TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

September 11, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Enclosed is additional information concerning the comparison of the Watts Bar and Sequoyah Nuclear Plants' Containment Systems. This evaluation of the Final Safety Analysis Report section 6.2 provides comparisons of the following.

6.2.1	Containment Functional Analysis
6.2.5	Combustible Gas Control and Hydrogen Purge
6.2.6	Containment Leak Rate Testing
6.2.7	Containment Isolation

Information in response to the NRC's September 2, 1981 informal request for a comparison of the containment heat removal systems is being compiled and will be forwarded upon completion.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 11th day of Sept 1981
Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

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Enclosure

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ENCLOSURE
WATTS BAR NUCLEAR PLANT UNITS 1 AND 2
WATTS BAR-SEQUOYAH CONTAINMENT SYSTEMS COMPARISON

6.2.1 Containment Functional Analysis

The primary containment at SQN and WBN are of the same basic design and have been functionally analyzed by essentially the same method. The prime design differences are the following:

	SQN	WBN
Containment Design Pressure	12 psig	15 psig
Net External Design Pressure	.5 psig	2.0 psig
Vacuum Relief System	Yes (auto)	No (manually open if reqd)
Containment Spray Flow Rate	4750 gpm	4000 gpm
Thermal Power	3423 MWT_6	3411 MWT
Design Blowdown Energy	334.6×10^6 Btu	318×10^6 Btu
Design Blowdown Mass	543300 LBm_3	493000 LBm_3
Containment Net Free Volume	1191500 ft^3	1191414 ft^3
Containment Heat Sinks	Slight differences in each	

The primary difference in the functional analysis concerns the MSLB analysis. The WBN analysis was a plant specific analysis whereas the SQN analysis was based on a comparison to WBN. The methodology used in the LOCA analysis for each plant is the same.

All of the key differences in containment parameters used in each LOCA and MSLB analysis are found in NRC questions Q6.56 and Q6.56A for Sequoyah. These questions and responses are attached.

6.56 Describe and justify the analytical model used to conservatively determine the maximum containment temperature and pressure for a spectrum of postulated main steam line breaks for various reactor power levels. Interim staff positions regarding postulated main steam line breaks are discussed in NUREG-0138, Issue No. 1 and NUREG-0153, Issue No. 25. Include the following in the discussion:

a. Provide single active failure analyses which specifically identify those safety grade systems and components relied upon to limit the mass and energy release and containment pressure/temperature response. The single failure analysis should include, but not necessarily be limited to: main steam and connected systems isolation; feedwater, auxiliary feedwater, and connected systems isolation; feedwater, condensate, and auxiliary feedwater pump trip, and auxiliary feedwater run-out control system; the loss of or availability of offsite power; diesel failure when loss of offsite power is evaluated; and partial loss of containment cooling systems.

b. Discuss and justify the assumptions made regarding the time at which active containment heat removal systems become effective.

c. Discuss and justify the heat transfer correlation(s) (e.g., Tagami, Uchida) used to calculate the heat transfer from the containment atmosphere to the passive heat sinks, and provide a plot of the heat transfer coefficient versus time for the most severe steam line break accident analyzed.

d. Specify and justify the temperature used in the calculation of condensing heat transfer to the passive heat sinks; i.e., specify whether the saturation temperature corresponding to the partial pressure of the vapor, or the atmosphere temperature which may be superheated was used.

e. Discuss and justify the analytical model including the thermodynamic equations used to account for the removal of the condensed mass from the containment atmosphere due to condensing heat transfer to the passive heat sinks;

f. Provide a table of the peak values of containment atmosphere temperature and pressure for the spectrum of break areas and power levels analyzed;

g. For the case which results in the maximum containment atmosphere temperature, graphically show the containment atmosphere temperature, the containment liner temperature, and the containment concrete temperature as a function of time. Compare the calculated containment atmosphere temperature response to the temperature profile used in the environmental

qualification program for those safety-related instruments and mechanical components needed to mitigate the consequences of the assumed main steam line break and effect safety reactor shutdown;

h. For the case which results in maximum containment atmosphere pressure, graphically show the containment pressure as a function of time; and

i. For the cases which result in the maximum containment atmosphere pressure and temperature, provide the mass and energy release data in tabular form.

j. For the instrumentation and equipment located inside the containment and required to (1) detect the steam line break; (2) initiate safety systems and (3) monitor the course of the accident, provide the following:

1. A description of the tests which were/or will be performed to show that this instrumentation and equipment are/or will be qualified to perform their function before, during, and after the accident. Include the spectrum of environmental conditions for which tests were/will be performed and state the acceptance criteria. The instrumentation and equipment to be considered includes, but is not limited to the following: (a) pressurizer pressure and level sensors and transmitters; (b) steam generator pressure and level sensors and transmitters; (c) main steam line pressure, differential pressure and flow sensors and transmitters; (d) primary system hot leg and cold leg temperature sensors and transmitters; (e) primary system pressure sensors and transmitters; (g) feed water flow sensors and transmitters; (h) containment pressure sensors and transmitters; (i) valve operators and position switches; (j) electrical cables, motors and penetrations; (k) containment coolers. Also identify any additional instruments and equipment required.

2. A description of the separation and independence between redundant sensors, cables and other equipment associated with each steam generator and steam line.

3. A description of the independence and separation between each steam generator and between each steam line.

Response:

a. Refer to section 6.2.1.3.11.

b. The active containment heat removal systems are the containment spray system and the air return fans. The air return fans begin

operation 600 seconds after a high-high containment pressure signal, and the containment sprays begin operation 25 seconds after a containment high-high pressure signal. The high-high containment setpoint is 3.0 psig. The containment sprays are discussed in FSAR section 6.2.2.2 and the air return fans in section 6.6. The times at which these systems become effective in the analysis are discussed in the reponse for parts c through i.

c to i. The response to these parts of the question are containment in the following description of the main steam line break MSLB analyses for an ice condenser plant.

MSLB ANALYSIS

Introduction

The LOTIC-3 computer code has been developed to analyze steamline breaks in an ice condenser plant. During the development of this computer code, discussion and justification of the heat transfer coefficients and of the thermodynamic equations have been presented. Details of the LOTIC-3 computer code are given in References 1 to 3. The LOTIC-3 computer code has been found to be acceptable for the analysis of steamline breaks (Reference 4) with the following restrictions:

1. Mass and energy release rates are calculated with an approved model.

2. Complete break spectrums are analyzed.

3. Convective heat flux calculations as described in Reference 2, Q7, are performed for all break sizes.

Two separate condensation models are used by the LOTIC-3 computer. The 100% condensate revaporization model is used for large breaks, and for small breaks the conservative 0% condensate revaporization and convective heat flux models are used. As pointed out in previous LOTIC-3 submittals, this position is felt to be justified. However, it has also been shown that the small steamline break temperature transients are more severe than large break transients, even if the large break calculations assume no revaporization of the condensate and do not take credit for convective heat flux (Reference 3). Rather than performing a plant specific containment steamline accident calculation for the Sequoyah plant, it will be shown that the design temperature transients generated previously are conservative. This will be accomplished by comparing the Sequoyah containment parameters (Reference 6) and the mass and energy releases to those used in the Watts Bar large steamline break analysis (Reference 5), and performing the same type of comparison between the Sequoyah and the typical plants small break parameters. It will be shown that any significant differences between the Sequoyah parameters and the reference case are conservative.

Large Break Analysis

An analysis has not been done specifically for Sequoyah, but a spectrum of large steamline breaks has been analyzed for the Watts Bar plant and reported in Watts Bar FSAR section 6.2.1, which may be referenced for the Sequoyah Nuclear Plant. These blowdowns are dry steam blowdowns representative of a 3425 Mwt plant with Model D steam generators operating a 1000 psia steam pressure. Sequoyah is a 3423 Mwt plant operating at a steam pressure of 860 psia, and has Model 51 steam generators which have considerably less mass inventory than Model D steam generators. Thus the blowdowns are conservative for Sequoyah for breaks occurring downstream of the steam line flow restrictor. Breaks upstream of the steam line flow restrictor have not been analyzed for Sequoyah and were not analyzed for Watts Bar since the Watts Bar Plant has flow restrictors integral to the steam generator shell. However, significant differences in the temperature transient for these blowdowns are not expected, since it has also been shown (Reference 3) that small steam line breaks produce more severe temperature transients than the large breaks, even if the large break calculations do not assume condensate revaporization or convective heat flux. Thus a re-analysis of the large steam line breaks submitted previously for Watts Bar and now being referenced for Sequoyah or an analysis of the break upstream of the Sequoyah steam line flow restrictors is not necessary. Small breaks, however, must still be investigated.

Containment Parameters

The Watts Bar plant containment parameters have been presented in Watts Bar FSAR, section 6.2.1 and have been reproduced in Table Q6.56-3. The corresponding Sequoyah parameters have been presented in part VI of Sequoyah FSAR Table 6-1 (in Appendix 6T) and are reproduced in Table Q6.56-2. In order to justify the use of the Watts Bar analysis for Sequoyah, a comparison will be made to show that the differences between the physical data (i.e., heat sinks, sprays, etc.) are either insignificant or conservative in nature.

1. Containment net free volume - The Sequoyah plant containment volume is 1,191,500 ft^2, while the Watts Bar volume is 1,191,414 ft^3. The volumes are essentially the same.

2. Structural heat sinks in the major compartments (i.e. upper, lower, ice condenser) - The heat sinks in the lower compartment are of primary importance since this is where the break occurs and consequently has the most severe environment. The temperatures in the lower compartment are those which are used for the qualification of equipment. The heat sinks in the upper and ice condenser compartments are of secondary importance.

