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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
DUKE POWER COMPANY, et al.	)	Docket Nos. 50-413
	)	50-414
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

TESTIMONY OF A. LOWELL SNOW

Q. PLEASE STATE YOUR NAME.

A. Lowell Snow

Q. BY WHOM ARE YOU EMPLOYED AND IN WHAT CAPACITY?

A. I am a Design Engineer II employed by Duke Power Company, 422 S. Church Street, Charlotte, North Carolina 28242. A statement of my qualifications is attached to this testimony as Attachment B.

Q. DESCRIBE THE AREAS OF YOUR RESPONSIBILITIES AS A DESIGN ENGINEER II.

A. My job responsibilities include spent fuel decay heat and criticality evaluations. Prior to my present job I was responsible for Catawba fluid systems design.

Q. WHAT IS THE PURPOSE OF YOUR TESTIMONY?

A. The purpose of my testimony is to address those portions of Contention 16 dealing with (1) the ability of the spent fuel pool cooling system to maintain anticipated pool water temperatures at or below the NRC's acceptance criteria with Oconee and McGuire, as well as Catawba, spent fuel; and (2) the criticality aspects of the expanded fuel pools.

COOLING SYSTEM

Q. DESCRIBE THE PRESENT SPENT FUEL POOL COOLING SYSTEM.

A. The spent fuel pool cooling system consists of two cooling loops, one purification loop and one skimmer loop. The pool cooling subsystem consists of two full capacity pumps, each of which are designed to pump 2840 gallons/minute, two full capacity heat exchangers, each of which are designed to maintain the spent fuel pool temperatures below 150°F under normal heat load conditions (as described on p. 3, line 19), and associated piping and valves sufficient to take suction from the pool and return the cooled water to the pool. This equipment is arranged in two loops (sometimes referred to as trains), each with one pump, one heat exchanger and associated piping and valves. The details of this system are set forth in FSAR Section 9.1.3.

The pool purification subsystem consists of a fuel pool cooling pre-filter, fuel pool cooling demineralizer and fuel pool cooling post-filter. The spent fuel pool skimmer loop consists of a skimmer trough, strainer, skimmer pump, and filter.

Q. HOW DOES THE PRESENT DESIGN OF THE SPENT FUEL POOL COOLING SYSTEM COMPARE TO THE ORIGINAL DESIGN?

A. The design is essentially the same. Due to the increased length of the fuel pool the return lines from the spent fuel pool cooling heat exchangers were lengthened. The spent fuel pool cooling pump internals were also modified to increase the flow rate through the spent fuel pool cleanup filters and demineralizers so that one full pool volume per day could still pass through the purification loop.

The spent fuel cleanup filters and demineralizers were re-sized to accommodate the additional flow.

Q. IS THE PRESENT SPENT FUEL POOL COOLING SYSTEM ADEQUATE TO PROVIDE SUFFICIENT SPENT FUEL POOL COOLING?

A. Yes. Consistent with guidance found in American National Standards Institute (ANSI) N210-1976 "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," the present spent fuel pool cooling system is designed to maintain the temperature of the spent fuel pool below 150°F. (This assumes that both pool cooling trains are operating under abnormal heat load conditions.) The present spent fuel pool cooling system also complies with more recent NRC guidance found in Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis." In addition, it should be noted that the spent fuel pool cooling system is capable of maintaining the temperature of the spent fuel pool below boiling in all cases, normal and abnormal, with the loss of one cooling train.

Q. DEFINE NORMAL HEAT LOAD AND ABNORMAL HEAT LOAD CONDITIONS.

A. As shown in Figure 1 (attached), normal heat load is the postulated maximum achievable heat generation rate (Btu/Hr) resulting from the storage of spent fuel in a Catawba spent fuel pool which maintains a full core reserve.

The abnormal heat load is the postulated maximum achievable heat generation rate resulting from the storage of spent fuel in a spent fuel pool following a full core discharge.

Q. GIVEN THE INCREASED SIZE OF THE SPENT FUEL POOL WHY WERE NOT MODIFICATIONS REQUIRED TO SATISFY THE COOLING SYSTEM DESIGN CRITERIA?

A. Calculations indicated that no change in the cooling system was necessary and that the heat loads of the expanded pool could be met with the originally designed cooling system. This is due primarily to the greatly reduced heat output of spent fuel as it decays with time. For example, the heat output from a spent fuel assembly decreases by a factor of 10 during the first year of decay. It should be noted that an increase in pump flow rate for purposes of the purification loop was made to accommodate the larger volume of water necessary for purification. However, this increased flow rate is not included in the flow to heat exchangers for cooling purposes in cooling response calculations.

Q. DESCRIBE THE COOLING RESPONSE CALCULATIONS.

A. The spent fuel pool temperatures that result from the assumed heat loads described in Figure 1 are graphically presented in Figure 2. These temperatures are calculated by using heat transfer equations for heat exchanger evaluations using the effectiveness method found in standard textbooks. (See Attachment A). The calculations are conservative in that they assume only one mechanism for heat transfer, which is heat exchanger cooling. They also assume that heat exchanger cooling water enters heat exchangers at the maximum temperature of 100°F. No credit is taken in these calculations for convection, evaporation, radiation heat transfer to the fuel pool surroundings, conduction through the pool walls to

the environment, or heat load decay in the spent fuel pool during the heatup times to equilibrium conditions.

