Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428
Docket 50-302
Docket 72-1035
Operating License No. DPR-72

10 CFR 50.90

March 31, 2016
3F0316-01

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

Subject: Crystal River Unit 3 – License Amendment Request #320, Revision 0, Cyber Security Plan Implementation Schedule Milestone 8

References:

1. NRC letter, *Crystal River Unit 3 Nuclear Generating Plant Certification of Permanent Cessation of Operation and Permanent Removal of Fuel from the Reactor*, dated March 13, 2013 (ADAMS Accession No. ML13058A380)
2. Duke Energy letter, *License Amendment Request – Cyber Security Plan Implementation Schedule Milestone 8*, dated December 19, 2013 (ADAMS Accession No. ML13357A189)
3. NRC letter, *Crystal River Unit 3 Nuclear Generating Plant Issuance of License Amendments Regarding Revision to Cyber Security Plan Implementation Schedule Completion Date*, dated December 19, 2014 (ADAMS Accession No. ML14318A929)

Dear Sir:

Pursuant to 10 CFR 50.90, Duke Energy Florida, LLC, previously known as Duke Energy Florida, Inc. (DEF), hereby requests a license amendment pertaining to the Cyber Security Plan (CSP) implementation schedule, including a proposed revision to the existing Physical Protection license condition for the facility operating license. Specifically, the completion date for Milestone 8 is proposed to be changed from December 31, 2017 to December 31, 2018.

Crystal River Unit 3 (CR-3) has been shutdown since September 26, 2009. On February 5, 2013, DEF announced that CR-3 would be retired. DEF notified the Nuclear Regulatory Commission (NRC) on February 20, 2013 of the permanent cessation of power operations and that CR-3 had removed all fuel from the reactor. By letter dated March 13, 2013 (Reference 1), the NRC acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel. Accordingly, pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

By letter dated December 19, 2013 (Reference 2), CR-3 was part of a Duke Energy Fleet license amendment request pertaining to the CSP implementation Milestone 8 schedule, including a proposed revision to the existing Physical Protection license condition for the associated facility operating licenses.

By letter dated December 19, 2014 (Reference 3), the NRC approved the associated CSP implementation schedule and revised the CR-3 license condition, to require the licensee to fully implement and maintain in effect all provisions of the NRC-approved CSP by December 31, 2017.

The CSP implementation schedule provided in Reference 2 listed a completion date of December 31, 2017 for Milestone 8. Milestone 8 pertains to the date that full implementation of the CSP for all Safety, Security, and Emergency Preparedness (SSEP) functions will be achieved. As stated in Reference 3, subsequent changes to the NRC-approved CSP implementation schedule require prior NRC approval pursuant to 10 CFR 50.90. Accordingly, pursuant to the provisions of 10 CFR 50.4 and 10 CFR 50.90, DEF is submitting this request for an amendment to the CR-3 facility operating license to propose a change in the completion date for Milestone 8 to December 31, 2018.

Attachment A provides an evaluation of the proposed change. Attachment B provides a summary of engineering calculations that support the proposed schedule change. A marked-up and redline version of the facility operating license pages for the Physical Protection license condition, reflecting the commitment change proposed in this submittal, are included as Attachment C and D.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed change involves no significant hazards consideration. The bases for these determinations are included in Attachment A.

NRC approval of this license amendment application is requested within one year of the date of this submittal. Once approved, the license amendment will be implemented within 30 days.

This submittal contains a revised regulatory commitment as identified in Attachment E. In accordance with 10 CFR 50.91, DEF is notifying the State of Florida of this license amendment request by transmitting a copy of this letter and enclosures to the designated State Officials.

There are no new regulatory commitments made within this submittal.

If you have any questions regarding this submittal, please contact Mr. Mark Van Sicklen, Licensing Lead, Nuclear Regulatory Affairs, at (352) 563-4795.

Sincerely,

Ronald R. Reising, Senior Vice President
Operations Support

RRR/mvs

Attachments:

A. Evaluation of the Proposed Change
B. Engineering Calculations Summary
C. Facility Operating License Strikeout Pages
D. Facility Operating License Revision Bar Pages
E. Revised Regulatory Commitment

xc: NMSS Project Manager
Regional Administrator, Region I
State of Florida

DUKE ENERGY FLORIDA, LLC

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ATTACHMENT A

EVALUATION OF THE PROPOSED CHANGE

EVALUATION OF THE PROPOSED CHANGE

1. SUMMARY DESCRIPTION

This license amendment request (LAR) includes a proposed change to Milestone 8 of the Cyber Security Plan implementation schedule and a proposed revision to the existing Physical Protection license condition for the CR-3 facility operating license. The completion date for Milestone 8 is proposed to be changed from December 31, 2017 to December 31, 2018.

2. DETAILED DESCRIPTION

Pursuant to 10 CFR 50.90, DEF hereby requests a license amendment pertaining to the Cyber Security Plan (CSP) implementation schedule, including a proposed revision to the existing Physical Protection license condition for the facility operating license. The completion date for Milestone 8 is proposed to be changed from December 31, 2017, to December 31, 2018.

Crystal River Unit 3 (CR-3) has been shutdown since September 26, 2009. On February 5, 2013, DEF announced that CR-3 would be retired. DEF notified the Nuclear Regulatory Commission (NRC) on February 20, 2013 of the permanent cessation of power operations and that CR-3 had removed all fuel from the reactor. By letter dated March 13, 2013 (Reference 1), the NRC acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel. Accordingly, pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for CR-3 no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel.

By letter dated December 19, 2013 (Reference 2), CR-3 was part of a Duke Energy Fleet license amendment request pertaining to the CSP implementation Milestone 8 schedules, including a proposed revision to the existing Physical Protection license condition for the associated facility operating licenses.

By letter dated December 19, 2014 (Reference 3), the NRC approved the associated CSP implementation schedule and revised the CR-3 license condition, to require the licensees to fully implement and maintain in effect all provisions of the NRC-approved CSP by December 31, 2017.

The CSP implementation schedule provided in Reference 2 listed a completion date for Milestone 8. Milestone 8 pertains to the date that full implementation of the CSP for all Safety, Security, and Emergency Preparedness (SSEP) functions will be achieved. As stated in Reference 3, subsequent changes to the NRC-approved CSP implementation schedule require prior NRC approval pursuant to 10 CFR 50.90. Accordingly, pursuant to the provisions of 10 CFR 50.4 and 10 CFR 50.90, DEF is submitting this request for an amendment to the CR-3 facility operating license to propose a change in the completion date for Milestone 8.

Attachment B provides a summary of engineering calculations that support the proposed schedule change, from a risk perspective. A marked-up and redline version of the facility operating license pages for the Physical Protection license condition, reflecting the commitment change proposed in this submittal, are included as Attachment C and D.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed change involves no significant hazards consideration. The bases for these determinations are discussed in Section 4, Regulatory Evaluation.

3. TECHNICAL EVALUATION

CR-3 was part of a Duke Energy Fleet LAR pertaining to the CSP implementation Milestone 8 schedules, including a proposed revision to the existing Physical Protection license conditions for the associated facility operating licenses. The NRC approved the associated CSP implementation schedule and revised the license condition to the CR-3 facility operating license, to require the licensee to fully implement and maintain in effect all provisions of the NRC-approved CSP by December 31, 2017.

While reassessing the scope and resource requirements following the permanent cessation of operations of CR-3, it has been recognized that the previous CSP Milestones were completed as if CR-3 would still be an operating facility. Since that time, the nuclear facility has significantly changed, all safety related systems and most of the systems important to safety are abandoned. Connections to originally designed plant cooling systems have been severed and those systems are now abandoned. Currently, the spent fuel has decayed enough that the addition of inventory would be sufficient for cooling. A new nuclear island has been established to independently cool the spent nuclear fuel.

Attachment B provides a summary of the analysis performed in 2013. It demonstrates that the current condition of the Spent Fuel Pool (SFP) poses a very low risk to the health and safety of the public. Because the CR-3 reactor was last critical on September 26, 2009 and all spent fuel has been stored in the SFPs since May 28, 2011, the spent fuel has significantly decayed. As a result, CR-3 has determined that the only remaining credible design basis accident in the permanently defueled condition is the Fuel Handling Accident (FHA).

Accident source terms no longer contain particulates (alkali metals) or halogens (iodines); the only source term is from noble gas Krypton-85 that contributes to dose consequences. Based on analysis, with no ventilation across the SFP, the time to uncover fuel in the SFP during a complete loss of forced cooling is estimated at 34 days with an estimated average heat-up rate of the SFP of 0.43°F/hr and an estimated boil off rate of approximately 6 gpm. Analysis for the Fuel Handling Accident shows the following dose consequences compared to the limits (in brackets):

Control Room	1.3E-04 rem TEDE	[5 rem TEDE]
Exclusion Area Boundary	5.9E-05 rem TEDE	[6.3 rem TEDE]
Low Population Zone	1.0E-05 rem TEDE	[6.3 rem TEDE]

These results demonstrate that the licensing basis accident scenario generates very low doses. The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) limit of 6.3 rem TEDE is taken from Regulatory Guide 1.183 and is 25% of the 25 rem offsite dose limits of 10 CFR 50.67. The 5 rem TEDE Control Room dose limit is also established in 10 CFR 50.67.

Fire risk at CR-3 has been significantly reduced compared to an operating unit due to a number of factors. Fire Protection Program administrative controls (hot work, transient combustibles, compensatory measures, etc.) remain in effect while fire hazards have been reduced. All large quantities of combustible liquids (cooling or lubricating oils, diesel fuel, etc.) have been removed from the plant except the remaining two backup diesel generators and one fire pump and there has been a significant reduction in operating equipment and their power supplies. Plant systems within yard area buildings have been decommissioned and normal access to those areas is restricted. With the exception of limited term projects (fuel handling building structural upgrades and Ready Stores Building demolition), most plant activities are related to surveillances or minor maintenance. The Permanently Defueled Emergency Plan (PDEP) has been significantly reduced from the original Emergency Plan. It no longer has any Emergency Action Level that is at the Site Area or General Emergency classification level.

DEF believes that actions taken in accordance with Milestones 1 through 7 provide a high degree of protection against cyber security related attacks until the full program (Milestone 8) would be completed or all fuel would be removed from the plant and placed in the Independent Spent Fuel Storage Installation (ISFSI). The ISFSI construction project is ongoing and planned completion of SFP offload is currently scheduled for February 2018. These actions include, in part, implementation of the Portable Media/Mobile Device (PMMD) Program, design process controls on digital modifications to identify critical systems and critical digital assets and the assessment and remediation of the Plant Security System. The Plant Security System was chosen first to assess and determine remediations. CR-3 is focused on the cyber-security remediations of the Plant Security System which are scheduled to be completed on or before December of 2017.

Based on the status of the cyber security program, the completion of the cyber-security remediations of the Plant Security System, the reduced fire risk, the reduced PDEP Emergency Action Levels, the time since last reactor operation and results of the engineering calculations mentioned above, DEF believes that the proposed revised completion date for Milestone 8 of December 31, 2018, is justified.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 73.54 requires licensees to maintain and implement a Cyber Security Plan. The Crystal River Unit 3 Nuclear Generating Plant Facility Operating License (No. DPR-72), includes a Physical Protection license condition that requires the respective licensees to fully implement and maintain in effect all provisions of the Commission-approved Cyber Security Plan, including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p).

4.2 No Significant Hazards Consideration Determination

DEF is requesting an amendment to the Crystal River Unit 3 Nuclear Generating Plant Facility Operating License to revise the Physical Protection license condition 2.D as it relates to the Cyber Security Plan. The current license condition reflects the Cyber Security Plan implementation schedule previously approved by the NRC, which listed a completion date of December 31, 2017, for Milestone 8. Milestone 8 pertains to the date that full implementation of the Cyber Security Plan for all Safety, Security, and Emergency Preparedness (SSEP) functions will be achieved. A revised Milestone 8 completion date of December 31, 2018, is requested.

DEF has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed one year extension to the Cyber Security Plan implementation schedule for Milestone 8 does not alter the Fuel Handling Accident analysis, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. The proposed change does not require any plant modifications that affect the performance capability of the structures, systems, and components relied upon to mitigate the consequences of postulated accidents and have no impact on the probability or consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change to the Cyber Security Plan implementation schedule for Milestone 8 does not alter accident analysis assumptions, add any initiators, or affect the function of plant systems or the manner in which systems are operated, maintained, modified, tested, or inspected. The proposed change does not require any plant modifications that affect the performance capability of the structures, systems, and components relied upon to mitigate the consequences of postulated accidents and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Plant safety margins are established through limiting conditions for operation and safety analysis described in the FSAR. The proposed change revises the Cyber Security Plan implementation schedule. The proposed Cyber Milestone 8 schedule change does not involve a significant reduction in a margin of safety because the proposed change does not involve changes to the initial conditions contributing to accident severity or consequences, or reduce response or mitigation capabilities. Because there is no change to these established safety margins as result of this change, the proposed change does not involve a significant reduction in a margin of safety.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, DEF concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission’s regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

The proposed amendment provides a change to the Cyber Security Plan implementation schedule for Milestone 8. The proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(12). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

DUKE ENERGY FLORIDA, LLC

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ATTACHMENT B

ENGINEERING CALCULATIONS SUMMARY

PERMANENTLY DEFUELED ACCIDENT ANALYSIS

1.0 SUMMARY DESCRIPTION

Crystal River Unit 3 (CR-3) has been shutdown since September 26, 2009, when the plant entered the Cycle 16 refueling outage. In the process of creating a construction opening for replacement of steam generators during that outage, a delamination of the outer concrete shell of the containment was discovered. The construction opening and adjacent concrete shell of the containment were repaired during 2010 and 2011. During the repair, fuel was reloaded into the reactor vessel in anticipation of plant restart. During tensioning of the containment prestressing tendons following the concrete repair, delaminations occurred in two other sections of the containment shell. In consideration of performing a second repair of the containment shell, all fuel was removed from the reactor vessel and placed in storage in the CR-3 spent fuel pools on May 28, 2011. Following a comprehensive analysis, DEF, a subsidiary of Duke Energy Corporation, announced on February 5, 2013, that CR-3 would be retired. By letter dated February 20, 2013 (ADAMS Accession No. ML13056A005), DEF, the licensee, notified the NRC of the permanent cessation of operation at CR-3 and that fuel had been permanently removed from the reactor vessel.

Concurrent with the submittal of this License Amendment Request, all fuel stored in the CR-3 spent fuel pools will have been in the pools for at least six and a half years. The calculations described in this enclosure are based on the radioactive inventories and heat generation rates as of September 26, 2013, or are for an earlier date and are conservative for that date.

In the calculations summarized in this enclosure, doses were determined at the Exclusion Area Boundary (EAB) which is defined as the area that extends 4400 feet in a circle around the Reactor Building at CR-3. The site boundary is the line beyond which the land is not owned, leased, or otherwise controlled by Duke Energy Florida, Inc. The EAB radius is equal to the distance between the center of the Reactor Building and the closest point of the site boundary. All other points on the site boundary are further from the center of the Reactor Building. Dose projections in the accident analyses are provided for the EAB. Therefore, dose projections at the site boundary are equal to or less than the dose projections at the EAB.

The accidents evaluated for the CR-3 Cyber Security Milestone 8 schedule extension are listed below by their section number in this enclosure. These sections contain excerpts of the full calculation for the purposes of this submittal, the entire calculations are available upon request.

2.0 Public and Control Room Dose from a Fuel Handling Accident (Calc N13-0001)
3.0 CR-3 Spent Fuel Pool Time to Uncover Fuel Analysis (Calc F13-0003)
4.0 Maximum Cladding and Fuel Temperature for Uncovered Spent Fuel Pool (Calc F13-0002)
5.0 Adiabatic Spent Fuel Bundle Heat-up (Calc F13-0004)
6.0 Dose Rates Due to Spent Fuel Assemblies in the CR3 Spent Fuel Pool Following Drain Down (Calc N13-0002)

In these calculations, dose results are compared to two different standards applied to Emergency Planning. First, the results are compared to the Environmental Protection Agency (EPA) Protective Action Guides (PAGs) to support the exemption from requirements for offsite planning zones. Second, the results are used to establish conformance to the guidance in Nuclear Energy Institute (NEI) 99-01, “Development of Emergency Action Levels for Non

Passive Reactors," Appendix C for Permanently Defueled plants that, in general, the source term and release motive forces are not expected to be sufficient to require declaration of a Site Area Emergency.

