NMC
Committed to Nuclear Excellence

Duane Arnold Energy Center
3313 DAEC Road
Palo, IA 52324-9646

Operated by Nuclear Management Company, LLC

November 17, 2000
NG-00-1904

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Request for Operating License Change (TSCR-040) – Revised Thermal-Hydraulic Analysis for the Spent Fuel Pool

References:
1. Letter, Liu (IES) to Murley (NRC), dated March 26, 1993, Spent and New Fuel Storage
2. Letter, Pulsifer (NRC) to Liu (IES), dated February 24, 1994, Duane Arnold Energy Center License Amendment 195
3. Letter, Franz (IES) to Collins (NRC), dated October 3, 1997, Refueling Operations
4. Letter, Laufer (NRC) to Liu (IES), dated April 2, 1998, Duane Arnold Energy Center License Amendment 222

File: A-117, J-80a

In accordance with the Code of Federal Regulations, Title 10, Sections 50.59 and 50.90, the Nuclear Management Company, LLC (NMC) hereby requests revision to the Operating License (OL) for the Duane Arnold Energy Center (DAEC).

In Reference 1, IES Utilities Inc. submitted a request for an Operating License Amendment to rerack the DAEC fuel pool. That request was granted as Amendment 195 (Reference 2). IES Utilities Inc. submitted Reference 3 requesting revised refueling operations. That request was granted as Amendment 222 (Reference 4). The purpose of this Operating License Amendment request is to revise the thermal-hydraulic analysis included as a license report in Reference 3. Specifically, this request proposes advanced core designs including the use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size and supercedes the thermal-hydraulic analysis included in Reference 3.

The proposed Operating License change is described in Attachment 3. A licensing report containing our revised thermal-hydraulic analysis is included as Attachment 5.

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Additionally, this request corrects some discrepancies which existed in our Reference 1 and 3 submittals. The revised thermal-hydraulic analysis included as Attachment 5 corrects these discrepancies. Correction of these discrepancies produces results that are not significantly different than the results of Reference 3.

This application has been reviewed by the DAEC Operations Committee and the Safety Committee. A copy of this submittal, along with the evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR Section 50.91. We request approval of this application no later than April 1, 2001 with a period of 60 days after the approval for implementation.

Should you have any questions regarding this matter, please contact this office.

This letter is true and accurate to the best of my knowledge and belief.

Nuclear Management Company, LLC

By Gary Van Middlesworth
Gary Van Middlesworth
DAEC Site General Manager

State of Iowa
(County) of Linn

Signed and sworn to before me on this 17th day of November, 2000,
by Gary Van Middlesworth.

NANCY S. FRANCK

Nancy S. Franck
Notary Public in and for the State of Iowa

9-28-01
Commission Expires

Attachments:
1) Evaluation Of Change Pursuant To 10 CFR Section 50.92
2) Safety Assessment
3) Proposed Change TSCR-040 To The Duane Arnold Energy Center Operating License
4) Environmental Consideration
5) Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie (Revision 4), prepared by Holtec International

cc: L. B. Swenzinski (w/a)
M. Wadley (w/o)
Docu (w/a)
B. Mozafari (NRC-NRR) (w/a)
J. Dyer (Region III) (w/a)
D. McGhee (State of Iowa) (w/a)
NRC Resident Office (w/a)

EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92

Background:

IES Utilities Inc., on March 26, 1993, requested an Operating License/Technical Specification Amendment to allow rerack of the Duane Arnold Energy Center (DAEC) fuel pool. In support of that request, the DAEC performed a thermal-hydraulic analysis for four refueling scenarios. The most limiting scenario represented a full core offload, starting 120 hours after shutdown, following an 18 month cycle, using two trains of Fuel Pool Cooling and Clean Up (FPCCU) to provide Spent Fuel Pool (SFP) cooling. The March 26, 1993 Operating License/Technical Specification Amendment Request was granted as Amendment 195.

IES Utilities Inc., on October 3, 1997, requested an Operating License Amendment to allow the start of core offload as soon as 60 hours after shutdown. In support of that request, the DAEC performed a thermal-hydraulic analysis for three additional scenarios. The most limiting scenario assumed an unplanned full core offload with two FPCCU trains running prior to initiation of Residual Heat Removal Supplemental Fuel Pool Cooling (RHR-SFPC). This scenario consisted of a normal 36 day refueling outage followed by 45 days of full power operation and a subsequent unplanned transfer of the entire core to the SFP. The analysis conservatively assumed the recently loaded fresh fuel had undergone 18 months of exposure even though it has been in the reactor for only 45 days.

In support of the October 3, 1997 Operating License Amendment Request, IES Utilities, Inc. submitted Revision 2 of the "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie." The Nuclear Management Company, LLC (NMC) hereby submits Revision 4 of the "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie." These thermal-hydraulic analyses will hereafter be referred to as Revision 2 and Revision 4, respectively. Revision 4 of the thermal-hydraulic analysis included as Attachment 5 supercedes Revision 2 submitted on October 3, 1997.

Recently, several discrepancies were discovered which existed in our March 26, 1993 and October 3, 1997 submittals. The revised thermal-hydraulic analysis included as Attachment 5 corrects these discrepancies. Correction of these discrepancies produces results that are not significantly different than the results submitted on October 3, 1997.

Nuclear Management Company, LLC, Docket No. 50-331
Duane Arnold Energy Center, Linn County, Iowa
Date of Amendment Request: November 17, 2000

Description of Amendment Request:

The March 26, 1993 and October 3, 1997 submittals contained several discrepancies. These discrepancies were discovered while preparing Revision 4 of the thermal-hydraulic analysis in preparation for this Operating License Amendment Request. Six discrepancies were

identified and are presented in tabular form below:

Calculation Input Parameter	Value used in the 1993 and 1997 submittals	Design value
Reactor Cavity Volume	157,000 gallons	136, 000 gallons
RHR Service Water Flow Rate	4800 gpm	4080 gpm
Emergency Service Water Makeup Flow	75 gpm	59.5 gpm
Height of fuel racks above bottom of SFP	14.67 feet	15.146 feet
Height of active fuel above bottom of SFP	13.52 feet	13.96 feet
RHR Flow Rate to SFP	1300 gpm	2000 gpm

The design values listed above have been used in Revision 4 of the thermal-hydraulic analysis included as Attachment 5. The error in RHR flow rate to the SFP was conservative. The first five discrepancies listed above were nonconservative, however, correction of these discrepancies produces results that are not significantly different than the results of Revision 2 of the thermal-hydraulic analysis submitted on October 3, 1997.

Beginning with Cycle 18, the DAEC plans to pursue advanced core designs including the use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size. The purpose of this Operating License Amendment Request is to revise the thermal-hydraulic analysis submitted to the NRC on October 3, 1997. Specifically, Revision 4 of the thermal-hydraulic analysis incorporates the use of advanced core designs and supercedes Revision 2 of the thermal-hydraulic analysis submitted on October 3, 1997.

Calculation results from Revision 4 of the thermal-hydraulic analysis are given below. The acceptability of these results is determined as follows:

- The maximum heat load is calculated to be 23.92×10^6 Btu/hr. This represents a slight increase from the maximum heat load value calculated in the October 3, 1997 and March 26, 1993 submittals. The increase in maximum heat load does not significantly alter the thermal-hydraulic analysis results.
- The maximum bulk pool temperature is calculated to reach 152.5°F. This is found to be acceptable since this temperature is below the temperature of 165°F previously reviewed and approved for Amendment 195.
- The maximum local water temperature is calculated to reach 177.53°F assuming 66% downcomer blockage. This is found to be acceptable since this temperature is below the temperatures of 216.3°F (no downcomer blockage) and 227.2°F (50% downcomer blockage) previously reviewed and approved for Amendment 195.
- The maximum fuel clad temperature is calculated to reach 211.28°F assuming 66% downcomer blockage. This is found to be acceptable since this temperature is below the temperatures of 264.4°F (no downcomer blockage) and 273.2°F (50% downcomer blockage) previously reviewed and approved for Amendment 195.
- The required makeup flow is calculated to be 53.05 gpm. This is not significantly changed from the flow of 52.78 gpm previously reviewed and approved for Amendment

222. The DAEC showed in the October 3, 1997 submittal that a makeup flow of 59.5 gpm could be achieved.

Revision 4 of the thermal-hydraulic analysis results in less time to take action following a loss of forced cooling with cask pit isolation gate failure event in order to prevent fuel in the SFP from being uncovered (hereafter referred to as "minimum time to action"). Revision 4 of the thermal-hydraulic analysis results in a minimum time to action of 3.8 hours compared to a minimum time to action of 4.5 hours previously reviewed and approved for Amendment 222. This change is addressed in the Basis for No Significant Hazards Consideration below.

Basis for proposed No Significant Hazards Consideration:

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

After reviewing this proposed amendment, we have concluded:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Revision 4 of the thermal-hydraulic analysis calculates a minimum time to action of 3.8 hours versus 4.5 hours previously reviewed and approved for Amendment 222. In the event of a loss of forced cooling with cask pit isolation gate failure event, the DAEC will use Emergency Service Water (ESW), a Seismic Category I system, to provide makeup to the SFP. It is estimated to take no more than 2 hours to provide ESW makeup to the SFP, therefore the minimum time to action of 3.8 hours is sufficient time to prevent uncovering the fuel in the SFP.

   Although Revision 4 of the thermal-hydraulic analysis shows an increase in maximum heat load, the proposed changes will not result in maximum bulk SFP, local water, or fuel clad temperatures in excess of those previously reviewed and approved for Amendment 195. Furthermore, the proposed changes will not result in maximum bulk SFP, local water, or fuel clad temperatures which would initiate bulk pool boiling, challenge fuel rod integrity, exceed the design limits of the fuel racks or jeopardize the structural integrity of the pool.

   Therefore, this proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Revision 4 of the thermal-hydraulic analysis shows that the proposed changes will not result in maximum bulk SFP, local water, or fuel clad temperatures which would initiate bulk pool boiling, challenge fuel rod integrity, exceed the design limits of the fuel racks or jeopardize the structural integrity of the pool.

   As stated above, the minimum time to action of 3.8 hours allows sufficient time to provide ESW makeup to the SFP.

   Therefore this proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment will not involve a significant reduction in a margin of safety.

   This change will not result in maximum bulk SFP, local water, and fuel clad temperatures in excess of those previously evaluated and accepted per Amendment 195. Revision 4 of the thermal-hydraulic analysis does result in a reduction in the minimum time to action by 0.7 hours. However, 3.8 hours does provide sufficient time to provide ESW makeup to the SFP as this task is estimated to require no more than 2 hours. Furthermore, this change does not result in any change to the Technical Specifications.

   Therefore, this proposed amendment will not involve a significant reduction in a margin of safety.

Based upon the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

Local Public Document Room Location: Cedar Rapids Public Library, 500 First Street SE, Cedar Rapids, Iowa 52401.

Attorney for Licensee: Al Gutterman; Morgan, Lewis & Bockius, 1800 M Street NW, Washington, D.C. 20036-5869.

SAFETY ASSESSMENT

By letter dated November 17, 2000, the Nuclear Management Company, LLC (NMC) submitted a request for revision of the Operating License for the Duane Arnold Energy Center (DAEC). The purpose of this amendment request is to revise the thermal-hydraulic analysis submitted to the NRC on October 3, 1997. Specifically, the revisions to that thermal-hydraulic analysis incorporate the use of advanced core designs including the use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size and seeks to correct several discrepancies in previous submittals. The revised thermal-hydraulic analysis supercedes that submitted to the NRC on October 3, 1997.

Background:

The DAEC Fuel Pool Cooling and Clean Up (FPCCU) system removes the decay heat and radioactivity released from the spent fuel elements stored in the Spent Fuel Pool (SFP). The system maintains specific fuel pool water temperature, purity, clarity and level. The FPCCU system consists of the SFP, two full capacity pumps, two heat exchangers, two filter demineralizers, two skimmer surge tanks and associated piping, valves and instrumentation. The FPCCU heat exchangers are cooled by the Reactor Building Closed Cooling Water system. Either loop of the Residual Heat Removal (RHR) system can be operated in the Supplemental Fuel Pool Cooling (RHR-SFPC) mode to provide additional cooling to the SFP.

