)*10/2+I32	=I33/60		60
34	70	=B$30*EXP(-E$16*60*(A34-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A34-30)))	=(B34+B35)*10/2+C33	=C34/60			70	=H$30	=(H34+H35)*10/2+I33	=I34/60		70
66	390	=B$30*EXP(-E$16*60*(A66-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A66-30)))	=(B66+B67)*10/2+C65	=C66/60			390	=H$30	=(H66+H67)*10/2+I65	=I66/60		390
67	400	=B$30*EXP(-E$16*60*(A67-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A67-30)))	=(B67+B68)*10/2+C66	=C67/60			400	=H$30	=(H67+H68)*10/2+I66	=I67/60		400
68	410	=B$30*EXP(-E$16*60*(A68-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A68-30)))	=(B68+B69)*10/2+C67	=C68/60			410	=H$30	=(H68+H69)*10/2+I67	=I68/60		410
69	420	=B$30*EXP(-E$16*60*(A69-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A69-30)))	=(B69+B70)*10/2+C68	=C69/60			420	=H$30	=(H69+H70)*10/2+I68	=I69/60		420
70	430	=B$30*EXP(-E$16*60*(A70-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A70-30)))	=(B70+B71)*10/2+C69	=C70/60			430	=H$30	=(H70+H71)*10/2+I69	=I70/60		430
71	440	=B$30*EXP(-E$16*60*(A71-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A71-30)))	=(B71+B72)*10/2+C70	=C71/60			440	=H$30	=(H71+H72)*10/2+I70	=I71/60		440
72	450	=B$30*EXP(-E$16*60*(A72-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A72-30)))	=(B72+B73)*10/2+C71	=C72/60			450	=H$30	=(H72+H73)*10/2+I71	=I72/60		450
73	460	=B$30*EXP(-E$16*60*(A73-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A73-30)))	=(B73+B74)*10/2+C72	=C73/60			460	=H$30	=(H73+H74)*10/2+I72	=I73/60		460
74	470	=B$30*EXP(-E$16*60*(A74-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A74-30)))	=(B74+B75)*10/2+C73	=C74/60			470	=H$30	=(H74+H75)*10/2+I73	=I74/60		470
75	480	=B$30*EXP(-E$16*60*(A75-30))+(D$12*I$16/E$16)*(1-EXP(-E$16*60*(A75-30)))	=(B75+B76)*10/2+C74	=C75/60			480	=H$30	=(H75+H76)*10/2+I74	=I75/60		480

