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

June 22, 2001

L-2001-083
10 CFR 50.90

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

RE: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment
Containment Equipment Door and Containment
Airlock Doors Open During Core Alterations

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) proposes to revise the St. Lucie Unit 2 Technical Specification (TS) 3.9.4, Containment Penetrations. TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door in each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each open containment airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close the door. The capability to close the containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock 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 and TS Bases changes. 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 2 Tech Spec 3.9.4*, prepared by Westinghouse Nuclear Systems. This proposed change is similar to License Amendment 172 for St. Lucie Unit 1 for the containment personnel airlock.

A001

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.

Approval of this proposed license amendment is requested by October 19, 2001 to support planning for the fall 2001 Unit 2 refueling outage (SL2-13).

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

Very truly yours,

Donald E. Jernigan
Vice President
St. Lucie Plant

DEJ/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)

Donald E. Jernigan 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.

Donald E. Jernigan

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 22 day of June, 2001
by Donald E. Jernigan , who is personally known to me.

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

Leslie J. Whitwell
MY COMMISSION # DD020212 EXPIRES
May 12, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

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

ATTACHMENT 1

SAFETY ANALYSIS

INTRODUCTION

Florida Power and Light Company (FPL) proposes to revise the St. Lucie Unit 2 Technical Specification (TS) 3.9.4, "Containment Building Penetrations." TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door in each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open airlock to close the door. The capability to close a containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner. Similar controls and procedures are already in place to support reactor coolant systems (RCS) operation at reduced inventory.

BACKGROUND

Technical Specification 3.9.4, "Containment Building Penetrations," requires the equipment door and at least one door in each containment airlock to be closed during core alterations and fuel movements (MODE 6). The basis for this requirement is to limit the effects of a fuel handling accident inside containment. The consequences of the fuel handling accident (FHA) for the reactor containment building is bounded by the effects of the accident occurring in the fuel handling building since the fuel handling 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 equipment door and all the containment airlock doors remain open for a two-hour period subsequent to the FHA.

FPL recalculated the doses resulting from the original design basis fuel handling accident incorporating the assumptions of Regulatory Guide (RG) 1.25 (Reference 4), using bounding values for source term inventories, and assuming no credit for ventilation system filtration.

The original design basis fuel handling accident analysis occurring in the fuel handling building 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 16 fuel rods in a single fuel assembly.

- In calculating the dose consequence, it is assumed that the incident occurs in the fuel handling building and that the activity released triggers the airborne radiation monitors to isolate the normal fuel handling building ventilation system and automatically initiates the filtration systems.

- Limiting Site Boundary Dose

  3.0 rem - thyroid and 0.11 rem - whole body

- Low Population Zone

  1.3 rem - thyroid and 0.046 rem - whole body

REVISED DESIGN BASIS ANALYSIS

In support of this submittal, FPL is revising the design basis for the St. Lucie Unit 2 FHA analysis to include the effects of a FHA inside the reactor containment building. The dose calculations use the methodology of Regulatory Guide 1.25. In the revised analysis, the equipment door and all the containment airlock doors (the source is not bounded by the size of any opening) are assumed open with the refueling cavity filled with 23 feet of water above the reactor pressure vessel flange. The consequences of this event bound those from a FHA in the fuel handling building. The methodology used in calculating the control room doses is derived from an expression provided in *Nuclear Power Plant Control Room Ventilating System Design for Meeting General Design Criteria (GDC) 19*, 13th AEC Air Cleaning Conference, CONF740-807, Vol. 1, which determines the radiological doses based on an activity balance within the control room. 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 damaged and its gap activity is assumed released to the water either in the reactor vessel or the spent fuel pool. This assumption is consistent with the recommendation of RG 1.25 (Reference 4).

2. The hottest fuel assembly with the highest radial peaking factor is assumed damaged. This assumption is consistent with the recommendation of RG 1.25 (Reference 4).

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 RG 1.25 (Reference 4).

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

5. All of the gap activity in the damaged fuel rods is assumed to be released and consists 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 RG 1.25, (Reference 4) except for item (d). Item (d) uses a higher gap activity for I-131 isotope that is consistent with the guidance provided in NUREG/CR-5009 (Reference 8) for extended burn-up 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 RG 1.25 (Reference 4).

7. Iodine gas inventory is 99.75% inorganic and 0.25% organic. This assumption is consistent with regulatory position C.1.f of RG 1.25 (Reference 4).

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 RG 1.25 (Reference 4).

9. For the exclusion area boundary (EAB) doses, the radioactive material that escapes from the spent fuel pool to the building is assumed to be released from the building over a two-hour time period. This assumption is consistent with regulatory position C.1.i of RG 1.25 (Reference 4).

10. Building exhaust system absorbers are not credited in the analysis. This is conservative in relation to regulatory position C.1.j of RG 1.25 (Reference4).

11. No mixing of activity with fuel handling building air is assumed. This assumption is consistent with regulatory position C.1.k of RG 1.25 (Reference 4).

12. No credit is assumed for depletion of the effluent plume due to deposition or decay. This assumption conforms to regulatory position 3.a.(2) of RG 1.25 (Reference 4).

13. Consistent with the guidance of RG 1.25 (Reference 4), the following iodine isotopes are 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 are neglected due to the short half-life (52.6 min, from Reference 9) for this isotope.

14. The reactor would be subcritical for at least 72 hours prior to fuel movement before commencing refueling operations. This assumption is consistent with St. Lucie Unit 2 TS 3.9.3.

15. Control room intake and exhaust flow rates are assumed to be equal. The total in-leakage is assumed to be 450 cfm.

16. The location specific atmospheric dispersion factors that are provided in Reference 8 are assumed to be applicable for the EAB, low population zone (LPZ), and the control room.

17. A maximum average core burn-up of 41.35 GWD/MTU is assumed consistent with item 58 on page B-19 of Reference 7. This value corresponds to a maximum batch average discharge burn-up of 55 GWD/MTU consistent with item 102 on page B-26 of Reference 7. Since this batch is made up of assemblies that would be at burn-up levels higher and lower than this value, the peak assembly value is assumed to be at a higher value (about 58 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 (Reference 10). These dose conversion factors are conservative relative to the TS 1.10 stipulated ICRP-30 thyroid dose conversion factors.

20. Part of the control room in-leakage (450-cfm) is assumed to be unfiltered (100 cfm) with the remainder (350-cfm) being filtered leakage. At the time of containment isolation on a containment isolation signal (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 occurs for noble gases.

21. The fission product inventory calculation uses a multiplication factor of 30% on the activity calculated using the burn-up value in assumption 18 for additional conservatism.

22. The atmospheric dispersion factors used are those for ground level releases. These values are more conservative than those for elevated releases are (see, for example, Figures 1 and 3 of RG 1.25 (Reference 4)). Note that releases from the containment equipment door are elevated releases and, as such, the atmospheric dispersion

factors characteristic of these releases are expected to be smaller than the ground level release values.

The results of this re-analysis are as follows:

- Control Room Dose

  9.39 rem – thyroid and 0.02 rem – whole body

- Site Boundary Dose (EAB)

  61.6 rem – thyroid and 0.75 rem – whole body

- Low Population Zone (LPZ) Dose

  26.7 rem – thyroid and 0.33 rem – whole body

These values remain well within the acceptance 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 61.6 rem and 26.7 rem, respectively. The EAB and LPZ whole body doses are calculated to be 0.75 rem and 0.33 rem, respectively. The NRC acceptance criteria on offsite doses are given in Reference 3 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, the calculated inhalation thyroid dose is 9.39 rem and the whole body is 0.02 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 Updated Final Safety Analysis Report (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 a. and TS 3.9.4 b. to read (with the proposed new requirements in bold).

a. The equipment door closed and held in place by a minimum of four bolts, **or the equipment door may be open if:**

   1) it is capable of being closed with four bolts within 30 minutes,

   2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

   3) a designated crew is available at the equipment door to close the door.

