LICENSEE: Tennessee Valley Authority

FACILITY: Watts Bar Nuclear Station, Unit 1 January 16, 1997

SUBJECT: SUMMARY OF JANUARY 14, 1997 MEETING - SPENT FUEL POOL RERACKING

On January 14, 1997, Nuclear Regulatory Commission (NRC) and Tennessee Valley Authority (TVA) representatives met at the NRC office (Two White Flint North) to discuss TVA's application dated October 23, 1996. TVA proposes to replace the current spent fuel pool racks with racks previously used in the Sequoyah plant. A list of attendees is provided in Enclosure 1.

TVA made a presentation based on the handouts in Enclosure 2. TVA's presentation addressed the areas of the NRC staff review: reactor systems, plant systems, materials engineering, environmental, radiation protection and structural engineering.

The NRC staff has issued requests for additional information on November 7, 1996 (reactor systems) and January 2, 1997 (materials and environmental reviews). The staff identified additional needs for information as set forth in Enclosure 3 (plant systems and radiological protection). TVA agreed to provide docketed responses to these issues as well as to several other questions raised during the meeting. The staff will identify further needs for additional information in the Structural Engineering area.

Regarding the schedule, TVA indicated that, based on recent improvements in the schedule for Cycle 1, they would need an NRC staff decision on the application by early March 1997 to support the next refueling outage which is now scheduled to begin in early September. A timely decision supporting use of the replacement racks would result in (a) the generation of less radwaste (b) lower personnel exposure and (c) simpler heavy loads handling due to the rack changeouts.

The presentation by TVA and its contractor, Holtec, was very informative. The staff noted that an application of this complexity should have been submitted by TVA much further in advance of TVA's need for an NRC staff decision on the application than was the case for the reracking application. TVA indicated that it would respond expeditiously to the needs for additional information identified by the NRC staff.

Original signed by

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000390
P PDR

Docket No. 50-390

Enclosures: 1. Attendance List
2. TVA Handout
3. Request for Additional Information

cc w/Enclosure: See next page

DOCUMENT NAME: G:\WBN\MEETING.SUM

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DATE	01/16/97	01/15/97	01/16/97	01/ /97	01/ /97

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LIST OF PARTICIPANTS

NRC - TVA MEETING ON WATTS BAR UNIT 1

JANUARY 14, 1997

Name	Affilation
Bob Martin	NRC Project Manager
Larry Kopp	NRC Reactor Systems
Yong S. Kim	NRC/NRR/DE/ECGB
Kris Singh	Holtec Int.
Paul Pace	WBN Licensing Manager
Alan Soler	Holtec Int.
John Minns	NRC/DRPM/PERB
Rebecca Mays	TVA - WBN Licensing
David Shum	NRC/NRR/SPLB
Ron Hernan	NRC/DRPE
Bob Mullin	TVA - WBN Site Engineer
Richter Wiggall	TVA - System Engineering
James Adair	TVA - WBN Site Engineer
J. E. Maddox	TVA - WBN Engineer & Materials Manager
R. L. Rothman	NRC/ECGB
Claudia Craig	NRC/PGEB
Kris Parczewski	NRC/NRR/DE/EMCB
George Hubbard	NRC/NRR/DSSA/SPLB
Bob James	TVA - WBN Site Engineer/Civil Engineer
Rick Cutsinger	TVA - Chief Civil Engineer

ENCLOSURE 1

AGENDA
JANUARY 14, 1997

I.	INTRODUCTION AND PURPOSE OF MEETING	Pace
II.	PURPOSE AND SCOPE OF PROJECT	Maddox/Adair
III.	INSTALLATION CONSIDERATIONS	Mullin
IV.	THERMAL HYDRAULIC CONSIDERATIONS	Singh
V.	STRUCTURAL AND SEISMIC CONSIDERATIONS	Soler
VI.	SUMMARY AND CONCLUSION	Maddox/Adair

