ATTACHMENT 2

Westinghouse Application for Withholding, Affidavit, and Non-Proprietary Version of Attachment 1

Westinghouse

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Our ref: CAW-06-2103 Rev. 1

February 22, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: NF-BEX-06-8-P Rev. 1, "Task Report for TSD DQW04-21 LOCA Analysis for Quad Cities 1 & 2 and Dresden 2 & 3" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-06-2103 Rev. 1 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Exelon Nuclear.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-06-2103 Rev. 1 and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz/NRR
P. M. Clifford/NRR
M. Banerjee/NRR
G. S. Shukla/NRR
L. M. Feizollahi/NRR (affidavit only)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared R. B. Sisk, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

R B Sisk

R. B. Sisk, Manager
Fuel Engineering Licensing

Sworn to and subscribed
before me this 22nd day
of February, 2006

Sharon L Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007

Member, Pennsylvania Association Of Notaries

(1) I am Manager, Fuel Engineering Licensing, in Nuclear Fuel, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in NF-BEX-06-8-P Rev. 1, "Task Report for TSD DQW04-21 LOCA Analysis for Quad Cities 1 & 2 and Dresden 2 & 3" (Proprietary), dated February 2006, for review and comment by the NRC, being transmitted by Exelon Nuclear letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Quad Cities Units 1 and 2 and Dresden Units 2 and 3 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of SVEA-96 Optima2 License Amendment Request.

This information is part of that which will enable Westinghouse to:

(a) Provide technical information in support of License Amendment Request.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this information to further enhance their licensing position with their competitors.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar analyses and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice

NF-BEX-06-8-NP
Revision 1

February 2006

Task Report for TSD DQW04-21
LOCA Analysis for
Quad Cities 1 & 2 and Dresden 2 & 3

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Acronyms

Acronym	Definition
ADS	Automatic Depressurization System
AVZ	Above Vessel Zero
BWR	Boiling Water Reactor
CPR	Critical Power Ratio
DEFLB	Double-Ended Feed Line Break
DEG	Double-Ended Guillotine
DEGPD	Double-Ended Guillotine Pump Discharge
DEGPS	Double-Ended Guillotine Pump Suction
DESLB	Double-Ended Steam Line Break
DESPB	Double-Ended Spray Line Break
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EM	Evaluation Model
FW	Feed Water
HPCI	High Pressure Coolant Injection
ICF	Increase Core Flow
IV	Injection Valve
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LSL	Loop Select Logic
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
PCT	Peak Cladding Temperature
PD	Pump Discharge
PS	Pump Suction
RDV	Recirculation Discharge Valve
SLB	Steam Line Break

Contents

Tables

Figures

1.0 Description of LOCA Event and Acceptance Criteria

1.1 Description of LOCA Event

The LOCA event is postulated as a rupture of piping connected to the reactor pressure vessel within the primary containment. A spectrum of piping breaks is considered to encompass all sizes and locations of breaks up to and including the circumferential failure of the largest connected pipe. A LOCA inside containment would result in the heating and pressurization of containment, a challenge to the emergency core cooling system (ECCS), and the potential release of radioactive material to the environment. By evaluating the entire spectrum of postulated breaks, the most severe challenge to the ECCS and primary containment can be determined.

This report evaluates the fuel thermal response and the ECCS performance as a result of a postulated LOCA. The plant maximum average planar linear heat generation rate (MAPLHGR) operating limit is established, in part, to ensure compliance with the LOCA ECCS design bases. The function of each ECCS subsystem is to ensure adequate core cooling across the entire spectrum of line break accidents when operated with other available ECCS subsystems with consideration of the Appendix K single failure criterion and without reliance on external sources of electrical power.

The LOCA event described below is for a large break in one of the two external recirculation loops. Other break locations have slightly varying transient characteristics similar to that outlined here.