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

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

   2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

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

2. Bases for Section 3.9.4: Revise the Bases for TS 3.9.4 to add the following paragraph.

These restrictions include the administrative controls to allow the opening of both doors of each airlock (emergency and/or personnel) and the containment equipment door during CORE ALTERATIONS provided that: a) at least one door of each airlock is capable of being closed; b) the plant is in Mode 6 with at lest 23 feet of water above the reactor pressure vessel flange; c) a designated individual is available outside each open airlock to close the door; d) the equipment door can be closed with four bolts within 30 minutes; and e) an equipment door closure crew is available to close the equipment door.

Justification

The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door of each open

containment airlock is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close the door. The capability to close a containment a airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner. Similar controls and procedures are already in place to support reactor coolant system (RCS) operation at reduced inventory.

The regulatory basis for TS 3.9.4, "Containment Building Penetrations," is to ensure that the primary containment is capable of containing fission product radioactivity that may be released 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 minimize the release 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 2 UFSAR, Chapter 15.7.4.1.2, "Fuel Handling Accident" are met and the resulting doses are lower than calculated.

The original analysis of the fuel handling accident for St. Lucie Unit 2, assumed that the in-containment fuel handling accident was bounded by the fuel handling building accident. In that event the entire amount of radioactivity released from the spent fuel pool is assumed to escape and that the activity released triggers the airborne radiation monitors to isolate the normal fuel handling building ventilation system and automatically initiates the filtration systems. The revised analysis estimates the dose with the containment equipment door and both doors of each containment airlock open. In the revised analysis, it is also assumed that the entire radioactivity released from the reactor cavity leaves the reactor containment building through the equipment door and both doors of each containment airlock, with no credit taken for filtration.

The proposed change contains restrictions on allowing the containment equipment door and both doors of each containment airlock to be open, provided that at least one door on each open containment airlock and equipment 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, ensures a minimum water level exists to provide sufficient shielding during fuel movement, and reduces the radioactivity released in the event of a fuel handling accident. The capability to close the containment equipment door and a door of each open containment airlock includes the requirement that the doors are capable of being closed and that any cables or hoses crossing through the doors have quick-disconnects to ensure the doors are capable of being closed in a timely manner. Requiring that a designated individual be available to close the equipment door and a door of each open containment airlock following evacuation of the containment will minimize the release of radioactive material. Administrative requirements will be established for the responsibilities and appropriate actions of the designated individuals in the event of an in-containment fuel handling accident. These requirements will include the responsibility to be able to communicate with the control room, responsibility to ensure that the doors are capable of

being closed in the event of an in-containment fuel handling accident, door closure, and to implement single containment airlock door open operations in the event of a fuel handling accident. These administrative controls will ensure refueling containment integrity would be established in the event of an in-containment fuel handling accident.

The revised calculations and analysis indicate that the basis for the Technical Specification requirements will be met with the equipment door and both doors of each containment airlock open during core alterations with the ability to close the equipment door and one door on each open containment airlock following a FHA.

EVALUATION

Containment Integrity

Technical Specification 3.6.1, "Containment Integrity" requires that containment integrity be maintained while in MODES 1 to 4. During MODES 1 to 4, the reactor coolant system contains significant energy that 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 relaxation described in this evaluation is being sought for MODE 6 where the effects of a fuel handling accident inside containment are the event of concern and are bounded by the DBA.

Containment Closure

Technical Specification 3.9.4, "Containment Building Penetrations," requires that a minimum of one door on each open containment airlock, the equipment 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 revised St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR), Chapter 15.7.4.1.2, "Fuel Handling Accident." The revised accident analysis assumes that, in the event of a fuel handling accident in containment, all of the iodine 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. The revised fuel handling accident analysis also assumes a minimum water level of 23 feet above the top of the fuel in the core and a minimum post-reactor shutdown 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 are procedurally required to inform the control room that the containment evacuation alarm be sounded in the event of a fuel handling accident. The personnel inside the reactor containment building will evacuate. This requires that personnel operate the personnel airlock doors to exit the containment. The revised analysis assumes that the reactor cavity 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 through 4 from a pressurized RCS would have a greater effect since the reactor containment building would become pressurized.

The containment emergency airlock opens into the fuel handling building, which has an air filtration system that releases through a monitored plant vent stack. The opening of these doors will allow control element assembly extension shafts to be passed directly from the containment to the fuel handling building or from the fuel handling building into containment. The extension shafts are normally stored in containment until refueling containment integrity is no longer required by TS. This creates an unnecessary radioactive source inside containment for this period of time. Elimination of the extension shaft storage will reduce personnel exposure of the plant workers near the storage area.

The containment equipment door will have a closure crew available to close this door. The closure crew is trained for timely equipment door closure. The door can be closed without electrical power available and within 20 minutes of notification. The equipment door closure crew currently provides this function during RCS reduced inventory operations in accordance with FPL commitments made as part of Generic Letter (GL) 88-17.

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 may be a 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 containment airlock cycle, more containment air would be released. Under the proposed change, the containment could be evacuated more quickly with timely refueling integrity being established subsequently. This would reduce dose to workers.

Control Room Ventilation

The FSAR discusses St. Lucie Unit 2 compliance with GDC 19. The NRC Safety Evaluation of the St. Lucie Plant Unit No. 2, dated October 1981, concluded that the proposed design of the control room and the ventilation system would meet GDC 19 criteria. The St. Lucie Unit 2 control room is designed with an emergency cleanup system, which is actuated by a containment isolation actuation signal (CIAS) from either unit or a control room outside air intake (CROAI) high radiation signal. The filter trains filter a portion of the recirculated air. Outside air make up and toilet and kitchen exhaust flows are isolated by butterfly valves actuated by a CIAS (either unit) or CROAI 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 which prevents the ingress of unfiltered (i.e., potentially contaminated) outside air.

The CIAS was designed to control the radioactive release from the plant under accident conditions such as a loss of coolant accident (LOCA). Since the doses conservatively calculated in the event of a LOCA event are significantly higher than the doses calculated in the event of an in-containment fuel handling accident, the requirements of GDC-19 are satisfied. The control room dose is bounded by the large break LOCA. 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 takes credit for the containment building barriers. The results of the calculations performed (Attachment 4, page 16) 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.

U. S. NRC 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. Thc 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 FHA and GDC 61 with respect to appropriate containment, confinement, and filtering systems.

NUREG/CR 5009, "Assessment of the Use of Extended Burn-up 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 burn-up 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 changes comply 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," requires that 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 (1) with 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) with 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 analysis, coupled with the proposed Technical Specification changes demonstrate comply with the above GDCs.

EVALUATION CONCLUSIONS

Based on review of the licensing bases documentation and the results of the reanalysis of the fuel handling accident inside the reactor containment building, it is concluded that the proposed license amendment is acceptable and that code requirements are maintained.