The lower compartment comparison illustrates that the Sequoyah and Watts Bar heat sinks are nearly identical.

The areas and volumes for the concrete of the Watts Bar plant are 35553 ft^2 and 49867 ft^3, compared to 35790 ft^2 and 49755 ft^3 for Sequoyah. The average concrete thickness for each of the plants are 1.40 ft for Watts Bar and 1.39 ft for Sequoyah. For the steel heat sink comparison the volume, or thermal penetration, does have a significant effect. The steel heat sink area and volume comparisons are 30770 ft^2 and 1929 ft^3 for Watts Bar compared to 30770 ft^2 and 1920 ft^2 for Sequoyah. The steel heat transfer data comparison shows that the areas and volumes are essentially identical. The heat sinks in the ice condenser are identical for both plant cases.

The last comparison of compartment heat sinks will be the upper compartment, which is also comprised of steel and concrete materials. For the Watts Bar plant, the concrete area and volume are 23900 ft^2 and 32100 ft^3 respectively, compared to 23920 ft^2 and 24943 ft^3 for Sequoyah. The steel heat transfer are and volume for Watts Bar is 60170 ft^2 and 4586 ft^3 compared to 56060 ft^2 and 2365 ft^3 for Sequoyah. Although the Watts Bar plant does have a greater upper compartment structural heat removal capability, this is of secondary importance in the analysis since the ice condenser allows very little steam into the upper compartment, and the spray system has the capability to cool the upper compartment. Consequently, the upper compartment's environment is not a severe one and has little impact on the analysis.

3. Spray parameters - The spray pump flows are 4750 gpm at 95°F for Sequoyah and 4000 gpm at 100°F for Watts Bar. Since the flows for Sequoyah are greater and the temperature lower, application of the Watts Bar parameters is conservative with respect to Sequoyah.

In conclusion, the Watts Bar large steam line break temperature transients are more conservative (result in higher calculated containment temperatures) than those which would be calculated specifically for Sequoyah. Included are tables which give the volumes, areas, materials and a description of the compartment.

Small Break

The method used for this evaluation will be the same as that used for the large break. It will be shown that differences between the generic containment parameters and blowdowns, and those for Sequoyah are either conservative or unimportant. Rather than performing a plant specific small steamline break calculation for Sequoyah, this evaluation will justify that referencing the generic LOTIC-3 work is conservative.

These transients were generated for a typical 4-loop, 3425 Mwt, Westinghouse design nuclear plant. Specific characteristics of the plant design assumed when calculating these blowdowns are given in section 3 of Reference 5 and are representative of the Sequoyah Plant design. The break sizes used in the LOTIC-3 report are the largest split ruptures which do not generate a steam line isolation signal due to low steam line pressure/high steam flow conditions (See Section 2.3 of Reference 5). See the response to Q6.56A for a discussion of the containment temperature response to a spectrum of small split breaks. Since the Sequoyah plant has the same type of protection system, these break sizes are also appropriate for their design. 61

Key parameters which should be compared to ensure the conservatism of these blowdowns for Sequoyah are shown in Table Q6.56-4. As can be seen from this table, the initial steam pressure for Sequoyah is lower making the LOTIC-3 blowdowns greater than those expected for Sequoyah. Also the power levels for both analyses are essentially equivalent, and the expected steam and feed line isolation signals would occur much earlier for the Sequoyah plant than was assumed in the LOTIC-3 analyses. The results of the generic plant analyses are presented in Figures Q6.56-1, -2, -3. Thus, it can be concluded that the small break analyses in the LOTIC-3 report are conservative compared to a Sequoyah nuclear plant small steam line break analyses. 61

Beginning with the total containment net free volume, each compartment's heat sinks will be evaluated by its heat transfer area and volume.

1. Containment net free volume - The Sequoyah total net free volume is 1,191,500 ft^3. The generic plant's volume is 1,193,971 ft^3. The difference is 2470 ft^3, which is less than 0.5% and therefore will not have any significant affect on the analysis.

2. Structural heat sinks in the major compartments (i.e. upper, lower, ice condenser) - The heat sinks in the lower compartment are of primary importance since this compartment is where the break occurs and consequently sees the most severe environment. The temperatures in the lower compartment are those which are used to qualify equipment. The heat sinks in the upper and ice condenser compartments are of secondary importance.

The lower compartment comparison illustrates the conservatism of the generic plants heat sinks. The areas and volumes for the concrete of the generic plant are 25670 ft^2 and 47808 ft^3 compared to 35790 ft^2 and 49755 ft^3. This comparison reveals a greater concrete heat transfer area and heat removal capability for Sequoyah, which would result in a lower calculated peak temperature. Likewise, the steel heat sink comparison display a greater heat transfer area and heat removal capability. The generic plant's parameters are 3955 ft^2 and 167 ft^3 compared to 30770 ft^2 and 1920 ft^2 for Sequoyah. The heat sinks in the ice condenser are identical for both plant uses.

The last comparison of compartment heat sinks is in the upper compartment. The generic plants upper compartment concrete area and volume are 26123 ft^2 and 41722 ft^3 respectively, compared to 23920 ft^2 and 29942 ft^3 for Sequoyah. The steel heat transfer areas and volumes are 41302 ft^2 and 2034 ft^3 for the generic plant and 56060 ft^2 and 23645 ft^3 for Sequoyah. Although the generic plant does have a greater upper compartment structural heat removal capability, this is of secondary importance in the analysis since the ice condenser allows very little steam into the upper compartment, and the spray system has the capability to cool the upper compartment. Consequently, the upper compartment's environment is not a severe one and has little impact on the analysis.

3. Spray parameters - The spray pump flows at 4750 gpm at 95°F for Sequoyah and 3400 gpm at 100°F for the generic plant. Since the Sequoyah flows are greater than that of the generic plant and the temperature is cooler, application of the generic parameters is conservative with respect to Sequoyah.

In conclusion, the generic plant's small steamline break temperature transients are more conservative (result in higher calculated containment temperatures) than those which would be calculated specifically for Sequoyah.

j. The instrumentation inside containment which provides automatic protective signals for a main steam line break is (1) pressurizer pressure; (2) pressurizer level; (3) and steam line flow. The instrumentation inside containment which allows the operator to monitor the steam line break accident is (1) narrow range steam generator level; (2) pressurizer level; (3) reactor coolant system wide-range pressure; and (4) reactor coolant system wide-range temperature.

Qualification type testing for the protype pressure and differential pressure transmitters to be used for the above instrumentation is described in Westinghouse letter NS-CE-1384 (Eicheldinger to Stolz, March 23, 1977). Qualification type testing of the resistance temperature detector's for the reactor coolant system is described in section 3-4 of WCAP 9157, "Environmental Qualification of Safety Related Class 1E Process Instrumentation." The qualification testing program for the valve motor operator and position switches is described in Westinghouse letter NS-CE-692. TVA-purchased valve operators are identical to the Westinghouse supplied operators described in NS-CE-692. TVA-purchased Namco limit switches were qualified to IEEE Standard 382-1972 and were subjected to thermal and mechanical aging, radiation exposure, LOCA, and MSLB environmental tests. The results of the test are in Acme Cleveland Development Company Qualification reports on Namco limit switches, models EA-180 and EA-740. The tests for TVA-purchased air return fans are described in a reprint by Joy Manufacturing Company of a paper presented at the 10th Electrical Insulation Conference held September 20-23, 1971. The tests included accelerated heat aging and pressure temperature transients in a saturated steam, chemical mixture environment. 59

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The temperature inside containment for breaks in the main steam line will not exceed 327°F. The temperature is calculated for the worst case small break which is the controlling case for containment temperature response. This calculation was done for the generic plant whose parameters are conservative with respect to or are insignificantly different from those at Sequoyah. All safety-related electrical equipment that must perform in this environment is identified in FSAR table 3.11-3. Manufacturer model, and type numbers are identified in 59

all test reports. With the exceptions given below, all the testing subjected the equipment and instruments to temperature ranging from 330°F to 370°F. In most of these tests, these temperatures were brought about by rapid temperature increase requirements which resulted in the actual temperature profile overshooting the required maximum temperature. In other cases, unplanned elevated temperatures resulted in conservative overtesting. The actual testing of this equipment at or above 330°F qualifies it for service at 327°F. 59 59

Testing of the penetrations was performed to ensure seal integrity and insulation quality after being subjected to "accident" conditions for temperature and pressure. All penetrations were subjected to a temperature in excess of 325°F except the triax which was subjected to 323°F. This small difference was considered insignificant. The tests demonstrated penetrations integrity for the MSLB occurrence. The results of the test are in the following reports:

Low voltage power - PEN-RLK-3-16-01

Medium voltage power - PEN-ACD-4-72-03

Low voltage control and instrumentation - PEN-RLK-3-26-73 59

Triax (neutron monitoring) - Westinghouse triax incident report

The testing programs for electrical cables are described in a large number of separate reports. The test results have been summarized in Table Q6.56-5. 60

Recent testing developments and evaluations for production run pressure and differential pressure transmitters has shown these instruments to be acceptable for use at Sequoyah inside containment. An evaluation of the separation and independence of equipment necessary to mitigate the effects of a main steam line break inside containment was done according to requirements of the pipe break evaluation inside containment. Sensors, cables, and other equipment, were included in the evaluation. The safety-related instrumentation is channelized and the separation and independence requirements are found in FSAR section 7.1.2.2. See section 3.6 of the FSAR for details of the evaluation criteria. Where unacceptable interactions were identified, sensing lines, conduit, and other equipment were moved or protected to provide the required separation and redundancy of mitigation equipment. See Figure 1.2-11 for the physical layout of the steam generators and main steam lines. Independence between steam generators, as required by the plant safety analyses, is provided by the system design, the physical layout, and by modifications made as a result of the pipe break evaluation inside containment. See section 3.6 of the FSAR for details of the pipe break evaluation. 61 59