St. Lucie 1 Fuel Handling Accident - Control Room Whole Body Gamma Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M	N
1			SL1	CONTROL ROOM	WHOLE	BODY	DOSE							
2	Lu=	0.2123775												
3	fR=	0												
4	RC=	0.000592158												
5	VCR(ft3)=	56291.27379	VCR(m3)=	1594										
6	Lf=	0												
7	X/Q=	4.86E-04												
8	Lu/VCR(m3)=	0.000133236												
9	fR*RC=	0												
10	P=	1.65												
11	DFp=	1	DFf=	1										
12	C2=	1.03E-04												
13	B=	3.47E-04												
14														
15	Period/Isotope	HL	Lambda	C1	TCI	TCI/NFA	Fg	A2	IQM	DCF	CRO	Dwb		
16	0-8 hr CR/Xe-131m	1.02E+06	6.82E-07	1.34E-04	5.92E+05	2.73E+03	0.1	0.0156298	9.64E-02	2.90E-03	1.00E+00	2.17028E-05		
17	0-8 hr CR/Xe-133	4.54E+05	1.53E-06	1.35E-04	1.08E+08	4.98E+05	0.1	2.851382488	1.75E+01	9.32E-03	1.00E+00	0.01264441		
18	0-8 hr CR/Xe-133m	1.89E+05	3.66E-06	1.37E-04	1.52E+06	7.00E+03	0.1	0.040130568	2.42E-01	7.96E-03	1.00E+00	0.00014962		
19	0-8 hr CR/Xe-135	3.28E+04	2.12E-05	1.54E-04	1.48E+05	6.82E+02	0.1	0.00390745	2.09E-02	5.74E-02	1.00E+00	9.31484E-05		
20	0-8hr CR/Xe-135m	936	7.41E-04	8.74E-04	4.36E-78	2.01E-80	0.1	1.15111E-85	1.09E-85	9.89E-02	1.00E+00	8.35437E-88		
21	0-8hr CR/Kr-85	3.38E+08	2.05E-09	1.33E-04	1.30E+06	5.99E+03	0.3	0.10296659	6.38E-01	5.10E-04	1.00E+00	2.5272E-05		
22	0-8 hr CR/Kr-85m	16128	4.30E-05	1.76E-04	2.73E+02	1.26E+00	0.1	7.20766E-06	3.38E-05	3.71E-02	1.00E+00	9.73037E-08		
23										Total		0.01293425		
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St. Lucie 1 Fuel Handling Accident - Control Room Whole Body Gamma Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL1	CONTROL ROOM	WHOLE	BODY	DOSE					
2	Lu=	=(450*28317)/(60*1000000)										
3	fR=	0										
4	RC=	=(2000*28317)/(60*1000000*1										
5	VCR(ft3)=	=(1594*1000000)/28317	VCR(m3)=	1594								
6	Lf=	0										
7	X/Q=	0.000486										
8	Lu/VCR(m3)=	=B2/D5										
9	fR*RC=	=B3*B4										
10	P=	1.65										
11	DFp=	1	DFf=	1								
12	C2=	=B7*B2										
13	B=	0.000347										
14												
15	Period/Isotope	HL	Lambda	C1	TCI	TCI/NFA	Fg	A2	IQM	DCF	CRO	Dwb
16	0-8 hr CR/Xe-131m	1016928	=LN(2)/B16	=B8+B9+C16	592000	=E16/217	0.1	=(F16*G16*B10)/(B11*D11*8*3600)	=(B$12*H16*8)/D16	0.0029	1	=((B$5^0.338)*3600*J16*K16*I16)/(1173*D$5)
17	0-8 hr CR/Xe-133	453600	=LN(2)/B17	=B8+B9+C17	108000000	=E17/217	0.1	=(F17*G17*B10)/(B11*D11*8*3600)	=(B$12*H17*8)/D17	0.00932	1	=((B$5^0.338)*3600*J17*K17*I17)/(1173*D$5)
18	0-8 hr CR/Xe-133m	189216	=LN(2)/B18	=B8+B9+C18	1520000	=E18/217	0.1	=(F18*G18*B10)/(B11*D11*8*3600)	=(B$12*H18*8)/D18	0.00796	1	=((B$5^0.338)*3600*J18*K18*I18)/(1173*D$5)
19	0-8 hr CR/Xe-135	32760	=LN(2)/B19	=B8+B9+C19	148000	=E19/217	0.1	=(F19*G19*B10)/(B11*D11*8*3600)	=(B$12*H19*8)/D19	0.0574	1	=((B$5^0.338)*3600*J19*K19*I19)/(1173*D$5)
20	0-8hr CR/Xe-135m	936	=LN(2)/B20	=B8+B9+C20	4.36E-78	=E20/217	0.1	=(F20*G20*B10)/(B11*D11*8*3600)	=(B$12*H20*8)/D20	0.0989	1	=((B$5^0.338)*3600*J20*K20*I20)/(1173*D$5)
21	0-8hr CR/Kr-85	337666320	=LN(2)/B21	=B8+B9+C21	1300000	=E21/217	0.3	=(F21*G21*B10)/(B11*D11*8*3600)	=(B$12*H21*8)/D21	0.00051	1	=((B$5^0.338)*3600*J21*K21*I21)/(1173*D$5)
22	0-8 hr CR/Kr-85m	16128	=LN(2)/B22	=B8+B9+C22	273	=E22/217	0.1	=(F22*G22*B10)/(B11*D11*8*3600)	=(B$12*H22*8)/D22	0.0371	1	=((B$5^0.338)*3600*J22*K22*I22)/(1173*D$5)
23										Total		=SUM(L16:L22)