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 Building Penetrations. TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door on each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: a) the equipment door is capable of being closed with four bolts within 30 minutes, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b. would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: a) at least one door on each open containment airlock door is capable of being closed, b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and c) a designated individual is available outside each open containment airlock to close a door. The capability to close the containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

Introduction

The Nuclear Regulatory Commission provides standards for determining whether a significant hazards consideration will exist (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 equipment door and both doors of each containment airlock to be open during fuel movement or core alterations. Currently, the equipment door is closed with four (4) bolts and a single door on each containment airlock 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. Neither the containment equipment door nor either of the containment airlock doors is an initiator of an accident. Whether the containment equipment door or both doors of the containment airlocks are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the containment equipment door and the containment airlock 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 containment equipment door or the containment airlock doors after the containment is evacuated, which would be much less than the two hours assumed in the analysis. The proposed change would 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 the 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 Building Penetrations," affects a previously evaluated fuel handling accident. Both the current and the revised fuel handling accident analyses assume that all of the iodine and noble gases that become airborne 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. 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 significantly reduced. The calculated dose is well within 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.

Summary

Based on the above discussion, FPL has determined that the proposed amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (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; therefore, the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

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 2 Marked-Up Technical Specification Pages

3/4-9-4

B 3/4 9-1

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building 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, or

2. Be capable of being closed by an OPERABLE automatic containment isolation 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 building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building 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 building by:

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

b. Testing of containment isolation valves per the applicable portions of Specification 4.6.3.2.

Insert 1:

or the equipment door may be open if:

1) it is capable of being closed with four bolts within 30 minutes,

2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

3) a designated crew is available at the equipment door to close the door.

Insert 2:

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

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

2) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and

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

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value specified in the COLR for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value specified in the COLR includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

Insert:

These restrictions include the administrative controls to allow the opening of both doors of each airlock (emergency and/or personnel) and the containment equipment door during CORE ALTERATIONS provided that: a) at least one door of each airlock is capable of being closed; b) the plant is in Mode 6 with at lest 23 feet of water above the reactor pressure vessel flange; c) a designated individual is available outside each open airlock to close the door; d) the equipment door can be closed with four bolts within 30 minutes; and e) an equipment door closure crew is available to close the equipment door.

ATTACHMENT 4

WESTINGHOUSE NUCLEAR SYSTEMS

Determination of Fuel Handling Accident Radiological Releases

in Support of Relaxation

of St. Lucie Unit 2 Tech Spec 3.9.4

Revision 00

March 1, 2001

Analysis Number L-FSA-C-000001

Westinghouse Electric Company
Nuclear Systems

DESIGN ANALYSIS

ANALYSIS NUMBER: L-FSA-C-000001	REVISION NUMBER: 00
TITLE: Determination of Fuel Handling Accident Radiological Releases in Support of Relaxation of St. Lucie Unit 2 Tech Spec 3.9.4	SUMMARY OF CONTENTS: Calculation 25 Pages Appendices 20 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 Equipment door and Personnel Air Lock (PAL) doors open during refueling operations.

COGNIZANT ENGINEER: M. C. Jacob, Senior Consultant, MCJacob 2/21/01
Name, Title, Signature, Date

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: INCOMPLETE

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

✓ Design Review
____ Alternate Calculation –Copy Attached
____ Qualification Testing – Test Report No.

Independent Reviewer: R. E. Schneider, Senior Consultant, RESchneider 2/28/01
Name, Title, Signature, Date

MANAGEMENT APPROVAL: C. L. Kling, Manager – Fluid Systems Analysis, CLKling 3/01/01
Name, Title, Signature, Date

Westinghouse Electric Company
Nuclear Systems

RECORD OF REVISIONS

Rev. No.	Date	Revised Pages	Author	Independent Reviewer	Approved By	Total # Pages
00	3/01/01	Original Issue	M. C. Jacob	R. E. Schneider	C. L. Kling	45

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1.0 INTRODUCTION

A relaxation of St. Lucie Unit 2 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 open Equipment door and Personnel Air Lock (PAL) door. The only requirement would be that the air lock be capable of being closed under administrative control, when required.

Open containment doors have the potential to increase radiological releases beyond what is currently reported in the St Lucie Unit 2 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 amendment 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 the equipment door and 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

÷ = 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 to and within the control room

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

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 a particular chemical form of an individual iodine isotope,

R_e = 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 for time frame II should equal the activity 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:

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

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:

$$A_{CR}(t) = [(C_2 * L_{2f} * A_{CT}(t)) \div C_1] \qquad (9)$$

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:

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

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 2 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. For the EAB doses, the radioactive material that escapes from the pool to the building is assumed to be released from the building over a two hour time period. This assumption is consistent with regulatory position C.1.i of Reg Guide 1.25 (Ref. 3).

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 2 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. A maximum average core burnup of 41.35 GWD/MTU is assumed consistent with item 58 on page B-19 of Ref. 7. This value corresponds to a maximum batch average discharge burnup of 55 GWD/MTU consistent with item 102 on page B-26 of Ref. 7. Since this batch is made up of assemblies that would be at burnup levels higher and lower than this value, the peak assembly value is assumed to be at a higher value (about 58 GW/D/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 occurs for noble gases.

22. The fission product inventory calculation (see Section 5.0) uses a multiplication factor of 30% on the activity calculated using the burnup assumed in assumption 22 for additional conservatism.

23. The atmospheric dispersion factors used is those for ground level releases. These values are more conservative than those for elevated releases (see for example Figures 1 and 3 of RG 1.25 (Ref. 3)). Note that releases from the containment equipment door are elevated releases and as such the atmospheric dispersion factors characteristic of these releases are expected to be smaller than the ground level release values.

5.0 INPUT DATA

Most of the input data employed in this analysis were developed using the Equivalence Table for Physics Assessment Checklist (EPAC, Reference 7) and Non-Physics Assessment Checklist (NPAC, Reference 15) for St. Lucie Unit 2. In addition, data from Regulatory Guides, NRC reports, Technical Specifications and the St. Lucie Unit 2 FSAR were used. Table 1 contains a listing of the input parameter and parameter values used in the analysis and identifies the sources of the values.

The source term data contained in Reference 7 is applicable to a maximum core average fuel burnup of 41,350 MWD/MTU (from page B-19, item 58 of EPAC, Ref. 7). This corresponds to a maximum batch average discharge burnup of 55,000 MWD/MTU and a peak assembly burnup of about 58,000 MWD/MTU (see assumption 18) for St. Lucie Unit 2. The source term activities given in Table B-14 of Reference 7 represent activities in the fuel rods immediately after shutdown. Since source term activities at 72 hours after shutdown is required for the current FHA analysis, it was decided to generate this data using the data of Reference 7 and decaying it for 72 hours based on the half lives of the isotopes of interest. The following procedure was used to develop this data.

First, the source term activities at shutdown provided in Table B-14 of Reference 7 were increased by a factor of 30% for added conservatism. Then the activities at 72 hours after shutdown were calculated using the source term data at shutdown and decaying the activity of the isotopes for 72 hours based on the half lives of the isotopes of interest. Using Equation (1-27) of Reference 16, the relationship between the activities at time zero (immediately after shutdown) and any time "t" after shutdown can be written as:

$$(A/A_o) = \exp(-\lambda t),$$

where, A = Activity of the isotope at time "t",
A_o = Activity of isotope at time zero (immediately after shutdown)
λ = decay constant for the isotope, sec^{-1}.

From Equation (1-28) of Reference 16,

$$\lambda = (0.6931/t_{1/2})$$

where, $t_{1/2}$ = Half life of isotope, sec

The above calculations for the isotopes of interest were performed using the Microsoft Excel spreadsheet program. Table 3 shows these calculations along with the results.

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 2-hour EAB and 8-hour LPZ inhalation thyroid doses are determined to be 61.6 rems and 26.7 rems, respectively. The corresponding EAB and LPZ whole body doses are calculated to be 0.75 rem (including the dose due to beta radiation) and 0.321 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 9.[illegible]9 rems, the whole body gamma dose is 0.02 rem, and the beta skin dose is 0.58 rem. 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 equipment door and PAL doors fully open. Therefore, the relaxation of Tech Spec 3.9.4 to carry out refueling operations in Mode 6 with the equipment door and PAL doors open is justified.