REFERENCES

1. C. Eicheldinger, Letter of 10/22/76, #NS-CE-1250.

2. C. Eicheldinger, Letter of 6/14/77, #NS-CE-1453.

3. C. Eicheldinger, Letter of 12/7/77, #NS-CE-1626.

4. John F. Stolz, Letter of 5/3/78, "Evaluation of Proposed Supplement to WCAP-8354 (LOTIC-3)".

5. Land, R. E., "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822, September 1976.

TABLE Q6.56-1

GENERIC ICE CONDENSER PLANT DESIGN PARAMETERS

1. VOLUME

Reactor Containment Volume (Net free volume, ft^3)

Upper Compartment	670,101
Upper Plenum	47,000
Ice Condenser	86,300
Lower Plenum	24,200
Lower Compartment (Active)	235,481
Lower Compartment (Dead Ended)	130,899
Total Containment Volume	1,193,971

Tech Spec Weight of Ice in Condenser, lbs	2.45×10^6

Added by Amendment 58
December 22, 1978

TABLE Q6.56-1 (Continued)

2. STRUCTURAL HEAT SINKS

	Area (ft^2)	Material and Thickness (ft)	
A. Upper Compartment			
1. Polar Crane Wall, Containment Shell, and Miscellaneous Steel			
Slab 1	8915	0.000583 0.01017	Paint Carbon Steel
Slab 2	31667	0.000583 0.05758	Paint Carbon Steel
Slab 3	720	0.00167 0.1670	Paint Carbon Steel
2. Refueling Canal and Miscellaneous Concrete			
Slab 4	25443	0.00167 1.511	Paint Concrete
Slab 5	680	0.00167 4.82	Paint Concrete
B. Lower Compartment			
1. Platforms			
Slab 1	1,375	0.000583 0.007813	Paint Carbon Steel
2. Steam Generator Supports and Reactor Coolant Pump Supports			
Slab 2	1,580	0.00583 0.0605	Paint Concrete
3. Miscellaneous Concrete			
Slab 3	23,300	0.00167 1.645	Paint Concrete

Added by Amendment 58
December 22, 1978

TABLE Q6.56-1 (Continued)

STRUCTURAL HEAT SINKS

	Area (ft^2)	Material and Thickness (ft)	
4. Reactor Cavity			
Slab 4	2,370	0.00167 4.0	Paint Concrete
5. Base Floor*			
Slab 5	4,228	0.00167 2.0	Paint Concrete
C. Ice Condenser			
1. Ice Baskets			
Slab 1	180,628	0.00663	Steel
2. Lattice Frames			
Slab 2	76,650	0.0217	Steel
3. Lower Support Structure			
Slab 3	28,670	0.0267	Steel
4. Ice Condenser Floor			
Slab 4	3,336	0.000833 0.333	Paint Concrete
5. Containment Wall Panels and Containment Shell			
Slab 5	19,100	1.0 0.0625	Steel & Insulation Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 6	13,055	1.0 1.0	Steel & Insulation Concrete

* In contact with sump.

Added by Amendment 58
December 22, 1978

TABLE Q 6.56-2

SEQUOYAH ICE CONDENSER DESIGN PARAMETERS

REACTOR CONTAINMENT VALUE (NET FREE VOLUME)

Upper Compartment, Ft^3	651,000
Ice Condenser, Ft^3	
Lower Plenum	24,200
Ice Bed	86,300
Upper Plenum	47,000
Lower Compartment (active), Ft^3	289,000
Total Active Volume, Ft^3	1,097,590
Lower Compartment (dead ended), Ft^3	94,000
Total Containment Volume, Ft^3	1,191,500
Reactor Containment Air Compressor Ratio	1.43
Reactor Power, MWT	3,582
Design Energy Release to Containment	
Initial Blowdown Mass Release, lb.	543,330
Initial Blowdown Energy Release, BTU	$334.6\ 10^6$
Ice Condenser Parameters	
Weight of Ice in Condenser, lb.	$2.45\ 10^6$

Added by Amendment 58
December 22, 1978

TABLE Q6.56-2 (Continued)

SEQUOYAH
STRUCTURAL HEAT SINKS

Heat Sink	Material	Area (ft^2)	Layer and Thickness (ft)	
Upper Compartment				
1) Operating Deck	Concrete	4,880	1.07	concrete
2) Crane Wall	Concrete	18,280	0.0005 1.29	paint concrete
3) Refueling Canal	Steel-lined concrete	3,840	0.0208 1.5	stainless steel concrete
4)	Concrete	760	0.00125 1.5	paint concrete
5) Containment Shell & Misc. Steel	Steel	49,960	0.000625 0.0403	paint steel
6) Misc. Steel	Steel	2,260	0.000625 0.12	paint steel
Lower Compartment				
7) Operating Deck	Concrete	32,200	1.416	concrete
8) Area In Contact With Sump Water	Concrete	15,540	0.0005 1.6	paint concrete
9)	Concrete	2,830	.00125 1.0	paint concrete
10)	Concrete	760	0.0005 1.75	paint concrete
11) Reactor Cavity	Steel-lined concrete	2,270	0.02082 2.0	stainless steel concrete
12) Misc. Steel	Steel	19,500	0.000625 0.0495	paint steel
13) Misc. Steel	Steel	9,000	0.000625 0.1008	paint steel

Added by Amendment 58
December 22, 1978

TABLE Q6.56-3

Watts Bar Ice Condenser Design Parameters

Reactor Containment Volume (Net free volume, ft^3)	
Upper Compartment	651,000
Upper Plenum	47,000
Ice Condenser	86,200
Lower Plenum	24,200
Lower Compartment (Active)	289,014
Lower Compartment (Dead Ended)	94,000
Total Containment Volume	1,191,414
Reactor Power, MWt	3,579
Tech Spec Weight of Ice in Condenser, lbs.	2.45×10^6

Added by Amendment 58
December 22, 1978

TABLE Q6.56-3 (Continued)

WATTS BAR ICE CONDENSER DESIGN PARAMETERS

STRUCTURAL HEAT SINKS

A. Upper Compartment

	Area (ft^2)	Thickness (ft)	
1. Operating Deck			
Slab 1	4880	1.1	Concrete
Slab 2	18280	.0005 1.4	Paint Concrete
Slab 3	760	.00125 1.5	Paint
2. Shell & Misc.			
Slab 5	56331	.000625 .08	Paint Steel

B. Lower Compartment

1. Operating Deck, Crane Wall, and Interior Concrete			
Slab 6	31963	1.43	Concrete
2. Operating Deck			
Slab 7	2830	.00125 1.0	Paint Concrete
Slab 8	760	.0005 1.75	Paint Concrete

Added by Amendment 58
December 22, 1978

TABLE 06.56-3 (Continued)

WATTS BAR ICE CONDENSER DESIGN PARAMETERS

STRUCTURAL HEAT SINKS

B. Lower Compartment (cont'd)

	Area (ft^2)	Thickness (ft)	
4. Floor*			
Slab 10	15921	.0005 1.6	Paint Concrete
5. Misc. Steel			
Slab 11	28500	.000625 .066	Paint Steel
C. Ice Condenser			
1. Ice Baskets			
Slab 12	180,628	0.00663	Steel
2. Lattice Frames			
Slab 13	76,650	0.0217	Steel
3. Lower Support Structure			
Slab 14	28,670	0.0217	Steel
4. Ice Condenser Floor			
Slab 15	3,336	0.00833 0.333	Paint Concrete

*In contact with sump.

Added by Amendment 58
December 22, 1978

TABLE Q6.56-3 (Continued)

STRUCTURAL HEAT SINKS

	Area (ft^2)	Thickness (ft)	
5. Containment Wall Panels & Containment Shell			
Slab 16	19,101	1.0	Steel & Insulation
		0.0625	Steel Shell
6. Crane Wall Panels and Crane Wall			
Slab 17	13,055	1.0	Steel & Insulation
		1.0	Concrete

Added by Amendment 58
December 22, 1978

TABLE Q6.56-4

KEY PARAMETERS AFFECTING SPLIT STEAM LINE BREAKS

Variable	Values Used in LOTIC-3 Report	Values for Watts Bar	Values for Sequoyah
Full Load Steam Pressure (psia)	1000	1000	860
Plant Power (Mwt)	3425	3425	3423
Time Delay to Feedline Isolation (sec)	15	<10.9	<6.5
Time Delay to Steam Line Isolation (sec)	15	<13.7	<5.0

Added by Amendment 58
December 22, 1978

Added by Amendment 6[illegible],
March 2, 1979

TABLE Q6.56-5

SQNP - QUALIFICATION DATA ON CABLES INSIDE CONTAINMENT

TVA Mark No. 1	Applicable IEEE Std	Contract No.	Mfg	Env Qual Aging	Rad	LOCA	MSLB	Remarks
WVA	383-74	WB85259	Belden	Yes	2×10^8	300°F	*	Franklin Institute report F-C4039
	323-74	BLN 820991	Boston Insulated Wire	Yes	2×10^8	300°F	*	BIW report
	383-74	84211	ITT Surprenant	No	2×10^8	346°F	Yes	Franklin Institute report F-C3961
	383-74	822000	Brand Rex	Yes	2×10^8	340°F	yes	Franklin Institute report F-C4113
	383-74	824171 823265	Rockbestos Rockbestos	Yes	2×10^8	340°F	Yes	Rockbestos report on qual of fireway 111 Class 1E Elec Cables
	383-74	87232 824447	Anaconda	Yes	2×10^8	340°F	Yes	Franklin Institute report F-C4836
WVA-1	383-74	824171 823265	Rockbestos	Yes	2×10^8	340°F	Yes	Rockbestos report on qual of fireway 111 Class 1E Elec Cables
WVC	383-74	87232 82447	Anaconda	Yes	2×10^8	340°F	Yes	Franklin Institute report F-C4836

Added by Amendment 60,
March 2, 1979

TABLE Q6.56-5 (Cont'd.)