2.0 PUBLIC AND CONTROL ROOM DOSE FROM A FUEL HANDLING ACCIDENT - CALCULATION N13-0001

2.1 General Description

The purpose of this calculation is to determine the public dose at the EAB and LPZ, and dose to operators in the Control Room for a Fuel Handling Accident in the spent fuel pools (SFPs) under permanent shutdown conditions. Due to the amount of decay assumed (4 years), the results of this calculation may be applied as of September 26, 2013. The plant last operated until September 26, 2009, when it shut down for a refueling outage and steam generator replacement.

The method employed in this calculation involves use of NRC software program RADTRAD to implement the guidance of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. RADTRAD was used to perform the prior CR-3 Fuel Handling Accident dose calculation which was approved by the NRC in License Amendment 199, "Crystal River Unit 3 - Issuance of Amendment Regarding Alternative Source Term and Control Room Ventilation System (TAC NO. MB0241)," (ADAMS Accession No. ML012430210). The approach used in this calculation for some input parameters, such as χ/Q which has been recalculated according to NRC guidance and decay time which was changed to reflect the last date of plant operation for CR-3 is similar to the previously approved calculation.

Although sufficient time has passed since shutdown for the key dose contributing iodine radionuclides to have decayed to insignificance, the RADTRAD input files were nevertheless set up to include iodine specific inputs (e.g., pool decontamination factor) so that there would be documented proof there is no dose due to iodines.

A separate manual calculation using Excel was done as a check on the RADTRAD results for the EAB and provided predicted doses in agreement with the RADTRAD results.

2.2 Design Inputs

Inputs for RADTRAD

No.	Parameter	Value	Source & Notes
1	Thermal Power	2619 MWt	Reference FSAR 1.2.2 and Reference 2.5.1, 2609 MWt plus 10 MWt (0.4% uncertainty)
2	Number of assemblies in core	177	CR-3 Technical Specifications, Section 4.0 Design Features, Subsection 4.2.1
3	Number of rods per assembly	208	FSAR, Chapter 3, Section 3.17
4	Radial Peaking Factor	1.8	FSAR Table 3-113
5	Release Fraction for Noble Gases / Iodines (.rft file)	0.1 / 0.08	Reference 2.5.2, Table 3 Used most limiting values for noble gases and iodine, which are for Kr-85 and I-131
6	Release Timing (.rft file)	0.1E-05 hours	Reference 2.5.3
7	Compartment Volume for spent fuel area	1 $ft.^3$	Reference 2.5.3 Arbitrary low volume that when combined with flow rate to environment assures all release occurs within 2 hours
8	Compartment flow rate from spent fuel area to environment	300 cfm	Reference 2.5.3 Arbitrary high flow rate value that when combined with low compartment volume assures release of all activity from spent fuel area well within 2 hours

No.	Parameter	Value	Source & Notes
9	Source term fraction for 1 failed assembly	1.02E-02	References: 2.5.3 and 2.5.4 Same as previously approved licensing basis Fuel Handling Accident analysis - 1 entire assembly (208 rods) is assumed to fail; however a mechanical drop analysis (Reference 2.5.4) indicates that 43 of 208 rods would fail. Fraction of core experiencing cladding failure adjusted by radial peaking factor of 1.8. 1 failed assembly/177 assemblies x 1.8 = 1.02E-02.
10	Percent of gap activity released	100%	Reference 2.5.3
11	Pool decontamination factor – Noble Gases	1	Reference 2.5.2, Appendix B
12	Pool decontamination factor (DF) – Halogens	100	Reference 2.5.3: provides for using a DF of 100 and Reference 2.5.2: provides for a DF of 200 when the pool depth over the fuel is at least 23 feet
13	Pool decontamination factor – Alkalis	infinite	Reference 2.5.2, Appendix B, Section 2
14	Composition of airborne halogens – Elemental	57%	Reference 2.5.2, Appendix B, Section 2: Assumed same as with 23 feet of water
15	Composition of airborne halogens – Organic	43%	Reference 2.5.2, Appendix B, Section 2: Assumed same as with 23 feet of water

No.	Parameter	Value	Source & Notes
16	Nuclide Inventory File (.nif file)	Same as FSAR 3.5.3 except for Kr-85 and I-131	Reference 2.5.3 and Reference 2.5.5, Table 1.4.3.2-2
17	Nuclide Inventory File (.nif file) for Kr-85 & I-131	4.20E+02 Ci/MWt 2.902E+04 Ci/MWt	Reference 2.5.6, Table 4-1 Divide Table 4-1 total core Curies by MWt 1.1E6 Ci/2619 MWt = 420 Ci/MWt Kr-85 7.6E7 Ci/2619 MWt = 2.902E+04 Ci/MWt I-131
18	Control Complex Volume	3.649E+05 $ft.^3$	Reference 2.5.3
19	Control Complex Intake Flow	5.7E+03 cfm	Reference 2.5.3
20	Control Room χ/Q	5.14E-03 sec/m^3	Reference 2.5.3
21	EAB χ/Q	1.54E-04 sec/m^3	Reference 2.5.3
22	LPZ χ/Q	2.70E-05 sec/m^3	Reference 2.5.3
23	Time from Shutdown to Accident	35040 hours	Reference 2.5.1 Assumed earliest possible accident time as of September 26, 2013, which is 4 years (y) from time of last shutdown. Input is in hours (h): 4 y x 365 day/y x 24 h/day = 35040 hours
24	Breathing Rate – Offsite	3.5E-04 m^3/s	Reference 2.5.2, Section 4.1.3 Use this value for the entire period due to short duration of release
25	Breathing Rate – Control Room	3.5E-04 m^3/s	Reference 2.5.2, Section 4.2.6
26	Occupancy Factor – Control Room	100% for first 24 hours, 60% from 24 to 96 hours, and 40% for remainder of 30 days.	Reference 2.5.2, Section 4.2.6 Due to short duration of release, the dose will have been accumulated in the first few hours.

2.3 Assumptions

As all used fuel resides in the SFPs, the Fuel Handling Accident occurs in the Auxiliary Building (Reference 2.5.8) and hence, the release path is out the Auxiliary Building vent.

Regulatory Guide 1.183 (Reference 2.5.2) provides for an iodine reduction factor of 200 with at least 23 feet of water above the damaged fuel. This would be the case for fuel in the spent fuel racks which are damaged by a heavy load drop, but for a damaged assembly which lies horizontally across the top of the spent fuel racks, the water depth could be slightly less than 23 feet. Therefore, an iodine removal factor of 100 will be used. This is consistent with the previously approved licensing basis analysis (Reference 2.5.3).

Release pathway filters (HVAC HEPA and carbon) are not considered.

The library of nuclides used by RADTRAD includes those generally recognized as the major dose contributors should an accident occur while a nuclear plant is operating or during the period shortly after shutdown (Reference 2.5.5).

Consistent with the previously approved Fuel Handling Accident (Reference 2.5.3), all 208 rods in an assembly are assumed to fail; however, a mechanical drop analysis (Reference 2.5.4) indicates that 43 rods (approximately 20% of a full assembly) will fail. The difference in dose is proportional to the difference in the number of rods which fail.

2.4 Conclusions/Results

Dose results for primary receptor locations are given here. Dose limits for each receptor are in brackets on the same line.

Control Room:	1.3E-04 rem TEDE	[5 rem TEDE]
Exclusion Area Boundary:	5.9E-05 rem TEDE	[100 mrem TEDE]
Low Population Zone:	1.0E-05 rem TEDE	[100 mrem TEDE]

These results show that the previously approved licensing basis scenario (Reference 2.5.3), as updated in this calculation, generates very low doses. The 5 rem TEDE Control Room dose limit is established in 10 CFR 50.67. The site boundary and low population zone dose limits are established to verify that site boundary doses will be below the threshold for a Site Area Emergency.

2.5 References

2.5.1 Calculation M07-0002, Revision 0, “Heat Balance Uncertainty for CR-3 MUR”

2.5.2 NRC Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”

2.5.3 Calculation N00-0001, Revision 0, “Public and Control Room Dose from a Fuel Handling Accident Using the Alternative Source Term”

2.5.4 AREVA NP Document 32-1244997-001, Fuel Assembly Drop (Slap Down), Sept 2008

2.5.5 NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” including Supplements 1 and 2.

2.5.6 Calculation M98-0108, “CR-3 Core and Gap EQ and EOP MHA Dose Activities”

2.5.7 Calculation N12-0002, Revision 0, “Calculation of Atmospheric Relative Concentrations, χ/Q”

2.5.8 CR-3 to NRC letter, “Crystal River Unit 3 - Certification of Permanent Cessation of Power Operations and that Fuel Has Been Permanently Removed from the Reactor,” dated February 20, 2013. (ADAMS Accession No. ML13056A005)

3.0 CR-3 SPENT FUEL POOL TIME TO UNCOVER FUEL ANALYSIS – CALCULATION F13-0003

3.1 General Description

CR-3 calculation F13-0003, Revision 0, determined how much time it takes to reach certain levels in the spent fuel pools following a loss of active spent fuel pool (SFP) cooling based on an initial water temperature of 110°F.

This calculation provides more realistic, yet still conservative results by using sophisticated tools and methods to credit heat and mass transfer out of the SFPs. Most importantly, the effect of evaporative cooling is taken into account. Additionally, natural convection heat transfer into the SFP walls and floor and into the air above the water surface is taken into account. Mass transfer between the pools through the open gate and convection/conduction through the pool dividing wall are also included. To illustrate the benefit of these more sophisticated methods, an adiabatic computation is also performed in this calculation such that a quantitative comparison can be made.

The response of the SFPs to a loss of cooling based on an initial water temperature of 110°F is evaluated for a range of scenarios. Two variables define each scenario: the condition of the fuel handling building ventilation system and the time after September 26, 2013. Two conditions of the ventilation system are evaluated: operational and inoperable. Two times after September 26, 2013 are evaluated: 0 years and 5 years. Every unique combination of these variables is analyzed with scenarios designated as follows:

Table 3.1: Analyzed Scenarios

Scenario	Ventilation	Time After 9/26/13
inop-0y	Inoperable	0 years
op-0y	Operational	0 years
inop-5y	Inoperable	5 years
op-5y	Operational	5 years

3.2 Methodology for F13-0003

The transient resulting from an unmitigated loss of active cooling to the spent fuel pools is characterized by two phases: a transient heat-up phase and a steady boil-off / evaporation phase. The heat-up phase is a non-linear process wherein the temperature of the water increases asymptotically. This behavior is due to evaporative cooling and heat absorption by the concrete walls and floor and by the air space above the pools. Sophisticated tools must be used to capture these effects. After a time, the heat-up phase either ceases due to the boiling point being reached or becomes linear below the boiling point due to evaporation. In either case, the rate of vaporization from the pools, whether by boiling or subcooled evaporation, reaches a steady state and sophisticated tools are no longer needed. Thus, the heat-up phase is modeled using sophisticated tools that take into account the various heat transfer modes, and the steady boil-off/evaporation phase is modeled using a simplistic approach.

3.3 Assumptions for F13-0003

3.3.1 The GOTHIC model of the FHB documented in Calculation F13-0002 [Ref. 2.1] is used in this analysis per Section 3.1. All assumptions documented in Reference 2.1 that are related to the GOTHIC model are inherited by this analysis. These assumptions were reviewed and determined to be valid for this analysis.

3.4 Spent Fuel Pool HU Rates

The heat-up phase is modeled using the GOTHIC computer program. GOTHIC (Generation of Thermal Hydraulic Information for Containments) is a general purpose thermal-hydraulics software package for the analysis of nuclear power plant containments, confinement buildings, and system components.

Two GOTHIC models of the Fuel Handling Building (FHB) were previously developed in Calculation F13-0002 [Ref. 5.6.1]. One of the models includes normal ventilation (filename: “Fuel Building (ventilation on).GTH”) and the other does not (filename: “Fuel Building (ventilation off).GTH”). These models were reviewed and determined to be valid for this analysis with minor modifications made to suit its objectives. The necessary modifications to each model are discussed below. In the modified models, evaporative cooling, heat and mass transfer between the pools, and heat absorption by the concrete walls and floor and by the air space above the pools are taken into account.

3.5 Results

3.5.1 Ventilation Inoperable, 0 Years After 9/26/13

3.5.1.1 Heat-Up Phase

The SFP water temperature and level computed for this scenario are shown in Figure 3.5-1. The data show that boiling commences at 10.0 days after the start of the transient, and the water level at the onset of boiling is 153.8 ft.

3.5.1.2 Steady Boil-Off Phase

Entering the above results into Eqs. 3-1 through 3-3, with other inputs to the equations taken from Table 3.1, yields:

$$\dot{m} = -Q_{fuel}/h_{fg} = -\left(2.842 \times 10^6\right)/970.3 = -2929\ lb/hr$$

$$\dot{L} = \frac{\dot{m}}{\rho A} = \frac{-2929}{59.81 \cdot (772 + 613 + 12)} = -0.03505\ ft/hr = -0.841\ ft/day$$

$$L(t) = L(t_{boil}) + \dot{L} \cdot (t - t_{boil}) = 153.8 - 0.841 \cdot (t - 10.0)$$

There are two water levels of interest in this analysis: 142.88’ and 134’. The latter is the elevation of the bottom of the gate between the two pits (Design Input 4.5), which is of interest because the two SFPs will be isolated from each other at this point, and the former is 10 feet above the top of the fuel assembly storage cell racks in the west pool (Design Input 4.3), which are higher than the racks in the east pool. Entering these water levels into the equation for L(t) and solving for t yields:

$$t_{142.88'} = \frac{153.8 - 142.88}{0.841} + 10.0 = 23.0\ days$$

$$t_{134'} = \frac{153.8 - 134}{0.841} + 10.0 = 33.5\ days$$

These results are illustrated in Figure 3.5-2 and summarized below.