9.0 REFERENCES

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TABLE 1

Input Data for Analysis of Fuel Handling Accident for St. Lucie Unit 2 with the Containment Equipment Door and 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:	236	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	10	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 immediately after shutdown from full power for the iodine and noble gas isotopes of interest (curies):		Compiled from from Ref. 7 with 30 % increase for added conservatism
I-131	9.847E+07	
I-132	1.433E+08	

I-133	1.997E+08
I-135	1.863E+08
Xe-131m	1.102E+06
Xe-133	1.937E+08
Xe-133m	6.219E+06
Xe-135	6.489E+07
Xe-135m	3.955E+07
Kr-85	1.335E+06
Kr-85m	3.096E+07

Core inventory source term for the iodine and noble gas isotopes of interest 72 hours after shutdown (curies) :	Calculated by decaying activities at full power
I-131	7.603E+07
I-132	2.016E-02
I-133	1.834E+07
I-135	9.799E+04
Xe-131m	9.240E+05
Xe-133	1.303E+08
Xe-133m	2.407E+06
Xe-135	2.694E+05
Xe-135m	1.718E-76
Kr-85	1.334E+06
Kr-85m	5.735E+02

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

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Atmospheric Dispersion Factors (sec/m^3):		
0-2 hr Exclusion Area Boundary	1.64E-04	FSAR
0-8 hr Low Population Zone	7.10E-05	FSAR
Control Room		FSAR
0-8 hrs	5.00E-04	
8-24 hrs	3.00E-04	
1-4 days	1.17E-04	
4-30 days	3.35E-05	
Control Room in-leakage rate (cfm):		TS
Total	450	
Unfiltered	100	
Filtered	350	
Maximum Time after Initiation of FHA of Containment Isolation Signal (CIS) Actuation on High Containment Radiation (min):	30	Assumed
Control Room Charcoal Adsorber Efficiency (%):	90	Conservative with respect to TS value of 99 %
Control Room Recirculation Flow Rate (cfm)	2000	TS
Breathing Rate, B (m^3/sec)	3.47 x 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	

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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	.48E+6	
I-132	5.35E+4	
I-133	4.00E+4	
I-135	.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	.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	

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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)	2763.74	FSAR Section 6.4.2.2

* Derived from "standard man" parameters recommended in ICRP Publication 2.
TS: Technical Specifications
RG: Regulatory Guide
FSAR: St. Lucie Unit 2 Final Safety Analysis Report, Amendment 12

TABLE 2

SUMMARY OF RESULTS*** VS. ACCEPTANCE CRITERIA
Radiological Releases for the Fuel Handling Accident for St. Lucie Unit 2
with the Containment Equipment Door and 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)	61.6	75	0.75	6	*	**
LPZ Doses (rems)	26.7	75	0.33	6	*	**
Control Room Doses (rems)	9.39	30	0.02	5	0.58	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.

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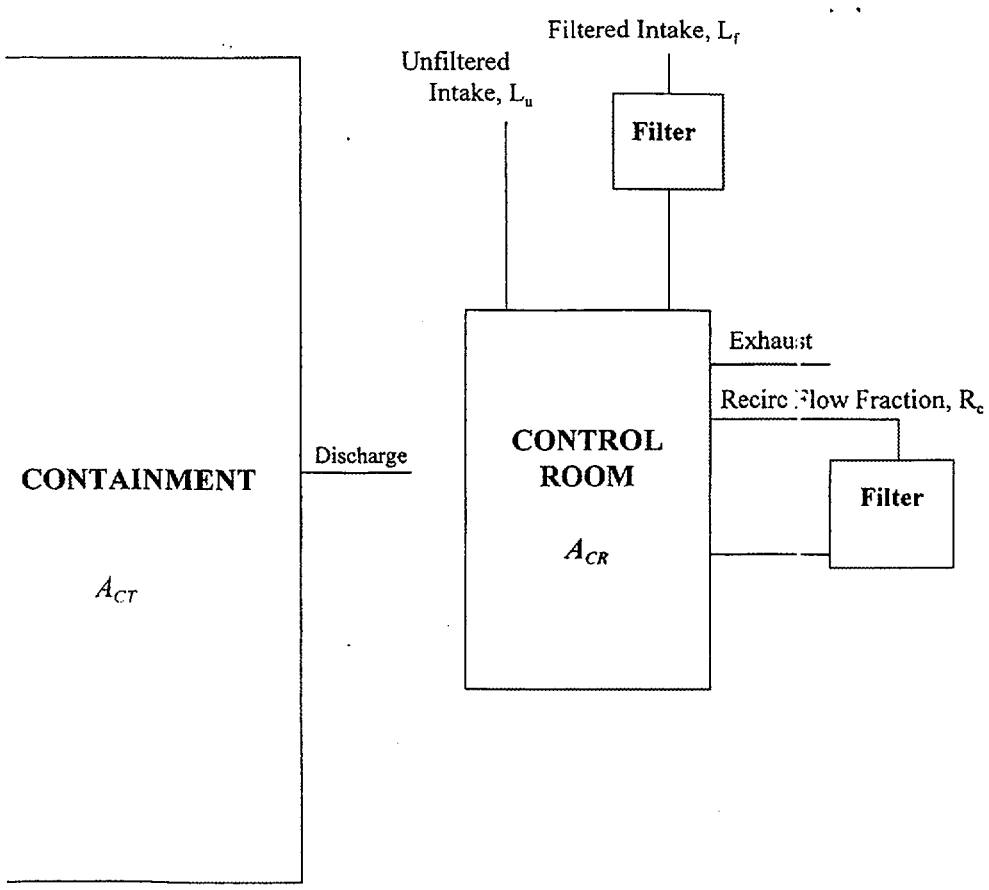
TABLE 3
St. Lucie Unit 2 Source Term Calculation for time zero and 72 hours after Shutdown
(Average Bundle Values)

		SL2 SOURCE TERM	ZERO & 72 HRS	AFTER SHUTDOWN			
F=			1				
P=			1.65				
B=			3.47E-04				
X/Q EAB=			1.84E-04				
X/Q LPZ=			7.10E-05				
DFp=			100				
DFI=			1				
NFA=			217				
RATIO1=	1.3						
RATIO2=	1						
NFR=	51212						
	SL2-FE-0181, Rev 9						
Isotope	CIROD-41350	CIROD- 41350+30%	TCI_D	HL	LAMBDA	TCI_72	TCI/NFA
I-131	1.479E+03	1.923E+03	9.847E+07	6.947E+05	9.978E-07	7.60[illegible]5387E+07	3.503E+05
I-132	2.153E+03	2.799E+03	1.433E+08	7.920E+03	8.752E-05	2.0[illegible]58971E-02	9.290E-05
I-133	3.000E+03	3.900E+03	1.997E+08	7.524E+04	9.212E-06	1.83[illegible]19826E+07	8.452E+04
I-135	2.798E+03	3.637E+03	1.863E+08	2.360E+04	2.913E-05	9.[illegible]9868E+04	4.516E+02
Xe-131m	1.656E+01	2.153E+01	1.102E+06	1.017E+06	6.616E-07	9.[illegible]3948E+05	4.258E+03
Xe-133	2.909E+03	3.782E+03	1.937E+08	4.536E+05	1.528E-06	1.303[illegible]9182E+08	6.006E+05
Xe-133m	9.342E+01	1.214E+02	6.219E+06	1.892E+05	3.663E-06	2.4[illegible]6496E+06	1.109E+04
Xe-135	9.747E+02	1.267E+03	6.489E+07	3.276E+04	2.116E-05	2.[illegible]9408E+05	1.242E+03
Xe-135m	5.940E+02	7.722E+02	3.955E+07	9.360E+02	7.405E-04	1.[illegible]71771E-76	7.916E-79
Kr-85	2.005E+01	2.607E+01	1.335E+06	3.377E+08	2.053E-09	1.3[illegible]4131E+06	6.148E+03
Kr-85m	4.651E+02	6.046E+02	3.096E+07	1.649E+04	4.204E-05	5.[illegible]3472E+02	2.643E+00