SQNP - QUALIFICATION DATA ON CABLES INSIDE CONTAINMENT

TVA Mark No. 1	Applicable IEEE Std	Contract No.	Mfg	Env Qual Aging	Rad	LOCA	MSLB	Remarks
WPA, WPB, WPC, & WPG	383-74	824308 823428	Rockbestos	Yes	2×10^8	340°F	Yes	Rockbestos report on Qual of Fireway SR Class 1E elec cables
		85861 83999 85112	Continental Wire	No	1×10^8	340°F	Yes	Franklin Institute report F-C2935
WPJ WPH-1		85861 83999 85112	Continental Wire	No	1×10^8	340°F	Yes	Franklin Institute report F-C2935

Note 1 - Description of cable is as follows:

Mark Nos. WVA, WVA-1, & WVC - Cable, multiple conductor, twisted, stranded, cross-linked polyethylene insulation, color coded, shielded, and 90°C minimum black light stabilized chlorsulfonated polyethylene jacket.

Mark Nos. WPA, WPB, WPC, WPG WPH-1, and WPJ - Cable, multiple conductor, stranded, 600V, 125°C, silicon rubber insulation with glass braid over silicon. Overall cables has mylar binder tape and asbestos braid finished with flame and heat resistant finish.

Note 2 (*) - An arrhenius curve was obtained for these cables which indicate usage in excess of 3 weeks at 325°F. It is felt that due to the short duration of this temperature extreme for a MSLB occurrence that the cables would experience neglible degradation.