Table 3.5: Results of Scenario inop-0y

Time to Boil (days from accident)	Avg. Heat-Up Rate* (°F/hr)	Steady-State Boil-Off Rate		Time to El. 142.88’ (days from accident)	Time to El. 134’ (days from accident)
		ft/day	gpm**		
10.0	0.43	0.841	6.1	23.0	33.5

* Based on the total temperature rise (212-110°F) divided by the time to boil (10.0 days). Care should be taken in using this value because instantaneous heat-up rates vary greatly with time.
** gpm = ft/day x (1 day / 1440 minutes) x (area of pool = 772 + 613 + 12 = 1397 ft^2) x (7.48052 gal / ft^3)

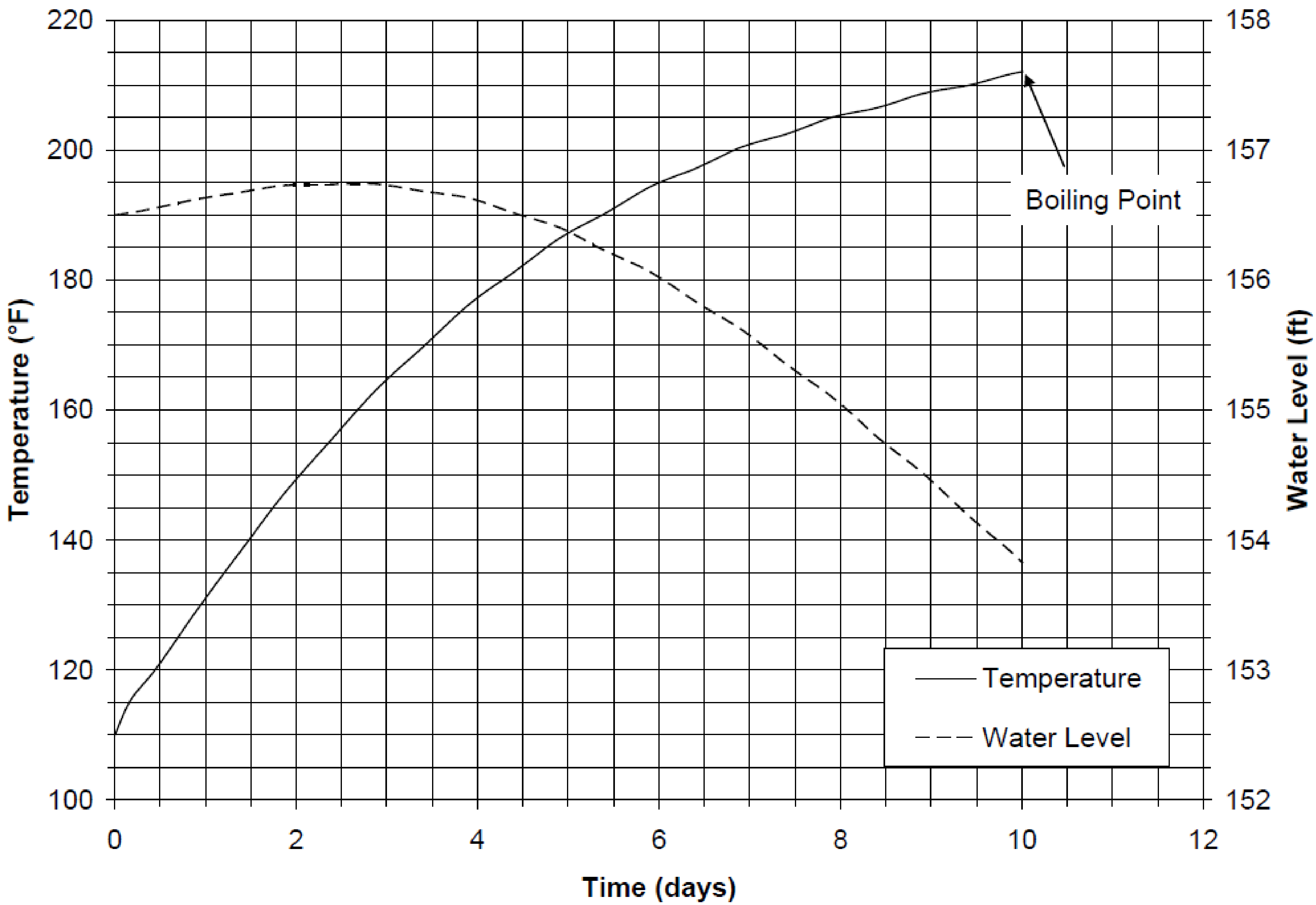

Figure 3.5-1: SFP Temperature and Level for Scenario inop-0y

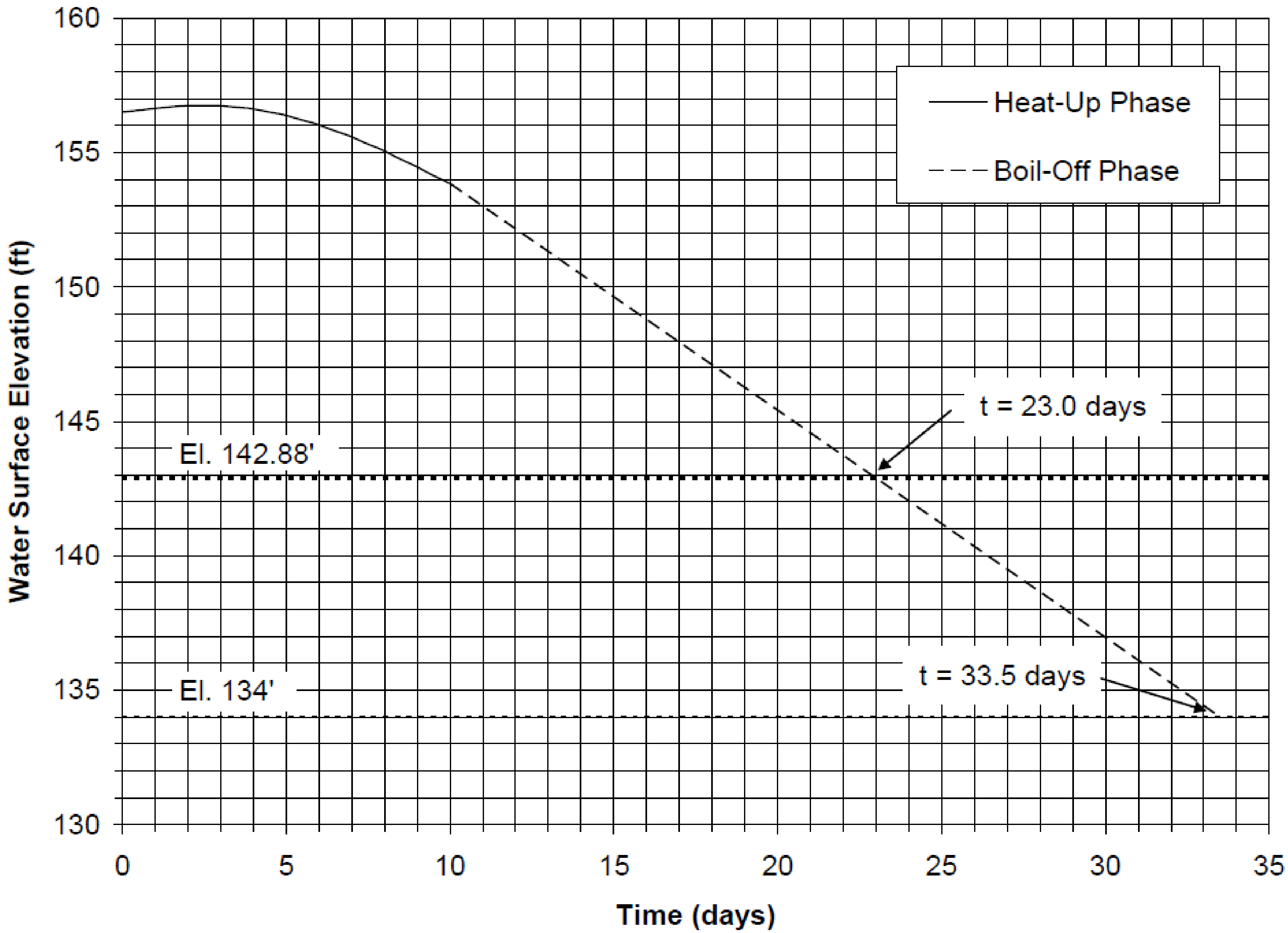

Figure 3.5-2: Extended SFP Water Level for Scenario inop-0y

3.6 Ventilation Operational, 0 Years After 9/26/13

3.6.1 Heat-Up Phase

The SFP water temperature and level computed for this scenario are shown in Figure 3.6-1. The data show that with the ventilation system operating, the spent fuel pools are maintained below the boiling point for a very long period of time (>50 days by extrapolation). Nevertheless, the water level decreases due to evaporation. The evaporation rate, which is shown in Figure 3.6-2, reaches a quasi steady-state condition after approximately 8 days.

3.6.2 Steady Evaporation Phase

The steady water level change rate is computed from the GOTHIC output provided in Attachment 4 using Eq. 3-4 as follows:

$$\dot{L} = \frac{L(t_2) - L(t_1)}{t_2 - t_1} = \frac{150.4 - 152.8}{14.7 - 10.3} = -0.545 \ ft/day$$

where t1 and t2 are selected based on Figure 3.6-2 to capture several days of data in the quasi steady- state regime. From Figure 3.6-1, water temperature at t2 is 171°F. Using the above water level change rate in Eq. 3-5 yields:

$$L(t) = L(t_2) + \dot{L} \cdot (t - t_2) = 149.7 - 0.708 \cdot (t - 12.9)$$

As discussed earlier, there are two water levels of interest in this analysis: 142.88’ and 134’. The latter is the elevation of the bottom of the gate between the two pits (Design Input 4.5), which is of interest because the two SFPs will be isolated from each other at this point, and the former is 10 feet above the top of the fuel assembly storage cell racks in the west pool (Design Input 4.3). Entering these water levels into the equation for L(t) and solving for t yields:

$$t_{142.88'} = \frac{150.4 - 142.88}{0.545} + 14.7 = 28.5 \ days$$

$$t_{134'} = \frac{150.4 - 134}{0.545} + 14.7 = 44.8 \ days$$

These results are illustrated in Figure 3.6-3 and summarized below. Note that without makeup, the water level will drop to elevation 134’ before boiling occurs.

Table 3.6: Results of Scenario op-0y

Time to Boil (days from accident)	Avg. Heat-Up Rate* (°F/hr)	Steady-State Evaporation Rate		Time to El. 142.88' (days from accident)	Time to El. 134' (days from accident)
		ft/day	gpm***		
N/A**	0.20	0.708	5.1	22.5	35.1

*Based on the total temperature rise (171-110°F) divided by the total elapsed time in GOTHIC (12.9 days). Care should be taken in using this value because instantaneous heat-up rates vary greatly with time.