Q. HOW DO YOU DETERMINE NORMAL AND ABNORMAL HEAT LOADS?

A. As set forth in Figure 1, the heat loads for various conditions are determined using the NRC decay heat curve (Branch Technical Position ASB 9-2) as set forth in S.R.P. 9.1.3. The 7 day block under the three normal cases described in Figure 1 results from a refueling (i.e. 61 Catawba spent fuel assemblies being transferred to the spent fuel pool all being decayed for 7 days, which is a conservative estimate of the time required to accomplish the transfer of the fuel). The remainder of the normal design heat load and the normal expanded Catawba-only heat load are comprised of additional yearly spent fuel discharges such that all available storage spaces (excluding full core discharge reserve) are filled with Catawba fuel from previous refuelings. This results in an increase in heat load from 13.8 million BTU/HR for the normal fuel pool design to 17 million BTU/HR for the normal expanded case.

For the normal expanded combined case (that is, Catawba, Oconee and McGuire spent fuel in the Catawba spent fuel pool), heat load calculations assume an increase due to other units' (e.g., McGuire and Oconee) spent fuel being added earlier in the decay sequences than was assumed for the Catawba-only sequences. Spent fuel from these stations is assumed to be decayed for 270 days, which is conservative when compared to the stipulated 5-year old spent fuel. It should be noted that if 5-year old spent fuel is assumed, the resulting value will be 17.3 million BTUs/hour rather than the 20.6 million BTUs/hour shown in Figure 1.

For the three abnormal cases shown in Figure 1, the 7 day block now results from a full core discharge (FCD) of 193 assemblies which have been decayed for 7 days. The 25 day block results from an immediate past refueling outage that has decayed for 25 days. Twenty-five days decay for a refueling batch (with an assumed full core discharge) was selected because an evaluation of decay heat curves reflects that an assumed 11 day irradiation time for the full core prior to discharge approximates the maximum combined heat load of the full core discharge plus the immediate past refueling outage.

The remainder of the abnormal heat load for both the current and earlier fuel pool designs is comprised of additional yearly spent fuel discharges such that all available storage spaces are filled with Catawba fuel. As shown in Figure 1, this results in an increase in heat load from 35.9 million BTU's to 39.0 million BTU/HR.

For the abnormal expanded combined case (that is, Catawba, Oconee and McGuire spent fuel pool), heat load calculations assume the 7 day and 25 day blocks discussed immediately above. They also assume an increase due to other units' (e.g., McGuire and Oconee) spent fuel being added earlier in the decay sequences than was assumed for the Catawba-only sequences. Spent fuel from Oconee and McGuire is assumed to be decayed for 270 days, which is conservative when compared to the stipulated 5-year old spent fuel. It should be noted that if 5-year old spent fuel is assumed, the resulting value will be 39.4 million BTUs/hour rather than the 42.7 million BTUs/hour shown in Figure 1.

It should be noted that these heat load calculations are conservative in that they assume (1) that only 7 days is required to refuel the reactor; (2) that the reactor units operate continuously (except during assumed 7 day refueling outage); and (3) that there is maximum burnup of all discharged fuel. In addition, the combined heat load cases assume that the freshest (>270 days) spent fuel available at the McGuire and Oconee sites is transhipped to Catawba rather than the older, cooler, spent fuel available at these sites.

Q. DESCRIBE THE SPENT FUEL POOL TEMPERATURES THAT YOU HAVE CALCULATED.

A. Figure 2 (attached) illustrates the results of taking six of the heat load values presented in Figure 1 and evaluating each of them first with one cooling train in operation and then with two cooling trains in operation. For each of the normal cases it is observed that with one cooling train available the resulting maximum fuel pool temperature is less than 140°F. As would be expected, the increased heat load associated with the expanded pool results in temperatures higher than the original design, but in all cases well below the design basis. Since the design basis is still met, these changes in temperature do not represent a decrease in safety margin, nor have they resulted in a change of fuel pool cooling system capacity.

With respect to the abnormal cases, operation with two cooling trains results in maximum fuel pool temperatures below 150°F. (The two cooling train assumption is consistent with the above-referenced Standard Review Plan and Regulatory Guide). In addition, the

abnormal cases with only one cooling train in operation are shown to be well below 212°F (boiling). Since this additional design consideration is met, these changes in temperature do not represent a decrease in safety margin, nor have they resulted in a change of fuel pool cooling system capacity.

Q. GIVEN THE INCREASE IN THE STORAGE CAPABILITY OF THE EXPANDED FUEL POOL, WHY IS THE TEMPERATURE INCREASE RELATIVELY LOW?

A. The calculated design temperature of the fuel pool is a function of the decay heat load and the performance of the spent fuel pool cooling heat exchangers. Equilibrium conditions are assumed in fuel pool temperature calculations (i.e., for an assumed constant heat load, a constant fuel pool temperature is calculated). At equilibrium, the spent fuel pool cooling heat exchangers are rejecting an amount of heat equal to the spent fuel pool heat load. When additional heat load is added to the spent fuel pool, the temperature of the pool rises, resulting in additional heat being transferred across the heat exchangers. (This assumes constant cooling water temperature and heat exchanger flow rates). A new equilibrium temperature is established when the heat exchanger heat rejection again equals the fuel pool heat load. The relative increase in the calculated equilibrium temperature, however, is not equal to the relative increase in heat load since heat exchangers reject heat more effectively at increased "hot side" temperatures (assuming all other parameters are constant).

As observed in Figure 1, the design heat loads for the expanded pools did not increase proportionately with the pool

volume increase due to the dominating effect of assumed fresh Catawba discharges. This fact, coupled with the increased heat exchanger heat rejection capability at increased fuel pool temperatures, results in calculated pool temperatures for the expanded fuel pool design only slightly increased over the original design.