IES Utilities Inc., on March 26, 1993, requested an Operating License/Technical Specification Amendment to allow rerack of the DAEC fuel pool. In support of that request, the DAEC performed a thermal-hydraulic analysis for four refueling scenarios or "cases". Case 1 simulated the Standard Review Plan (SRP) discharge condition labeled as the normal scenario with appropriate modification for 18 month cycles; Case 2 simulated the full core offload scenario of the SRP. The purpose of these cases was to demonstrate compliance of the FPCCU system with the SRP. Case 3, the most limiting scenario, represented a full core offload, starting 120 hours after shutdown, following an 18 month cycle, using two trains of FPCCU to provide SFP cooling. Case 4 represented a full core offload using two trains of FPCCU, starting 120 hours after shutdown, following 36 days of power operation. The March 26, 1993 Operating License/Technical Specification Amendment Request was granted as Amendment 195.

IES Utilities Inc., on October 3, 1997, requested an Operating License Amendment to allow the start of core offload as soon as 60 hours after shutdown. This request was granted as Amendment 222. In support of that request, the DAEC performed a thermal-hydraulic analysis for three additional scenarios or "cases". Case A assumed a planned (i.e., follows 18 months of power operation) full core offload with one FPCCU train running prior to initiation RHR-SFPC. RHR-SFPC was assumed to be in service when SFP temperature reached 120°F. RHR-SFPC provides cooling to the SFP and reactor cavity by taking suction from the RHR Shutdown Cooling system piping and diverting a portion of flow to the SFP. It is assumed that when RHR-SFPC is in operation, the transfer canal gates are open and

water is able to flow from the SFP to the flooded reactor cavity. Further, it was assumed that only one RHR system cooling train is in operation, diverting flow from this RHR train. The analysis conservatively assumed FPCCU is secured when RHR-SFPC is used. Case B was identical to Case A except it assumed two FPCCU trains are running prior to initiation of RHR-SFPC. Case C, the most limiting scenario, assumed an unplanned full core offload with two FPCCU trains running prior to initiation of RHR-SFPC. This scenario consisted of a normal 36 day refueling outage followed by 45 days of full power operation and a subsequent unplanned transfer of the entire core to the SFP. The analysis conservatively assumed the recently loaded fresh fuel had undergone 18 months of exposure even though it would have been in the reactor for only 45 days.

In support of the October 3, 1997 Operating License Amendment Request, IES Utilities, Inc. submitted Revision 2 of the "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie." NMC hereby submits Revision 4 of the "Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie" as Attachment 5 to this Operating License Amendment Request. These thermal-hydraulic analyses will hereafter be referred to as Revision 2 and Revision 4, respectively. Revision 4 of the thermal-hydraulic analysis supercedes Revision 2 submitted on October 3, 1997.

Recently, several discrepancies were discovered which existed in the March 26, 1993 and October 3, 1997 submittals. Revision 4 of the thermal-hydraulic analysis included as Attachment 5 corrects these discrepancies. Correction of these discrepancies produces results that are not significantly different than the results of Revision 2 of the thermal-hydraulic analysis submitted on October 3, 1997.

Basis for Change:

The March 26, 1993 and October 3, 1997 submittals contained several discrepancies. These discrepancies were discovered while preparing Revision 4 of the thermal-hydraulic analysis in preparation for this Operating License Amendment Request. Six discrepancies were identified and are presented in tabular form below:

Calculation Input Parameter	Value used in the 1993 and 1997 submittals	Design value
Reactor Cavity Volume	157,000 gallons	136, 000 gallons
RHR Service Water Flow Rate	4800 gpm	4080 gpm
Emergency Service Water Makeup Flow	75 gpm	59.5 gpm
Height of fuel racks above bottom of SFP	14.67 feet	15.146 feet
Height of active fuel above bottom of SFP	13.52 feet	13.96 feet
RHR Flow Rate to SFP	1300 gpm	2000 gpm

The design values listed above have been used in Revision 4 of the thermal-hydraulic analysis included as Attachment 5. The error in RHR flow rate to the SFP was conservative. The first five discrepancies listed above were nonconservative, however, correction of these discrepancies produces results that are not significantly different than the results of Revision

2 of the thermal-hydraulic analysis submitted on October 3, 1997.

Beginning with Cycle 18, the DAEC plans to pursue advanced core designs including the use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size. The purpose of this amendment request is to revise and supercede the thermal-hydraulic analysis submitted to the NRC on October 3, 1997. Specifically, Revision 4 of the thermal-hydraulic analysis incorporates the use of advanced core designs and supercedes Revision 2 submitted on October 3, 1997.

As stated earlier, Case C assumes a normal 36 day refueling outage followed by 45 days of full power operation and a subsequent unplanned transfer of the entire core to the SFP. Revision 4 assumes, for Case C, that recently loaded fresh fuel has undergone 45 days of exposure, rather than the overly conservative assumption of 18 months of exposure assumed in Revision 2. This assumption gives representative results since decreases in the refueling outage duration or increases in the days of operation before the unplanned offload of Case C result in less than a 1% change in SFP temperatures.

Revision 4 of the thermal-hydraulic analysis uses revised versions of the FLUENT and TBOIL computer codes. The more recent codes replace the simplified, conservative methods used in Revision 2, with state of the art techniques that yield more realistic results while remaining conservative. The NRC has reviewed and approved the use of these revised codes for other licensees.

Revision 4 of the thermal-hydraulic analysis results in a maximum heat load of 23.92×10^6 Btu/hr. This represents a slight increase from the maximum heat load of 22.54×10^6 Btu/hr calculated in Revision 2 of the thermal-hydraulic analysis submitted on October 3, 1997 and the maximum heat load of 18.73×10^6 Btu/hr calculated in the thermal-hydraulic analysis submitted on March 26, 1993. The increase in maximum heat load does not significantly alter the thermal-hydraulic analysis results.

The maximum bulk pool temperature previously reviewed and approved for Amendment 195 was 165°F. Revision 2 of the thermal-hydraulic analysis, submitted on October 3, 1997 calculated a maximum bulk pool temperature of 159.87°F. The decrease in maximum bulk pool temperature resulted from the October 3, 1997 analysis assuming RHR-SFPC is initiated as SFP temperature approaches 120°F versus the March 26, 1993 analysis which assumed two trains of FPCCU in use. Revision 4 of the thermal-hydraulic analysis results in a maximum bulk pool temperature of 152.5°F. This is found to be acceptable since this temperature is below the temperatures previously reviewed and approved.

The maximum local water temperature previously reviewed and approved for Amendment 195 was 216.3°F assuming no downcomer blockage and 227.2°F assuming 50% downcomer blockage. Revision 2 of the thermal-hydraulic analysis, submitted on October 3, 1997 calculated a maximum local water temperature of 171.74°F and assumed 66% downcomer blockage. The decrease in maximum local water temperature resulted from the October 3,

1997 analysis assuming RHR-SFPC is initiated as SFP temperature approaches 120°F versus the March 26, 1993 analysis which assumed two trains of FPCCU in use. Revision 4 of the thermal-hydraulic analysis results in a maximum local water temperature of 177.53°F and assumes 66% downcomer blockage. This is found to be acceptable since this temperature is below the temperatures previously reviewed and approved.

The maximum fuel clad temperature previously reviewed and approved for Amendment 195 was 264.4°F assuming no downcomer blockage and 273.2°F assuming 50% downcomer blockage. Revision 2 of the thermal-hydraulic analysis, submitted on October 3, 1997 calculated a maximum fuel clad temperature of 230.46°F and assumed 66% downcomer blockage. The decrease in maximum fuel clad temperature resulted from the October 3, 1997 analysis assuming RHR-SFPC is initiated as SFP temperature approaches 120°F versus the March 26, 1993 analysis which assumed two trains of FPCCU in use. Revision 4 of the thermal-hydraulic analysis results in a maximum fuel clad temperature of 211.28°F and assumes 66% downcomer blockage. The further reduction in maximum fuel clad temperature in Revision 4 results from using the revised computer codes previously discussed. The maximum fuel clad temperature of 211.28°F calculated in Revision 4 is found to be acceptable since this temperature is below the temperatures previously reviewed and approved.

Therefore, Revision 4 of the thermal-hydraulic analysis shows that the proposed change will not result in maximum bulk SFP, local water, or fuel clad temperatures which would initiate bulk pool boiling, challenge fuel rod integrity, exceed the design limits of the fuel racks or jeopardize the structural integrity of the pool.

Revision 4 of the thermal-hydraulic analysis results in a required makeup flow of 53.05 gpm. This is not significantly changed from the flow of 53 gpm previously reviewed and approved in Amendment 222. The DAEC showed, in the October 3, 1997 submittal, that a makeup flow of 59.5 gpm could be achieved.

Revision 4 of the thermal-hydraulic analysis results in slightly less time to take action following a loss of forced cooling with cask pit isolation gate failure event in order to prevent fuel in the SFP from being uncovered (hereafter referred to as "minimum time to action"). Revision 4 of the thermal-hydraulic analysis results in a minimum time to action of 3.8 hours versus 4.5 hours previously reviewed and approved for Amendment 222. In the event of a loss of forced cooling with cask pit isolation gate failure event, the DAEC will use Emergency Service Water (ESW), a Seismic Category I system, to provide makeup to the SFP. It is estimated to take no more than 2 hours to provide ESW makeup to the SFP, therefore the minimum time to action of 3.8 hours is sufficient time to prevent uncovering the fuel in the SFP.

Therefore, we have concluded that the proposed amendment to the DAEC Operating License is acceptable.

PROPOSED CHANGE TSCR-040 TO THE DUANE ARNOLD ENERGY CENTER OPERATING LICENSE

The holders of license DPR-49 for the Duane Arnold Energy Center (DAEC) propose to amend the Operating License by deleting the referenced page and replacing it with the attached, new page. The affected page is given below.

AFFECTED PAGE

Operating License page 3

SUMMARY OF CHANGES:

Page	Description of Change
OL-3	Revise Operating License item 2.B (2) to allow refueling activities pursuant to letter dated March 26, 1993 and letter dated November 17, 2000.

2.B.(2) NMC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000 ~~October 3, 1997~~;

2.B.(3) NMC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

2.B.(4) NMC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated radioactive apparatus components;

2.B.(5) NMC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

Maximum Power Level

2.C.(1) NMC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1658 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 234, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. The Nuclear Management Company, LLC (NMC) has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment supercedes the thermal-hydraulic analysis submitted to the NRC on October 3, 1997, incorporates advanced fuel designs including use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size and seeks to correct several discrepancies in previous submittals. Attachment 5 shows maximum Spent Fuel Pool (SFP) temperatures, minimum time to action following loss of forced cooling with cask pit isolation gate failure, and local water and fuel clad temperatures will remain within acceptable limits and will not initiate bulk pool boiling, challenge fuel rod integrity, exceed the design limits of the fuel racks or jeopardize the structural integrity of the SFP. Therefore, there will be no significant change in the types of effluents and no significant increase in the amounts of any effluents released offsite.

3. The proposed amendment supercedes the thermal-hydraulic analysis submitted to the NRC on October 3, 1997, incorporates advanced fuel designs including use of GE-14 fuel, increased fuel burnup, increased cycle length and increased reload batch size and seeks to correct several discrepancies in previous submittals. Attachment 5 shows maximum SFP temperatures, minimum time to action following loss of forced cooling with cask pit isolation gate failure, and local water and fuel clad temperatures will remain within acceptable limits and will not initiate bulk pool boiling, challenge fuel rod integrity, exceed the design limits of the fuel racks or jeopardize the structural integrity of the SFP. Therefore, there will be no significant increase in either individual or cumulative occupational radiation exposure.

Attachment 5 to NG-00-1904

Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie

Prepared by Holtec International
Revision 4

HOLTEC
INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797- 0900
Fax (856) 797 - 0909

Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie

FOR

IES Utilities

Holtec Report No: HI-971746

Holtec Project No: 60884

Report Category: A

Report Class : SAFETY RELATED

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HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (609) 797- 0900
Fax (609) 797 - 0909

REVIEW AND CERTIFICATION LOG

DOCUMENT NAME :	Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie
HOLTEC DOCUMENT I.D. NUMBER :	971746
HOLTEC PROJECT NUMBER :	60884
CUSTOMER/CLIENT:	IES Utilities

REVISION BLOCK

REVISION NUMBER *	AUTHOR & DATE	REVIEWER & DATE	QA & DATE	APPROVED & ! DATE	DIST. x
ORIGINAL	E Rosenbaum 7/24/97	Indresh Rampall INDRESH RAMPALL 7/24/97	S. SHAH FOR V. GUPTA 7/24/97	A. S. 7/24/97	
REVISION 1	E Rosenbaum 8/18/97	Indresh Rampall INDRESH RAMPALL 8/18/97	V. Gupta V. GUPTA 8-19-97	A.S. 8/19/97	C
REVISION 2	E Rosenbaum 9/17/97	Indresh Rampall INDRESH RAMPALL 9/17/97	S SHAH FOR V. GUPTA 9/18/97	A.S. 9/18/97	C
REVISION 3	E Rosenbaum 07/07/2000	DMM 7/13/00	M Phipps 7/13/00	E Rosenbaum 7/13/2000	C
REVISION 4	E Rosenbaum 10/30/2000	DMM 11/1/00	N/A	E Rosenbaum 11/1/2000	C
REVISION 5					
REVISION 6					

This document conforms to the requirements of the design specification and the applicable sections of the governing codes.