FIGURE 1
ACTIVITY TRANSPORT FOR THE CONTROL ROOM



APPENDIX A

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

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Spread Sheet for EAB & LPZ Thyroid Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K
1	SL2 EAB & LPZ	INHALATION	THYROID DOSES								
2	F=	1									
3	P=	1.65									
4	B=	3.47E-04									
5	X/Q EAB=	1.64E-04									
6	X/Q LPZ=	7.10E-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	7.603E+07	3.50E+05	1.48E+06	5.84E+01					
17	0-2 hr/I-132	0.1	2.016E-02	9.29E-05	5.35E+04	4.67E-10					
18	0-2 hr/I-133	0.1	1.834E+07	8.45E+04	4.00E+05	3.17E+00					
19	0-2 hr/I-135	0.1	9.799E+04	4.52E+02	1.24E+05	5.26E-03					
20	Total					6.16E+01					
21											
22	0-8 hr/I-131	0.12	7.603E+07	3.50E+05	1.48E+06	2.53E+01					
23	0-8 hr/I-132	0.1	2.016E-02	9.29E-05	5.35E+04	2.02E-10					
24	0-8 hr/I-133	0.1	1.834E+07	8.45E+04	4.00E+05	1.37E+00					
25	0-8 hr/I-135	0.1	9.799E+04	4.52E+02	1.24E+05	2.28E-03					
26	Total					2.67E+01					
27											
28											

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Spread Sheet for EAB & LPZ Thyroid Dose Calculation with formulas

	A	B	C	D	E	F
1	SL2 EAB & LPZ	INHALATION	THYROID DOSES			
2	F=	1				
3	P=	1.65				
4	B=	0.000347				
5	X/Q EAB=	0.000164				
6	X/Q LPZ=	0.000071				
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	76025387	=C16/B9	1480000	=(D16*B2*B3*B16*B4*E16*B5)/(B7*B8)
17	0-2 hr/I-132	0.1	0.020158971	=C17/B9	53500	=(D17*B2*B3*B17*B4*E17*B5)/(B7*B8)
18	0-2 hr/I-133	0.1	18339826	=C18/B9	400000	=(D18*B2*B3*B18*B4*E18*B5)/(B7*B8)
19	0-2 hr/I-135	0.1	97986.8	=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	76025387	=C22/B9	1480000	=(D22*B2*B3*B22*B4*E22*B6)/(B7*B8)
23	0-8 hr/I-132	0.1	0.020158971	=C23/B9	53500	=(D23*B2*B3*B23*B4*E23*B6)/(B7*B8)
24	0-8 hr/I-133	0.1	18339826	=C24/B9	400000	=(D24*B2*B3*B24*B4*E24*B6)/(B7*B8)
25	0-8 hr/I-135	0.1	97986.8	=C25/B9	124000	=(D25*B2*B3*B25*B4*E25*B6)/(B7*B8)
26	Total					=SUM(F22:F25)
27						
28						

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Spread Sheet for EAB & LPZ Whole Body Gamma Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M
1		SL2	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	9.239E+05	1.00E-01	7.03E+02	5.76E-04						
5	0-2 hr/Xe-133	0.25	4.50E-02	1.303E+06	1.00E-01	9.91E+04	1.89E-01						
6	0-2 hr/Xe-133m	0.25	4.16E-02	2.406E+06	1.00E-01	1.83E+03	3.12E-03						
7	0-2 hr/Xe-135	0.25	2.47E-01	2.694E+05	1.00E-01	2.05E+02	2.07E-03						
8	0-2 hr/Xe-135m	0.25	4.32E-01	1.718E-76	1.00E-01	1.31E-79	2.31E-84						
9	0-2 hr/Kr-85	0.25	2.00E-03	1.334E+06	3.00E-01	3.04E+03	2.50E-04						
10	0-2 hr/Kr-85m	0.25	1.59E-01	5.735E+02	1.00E-01	4.36E-01	2.84E-06						
11													
12	Total						1.89E-01						
13													
14	0-8 hr/Xe-131	0.25	2.00E-02	9.239E+05	1.00E-01	7.03E+02	2.49E-04						
15	0-8 hr/Xe-133	0.25	4.50E-02	1.303E+06	1.00E-01	9.91E+04	7.92E-02						
16	0-8 hr/Xe-133m	0.25	4.16E-02	2.406E+06	1.00E-01	1.83E+03	1.35E-03						
17	0-8 hr/Xe-135	0.25	2.47E-01	2.694E+05	1.00E-01	2.05E+02	8.98E-04						
18	0-8 hr/Xe-135m	0.25	4.32E-01	1.718E-76	1.00E-01	1.31E-79	1.00E-84						
19	0-8 hr/Kr-85	0.25	2.00E-03	1.334E+06	3.00E-01	3.04E+03	1.08E-04						
20	0-8 hr/Kr-85m	0.25	1.59E-01	5.735E+02	1.00E-01	4.36E-01	1.23E-06						
21													
22	Total						8.18E-02						
23													
24	DCF per Mev=	0.25											
25	PF=	1.65											
26	X/Q 2 hr =	1.64E-04											
27	X/Q 8 hr =	7.10E-05											
28	NFA=	217											
29													
30													

Note: The whole-body dose contribution due to the Iodines is negligible in comparison to that due to the noble gases, since their DF's differ by a factor of 100.

Spread Sheet for EAB & LPZ Whole Body Gamma Dose Calculation with Formulas

	A	B	C	D	E	F	G
1		SL2	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	923948	0.1	=(D4/B$28)*B$25*E4	=B4*C4*B$26*F4
5	0-2 hr/Xe-133	0.25	0.045	130329182	0.1	=(D5/B$28)*B$25*E5	=B5*C5*B$26*F5
6	0-2 hr/Xe-133m	0.25	0.0416	2406496	0.1	=(D6/B$28)*B$25*E6	=B6*C6*B$26*F6
7	0-2 hr/Xe-135	0.25	0.247	269408	0.1	=(D7/B$28)*B$25*E7	=B7*C7*B$26*F7
8	0-2 hr/Xe-135m	0.25	0.432	1.71771E-76	0.1	=(D8/B$28)*B$25*E8	=B8*C8*B$26*F8
9	0-2 hr/Kr-85	0.25	0.002	1334131	0.3	=(D9/B$28)*B$25*E9	=B9*C9*B$26*F9
10	0-2 hr/Kr-85m	0.25	0.159	573.472	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	923948	0.1	=(D14/B$28)*B$25*E14	=B14*C14*B$27*F14
15	0-8 hr/Xe-133	0.25	0.045	130329182	0.1	=(D15/B$28)*B$25*E15	=B15*C15*B$27*F15
16	0-8 hr/Xe-133m	0.25	0.0416	2406496	0.1	=(D16/B$28)*B$25*E16	=B16*C16*B$27*F16
17	0-8 hr/Xe-135	0.25	0.247	269408	0.1	=(D17/B$28)*B$25*E17	=B17*C17*B$27*F17
18	0-8 hr/Xe-135m	0.25	0.432	1.71771E-76	0.1	=(D18/B$28)*B$25*E18	=B18*C18*B$27*F18
19	0-8 hr/Kr-85	0.25	0.002	1334131	0.3	=(D19/B$28)*B$25*E19	=B19*C19*B$27*F19
20	0-8 hr/Kr-85m	0.25	0.159	573.472	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 =	0.000164					
27	X/Q 8 hr =	0.000071					
28	NFA=	217					