Q. HAVE YOU ANALYZED THE CONSEQUENCES OF FAILURE OF BOTH COOLING TRAINS, ASSUMING THAT NO MAKEUP WATER IS SUPPLIED AND THE MAXIMUM DECAY HEAT PRODUCTION RATE IS UTILIZED?

A. Yes.

Q. PLEASE EXPLAIN YOUR ANALYSES AND RESULTS.

A. In Figure 3 (attached), the previously described decay heat loads are evaluated to determine the minimum time for onset of boiling and the minimum time for assembly uncovery. The evaluation assumes conservative heat transfer mechanisms of maximum heat generation and no heat losses, such that all of the heat is transferred into increasing the temperature of the water. At the onset of boiling, it is assumed that all of the heat generated is utilized in evaporating the water. Standard heat transfer equations, which are set forth in Attachment A, were used to develop these minimum times. The results of these calculations reflect that for the normal expanded heat load case, at least 25.3 hours will elapse before the onset of boiling, and moreover, that at least 150 hours will elapse before assembly uncovery, assuming no cooling or makeup. With regard to the normal expanded combined mode, about 21.0 hours will elapse before the onset of boiling and about 138 hours will elapse prior to

assembly uncovery. With regard to the abnormal expanded mode, the values are 9.8 and 116 hours, respectively, and with regard to the abnormal expanded combined mode the results are 8.9 hours until the onset of boiling and 106 hours until assembly uncovery.

Q. GIVEN THE ABOVE TIMES TO ASSEMBLY UNCOVERY, CAN ACTIONS BE TAKEN AT CATAWBA TO PROVIDE REASONABLE ASSURANCE THAT ASSEMBLY UNCOVERY DOES NOT OCCUR?

A. Yes. There are several sources of water that can provide makeup to the spent fuel pool. These sources include the Refueling Water Storage Tank (FWST), which can be used for normal makeup requirements, and the assured fuel pool makeup source of Nuclear Service Water (lake water), which provides fully redundant makeup in the unlikely event of loss of all cooling capacity. Both of the above referenced sources are classified as Nuclear Safety Related sources of makeup. The 395,000 gallon Refueling Water Storage Tank can provide a gravity fed source of borated water at a rate sufficient to maintain the fuel pool water level during a postulated boiling event to the extent of the tank volume.

The assured fuel pool makeup source can provide virtually unlimited makeup to the spent fuel pool at a rate well in excess of the maximum expected boiloff rate, which is less than 100 gallons per minute (gpm). Also, a half full Storage Tank can provide makeup at approximately 370 gpm and the assured makeup source can provide makeup at approximately 500 gpm from each of two independent redundant trains. The 500 gallons/minute is available from the Catawba River, which flows at an average rate of ~1.97

million gpm. It is my understanding that these sources can be called upon well within the time calculated to assembly uncovery.

CRITICALITY

Q. HAVE YOU PERFORMED A DESIGN SPECIFIC CALCULATION OF THE REACTIVITY OF THE EXPANDED CATAWBA SPENT FUEL POOL?

A. Yes.

Q. PLEASE EXPLAIN YOUR CALCULATION AND RESULTS.

A. As set forth in FSAR Section 9.1.2.3.1, two criticality analyses were performed, one for Catawba/McGuire spent fuel (which are identical), and one for Oconee spent fuel. For the Catawba/McGuire spent fuel case, the assumptions are: (1) an initial enrichment of 3.5 weight percent U_{235} (this assumes no credit for boron concentration); (2) infinite storage arrays in lateral directions to establish the "worst case" K_{eff} for the storage rack configuration; and (3) 13½" center to center spacing. For the Oconee analysis, the assumptions were identical except that the initial enrichment of 3.3 weight percent U_{235} was assumed. The methodology outlined in Standard Review Plan 9.1.2, "Spent Fuel Storage"; ANSI N210, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"; and ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants" was used. This methodology was used for both the Catawba/McGuire and the Oconee cases. It should be noted that this methodology directs consideration of:

a. Accidental tipping, falling or dropping of a spent fuel assembly;

b. Accidental tipping, falling or sliding of storage racks during fuel transfer or during seismic events;

c. Stuck fuel assembly/crane uplift forces,

d. Objects that may fall on stored fuel assemblies.

The analyses considered these accident situations. In each of these accident situations, the normal storage configurations (~13.5" center-to-center spacing in racks) with the assumed infinite array and assumed lack of boron yield the highest K_{eff} values. The calculated worst case K_{eff} values for these assumptions of each specific fuel type are:

Catawba and McGuire fuel - $K_{eff} = 0.922$

Oconee fuel - $K_{eff} = 0.915$

These values demonstrate design compliance with the referenced NRC criteria by maintaining a storage array neutron multiplication factor (K_{eff}) $\leq$ 0.95 under all credible normal and accident conditions.

Q. ARE THERE UNCERTAINTIES ASSOCIATED WITH THESE VALUES?

A. Yes.

Q. PLEASE EXPLAIN.

A. Figure 4 (attached) is a graphical representation of the calculated K_{eff} values. Figure 4 illustrates how calculational and geometric uncertainties are treated in a conservative manner.