**Evaporative cooling in this scenario is such that the water level drops to elevation 134' prior to boiling.

***gpm = ft/day x (1 day/1440 minutes) x (area of pool = 772 + 613 + 12 = 1397 ft^2) x (7.48052 gal/ ft^3)

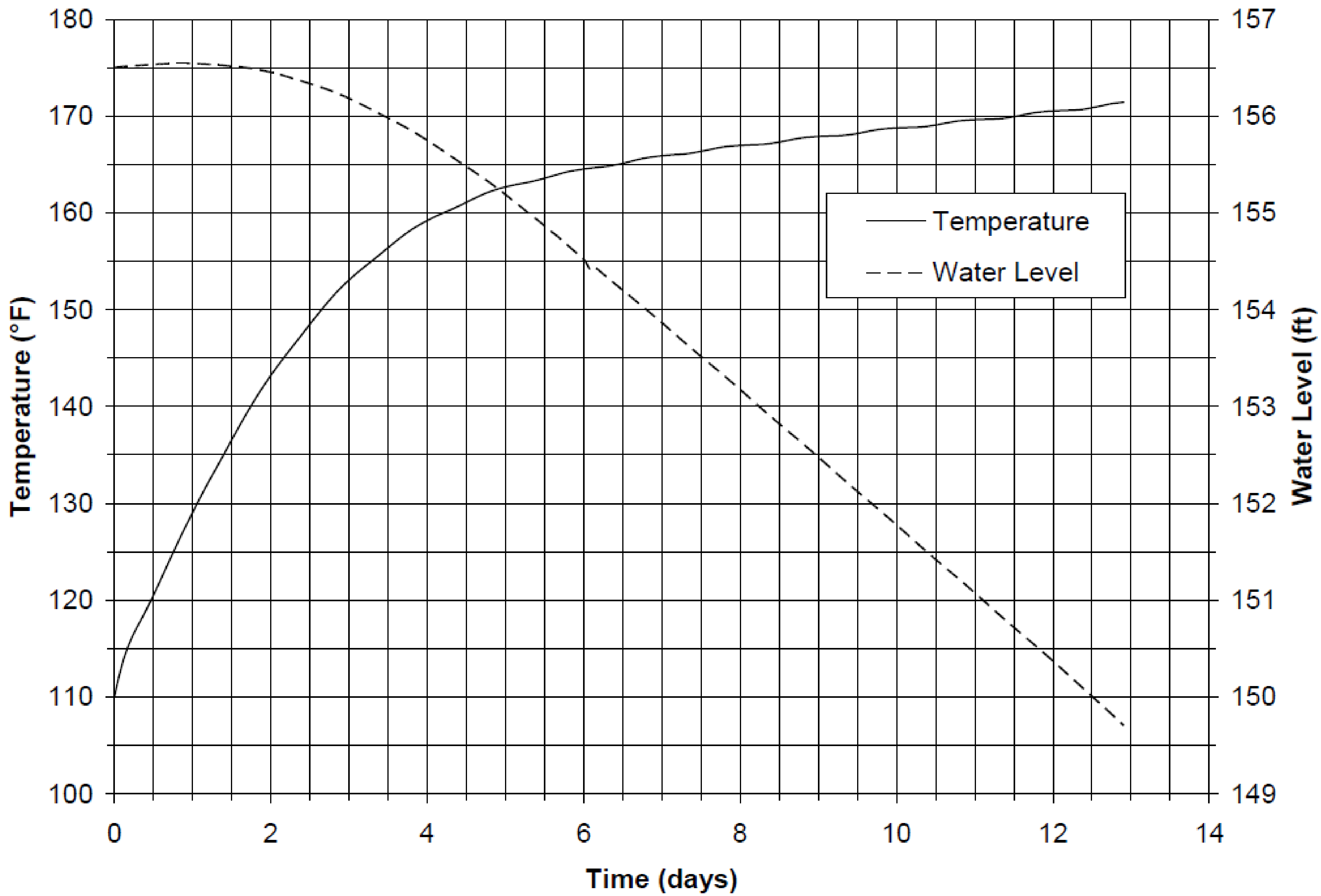

Figure 3.6-1: Extended SFP Water Level for Scenario op-0y

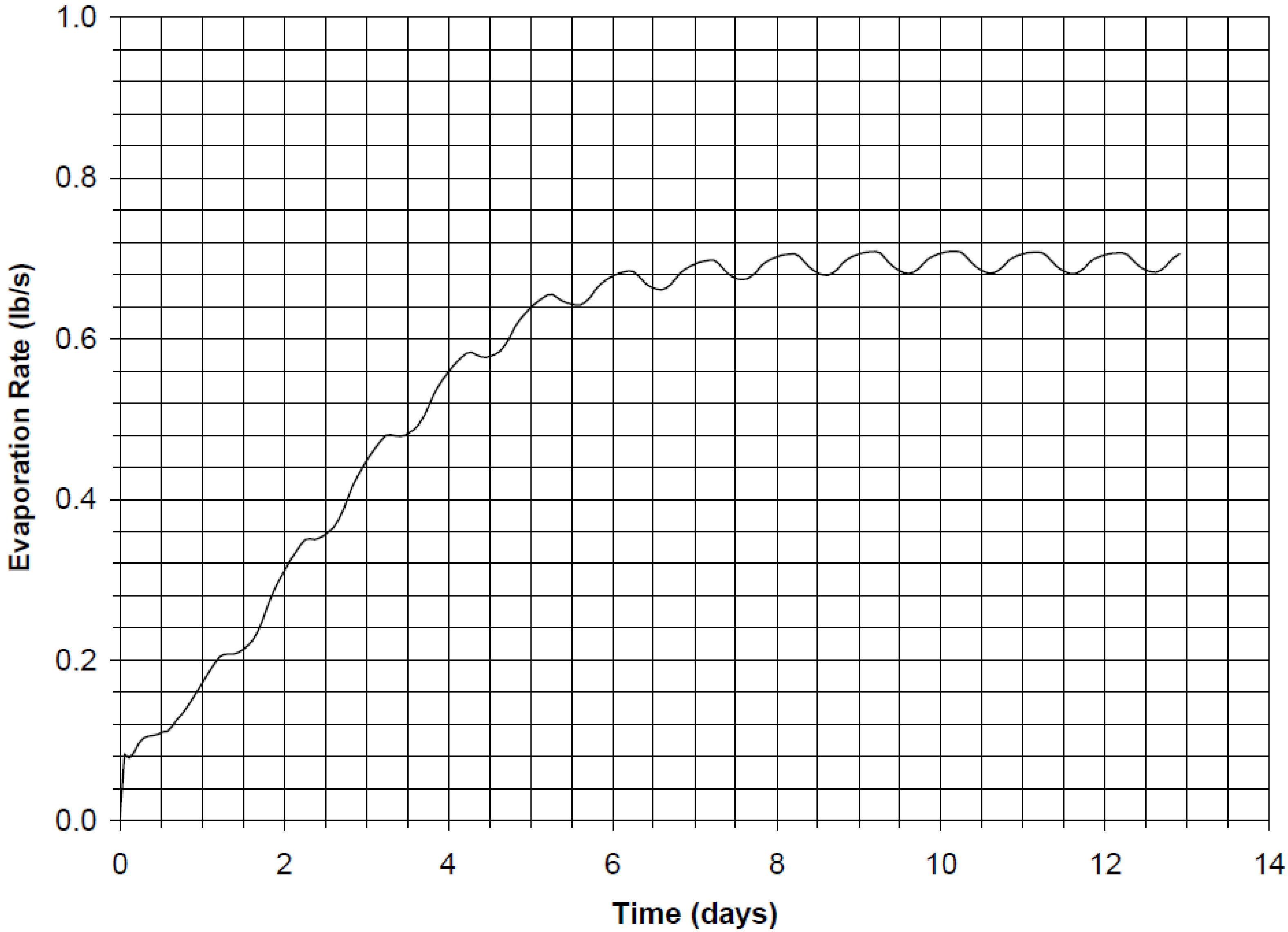

Figure 3.6-2: Extended SFP Water Level for Scenario op-0y

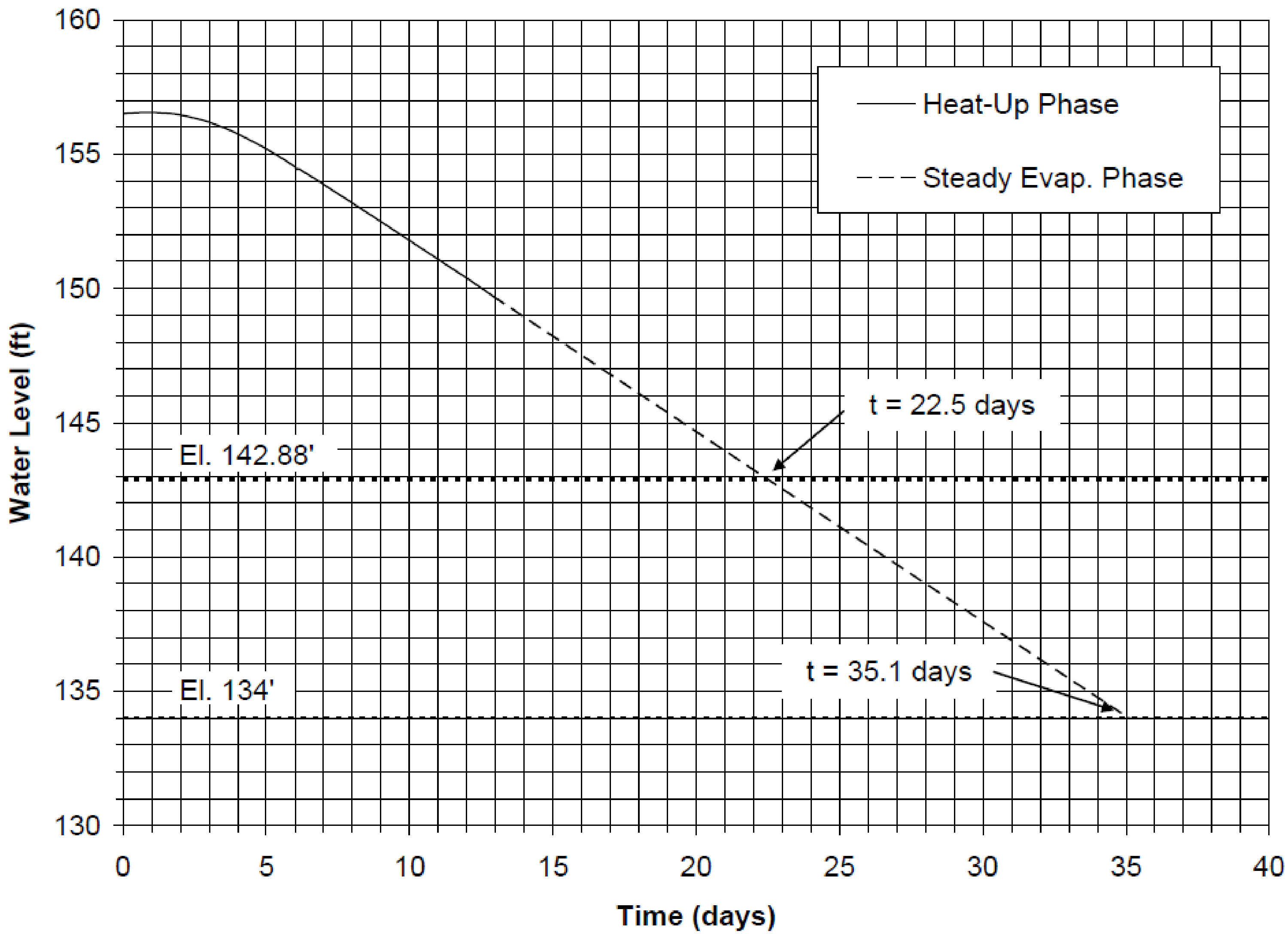

Figure 3.6-3: Extended SFP Water Level for Scenario op-0y

3.7 Ventilation Inoperable, 5 Years After 9/26/13

3.7.1 Heat-Up Phase

The SFP water temperature and level computed for this scenario are shown in Figure 3.7-1. The data show that the boiling point is nearly reached within 18.0 days (the final temperature at the end of the GOTHIC run is 207°F). The GOTHIC run was terminated at this point to avoid numerical instabilities that occur when the water level approaches one of the grid lines in the z-direction. The water level after 18.0 days is 150.5 ft.

3.7.2 Steady Boil-Off Phase

Although the GOTHIC run was terminated prior to the onset of boiling at 18.0 days, the margin to boiling at the end of the run is relatively small (5°F). Therefore, it is conservatively assumed that boiling commences at 18.0 days, and the SFP conditions at that time are entered into Eqs. 3-1 through 3-3, with other inputs to the equations taken from Table 3.1, to yield the following results:

$$\dot{m} = -Q_{fuel}/h_{fg} = -\left(2.233\times10^{6}\right)/970.3 = -2301\ lb/hr$$

$$\dot{L} = \frac{\dot{m}}{\rho A} = \frac{-2301}{59.81\cdot(772+613+12)} = -0.02754\ ft/hr = -0.661\ ft/day$$

$$L(t) = L(t_{boil}) + \dot{L}\cdot(t - t_{boil}) = 150.5 - 0.661\cdot(t - 18.0)$$

As discussed earlier, there are two water levels of interest in this analysis: 142.88’ and 134’. The latter is the elevation of the bottom of the gate between the two pits (Design Input 4.5), which is of interest because the two SFPs will be isolated from each other at this point, and the former is 10 feet above the top of the fuel assembly storage cell racks in the west pool (Design Input 4.3). Entering these water levels into the equation for L(t) and solving for t yields:

$$t_{142.88'} = \frac{150.5-142.88}{0.661} + 18.0 = 29.5\ days$$

$$t_{134'} = \frac{150.5-134}{0.661} + 18.0 = 43.0\ days$$

These results are illustrated in Figure 3.7-2 and summarized below.

Table 3.7: Results of Scenario inop-5y

Time to Boil (days from accident)	Avg. Heat- Up Rate* (°F/hr)	Steady-State Boil-Off Rate		Time to El. 142.88’ (days from accident)	Time to El. 134’ (days from accident)
		ft/day	gpm**		
18.0	0.22	0.661	4.8	29.5	43.0

* Based on the total temperature rise (207-110°F) divided by the total elapsed time in GOTHIC (18.0 days). Care should be taken in using this value because instantaneous heat-up rates vary greatly with time.
** gpm = ft/day x (1 day / 1440 minutes) x (area of pool = 772 + 613 + 12 = 1397 ft^2) x (7.48052 gal / ft^3)

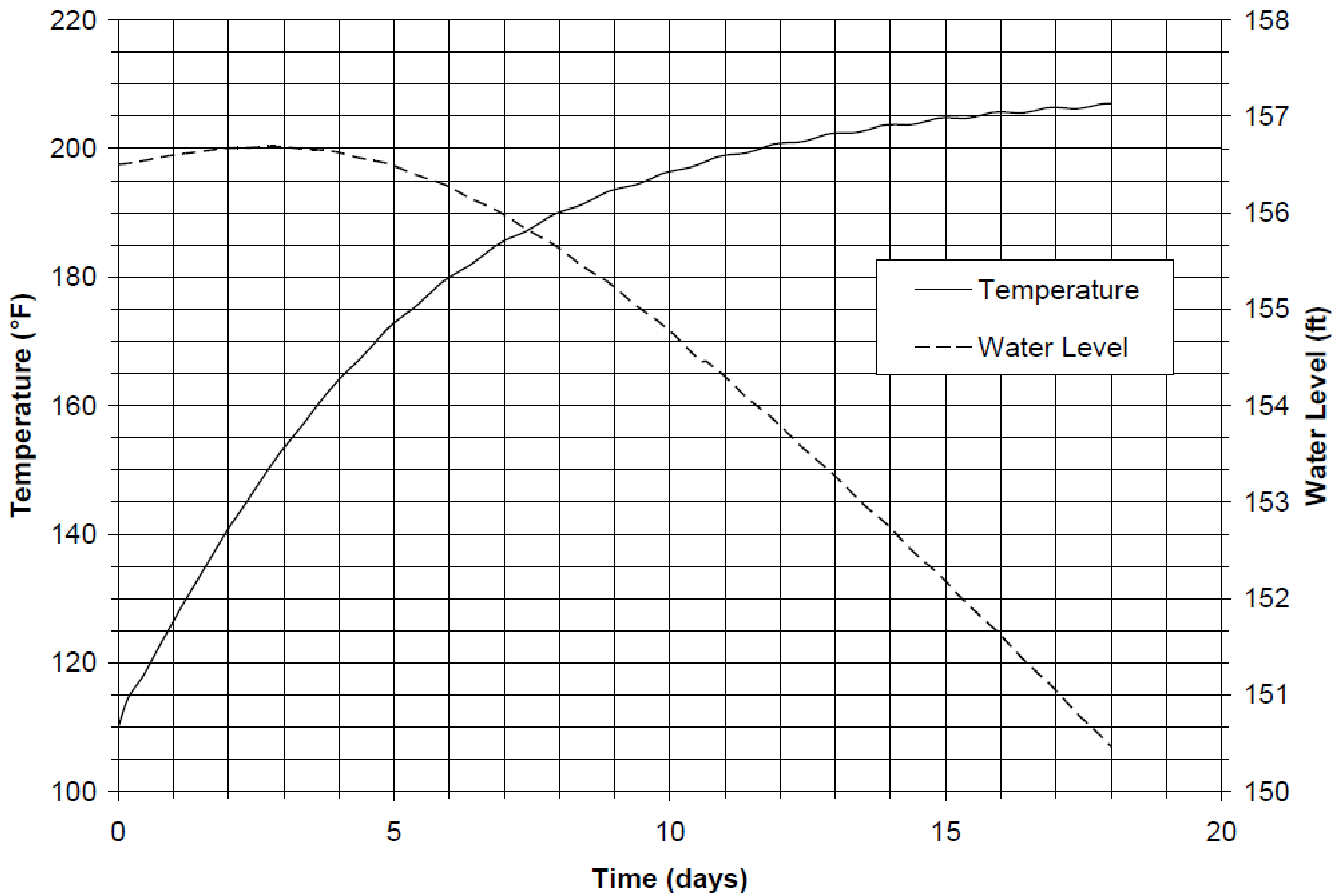

Figure 3.7-1: SFP Temperature and Level for Scenario inop-5y

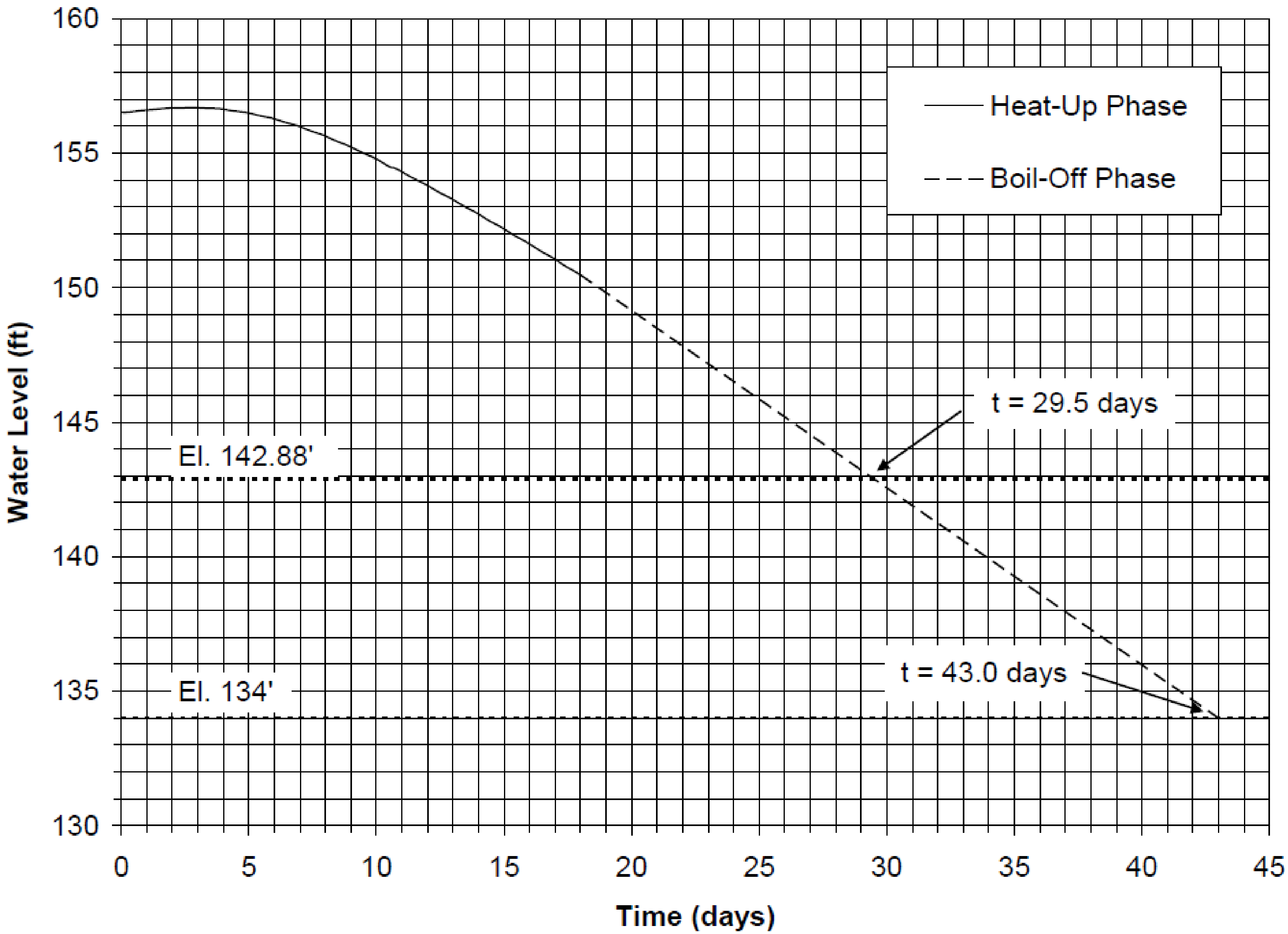

Figure 3.7-2: Extended SFP Water Level for Scenario inop-5y

3.8 Ventilation Operational, 5 Years After 9/26/13

3.8.1 Heat-Up Phase

The SFP water temperature and level computed for this scenario are shown in Figure 3.8-1. The data show that with the ventilation system operating, the spent fuel pools are maintained below the boiling point for a very long period of time (>80 days by extrapolation). Nevertheless, the water level decreases due to evaporation. The evaporation rate, which is shown in Figure 3.8-2, reaches a quasi steady-state condition after approximately 10 days.