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Spread Sheet for EAB & LPZ Whole Body Beta Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M
1	SL2	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	9.239E+05	1.00E-01	7.03E+02	3.79E-03						
5	0-2 hr/Xe-133	0.23	1.35E-01	1.303E+08	1.00E-01	9.91E+04	5.05E-01						
6	0-2 hr/Xe-133m	0.23	1.90E-01	2.406E+06	1.00E-01	1.83E+03	1.31E-02						
7	0-2 hr/Xe-135	0.23	3.16E-01	2.694E+05	1.00E-01	2.05E+02	2.44E-03						
8	0-2 hr/Xe-135m	0.23	9.50E-02	1.718E-76	1.00E-01	1.31E-79	4.68E-85						
9	0-2 hr/Kr-85	0.23	2.51E-01	1.334E+06	3.00E-01	3.04E+03	2.88E-02						
10	0-2 hr/Kr-85m	0.23	2.53E-01	5.735E+02	1.00E-01	4.36E-01	4.16E-06						
11													
12	Total						5.53E-01						
13													
14	0-8 hr/Xe-131	0.23	1.43E-01	9.239E+05	1.00E-01	7.03E+02	1.64E-03						
15	0-8 hr/Xe-133	0.23	1.35E-01	1.303E+08	1.00E-01	9.91E+04	2.18E-01						
16	0-8 hr/Xe-133m	0.23	1.90E-01	2.406E+06	1.00E-01	1.83E+03	5.68E-03						
17	0-8 hr/Xe-135	0.23	3.16E-01	2.694E+05	1.00E-01	2.05E+02	1.06E-03						
18	0-8 hr/Xe-135m	0.23	9.50E-02	1.718E-76	1.00E-01	1.31E-79	2.03E-85						
19	0-8 hr/Kr-85	0.23	2.51E-01	1.334E+06	3.00E-01	3.04E+03	1.25E-02						
20	0-8 hr/Kr-85m	0.23	2.53E-01	5.735E+02	1.00E-01	4.36E-01	1.80E-06						
21													
22	Total						2.39E-01						
23													
24	DCF per4 Mev=	0.23											
25	PF=	1.65											
26	X/Q 2 hr =	1.64E-04											
27	X/Q 8 hr =	7.10E-05											
28	NFA=	217											

Note: The whole-body beta dose contribution due to the Iodines is negligible in comparison to that due to the noble gases, since their DF's differ by a factor of 100.

Westinghouse Nuclear Systems

Spread Sheet for EAB & LPZ Whole Body Beta Dose Calculation with Formulas

	A	B	C	D	E	F	G	H
1	SL2	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	923948	0.1	=(D4/B$28)*B$25*E4	=B4*C4*B$26*F4	
5	0-2 hr/Xe-133	0.23	0.135	130329182	0.1	=(D5/B$28)*B$25*E5	=B5*C5*B$26*F5	
6	0-2 hr/Xe-133m	0.23	0.19	2406496	0.1	=(D6/B$28)*B$25*E6	=B6*C6*B$26*F6	
7	0-2 hr/Xe-135	0.23	0.316	269408	0.1	=(D7/B$28)*B$25*E7	=B7*C7*B$26*F7	
8	0-2 hr/Xe-135m	0.23	0.095	1.71771E-76	0.1	=(D8/B$28)*B$25*E8	=B8*C8*B$26*F8	
9	0-2 hr/Kr-85	0.23	0.251	1334131	0.3	=(D9/B$28)*B$25*E9	=B9*C9*B$26*F9	
10	0-2 hr/Kr-85m	0.23	0.253	573.472	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	923948	0.1	=(D14/B$28)*B$25*E14	=B14*C14*B$27*F14	
15	0-8 hr/Xe-133	0.23	0.135	130329182	0.1	=(D15/B$28)*B$25*E15	=B15*C15*B$27*F15	
16	0-8 hr/Xe-133m	0.23	0.19	2406496	0.1	=(D16/B$28)*B$25*E16	=B16*C16*B$27*F16	
17	0-8 hr/Xe-135	0.23	0.316	269408	0.1	=(D17/B$28)*B$25*E17	=B17*C17*B$27*F17	
18	0-8 hr/Xe-135m	0.23	0.095	1.71771E-76	0.1	=(D18/B$28)*B$25*E18	=B18*C18*B$27*F18	
19	0-8 hr/Kr-85	0.23	0.251	1334131	0.3	=(D19/B$28)*B$25*E19	=B19*C19*B$27*F19	
20	0-8 hr/Kr-85m	0.23	0.253	573.472	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 =	0.000164						
27	X/Q 8 hr =	0.000071						
28	NFA=	217						
29								
30								

APPENDIX B

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

Westinghouse Nuclear Systems

Spread Sheet for Control Room Thyroid Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q
1			SL2	CONTROL ROOM		THYROID DOSE											
2	Lu=	0.047195			Luprime=	0.047195											
3	fR=	0.9			fRprime=	0.9											
4	RC=	0			Rcprime=	0.0003415											
5	VCR(ft3)=	97600 028	VCR(m3)=	2763 74													
6	Lf=	0.1651825	fCR=	0.9	Lfprime=	0											
7	X/Q=	5.00E-04	Lf/VCR=	5.97677E-05	Lfprime/VCR=	0											
8	Lu/VCR(m3)=	1.708E-05	FCR*Lf=	0 01651825	Luprime/VCR=	1.708E-05											
9	fR*RC=	0	FCR*Lfprime=	0	fRprime*RCprime=	0.0003074											
10	P=	1.65	NFA=	217													
11	DFp=	100	DFf=	1													
12	C2=	3.19E-05	C2prime=	2 35975E-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	7.76E-05	3.25E-04	7.603E+07	3 50E+05	0 12	0 024086384	1 33E-02	1.48E+06	8.914732633					
17	0-8 hr CR/I-132	8.21E+03	8.44E-05	1.61E-04	4.09E-04	2 016E-02	9.29E-05	0.1	5.32231E-12	2 40E-12	5.35E+04	5 79156E-11					
18	0-8 hr CR/I-133	7.52E+04	9 21E-06	8.61E-05	3.34E-04	1.834E+07	8.45E+04	0.1	0.004842024	2 62E-03	4.00E+05	0 473683737					
19	0-8 hr CR/I-135	2.38E+04	2.91E-05	1.06E-04	3.54E-04	9.799E+04	4.52E+02	0.1	2.58702E-05	1.33E-05	1.24E+05	0.000744729					
20										Total		9 389161099					
21																	
22	I-131						I-132					I-133					I-135
23	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		ACR Time, min
24	0	[illegible]	[illegible]	[illegible]			0	[illegible]	[illegible]	[illegible]		0	[illegible]	[illegible]	[illegible]		0
25	10	4 50E-04	0 0088934	0 000148223			10	9 70E-14	1.89454E-12	3.158E-14		10	9.02E-05	0.001781335	2.96889E-05		10
26	20	8.79E-04	0 0197326	0 000328877			20	1 85E-13	4.14386E-12	6.906E-14		20	1.76E-04	0.00394673	6.57788E-05		20
27	30	1.29E-03	0 0330262	0 000550436			30	2.65E-13	6.83866E-12	1.14E-13		30	2 57E-04	0.006596193	0 000109937		30
28	40	1.37E-03	0 0470596	0 000784326			40	2 74E-13	9 61542E-12	1.603E-13		40	2 73E-04	0.00938625	0.000156438		40
29	480	1.75E-03	0.7995821	0 013326368			480	3 07E-13	1.43701E-10	2 395E-12		480	3.42E-04	0.157197249	0 002619954		480
30	490	1.75E-03	0 8170459	0.013617432			490	3.07E-13	1.46772E-10	2.446E-12		490	3.42E-04	0 160621545	0 002677026		490
31	500	1 75E-03	0 8257778	0.013762964			500	3 07E-13	1.48308E-10	2.472E-12		500	3.42E-04	0.162333696	0.002705562		500