For the Catawba and McGuire reactivity calculation, the $K_{eff} = 0.922$ value represents the calculated worst case K_{eff} including consideration of uncertainties. The calculational uncertainties contribute 0.024 to this K_{eff} value. The calculational uncertainties (determined by benchmark and statistical studies) are

applied conservatively by assuming they occur in the positive reactivity direction. The assumption that storage rack dimensional tolerances and assembly positioning in the rack (i.e., geometric uncertainties) occur in the worst possible combination contributes 0.014 K_{eff} to the 0.922 value. This is conservative since dimensional tolerances and assembly positioning do not occur in the worst combinations, but tend to cancel out over the storage array.

Figure 4 also illustrates how the conservative assumptions of infinite array and of no boron affect the calculated K_{eff} value. The infinite array assumption contributes ~0.01 to the total 0.922 K_{eff} value. The assumed absence of boron (the normal boron concentration will be 2,000 ppm) in the pool contributes ~0.20 to the total 0.922 K_{eff} value. The K_{eff} for a Catawba storage array containing new (unirradiated) fuel from McGuire or Catawba, therefore, is on the order of 0.674 without the above uncertainties or conservatisms.

The Oconee reactivity calculation worst case K_{eff} value is 0.915. The calculational method uncertainties contribute 0.028 K_{eff} to this value. The contribution due to worst case treatment of dimension tolerances and assembly positioning has not been separately evaluated for Oconee fuel, but is included in the present results. The reactivity contributions resulting from the assumed infinite array and assumed lack of boron are 0.01 and 0.20, respectively. The reactivity value for the storage of new Oconee fuel in the Catawba storage racks is therefore on the order of 0.677 without the above uncertainties or conservatisms but assuming worst case geometry.

It should be noted that additional conservatism exists in the reactivity calculations in that new fuel enrichments are assumed. Spent fuel is depleted in fissile U-235 and represents a much less reactive configuration in storage arrays than that assumed in the calculations.

Q. DO THESE VALUES PROVIDE REASONABLE ASSURANCE THAT THE PUBLIC HEALTH AND SAFETY IS PROTECTED?

A. Yes. They are below the values suggested in above-referenced NRC regulatory guidance which establishes 0.95 as a maximum K_{eff} value.

Q. WHAT IS SIGNIFICANT ABOUT THE 0.95 K EFFECTIVE VALUE?

A. There is an arbitrary factor of safety between 0.95 and 1.00 K_{eff}. It should be noted that criticality does not occur until 1.00 K_{eff} is reached. Accordingly, any value below 1.00 K_{eff} is subcritical.

Q. WHAT STEPS HAVE BEEN TAKEN IN THE DESIGN OF THE SPENT FUEL POOL COOLING SYSTEM TO KEEP WORKER EXPOSURE AS LOW AS REASONABLY ACHIEVABLE (ALARA)?

A. Steps taken to reduce worker exposure include the following. First, the depth of the water in the spent fuel pools provides shielding. Also, the provision for filtration and demineralization removes fission products. Third, the spent fuel pool heating, ventilation and air-conditioning system removes radioactive particulates in the spent fuel pool area. Fourth, the layout of the piping and components in the spent fuel pool cooling and cleaning system is such that worker exposure is controlled. Fifth, radiation monitors are provided in the spent fuel pool area to alert plant personnel to potential radiation hazards.

I hereby certify that I have read and understand this document and believe it to be my true, accurate and complete testimony.

A. Lowell Snow

A. Lowell Snow

Sworn to and subscribed before me this 30 day of September, 1983.