3.8.2 Steady Evaporation Phase

The steady water level change rate is computed from the GOTHIC output provided in Attachment 4 using Eq. 3-4 as follows:

$$\dot{L} = \frac{L(t_2) - L(t_1)}{t_2 - t_1} = \frac{150.4 - 152.8}{14.7 - 10.3} = -0.545\ ft/day$$

where t1 and t2 are selected based on Figure 3.8-2 to capture several days of data in the quasi steady- state regime. From Figure 3.8-1, water temperature at t2 is 162°F. Using the above water level change rate in Eq. 3-3 yields:

$$L(t) = L(t_2) + \dot{L} \cdot (t - t_2) = 150.4 - 0.545 \cdot (t - 14.7)$$

As discussed earlier, there are two water levels of interest in this analysis: 142.88’ and 134’. The latter is the elevation of the bottom of the gate between the two pits (Design Input 4.5), which is of interest because the two SFPs will be isolated from each other at this point, and the former is 10 feet above the top of the fuel assembly storage cell racks in the west pool (Design Input 4.3). Entering these water levels into the equation for L(t) and solving for t yields:

$$t_{142.88'} = \frac{150.4 - 142.88}{0.545} + 14.7 = 28.5\ days$$

$$t_{134'} = \frac{150.4 - 134}{0.545} + 14.7 = 44.8\ days$$

These results are illustrated in Figure 3.8-3 and summarized below. Note that without makeup, the water level will drop to elevation 134’ before boiling occurs.

Table 3.8: Results of Scenario op-5y

Time to Boil (days from accident)	Avg. Heat-Up Rate* (°F/hr)	Steady-State Evaporation Rate		Time to El. 142.88’ (days from accident)	Time to El. 134’ (days from accident)
		ft/day	gpm***		
N/A**	0.15	0.545	4.0	28.5	44.8

* Based on the total temperature rise (162-110°F) divided by the total elapsed time in GOTHIC (14.7 days). Care should be taken in using this value because instantaneous heat-up rates vary greatly with time.
** Evaporative cooling in this scenario is such that the water level drops to elevation 134’ prior to boiling.
*** gpm = ft/day x (1 day / 1440 minutes) x (area of pool = 772 + 613 + 12 = 1397 ft^2) x (7.48052 gal / ft^3)

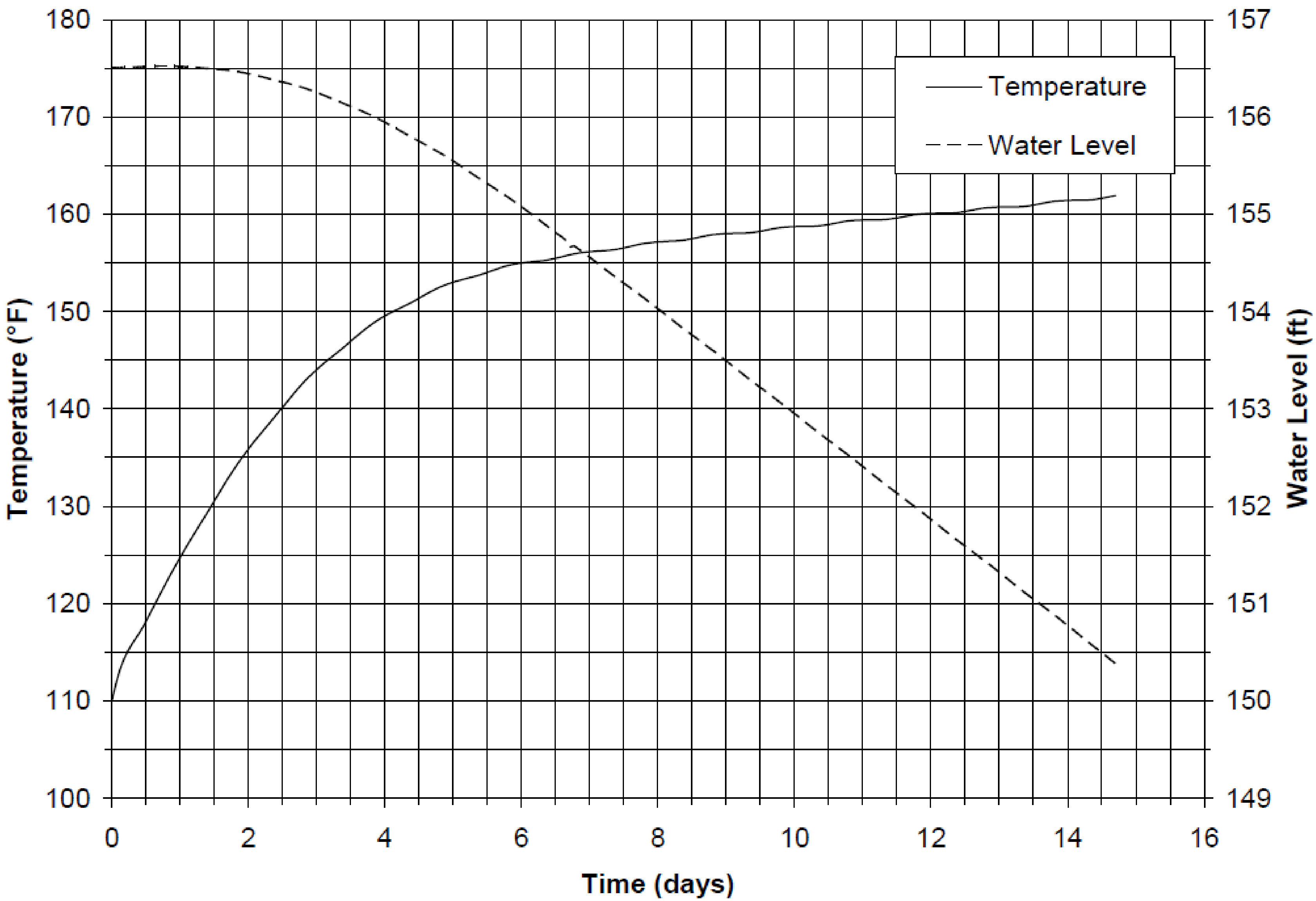

Figure 3.8-1: SFP Temperature and Level for Scenario op-5y

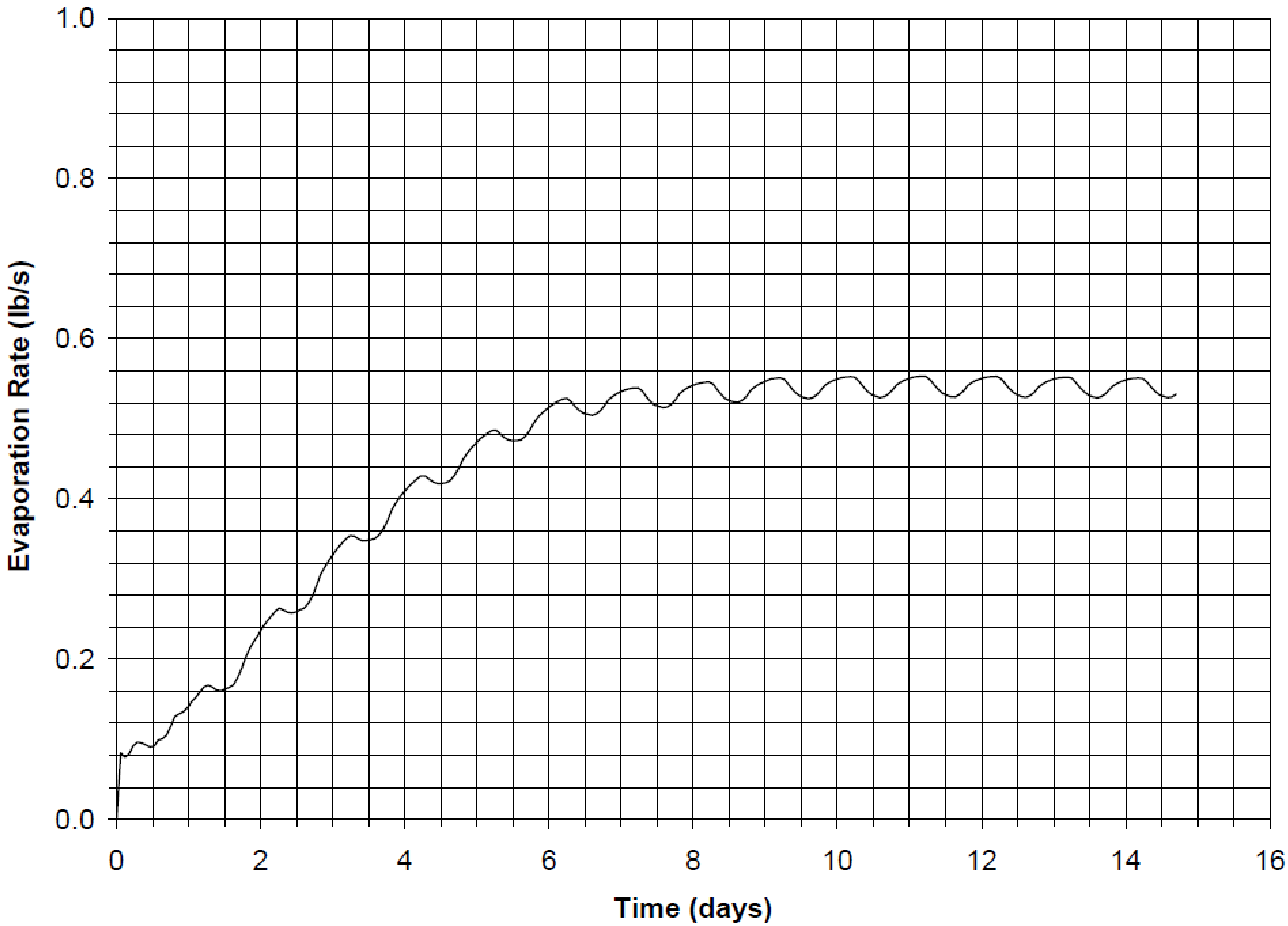

Figure 3.8-2: SFP Evaporation Rate for Scenario op-5y

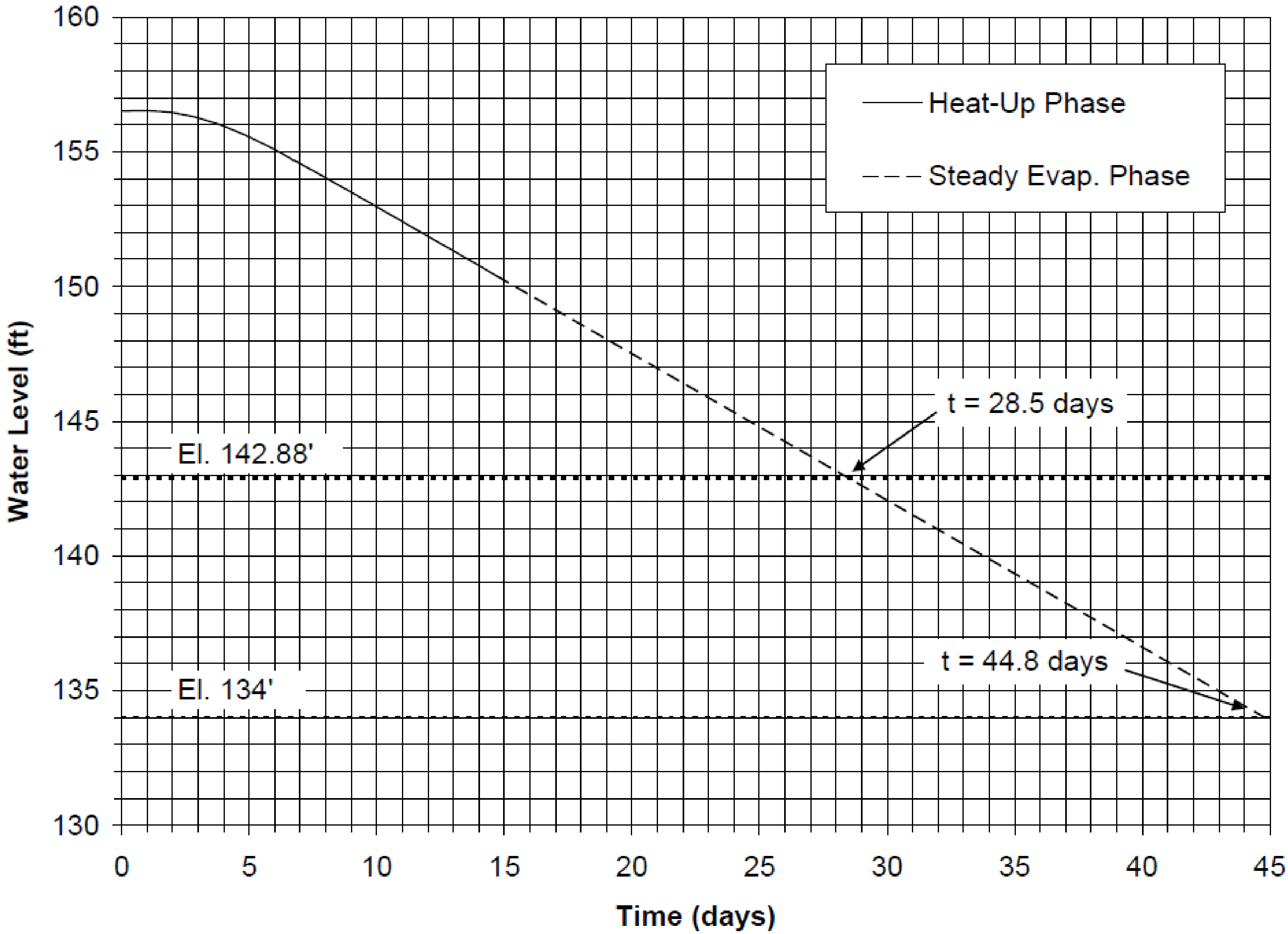

Figure 3.8-3: Extended SFP Water Level for Scenario op-5y

3.9 CONCLUSIONS

The length of time after an unmitigated loss of active cooling to the SFPs at which certain water levels are reached has been determined based on an initial water temperature of 110°F for a range of scenarios. Compensatory actions by the operator, including the addition of makeup water to maintain a stable water level, were not evaluated. The accident is characterized by two distinct phases: a transient heat-up phase and a steady boil-off/ evaporation phase. The heat-up phase is a non-linear process wherein the temperature of the water increases asymptotically. This behavior is due to evaporative cooling and heat absorption by the concrete walls and floor and by the air space above the pools. Sophisticated tools (GOTHIC) were used to capture these effects. After a time, the heat-up phase either ceases due to the boiling point being reached or becomes linear below the boiling point. In either case, the rate of vaporization from the pools, whether by boiling or subcooled evaporation, reaches a steady state and sophisticated tools are no longer needed. Thus, the steady boil-off/evaporation phase was modeled using a simplistic approach. Table 3.9 below provides a summary of the results of this analysis.

Table 3.9: Summary of Results

Scenario	Time to Boil (days from accident)	Avg. Heat-Up Rate* (°F/hr)	Steady-State Boil-Off / Evaporation Rate		Time to El. 142.88' (days from accident)	Time to El. 134' (days from accident)
			ft/day	gpm***		
inop-0y	10.0	0.43	0.841	6.1	23.0	33.5
op-0y	N/A**	0.20	0.708	5.1	22.5	35.1
inop-5y	18.0	0.22	0.661	4.8	29.5	43.0
op-5y	N/A**	0.15	0.545	4.0	28.5	44.8

* Care should be taken in using these values because instantaneous heat-up rates vary greatly with time.
** Evaporative cooling in this scenario is such that the water level drops to elevation 134' prior to boiling.
*** gpm = ft/day x (1 day / 1440 minutes) x (area of pool = 772 + 613 + 12 = 1397 ft^2) x (7.48052 gal / ft^3)

Note that the steady-state boil-off/evaporation rates are relatively low. Based on this, only minimal makeup flow would be required to maintain a stable water level indefinitely. If the makeup is at the same temperature as the SFP, then the required makeup flow rates are equal to the boil-off/evaporation rates given in the table. If the makeup is colder than the SFP, then the required makeup flow rates are less. This is in part due to the fact that cold makeup water will expand slightly when it mixes with and reaches the temperature of the SFP. Additionally, the makeup water will cool the SFP slightly, thereby reducing the steady-state boil-off/ evaporation rate. Neglecting the second effect, the required makeup flow rate can be computed by multiplying the boil-off/evaporation rate by the ratio of SFP density to makeup water density. This is illustrated in the table below, which is based on 80°F makeup water (ρ = 62.22 lb/ft^3).