Westinghouse Nuclear Systems

Spread Sheet for Control Room Thyroid Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL2	CONTROL ROOM		THYROID DOSE						
2	Lu=	=(100*28317)/(60*1000000)			Luprime=	=(100*28317)/(60*1000000)						
3	fR=	0.9			fRprime=	0.9						
4	RC=	=(0*28317)/(60*1000000*2763.74)			Rcprime=	=(2000*28317)/(60*1000000*2763.74)						
5	VCR(ft3)=	=(2763.74*1000000)/28317	VCR(m3)=	2763.74								
6	Lf=	=(350*28317)/(60*1000000)	fCR=	0.9	Lfprime=	=(0*28317)/(60*1000000)						
7	X/Q=	0.0005	Lf/VCR=	=B6/D5	Lfprime/VCR=	=F6/D5						
8	Lu/VCR(m3)=	=B2/D5	FCR*Lf=	=(1-D6)*B6	Lupnme/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								
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	76025387	=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	8200	=LN(2)/B17	=B$8+B$9+D$7+C17	=F$8+F$7+F$9+C17	0.020158971	=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	18339826	=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	97986 8	=F19/217	0 1	=(G19*H19*B10)/(B11*D11*8*3600)	=T$75	124000	=(3600*K19*B$13*D$13*J19)/(D$5)
20										Total		=SUM(L16:L19)
22	I-131						I-132					I-133
23	ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr			ACR Time, min	ACR	IQM Ci-min	IQM Ci-hr		ACR Time, min
24	0	=((B$12*I$16)/(D$16)*(1-EXP(-D$16*60*A27))	=(B27+B28)*10/2+0	=C27/60			0	=((B$12*I$17)/(D$17)*(1-EXP(-D$17*60*G27))	=(H27+H28)*10/2+0	=I27/60		0
25	10	=((B$12*I$16)/(D$16)*(1-EXP(-D$16*60*A28))	=(B28+B29)*10/2+C27	=C28/60			10	=((B$12*I$17)/(D$17)*(1-EXP(-D$17*60*G28))	=(H28+H29)*10/2+I27	=I28/60		10
26	20	=((B$12*I$16)/(D$16)*(1-EXP(-D$16*60*A29))	=(B29+B30)*10/2+C28	=C29/60			20	=((B$12*I$17)/(D$17)*(1-EXP(-D$17*60*G29))	=(H29+H30)*10/2+I28	=I29/60		20
27	30	=((B$12*I$16)/(D$16)*(1-EXP(-D$16*60*A30))	=(B30+B31)*10/2+C29	=C30/60			30	=((B$12*I$17)/(D$17)*(1-EXP(-D$17*60*G30))	=(H30+H31)*10/2+I29	=I30/60		30
28	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*EXP(-E$17*60*(G31-30))+(D$12*I$17/E$17)*(1-EXP(-E$17*60*(G31-30)))	=(H31+H32)*10/2+I30	=I31/60		40

Westinghouse Nu. stems

Spread Sheet for Control Room Whole Body Gamma Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M
1			SL2	CONTROL ROOM	WHOLE	BODY	DOSE						
2	Lu=	0.2123775											
3	fR=	0											
4	RC=	0.00034153											
5	VCR(ft3)=	97600.028	VCR(m3)=	2763.74									
6	Lf=	0											
7	X/Q=	5.00E-04											
8	Lu/VCR(m3)=	7.684E-05											
9	fR*RC=	0											
10	P=	1.65											
11	DFp=	1	DFf=	1									
12	C2=	1.06E-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	7.75E-05	9.239E+05	4.26E+03	0.1	0.024393788	2.67E-01	2.90E-03	1.00E+00	4.18158E-05	
17	0-8 hr CR/Xe-133	4.54E+05	1.53E-06	7.84E-05	1.303E+08	6.01E+05	0.1	3.440910623	3.73E+01	9.32E-03	1.00E+00	0.018751515	
18	0-8 hr CR/Xe-133m	1.89E+05	3.66E-06	8.05E-05	2.406E+06	1.11E+04	0.1	0.063535561	6.70E-01	7.96E-03	1.00E+00	0.000287875	
19	0-8 hr CR/Xe-135	3.28E+04	2.12E-05	9.80E-05	2.694E+05	1.24E+03	0.1	0.007112826	6.17E-02	5.74E-02	1.00E+00	0.000190909	
20	0-8hr CR/Xe-135m	936	7.41E-04	8.17E-04	1.718E-76	7.92E-79	0.1	4.53504E-84	4.71E-84	9.89E-02	1.00E+00	2.51455E-86	
21	0-8hr CR/Kr-85	3.38E+08	2.05E-09	7.68E-05	1.334E+06	6.15E+03	0.3	0.105669938	1.17E+00	5.10E-04	1.00E+00	3.21372E-05	
22	0-8 hr CR/Kr-85m	16128	4.30E-05	1.20E-04	5.735E+02	2.64E+00	0.1	1.51406E-05	1.07E-04	3.71E-02	1.00E+00	2.14828E-07	
23										Total		0.019304866	
24													
25													
26													
27													
28													

Note: The whole-body dose contribution due to the Iodines is negligible in comparison to that due to the noble gases, since their DF's differ by a factor of 100.