Following the postulated pipe rupture, coolant discharges rapidly through both sides of the break, with greater flow from the vessel side. The pump side flow is restricted by the reduced flow area of the jet pump nozzles and friction losses in the recirculation loop and pump. Loss of all off-site electrical power is assumed coincident with the break, resulting in a coastdown of all recirculation pumps and a rapid closure of the turbine stop valves. The closure of the turbine stop valves will cause a momentary increase in reactor vessel pressure and a power increase due to void collapse. Automatic reactor scram occurs as a result of high drywell pressure, high reactor pressure or low reactor water level. Following reactor shutdown, the steam production in the core is reduced and the reactor pressure decreases rapidly. After several seconds, the water level in the downcomer falls to the jet pump suction elevation, which allows steam to flow to the break, and the break mass flow rate decreases significantly.

Flashing in the jet pumps and subsequently in the lower plenum occurs as the pressure continues to decrease. This results in a short-term level rise in the core and downcomer. Following this level swell, the continued inventory decrease results in decreasing reactor water level and system pressure. The core two-phase mixture level will drop exposing the fuel rods to a mostly steam environment and the heat transfer mode in the core transitions from nucleate boiling to film boiling and finally to steam cooling. The transition from nucleate boiling results in a fuel rod cladding heat up. By the time this occurs, the reactor will have scrammed. However, fission product decay heat will cause both the fuel and cladding temperatures to increase. The cladding temperature increase is terminated when two-phase cooling conditions are restored in the core by the ECCS equipment.

The low pressure ECCS equipment is actuated by either a high drywell pressure signal or the combination of a low-low reactor water level signal and a low reactor pressure signal. For most breaks inside containment, the high drywell pressure signal occurs first. The emergency diesel generators, which start on loss of off-site power, provide power to the ECCS equipment, which direct emergency cooling water into the reactor pressure vessel. The timing of recovery of the core depends on the ECCS equipment actuated.

Since the high pressure coolant injection system is turbine driven, it may not be actuated for large breaks as the reactor pressure decreases below the pressure required by the turbine.

1.2 Acceptance Criteria

The Code of Federal Regulations Title 10 Part 50.46 provides five specific design acceptance criteria for the plant ECCS. The acceptance criteria are:

(1) Peak cladding temperature shall not exceed 2200 °F.

(2) The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) After any calculated successful operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Demonstration that the first three criteria are satisfied ensures that the fourth criterion is satisfied.

2.0 Westinghouse BWR LOCA Methodology

The Westinghouse BWR LOCA methodology is described in References 1 through 6. The methodology used in support of the Dresden and Quad Cities units is the USA5 evaluation model (EM), which is described in Reference 6. This methodology makes use of the GOBLIN series of computer codes to calculate the BWR transient response to both large and small break LOCAs.

2.1 Analysis Codes

The GOBLIN series of computer codes is comprised of two major computer codes – GOBLIN and CHACHA-3D.

<u>GOBLIN</u> performs the analysis of the LOCA blowdown and reflood thermal hydraulic transient for the entire reactor, including the interaction with various control and safety systems. GOBLIN may also be run in the 'DRAGON' mode to perform hot fuel assembly transient calculation using boundary conditions from a previous GOBLIN system analysis. Alternatively, the hot assembly analysis may be performed as a parallel channel in the GOBLIN system analysis. In this case, there is no need to drive the DRAGON analysis with boundary conditions from the system analysis. The GOBLIN code is described in detail in Reference 1. The GOBLIN code can be divided into four main sections as described below.

The <u>hydraulic model</u> solves the mass, energy and momentum conservation equations together with the equation of state for each control volume. This model includes empirical constitutive correlations for the calculation of pressure drops, [

][a,c]

The <u>system models</u> contain detailed models of the various reactor components, and the safety systems that are activated after a LOCA. They include the [

][a,c]

The <u>thermal model</u> calculates the heat conduction and heat transfer from the fuel rods, pressure vessel, and internals to the coolant. This model solves the material heat conduction equation and calculates the heat transfer from the fuel and structures to the coolant.

The <u>power generation models</u> calculate the heat generation due to fission, decay heat, and metal-water reaction. Fission power is determined by a point kinetics model.

These models are described in detail in References 1, 4 and 6. The most recent changes to the GOBLIN code, which are described in detail in Reference 6, include an improvement to the [ ][a,c] model and the implementation of the approved CPR correlation for SVEA-96 Optima2 fuel (Reference 7).