PURPOSE OF RERACK PROJECT

SCOPE OF PROJECT

PROJECT SCHEDULE

PURPOSE OF

WATTS BAR

RERACK PROJECT

1. INCREASE SPENT FUEL STORAGE CAPACITY

2. ELIMINATE BORAFLEX

BENEFITS OF RERACKING

BEFORE REFUELING OUTAGE

1. MINIMIZE GENERATION OF NUCLEAR WASTE

2. MINIMIZE PERSONNEL RADIATION EXPOSURE (ALARA)

3. NO MOVEMENT OF HEAVY LOADS NEAR SPENT FUEL

4. REPLACE BORAFLEX WITH BORAL

5. LOWER COST

SCOPE OF PROJECT

TRANSPORT RACKS TO WATTS BAR

REMOVE EXISTING RACKS

REMOVE PORTION OF SPARGER

MODIFY WALL OBSTRUCTIONS

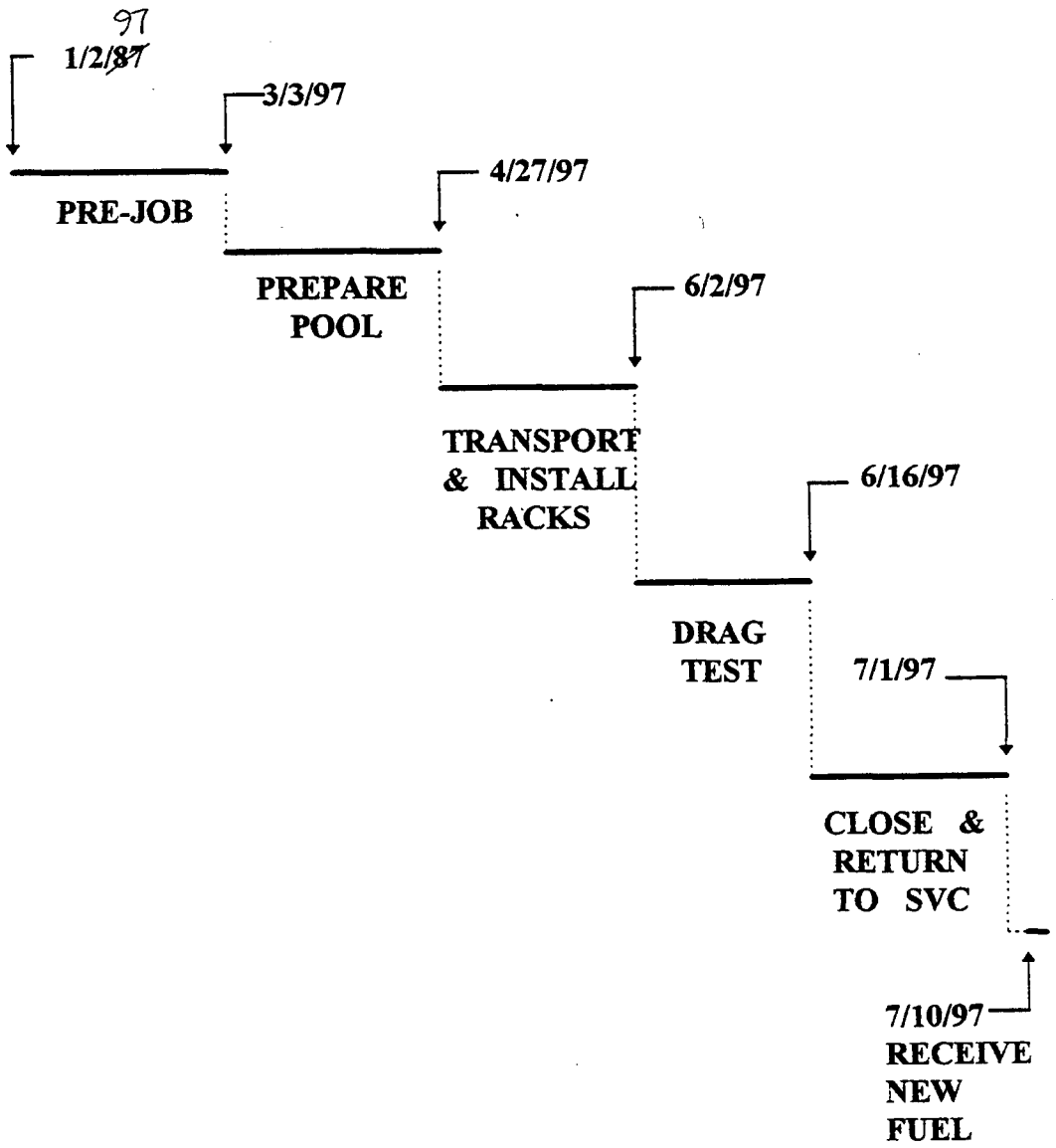
INSTALL NEW BEARING PLATES

INSTALL BABY RACK ATTACHMENTS

INSTALL REPLACEMENT RACKS

DRAG TEST 100% OF CELLS

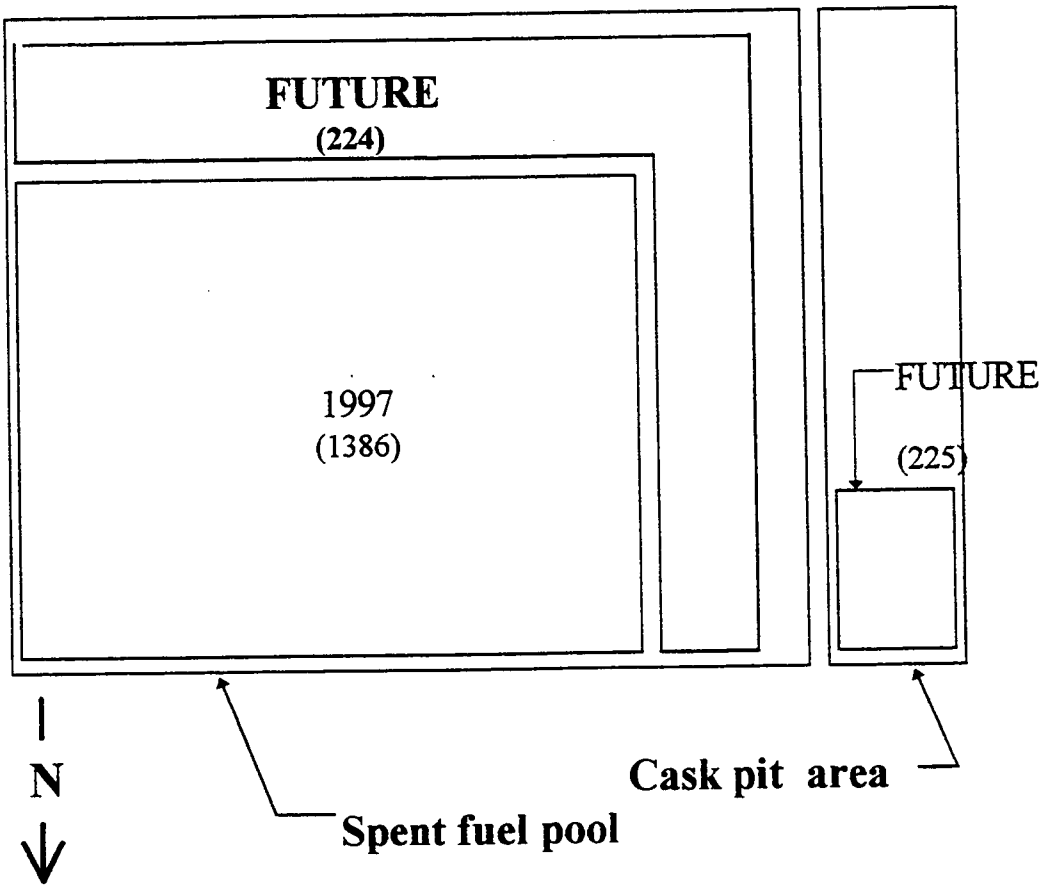
PROJECT SCHEDULE



INSTALLATION CONSIDERATIONS

- POOL LAYOUT AND RACK CONSTRUCTION
- CRITICALITY
- RADIOLOGICAL EVALUATION
- ENVIRONMENTAL ASSESSMENT

SCOPE OF RERACK PROJECT



RACK MODULE DATA, EXISTING AND PROPOSED RACKS

ITEM	EXISTING RACKS	PROPOSED RACKS	
	Watcher	PaR	Holtec
Number of Cells	1312 (484 usable)	1386	449
Number of Modules	16	24	11
Neutron Absorber	Boraflex	Boral	Boral
Nominal Cell Pitch, (inches)	10.72	10.375	8.972
Nominal Cell Envelope (inches)	8.99 x 8.99	8.75 x 8.75	8.75 x 8.75
Maximum Initial Enrichments (Weight % U-235)	3.50	5.0(1)	5.0(1)

RACK CONSTRUCTION MATERIALS

STRUCTURAL MATERIALS

THE FOLLOWING STRUCTURAL MATERIALS ARE UTILIZED IN THE FABRICATION OF THE BURNUP CREDIT RACKS.

A. ASME SA240-304L FOR SHEET METAL STOCK FOR THE STORAGE CELL STRUCTURES.

B. INTERNALLY THREADED SUPPORT LEGS: ASME SA240-304L.

C. EXTERNALLY THREADED SUPPORT SPINDLE: ASME SA564-630 PRECIPITATION HARDENED STAINLESS STEEL.

D. WELD MATERIAL PER THE FOLLOWING ASME SPECIFICATION: SFA 5.9 ER308L.

THE PAR FLUX TRAP RACKS BEING TRANSFERRED FROM SQN CONTAIN THE FOLLOWING PROVEN MATERIALS:

A. POISON CAN INNER AND OUTER TUBES: 304 STAINLESS STEEL, ASTM STANDARD A-666-72 GRADE B.

B. TOP AND BOTTOM GRID CASTING: CF-3M, ASTM STANDARD A-296-77.

C. THREADED PEDESTAL FOOT: 17-4 PH, ASTM STANDARD A-564-66.

POISON MATERIAL

BORAL

MATERIALS
REQUEST FOR ADDITIONAL INFORMATION

DOES TVA PLAN TO ESTABLISH A BORAL SURVEILLANCE PROGRAM?

- PAR REGION 1 RACKS PERFORMED SUCCESSFULLY AT SEQUOYAH FOR 13 YEARS.
- 100% DRAG TEST WILL BE PERFORMED DURING INSTALLATION AT WATTS BAR.
- AT EACH REFUELING OUTAGE, DRAG IS MONITORED DURING INSERTION AND WITHDRAWAL OF APPROXIMATELY 275 ASSEMBLIES.
- NO VINTAGE COUPON MATERIAL AVAILABLE FOR TRANSFER WITH SEQUOYAH RACKS.
- CORROSION DAMAGE AND LOSS OF BORON CARBIDE CONSIDERED EXTREMELY REMOTE.

LAYOUT AND CONSTRUCTION SUMMARY

- SEQUOYAH AND WATTS BAR POOL AREAS ARE ESSENTIALLY IDENTICAL.

- FUEL TO BE STORED IS THE SAME AT BOTH PLANTS.

- STORAGE CONFIGURATION AND MATERIALS PROPOSED FOR WATTS BAR ARE A COMBINATION OF PREVIOUS AND CURRENT SEQUOYAH RACK DESIGNS WHICH HAVE PERFORMED SUCCESSFULLY AT THAT PLANT.

HIGHLIGHTS OF CRITICALITY SAFETY ANALYSIS

- DESIGN BASIS: $K_{EFF} \leq 0.95$ WITH 95% PROBABILITY AT 95% CONFIDENCE LEVEL AND EVALUATED IN ABSENCE OF SOLUBLE BORON FOR WESTINGHOUSE VANTAGE 5H FUEL ASSEMBLIES UP TO 4.95 ± 0.05 WT% U-235 ENRICHMENT.

- ADMINISTRATIVE PLACEMENT CONTROLS AND BURNUP CREDIT APPLIED. NO CREDIT TAKEN FOR SOLUBLE BORON.

- ANALYSIS PERFORMED USING THE ACCEPTED SCALE SYSTEM OF CODES FOR CROSS-SECTION GENERATION AND REACTIVITY CALCULATIONS, AND CASMO WAS USED FOR DEPLETION CALCULATIONS.

- TVA'S PAR RACK (REGION 1) CRITICALITY ANALYSIS FOR SEQUOYAH VERIFIED BY HOLTEC INTERNATIONAL AND APPLIED AT WATTS BAR.

- REGION 2 ANALYSIS BY HOLTEC IS ESSENTIALLY AN APPLICATION OF WHAT WAS ANALYZED AND LICENSED AT SEQUOYAH FOR 1994 RERACK.

- INTERFACE OF PAR AND HOLTEC RACK DESIGNS EVALUATED TO ASSURE REACTIVITY IMPACTS WERE ASSESSED.

- RESTRICTIONS ON BOTH REGION 1 (PAR FLUX TRAP) RACKS AND REGION 2 (BURNUP CREDIT "BABY") RACKS ESSENTIALLY THE SAME AS FOR SEQUOYAH APPLICATIONS.

CRITICALITY
REQUEST FOR ADDITIONAL INFORMATION

QUESTION 1 ANY SUBSEQUENT CHECKS FOR PROPER STORAGE?

QUESTION 2 WILL ENRICHMENT LIMIT FOR FRESH FUEL BE CHANGED?

QUESTION 3 IS THE BORAL VENTED?

QUESTION 4 DID REGION 2 CALCULATIONS CONSIDER BORON PARTICLE SELF-SHIELDING?

QUESTION 5 WAS KENO METHOD BIAS DETERMINED CORRECTLY?

QUESTION 6 SHOULD STORAGE CAPACITIES BE CLARIFIED?

RADIOLOGICAL EVALUATION

- FUEL HANDLING ACCIDENT DOSES AT SITE BOUNDARY AFTER 2 HOURS:

THYROID DOSE (REM)	=	1.81
BETA DOSE (REM)	=	1.96
GAMMA DOSE (REM)	=	0.675

10 CFR 100 GUIDELINES FOR "WELL WITHIN":

75 RAD FOR THYROID DOSES
6.25 REM FOR WHOLE BODY DOSES

- SOLID RADWASTE

NO SIGNIFICANT INCREASE EXPECTED.

- GASEOUS RELEASES

NORMALLY NEGLIGIBLE FROM FUEL STORAGE AREA. NO SIGNIFICANT INCREASES EXPECTED.

- PERSONNEL EXPOSURES

NO SIGNIFICANT EFFECTS BASED ON PWR OPERATING EXPERIENCE. NO CHANGES NECESSARY IN HEALTH PHYSICS PROGRAM OR AREA RADIATION MONITORING SYSTEM.

- ANTICIPATED EXPOSURE DURING RERACKING

PLANNING ESTIMATE IS 1.5 TO 2.5 PERSON - REM.

RADIOLOGICAL
REQUEST FOR ADDITIONAL INFORMATION

DOES 12% GAP RELEASE FRACTION FOR I-131 ADDRESSED IN NUREG/CR 5009 WARRANT RECALCULATING THE FUEL HANDLING ACCIDENT DOSE?

ENVIRONMENTAL ASSESSMENT

- STORAGE EXPANSION NEED

  FULL CORE DISCHARGE CAPABILITY LOST IN MARCH 2002; COMPONENT SHUFFLING WITH NEW FUEL IN POOL ADVANCES NEED DATE TO SEPTEMBER 2000.

- CONSIDERATIONS IN EVALUATING STORAGE OPTIONS

  MINIMIZE FUEL HANDLING AND EXPOSURES
  MATURITY OF TECHNOLOGY, INDUSTRY EXPERIENCE
  MAINTAIN FLEXIBILITY FOR FUTURE STORAGE ACTIONS
  MINIMIZE OVERALL CAPITAL AND O&M COSTS.

- MINIMAL COMMITMENT OF PRIMARY RESOURCES IS REQUIRED

- OTHER CONSIDERATIONS

  NEGLIGIBLE INCREASE IN HEAT LOSS AND WATER VAPOR EMISSION TO ENVIRONMENT.