Note : Signatures and printed names required in the review block.

* A revision of this document will be ordered by the Project Manager and carried out if any of its contents is materially affected during evolution of this project. The determination as to the need for revision will be made by the Project Manager with input from others, as deemed necessary by him.

! Must be Project Manager or his designee.

x Distribution : C : Client
M : Designated Manufacturer
F : Florida Office

*** Report category on the cover page indicates the contractual status of this document as ***
A = to be submitted to client for approval I = for client's information N = not submitted externally

THE REVISION CONTROL OF THIS DOCUMENT IS BY A "SUMMARY OF REVISIONS LOG" PLACED BEFORE THE TEXT OF THE REPORT.

SUMMARY OF REVISIONS

REVISION 4: Revision 4 is issued to update the calculations to reflect several updated input parameters including (1) modified reactor well volume, (2) increase in intertie flow rate, (3) decrease in RHR coolant water flow rate and (4) increase in maximum fuel assembly burnup. Several editorial changes to the text are also incorporated.

All revisions to the text and appendices are marked by vertical bars in the margins.

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* Note: Appendices B, C and D contain Holtec proprietary material. These appendices are not provided to the client, but are available for review at Holtec's corporate offices.

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* Note: Appendices B, C and D contain Holtec proprietary material. These appendices are not provided to the client, but are available for review at Holtec's corporate offices.

1.0 INTRODUCTION

The Duane Arnold Energy Center is a boiling water reactor (BWR) style nuclear electric generating station owned and operated by IES Utilities. The plant has the ability to provide cooling to the spent fuel pool (SFP) by diverting a portion of the flow discharged from the Residual Heat Removal (RHR) system heat exchangers to the SFP. It must be demonstrated that the RHR system is capable of cooling both the reactor vessel cavity and the SFP simultaneously during both planned and unplanned full core discharge scenarios.

The primary purpose of the RHR system is to reject heat from the reactor vessel cavity during a plant shutdown. The RHR heat exchangers are TEMA type CEU shell-and-tube heat exchangers with reactor water on the shell side and river water on the tube side. If the RHR system is to be used to provide cooling to the SFP, the system must be capable of rejecting the decay heat of the core plus the residual decay heat in the previously discharged fuel assemblies in the SFP.

If RHR Supplemental Fuel Pool Cooling is operational, then the water which is discharged from the RHR system to the SFP must be returned to the reactor cavity to prevent the SFP or skimmer surge tanks from overflowing. Two available options (others may exist) include:

1. Opening the transfer canal gates and allowing water to flow from the SFP to the flooded reactor cavity.

2. Operating the FPCCU system with suction from the skimmer surge tanks and discharge to the reactor cavity.

Operating the FPCCU system will result in a larger total cooling capacity for the combined SFP-reactor cavity system, which could only serve to reduce the bulk temperatures. Therefore, all analyses presented in this report are assumed to have the transfer canal gates open and the FPCCU system off.

The total decay heat generation rate in the SFP and the reactor will vary based on the fuel power history, actual SFP inventory, etc. The design basis heat rejection capability of the RHR is defined by the design basis RHR service water temperature and flow rate. Boiling must be prevented from occurring while forced cooling is available. Local water and fuel clad temperatures must be sufficiently low to preclude nucleate boiling on the fuel rod surfaces.

In order to demonstrate the sufficiency of the RHR to remove total SFP/reactor decay heat load, a series of thermal evaluations are performed. These evaluations examine the following parameters:

1. Maximum Spent Fuel Pool Temperatures
2. Minimum Time-to-Boil After a Loss of Forced Cooling
3. Local Water and Fuel Cladding Temperatures

The maximum SFP temperatures are calculated to determine if the bulk SFP temperature will exceed safe temperature limits. The time-to-boil calculations are performed with the initial temperatures set to the maximum calculated SFP bulk temperatures. *These calculations are performed two ways, with and without a* coincident cask pit isolation gate failure. The local water and fuel cladding temperature analyses are performed with the net SFP heat generation set at the maximum calculated values.

It should be noted that the analyses presented in this report are performed, in part, to update the licensing basis analyses originally performed for Holtec Project 20170 and documented in Holtec Report HI-93972, Revision 3 [1].

2.0 METHODOLOGY

2.1 Maximum Spent Fuel Pool Temperatures

The transient thermal response of the SFP, reactor and RHR system to decay heat load transients is governed by the following equations:

$$C_{RXW} \times \frac{\partial T_{RXW}}{\partial \tau} = Q_{RXW}(\tau) + W \times Cp \times (T_{SFP} - T_{RXW}) - Q_{RHR}(T_{RXW})$$

$$C_{SFP} \times \frac{\partial T_{SFP}}{\partial \tau} = Q_{SFP}(\tau) - W \times Cp \times (T_{SFP} - T_{RHR}) - Q_{ENV}(T_{SFP})$$

where:

C_{RXW} = Reactor vessel cavity thermal capacity, Btu/°F
C_{SFP} = Spent fuel pool thermal capacity, Btu/°F
T_{RXW} = Bulk reactor vessel cavity temperature, °F
T_{SFP} = Bulk spent fuel pool temperature, °F
τ = Time after reactor shutdown, hr
$Q_{RXW}(\tau)$ = Transient decay heat generation rate in reactor, Btu/hr
$Q_{SFP}(\tau)$ = Transient decay heat generation rate in SFP, Btu/hr
W = RHR Supplemental Fuel Pool Cooling mass flow rate, lb/hr
Cp = Specific heat capacity of water, Btu/(lb×°F)
$Q_{RHR}(T_{RXW})$ = RHR heat exchangers heat rejection rate, Btu/hr
T_{RHR} = RHR heat exchangers exit temperature, °F
$Q_{ENV}(T_{SFP})$ = SFP Heat losses to the environment, Btu/hr

The total transient decay heat in the SFP (Q_{SFP}) consists of both an invariant contribution and a time-varying contribution. The invariant contribution is due to the heat generation from previously discharged fuel assemblies. Due to the nature of exponential decay, the decay heat for "old" fuel assemblies is relatively constant over short periods of time. Fuel transfer from the reactor to the SFP will increase Q_{SFP} and correspondingly decrease Q_{RXW}.

The invariant decay heat generation rates of both the previously and freshly discharged fuel assemblies is determined using the Holtec QA validated computer program DECOR [2]. This program can perform decay heat calculations using either Branch Technical Position ASB 9-2 [3] or the ORIGEN2 [4] computer code from the Radiation Shielding Information Center (RSIC) at the Oak Ridge National Laboratory (ORNL). For this analysis, the ORIGEN2 option is used. The fuel discharge schedule for this analysis [*24*] is included in Table 1. All fuel assemblies are assumed to have been irradiated to the appropriate maximum burnup level.

The heat exchanger heat rejection rate (Q_{RHR}) is determined by developing a thermal-hydraulic model directly from the geometry of the heat exchanger, using the Holtec QA validated computer program STER [22]. The STER program is then used to determine outlet temperatures

corresponding to specified inlet temperature and flow rate conditions. The resulting temperature results are used to calculate the temperature effectiveness (p) as:

$$p = \frac{T_{co} - T_{ci}}{T_{hi} - T_{ci}}$$

where:

T_{co} is the coolant water outlet temperature, °F
T_{ci} is the coolant water inlet temperature, °F
T_{hi} is the heated water inlet temperature, °F

The calculated temperature effectiveness value is used in the following formulation for the heat rejection rate:

$$Q_{RHR}(T_{RXW}) = W_c \times Cp \times p \times (T_{RXW} - T_{ci})$$

where Wc is the coolant water flow rate, in lb/hr, and all other terms are as defined above.

The transient solution of the above differential equations, including the transient decay heat contribution and the temperature response, is performed using the Holtec QA validated computer program MULPOOLD [6]. This evaluation is performed for a number of different discharge scenarios, which are described later in this report.

2.2 Minimum Time-to-Boil After a Loss of Forced Cooling

The second of the two differential equations presented in Section 2.1 is used to determine the time-to-boil. The RHR Supplemental Fuel Pool Cooling mass flow rate is set to zero and the initial temperature is set as the maximum calculated bulk SFP temperature. This reduced differential equation is:

$$C_{SFP} \times \frac{\partial T}{\partial \tau} = Q_{SFP}(\tau) - Q_{ENV}(T_{SFP})$$

where τ is now the time after loss of forced cooling and the initial condition is given as:

$$T_0 = T_{bulk,\max}$$

In addition, this scenario is assumed to occur *both with and without a* coincident failure of the seals on the gate *that* separates the SFP from the cask pit. Thus, *for the scenarios with the coincident gate failure,* the value of C_{SFP} is reduced to consider this considerable loss of water. The boiloff rate once 212°F has been reached can be calculated by dividing the net decay heat by the latent heat of vaporization of water.

This evaluation is performed iteratively, to determine the maximum time available, after loss of cooling, to provide makeup water to prevent the water level from dropping to the active fuel

height. This evaluation is performed using the Holtec QA Validated computer program TBOIL [7]. The version of TBOIL used for this analysis permits a curve-fit decay heat function to be specified. Decay heat functions are fit to the decay heat versus time results of the maximum SFP temperature analyses described in Section 2.1. This evaluation is performed for each discharge scenario as described later in this report.

2.3 Local Water and Fuel Cladding Temperatures

The decay heat generated by the fuel assemblies stored in the SFP induces a buoyancy driven flow field upward through the fuel rack cells. Cooler water is supplied to the bottom of the racks cells through the rack-to-wall gaps and rack-to-floor plenum. The presence of strong buoyancy forces results in a coupling of the SFP flow and temperature fields. As with any buoyancy dominated flow field, local effects can effect the global flow field. Quantification of the coupled flow and temperature fields can only be accomplished through the use of a computational fluid dynamics (CFD) analysis.

Other local temperature analysis techniques do exist, and have been used previously for evaluating the DAEC SFP and FPCCU for reracking [1]. The inherent simplifications in these other methods render them extremely overconservative. These oversimplified methods can be replaced by state of the art techniques, which yield more realistic results while remaining conservative. The CFD analysis methodology is outlined in the following paragraphs.

From a fluid flow modeling standpoint, there are two regions to be considered. One region is the bulk SFP region where the classical Navier-Stokes equations are solved with turbulence effects included. The other region is the heat generating fuel assemblies located in the spent fuel racks sitting near the bottom of the SFP. This second region is modeled as a porous solid region in which fluid flow is governed by Darcy's Law:

$$\frac{\partial P}{\partial X_i} = -\frac{\mu}{K(i)} \times V_i - C \times \rho \times |V| \times \frac{V_i}{2}$$

where $\partial P/\partial X_i$ is the pressure gradient, K(i), V_i and C are the corresponding permeability, velocity and inertial resistance parameters and μ is the fluid viscosity.

The CFD analysis will be performed on an industry standard FLUENT [8] fluid flow and heat transfer modeling program. The FLUENT code enables buoyancy flow and turbulence effects to be included in the CFD analysis. Turbulence effects are modeled by relating time-varying "Reynolds's Stresses" to the mean bulk flow quantities with the following turbulence modeling options:

i. k-ε Model
ii. RNG k-ε Model
iii. Reynolds Stress Model

The k-ε model is a time-tested, general purpose turbulence model. The RNG and Reynolds Stress models are newer models developed for applications where the k-ε model gives unacceptable

results, such a high speed flows and flows of highly viscous fluids. The k-ε modeling option is used for the DAEC local water temperature analysis.