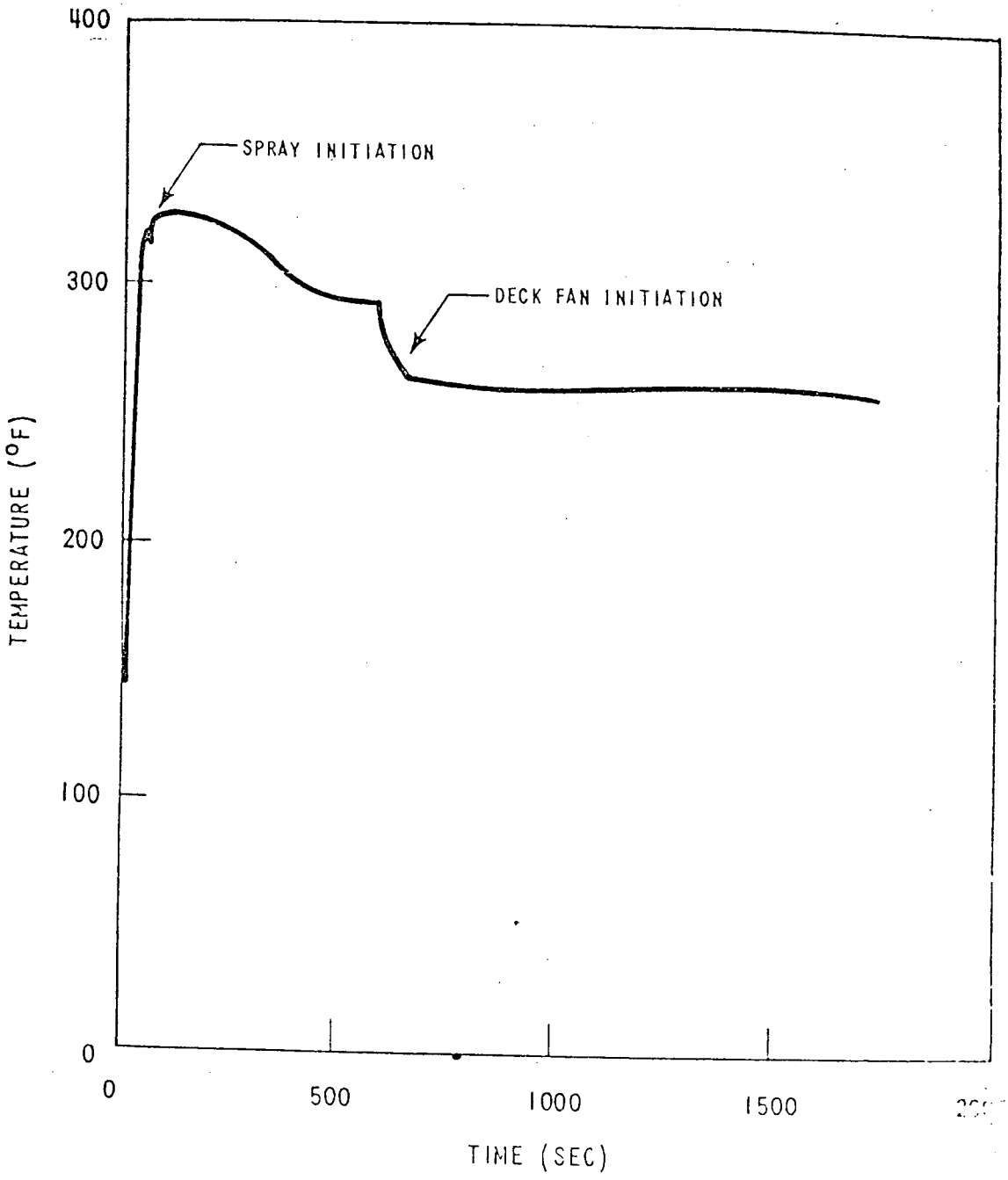

Figure Q6.56-1 Containment Temperature Versus Time for Break at 100% Power

Added by Amendment 58
December 22, 1978

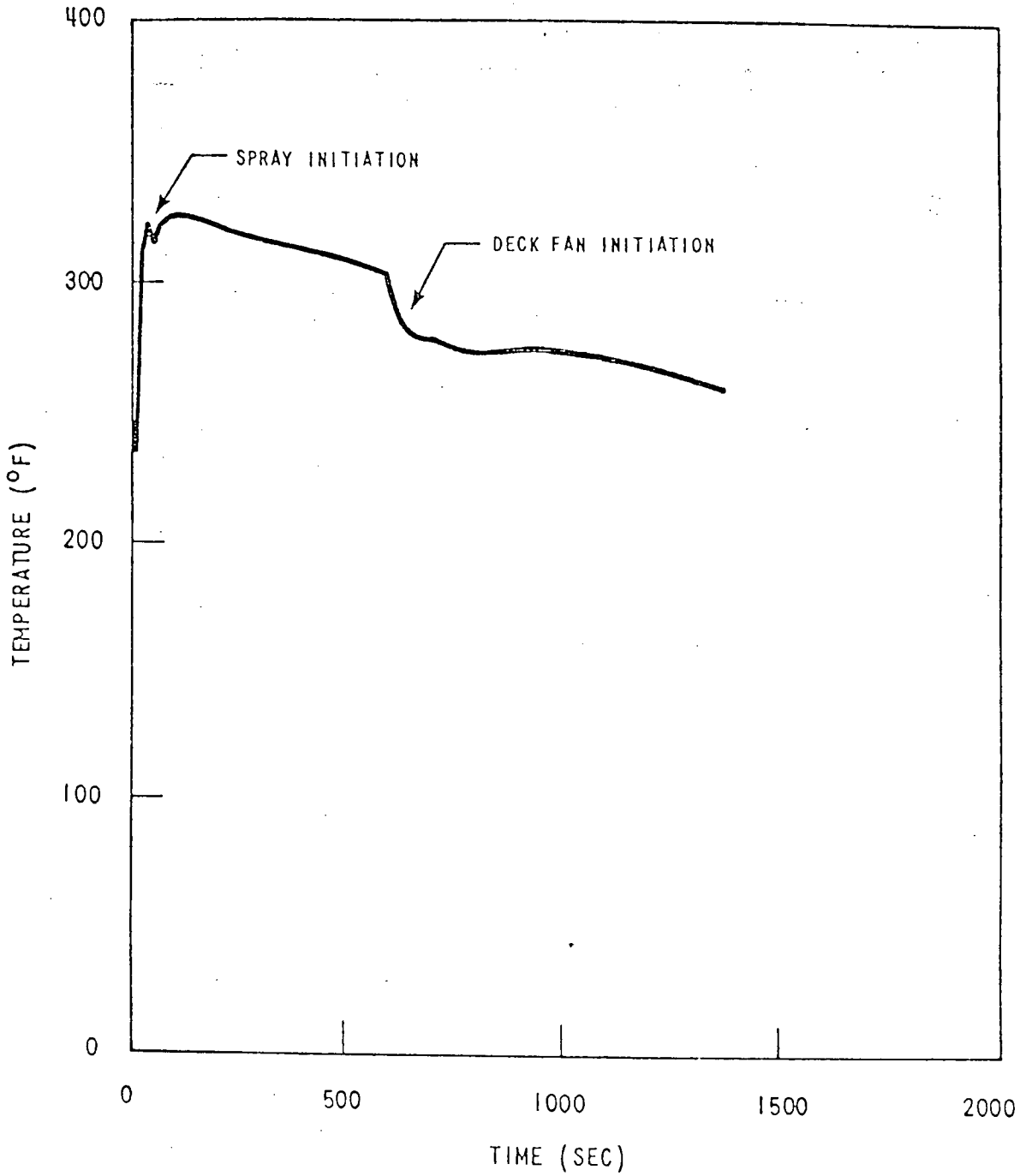

Figure 06.56-2 Containment Temperature Versus Time for Break at 70% Power

Added by Amendment 58
December 22, 1978

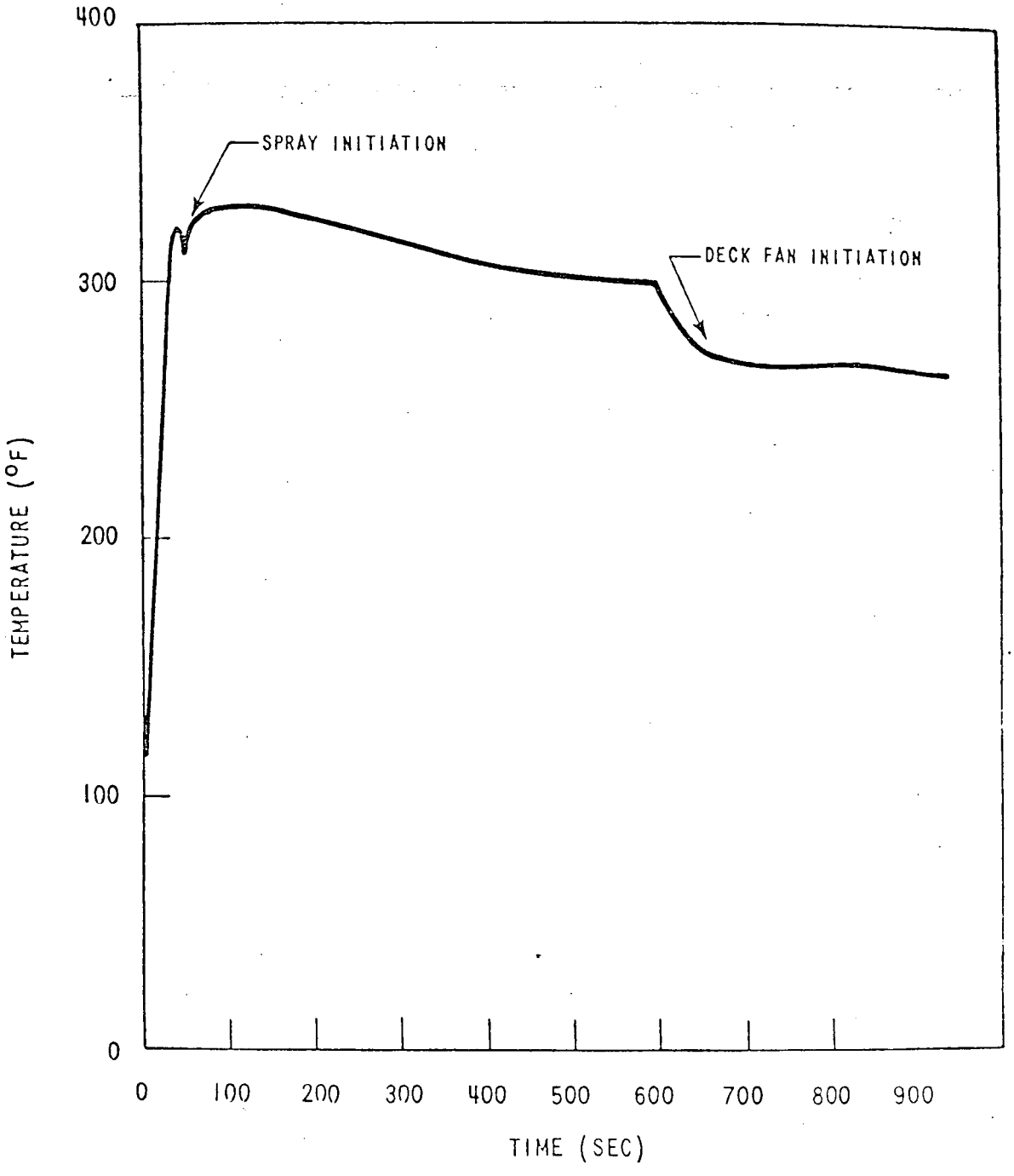

Figure Q6.56-3 Containment Temperature Versus Time for Break at 30% Power

Added by Amendment 58
December 22, 1978

6.56A Your response to Staff question 6.56 (TVA letter J. Gilleland To S. Varga - November 14, 1978) regarding the effects of a postulated main steam line break inside the reactor containment building indicates that TVA believes that the results of analyses performed for a "generic" ice condenser plant as a part of the LOTIC-3 topical report program provided a bounding containment temperature profile for the Sequoyah Nuclear Plant. We have reviewed the information you have provided and agree that the results of analyses performed for a "generic" plant for the review of the LOTIC-3 long term ice condenser code would be conservative with respect to the expected Sequoyah plant long term response. However, the topical report analyses did not include a spectrum of small main steam ling split breaks to assure that the worst temperature profile for the containment was obtained.

We will therefore require additional information regarding the analyses of the containment temperature response to postulated main steam line break(s) which you reference in your response to NRC question 6.56. Specifically, we will require the following information to complete our review of the containment response to postulated ruptures of the main steam line inside containment.

For the worst small split break (i.e., 30% power level with assumed failure of the auxiliary feedwater runout protection system), provide the results of containment response analyses using the LOTIC-3 code for a spectrum of break sizes ranging in size up to the small split break previously analyzed. The spectrum of breaks analyzed should include the largest split break which would not result in automatic initiation of the containment spray system and the largest split break which would not result in automatic initiation of the containment return air fan(s).

For each break analyzed provide: 1) a figure similar to Figure 12.2 (WCAP-8354 Supplement 2) showing upper and lower compartment temperature as a function of time; 2) a figure showing containment pressure as a function of time; 3) a table similar to Table 12.3 (WCAP-8354 Supplement 2) identifying the mass and energy release rate data used in the containment analyses; and 4) identification of any actions assumed to be performed by a control room operator during the course of the accident and the time at which operator actions are assumed to occur including justification for the assumed operator actions.

For all breaks of the spectrum analyzed provide a table showing the following: 1) size of the break; 2) maximum lower compartment temperature (T_L-max); 3) time of T_L-max; 4) time of containment spray initiation; and 5) time of containment return air fan operation.

Response

In response to the request, to provide for the containment responses for a spectrum of small breaks (at the 30% power level with assumed failure of the auxiliary feedwater runout protection system), enclosed are the results from an analysis using the LOTIC-3 computer code. The analyses studied a spectrum of breaks ranging in size from 0.1 ft^2 up to the break identified as the most severe small split break in FSAR Amendment 58, Response 6.56.

This spectrum analyzed breaks of 0.6 ft^2, 0.35 ft^2 and 0.10 ft^2. Attached Figures Q6.56A-1 and Q6.56A-2 provide the upper compartment temperature and lower compartment pressure transients. Figure Q6.56A-3 provides the lower compartment temperature transients. As Figure Q6.56A-3 shows, similar lower compartment temperature transients were calculated for the spectrum of breaks analyzed. However, the 0.6 ft^2 break resulted in a slightly higher maximum lower compartment temperature. (See attached Table Q6.56A-1). When this transient was compared to the transient identified as the most severe small break at 30% power in the previous analysis, it was found to result in a slightly lower peak temperature (See Figure Q6.56A-4).

In the analyses, spray and fan initiation will be automatic after reaching the containment Hi-2 setpoint. Associated times assumed in the analyses are also included in Table Q6.56A-1. Tables Q6.56A-2, Q6.56-3, and Q6.56A-4 provide the mass and energy release rates for the transients analyzed.

These results demonstrate the conservatism of the results previously submitted in FSAR Amendment 58, Question 6.56, and also the relative insensitive nature of the containment response to break size.

TABLE Q6.56A-1

CASE	MAXIMUM LC TEMP °F	TIME t_{max} SEC.	TIME OF CONTAINMENT* SPRAY	FAN
0.6 FT^2	326.1	140.78	52.	605.
0.35 FT^2	325.7	305.67	59.	616.
0.1 FT^2	319.3	649.33	103.	661.

* Hi-2 Pressure Setpoint used was 3.5 psig,

Relay time used for spray actuation after Hi-2 signal was 45 sec
Relay time used for fan actuation after Hi-2 signal was 600 sec.