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St. Lucie 1 Fuel Handling Accident - Control Room Beta Skin Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
1			SL1	CONTROL ROOM	SKIN DOSE										
2	Lu=	0.2123775													
3	fR=	0													
4	RC=	0.000592158													
5	VCR(ft3)=	56291.27379	VCR(m3)=	1594											
6	Lf=	0													
7	X/Q=	4.86E-04													
8	Lu/VCR(m3)=	0.000133236													
9	fR*RC=	0													
10	P=	1.65													
11	DFp=	1	DFf=	1											
12	C2=	1.03E-04													
13	B=	3.47E-04													
14															
15	Period/Isotope	HL	Lambda	C1	TCI	TCI/NFA	Fg	L21A2	IQM	DCF	CRO	Dwb			
16	0-8 hr CR/Xe-131m	1.02E+06	6.82E-07	1.34E-04	5.92E+05	2.73E+03	0.1	0.0156298	9.64E-02	1.51E-02	1.00E+00	0.003286571			
17	0-8 hr CR/Xe-133	4.54E+05	1.53E-06	1.35E-04	1.08E+08	4.98E+05	0.1	2.851382488	1.75E+01	9.70E-03	1.00E+00	0.382739528			
18	0-8 hr CR/Xe-133m	1.89E+05	3.66E-06	1.37E-04	1.52E+06	7.00E+03	0.1	0.040130568	2.42E-01	3.15E-02	1.00E+00	0.017220076			
19	0-8 hr CR/Xe-135	3.28E+04	2.12E-05	1.54E-04	1.48E+05	6.82E+02	0.1	0.00390745	2.09E-02	5.90E-02	1.00E+00	0.002784609			
20	0-8hr CR/Xe-135m	936	7.41E-04	8.74E-04	4.36E-78	2.01E-80	0.1	1.15111E-85	1.09E-85	2.25E-02	1.00E+00	5.52776E-87			
21	0-8hr CR/Kr-85	3.38E+08	2.05E-09	1.33E-04	1.30E+06	5.99E+03	0.3	0.10296659	6.38E-01	4.24E-02	1.00E+00	0.061106056			
22	0-8 hr CR/Kr-85m	16128	4.30E-05	1.76E-04	2.73E+02	1.26E+00	0.1	7.20766E-06	3.38E-05	4.63E-02	1.00E+00	3.53171E-06			
23										Total		0.467140372			
24															
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26															
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St. Lucie 1 Fuel Handling Accident - Control Room Beta Skin Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL1	CONTROL ROOM	SKIN DOSE							
2	Lu=	=(450*28317)/(60*1000000)										
3	fR=	0										
4	RC=	=(2000*28317)/(60*1000000*1										
5	VCR(ft3)=	=(1594*1000000)/28317	VCR(m3)=	1594								
6	Lf=	0										
7	X/Q=	0.000486										
8	Lu/VCR(m3)=	=B2/D5										
9	fR*RC=	=B3*B4										
10	P=	1.65										
11	DFp=	1	DFf=	1								
12	C2=	=B7*B2										
13	B=	0.000347										
14												
15	Period/Isotope	HL	Lambda	C1	TCI	TCI/NFA	Fg	L21A2	IQM	DCF	CRO	Dwb
16	0-8 hr CR/Xe-131m	1016928	=LN(2)/B16	=B8+B9+C16	592000	=E16/217	0.1	=(F16*G16*B10)/(B11*D11*8*3600)	=(B$12*H16*8)/D16	0.0151	1	=(3600*J16*K16*I16)/D$5
17	0-8 hr CR/Xe-133	453600	=LN(2)/B17	=B8+B9+C17	108000000	=E17/217	0.1	=(F17*G17*B10)/(B11*D11*8*3600)	=(B$12*H17*8)/D17	0.0097	1	=(3600*J17*K17*I17)/D$5
18	0-8 hr CR/Xe-133m	189216	=LN(2)/B18	=B8+B9+C18	1520000	=E18/217	0.1	=(F18*G18*B10)/(B11*D11*8*3600)	=(B$12*H18*8)/D18	0.0315	1	=(3600*J18*K18*I18)/D$5
19	0-8 hr CR/Xe-135	32760	=LN(2)/B19	=B8+B9+C19	148000	=E19/217	0.1	=(F19*G19*B10)/(B11*D11*8*3600)	=(B$12*H19*8)/D19	0.059	1	=(3600*J19*K19*I19)/D$5
20	0-8hr CR/Xe-135m	936	=LN(2)/B20	=B8+B9+C20	4.36E-78	=E20/217	0.1	=(F20*G20*B10)/(B11*D11*8*3600)	=(B$12*H20*8)/D20	0.0225	1	=(3600*J20*K20*I20)/D$5
21	0-8hr CR/Kr-85	337666320	=LN(2)/B21	=B8+B9+C21	1300000	=E21/217	0.3	=(F21*G21*B10)/(B11*D11*8*3600)	=(B$12*H21*8)/D21	0.0424	1	=(3600*J21*K21*I21)/D$5
22	0-8 hr CR/Kr-85m	16128	=LN(2)/B22	=B8+B9+C22	273	=E22/217	0.1	=(F22*G22*B10)/(B11*D11*8*3600)	=(B$12*H22*8)/D22	0.0463	1	=(3600*J22*K22*I22)/D$5
23										Total		=SUM(L16:L22)
24												
25												
26												
27												
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APPENDIX C