  INSIGNIFICANT INCREASE IN POOL-RELATED RADWASTE.

ENVIRONMENTAL
REQUEST FOR ADDITIONAL INFORMATION

QUESTION 1	ANY NPDES PERMIT CHANGES REQUIRED?
QUESTION 2	ANY CHANGES TO WASTE TREATMENT SYSTEMS OR FLOWRATES?
QUESTION 3	ANY EFFLUENT TEMPERATURE CHANGES RESULTING FROM INCREASES IN POOL TEMPERATURE OR EVAPORATION?
QUESTION 4	DISCUSS SUBSEQUENT ACTIONS FOR INCREASING SPENT FUEL STORAGE CAPACITY.
QUESTION 5	DISCUSS LENGTH OF CURRENT AND FUTURE FUEL CYCLES.

FUEL DISCHARGE SCHEDULE

CYCLE NO.	SHUTDOWN DATE	EFFECTIVE FULL POWER DAYS (EFPDS)	NUMBER OF DISCHARGE ASSEMBLIES	ASSEMBLIES IN POOL	USABLE STORAGE CELLS REMAINING
1	9/97	450	84	84	119[1]
2	2/99	479	88	172	39
3	9/00	499	80	252	-41
4	3/02	485	80	332	
5	9/03	487	80	412	
6	3/05	487	80	492	

CYCLE 2 IS PLANNED AS A TRANSITION TO 18-MONTH CYCLES.

CYCLE 5 AND SUBSEQUENT FUEL CYCLES ARE PROJECTED TO BE 18-MONTH CYCLES WITH APPROXIMATELY 487 EFPDS.

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[1] 484 - (84+193+88) = 119

THERMAL - HYDRAULIC EVALUATIONS

- POOL BULK TEMPERATURE AS A FUNCTION OF TIME;

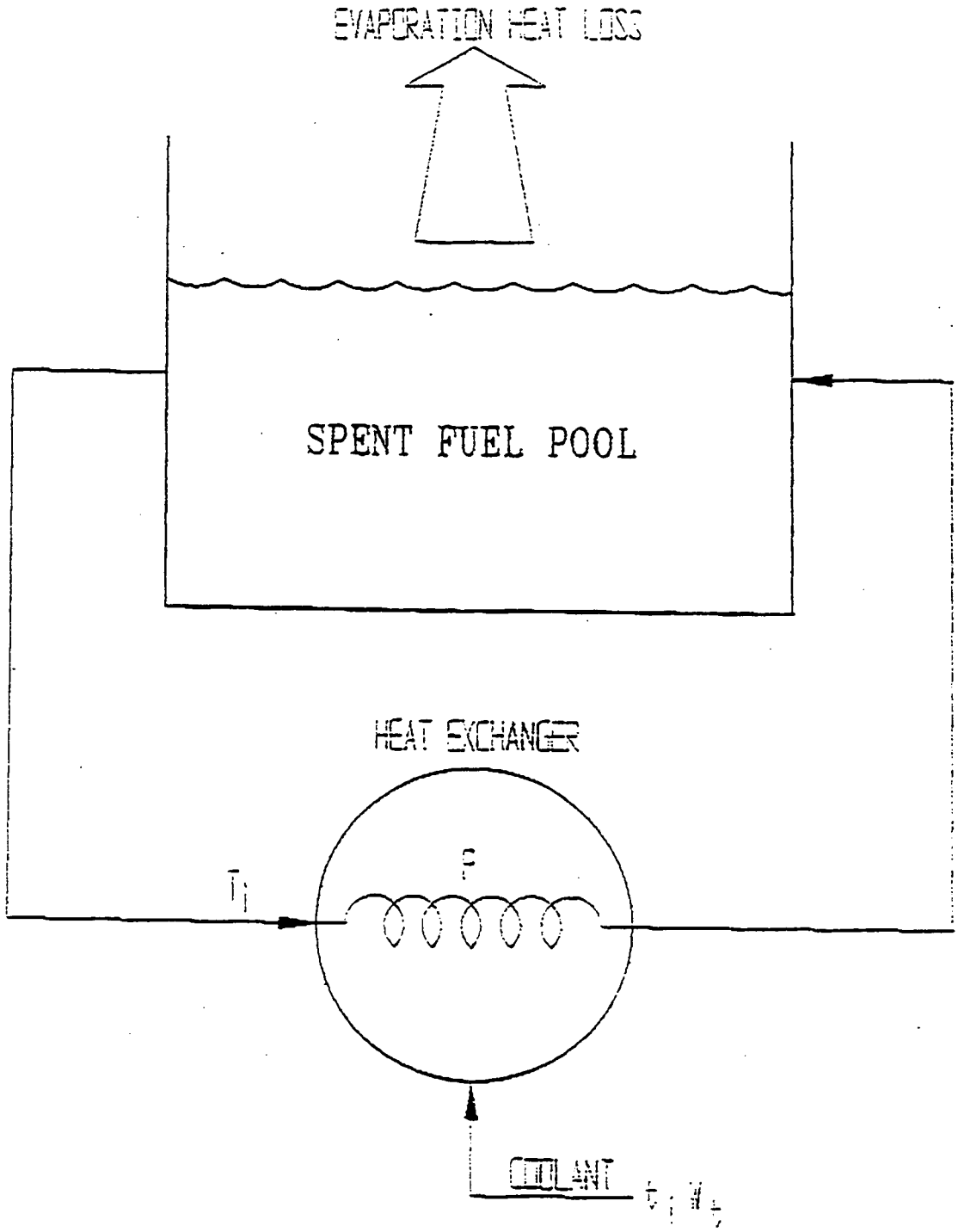
  BULKTEM

- TIME -TO-BOIL; TBOIL

- MAXIMUM LOCAL WATER TEMPERATURE; FLUENT

  - UNBLOCKED CELL
  - PARTIALLY BLOCKED CELL
  - ECCENTRIC FUEL PLACEMENT

- FUEL CLADDING TEMPERATURE



SPENT FUEL COOLING MODEL

ENERGY CONSERVATION RELATIONSHIP:

$$C \frac{dT}{dt} = B \; P_o + Q(t) - M_t \; p \, (T - T_C) - Q_{ev}(T)$$

WHERE:

C:	POOL THERMAL CAPACITY
B:	DIMENSIONLESS DECAY HEAT FACTOR
P_O:	OPERATING POWER PER ASSEMBLY
Q(t):	DECAY HEAT LOAD COMPUTED USING ORIGEN-2
M_t:	COOLANT THERMAL FLOW RATE
p:	TEMPERATURE EFFECTIVENESS OF FUEL POOL COOLER
T:	POOL WATER TEMPERATURE AT TIME t
Qev:	EVAPORATIVE HEAT LOSS

NOTES:

- p is a function of pool water temperature
- B is assumed to be constant for the duration of the transient study

COMPUTER CODE BULKTEM

FUEL POOL COOLER MARGINS

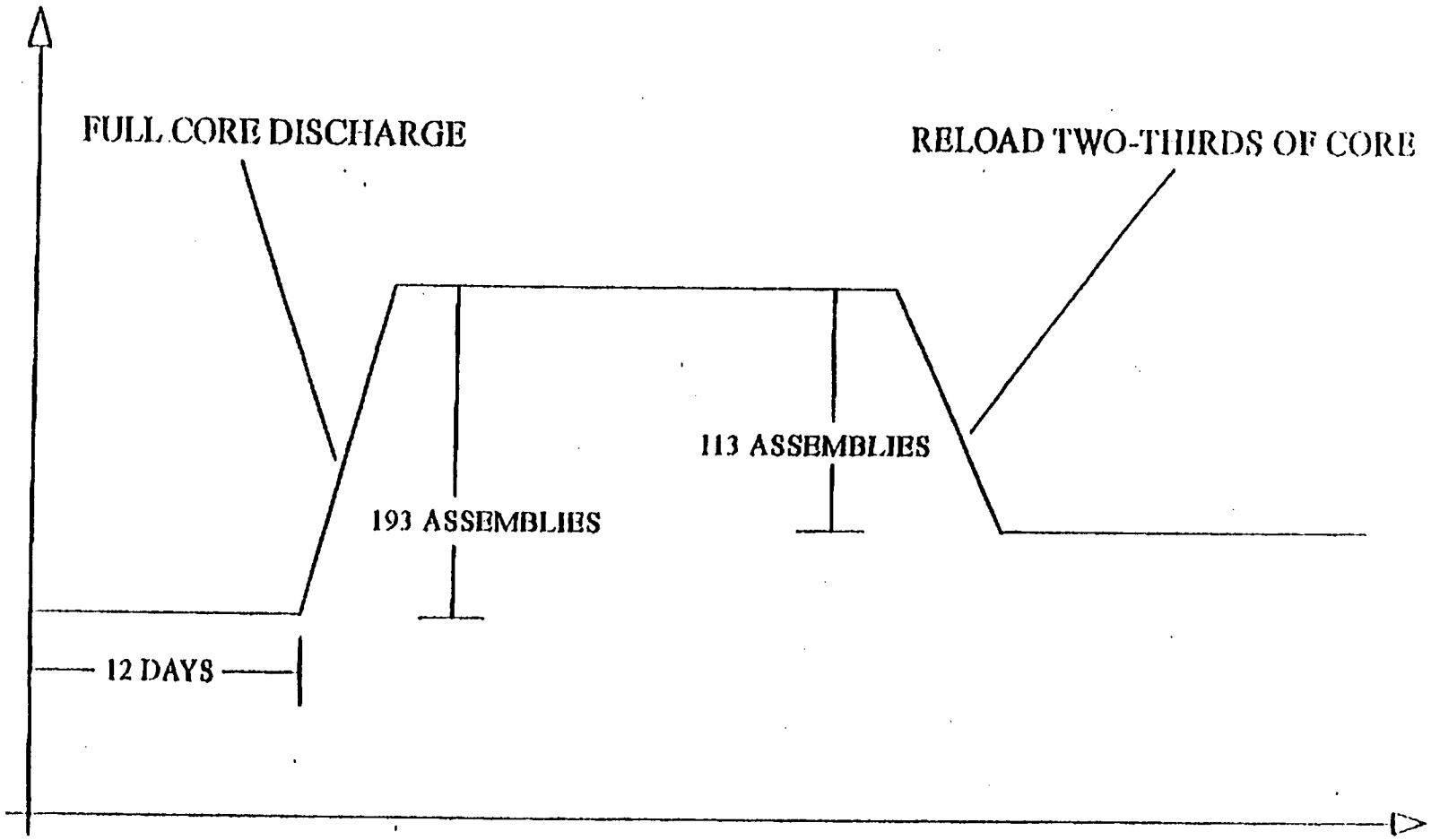
- **OVERALL HEAT TRANSFER COEFFICIENT USED: 351 Btu (hrxft^{2}x°F)**