Added by Amendment 61
May 25, 1979

TABLE Q6.56A-2

0.1 FT^2 SPLIT 30 PERCENT POWER

TIME	m (lb/sec)	e (BTU/SEC)
.1000E-01	.2550E+03	.3032E+06
.5000E+00	.2280E+03	.2711E+06
.3000E+01	.2260E+03	.2688E+06
.7000E+01	.2250E+03	.2677E+06
.1400E+02	.2240E+03	.2665E+06
.2400E+02	.2220E+03	.2642E+06
.2600E+02	.2250E+03	.2677E+06
.2800E+02	.2280E+03	.2711E+06
.3000E+02	.2290E+03	.2722E+06
.3600E+02	.2300E+03	.2734E+06
.4600E+02	.2300E+03	.2734E+06
.5200E+02	.2290E+03	.2723E+06
.5400E+02	.2280E+03	.2711E+06
.5600E+02	.2280E+03	.2712E+06
.6000E+02	.2230E+03	.2654E+06
.6600E+02	.2180E+03	.2597E+06
.7000E+02	.2150E+03	.2562E+06
.7400E+02	.2120E+03	.2527E+06
.8200E+02	.2080E+03	.2481E+06
.9400E+02	.2030E+03	.2423E+06
.1100E+03	.1980E+03	.2365E+06
.1300E+03	.1930E+03	.2307E+06
.1560E+03	.1880E+03	.2249E+06
.1850E+03	.1830E+03	.2190E+06
.2200E+03	.1780E+03	.2132E+06
.2600E+03	.1730E+03	.2073E+06
.3000E+03	.1680E+03	.2014E+06
.3600E+03	.1620E+03	.1944E+06
.4000E+03	.1570E+03	.1884E+06
.4600E+03	.1510E+03	.1813E+06
.5000E+03	.1460E+03	.1773E+06
.5600E+03	.1420E+03	.1717E+06
.6400E+03	.1360E+03	.1636E+06
.6600E+03	.1340E+03	.1612E+06
.7000E+03	.1310E+03	.1576E+06
.7400E+03	.1280E+03	.1540E+06
.8200E+03	.1220E+03	.1468E+06
.8800E+03	.1180E+03	.1421E+06
.9600E+03	.1130E+03	.1361E+06
.1000E+04	.1110E+03	.1337E+06

Added by Amendment 61
May 25, 1979

TABLE Q6.56A-3

0.35 FT^2 SPLIT 30 PERCENT POWER

TIME	m (lb/sec)	e (BTU/SEC)
.1000E-01	.5400E+03	.6422E+06
.5000E+00	.8000E+03	.9515E+06
.2000E+01	.7930E+03	.9434E+06
.4000E+01	.7850E+03	.9342E+06
.6000E+01	.7780E+03	.9261E+06
.8000E+01	.7720E+03	.9192E+06
.1000E+02	.7660E+03	.9122E+06
.1200E+02	.7730E+03	.9203E+06
.1500E+02	.7790E+03	.9272E+06
.1900E+02	.7780E+03	.9261E+06
.2000E+02	.7720E+03	.9195E+06
.2100E+02	.7570E+03	.9022E+06
.2200E+02	.7440E+03	.8871E+06
.2300E+02	.7310E+03	.8721E+06
.2500E+02	.7090E+03	.8466E+06
.2800E+02	.6800E+03	.8123E+06
.3000E+02	.6630E+03	.7930E+06
.3400E+02	.6340E+03	.7590E+06
.3800E+02	.6120E+03	.7333E+06
.4200E+02	.5930E+03	.7109E+06
.4600E+02	.5770E+03	.6921E+06
.5200E+02	.5580E+03	.6697E+06
.6200E+02	.5330E+03	.6401E+06
.7000E+02	.5170E+03	.6211E+06
.8000E+02	.5010E+03	.6022E+06
.1000E+03	.4750E+03	.5712E+06
.1250E+03	.4510E+03	.5427E+06
.1400E+03	.4360E+03	.5271E+06
.1600E+03	.4230E+03	.5092E+06
.2000E+03	.3960E+03	.4769E+06
.2500E+03	.3670E+03	.4421E+06
.3000E+03	.3400E+03	.4096E+06
.3800E+03	.3010E+03	.3626E+06
.4800E+03	.2690E+03	.3240E+06
.5400E+03	.2420E+03	.2913E+06
.6000E+03	.2250E+03	.2707E+06
.6400E+03	.2140E+03	.2574E+06
.7600E+03	.1880E+03	.2259E+06
.8800E+03	.1680E+03	.1892E+06
.1000E+04	.1490E+03	.1786E+06

Added by Amendment 61
May 25, 1979

TABLE Q6.56A-4

0.6 FT^2 SPLIT 30 PERCENT POWER

TIME	m (lb/sec)	e (BTU/SEC)
.1000E-01	.3525E+04	.4192E+07
.1000E+01	.1364E+04	.1623E+07
.3000E+01	.1340E+04	.1595E+07
.5000E+01	.1320E+04	.1572E+07
.6500E+01	.1305E+04	.1555E+07
.8000E+01	.1293E+04	.1541E+07
.1000E+02	.1297E+04	.1545E+07
.1200E+02	.1298E+04	.1546E+07
.1300E+02	.1296E+04	.1544E+07
.1400E+02	.1267E+04	.1511E+07
.1600E+02	.1196E+04	.1429E+07
.1800E+02	.1132E+04	.1354E+07
.2000E+02	.1078E+04	.1291E+07
.2200E+02	.1032E+04	.1237E+07
.2400E+02	.9930E+03	.1191E+07
.2700E+02	.9440E+03	.1133E+07
.3200E+02	.8810E+03	.1059E+07
.3600E+02	.8430E+03	.1013E+07
.4000E+02	.8130E+03	.9778E+06
.4600E+02	.7760E+03	.9337E+06
.5000E+02	.7570E+03	.9110E+06
.6000E+02	.7170E+03	.8632E+06
.7600E+02	.6700E+03	.8069E+06
.9600E+02	.6290E+03	.7577E+06
.1200E+03	.5900E+03	.7108E+06
.1500E+03	.5490E+03	.6614E+06
.1800E+03	.5130E+03	.6180E+06
.2100E+03	.4800E+03	.5782E+06
.2400E+03	.4490E+03	.5407E+06
.2600E+03	.4300E+03	.5177E+06
.3000E+03	.3950E+03	.4754E+06
.3600E+03	.3150E+03	.3784E+06
.4200E+03	.2620E+03	.3142E+06
.5000E+03	.2200E+03	.2633E+06
.5600E+03	.2050E+03	.2451E+06
.6000E+03	.1960E+03	.2342E+06
.6600E+03	.1850E+03	.2209E+06
.8600E+03	.1660E+03	.1979E+06
.9600E+03	.1590E+03	.1894E+06
.1000E+04	.1550E+03	.1846E+06