<u>CHACHA-3D</u> performs detailed fuel rod mechanical and thermal response calculations at a specified axial level within the hot assembly. All necessary fluid boundary conditions are obtained from the hot assembly thermal hydraulic analysis described above. CHACHA-3D determines the [ ][a,c] analyzed. These results are used to determine the peak cladding temperature and cladding oxidation at the axial plane under investigation. CHACHA-3D also provides input for the calculation of total hydrogen generation.

The major components of the CHACHA-3D code are the fuel rod conduction model, the channel temperature model, the heat generation model, the metal-water reaction model, the thermal radiation model, the gas plenum temperature and pressure model, the channel rewet model, the pellet/cladding gap heat transfer model and the cladding strain and rupture model. These models are described in detail in References 1, 4 and 6.

The most recent changes to the CHACHA-3D computer code, which are described in more detail in Reference 6, are the addition of a new fuel rod plenum model for part-length rods and the addition of applicable fuel performance models from the approved STAV7.2 fuel performance code (Reference 8).

2.2 Analysis Process

The Westinghouse analysis process as described below is in accordance with NRC approved methodology and codes. There are no deviations from the approved methodology.

The flow of information between the various analyses is shown in Figure 2-1. In the case shown in the figure, the hot assembly is analyzed using GOBLIN in the so-called DRAGON mode. In the event that the hot assembly analyses is done in conjunction with the system analysis, the intermediate step shown in the figure is not necessary and the information provided to CHACHA-3D is derived from the hot assembly analysis performed in the first step. For this application, the three step process is used.

The system analysis determines the overall response of the system to the event analyzed. The discussion in Section 1.1 provides an example for one such scenario. In addition to the boundary conditions that are used by the CHACHA-3D heat-up analysis, the GOBLIN system analysis determines the actuation of the ECCS components that provide cooling to the core after the system is depressurized.

The hot assembly analysis determines the thermal-hydraulic conditions in the hot channel. [

]a,c

As shown in Figure 2-1, the convective heat transfer coefficients predicted by the hot assembly analysis are provided to the CHACHA-3D heat-up calculation. [

]a,c and described in Reference 6.

Figure 2-1 also shows fuel performance data being applied in the form of gap coefficients for the system and hot channel analyses. [

]a,c

Figure 2-1 also shows that fuel performance data are provided to the CHACHA-3D heat-up analysis. These data are comprised of initial conditions for each type of fuel rod (e.g., [

]a,c

[ ]a,c and the most conservative result is selected.

a,c

Figure 2-1 Flow of Information Between Analyses

3.0 Plant-Specific Information

The Dresden 2 & 3 and Quad Cities 1 & 2 units are similar in many respects. They are all BWR/3 designs that have the same rated power and flow. Their ECCS components, although slightly different in performance, are the same. Because of this similarity, the LOCA analysis is based on a single bounding 'Unit 5' model that is conservative with respect to all four units. The inputs to the LOCA model are described in detail in Reference 10.

3.1 ECCS Description

As shown in Figure 3-1, the ECCS is comprised of one High Pressure Coolant Injection (HPCI) system, two Low Pressure Coolant Injection (LPCI) systems, and two Low Pressure Core Spray (LPCS) systems. The Automatic Depressurization System (ADS), which is used to depressurize the reactor system for certain small breaks, is not shown in the figure. Each of these systems is described below.