3.10 References

3.10.1 Calculation No. F13-0002, "Maximum Cladding and Fuel Temperature Analysis for Uncovered Spent Fuel Pool," Rev. 0.

3.10.2 FSAR Rev. 31.3, Section 9.3.2.7, "Spent Fuel Cooling System Operational Limits."

3.10.3 Calculation No. F97-0014, "CR-3 Spent Fuel Pool Temperature Rise From Fuel in the Pool", Rev. 8.

3.10.4 Computer Programs and User Manuals

3.10.4.1 GOTHIC Version 8.0(QA), Numerical Applications Inc., S&L Program No. 03.7.759-8.0.

3.10.4.2 NAI 8907-02, "GOTHIC Thermal Hydraulic Analysis Package User Manual," Rev. 20.

3.10.5 American Society of Mechanical Engineers, *ASME Steam Tables*, 5th Edition, 1992

3.10.6 Calculation No. F13-0001, "Radiation Source Term and Heat Generation Rate in the CR3 Spent Fuel Pool Following Drain Down," Revision 1.

3.10.7 Procedure No. OP-700C, "120/240/480 Volt AC Distribution Panels," Revision 117.

3.10.8 Drawing No. SC-421-141, "Auxiliary Building North Spent Fuel Pit Concrete Outline Plans & Sections," Rev. 14.

3.10.9 RMS-NGG Record Number 5222537 (Quality Record that contains Spent Fuel Pool map). Included as Attachment 7.

3.10.10 Drawing No. 80E1502, "Pool A Fuel Storage Rack Installation," Revision 3.

4.0 MAXIMUM CLADDING and FUEL TEMPERATURE for UNCOVERED SPENT FUEL POOL – CALCULATION F13-0002

4.1 General Description

This analysis is provided to compare the conditions for the irradiated fuel assemblies stored in the CR-3 fuel pools to the first criterion proposed in SECY-99-168, "Improving Decommissioning Regulations for Nuclear Power Plants," (Reference 4.9.1) applicable to offsite emergency response for the unit in the decommissioning process. That criterion considers whether the spent fuel decay heat is sufficiently low so that air cooling is adequate to maintain the clad temperature below the point of self-sustained zirconium oxidation. If the fuel cladding is predicted to remain below the temperature for the onset of rapid oxidation (a zirconium fire), then offsite emergency preplanning involving the plant is not necessary. Temperatures in excess of 800^{o}C are required for self-sustained zirconium oxidation. Cladding temperatures below 565^{o}C do not produce cladding swelling.

The analysis of this beyond design basis event was performed applying two major steps. First, maximum steady-state long term air temperatures throughout the Fuel Handling Floor (FHF) elevation of the Auxiliary Building, including the space below the storage racks, were determined following a loss of water inventory from the SFP. The maximum air temperature below the storage racks was then used in the second step, the calculation of the maximum temperature of the zirconium cladding.

4.2 <u>Methodology</u>

Maximum quasi steady-state long term FHF air temperatures are determined using the GOTHIC computer program (Reference 4.9.2). Three control volumes are included in the GOTHIC model. Two of these control volumes represent the FHF elevation and one of the control volumes represents the outside environment. Both FHF control volumes are subdivided, meaning that dimensions in the x, y, and z directions are identified and a mesh grid is overlaid to subdivide the control volume into sub-volumes. The main advantage of using subdivided volumes is that spatial temperature variations and flow patterns are captured. The second advantage of using subdivided volumes is that GOTHIC automatically calculates the pertinent parameters (volume, hydraulic diameter, flow area, etc.) using the dimensions supplied by the user.

The first control volume (CV1) represents both SFPs and the air above them extending to the ceiling above the FHF. This control volume is wholly contained within CV2, which represents the entire FHF (see below). The reason for modeling the pools and the air above them separately in CV1 is so that a fine mesh grid can be applied to this region and a coarse mesh grid can be applied to the rest of the building. The finer mesh grid provides a higher degree of spatial resolution in the area of most interest, and the coarser mesh grid reduces run-time.

The CV1 control volume is modeled to account for a downcomer region of 1 ft. for air circulation which accounts for the space between the fuel racks and the fuel pool walls and the fact that. in general, the outer fuel rack cells are empty. This downcomer region supplies air to the area under the fuel racks to replace air that rises through the fuel elements and racks as it is heated.

The general arrangement of the FHF elevation and the fuel pools is shown in Figure 4.2.

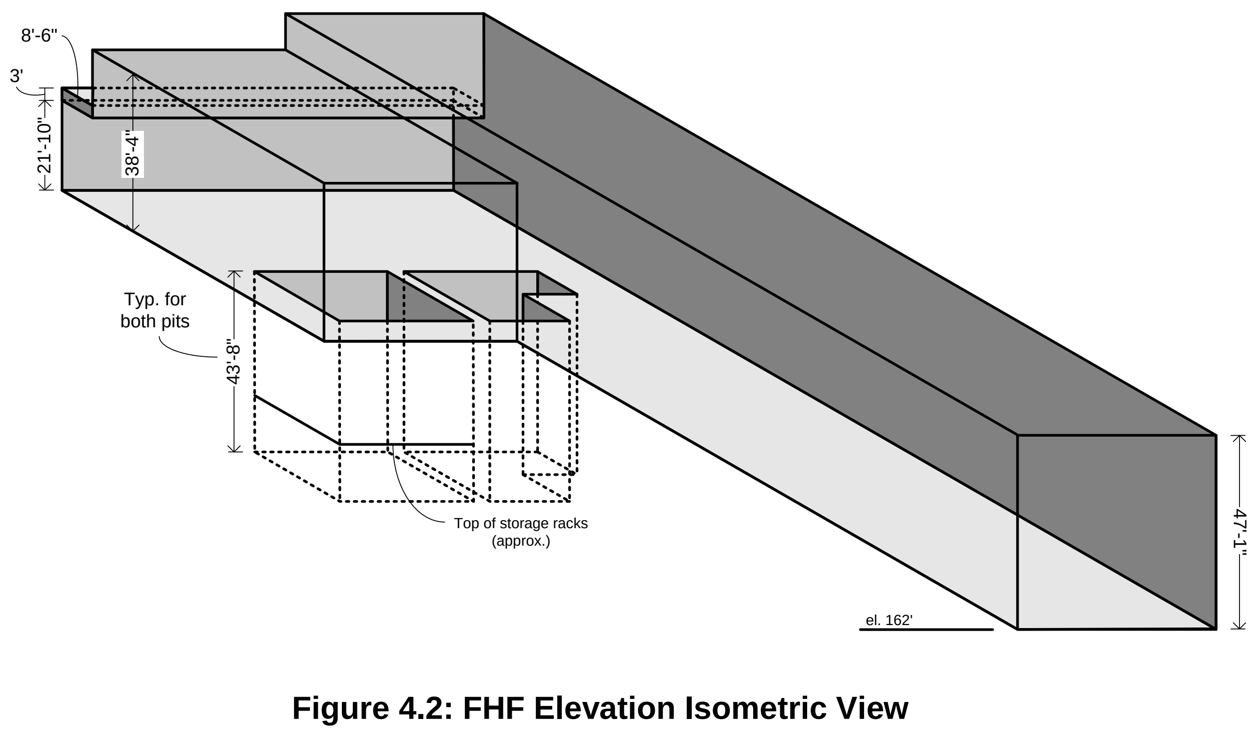

Figure 4.2: FHF Elevation Isometric View

4.3 Inputs

Heat Sinks

The heat sinks included in the model are the FHF aluminum siding walls, the roof, the walls and floors of each spent fuel pit (excluding the dividing wall), a portion of the 162 ft. elevation floor, the Reactor Building wall (west side), and the Control Complex wall (north side). For conservatism, the surface area of the corrugated aluminum siding is computed as though it is a flat plate. That is, no credit is taken for the additional surface area due to the corrugations.

The outer surfaces of the spent fuel pit walls and floors, the general floor area, the Reactor Building wall, and the Control Complex wall, all of which are exposed to enclosed areas of the plant, are taken to be adiabatic. This is conservative since, in reality, other areas of the plant will be cooler than the FHF elevation in the postulated scenario. The thicknesses of walls are taken from plant drawings.

Solar Radiation

The combined surface heat flux from the three modes of solar irradiance (direct, sky shine, and ground reflected solar radiation) are computed for the aluminum siding and the roof using the UHSSIM computer program (Reference 4.9.3), a Sargent & Lundy (S&L) proprietary software package that follows the American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) methodology for calculating solar irradiance. Solar heat flux is determined separately for each heat sink exposed to solar radiation based on its orientation. Shading from the containment building and other structures is conservatively ignored.

Initial Conditions

The initial temperature of the FHF and heat sinks is set to the maximum normal Auxiliary Building temperature (122°F per Reference 4.9.4, Section 2.0), with the exception that concrete heat sinks are initialized at a temperature close to the final average building temperature such that quasi steady-state is reached sooner. Concrete heat sinks provide significant thermal inertia and initializing them at a low temperature would require that the model be run for several more days to reach quasi steady-state. The initial humidity of the FHF elevation is set to 10%.

The SFPs are modeled as initially dry (no liquid water).

Outside air temperatures are calculated below using the methodology provided in Chapter 14 of ASHRAE 2009 (Reference 4.9.5). Temperature calculations are for August, the hottest month. The results are given below in Table 4.3.

Table 4.3: Outside Air Temperature

Hour of Day	Outside Air Temperature (°F)	Hour of Day	Outside Air Temperature (°F)
0	87.4	13	98.3
1	86.5	14	99.0
2	85.9	15	99.0
3	85.5	16	98.1

Hour of Day	Outside Air Temperature (^oF)	Hour of Day	Outside Air Temperature (^oF)
4	85.1	17	97.0
5	84.8	18	95.6
6	85.1	19	93.5
7	86.1	20	91.9
8	88.5	21	90.6
9	91.2	22	89.3
10	93.6	23	88.4
11	95.7	24	87.4
12	97.2		

The above daily temperature cycle is applied to the outside air (CV3) by means of the "flow through" method, which involves inundating the control volume with air set to the temperatures given above, atmospheric pressure (14.7 psia), and 20% relative humidity.

<u>Maximum Cladding Surface Temperature</u>

The maximum steady-state temperature of the spent fuel zirconium cladding is calculated using the computer program COBRA-SFS (Reference 4.9.6). COBRA models the heat transfer inside of the single worst case assembly/storage cell. From the results of the GOTHIC analysis, SFP B has a higher temperature. Therefore, the geometry for SFP B is used in this analysis. The storage cell width and length of the storage cell cross section are set to 8.906 inches which are the smallest dimensions of the two rack designs. It is conservative to minimize the total cross sectional area of the storage cell because the flow rate of air through the storage cell is reduced.

4.4 Calculation

Two runs of the GOTHIC model were performed. One run models normal ventilation and the second run credits no ventilation. Both runs proceed until quasi steady-state conditions are achieved. This description only discusses the more conservative run with no ventilation.

The following sources supply heat to CV2:

- Spent fuel decay heat

  The spent fuel decay heat loads in Pools A and B are evenly distributed across the corresponding pool. No credit is taken for the lessening of the decay heat load with time. The predicted spent fuel decay heat load, as of September 26, 2013, is 1.342×10^6 Btu/hr in Pool A and 1.500×10^6 in Pool B.

- Electrical equipment heat loads

  The electrical equipment heat loads are positioned in the model in accordance with their approximate physical location in the fuel handling building.

No Ventilation System Operating

Ventilation is not credited by de-activating the volumetric fans in GOTHIC which model the flows through the multiple ventilation supply and return registers in the fuel handling building. It should be noted that de-activating a volumetric fan in GOTHIC effectively closes the flow path in which the fan (register) is located. Thus, natural circulation that may in reality develop through the ventilation system is conservatively ignored. To further limit natural circulation, one of the two staircase flow paths to the lower elevations of the Auxiliary Building is closed in the run without ventilation. In reality, neither of the staircases can be hydraulically isolated. However, leaving both staircases open in the model would allow GOTHIC to compute natural circulation between the FHF and the floor below, a condition which is likely to occur in reality, but which cannot be credited analytically without extending the scope of the model to include the lower elevation so that a conservative value for its temperature is known. Closing only one of the staircase flow paths allows air to escape the building as it expands (heats up) in the daytime, thereby reducing the building's heat capacity and producing higher temperatures. The staircase furthest from the spent fuel pits (the South staircase) was selected to be the staircase that is kept open in this run.

4.5 Assumptions

4.5.1 The heat generation rate is assumed to be distributed uniformly along the heated length of the rod (143 inches per Input 4.2). It is conservative to assume a uniform heat generation rate compared to a sine distribution because the heat generation will be greater at the top of the fuel rod for the uniform heat generation rate. This will maximize the temperature of the cladding at the top of the hottest rods. This assumption applies to the COBRA model.

4.5.2 The bulk pressure on the FHF is assumed to be 14.7 psia. This is appropriate because Crystal River is at sea level and the FHF elevation is not air tight.

4.5.3 For the purposes of calculating the mass flux of air in the storage cell, the temperature of air in the storage cell is assumed to vary linearly. This approximation is appropriate because the heat generation rate in the rod is modeled as being uniform, and because the enthalpy of air is linear through the range of temperatures in this analysis.

4.5.4 The heat transfer to the downcomer is assumed to be 50% of the heat generated by the assemblies in the storage cells on the edge of the racks. This heat transfer is used to calculate the temperature increase of the air going through the downcomer. This assumption is conservative because most of the heat will be removed by the air going through the storage cells of those assemblies, as opposed to being transferred through the stainless steel sides of the storage cells and into the downcomer.

4.5.5 It is assumed that the fuel handling crane (FHCR) is not being operated at any time during the event. Thus, heat loads associated with crane equipment, including the disconnect switch (FHCR-5), are not included in the GOTHIC model.

4.6 Maximum Air Temperature Below Storage Racks

No Ventilation System Operating

Figure 4.6 shows the range of air temperatures computed by GOTHIC in the run without ventilation for the space below the storage racks in both spent fuel pits. The maximum temperature is 328°F (164°C).

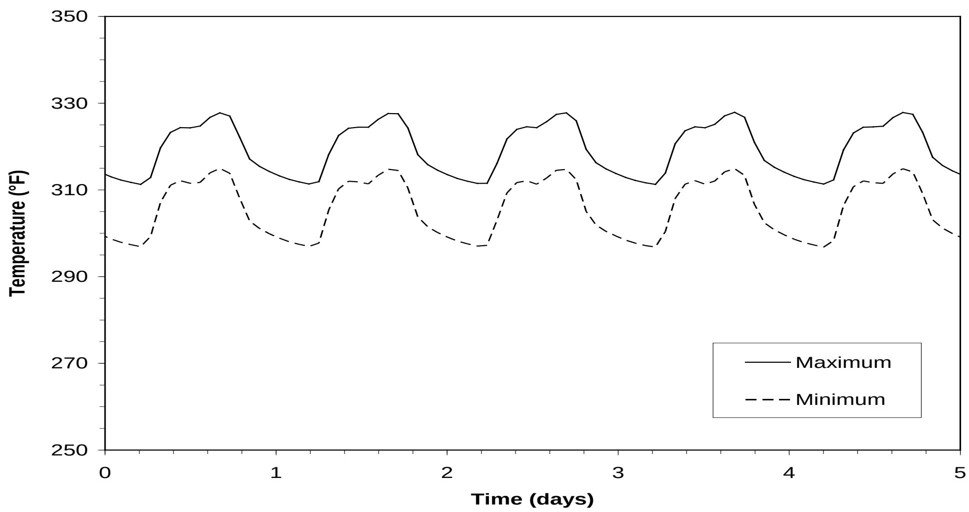

Figure 4.6: Range of Air Temperatures below Storage Racks (No Ventilation)

4.7 Maximum Cladding Surface Temperature

Downcomer Temperature Rise

Based on the results of a determination of heating in the downcomer, the temperature increase for air in the downcomer is 16°F (9°C). The air properties used in that analysis are for the no ventilation case. Therefore, the temperature at the inlet of the storage cell used for the COBRA model for the no ventilation case is 344°F (= 328°F from above + 16°F, or 173°C).