Added by Amendment 61
May 25, 1979

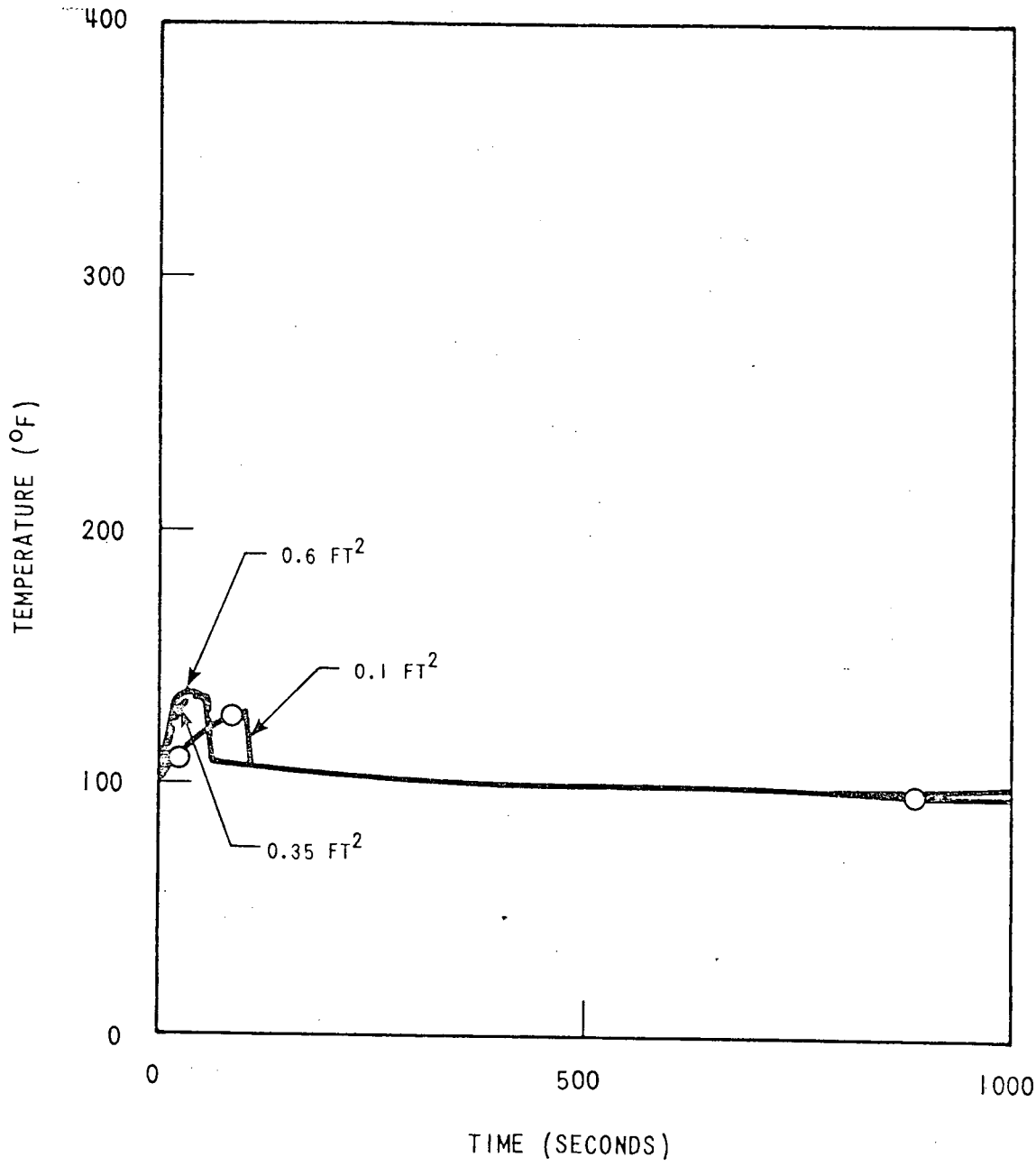

Figure Q6.56A-1 Upper Compartment Temperature 30% Power Level

Added by Amendment 61
May 25, 1979

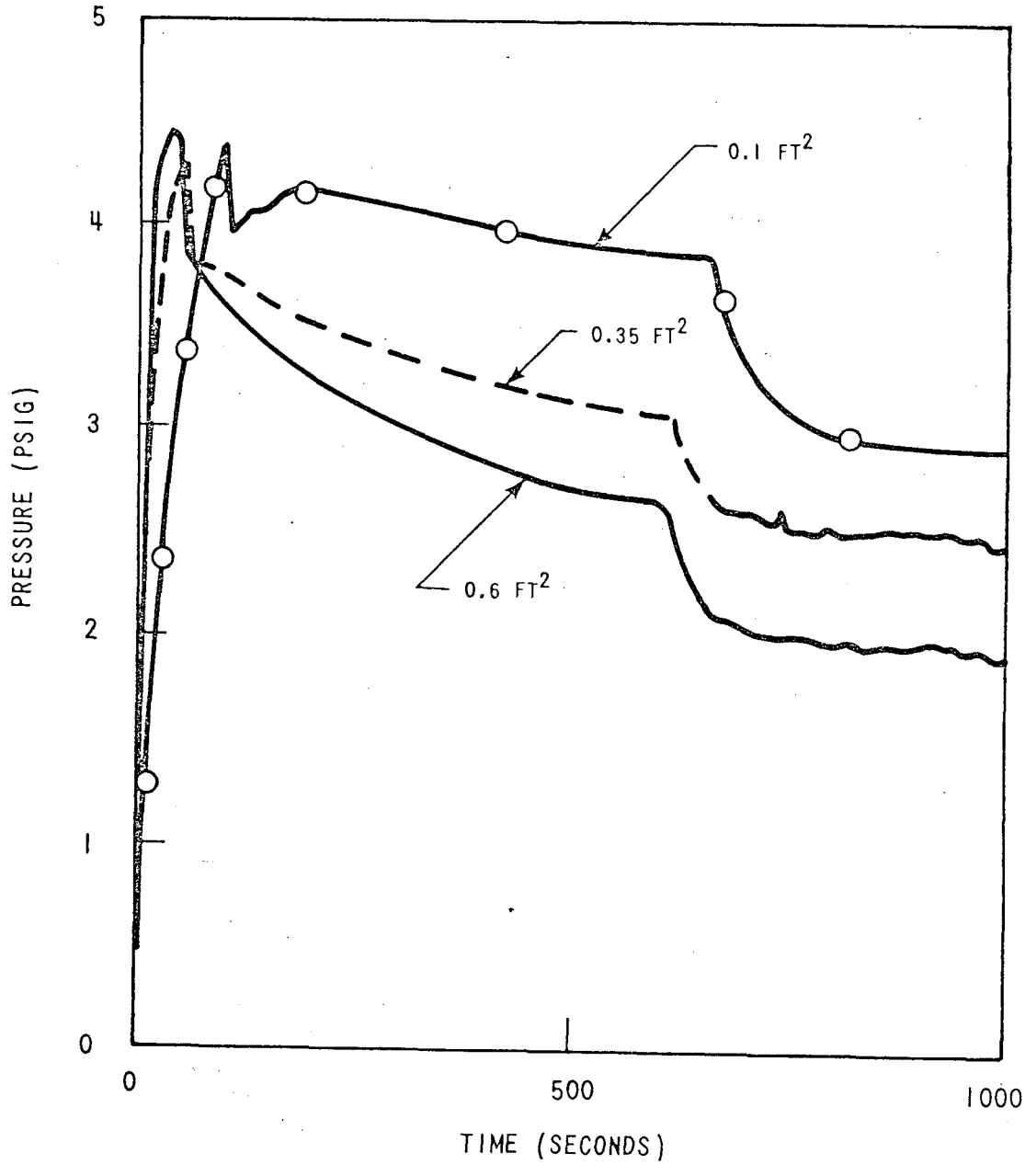

Figure Q6.56A-2 Lower Compartment Pressure 30% Power Level

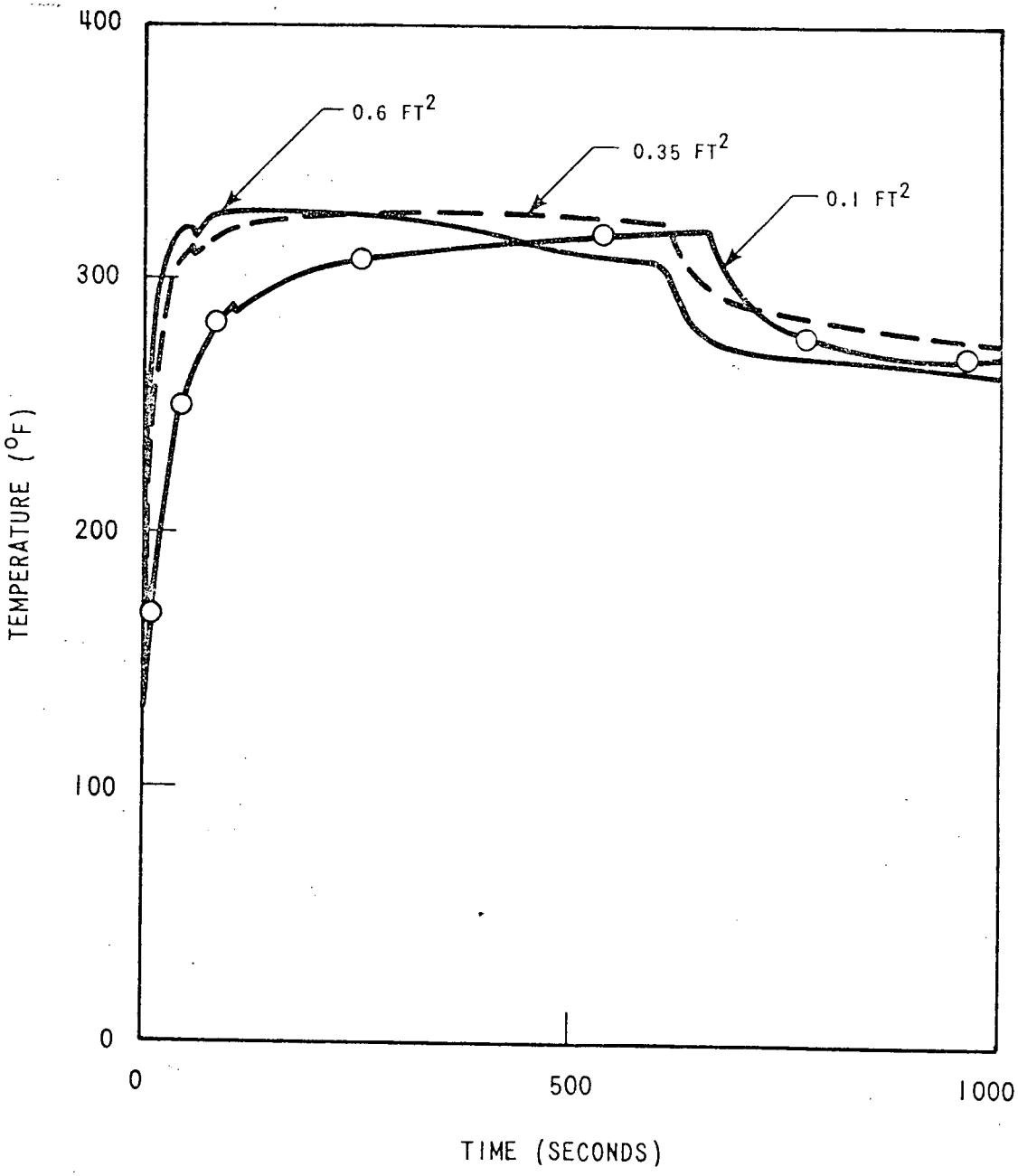

Figure Q6.56A-3 Lower Compartment Temperature 30% Power

Added by Amendment 6
May 25, 1979

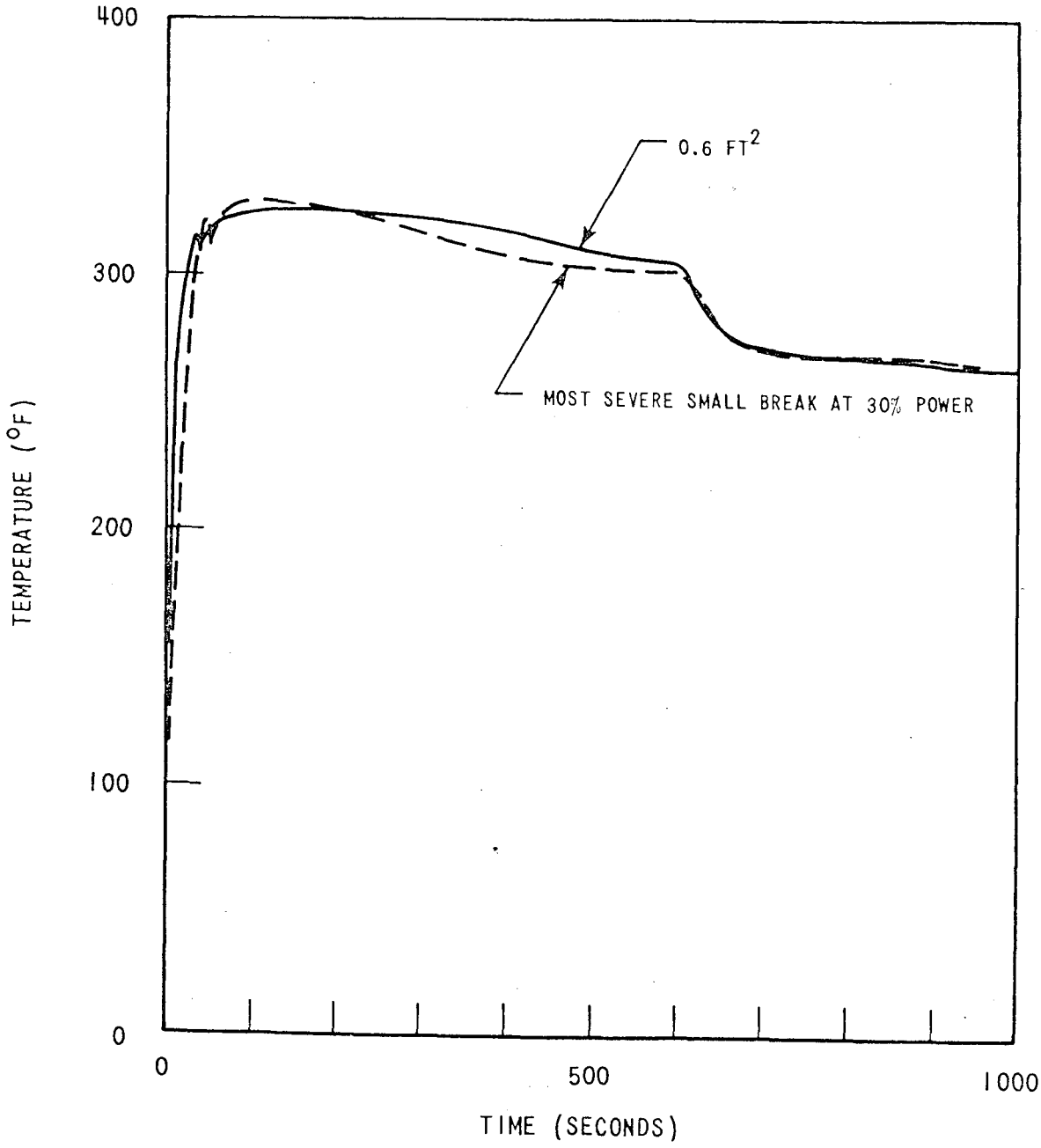

Figure Q6.56A-4 Worst Break Lower Compartment Temperature Comparison

Added by Amendment 61,
May 25, 1979

6.2.5 SQN/WBN Comparison Combustible Gas Control (Beyond Hydrogen Recombiners)

As a result of the TMI-2 accident, NRC is presently requiring plants with relatively low containment volumes and design pressures (ice condenser and Mark III containments in particular) to provide additional design features for controlling post-LOCA hydrogen releases in excess of the quantities presently required by 10CFR50.44. To resolve this issue for Sequoyah, TVA incorporated a controlled ignition system (Interim Distributed Ignition System - IDIS) into the plant design. This system utilizes a number of glow plug igniters strategically placed within the various containment compartments to burn the hydrogen at low concentrations (6-8%) as it is released, thereby preventing the formation of localized "pockets" of hydrogen in potentially detonable concentrations (approximately 18 v/o). The NRC's assessment of this system is provided in Supplements 3 and 4 to the Sequoyah Safety Evaluation Report (NUREG-0011).

As part of a system upgrade to address the Sequoyah license requirements, TVA will be replacing the IDIS with another controlled ignition system. This system is presently called the Permanent Hydrogen Mitigation System (PHMS): The IDIS and PHMS are essentially the same with respect to functional operating capability. The major difference between the systems is that the PHMS provides complete redundancy. Pending the identification of major problems during the present licensing review by NRC, TVA intends to install the PHMS at both Sequoyah and Watts Bar. It should be noted that the hydrogen research program being conducted by TVA to verify the adequacy of PHMS for Sequoyah is directly applicable to Watts Bar.

6.2.5.2 Hydrogen Purge Exhaust Subsystem

The postaccident hydrogen purge system to be used as a backup to the redundant hydrogen recombiner system as described below for WBN is identical to the system used in SQN as described in section 6.2.5.2 of the Sequoyah FSAR.

The hydrogen purge exhaust subsystem consists of a single penetration (X-80) in the primary containment wall equipped with two normally closed, remote manually operated isolation valves (one on either side of the containment wall); one pneumatically operated annulus purge exhaust valve located within the annulus; and two 1/2-inch leakoff nipples located between the outboard isolation valves and the annulus purge exhaust valve. With the containment isolation valves open and the annulus purge exhaust valve closed, a flow path is established from the primary containment through the leakoffs and into the annulus, which will permit purging of the containment for hydrogen control subsequent to a LOCA.

The impetus for flow will be provided by the differential pressure between the primary containment and annulus. If the concentration cannot be maintained below 4 percent through the leakoff path, the annulus purge valve will be opened to exhaust dilutant air for a minimum time sufficient to maintain the hydrogen concentration below 4 percent. The containment effluent purged will flow directly to the annulus where it will mix with the annulus atmosphere and be filtered by the air cleanup system prior to discharge to the outside environment.

6.2.6 Comparison of Leak Rate Testing at Sequoyah and Watts Bar Nuclear Plants

Leak rate testing of the containment isolation systems at both SQN and WBN is performed according to 10CFR50, Appendix J, with one notable exception. The exception allows SQN unit 1 to perform several preoperational type C leak rate tests using water as the medium instead of air as is required in 10CFR50, Appendix J. During the first refueling outage, modifications will be made to allow air testing in subsequent type C tests at SQN unit 1.

The containment leak test in accordance with 10CFR50, Appendix J, and the one time exemption for SQN unit 1 are described in the respective FSAR for each plant.