Westinghouse Nuclear Systems

Spread Sheet for Control Room Whole Body Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL2	CONTROL ROOM	WHOLE	BODY	DOSE					
2	Lu=	=(450*28317)/(60*1 000000)										
3	IR=	0										
4	RC=	=(2000*28317)/(60* 1000000*2763 74)										
5	VCR(ft3)=	=(2763.74*1000000)/28317	VCR(m3)=	2763 74								
6	Lf=	0										
7	XQ=	0 0005										
8	Lu/VCR(m3)=	=B2/D5										
9	IR*RC=	=B3*B4										
10	[illegible]	1 65										
11	DF[illegible]=	1	OFF=	1								
12	C2=	=B7*B2										
13	B=	0 000347										
14												
15	Period/Isotope	HL	Lambda	C1	TCI	TCIVNFA	Fg	A2	IOM	DCF	CRO	Dwb
16	0-8 hr CR/Xe-131m	1016928	=LN(2)/B16	=B8+B9+C16	923948	=E16/217	0 1	=(F16*G16*B10)/(B11*D11*B*360 0)	=(B$12*H16*B)/D16	0.0029	1	=((B$5*0 338)*3600*J16*K16*116)/(1173*D$5)
17	0-8 hr CR/Xe-133	453600	=LN(2)/B17	=B8+B9+C17	130329182	=E17/217	0 1	=(F17*G17*B10)/(B11*D11*B*360 0)	=(B$12*H17*B)/D17	0 00932	1	=((B$5*0.338)*3600*J17*K17*117)/(1173*D$5)
18	0-8 hr CR/Xe-133m	189216	=LN(2)/B18	=B8+B9+C18	2408496	=E18/217	0 1	=(F18*G18*B10)/(B11*D11*B*360 0)	=(B$12*H18*B)/D18	0.00796	1	=((B$5*0 338)*3600*J18*K18*118)/(1173*D$5)
19	0-8 hr CR/Xe-135	32760	=LN(2)/B19	=B8+B9+C19	269408	=E19/217	0 1	=(F19*G19*B10)/(B11*D11*B*360 0)	=(B$12*H19*B)/D19	0 0574	1	=((B$5*0 338)*3600*J19*K19*119)/(1173*D$5)
20	0-8hr CR/Xe-135m	936	=LN(2)/B20	=B8+B9+C20	1.71771E-76	=E20/217	0 1	=(F20*G20*B10)/(B11*D11*B*360 0)	=(B$12*H20*B)/D20	0 0989	1	=((B$5*0 338)*3600*J20*K20*120)/(1173*D$5)
21	0-8hr CR/Kr-85	337666320	=LN(2)/B21	=B8+B9+C21	1334131	=E21/217	0 3	=(F21*G21*B10)/(B11*D11*B*360 0)	=(B$12*H21*B)/D21	0 00051	1	=((B$5*0 338)*3600*J21*K21*121)/(1173*D$5)
22	0-8 hr CR/Kr-85m	16128	=LN(2)/B22	=B8+B9+C22	573.472	=E22/217	0 1	=(F22*G22*B10)/(B11*D11*B*360 0)	=(B$12*H22*B)/D22	0 0371	1	=((B$5*0 338)*3600*J22*K22*122)/(1173*D$5)
23										Total		=SUM(L16:L22)
24												
25												

Westinghouse Nuclear Systems

Spread Sheet for Control Room Beta Skin Dose Calculation

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
1			SL2	CONTROL ROOM	SKIN DOSE										
2	Lu=	0.2123775													
3	IR=	0													
4	RC=	0.00034153													
5	VCR(ft3)=	97600.028	VCR(m3)=	2763.74											
6	Lf=	0													
7	X/Q=	5.00E-04													
8	Lu/VCR(m3)=	7.684E-05													
9	IR*RC=	0													
10	P=	1.65													
11	DFp=	1	DFf=	1											
12	C2=	1.06E-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	7.75E-05	9.239E+05	4.26E+03	0.1	0.024399788	2.67E-01	1.51E-02	1.00E+00	0.005257551			
17	0-8 hr CR/Xe-133	4.54E+05	1.53E-06	7.84E-05	1.303E+08	6.01E+05	0.1	3.440910623	3.73E+01	9.70E-03	1.00E+00	0.471255153			
18	0-8 hr CR/Xe-133m	1.89E+05	3.68E-06	8.05E-05	2.406E+06	1.11E+04	0.1	0.063535551	6.70E-01	3.15E-02	1.00E+00	0.027508373			
19	0-8 hr CR/Xe-135	3.28E+04	2.12E-05	9.80E-05	2.694E+05	1.24E+03	0.1	0.007112826	5.17E-02	5.90E-02	1.00E+00	0.004738387			
20	0-8hr CR/Xe-135m	936	7.41E-04	8.17E-04	1.718E-76	7.92E-79	0.1	4.53504E-84	4.71E-84	2.25E-02	1.00E+00	1.38137E-85			
21	0-8hr CR/Kr-85	3.38E+08	2.05E-09	7.68E-05	1.334E+06	6.15E+03	0.3	0.105669938	1.17E+00	4.24E-02	1.00E+00	0.064516114			
22	0-8 hr CR/Kr-85m	16128	4.30E-05	1.20E-04	5.735E+02	2.64E+00	0.1	1.51406E-05	1.07E-04	4.63E-02	1.00E+00	6.47384E-06			
23										Total		0.573282051			
24															
25															
26															
27															

Note: The beta skin dose contribution due to the Iodines is negligible in comparison to that due to the noble gases, since their DF's differ by a factor of 100.

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Spread Sheet for Control Room Beta Skin Dose Calculation with Formulas

	A	B	C	D	E	F	G	H	I	J	K	L
1			SL2	CONTROL ROOM	SKIN DOSE							
2	Lu=	=(450*28317)/(60*1000000)										
3	fR=	0										
4	RC=	=(2000*28317)/(60*1000000*2763.74)										
5	VCR(ft3)=	=(2763.74*1000000)/28317	VCR(m3)=	2763.74								
6	Lf=	0										
7	X/Q=	0.0005										
8	Lu/VCR(m3)=	=B2/D5										
9	fR*RC=	=B3*B4										
10	P=	1.65										
11	DFp=	1	DFI=	1								
12	C2=	=B7*B2										
13	B=	0.000347										
14												
15	Period/Isotope	HL	Lambda	C1	TCI	TCWNFA	Fg	L21A2	IQM	DCF	CRO	Dwb
16	0-8 hr CR/Xe-131m	1016928	=LN(2)/B16	=B8+B9+C16	923948	=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	130329182	=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	2406496	=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	269408	=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	1.71771E-76	=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	1334131	=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	573.472	=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												

APPENDIX C

DESIGN VERIFICATION CHECKLIST

QPF 0306-1

Westinghouse Nuclear Systems

CE Nuclear Power LLC DESIGN VERIFICATION CHECKLIST QPF 0306-1 DESIGN DOCUMENT NO: L-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.	✓	

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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.		

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The use of 30% increase in fission product inventory is another significant conservatism in the calculation. EAB dose calculation assumes all activity released from fuel is transported tc the exclusion area boundary in 2 hours. Results are consistent with general predictions for St. Lucie Unit 1 Fuel Handling Accident analysis with the containment equipment door and PAL doors open (Calculation No. F-FSA-C-000001, Rev. 00).		
Independent Reviewer: R. E. Schneider, Senior Consultant, 2/28/01 Name, Title, Signature, Date		

APPENDIX D

DESIGN DOCUMENT REVIEWER'S COMMENT CHECKLIST/FORM

QPF 0302-1

Westinghouse Nuclear Systems

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 2 Tech Spec 3.9.4		**PAGE** D-2 **OF** D-2	
DOCUMENT NUMBER: L-FSA-C-000001		**REVISION NUMBER:** 00	
Comm. No.	**Reviewer's Comment**	**Resp. Req'd?**	**Author's Response**	**Resp. Accept-ed?**
1	The use of 30% increase in fission product should be indicated as a conservatism and should be included in Section 4.0 "Assumptions & Justification".	Yes	Assumption 23 is added in Section 3.0 indicating this conservative assumption.	Yes
2	The use of a peak assembly burnup of 58 GWD/MTU should be identified somewhere in the text to tie in the average burnup of 41.35 GWD/MTU.	Yes	Assumption 22 is added to Section 4.0 to indicate the connection between the 41.35 GWD/MTU to the batch average discharge burnup of 55 GWD/MTU and the peak assembly burnup of 58 GWD/MTU.	Yes
3	Various editorial changes identified on the marked-up hardcopy document are recommended to improve clarity and readability.	Yes	Editorial changes recommended are incorporated.	Yes
4	Values of doses contained in the "Conclusions" Section and Table 2 should be consistent with the values calculated in the Excel spreadsheets.	Yes	Values of doses were made consistent with the values calculated in the Excel spreadsheets.	Yes

Checklist Completed By Reviewer:		
R. E. Schneider	R.E. Schneider	2/28/0
Printed Name	**Signature**	**Date**