Jo Ann D. Bowman

Notary Public

Commission Expires 7-12-88

FIGURE 1 - HEAT LOAD CASES

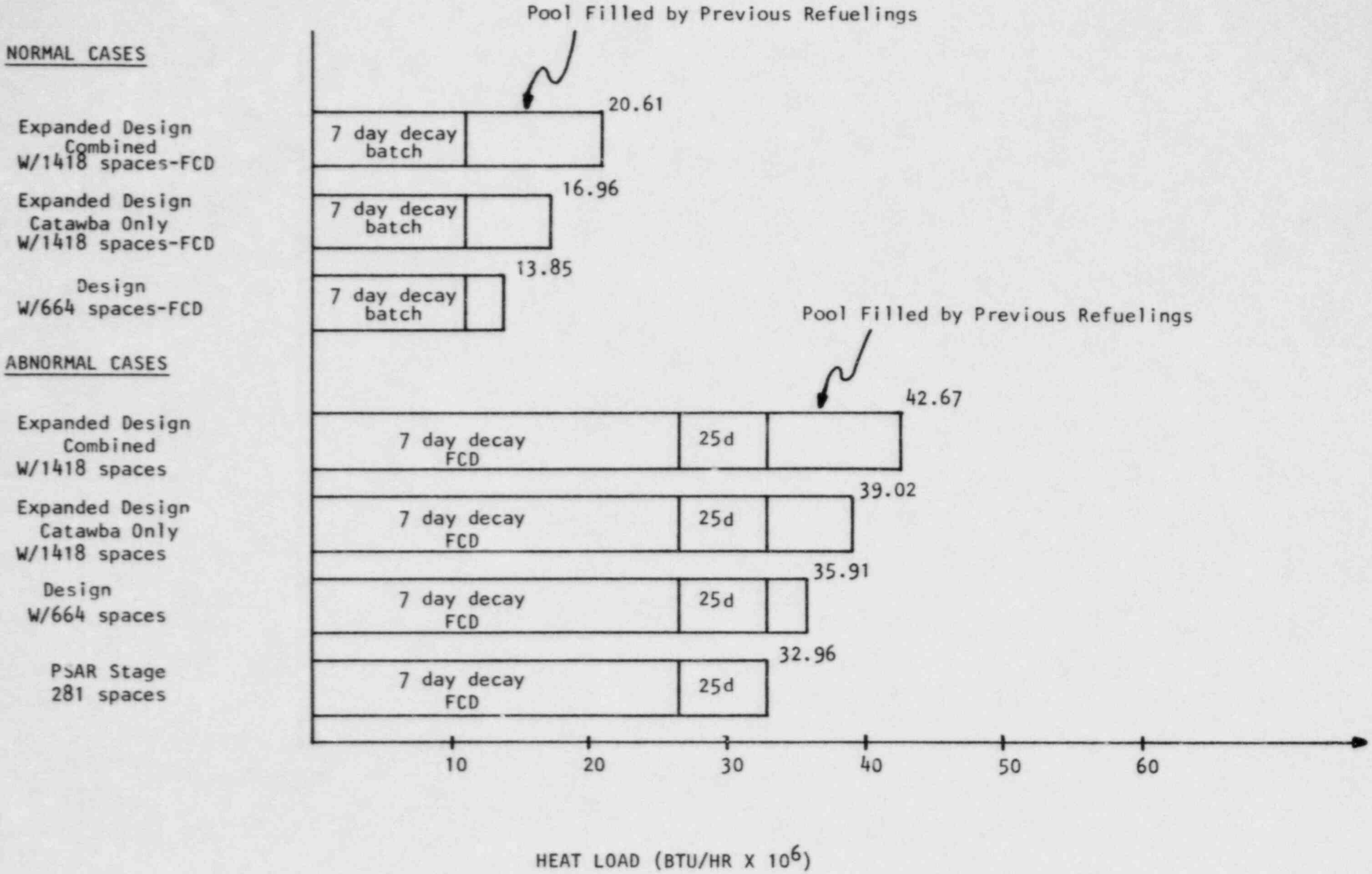