6.2.4 Comparison of Containment Isolation at Sequoyah and Watts Bar Nuclear Plants

Containment isolation for both Sequoyah (SQN) and Watts Bar (WBN) Nuclear Plants has, in general, been designed to meet the criteria of 10CFR50 Appendix A and Standard Review Plan 6.2.4. These plants are very similar in their containment isolation, having only a few differences resulting from the plant system differences. The major differences in the plants are listed below:

1. Penetrations X-8A, 8B, 8C, and 8D - In addition to the main and auxiliary feedwater lines, WBN has four feedwater bypass lines (one per steam generator). These feedwater bypass lines provide yet another path for supplying feedwater to the steam generator. These feedwater bypass lines penetrate primary containment (penetrations X-8A-X-8D) and attach to the secondary side of the steam generator. The auxiliary feedwater lines at WBN connect to the feedwater bypass lines. The auxiliary feedwater lines to steam generators 2 and 3 join to the corresponding feedwater bypass lines outside containment and the auxiliary feedwater lines to steam generators 1 and 4 penetrate containment (X-40A, B) and connect to the respective feedwater bypass line inside containment.

   Containment isolation for the feedwater bypass lines is accomplished in the same manner as the main and auxiliary feedwater lines by the filling of the secondary side of the steam generator with water post-LOCA.

   At SQN, feedwater bypass lines are not used and penetrations X-8A through X-8D do not exist. For this plant, the auxiliary feedwater lines connect directly to the main feedwater lines.

2. Penetration X-17 - The remote manual isolation valve in this RHR line is inside containment at WBN and outside containment at SQN. Containment isolation for both systems is composed of a closed system outside containment and check valves inside containment.

   Additionally, at both plants, a water seal is placed on the penetration by using the remote manual valves and the RHR pumps.

3. Penetrations X-19A and 19B - The containment spray suction lines connect to the RHR sump suction lines outboard of the RHR sump suction isolation valves at SQN. At WBN, however, the containment spray suction lines connect inboard of the RHR sump suction isolation valves. For this reason, each containment spray suction lines at WBN has a motor operated containment isolation valve which is remote manually operated from the main control room.

4. Penetrations X-27A, 27B, 27C, and 27D - At SQN, the steam generator sample lines connect to the steam generator blowdown lines outboard of primary containment and inboard of the outer isolation valve. Each of these sample lines are isolated by an air operated gate valve which receives the phase A isolation signal.

At WBN, the steam generator sample lines penetrate primary containment (penetrations X-27A-D) and connect directly to the steam generator. Each sample line has two containment isolation valves which receive the containment isolation signal - one valve immediately inside containment and one valve immediately outside containment.

These penetrations are seal welded spares at SQN.

5. Penetrations X-40A and 40B - The auxiliary feedwater lines at SQN connect to main feedwater lines, but at WBN, the auxiliary feedwater lines connect to feedwater bypass lines as described in (1) above. Containment isolation in these lines, however, is the same for both plants.

6. Penetrations X-102 and104 - At SQN, these penetrations are used by auxiliary feedwater test lines. Isolation of these lines is accomplished by a closed system inside containment (secondary side oif the steam generator) and a manually locked closed isolation valve outside containment. At this time, there are no corresponding lines at WBN and penetrations X-107 and X-104 are seal welded spares at this plant.

7. Penetration X-107 - The RHR supply line at WBN has a bypass line around the inner containment isolation valve. The bypass line, which does not penetrate containment, contains an inline containment isolation valve which is motor operated and is remote manually operated from the main control room.

   At SQN, there is no corresponding bypass line.

8. Penetrations X-111, 112, and 113 - At SQN, these penetrations are used by the vacuum relief lines which employ two isolation valves outside containment.

   At WBN, these penetrations are seal welded spares.