Rigorous modeling of fluid flow problems requires a solution to the classical Navier-Stokes equations of fluid motion [9]. The equations (in modified form for turbulent flows with buoyancy effects included) are provided below:

$$\frac{\partial \rho_o u_i}{\partial t} + \frac{\partial \rho_o u_i u_j}{\partial X_i} = \frac{\partial}{\partial X_j}\left[\mu\left(\frac{\partial u_i}{\partial X_j} + \frac{\partial u_j}{\partial X_i}\right)\right] - \frac{\partial P}{\partial X_i} - (\rho - \rho_o) g_i + \frac{\partial \rho_o \langle u_i' u_j' \rangle}{\partial X_j}$$

where u_i are the three time-averaged velocity components. $\rho \langle u'_i u'_j \rangle$ are time-averaged Reynolds stresses derived from the turbulence induced fluctuating velocity components u'_i, P is the static pressure head, ρ_o is the fluid density at temperature T_o, β is the coefficient of thermal expansion, μ is the fluid viscosity, g_i are the components of gravitational acceleration and x_j are the Cartesian coordinate directions. The Reynolds stress tensor is expressed in terms of the mean flow quantities by defining a turbulent viscosity μ_t and a turbulent velocity scale $k^{1/2}$ as shown below [10]:

$$\rho \langle u_i' u_j' \rangle = \frac{2}{3}\rho \left(k^{1/2}\right)^2 \delta_{ij} - \mu_t \left[\frac{\partial u_i}{\partial X_j} + \frac{\partial u_j}{\partial X_i}\right]$$

The procedure to obtain the turbulent viscosity and velocity length scales involves a solution of two additional transport equations for kinetic energy (k) and rate of energy dissipation (ε). This methodology, known as the k-ε model for turbulent flows, is described by Launder and Spalding [11].

Once the spatial temperature distribution in the SFP is obtained, the fuel clad to local water temperature difference (T_{CL}), also called the clad superheat, is calculated from principles of laminar flow heat transfer. The Nusselt number for laminar flow with a constant heat rate can be obtained from various heat transfer handbooks. The following correlation for Nusselt number can then be solved for the convective heat transfer coefficient (h).

$$Nu = \frac{h \times D_h}{k}$$

where:

h = Convective heat transfer coefficient, Btu/(hr×ft^2×°F)
D_h = Hydraulic diameter of fuel assembly, ft
k = Thermal conductivity of water, Btu/(hr×ft^2×°F)

Once the heat transfer coefficient is known, the overall heat transfer coefficient (U), which includes the resistance of any crud on the rod surfaces can be calculated as:

$$U = \left(\frac{1}{h} + R_{crud} \right)^{-1}$$

where R_{crud} is the crud heat transfer resistance, (hr×ft^2×°F)/Btu. The clad superheat can now be calculated using the relationship:

$$T_{CL} = \frac{q_{MAX}}{U}$$

where q_{MAX} is the maximum heat flux (including peaking), Btu/(hr×ft^2).

3.0 ACCEPTANCE CRITERIA

The following acceptance criteria are applied to the analyses performed in this report.

1. The bulk SFP temperature must not exceed the following maximum temperature limits:

Discharge Type	# of SFPCS Trains Prior to RHR Supplemental Fuel Pool Cooling	Temperature Limit
Planned Full Core	2	180°F
Planned Full Core	1	180°F
Unplanned Full Core	2	180°F

The 180°F temperature limit is consistent with the previously performed thermal-hydraulic analyses performed to support the DAEC reracking license amendment request [1]. This temperature limit is selected primarily to provide a large margin against bulk SFP boiling.

2. Localized boiling is not permitted for any scenario where forced cooling is available.

4.0 ASSUMPTIONS

Several conservative assumptions are required to perform the analyses documented in this report. In addition to those assumptions set forth in the previously performed work [1] the following assumptions are used:

- The minimum in-core hold time is the time between the loss of criticality in the reactor and the transfer of the first fuel assembly from the reactor into the SFP.

- A single RHR system cooling train is in operation throughout the entire transient thermal analysis. The RHR Supplemental Fuel Pool Cooling diverts flow from this RHR train only.

- Prior to aligning RHR Supplemental Fuel Pool Cooling, the SFP is cooled by either one or two FPCCU trains. Subsequent to RHR Supplemental Fuel Pool Cooling, the FPCCU is not operating.

- The transfer canal gates are open and water is able to flow through the canal to the flooded reactor cavity. This allows the water level in the SFP, the reactor cavity and the skimmer surge tanks to be maintained.

- The RHR Supplemental Fuel Pool Cooling is not performed until the bulk SFP temperature approaches 120°F. This allows sufficient time for operator action (the RHR Supplemental Fuel Pool Cooling is aligned manually) before temperature limits are exceeded.

- In the unplanned full core discharge scenario, the recently loaded fresh fuel is assumed to have undergone 45 days of exposure at a capacity factor of 100%. The remaining fuel assemblies in the core are assumed to have undergone two full cycles plus 45 days or one full cycle plus 45 days of exposure, depending on whether the fuel had been once- or twice-burned prior to reactor restart.

- The thermal capacity of the reactor cavity conservatively neglects the volume of the water in the reactor vessel itself. This conservatively minimizes the reactor cavity thermal capacity, thereby minimizing the time-to-boil and maximizing bulk temperatures.

- The loss of forced cooling of the SFP occurs at the maximum calculated bulk SFP temperatures. This conservatively minimizes the times-to-boil.

- All fuel assemblies discharged to the SFP through Cycle 17 have the maximum burnup of 41,400 MWd/MTU. All fuel assemblies discharged to the SFP in Cycle 18 and thereafter have the maximum burnup of 50,000 MWd/MTU. This conservatively maximizes the decay heat load associated with these assemblies.

- *Bounding* radial and axial peaking are included in the local water and fuel cladding temperature analysis. This conservatively increases the decay heat generation rates used to perform these calculations.

- The local water and fuel cladding temperatures analysis uses the geometry of a GE-14 fuel assembly. Of all the fuel assembly types stored in the DAEC pool, the GE-14 fuel assembly *has* the limiting hydraulic resistance. Future fuel assemblies will be the GE-14 type. This will conservatively maximize the rack cells hydraulic resistance and resulting local temperatures.

- In the local water and fuel cladding temperatures analysis, every storage location in the SFP is assigned the resistance of a cell above a rack support pedestal. These pedestal cells have a higher resistance to flow as a result of the pedestal blocking the primary baseplate flow hole. This will conservatively maximize the rack cells hydraulic resistance and resulting local temperatures.

- The local water temperature (CFD) analysis incorporated the effects of blocking the east and west rack-to-wall downcomers. This blocks approximately 2/3 of the total downcomer area. As the total upcomer flow area through the assemblies in the rack cells is much greater than the total downcomer flow area, blockage of the downcomers is much more severe than and equivalent percentage blockage of any cell outlet regions. This results in a local water temperature evaluation that bounds any realistic blockage scenario.

- *The local water temperature (CFD) analysis is performed utilizing a SFP bulk temperature considerably (approximately 10.5°F) greater than the highest calculated value. This conservatively maximizes the resulting local water and fuel cladding temperatures.*

- A fuel rod crud layer with a heat transfer resistance of 0.0005 (hr×°F)/Btu is applied to all rod surfaces in the fuel cladding temperature analysis. This conservatively maximizes the fuel clad surface temperature.

- *The local water temperature (CFD) analysis is performed utilizing a bounding minimum rack-to-floor flow space (bottom plenum) height. The actual minimum bottom plenum height is several inches larger. This will conservatively maximize the plenum flow resistance and resulting local temperatures.*

5.0 INPUT DATA

5.1 Maximum SFP Temperatures and In-Core Hold Time Requirements

The input data and corresponding references for the calculations required to determine the maximum bulk SFP temperatures and in-core hold time requirements are:

PARAMETER	VALUE	SOURCE
SFP Storage Capacity (# of assemblies)	3152	Reference *23*
Fuel Assembly Discharge Schedule	see Table 1	Reference *24*
Fuel Maximum Average Exposure Cycle 1 through Cycle 17 *(see Item 9 in Section 4.0)* Cycle 18 and thereafter	 41,400 MWd/MTU 50,000 MWd/MTU	 Reference *24* Reference *25*
Coolant Flow Rate per FPCCU Heat Exchanger	397,000 lb/hr	Reference *26*
Coolant Temperature to FPCCU Heat Exchangers	95°F	Reference *26*
Coolant Flow Rate per RHR Heat Exchanger	2,*034*,000 lb/hr	Reference *27*
Coolant Temperature to RHR Heat Exchangers	85°F	Reference *27*
Number of Assemblies in Reactor Core	368 assemblies	Reference *28*
Reactor Thermal Power (102% of actual) Cycle 1 through Cycle 17 Cycle 18 and thereafter	 1691 MWt 1950 MWt	 Reference *29* Reference *42*
Minimum In-Core Hold Time	60 hr	Reference *30*
Fuel Assembly Transfer Rate	6 per hr	Reference *31*
SFP Thermal Capacity	1,320,000 Btu/°F	Reference *31*
Reactor Cavity Water Volume	1*36*,000 gallons	Reference *32*
Flow Rate through RHR Supplemental Fuel Pool Cooling to SFP	*2,000* gpm	Reference *33*
Flow Rate from Reactor Cavity to RHR	4,800 gpm	Reference *27*
SFP Surface Area	800 ft^2	Reference *32*
Maximum SFP Building Ambient Temperature	100°F	Reference *31*

5.2 Minimum Time-to-Boil After a Loss of Forced Cooling

The input data and corresponding references (in addition to any listed in Section 5.1) for the calculations required to determine the time-to-boil without forced cooling are:

PARAMETER	VALUE	SOURCE
Maximum Makeup Water Flow Rate	*59.5* gpm	Reference *34*
Makeup Water Temperature	90°F	Reference *35*
Specific Volume of Water at 212°F	0.01672 ft^3/lb	Reference 15
Specific Heat Capacity of Water at 212°F	1.0 Btu/(lb×°F)	Reference 15
Fuel Storage Rack Height	*15.146* ft	References *37, 38*
SFP Minimum Normal Water Depth	*36 ft*	*Reference 41*
Top of Active Fuel Height above SFP Floor	*13.96 ft*	*Refs. 24, 37, 38*
SFP Water Volume after Gate Seal Failure	6,955 ft^3	Reference *31*

5.3 Local Water and Fuel Cladding Temperatures

The input data and corresponding references (in addition to any listed in Sections 5.1 and 5.2) for the calculations required to determine the local water and fuel cladding temperatures are:

PARAMETER	VALUE	SOURCE
Maximum Fuel Assembly Burnup	*56,000* MWd/MTU	Reference *24*
Bounding Axial Peaking Factor	2.0	Reference *40*
Fuel Rod Outer Diameter (GE-14)	0.404 in	Reference *24*
Fuel Assembly Rod Pitch (GE-14)	0.510 in	Reference *43*
Rack Cell Inner Dimension	5.90 in	Reference *36*
Fuel Assembly Active Fuel Length	150 in	Reference *24*
Number of Fuel Rods per Assembly (GE-14)	92	Reference *24*
Bounding Support Grid Thickness	0.02 in	Reference *24*
Rack Cell-to-Cell Pitch	6.06 in	Reference *36*
Fuel Rack Cell Length	169 in	Reference *37*
Bounding Minimum Bottom Plenum Height	4.5 in	References *37, 38*
Bounding Rack-to-North Wall Gap	3.0 in	Reference *23*
Bounding Rack-to-South Wall Gap	2.0 in	Reference *39*

6.0 CALCULATIONS

6.1 Maximum Spent Fuel Pool Temperatures

Three separate discharge and cooling scenarios are evaluated in these calculations. These three scenarios are:

Case A: Planned Full Core Discharge, One FPCCU Train Before Aligning RHR Supplemental Fuel Pool Cooling

Case B: Planned Full Core Discharge, Two SFPCS Trains Before Aligning RHR Supplemental Fuel Pool Cooling

Case C: Unplanned Core Discharge, Two SFPCS Trains Before Aligning RHR Supplemental Fuel Pool Cooling

The unplanned full core discharge (Case C) scenario consists of a normal refueling outage with a length of 36 days followed by 45 days of full power operation and a subsequent unplanned transfer of the entire core to the SFP.

The three cases analyzed evaluate the control and removal of decay heat generated by full core discharges. During a core shuffle discharge, the rate of heat transfer to the SFP as well as the total decay heat transferred to the SFP would be less than would be expected for a full core discharge. It is not anticipated, therefore, that initiating the RHR Supplemental Fuel Pool Cooling would be required. However, the RHR Supplemental Fuel Pool Cooling could be initiated if it was determined that the FPCCU system was not adequately removing decay heat or if the FPCCU system was removed from service (i.e. maintenance). The decay heat removal capacity of RHR Supplemental Fuel Pool Cooling is larger than that of the FPCCU system, and would therefore be more than adequate to remove the SFP heat generated during a core shuffle.