DESIGN VERIFICATION CHECKLIST

QPF 0306-1

CE Nuclear Power LLC DESIGN VERIFICATION CHECKLIST QPF 0306-1 DESIGN DOCUMENT NO: F-FSA-C-000001	REV. No.: 00	
A. GENERAL	**OK**	**N/A**
1. Design inputs were correctly selected and incorporated.	✓	
2. An appropriate design method was used.	✓	
3. Assumptions necessary to perform the design have been adequately described and are reasonable. Where necessary, assumptions are identified for subsequent re-verification when the detailed design activities are completed.	✓	
4. Applicable codes, standards, and regulatory requirements, including issue and addenda, have been properly identified, and their requirements have been met.	✓	
5. Technical Change Requests (TCR) and other design changes approved to date have been considered and incorporated where appropriate/required.		✓
6. Applicable construction and operating experience has been considered.		✓
7. Requirements for identification of items and materials have been specified.		✓
8. Versions of computer codes employed in the design have been certified for application.		✓
9. Appropriate quality and quality assurance requirements have been specified.	✓	
10. Specified parts, equipment, and processes are suitable for the required application		✓
11. Adequate handling, storage, cleaning and shipping requirements have been specified.		✓
12. Design input and verification requirements for interfacing organizations have been specified, where necessary.		✓
13. Specified materials are compatible with each other and with the design environmental conditions to which the material will be exposed.		✓
14. Provisions have been made for accessibility for needed maintenance, repair and in-service inspection, including consideration of radiation exposure to personnel.		✓
15. Acceptance criteria incorporated in the design documents are sufficient to allow verification that design requirements have been satisfactorily met.		✓
16. Adequate pre-operational and subsequent periodic test requirements have been appropriately specified.		✓