  ACTUAL (CLEAN) OVERALL COEFFICIENT: 565 Btu/(hrxft^{2}x°F)

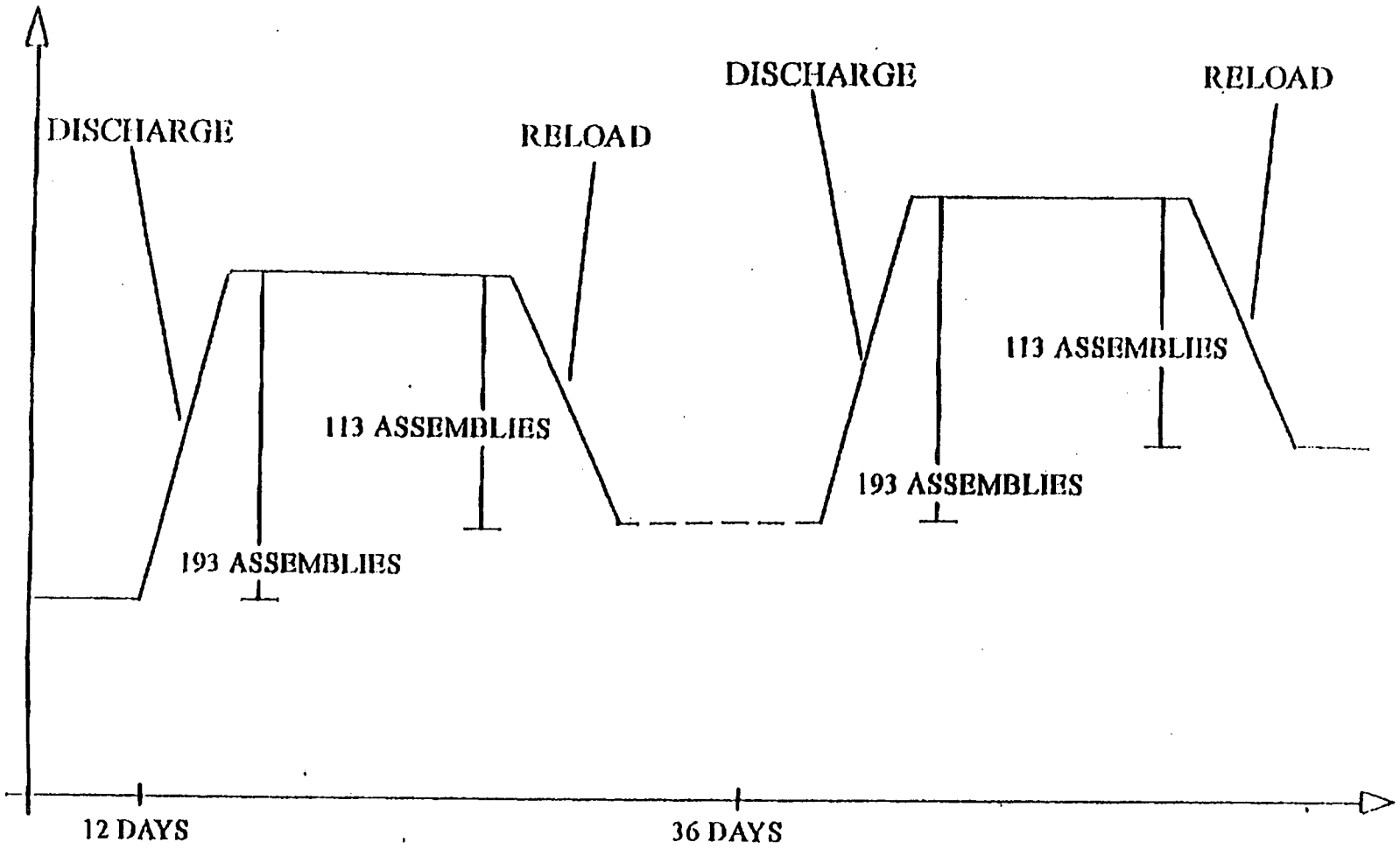
  i.e. 62.12% OF THE CLEAN COEFFICIENT.

- **5% OF THE TUBES ASSUMED TO BE PLUGGED.**

FULL CORE DISCHARGE SCENARIO FUEL INVENTORY PROFILE



UNPLANNED DISCHARGE SCENARIO FUEL INVENTORY PROFILE



DATA FOR DISCHARGE CASES 1A THROUGH 2B

CASE ID	DISCHARGE SCENARIO	NUMBER OF ASSEMBLIES RECENTLY DISCHARGED	TRANSFER START TIME (HOURS AFTER INITIAL SHUTDOWN)	FUEL TRANSFER TIME (HRS)	BURNUP PER BATCH (MWD/MTU)
1A	FULL CORE	193	288	48.25	48,000
1B	FULL CORE	193	288	48.25	48,000
2A	UNPLANNED	64 + 193	288 & 1152	16 + 48.25	48,000 16,000 (64) 32,000 (64) 48,000 (65)
2B	UNPLANNED	64 + 193	288 & 1152	16 + 48.25	48,000 16,000 (64) 32,000 (64) 48,000 (65)
2A'	UNPLANNED	80 + 193	288 & 1152	20 + 48.25	48,000 1,400 (80) 32,000 (48) 48,000 (65)
2B'	UNPLANNED	80 + 193	288 & 1152	20 + 48.25	48,000 1,400 (80) 32,000 (48) 48,000 (65

POOL BULK TEMPERATURE AND HEAT LOAD DATA

Case 1A: Full Core Discharge with Two SFPCS Trains

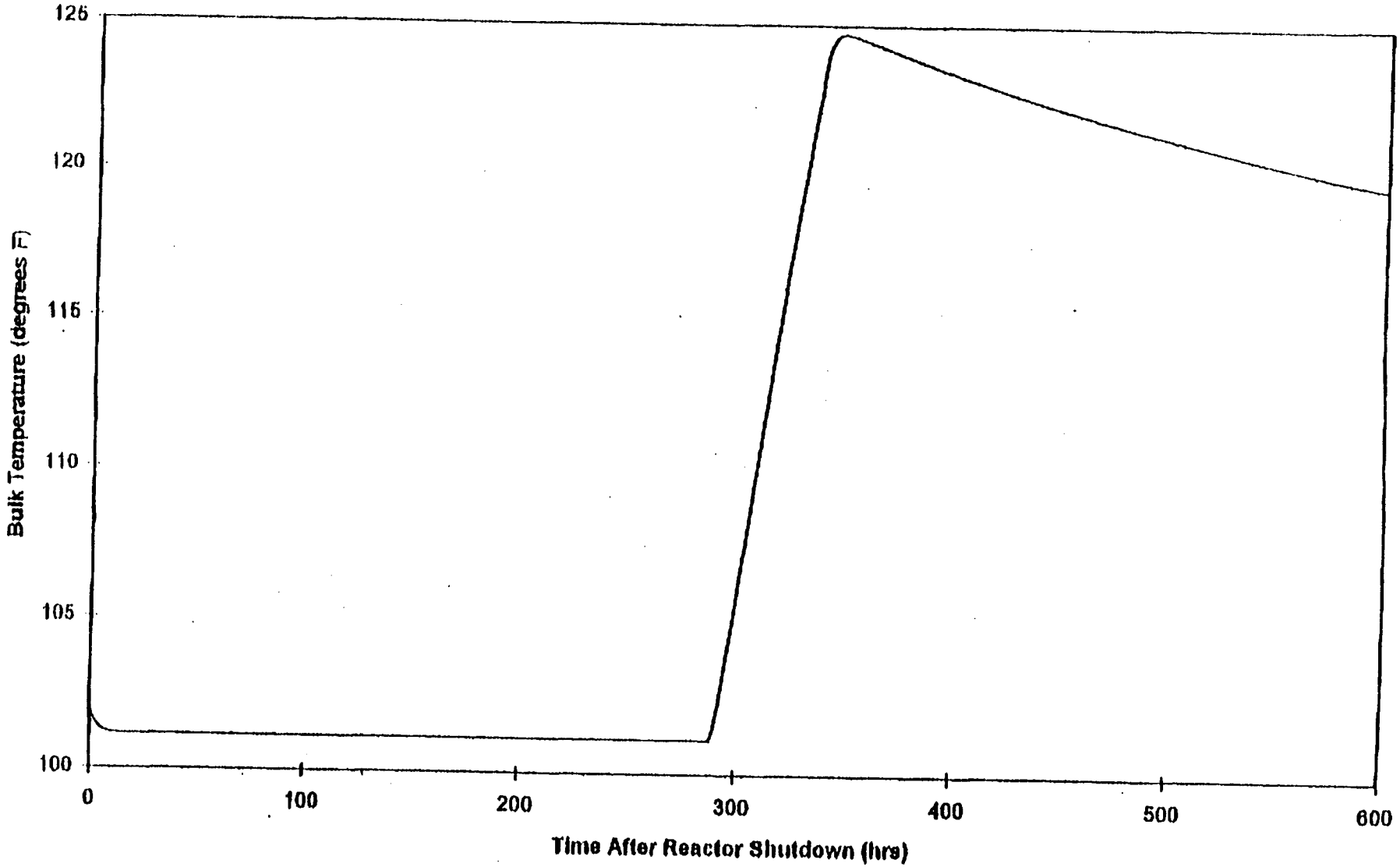
Case 1B: Full Core Discharge with One SFPCS Train