As stated in the assumptions (Section 4.0), the SFPCS is used to cool the SFP until the bulk SFP temperature reaches 120°F. Once RHR Supplemental Fuel Pool Cooling is aligned, FPCCU operation is discontinued.

These calculations are performed, using the analysis methodologies described in Section 2.1 of this report, in Appendix B. A listing of the computer files for these calculations is included on page B-2.

The STER output reports for the FPCCU and RHR heat exchanger models are included on pages B-3 through B-9 and B-10 through B-16, respectively. The temperature effectiveness calculations for these units are included on pages B-17 through B-20. The DECOR input files are included on pages B-21 and B-22. The MULPOOLD input files are included on page B-23.

6.2 Minimum Time-to-Boil After a Loss of Forced Cooling

The minimum time between the loss of forced cooling and the onset of bulk SFP boiling is calculated for the limiting case of each of the scenarios evaluated in Section 6.1, *both with and without a coincident cask pit gate seal failure*. The initial bulk SFP temperature for each time-to-boil scenario is the maximum temperature for each discharge/cooling scenario. Cases 1 and 2 are identical from the time-to-boil standpoint since their maximum bulk SFP temperatures *are the same*, and a single evaluation is performed for both of these cases.

These calculations are performed, using the analysis methodology described in Section 2.2 of this report, in Appendix C. A listing of the computer files for these calculations is included on page C-2. The TBOIL input files are included on page C-3. The spreadsheet used to generate the curve-fit decay heat functions is included on pages C-4 through C-12.

6.3 Local Water and Fuel Cladding Temperatures

The maximum local water and fuel cladding temperatures are calculated for a limiting (i.e. highest decay heat and *bounding* bulk temperature) scenario evaluated in Section 6.1. The net decay heat generation for this bounding scenario must therefore bound the calculated net SFP heat load coincident with the maximum temperature for the appropriate discharge/cooling scenario.

These calculations are performed, using the analysis methodology described in Section 2.3 of this report, in Appendix D. A listing of the computer files for these calculations is included on page D-2. The effective porous media properties calculation is included on pages D-3 through D-5. The calculations to determine the decay heats of each full-core zone (i.e., hottest reload batch and balance of core) are included on pages D-6 through D-8. Calculations of miscellaneous FLUENT model input values (volumetric heat rates, SFPCS inlet velocity and temperature, etc.) are included on pages D-9 through D-11. The fuel cladding superheat calculation is included on pages D-12 and D-13.

7.0 RESULTS AND CONCLUSIONS

7.1 Maximum Spent Fuel Pool Temperatures

These evaluations were performed as described in Section 6.1. The results of these calculations are summarized in the following table.

Scenario ID	Maximum Bulk SFP Temperature @ Time	Coincident Net SFP Heat Load	Coincident SFP Environmental Losses
Case A	*152.50*°F @ 12*3* hrs	23.87×10^6 Btu/hr	*0.894*×10^6 Btu/hr
Case B	*152.50*°F @ 12*3* hrs	23.87×10^6 Btu/hr	*0.894*×10^6 Btu/hr
Case C	*152.46*°F @ 122 hrs	23.*92*×10^6 Btu/hr	*0.892*×10^6 Btu/hr

Note that the results of Case A and Case B are identical. This demonstrates that the effects of a limited amount of FPCCU cooling available before RHR Supplemental Fuel Pool Cooling is rapidly overcome by the large cooling capacity of the RHR system. Figures 1 through 3 contain temperature versus time profiles for these three cases. Figures 4 through 6 contain the corresponding net decay heat and environmental losses versus time profiles.

The temperature at time zero for each of the temperature profiles (Figures 1 through 3) is an initial guess expected to be at or near the steady-state temperature. The slight rise seen on each figure is the result of the initial guess being somewhat lower than the actual steady-state temperature value. This had no effect on the results of the analyses, as the temperature converges to the steady-state temperature.

The fuel transfer from the reactor to the SFP begins after 60 hours of in-core hold time. This point is designated as "A" on each of the three temperature profile figures. The temperature begins to rise rapidly as the fuel transfer starts, due to the large thermal power of the freshly discharged assemblies. As the bulk SFP temperature approaches 120°F (point "B" on the three temperature profile figures), the FPCCU system is shut off and RHR Supplemental Fuel Pool Cooling is aligned.

The bulk SFP temperature continues to rise as fuel is transferred from the reactor to the SFP. The maximum decay heat load in the SFP is obtained a*t about* 122 hours after shutdown, designated as "C" on each of the three temperature profile figures. The peak bulk SFP temperature occurs slightly after the peak decay heat load is reached, due to the finite thermal capacity of the water in the SFP. The magnitude of this temperature lag is approximately three hours for all cases.

Examining Figures 1 through 3, it is observed that the minimum time required for the SFP temperature to increase from 140°F to 150°F is approximately *25* hours. This corresponds to an average temperature rise of approximately *0.4*°F per hour. This observation leads to the conclusion that sufficient time exists for supplemental cooling to be provided, if necessary, to meet plant procedural limits.

7.2 Minimum Time-to-Boil After a Loss of Forced Cooling

The minimum time-to-boil following a loss of forced cooling were calculated for the limiting cases for the three scenarios described in Section 6.2. The results of these calculations are summarized in the following table.

Scenario ID	Initial Temp.	Time-to-Boil	Max. Boiling Water Loss	Time to Provide Makeup Water
Without Coincident Cask Pit Gate Seal Failure				
Cases A & B	*152.50°F*	*3.34 hrs*	*52.71 gpm*	*45 hrs*
Case C	*152.46°F*	*3.40 hrs*	*51.90 gpm*	*46 hrs*
With Coincident Cask Pit Gate Seal Failure				
Cases A & B	152.50°F	*1.07* hrs	*53.05* gpm	*3.8* hrs
Case C	152.46°F	*1.09* hrs	*52.23* gpm	*3.9* hrs

7.3 Local Water and Fuel Cladding Temperatures

The maximum local water and fuel cladding temperatures were calculated for the scenario described in Section 6.2. The results of these calculations are summarized in the following table.

Item ID	Parameter	Value
A	Calculated Maximum Local Water Temperature	177.53°F
B	Calculated Maximum Fuel Clad Superheat	33.75°F
C	**Bounding Fuel Clad Temperature (A+B)**	**211.28°F**

The top of the fuel storage racks are more than 23 feet below the normal water level. The local saturation temperature at this depth is approximately 239.29°F. Thus, a substantial margin against nucleate boiling exists on the fuel rod surfaces and in the fuel rack cells. As stated in Section 4.0, the CFD model used to compute the local water temperature assumes *a bounding high SFP bulk temperature and* that both the east and west downcomer regions are completely blocked. Because of the *elevated bulk temperature and* extreme water supply pathway blockage, the calculated local water temperature (Item C) bounds that for any realistic *SFP bulk temperature and any realistic* cell or plenum blockage scenario.

Figure 7 presents a physical grid plot of the two-dimensional FLUENT model. Figures 8 and 9, respectively, present converged temperature contours and flow vectors from the FLUENT model.

8.0 REFERENCES

[1] "Thermal-Hydraulic Analysis of Duane Arnold Energy Center Spent Fuel Pool with Maximum Density Storage", Holtec Report HI-93972, Revision 3.

[2] "QA Documentation for DECOR v1.0", Holtec Report HI-971734, Revision 0.

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[4] A.G. Croff, "ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code," ORNL-5621, Oak Ridge National Laboratory, 1980.

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[6] "QA Documentation for MULPOOLD v2.0", Holtec Report HI-92834, Revision 2.

[7] "QA Documentation for TBOIL v1.9", Holtec Report HI-92832, Revision 5.

[8] "QA Validation of FLUENT Version 5.1", Holtec Report HI-992276, Revision 0.

[9] Batchelor, G.K., "An Introduction to Fluid Dynamics", Cambridge Univ. Press, 1967.

[10] Hinze, J.O., "Turbulence", McGraw Hill Publishing Co., New York, NY, 1975.

[11] Launder, B.E., and Spalding, D.B., "Lectures in Mathematical Models of Turbulence", Academic Press, London, 1972.

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[14] *Deleted.*

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[17] *Deleted.*

[18] *Deleted.*

[19] *Deleted.*

[20] *Deleted.*

[21] *Deleted.*

[22] *"QA Documentation for STER v5.04," Holtec Report HI-92776, Revision 11.*

[23] *Duane Arnold Drawings M453-001, M453-005 and M453-006.*

[24] *Letter NG-00-1621 from M. Brandt to S. Kottenstette, dated 20 September 2000.*

[25] *"Task T0200: Reactor Core and Fuel Performance," General Electric Project Task Report GE-NE-A22-00100-03-01, Revision 0, April 2000.*

[26] *"Fuel Pool Heat Exchanger Technical Manual," DAEC Document APED-G41-2828-24.*

[27] *"Process Diagram - Residual Heat Removal System," DAEC Document APED-E11-008(1).*

[28] *"Top Guide," DAEC Document APED-B11-85(1).*

[29] *Modification Number CMP-16, Document No. OPL3/4, Rev. 0, 4 October 1999, "Operating Plant Limits," Index Item No. 3.01.*

[30] *"Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie," Document No. CAL-M97-019, Revision 1, 29 June 1998.*

[31] *"Thermal-Hydraulic Analysis of DAEC Spent Fuel Pool with Maximum Storage (Holtec Report HI-93972, Rev. 3)," Document No. CAL-M93-083, Revision 1, 14 July 1995.*

[32] *"Reactor Well, Spent Fuel and Dryer-Separator Pool General Arrangement," Bechtel Drawing C-492, Revision 6.*

[33] *"Process Diagram-Fuel Pool Cooling and Clean-Up System (Filter/Demin)," DAEC Document APED-G41-5-3.*

[34] *"Demonstration of the ESW Make-Up to the Spent Fuel Pool," Special Test Procedure 198, 25 September 1996.*

[35] *Duane Arnold Energy Center Unit 1 Updated Final Safety Analysis Report (UFSAR), Section 9.2.3.2.1.*

[36] *Duane Arnold Drawing M453-002.*

[37] *Duane Arnold Drawing M453-003.*

[38] *Duane Arnold Drawing M453-004.*

[39] *Duane Arnold Drawing M453-0016.*

[40] *Modification Number CMP-17, Document No. FC#403R59, Rev. 0, 30 June 2000, "Fuel Management Summary," Index Item No. 3.03.*

[41] *Letter NG-00-1794 from B. Hendrickson (Alliant) to E. Rosenbaum (Holtec), dated 23 October 2000.*

[42] *"Task T0100: Nominal Reactor Heat Balance," General Electric Project Task Report GE-NE-A22-00100-01-01, Revision 0, April 2000.*

[43] *Letter NG-00-1834 from M. Brandt (Alliant) to E. Rosenbaum (Holtec), dated 30 October 2000.*

Table 1

DAEC OPERATING DATA

Cycle No.	Discharge Date	# of Assemblies	Total Stored Assemblies
1A	6/1975	4	4
1B	2/1976	84	88
2	3/1977	100	188
3	3/1978	88	276
4	2/1980	88	364
5	3/1981	84	448
6	2/1983	128	576
7	2/1985	120	696
8	3/1987	128	824
9	9/1988	120	944
10	6/1990	104	1,048
11	2/1992	104	1,152
12	7/1993	128	1,280
13	2/1995	128	1,408
14	9/1996	120	1,528
15	5/1998	120	1,648
16	12/1999	128	1,776
17	6/2001	136	1,912
18	6/2003	152	2,064
19	6/2005	152	2,216
20	6/2007	152	2,368
21	6/2009	152	2,520
22	6/2011	152	2,672
23	6/2013	152	2,824
24	6/2015	368	3,192