FIGURE 2 - FUEL POOL COOLANT TEMPERATURE CASES

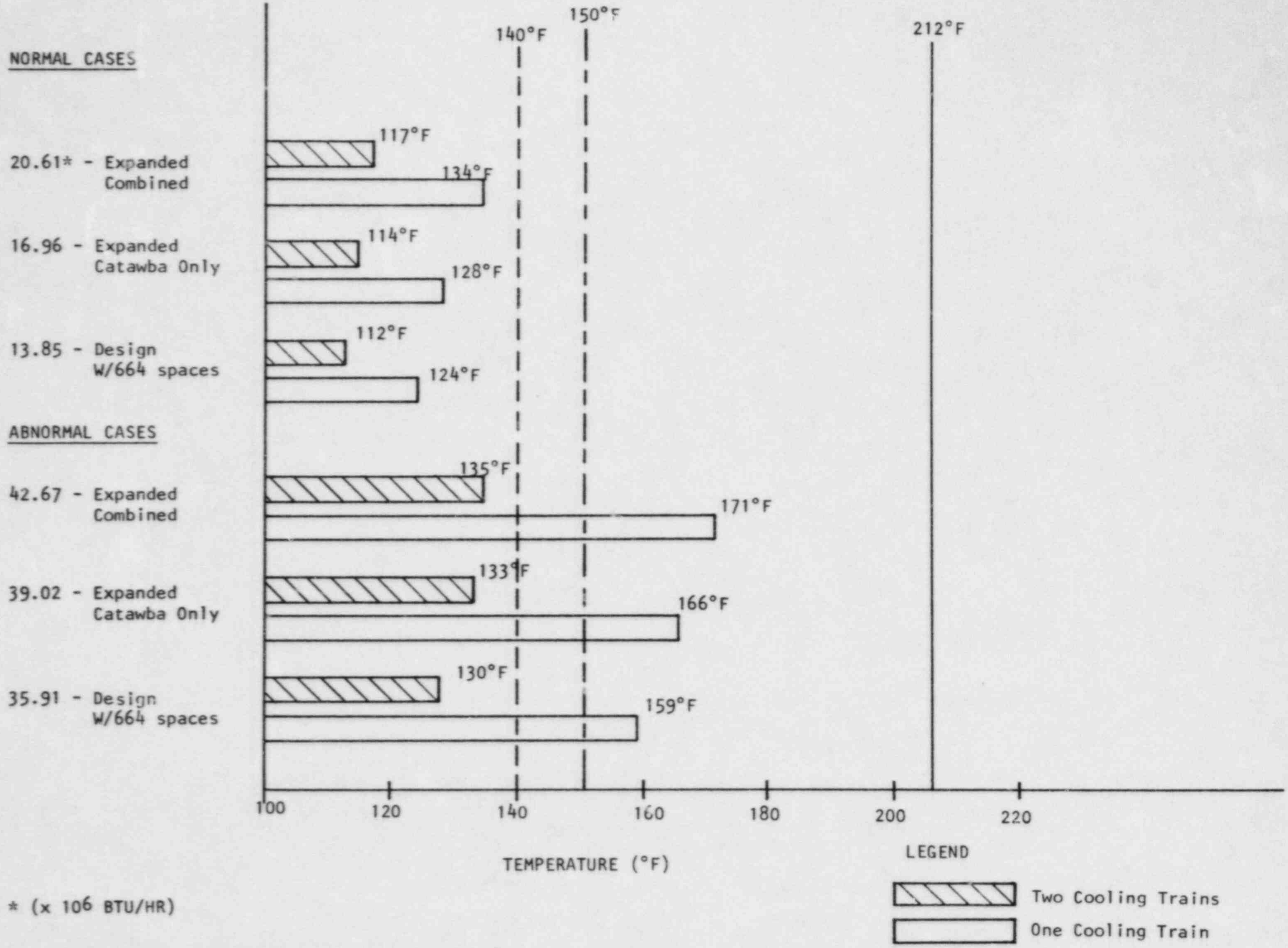
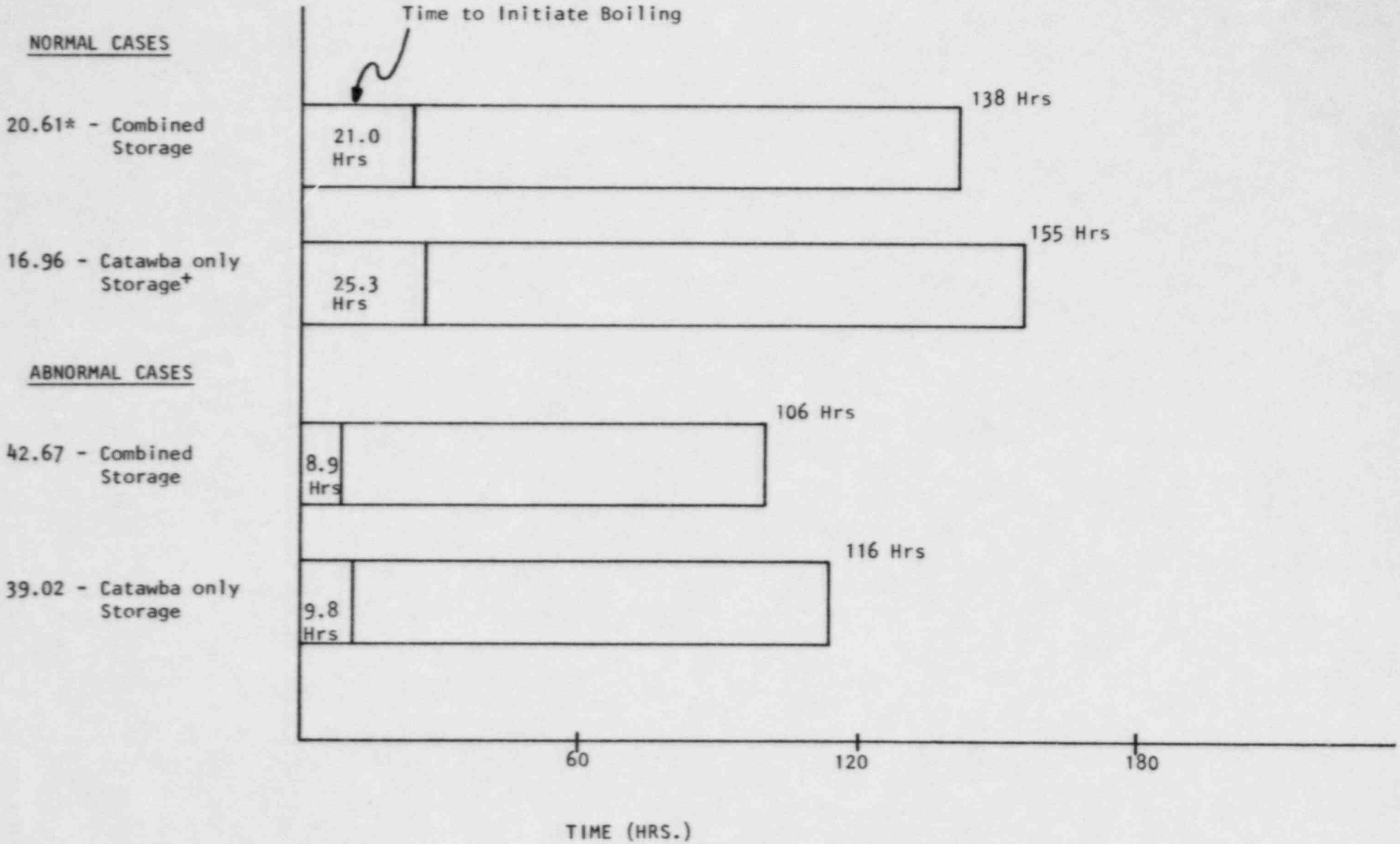