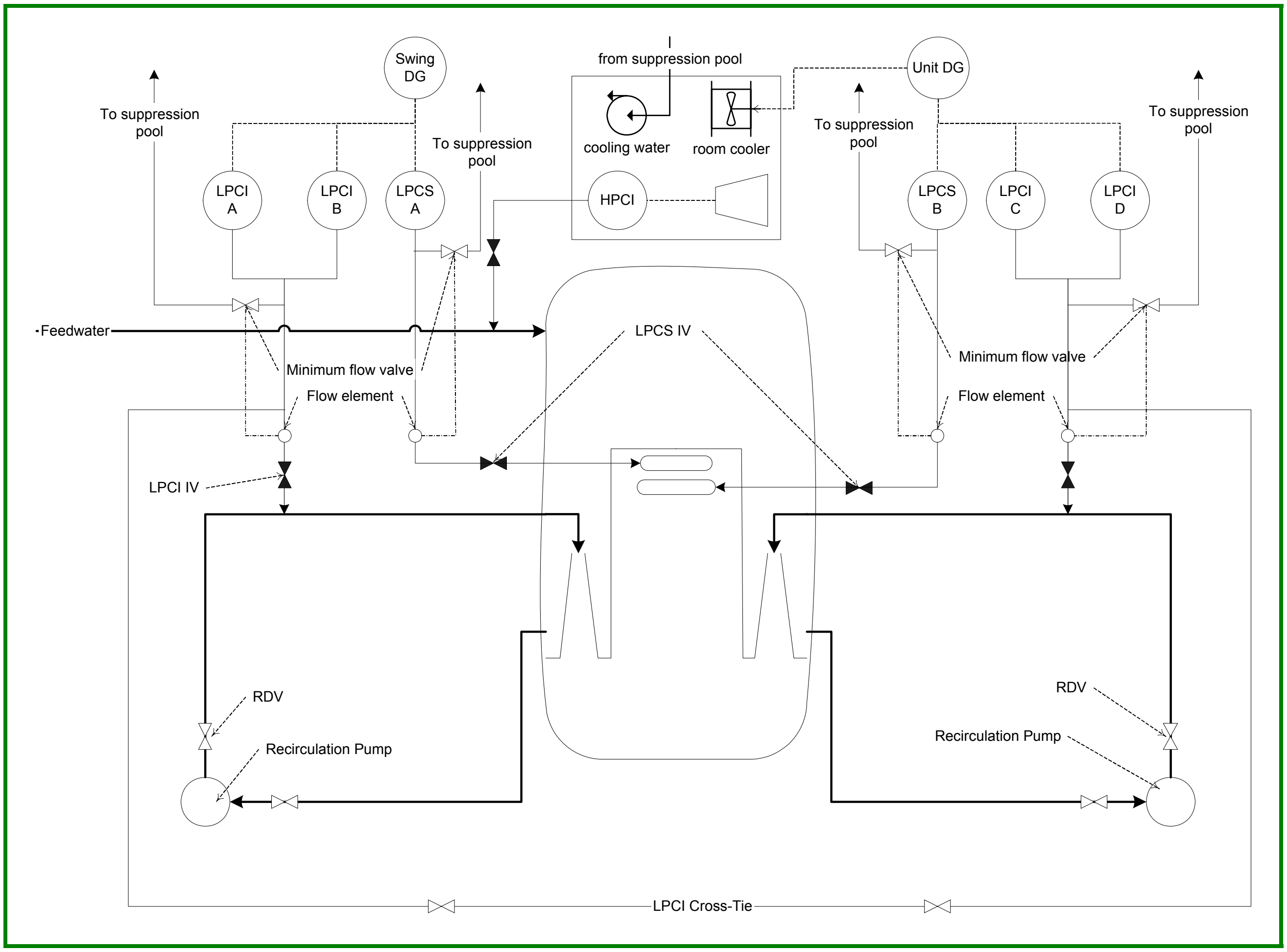

Figure 3-1 ECCS Schematic (short-term injection mode)

3.1.1 <u>High Pressure Coolant Injection</u>

The HPCI system consists of a steam turbine driving a multi-stage high-pressure pump and a gear-driven single stage booster pump. The HPCI system takes suction from either the contaminated condensate storage tank or the suppression pool and delivers flow to one of the feedwater lines. For the LOCA analysis, only the suppression pool is credited as a source of coolant since it is seismically designed.

The system performance and actuation parameters for the Unit 5 model are described in Table 3-1. The HPCI turbine oil cooler and gland seal condenser are cooled by water from the suppression pool. Since these components are rated at 140 °F, continued operation above a suppression pool temperature of 140 °F is not permitted. Also, operation of HPCI above 140 °F would exceed the current net positive suction head (NPSH) calculations for rated HPCI pump flows. Another limitation on the HPCI system is related to the dependence of the HPCI room cooler on the unit emergency diesel generator (EDG). Therefore, any single failures of the unit EDG need to assume consequential loss of the HPCI system after 10 minutes of operation. As a result of these considerations, the HPCI system is not credited when any of these conditions are exceeded.