Figure 1: SFP Bulk Temperature Profile - Case A

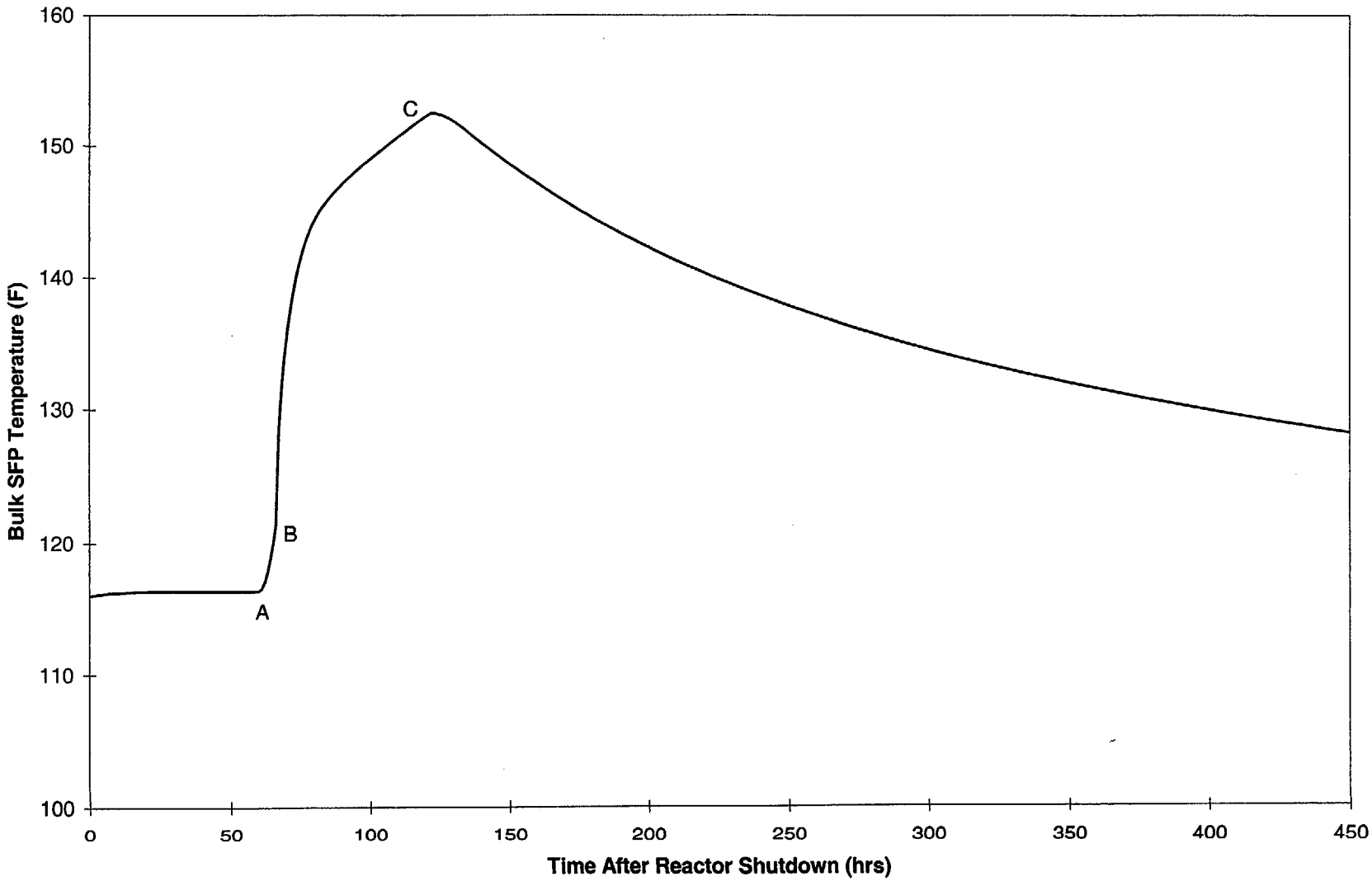


Figure 2: SFP Bulk Temperature Profile - Case B

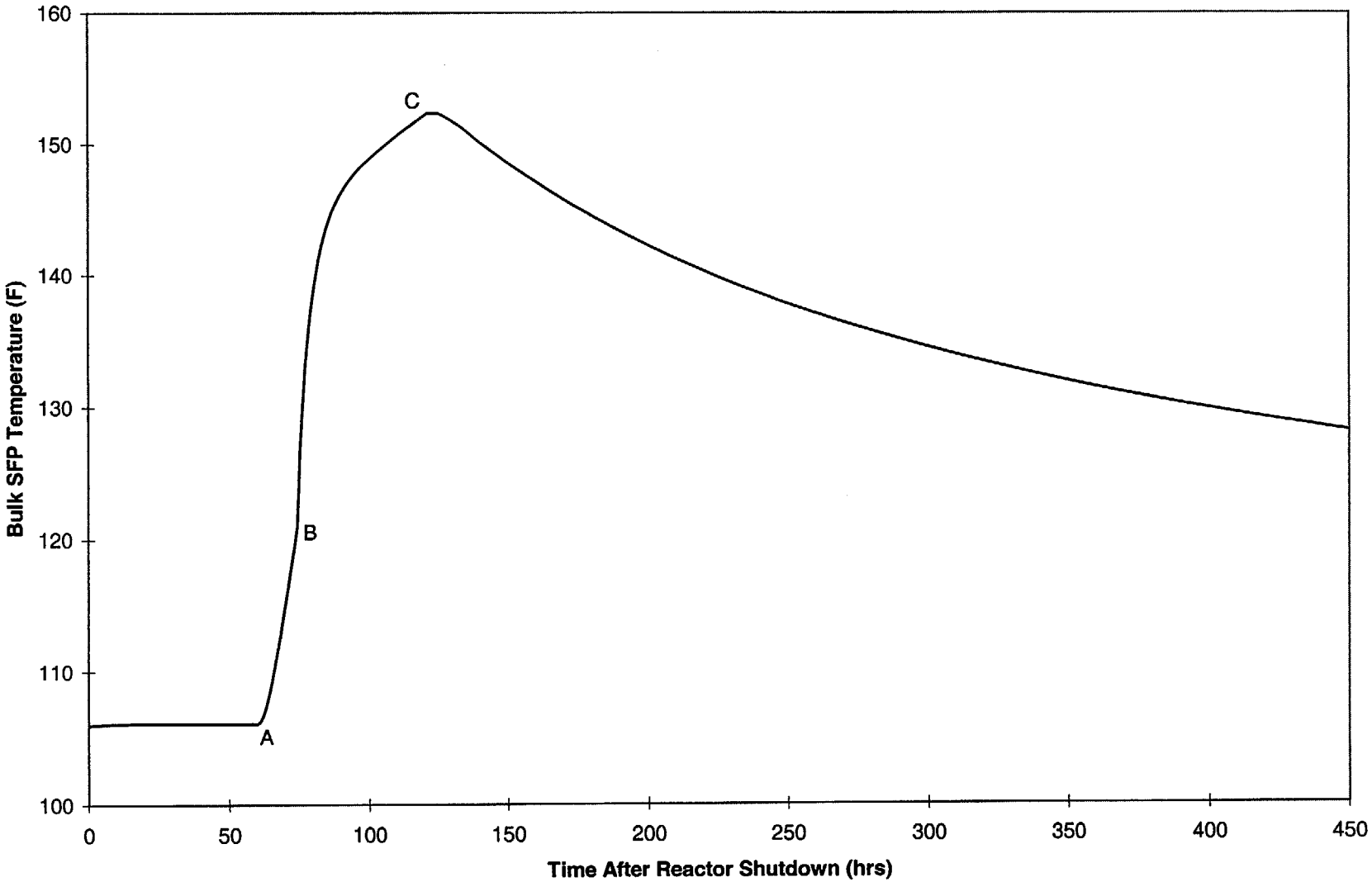


Figure 3: SFP Bulk Temperature Profile - Case C

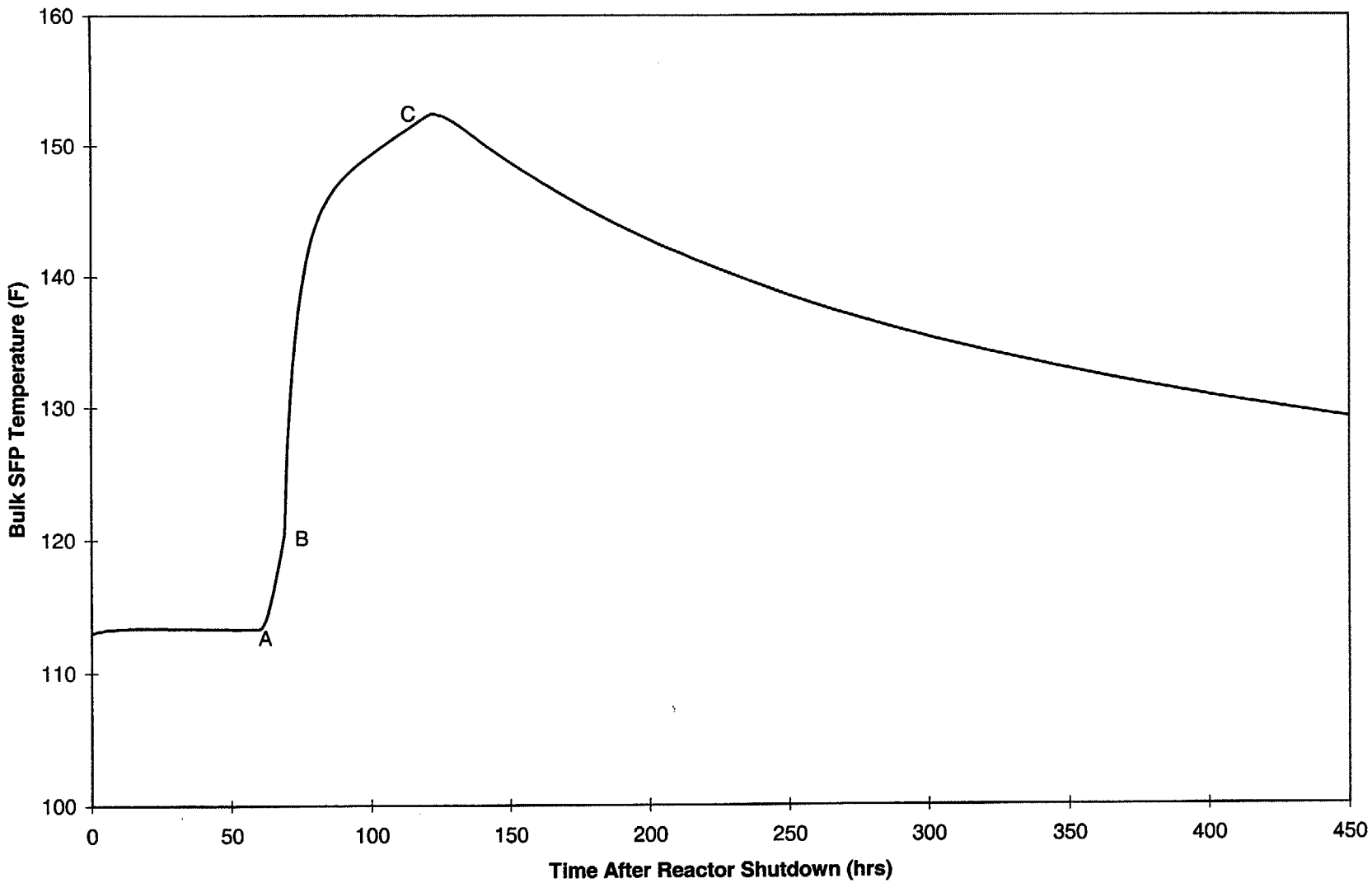


Figure 4: SFP Net Decay Heat Load and Heat Loss Profiles - Case A

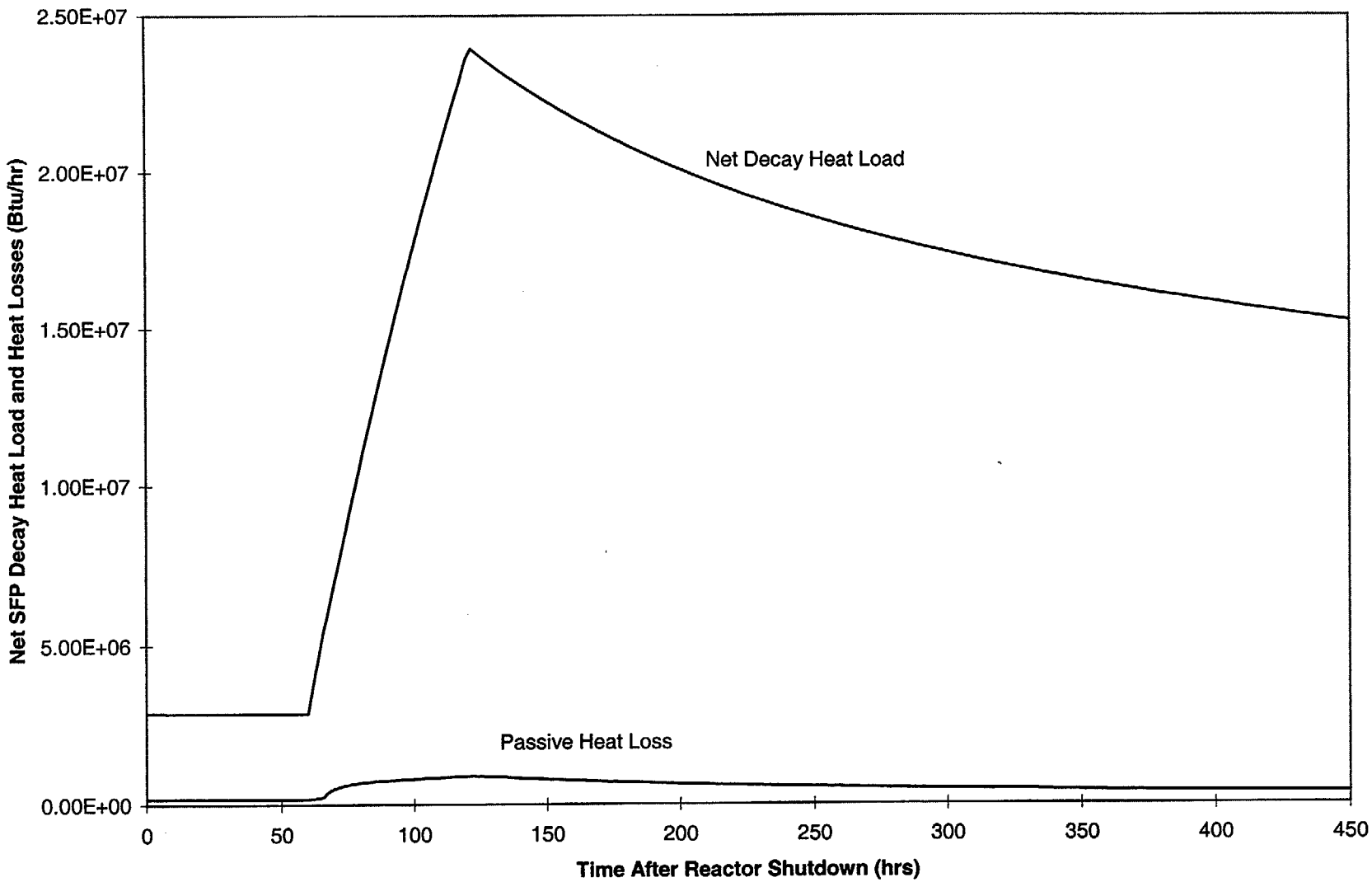