Case 2A: Unplanned Core Discharge with Two SFPCS Trains

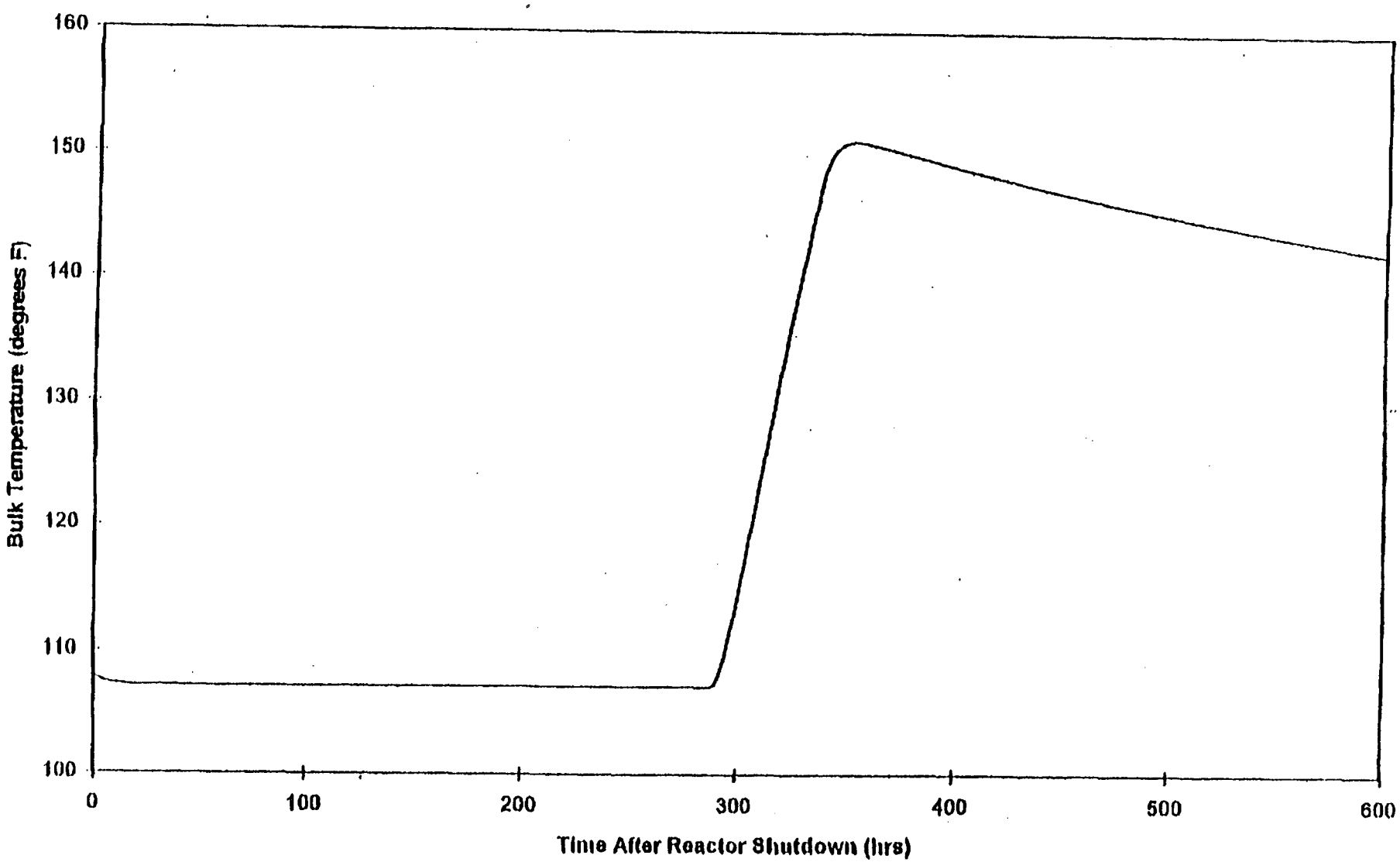
Case 2B: Unplanned Core Discharge with One SFPCS Train

Case ID	Max. Pool Bulk Temperature (F)	Coincident Net Decay Heat Load (Mbtu/hr)	Coincident Time (hours after initial shutdown)	Coincident Evaporative Heat Loss (MBtu/hr)
1A	124.69	27.843	345	0.256
1B	151.17	26.675	352	1.225
2A	129.30	32.240	1209	0.360
2B	159.24	30.618	1215	1.799
2A'	126.10	29.206	1208	0.286
2B'	153.71	27.906	1215	1.385

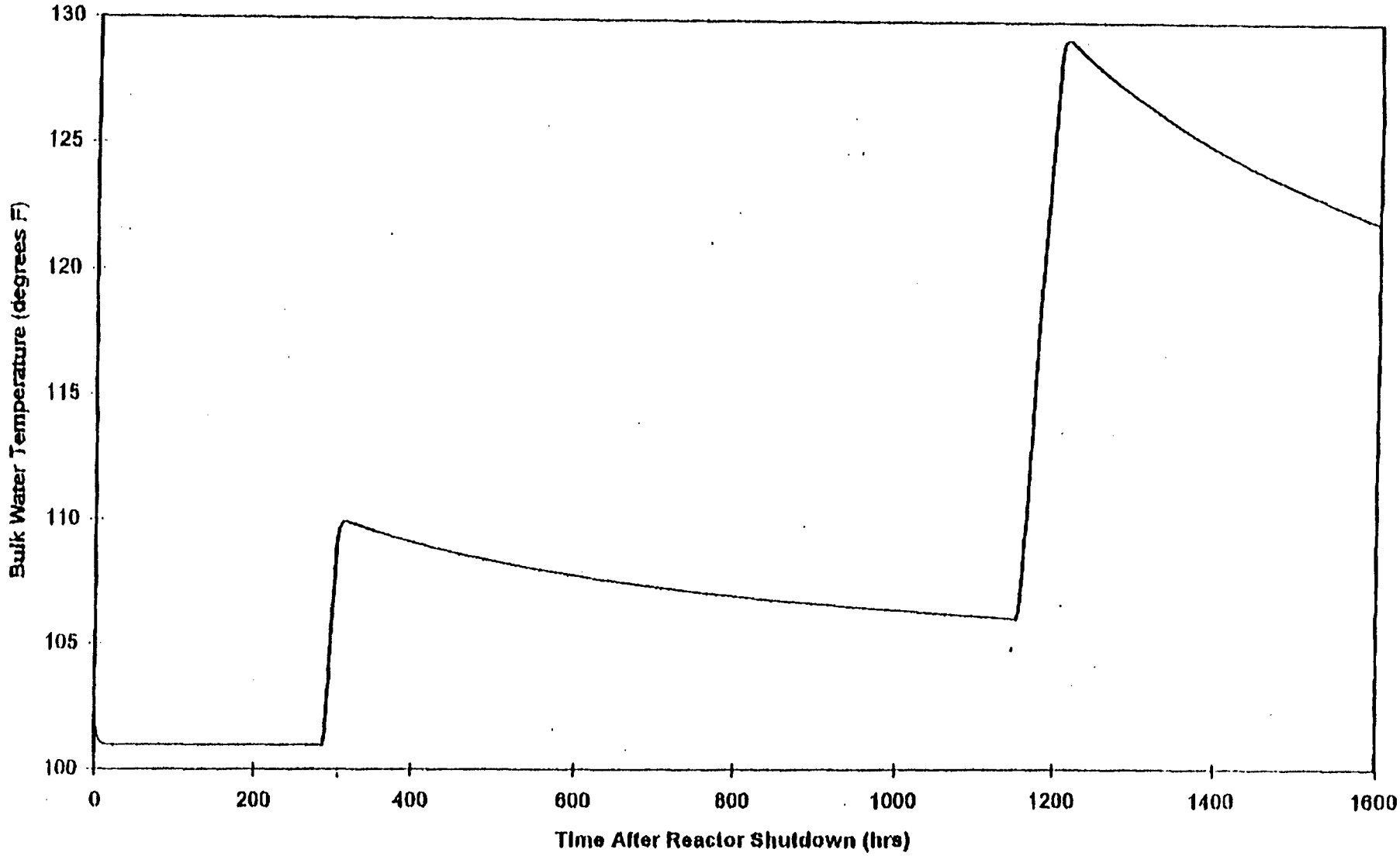
Bulk Temperature Profile for Full Core Discharge with Two SFPCS Trains



Bulk Temperature Profile for Full Core Discharge with One SFPCS Train



Bulk Temperature Profile for Unplanned Discharge with Two SFPCS Trains



TIME-TO-BOIL CALCULATIONS RESULTS

(WITH NO MAKE-UP WATER)