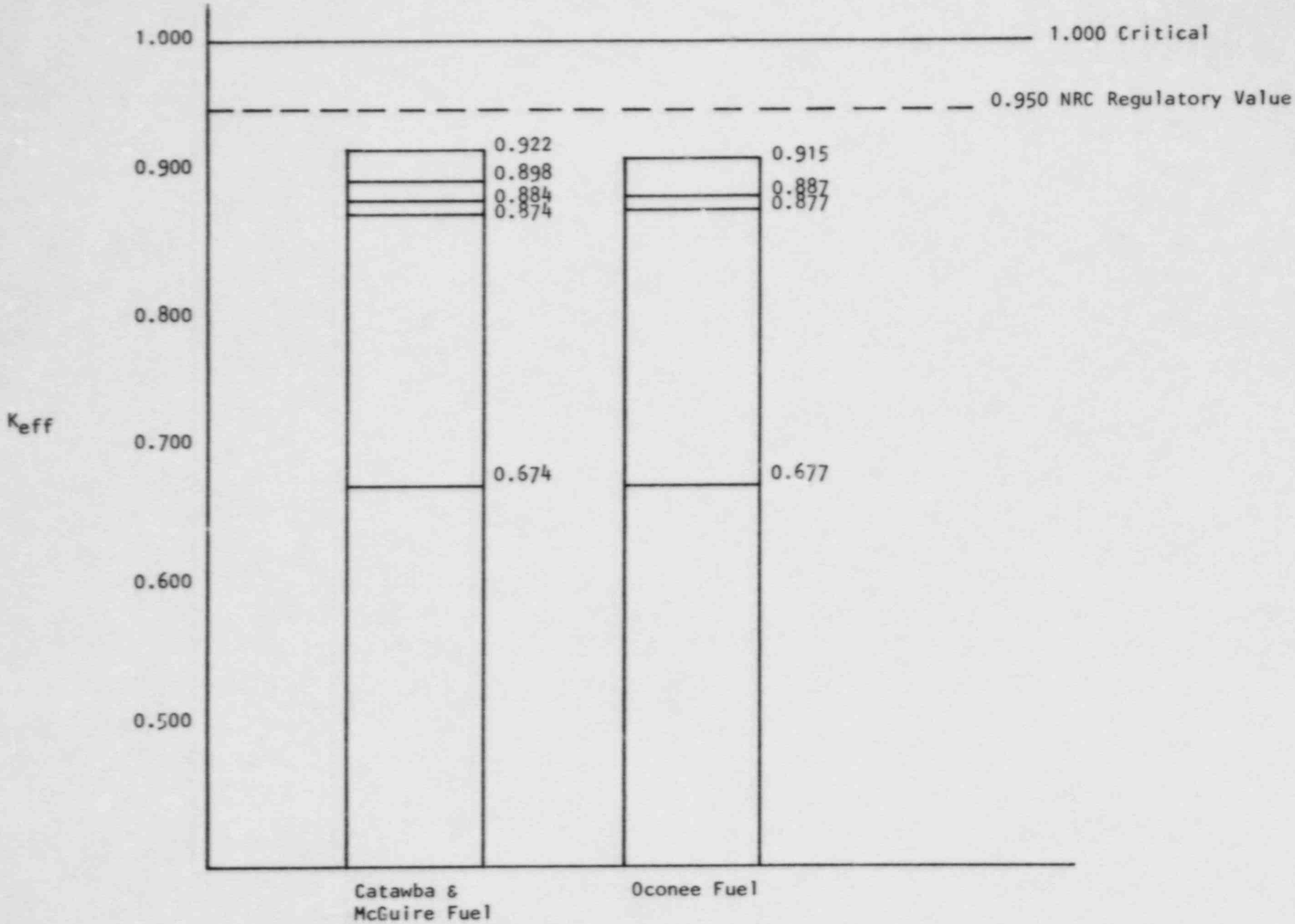
FIGURE 3 - LOSS OF COOLING/MAKEUP CASES



* ($x10^6$ BTU/HR)

+ FSAR Section 9.1.3.3.1 Reference Case

FIGURE 4 - REACTIVITY CALCULATIONS



Attachment A - Thermodynamic and Heat Transfer Equations

Basic Mass, Energy and Temperature Relationship for Adiabatic Heatup

1) $Q = MCp\Delta T$

where: Q = Energy gained or lost from system (Btu).

M = Mass of water in system (lbm).

Cp = Specific heat of water $(\frac{Btu}{lbm\text{-}°F})$.

ΔT = The total change in temperature of a system of mass M resulting from the addition or loss of a quantity of heat Q(°F).

The above equation can be modified to determine the change in temperature for a fluid stream through a heat exchanger:

2) $\dot{Q} = \dot{m}Cp\Delta T$

where: $\dot{Q}$ = Energy addition or rejection rate $(\frac{Btu}{Hr})$.

$\dot{m}$ = Mass flow rate through heat exchanger $(\frac{lbm}{Hr})$.

ΔT = Temperature difference of a fluid stream resulting from an energy gain or loss (°F).

Basic equation for boiloff rate determination

3) $\dot{Q} = \dot{m}(h_v - h_f)$

where: $\dot{Q}$ = Energy addition rate (i.e. Heat Load) $(\frac{Btu}{Hr})$.

$\dot{m}$ = Boiloff rate $(\frac{lbm}{Hr})$.

$(h_v - h_f)$ = Enthalpy (total energy) increase required for vaporization at 212°F. $(\frac{Btu}{lbm})$.

The Heat Exchanger effectiveness of a shell and tube HX w/one shell and even number of tube passes is given below. Effectiveness is defined as the ratio of actual heat exchanger heat transfer capability to the maximum theoretical heat transfer possible for given heat exchanger cooling water and hot fluid inlet temperatures.

4) $\varepsilon = 2\left\{1 + C + (1+C^2)^{\frac{1}{2}} \dfrac{1 + \exp[-N(1+C^2)^{\frac{1}{2}}]}{1 - \exp[-N(1+C^2)^{\frac{1}{2}}]}\right\}^{-1}$

where: ε = Effectiveness

$C = \dfrac{\dot{m}h}{\dot{m}c}$ - ratio of hot side (pool water to be cooled) mass flow rate to cooling water flow rate.

N is given by the relationship:

$$N = \frac{UA}{\dot{m}_c C_p}$$

where: U = Heat exchanger overall heat transfer coefficient. This value is referenced by the heat exchanger manufacturer $\left(\frac{Btu}{Hr\text{-}Ft^3\text{-}°F}\right)$.

A = Effective heat exchanger heat transfer area. This value is referenced by the heat exchanger manufacturer (Ft^2).