Figure 5: SFP Net Decay Heat Load and Heat Loss Profiles - Case B

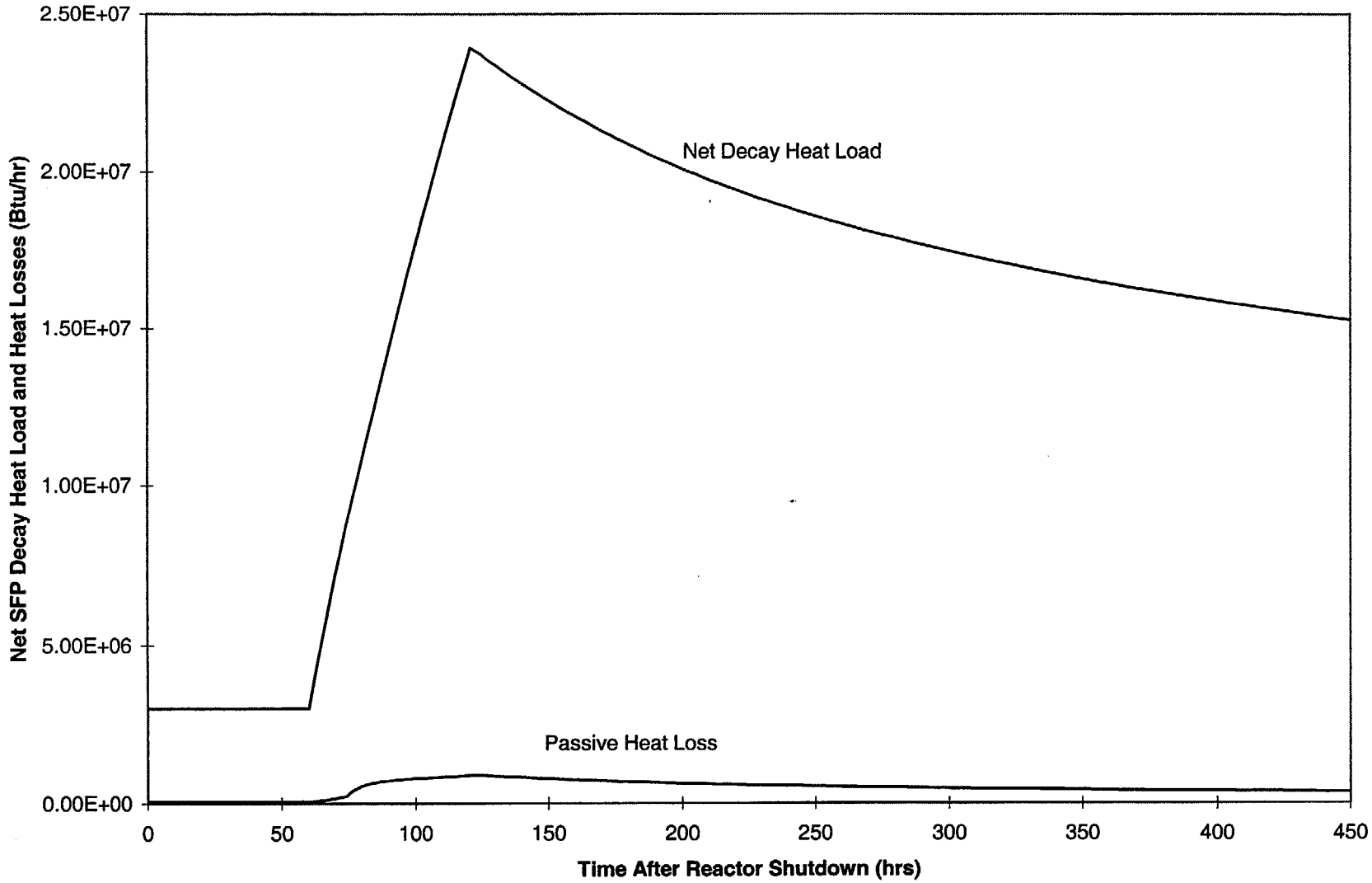


Figure 6: SFP Net Decay Heat Load and Heat Loss Profiles - Case C

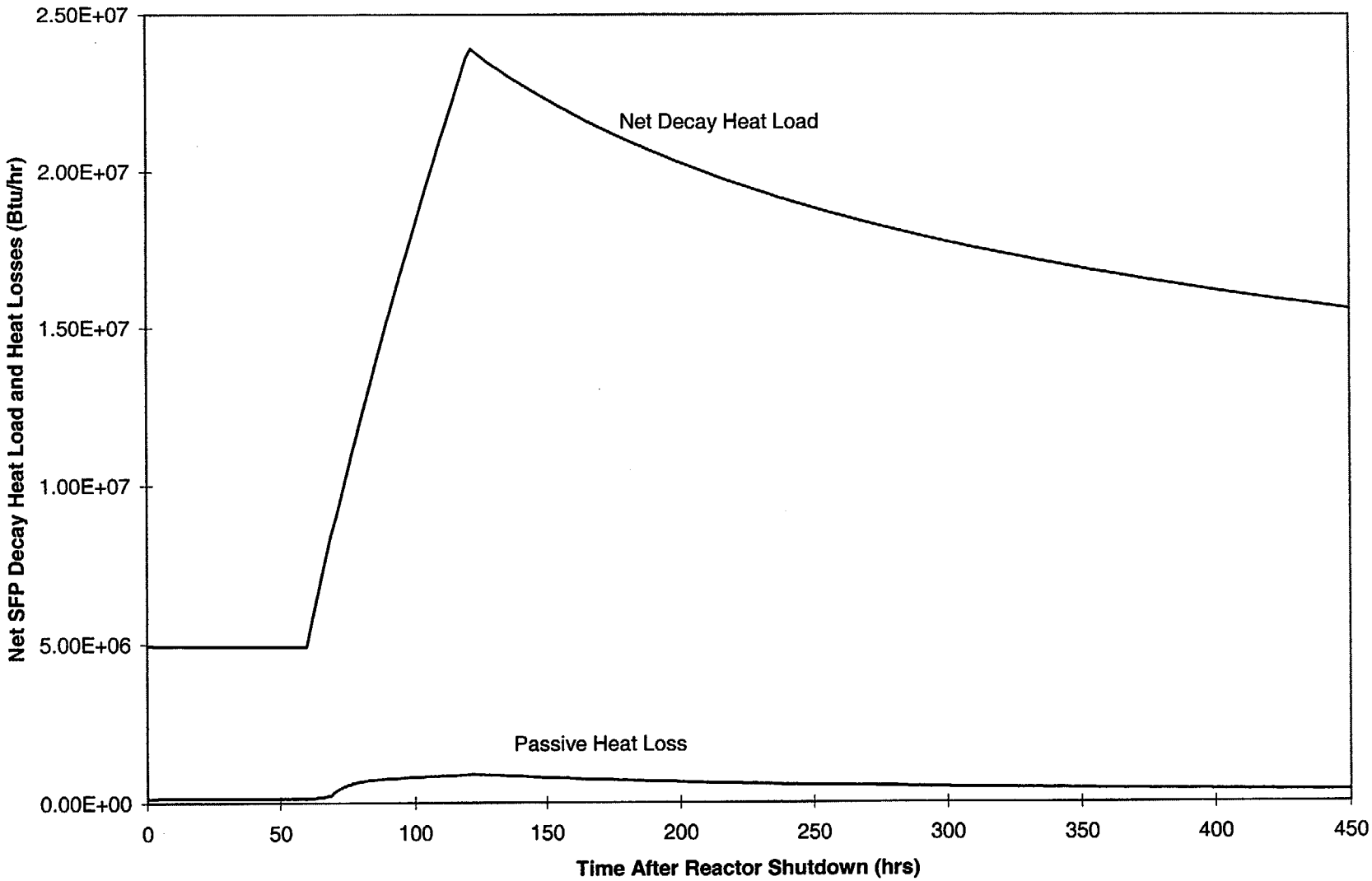
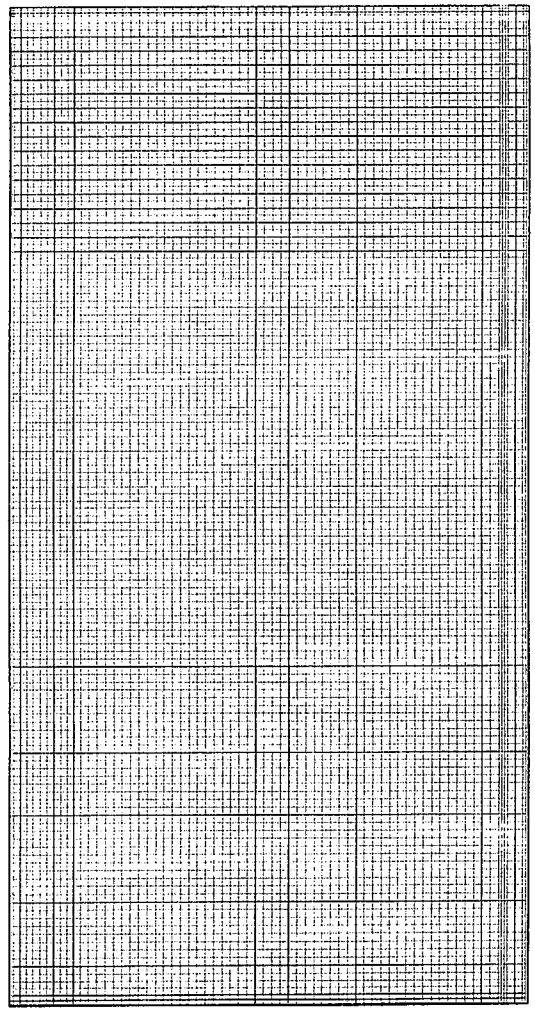


FIGURE 7: Local Temperature CFD Model - Physical Grid



Grid

Jul 11, 2000
FLUENT 5.1 (2d, segregated, ke)