CASE 1A: FULL CORE DISCHARGE WITH TWO SFPCS TRAINS

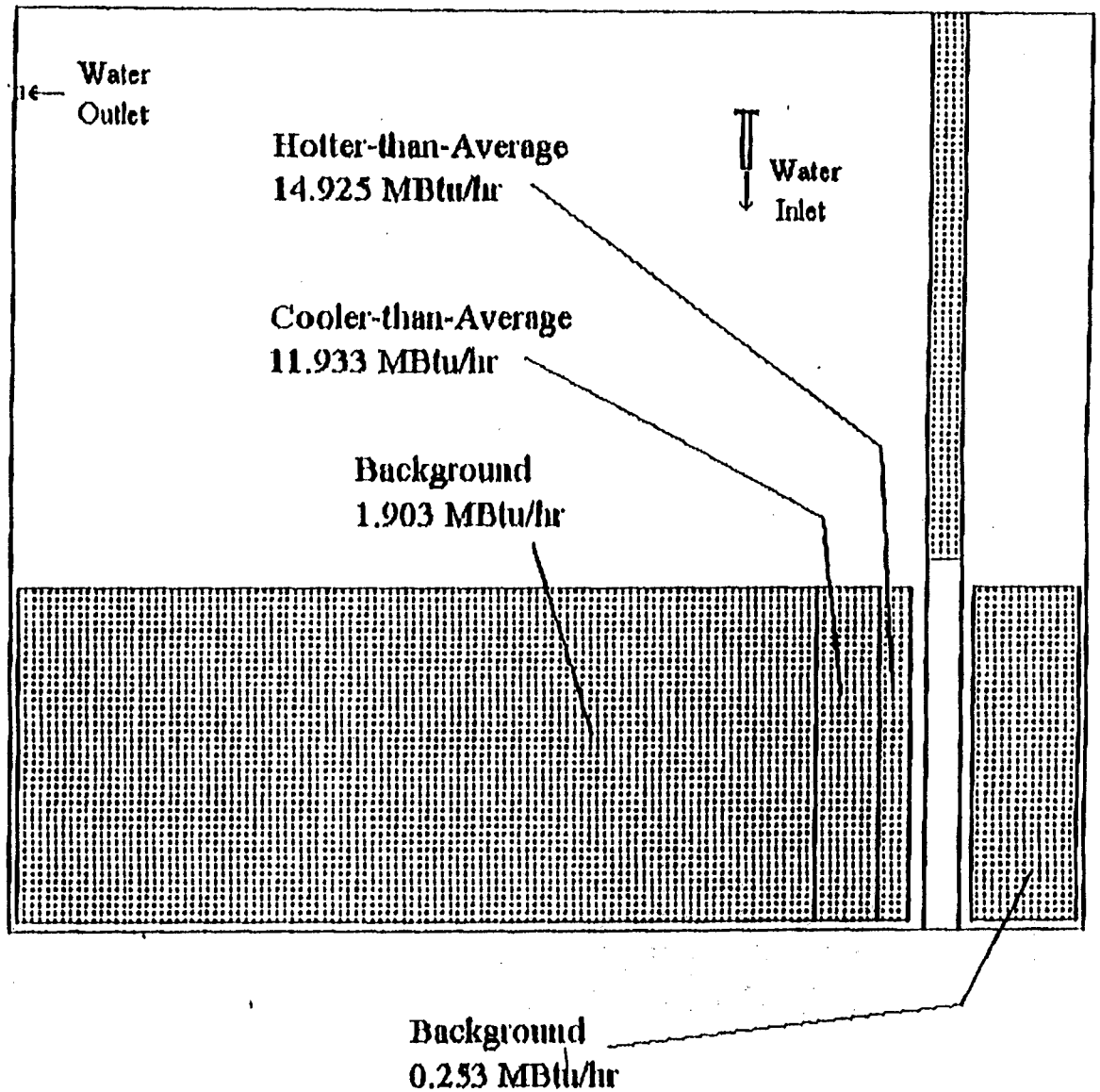
CASE 1B: FULL CORE DISCHARGE WITH ONE SFPCS TRAIN

CASE 2A: UNPLANNED DISCHARGE WITH TWO SFPCS TRAINS

CASE 2B: UNPLANNED DISCHARGE WITH ONE SFPCS TRAIN

Case Identifier	Time-to-Boil (hours)	Time-to-Ten Feet (hours)	Maximum Boil-Off Rate (gpm)	Average Heatup Rate (°F/hr)
1A	8.84	47.4	68.59	9.88
1B	6.27	45.0	68.32	9.70
2A	8.99	50.1	64.22	9.20
2B	5.86	47.1	64.08	9.00

SFP/Cask Pit CFD Model Decay Heat Distribution

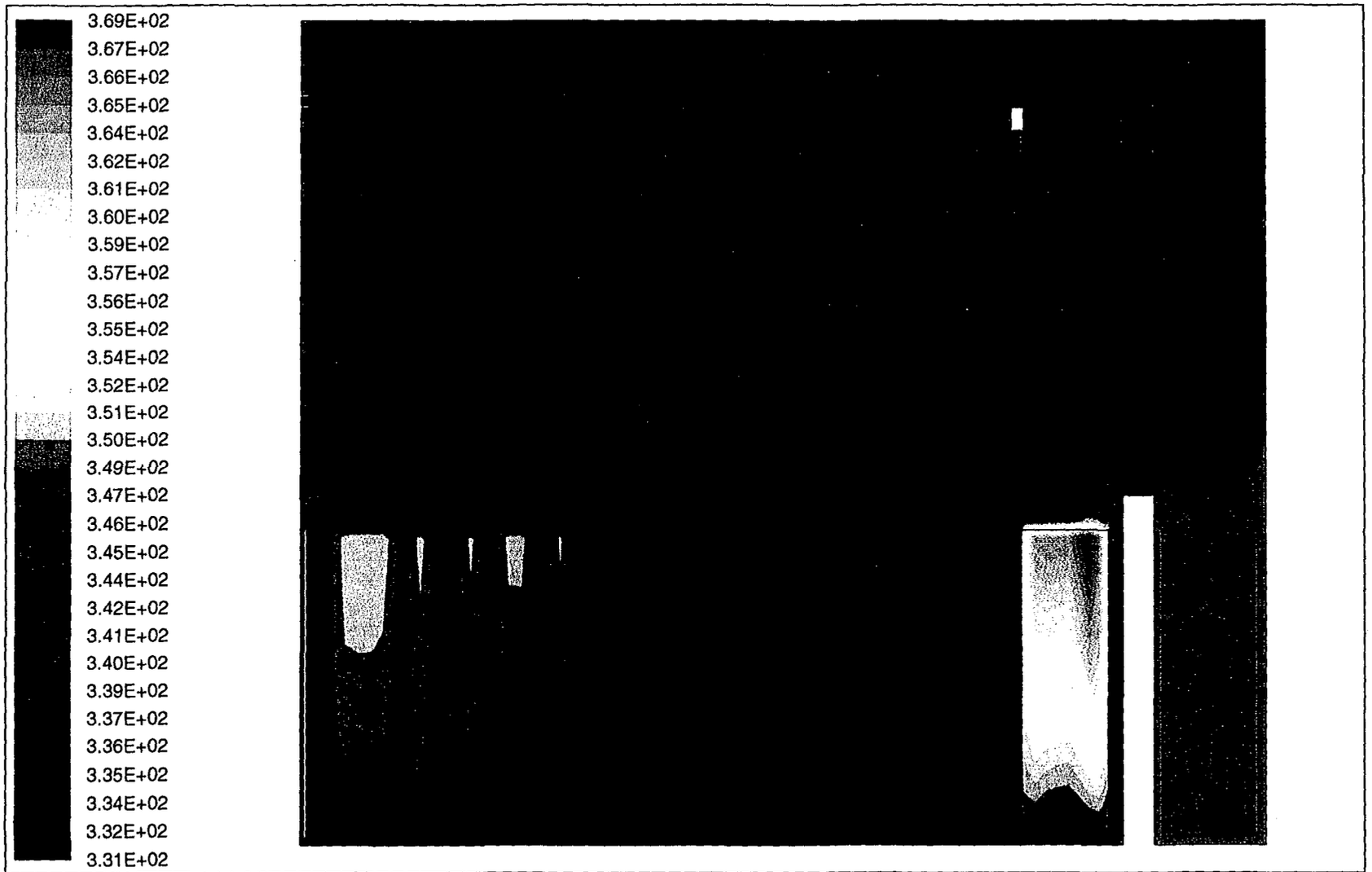


DATA FOR LOCAL TEMPERATURE

TYPE OF FUEL ASSEMBLY:	PWR
FUEL CLADDING OD (INCHES):	0.374
FUEL CLADDING ID (INCHES):	0.329
STORAGE CELL ID (INCHES):	8.80
ACTIVE FUEL LENGTH (INCHES):	144
NUMBER OF RODS PER ASSEMBLY	264
ASSEMBLY OPERATING POWER (Mbtu/hr):	60.32
CELL PITCH (INCHES):	8.972
CELL HEIGHT (INCHES):	172
BOTTOM PLENUM HEIGHT (INCHES):	5.25
RADIAL BUNDLE PEAKING FACTOR:	1.65
TOTAL BUNDLE PEAKING FACTOR:	2.50

MAXIMUM LOCAL TEMPERATURE CALCULATIONS RESULTS

PARAMETER	NO BLOCKAGE	PARTIALLY BLOCKED	OFF-CENTER
LOCAL MAXIMUM WATER TEMPERATURE (°F)	193.7	204.1	195.2
MID-HEIGHT FUEL CLADDING TEMPERATURE (°F)	208.2	217.1	208.9
BOUNDING MAXIMUM CLADDING TEMPERATURE (°F)	221.5	231.9	223.1