17. The design output is reasonable when compared to design input.	✓	
B. DESIGN ANALYSIS	**OK**	**N/A**
1. Adjustment factors, uncertainties, and empirical correlations used in the analysis have been correctly applied and an appropriate analysis or calculation method was used.	✓	
2. The purpose of the analysis is sufficiently clear, and the results and conclusions are reasonable when compared to inputs.	✓	
C. DRAWINGS	**OK**	**N/A**
1. The item(s) shown is not in conflict with design requirements and is compatible with the major component or system of which it is part.		✓
2. Sufficient dimensions and tolerance requirements have been specified to permit fabrication and inspection.		✓
3. The item(s) shown has been checked for interface agreement with mating components shown on complementary drawings.		✓
D. TESTING	**OK**	**N/A**
1. The test procedure includes provisions for assuring that prerequisites include such items as:		
Appropriate equipment and trained personnel		✓
Condition of the test rig and the item to be tested		✓
Suitable environmental conditions		✓
2. The test procedure describes the conduct of the test and:		✓
The type, range, accuracy, and location of instrumentation		✓
The requirements for data acquisition and instrument monitoring		✓
Acceptance criteria for evaluation of results		✓
3. The test report identifies the test procedure and changes thereto, adequately summarizes test results, and provides sufficient evidence to show that test requirements have been satisfied.		✓
E. COMMENTS/REMARKS: Review included review of equations and methodology and hand calculations of dominant releases and doses and validation of spreadsheet calculations. For thyroid dose calculations, several conservatisms used in the analysis are noted here for future reference. • Use of ICRP-2 instead of ICRP-30 dose conversion factors results in a 30% upward bias. • Use of 0.9 for Iodine filter efficiency underestimates the actual capability.		

• Integration scheme for IQ_M biases results one time step in the conservative direction. . Release time sensitivity study performed to support QA indicates small increases for shorter duration releases. Incremental changes are accommodated by the conservatisms mentioned above. EAB dose calculation assumes all activity released from fuel is transported to the exclusion area boundary in 2 hours.		
Independent Reviewer: R. E. Schneider, Senior Consultant, 9/29/00 Name, Title, Signature, Date		

APPENDIX D

DESIGN DOCUMENT REVIEWER'S COMMENT CHECKLIST/FORM

QPF 0302-1

CE Nuclear Power LLC

DESIGN DOCUMENT
REVIEWER'S COMMENT CHECKLIST/FORM **OPF 0302-1**

TITLE: Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 1 Tech Spec 3.9.4	**PAGE** D-2 **OF** D-3
DOCUMENT NUMBER: F-FSA-C-000001	**REVISION NUMBER:** 00

Comment Number	Reviewer's Comment	Response Required?	Author's Response	Response Accepted?
1	For the control room thyroid calculations of Section 3.2.1, the differential equation of Eq. (5) should be solved for two different time frames: time frame I: 0 to 30 minutes (time of CIS) and time frame II: 30 mins to 8 hrs, since the in-leakage and recirculation flows are different during these two time frames. The spreadsheet also should reflect the difference in the activity equation for these two time frames.	Yes	The text of Section 3.2.1 was modified to include the two time frames; Eqs. (6) and (7) are applicable to time frames I and II, respectively. In addition the spreadsheet was also changed to incorporate these two equations. However, only a very small increase in control room thyroid dose was noticed (8.48 rems vs. 8.23 rems)	Yes
2	In Eq. (7) in Section 3.2.2, the term (VCR*0.02832) should be (VCR/0.02832). This change should also be in the spreadsheet.	Yes	This is a typographical error and was corrected. The spreadsheet correctly used this term and hence no changes to the spreadsheet were required.	Yes

3	In the spreadsheet for calculating the control room thyroid dose, the units on the exponential term is missing multiplication by 60 seconds to make the exponent unitless. The use of the 60 seconds would increase the control room dose slightly above the limit of 30 rems.	Yes	This correction was made and the increase in dose noted here was confirmed. In discussions with FP&L it was leaned that the use 450cfm of in-leakage as unfiltered is extremely conservative, since the actual unfiltered leakage is 100 cfm. A change to the control room in-leakage modeling was made to include 100 cfm as the unfiltered in-leakage and 350 cfm as the filtered in-leakage. The filtered leakage was assumed to be terminated at 30 mins due to CIS on high radiation. This modeling change resulted in the control room thyroid dose remaining well below the acceptance criterion (8.48 rems vs. 30 rems.)	Yes
4	Various editorial changes are recommended.	Yes	Reasonable editorial changes as noted in the text were made.	Yes

Checklist Completed By Reviewer:		
R. E. Schneider	Raymond E. Schneider	09/29/2000
Printed Name	**Signature**	**Date**