FIGURE 8: Local Temperature CFD Model - Converged Temperature Contours

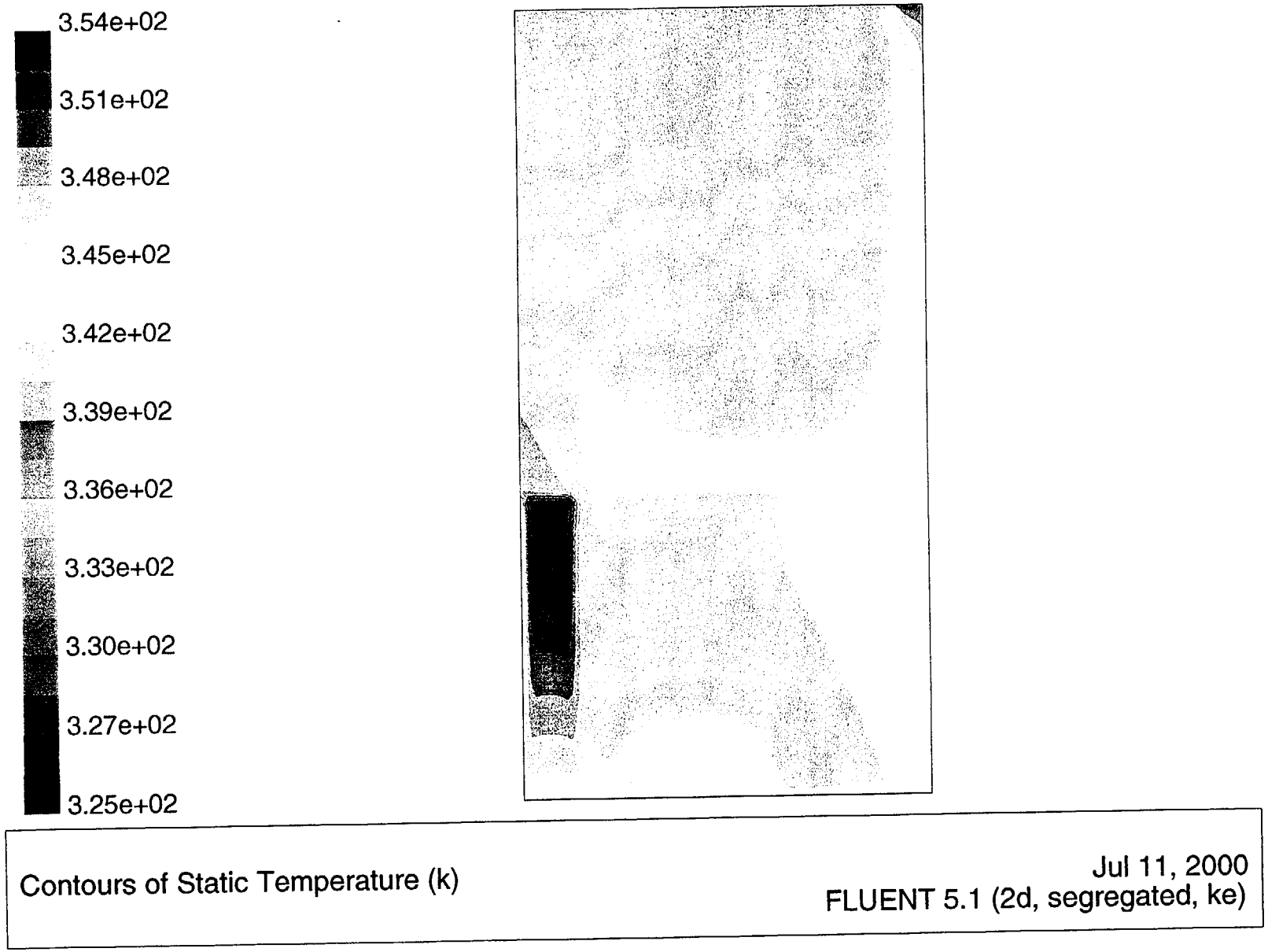
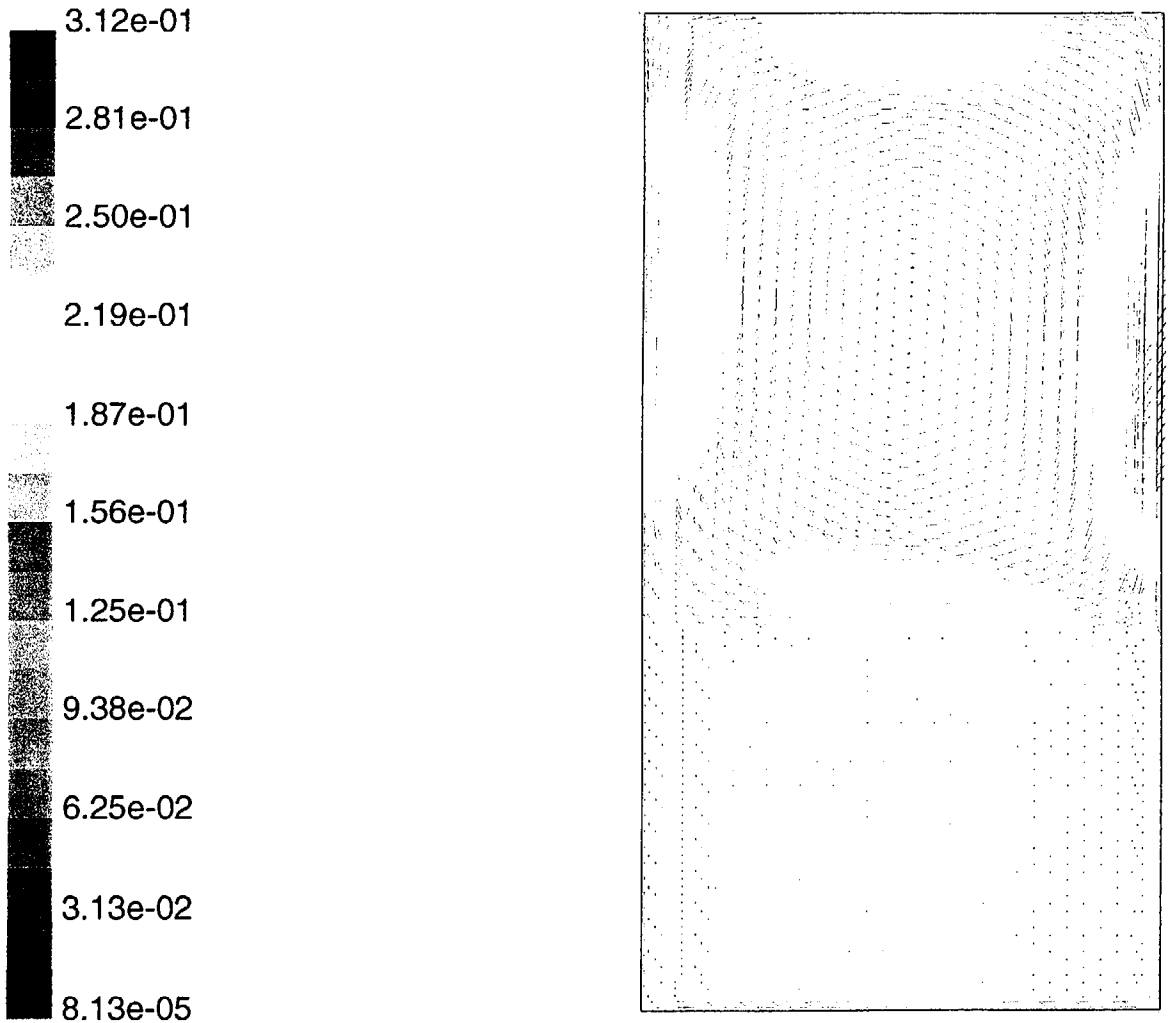


FIGURE 9: Local Temperature CFD Model - Converged Velocity Vectors



Velocity Vectors Colored By Velocity Magnitude (m/s)

Jul 11, 2000
FLUENT 5.1 (2d, segregated, ke)

APPENDIX A

Holtec QA Approved Computer Program Listing

HOLTEC APPROVED COMPUTER PROGRAM LIST				
PROGRAM(Category)	**VERSION**	**CERTIFIED USERS**	**CODE USED**	**Computer Environment**
ANSYS (A)	5.0, 5.0A, 5.1, 5.2, 5.3, 5.4, 5.6	JZ, EBR, IS, PKC, CWB, SPA, AIS, LT, IR. SP		
AC-XPERT	1.12			
AIRCOOL	5.01E, 5.01F, 5.02G, 5.11H, 5.2I, 6.1			
ANYSHEET	1.3			
AVESPC	1.0			
AXISOL	1.0			
BACKFILL	1.0, 2.0			
BONAMI (Scale)	4.3, 4.4			
BULKTEM	2.0, 3.0			
CASMO-4 (A)	I.13.04	ELR, SPA, DMM, KC, ST		
CASMO-3 (A)	4.4, 4.7	ELR, SPA, DMM, KC, ST		
CELLDAN	4.4.1			
CHANBP6 (A)	1.0	SJ, PKC, CWB, AIS, SP		
CHAP08 (CHAPLS10)	1.0			
CONPRO	1.0			
CORRE	1.3			
DECAY	1.4, 1.5			
DECOR	1.0		1.0	PC / DOS 6.0
DR.BEAMPRO	1.05			
DYNARACK (also known as DYNAMO XXXX) (A)	1.5 thru 1.13 inclusive	SJ, AIS, CWB, SP, PKC		
DYNAPOST	1.0, 2.0			
FIMPACT	1.0			
FLANGE	2.0			
FLUENT (A)	4.3, 4.32, 4.48, 5.1(see error notice), 4.2.8 (UNS)	EBR, IR, DMM, SPA	5.1	PC / NT 4.0
GENEQ	1.3			
HISPRAY	1.0			

HOLTEC APPROVED COMPUTER PROGRAM LIST				
PROGRAM(Category)	**VERSION**	**CERTIFIED USERS**	**CODE USED**	**Computer Environment**
HIVENT	1.0, 2.0			
HYSYST	1.01			
INSYST	2.01			
KENO-5A (A)	4.3, 4.4	ELR, SPA, DMM, KC, ST		
LONGOR	1.0			
LNSMTH2	1.0			
LS-DYNA3D (A)	936, 940, 950	JZ, IS, AIS, SPA, SP		
LUINVS	1.0			
MAXDIS16	1.0			
MCNP (A)	4A, 4B	ELR, SPA, KC, ST, DMM		
MASSINV	1.4, 1.5, 2.1			
MR2 (A)	1.4 THROUGH 1.9	AIS, SP, CWB, PKC		
MR216 (A)	1.0, 2.0	AIS, SP, CWB, PKC		
MR2 POST PROCESSORS	2.0			
MSREFINE	1.3, 2.1			
MULPOOLD	1.4, 1.3, 2.0, 2.1		2.0	PC / NT 4.0
MULTI1	1.3, 1.4, 1.5, 1.54, 1.55			
NITAWL (Scale)	4.3, 4.4			
ONEPOOL	1.4, 1.4.1, 1.5, 1.6			
ORIGEN	2.1			
ORIGENS (Scale)	4.3, 4.4			
PD16	1.1, 1.0, 2.0			
PIPE PLUS (A)	5.04-3H			
PREDYNA1	1.5, 1.4			
PREMULT2	1.0			
PSD1	1.0			
QAD	CGGP			

HOLTEC APPROVED COMPUTER PROGRAM LIST				
PROGRAM(Category)	**VERSION**	**CERTIFIED USERS**	**CODE USED**	**Computer Environment**
RELAP5/MOD3.2 (A)	3.2	RB		
RFPP	1.0			
SAS2H (Scale)	4.3, 4.4			
SCANS	1A			
SFMR2A	1.0			
SFMR2	1.1			
SIFATIG	1.0			
ST-XPERT	2.01			
SPG16	1.0, 2.0, 3.0			
STARDYNE (A)	4.4, 4.5	SP		
STER	3.12B, 3.22A, 3.22C, 3.24D, 3.3E, 4.12, 5.04		5.04	PC / Win 3.11
TBOIL	1.7, 1.9		1.9	PC / NT 4.0
THERPOOL	1.2, 1.2A			
TRIEL	2.0			
VERSUP	1.0			
VIB1DOF	1.0			
VMCHANGE	1.4, 1.3			
WEIGHT	1.0			
WORKING MODEL (A)	3.0 (2D); 4.0 (2D) 1.0 (3D); 6.0 (3D)	AIS		

NOTES:

1. XXXX = ALPHANUMERIC COMBINATION
2. GENERAL PURPOSES UTILITY CODES (MATHCAD, EXCEL, ETC.) MAYBE USED ANYTIME.