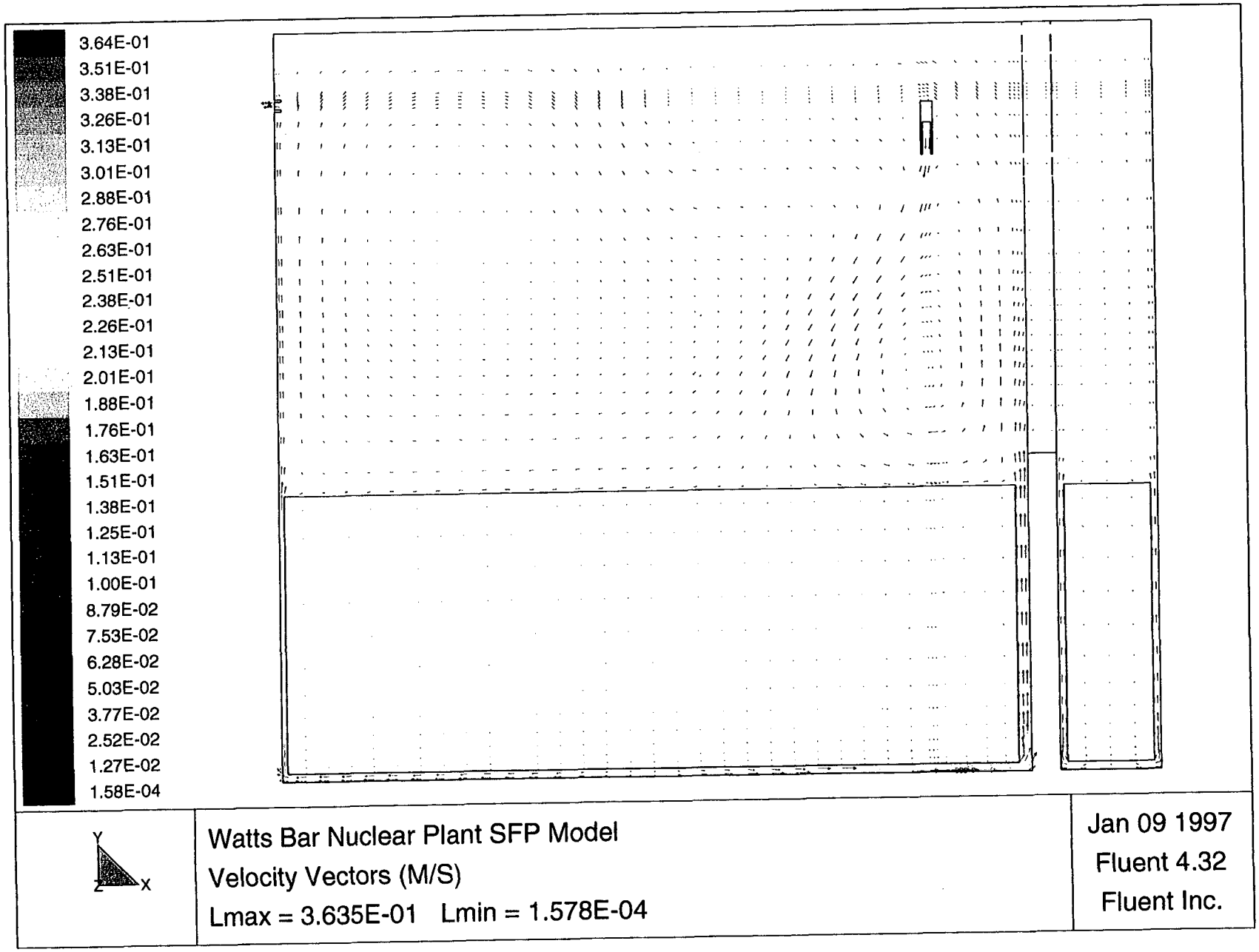


Y
Z X

Watts Bar Nuclear Plant SFP Model
Temperature (K)
Lmax = 3.686E+02 Lmin = 3.310E+02

Jan 09 1997
Fluent 4.32
Fluent Inc.

by Dr. K. P. Singh, Holtec International
January 14, 1997, Watts Bar Rerack
Meeting with USNRC.



by Dr. K. P. Singh, Holtec International
January 14, 1997, Watts Bar Rerack
Meeting with USNRC.

SEISMIC/STRUCTURAL ANALYSIS

WATTS BAR SPENT FUEL POOL RE-RACK

SUMMARY OF ANALYSIS

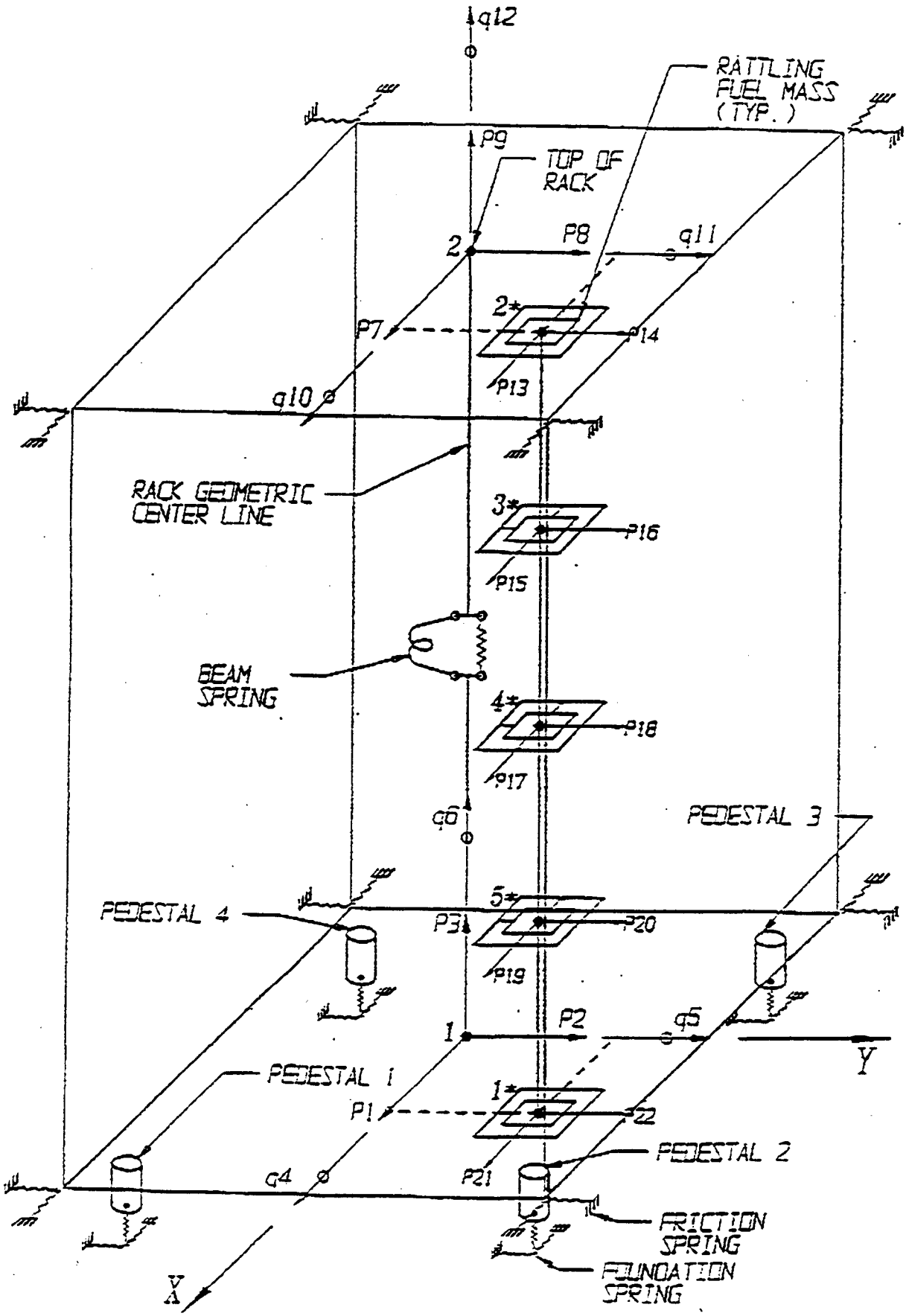
- TIME HISTORY DEVELOPMENT
- SPENT FUEL RACK MODELING
- SPENT FUEL RACK DYNAMIC ANALYSES/RESULTS
- FATIGUE ANALYSES
- ACCIDENT CALCULATIONS
- SPENT FUEL POOL STRUCTURAL CONSIDERATIONS

TIME HISTORY DEVELOPMENT

- DEVELOPED TIME HISTORIES MEET SRP 3.7.1
  - RE-GENERATED RESPONSE SPECTRA BOUNDS TARGET SPECTRA
  - RE-GENERATED PSD BOUNDS TARGET PSD
  - EACH TIME HISTORY IN A SET IS STATISTICALLY INDEPENDENT FROM OTHER TWO HISTORIES

SPENT FUEL RACK MODELING

- 12 DOF FOR RACK STRUCTURE.
- BEAM STIFFNESS ESTIMATED FROM CELLULAR GEOMETRY AND CONNECTIVITY.
- MULTIPLE FUEL MASSES WITH RATTLING.
- IMPACT CAPABILITY AND FRICTION INCLUDED.
- HYDRODYNAMIC EFFECTS COUPLE RACKS
- TRUE CAPTURE OF 3-D RESPONSE.



SCHEMATIC OF THE DYNAMIC MODEL FOR DYNARACK

CONSERVATISMS IN MODEL/SIMULATIONS

- ALL FUEL IN RACK AT ANY HEIGHT CONSERVATIVELY ASSUMED TO MOVE IN-PHASE.

- NO FLUID DRAG OR DAMPING.

- FLUID COUPLING EFFECTS BASED ON NOMINAL GAPS THUS MINIMIZING RESISTANCE AS GAPS CLOSE.

- STIFFNESS OVERESTIMATED IN ORDER TO MAXIMIZE INSTANTANEOUS PEAK LOADS.

ANALYSES/RESULTS

- SAME SIMULATION CODE AS USED IN PREVIOUS RACK LICENSING EFFORTS PRESENTED TO NRC.

- SINGLE RACK AND WHOLE POOL MULTI-RACK ANALYSES PERFORMED.

- FLUID COUPLING EFFECTS MODELED EXACTLY IN WPMR ANALYSES (34 RACKS).

DYNAMIC SIMULATIONS

- **BOUNDING SINGLE RACKS - VARIOUS FUEL LOAD ORIENTATIONS AND WEIGHT DISTRIBUTIONS (INTACT FUEL [1700 LB.] PLUS A POTENTIAL MIX OF HEAVIER FUEL)**

- **~~WBMP~~ WPMR ANALYSES - GAUSSIAN COEFFICIENT OF FRICTION DISTRIBUTION WITH VARIOUS LOADING SCENARIOS.**

- **2 TABLES SHOW SIMULATIONS PERFORMED.**

LIST OF SINGLE RACK SIMULATIONS

(Simulations include baby rack and consolidated fuel except as noted)

Run No.	Rack Size	Seismic Event	Fuel Loading	Coefficient of Friction
1	7X9	1.0 X SSE	Fully Loaded	0.2
2	7X9	1.0 X SSE	Half Loaded Along X Axis	0.2
3	7X9	1.0 X SSE	Half Loaded Along Y Axis	0.2
4	7X9	1.0 X SSE	Fully Loaded	0.8
5	7X9	1.0 X SSE	Half Loaded Along X Axis	0.8
6	7X9	1.0 X SSE	Half Loaded Along Y Axis	0.8
7	7X9	1.1 X SSE	Fully Loaded	0.8
8	7X9	1.2 X OBE (0.6 X SSE)	Limiting Case from Run Nos 1 through 6	
9	7X9*	1.0 X SSE	Limiting Case from Run Nos. 1 through 6	
10	7X9**	1.0 X SSE	Limiting Case from Run Nos. 1 through 6	
11	7X9***	1.0 X SSE	Limiting Case from Run Nos. 1 through 6	
12	15 X 15^	1.0 X SSE	Limiting Case from Run Nos. 1 through 6	

* 7 x 9 RACK IS LOADED WITH CONSOLIDATED FUEL. NO BABY RACK ATTACHED. EFFECTIVE FUEL ASSEMBLY WEIGHT EQUALS 3000 LBF.