Table 3-1 HPCI Performance Parameters

Description	**Units**	**Unit 5 Value**
Delivery Performance		
Vessel – wetwell ΔP	psid	< 150
Flow	gpm	0
Vessel – wetwell ΔP	psid	150 < ΔP < 1120
Flow	gpm	5000
Vessel – wetwell ΔP	psid	> 1120
Flow	gpm	0
Initiating Signals and Setpoints		
Low-low water level (AVZ), OR	inch	444
High drywell pressure	psig	2.5
Time Delay to Deliver Rated Flow	sec	45

3.1.2 Low Pressure Coolant Injection

As shown in Figure 3-1, there are two LPCI divisions. The major LPCI parameters are shown in Table 3-2. Each division consists of two LPCI pumps, associated piping and valves. During LPCI operation, the pumps take suction from the suppression pool through normally open suction valves and discharge to the reactor vessel through the discharge leg of the selected recirculation loop. The two LPCI divisions are cross-connected by piping containing two normally open motor operated valves. The cross-connection allows all of the LPCI pumps to discharge to the selected recirculation loop.

For recirculation breaks equal to or smaller than 0.15 ft^2, the loop selection logic is assumed not to detect the intact loop and LPCI is assumed to inject into the broken loop.

Each LPCI division is equipped with a minimum flow line that routes water from the pump discharge to the suppression pool. The LOCA analysis assumes the minimum flow valves remain open.

Table 3-2 LPCI System Parameters

Description	Units	Unit 5 Value
Delivery Performance (2 LPCI Pumps)		
Vessel – wetwell ΔP	psid	0
Flow	gpm	9300
Vessel – wetwell ΔP	psid	20
Flow	gpm	9000
Vessel – wetwell ΔP	psid	150
Flow	gpm	6000
Vessel – wetwell ΔP	psid	257
Flow	gpm	0
Delivery Performance (4 LPCI Pumps)		
Vessel – wetwell ΔP	psid	0
Flow	gpm	14700
Vessel – wetwell ΔP	psid	20
Flow	gpm	14200
Vessel – wetwell ΔP	psid	150
Flow	gpm	10000
Vessel – wetwell ΔP	psid	257
Flow	gpm	0
Initiating Setpoints		
Low-low water level (AVZ), AND	inch	444
Low RV pressure OR	psig	300
High drywell pressure OR	psig	2.5
sustained low-low level	sec	600
Permissive for opening injection valve and starting pumps	psig	300
Time Delays		
Signal processing delay	sec	1
Loop selected	sec	5.25
EDG output breaker closure	sec	17.3
Swing bus transfer	sec	26
Time to load pump A	sec	4
Time to load pump B	sec	7
Time for pump to reach rated speed	sec	7
Valve Stroke Times		
Recirculation discharge valve	sec	48
Injection valve	sec	30

Description	Units	Unit 5 Value
Other Parameters		
Minimum bypass flow (each division)	gpm	695

3.1.3 Low Pressure Core Spray

As shown in Figure 3-1, there are two independent LPCS systems, each with a pump, valves, piping and an independent spray sparger above the core. The major parameters associated with the LPCS system are shown in Table 3-3. The pumps take suction from the suppression pool through a common ECCS ring header, which has four suction lines located in the suppression chamber. The pumps receive an automatic start signal if a low-low reactor level signal exists concurrently with a low reactor pressure signal, or a high drywell pressure signal exists. The pumps will also start automatically if the low-low water level signal exists for a sustained time interval. Each pump is protected by a minimum flow recirculation line that prevents deadheading the pump prior to the opening of the injection valve. The minimum flow valve closes when the injection flow reaches a prescribed flow rate.

Table 3-3 LPCS System Parameters

Description	Units	Unit 5 Value
Delivery Performance		
Vessel – wetwell ΔP	psid	0
Flow	gpm	5650
Vessel – wetwell ΔP	psid	90
Flow	gpm	4500
Vessel – wetwell ΔP	psid	200
Flow	gpm	3000
Vessel – wetwell ΔP	psid	325
Flow	gpm	0
Initiating Setpoints		
Low-low water level (AVZ), AND	inch	444
Low RV pressure OR	psig	300
High drywell pressure OR	psig	2.5
sustained low-low level	sec	600
Permissive for opening injection valve and starting pumps	psig	300
Setpoint to close minimum flow valve	gpm	1250
Time delays		
Signal processing delay	sec	1
EDG output breaker closure	sec	17.3
Time to load pump	sec	12
Time for pump to reach rated speed	sec	5
Valve stroke times		