The results of the COBRA analysis for the case when there is no ventilation show that the maximum cladding temperature is 1016°F (547°C). The average air exit temperature is 1004°F (540°C).

The results discussed above are summarized below.

Case	GOTHIC Results for Maximum Temperature Below Storage Racks	Maximum Cladding Temperature
No Ventilation	328°F / 164°C	1016°F / 547°C

4.8 Conclusions

The results of this analysis show that the surface temperature of the cladding in the SFPs will not exceed the failure temperature for zirconium following a total loss of water from the pools, as of September 26, 2013, due to adequate natural circulation and heat rejection to the outside through the FHF elevation walls and roof.

There are several conservatisms in this calculation. The primary ones are listed below in order of judged significance.

- No credit is taken in the GOTHIC model for natural circulation between the FHF and the level below through the two open staircases or between the FHF and the outside through open ductwork. Natural circulation through the staircases or ducts will reduce the maximum temperature on the FHF.

- There is no heat transfer modeled between the limiting heat load assemblies/storage cells and the storage cells surrounding them in the COBRA model. Based on the assembly/storage cell map (Reference 4.9.7), the limiting Cycle 16 fuel assemblies are spread out through the pool and are not next to each other. In addition, based on calculation F13-0001 (Reference 4.9.8), there are only four assemblies with the limiting heat generation rate. Therefore, it is expected that the adjacent storage cells will be cooler than the limiting one. Heat transfer to these cooler cells would reduce the cladding temperature in the limiting assemblies/storage cells.

- All concrete heat sinks in the GOTHIC model are assumed to be adiabatic on the opposite side. In reality, heat will be transferred to the Reactor Building, the Control Complex, and the lower levels of the Auxiliary Building in the postulated scenario.

- Conservative values for surface radiative properties are used in the GOTHIC model (high solar absorptivity, low thermal emissivity). Depending on the pigment used to paint the exterior surfaces of the FHF elevation, absorbed solar radiation could be as much as approximately 40% less than what is modeled herein.

Additionally, the acceptance criterion for this analysis, that the cladding surface temperature is maintained below 565°C is also conservative. Per NUREG/CR-6451 (Reference 4.9.9), 565°C is the lowest temperature where incipient cladding failure might occur.

4.9 References

4.9.1 SECY-99-168, “Improving Decommissioning Regulations for Nuclear Power Plants,” dated June 30, 1999. (ADAMS Accession No. ML992800087)

4.9.2 GOTHIC Version 8.0(QA), Numerical Applications Inc., S&L Program No. 03.7.759-8.0

4.9.3 UHSSIM Version 1.0, S&L Program No. 03.7.870-1.0

4.9.4 Enhanced Design Basis Document for the Auxiliary Building Air Handling System, System Code: AH-XD-XE, Tab 8/7, Rev. 15

4.9.5 ASHRAE Handbook of Fundamentals, I-P Edition, 2009

4.9.6 COBRA-SFS Version 1.0, Pacific Northwest Laboratory, S&L Program No. 03.7.672-1.0

4.9.7 Records Management System-Nuclear Generation Group Record Number 5222537 (Quality Record that contains Spent Fuel Pool map)

4.9.8 Calculation F13-0001, “Radiation Source Term and Heat Generation Rate in the CR-3 Spent Fuel Pool Following Drain Down,” Revision 0

4.9.9 NUREG/CR-6451, “A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants,” dated April 1997.

5.0 ADIABATIC SPENT FUEL BUNDLE HEAT-UP – CALCULATION F13-0004

5.1 General Description

This analysis is provided to compare the conditions for the hottest fuel assembly stored in the CR-3 fuel pools to a criterion proposed in SECY-99-168, “Improving Decommissioning Regulations for Nuclear Power Plants,” applicable to offsite emergency response for the unit in the decommissioning process. That criterion considers the time for the hottest assembly to heat up from 30°C to 900°C adiabatically; if that time is greater than 10 hours, then offsite emergency preplanning involving the plant is not necessary. In SECY-99-168, this is identified as the second proposed criterion. This is a beyond design basis event and is bounding for any other loss of inventory event.

5.2 Methodology

Calculation F13-0004 identified the hottest fuel assembly in the CR-3 pools based on fuel management records and determined that its heat generation rate, as of September 26, 2013, will be 5700 BTU/hr. The bundle was analyzed as a closed system with no work or heat transfer out of the system. However, there is heat generation in the system. The fuel bundle is modeled as being insulated by a perfect insulator. The masses and specific heats were identified for the materials that make up the fuel assembly: uranium dioxide (UO_2), Zircaloy, stainless steel, and Inconel.

The fundamental equation for a closed system is:

$Q + W = \Delta U$ (Reference 5.5.1)

Since work (W) is zero the equation reduces to:

$Q = \Delta U = m \bullet Cp \bullet (\Delta T)$ (mass X specific heat of the materials X temperature change)

Q is a function of heat generation rate and time:

$Q = Q_{dot} \bullet t$, where $Q_{dot} = 5700$ BTU/hr

Solving for time (to heat from 30°C to 900°C), the equation becomes:

$t = m \bullet Cp \bullet (\Delta T) / Q_{dot}$

5.3 Results

The calculation is based on specific heat values (Reference 5.5.2) of 0.0356 BTU/lbm °F for UO_2 and 0.077 for Zircaloy. The calculation determined a time to heat up to 900°C to be 19.7 hours.

5.4 Conclusions

Based on this analysis, the second proposed criterion in SECY-99-168 is satisfied in that the adiabatic heat up of the hottest bundle is > 10 hours.

In the NRC Safety Evaluation for the Maine Yankee Atomic Power Station, dated September 3, 1998 (ADAMS Accession No. 9809140214), for exemptions to emergency preparedness requirements the NRC stated:

> "However, the staff reviewed the calculations and determined that the bounding scenario would be with the active fuel totally uncovered and water blocking the assembly inlet so that no natural circulation flowpath exists. The staff calculated that, for this case, as of August 1, 1998, it would take approximately 10 hours for the hottest location in the highest power assembly to reach 900°C. The heatup time was calculated assuming an adiabatic heatup of a fuel rod and using conservative decay heat assumptions. An adiabatic heatup is defined as one in which all heat generated is retained in the system, with no heat loss to the surroundings. This definition corresponds to a physical situation in which the spent fuel pool water is lost, no cooling mechanism is available, and the fuel is surrounded by a perfect insulator. The staff considers this scenario bounding for any loss-of-inventory scenario, since any other scenario would have some heat removal from the assembly and a longer heatup time. Consequently, the staff determined that, in view of the low likelihood of the bounding scenario, and the time elapsed since the shutdown of the facility, there would be sufficient time for mitigative actions and, if necessary, offsite protective measures to be initiated after a postulated loss of water and before a postulated release of radioactivity resulting from spent fuel overheating."

Given that the time for the hottest CR-3 fuel assembly to adiabatically heat up from 30°C to 900°C is almost double the time calculated for the Maine Yankee Atomic Power Station, then the same staff conclusion would apply to CR-3; that there would be sufficient time for mitigative actions and, if necessary, offsite protective measures to be initiated after a postulated loss of water and before a postulated release of radioactivity resulting from spent fuel overheating.

5.5 References

5.5.1 Marks Standard Handbook for Mechanical Engineers, 9th Edition

5.5.2 Incropera & DeWitt, "Fundamentals of Heat and Mass Transfer," Fifth Edition, John Wiley & Sons, Inc., 2002

6.0 DOSE RATES DUE to SPENT FUEL ASSEMBLIES in the CR3 SPENT FUEL POOL FOLLOWING DRAIN DOWN – CALCULATION N13-0002

6.1 General Description

The purpose of this calculation is to evaluate the effects of a loss of water inventory from the CR-3 SFP as of September 26, 2013. Specifically, the primary purpose of this calculation is to determine the radiological impact for the Control Room and to the public at the EAB of the CR-3

SFP for the event in which the spent fuel assemblies are uncovered following drain down. This is a beyond design basis event.

Dose rates are calculated at other locations to provide supplemental information regarding the impact to plant personnel. This information can be used to provide some level of preplanning in the event the spent fuel assemblies are uncovered following drain down, however those results are not reported in this summary.

Gamma dose rates for the CR-3 Control Room and EAB locations are reported in this summary.

6.2 Radiation Shine Dose Due to Fuel Uncovery

The Monte Carlo N-Particle version 5-1.60 (MCNP5) (Reference 6.9.1) radiation transport computer code is used for calculating the dose rates from the CR-3 SFP. MCNP5 was developed and is maintained by the Los Alamos National Laboratory and is widely used and accepted by the nuclear utility industry to perform radiological analysis. MCNP5 has undergone verification and validation under the vendor Nuclear QA Program, and therefore, verification and validation is not required as part of this analysis.

The source terms for neutron and gamma radiation are taken from CR-3 calculation F13-0001, “Radiation Source Term and Heat Generation Rate in the CR-3 Spent Fuel Pool Following Drain Down,” Revision 0, (Reference 6.9.2).

6.3 Design Inputs

The information and values used as design input to perform the calculations in this document are identified below.

6.3.1 SFP Geometry

The SFP geometry is determined from CR-3 concrete drawings. Concrete wall thicknesses and orientations are shown in Figure 6.3-1. The bottom of the SFP is a 5 ft. thick concrete slab between EL. 113’4” and 118’4”. The elevation of the top of the source region corresponds to the elevation at the top of the fuel racks and is taken from Reference 6.9.3.

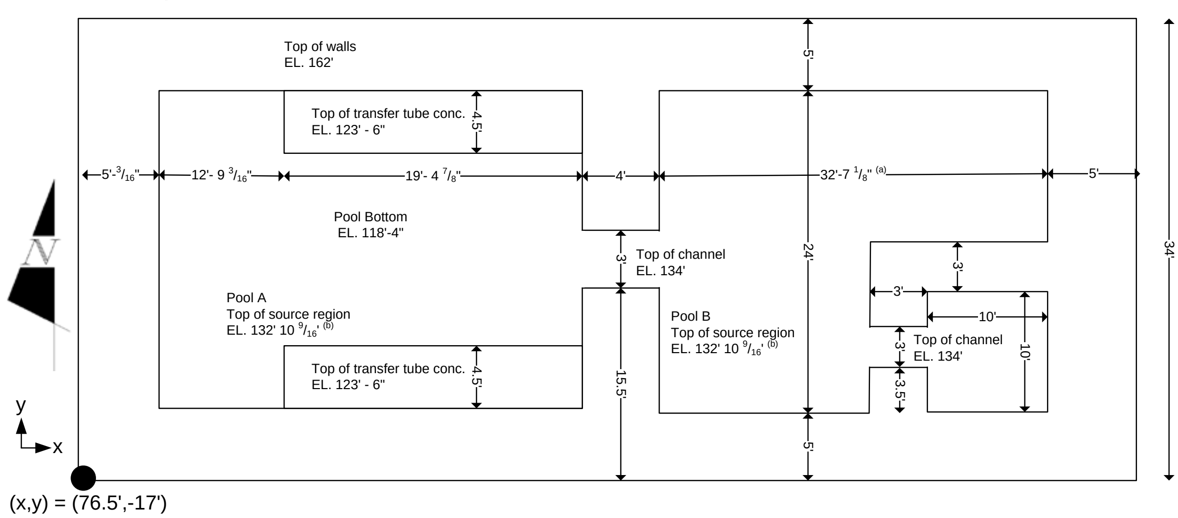

Figure 6.3-1: Plan view of the CR-3 SFP Concrete Outline and Sections

6.3.2 CR-3 Buildings Layout

Exterior concrete walls, floors at the highest elevations, and concrete roofs of several buildings within the CR-3 power block are included in the model. Figure 6.3-2 provides dimensions of the building geometries used in the model. In some instances, the walls, floors, or roofs are not continuously a single dimension, and therefore, the minimum concrete thickness is identified from drawings and used in this analysis.

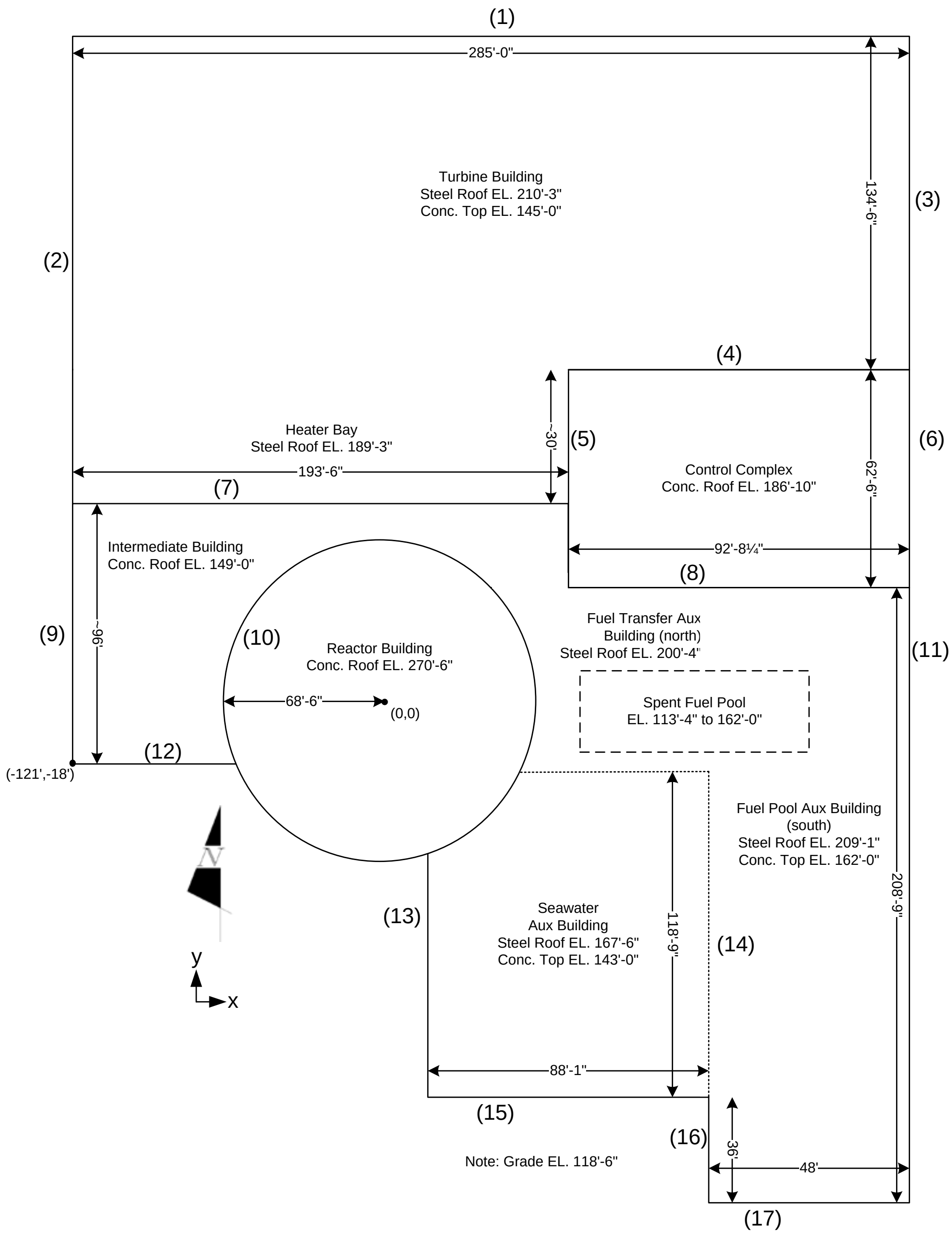

Figure 6.3-2: Plan view of CR-3 Buildings in Power Block

6.3.3 Materials

Four materials are defined in the MCNP5 input for the dose rate analysis of the CR-3 SFP. Concrete, air, and a homogenized mixture for the radiation source region for Pool A and Pool B are defined for input into MCNP5. Homogenized mixtures for the radiation source region of Pools A and B are defined using uranium dioxide (UO_2), Zircaloy, Stainless Steel 304 (SS304), Inconel, and air. Mass densities and compositions for concrete and air are taken from PNNL-15870 (Reference 6.9.4).