** 7 X 9 RACK AND ATTACHED BABY RACK ARE LOADED WITH INTACT FUEL. EFFECTIVE FUEL ASSEMBLY WEIGHT EQUALS 2424 LB.

*** EFFECTIVE FUEL ASSEMBLY WEIGHT EQUALS 2873 LBF. THREE LOADING SCENARIOS REPRESENTED BY THIS WEIGHT ARE LISTED BELOW. OTHER COMBINATIONS WHICH PRODUCE AN EFFECTIVE WEIGHT EQUAL TO 2873 LBF ARE ALSO POSSIBLE.

(1) A 7 X 9 RACK LOADED WITH INTACT FUEL (10%) AND CONSOLIDATED FUEL (90%). NO BABY RACK ATTACHED.

(2) A 7 X 9 RACK AND ATTACHED BABY RACK LOADED WITH INTACT FUEL (75%) AND CONSOLIDATED FUEL (25%).

(3) A 7 X 9 RACK LOADED WITH INTACT FUEL (65%) AND CONSOLIDATED FUEL (35%). ATTACHED BABY RACK LOADED WITH INTACT FUEL.

^ 15 X 15 RACK IN THE CASK PIT IS LOADED WITH CONSOLIDATED FUEL. NO BABY RACK ATTACHED. EFFECTIVE FUEL ASSEMBLY WEIGHT EQUALS 3000 LBF.

LIST OF WHOLE POOL MULTI-RACK (WPMR) SIMULATIONS

Run No.	Seismic Event	Fuel Loading	Coefficient of Friction
1	1.0 x SSE	Fully loaded with intact fuel, 1700 lbf per assembly	Gaussian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)
2	1.0 x SSE	Fully loaded with consolidated fuel, 3000 lbf per assembly	Gaussian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)
3	1.0 x SSE	Fully loaded with fuel (75% consolidated, 25% intact), 2675 lbf per assembly	Gaussian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)
4	1.0 x OBE (0.5 x SSE)	Fully loaded with intact fuel, 1700 lbf per assembly	Gaussian distribution with a mean of 0.5 (upper and lower limits of 0.8 and 0.2)

RESULTS

- PRIMARY STRESS EVALUATIONS PER ASME NF INTERACTION EQUATIONS.
- LICENSING BASIS INTACT FUEL
  - PEDESTAL LOADS - 120000 LB./86300 LB.
  - NO RACK-TO-WALL OR RACK-TO-RACK IMPACTS
  - LARGE MARGIN AGAINST ISOLATED RACK OVERTURNING
  - LARGEST INTERACTION FACTORS (.522/.684)<1.0.
- RACK-RACK INTERCONNECT LOAD 67% OF MAX.

FATIGUE ANALYSIS

- ESTIMATE PEAK STRESS INTENSITY FROM FINITE ELEMENT MODEL OF VICINITY NEAR PEDESTAL WITH STATICALLY APPLIED MAXIMUM PEDESTAL LOADS.

- ESTIMATE NUMBER OF CYCLES IN A SEISMIC EVENT FROM WPMR ARCHIVED TIME HISTORY RESULTS OF PEDESTAL WITH PEAK LOADS.

- DETERMINE CUMULATIVE DAMAGE FACTOR (1 SSE AND 5 OBE'S). 0.732<1.0.

MECHANICAL ACCIDENTS

- DROPPED FUEL ASSEMBLY PLUS HANDLING TOOL TO BASE OF RACK. DROP HEIGHT IS 36" ABOVE RACK TOP TO BOTTOM OF CELL.

- HOLTEC RACKS CONTAIN ASSEMBLY WITH NO DIRECT LINER IMPACT.

- PaR RACK STRUCTURE ABSORBS 12% OF ENERGY; ASSEMBLY THEN IMPACTS THE LINER.

- IMPACT ON LINER EXAMINED.

- DYNA3D (LARGE DEFLECTION/INELASTIC MATERIAL CODE USED FOR IMPACT ANALYSIS) MODELED ASSEMBLY/LINER/CONCRETE IMPACT PROBLEM.

- CONCLUSION (BASED ON STRAIN LEVELS) IS THAT NO LINER RUPTURE OCCURS.

MECHANICAL ACCIDENTS

- FUEL ASSEMBLY DROP ON RACK TOP CELL WALL INDICATES THAT RACK CELLULAR PERMANENT DEFORMATION EXTENDS DOWN 3.1"<DISTANCE TO BEGINNING OF ACTIVE FUEL REGION.

- GATE DROP, WHILE HEAVIER DROPPED OBJECT, HAS ENERGY ABSORPTION IN MORE CELL WALLS. PERMANENT DEFORMATION <5.4".

IMPACT SHIELD

- RACK IN CASK PIT PROTECTED BY REMOVABLE IMPACT SHIELD AFTER LOADING.

- DROP ANALYSES PERFORMED TO DEMONSTRATE THAT IMPACT SHIELD ABSORBS ENERGY AND HAS LOW STRESS LEVELS IN MAIN STEELWORK AND CONCRETE WALL SUPPORT.

- ALLOWABLE LOAD vs CARRY HEIGHT QUANTIFIED.

FUEL POOL STRUCTURE

- STRUCTURE IS IDENTICAL TO SEQUOYAH PLANT (SQN); COMPARISON OF LOADS/MARGINS PERMITS SQN TO BE USED TO QUALIFY WBN.
  - DEAD WEIGHT OF STRUCTURE, WATER, RACKS, FUEL
  - SSE OR OBE PEDESTAL LOADS AND STRUCTURE INERTIA
  - HYDRODYNAMIC LOADS CAUSED BY RACK MOVEMENT
  - MEAN TEMPERATURE PLUS THERMAL GRADIENTS
- ANALYSES FOLLOWS ACI CODE/WS METHOD.

CONCLUSIONS

- SPENT FUEL RACKS MEET STRUCTURAL REQUIREMENTS FOR LOADING WITH INTACT FUEL; NO RACK-TO-RACK OR RACK-TO-WALL IMPACT.

- USING ARRAYS WITH HEAVIER FUEL WEIGHTS STILL ACHIEVES STRUCTURAL MARGINS WITHOUT ANY IMPACTS OF RACKS WITH THE FUEL POOL WALLS.

- SPENT FUEL RACK MAINTAINS SUBCRITICALLITY OF FUEL IF THERE IS A FUEL OR GATE HANDLING ACCIDENT.

- FATIGUE DAMAGE UNDER 1 SSE AND 5 OBE'S IS WITHIN ASME LIMITS.

- CONCRETE STRUCTURAL MARGINS ARE MET IN THE SPENT FUEL POOL.

SUMMARY

- CONFORMS TO APPLICABLE NRC REQUIREMENTS

- SIMILAR TO SEVERAL OTHER RE-RACK PROJECTS

- EXTREME BENEFITS IN COMPLETING BEFORE FIRST REFUELING OUTAGE

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO SPENT FUEL POOL CAPACITY EXPANSION

Plant Systems

1. With regard to the calculated decay heat loads following the proposed pool expansion, provide the following information:

   (a) the decay heat generation rate as a function of decay time for both the maximum normal and maximum abnormal conditions (information should clearly show the decay heat generation rate from each batch of the previously discharged spent fuel assemblies and the freshly discharged full core in the spent fuel pool); and

   (b) a comparison of the assumptions used in the decay heat calculations to the guidance described in Section 9.1.3 of the Standard Review Plan (SRP, NUREG-0800) including the Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."

2. Identify and provide the basis for any deviations and exceptions to the guidance described in SRP Section 9.1.3 regarding the decay heat calculation and cooling of the spent fuel assemblies.

3. Figure 5.5.7 indicates that 600 hours after the reactor shutdown the decay heat generation rate in the spent fuel pool is approximately 25.5 x 10^6 Btu/Hr. However, Figure 5.5.8 indicates that 600 hours after the reactor shutdown the decay heat generation rate in the spent fuel pool is approximately 12.0 x 10^6 Btu/Hr. Provide detailed clarification for this discrepancy.

4. In the decay heat calculation for the case of normal full core discharge, fuel assemblies are assumed to be discharged to the spent fuel pool after 12 days of decay in the reactor. Has this restriction of 12-day duration for fuel assemblies to decay in the reactor prior to movement of a fuel assembly been incorporated in the Watts Bar Technical Specifications (TS)?

5. Note #1 for TS Section 3.9.4 states that the spent fuel pool transfer canal gate and the spent fuel pool cask pit gate may travel over assemblies in the spent fuel pool. Discuss the design features (such as interlocks or single-failure-proof crane design) to preclude these gates from dropping on spent fuel.

Radiological Protection

It is recommended that the licensee revise this section of the TS, Chapter 9, "Radiological Evaluation," Table 9.2. The core inventory for gap-release fractions should be 12 percent instead of 10 percent (normal burnup) in accordance with NUREG/CR-5009. The increase in I-131 is due to the burnup level of 60 GWD/T and the increase in fuel enrichment to 5.0 percent of U-235, whereas Regulatory Guide 1.25 assumes a release fraction of 0.10 (normal burnup). The licensee must recalculate his fuel handling accident dose for I-131 to address NUREG/CR-5009.