Once the value of ε for the specific heat exchanger is calculated by equation 4 the various inlet and outlet heat exchanger temperatures are determined using the following relationships and assumed values:

a) $\varepsilon = \dfrac{T_{hI} - T_{ho}}{T_{hI} - T_{cI}}$

b) $\dot{Q} = \dot{m}_c Cp\ (\Delta Tc) = \dot{m}_h Cp(\Delta T_h)$ (equation 2)

c) $\dot{Q} = \dot{m}_h Cp(\varepsilon(T_{hI} - T_{cI}))$

where: T_{hI} = Hot fluid inlet temperature (assumed fuel pool bulk water temperature) (°F).

T_{cI} = Cooling water inlet temperature. This temperature is set to the assumed maximum cooling water temperature of 100°F.

T_{ho} = Hot fluid outlet temperature (temperature of fluid being returned to pool) (°F).

Since $\dot{m}_h Cp$, ε and T_{cI} are known, an expression for the equilibrium fuel pool bulkwater temperature (Th_I) can be written:

5) $T_{Fuel\ Pool} = T_{hI} = \left(\dfrac{\dot{Q}}{\dot{m}_h Cp \varepsilon}\right) + 100°F$

The equations for the effectiveness method of evaluating heat exchanger performance can be found in:

J. P. Holman, Heat Transfer, 4th ed. (New York: McGraw-Hill Book Co., 1976), Chapter 10, Section 6.

DUKE POWER COMPANY

ARTHUR LOWELL SNOW

EDUCATION: B.S., Nuclear Engineering, University of Tennessee
M.E., Mechanical Engineering, University of South Carolina
Additional Courses:
Graduate Course Work in Mechanical Engineering toward PhD

CERTIFICATIONS/PROFESSIONAL AFFILIATIONS

Professional Engineer - North Carolina 7397
- South Carolina 6145
Member of the American Nuclear Society

YEARS
EXPERIENCE: 15

SUMMARY OF PERTINENT EXPERIENCE

- Supervisor, Mechanical and Nuclear Division, Nuclear Activities - All Duke Nuclear Power Stations: System-wide radwaste design review activities, licensing activities, probabilistic risk assessment and safety reviews, evaluation of nuclear accident scenario and corrective actions, radioactive effluents analysis, and nuclear fuel criticality analysis for spent fuel storage designs.

- Supervisor, Pipe Support/Restraint Design - Design, engineering and constructability of ASME III and B31.1.0 piping support/restraints for Catawba Nuclear Station. Development of design criteria and specifications, technical contract administration, scheduling and coordination of design activities.

- Assistant Design Engineer - Design and engineering of nuclear fluid process systems for Catawba Nuclear Station, review of operating procedures, testing procedures, start up assistance, operating parameters and cost evaluations.

- Assistant Design Engineer - Development of computer codes for radiation shielding, radioactive liquid and gaseous discharge. Preparation of Safety Analysis Reports/Environmental Reports. Radiation shielding designs. All this work for Oconee, McGuire and Catawba Nuclear Stations.

EXPERIENCE: DUKE POWER COMPANY since 1968

1979 to Present: Design Engineer II - In charge of Nuclear Sub-Group responsible for: system-wide radwaste design review activities; technical interface for Mechanical/Nuclear Division with licensing, probabilistic risk assessment and safety review groups; radioactive effluent analysis for normal and accident conditions; nuclear fuel criticality and generic engineering activities. Generic engineering activities include steam generator chemical cleaning, technical review of responses to TMI concerns, other regulatory and quality assurance matters.

ARTHUR LOWELL SNOW (Cont'd)

1977-1979	Design Engineer/Assistant Design Engineer - In responsible charge of Pipe Support/Restraint Group for Catawba. Multi-discipline group of Duke and contract personnel included clerks, draftsmen, designers, Mechanical and Civil Engineers (B.S., M.S., and PhD's). Activities included setting up initial organization, design criteria and specification preparation, contract administration, scheduling, interface with Construction Department, and other Design groups. Group produced in excess of 30,000 designs for ASME Section III, Class 2 and 3, and ANSI B31.1.0 piping systems and supports for HVAC seismically designed ducting.
1972-1977	Assistant Design Engineer - Responsible charge of fluid systems design for all Catawba nuclear process systems. Supervised preparation of Flow Diagrams, Design Criteria, System Descriptions, Data Sheets for Mechanical Equipment, Safety Analysis Report preparation for Mechanical/Nuclear systems. Developed operating parameters, costs, review of testing/operating procedures, piping system start up assistance.
1968-1972	Assistant Design Engineer/Associate Engineer/Jr. Engineer - Responsible charge of Radiation Analysis Group. Activities included direct effort and supervisory responsibility for: development of radiation shielding computer codes, development of radioactive liquid and gaseous discharge computer codes, cost evaluation of Turbine-Generator bids using part load heat rates, Safety Analysis Report and Environmental Report preparation, response to NRC (then AEC) questions, and appearances before ACRS and NRC Staff for Oconee, McGuire and Catawba Nuclear Stations, design radiation shielding for McGuire and Catawba.
PUBLICATIONS:	"Criteria For Evaluation of Interim Radwaste Solidification Systems" '83 Waste Management Symposium, Tuscon, Arizona 2/28/83.

NUCLEAR REGULATORY COMMISSION

Docket No. 50-413 Official Exh. No. 91

In the matter of Catawba

Staff ______ IDENTIFIED ✓

Applicant ✓ RECEIVED ✓

Intervenor ______ REJECTED ______

Cont'g Off'r ______

Contractor ______ DATE 12/8/83

Other ______ Witness ______

Reporter Mary Simond