Fuel Assembly and Fuel Rack Materials

Material compositions of UO_2, Zircaloy, SS304, and Inconel are used to define homogenized volumes that represent the radiation source region of Pools A and B. Atom densities of SS304 and Inconel are taken from PNNL-15870 (Reference 6.9.4). Atom densities of UO_2 and Zircaloy are calculated in Section 6.6. Mass densities and compositions for concrete and air are taken from PNNL-15870 (Reference 6.9.4). Homogenized mixture definitions for the radiation source region of both Pool A and Pool B are calculated in Section 6.6.

Fuel Assemblies and Racks – Weights and Densities

The weights and mass densities of materials within the fuel assemblies and fuel racks are used in Section 6.6 to determine material volumes and volume fractions for the mixtures within Pools A and B. The volume fractions are necessary for calculating the homogenized mixture definitions. The values pertaining to the assemblies and racks are taken from CR-3 calculation F97-0014.

6.3.4 Source Definition

The neutron and gamma energy spectra that represent the source term of the spent fuel assemblies in both Pools A and B is provided in Tables 65 and 64, respectively, of CR-3 calculation F13-0001 (Reference 6.9.2). The source terms in this analysis correspond to four years after CR-3 shutdown.

6.3.5 Distance to EAB

For calculating the offsite dose, the distance to the EAB is taken from Section 2.3.4.1 of the CR-3 FSAR (Reference 6.9.5). The distance from the center of the Reactor Building to the EAB in all directions is 1340 meters; also from Section 2.3.4.1 of the CR-3 FSAR.

6.3.6 Dose Conversion Factors

Neutron and gamma flux-to-dose conversion factors are used to calculate the doses. Flux-to-dose conversion factors are taken from ANSI/ANS-6.1.1-1977 as provided in the MCNP5 Manual (Reference 6.9.6).

6.4 Assumptions

6.4.1 For the purposes of this calculation, only exterior concrete walls and concrete floor slabs at the highest elevations are modeled. Any steel or other structural material within the buildings is not included in the MCNP5 model. This assumption is conservative since it ignores any added shielding effects.

6.4.2 For the purposes of calculating the dose rates beyond the power block, it is assumed that the CR-3 plant grade (EL. 118'6") extends beyond the power block. The MCNP5 point detectors are placed at four feet above the CR-3 plant grade elevation (i.e., at EL. 122'6"). The CR-3 power block grade is located above the grade elevation beyond the power block. This assumption is conservative since the dose rates will be larger at higher elevations due to radiation scattering in the air.

6.4.3 It is assumed that the top of the Reactor Building is a cylindrical slab rather than a dome and that the top of the Reactor Building ends at the base of the dome. This assumption will not have a significant impact on calculated doses since the dome does not present a significant radiation shield or scatter surface for radiation that contributes to the doses at specified dose point locations.

6.4.4 The top elevation of the source region for Pool B is assumed to be the same as for Pool A. The top of the radiation source is set at the top of the spent fuel storage racks and the length of the spent fuel storage racks in Pool B is less than that of Pool A. The length of the racks in Pool A is 171 ⅝" + 2 ¾" and the length of the racks in Pool B is less than that of Pool A (Reference 6.9.3). The rack length for Pool A is used for both pools since it is conservative as the source will be located nearer to the top of the SFP.

6.4.5 For the purposes of this analysis, all steel is modeled using the elemental composition of Stainless Steel 304. It is assumed that the fuel racks are composed entirely of stainless steel.

6.4.6 For buildings that do not impact the dose rate at the specified dose locations, the building dimensions in Figure 6.3-2 are approximate. The building lengths are determined from drawings based on the nearest column lines. The dose locations within the power block are on the FHF and the Control Complex roof. The dimensions of the Control Complex and SFP are exact. The building dimensions of the Turbine Building, Intermediate Building, and Auxiliary Building are approximated based on the nearest column lines since they will not affect the dose rates. Also, the concrete structures of these buildings end below the FHF which is at EL. 162'. The approximate building dimensions are included to aid in visualizing the CR-3 power block.

6.5 <u>Methodology</u>

The MCNP5 computer code is used to model the CR-3 SFP and surrounding buildings for the dose rate analysis. A complete MCNP5 model requires a definition of the geometry, materials, source term, and tallies.

<u>Geometry</u>

The CR-3 site is modeled using the plant coordinate system with the center of the Reactor Building, as the origin in the x-y directions and the z direction, corresponding to the plant elevation.

Not all buildings included in the model are expected to have an impact on the results, but they were included for completeness. The geometry of the MCNP5 model includes only the concrete sections of the major CR-3 buildings within the power block. The modeled buildings are the Turbine Building, Control Complex, Intermediate Building, Reactor Building, and Auxiliary Building. The exterior concrete walls and highest elevation concrete slabs or roofs are modeled. Any internal building details or structural material above the highest elevation concrete are not included in the model. For example, the south Auxiliary Building concrete extends to EL. 143'. There is a steel roof at EL. 167'6" that is not included in the model. This approach is conservative since it minimizes the amount of radiation shielding.

The modeled geometry extends outward to a radius of 1E+09 cm from the centerline of the Reactor Building. This approach accounts for any contribution to the dose rate from radiation back scatter in the air.

Materials

Other than the Pool A and Pool B homogenized source regions, concrete and air are the only shielding materials included in the MCNP5 model. The radiation source regions in Pool A and Pool B are each assigned a homogenized mixture of materials making up the fuel assemblies and fuel racks. The mixture definitions are used to represent all of the radiation source regions in Pools A and B. Figure 6.5 shows the layout of materials within Pools A and B of the MCNP5 model.

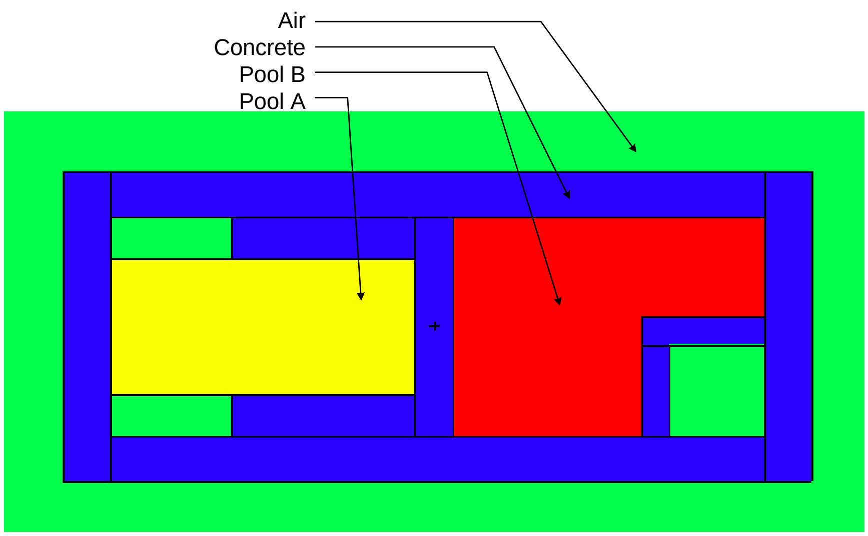

Figure 6.5: Plan View of CR-3 SFP Materials in MCNP5 Model

Source

Dose rates from three types of radiation are accounted for in this analysis. Contributions from gammas, neutrons, and secondary gammas from neutron interactions are calculated. The source terms represent the total activity in all assemblies in both pools of the CR-3 SFP.

Acceptance Criteria for Continuous Occupancy Dose

Section II.B.2 of NUREG-0737 (Reference 6.9.7) indicates that the dose rate for areas requiring continuous occupancy under accident conditions is to be <15 mrem/hr (averaged over 30 days).

Acceptance Criteria for Dose at the EAB

The acceptance criterion for exemption from requiring offsite emergency planning zones is less than 1 rem projected dose for a four day period. The acceptance criterion for establishing the EALs proposed in this request is less than 100 mrem for a two hour period consistent with the lower threshold for declaration of a Site Area Emergency.

6.6 Calculations

The calculations performed for the dose rate analysis of the CR-3 SFP are provided in this section.

6.6.1 Homogenized Mixture Definitions

Pools A and B are assigned a homogenized mixture containing UO_2, Zircaloy, SS304, Inconel, and air.

The total volumes of the radiation source regions within Pool A and Pool B in the MCNP5 model are needed to determine the volume fractions for the homogenized mixture. The calculation to determine the volume of the homogenized radiation source regions within the pools modeled in MCNP5 discussed in the following section.

6.6.2 Gamma Dose Rates

The neutron, n-gamma, and gamma dose rates were calculated and the results indicate that the dose rates are dominated by the contribution from the gamma radiation. Therefore, only the gamma radiation is analyzed at the dose locations of the CR-3 Control Room and EAB locations. Table 6-6 provides the dose rates at those point detector dose locations along with statistical information relevant to the acceptance criteria mentioned in Section 6.5. Since MCNP5 is a statistical code, the dose rate results are generally reported within two standard deviations of the MCNP5 calculated mean dose rate. Therefore, the upper boundary of the 95% confidence interval, which represents the maximum calculated dose rate, is also provided for each dose location.

Table 6-6: Point Detector Gamma Dose Rates

Tally Number	Dose Location	Mean Dose Rate (rem/hr)	Relative Error (fraction)	Upper Boundary of 95% Confidence Interval[(a)]
165	CR-3 Control Room[(b)]	6.45E-06	0.0983	7.72E-06
175	EAB - North[(b)]	7.44E-09	0.0991	8.91E-09
185	EAB - South[(b)]	2.85E-08	0.0963	3.40E-08
195	EAB - East[(b)]	1.56E-07	0.1300	1.97E-07
205	EAB - West[(b)]	1.81E-08	0.0999	2.17E-08

(a) The upper boundary of the 95% confidence interval of the dose rate is calculated by adding the mean value with the product of twice the mean value and the relative error. For the dose rate at the FHF North Stairwell, the upper boundary is 2.77E+01 rem/hr + (2 x 2.77E+01 rem/hr x 0.0048) = 2.80E+01 rem/hr.

(b) The relative errors for these dose locations are greater than 0.05 and pass less than 8 out of 10 statistical checks. Table 2-5 of the MCNP5 User's Manual indicates that relative errors between 0.05 and 0.5 imply that the tally results could be off by a factor of a few. However, the dose rates are insignificant and therefore, these results do not affect the conclusions of this calculation.

6.7 Results

A summary of the results from calculations performed in Section 6 of this calculation are provided in this section. The summary is based on the condition of CR-3 spent fuel assemblies as of September 26, 2013. The dose rate results decrease for dates after September 26, 2013.

Table 6.7 presents the dose rates at the dose locations based on a beyond design basis accident event (loss of water inventory in the SFP) due to direct and scattered radiation from spent fuel assemblies in the CR-3 SFP.

Table 6.7: Dose Rates in the Vicinity of the CR-3 SFP

Dose Location	Dose Rate (rem/hr)
CR-3 Control Room	7.72E-06
EAB - North	8.91E-09
EAB - South	3.40E-08
EAB - East	1.97E-07
EAB - West	2.17E-08

6.8 Conclusions

The results of the MCNP5 calculation have relative errors less than 0.05 or pass 8 out of 10 statistical checks, including the mean and relative error statistical checks.

Based on calculated direct and scattered dose rates from spent fuel assemblies in the CR-3 SFP following drain down, it is concluded that the CR-3 Control Room and offsite locations would not be considered as limited occupancy areas because the dose rates are less than 15 mrem/hr.

The doses at the EAB are well below the acceptance criteria stated in Section 6 for the changes requested in this document of less than 1 rem over four days and 100 mrem over two hours to a member of the public.

6.9 References

6.9.1 MCNP5, S&L Computer Program Number 03.7.511-5.0-1.60, Version 1.60.

6.9.2 Calculation F13-0001, “Radiation Source Term and Heat Generation Rate in the CR-3 Spent Fuel Pool Following Drain Down,” Revision 0.

6.9.3 EC 74407 Attachment Z26, "Spent Fuel Handling Using FHCR-3A,” Revision 0.

6.9.4 PNNL-15870, “Compendium of Material Composition Data for Radiation Transport Modeling,” Pacific Northwest National Laboratory, April 2006.

6.9.5 Crystal River Unit 3, Final Safety Analysis Report, Revision 31.1

6.9.6 LA-UR-03-1987, X-5 Monte Carlo Team, “MCNP: A General Monte Carlo N-Particle Transport Code, Version 5,” Volume I: Overview and Theory, Los Alamos National Laboratory, Revised 2/1/2008.

6.9.7 NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.

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ATTACHMENT C

FACILITY OPERATING LICENSE STRIKEOUT PAGES

(11) Deleted per Amendment No. 247

(12) Deleted per Amendment No. 237

(13) Deleted per Amendment No. 229

(14) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(1.) Fire fighting responses strategy with the following elements:
- a. Pre-defined coordinated fire response strategy and guidance
- b. Assessment of mutual aid fire fighting assets
- c. Designated staging areas for equipment and materials
- d. Command and control
- e. Training of response personnel

(2.) Operations to mitigate fuel damage considering the following:
- a. Protection and use of personnel assets
- b. Communications
- c. Minimizing fire spread
- d. Procedures for implementing integrated fire response strategy
- e. Identification of readily-available pre-staged equipment
- f. Training on integrated fire response strategy
- g. Spent fuel pool mitigation measures

(3.) Actions to minimize release to include consideration of:
- a. Water spray scrubbing
- b. Dose to onsite responders

(15) Deleted per Amendment No. 247

D. Physical and Cyber Security

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 2781.7 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Physical Security Plan, Revision 5," and "Safeguards Contingency Plan, Revision 4," submitted by letter dated May 16, 2006, and "Guard Training and Qualification Plan, Revision 0," submitted by letter dated September 30, 2004, as supplemented by letters dated October 20, 2004, and September 29, 2005.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 238, as supplemented by changes approved by License Amendment Nos. 242, 245 and ~~245~~ .

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ATTACHMENT D

FACILITY OPERATING LICENSE REVISION BAR PAGES

(11) Deleted per Amendment No. 247

(12) Deleted per Amendment No. 237

(13) Deleted per Amendment No. 229

(14) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions that include the following key areas:

(1.) Fire fighting responses strategy with the following elements:
- a. Pre-defined coordinated fire response strategy and guidance
- b. Assessment of mutual aid fire fighting assets
- c. Designated staging areas for equipment and materials
- d. Command and control
- e. Training of response personnel

(2.) Operations to mitigate fuel damage considering the following:
- a. Protection and use of personnel assets
- b. Communications
- c. Minimizing fire spread
- d. Procedures for implementing integrated fire response strategy
- e. Identification of readily-available pre-staged equipment
- f. Training on integrated fire response strategy
- g. Spent fuel pool mitigation measures

(3.) Actions to minimize release to include consideration of:
- a. Water spray scrubbing
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ATTACHMENT E

REVISED REGULATORY COMMITMENT

Revised Cyber Security Plan Implementation Schedule
(Milestone 8)

#	Implementation Milestone (Regulatory Commitment)	Completion Date	Basis
8	Full implementation of the *Cyber Security Plan* for all SSEP functions will be achieved in accordance with 10 CFR 73.54.	December 31, 2018	By the completion date, the *Carolina Power & Light Company and Florida Power Corporation Cyber Security Plan* will be fully implemented for all SSEP functions in accordance with 10 CFR 73.54. This date also bounds the completion of all individual asset security control design remediation actions including those that require a refuel outage for implementation.