Description	Units	Unit 5 Value
Injection valve	sec	53
Minimum flow valve	sec	32
Other parameters		
Minimum bypass flow	gpm	308

3.1.4 Automatic Depressurization System

Along with the HPCI system, the Automatic Depressurization System (ADS) is used to mitigate small break LOCAs. The major parameters associated with the ADS are shown in Table 3-4. The ADS is used to depressurize the reactor vessel for small breaks to enable coolant makeup using LPCS and/or LPCI pumps. ADS depressurization is accomplished by opening the four relief valves and the dual function safety / relief valve. There are two timers in the ADS. The first timer is actuated after the coincident indication of low-low water level and high drywell pressure. The second timer is actuated after indication of low-low water level alone. The second timer times out well after the first timer and is intended to mitigate very small breaks within the containment or small breaks outside of the containment, which would not generate a high drywell pressure signal. The single active failure of the ADS is due to the mechanical failure of one relief valve to open.

Table 3-4 ADS Parameters

Description	Units	Unit 5 Value
Number of ADS valves operable	--	5
Valve capacity		
4 relief valves (each)	Mlb/hr	0.540 @ 1135 psig
1 safety / relief valve	Mlb/hr	0.622 @ 1125 psig
Initiating Setpoints		
Low-low water level (AVZ) AND	inch	444
High drywell pressure AND	psig	2.5
Timer 1 delay OR	sec	120
Low-low water level (AVZ), AND	inch	444
Timer 2 delay	sec	600
Delay times		
Valve opening time	sec	0.4
Valve closing time	sec	10
Valve reopening time	sec	14.5
Re-close pressure	psig	50
Re-open pressure	psig	100

3.2 Special Considerations

3.2.1 Changes to Plant Inputs Relative to Current Analysis of Record

Some plant performance parameters are modeled in the Westinghouse Evaluation Model of the Quad Cities and Dresden units differently than the current analysis of record for the co-resident fuel. These variations are summarized here:

(1) The Westinghouse evaluation model assumes HPCI to function within its operability constraints.

(2) Similar to the current analysis of record, the Westinghouse evaluation model credits partial opening of the core spray injection valve. However, the discharge coefficient for this gate valve was recalculated by Westinghouse.

(3) The Westinghouse evaluation model simulates the pump performance characteristics for the LPCI and LPCS pumps as conservative polynomial fits rather than as a linear interpolation between specific points.

(4) The Westinghouse evaluation model of the LPCI system injects the coolant into the discharge leg of the selected recirculation loop. The discharge coefficient of the recirculation discharge valve is used to model the flow characteristics of the valve as it closes. Similarly, the discharge coefficient of the LPCI injection valve is used to model the flow characteristics of the valve as it opens.

(5) The Westinghouse evaluation model of the leakage from the lower head drain to the pump suction leg of one of the recirculation loops is modeled as a specific flow path.

3.2.2 Leakage

The ECCS piping inside the vessel (between the vessel wall and the shroud) has various leakage paths. Some of the water injected into the reactor vessel is lost to the downcomer region where it is not available for core cooling. In addition, there are leakage paths into the downcomer region through shroud cracks, shroud repairs and jet pump slip joints. These leakage paths affect the water level inside the shroud during long term recovery. Since the leakage quantities vary from unit to unit, they are grouped into types of leakage paths and bounded in the Unit 5 model. Table 3-5 describes the various leakage paths and how they are accounted for in the Unit 5 LOCA analyses.

Table 3-5 Description of Leakage Paths Affecting ECCS Performance

Description	Units	Unit 5 Value	Notes
LPCS (per pump)			
Outside shroud	gpm	250	Assumed constant, but not greater than injected flow. Leakage added to downcomer.
Inside shroud	gpm	403.5	Assumed constant, but not greater than injected flow. Leakage added to upper plenum.
LPCI			
Jet pump leaks	gpm	811.1	Assumed constant, but not greater than injected flow. Leakage added to downcomer.
Jet pump slip joints	gpm	225	Distributed over all 20 jet pumps and based on 2/3rd core height flooding. Leakage added to downcomer.

Shroud			
Upper	gpm	350	Based on prescribed initial operating conditions. Leakage varies with calculated pressure differences. Leakage added to downcomer.
Lower	gpm	262	Based on prescribed initial operating conditions. Leakage varies with calculated pressure differences. Leakage added to downcomer.
Lower head drain to recirculation suction line	gpm	104	Based on 2/3 core height flooding. Leakage added to recirculation suction line of the unselected loop.

3.2.3 Recirculation Flow Mismatch

The LOCA model is initiated with a 10% mismatch in recirculation flow and breaks in the recirculation piping are assumed to occur in the loop providing the largest initial flow to ensure that the credit for pump coastdown is not over-estimated.

3.2.4 Variations in Core Flow

The units are permitted to operate at rated power over a range of core flow rates. This range is between 95.3% and 108% of rated core flow. Since the variation in core flow does not affect the outcome of the study to determine the limiting break size and single failure, the break spectrum analysis was performed at 108% of rate core flow and 102% of rated core power. The limiting initial core flow condition was then determined for the limiting break size and single failure combination (see Section 5.2).

4.0 Break Spectrum Analysis

The importance of various single failures depends on break location as the location of the break can disable an ECCS component. Each ECCS subsystem is designed to ensure adequate core cooling across the entire spectrum of line break accidents when operated with other available ECCS subsystems determined from the Appendix K single active failure criterion. Table 4-1 lists the break locations considered and the ECCS equipment available under different postulated single active failures. Figure 4-1 shows the GOBLIN nodalization diagram that was used for the break spectrum study. Boundary conditions from GOBLIN were provided to a stand-alone DRAGON model for the hot assembly. The power of the hot assembly is established at a conservative initial CPR operating limit of 1.41. Figure 4-2 shows the DRAGON nodalization diagram that was used for the break spectrum analysis. Boundary conditions from the hot assembly analysis were provided to CHACHA to determine the peak cladding temperature and maximum oxidation. The same lattice, nodal exposure and nodal peaking were used for all CHACHA cases. This ensured that the LOCA system response could be compared on the same basis.

Table 4-1 Break Location, Single Failure and ECCS Availability

Case	Equipment Available				Break Locations to Consider				Failure / Comments
	LPCS	LPCI	HPCI	ADS	Recirc Line	Steam Line	LPCS	FW Line	
1	2	0	1	5	X	X			LPCI IV failure
2	1	2	1	5	X	X			EDG failure
3	2	4	0	5	X	X			HPCI failure
4	2	4	1	5	X	X			Loop select failure
5	2	4	1	4	X	X			ADS failure – important only for small breaks
6	1	0	1	5			X		LPCI IV failure
7	0	2	1	5			X		EDG failure + break in powered LPCS line
8	1	4	0	5			X		HPCI failure
9	1	4	1	5			X		Loop select failure has no effect due to break location
10	1	4	1	4			X		ADS failure – important only for small breaks
11	2	0	0	5				X	LPCI IV failure
12	1	2	0	5				X	EDG failure
13	2	4	0	5				X	HPCI failure has no effect since break location disables HPCI
14	2	4	0	5				X	Loop select failure has no effect due to break location
15	2	4	0	4				X	ADS failure – important only for small breaks

a,c

Figure 4-1 GOBLIN Nodalization for Break Spectrum Study

a,c

Figure 4-2 DRAGON Nodalization for Break Spectrum Study

4.1 Break Spectrum Results and Conclusions

The break spectrum study indicated that breaks in the recirculation lines were more limiting with regard to core cooling. The break spectrum study evaluated breaks in different locations and single active failures (consistent with Table 4-1). Depending on the single failure assumption, break sizes were varied until the limiting break area could be determined. As shown in Figure 4-3, break configurations between the full double-ended break of the recirculation line and slot breaks having an area of 0.1 ft^2 were evaluated.

As shown, the limiting break is a large double-ended guillotine break in the pump suction line with the single failure of the LPCI injection valve. The limiting small break is a 0.15 ft^2 break in the pump discharge line of the selected loop with single failure HPCI.

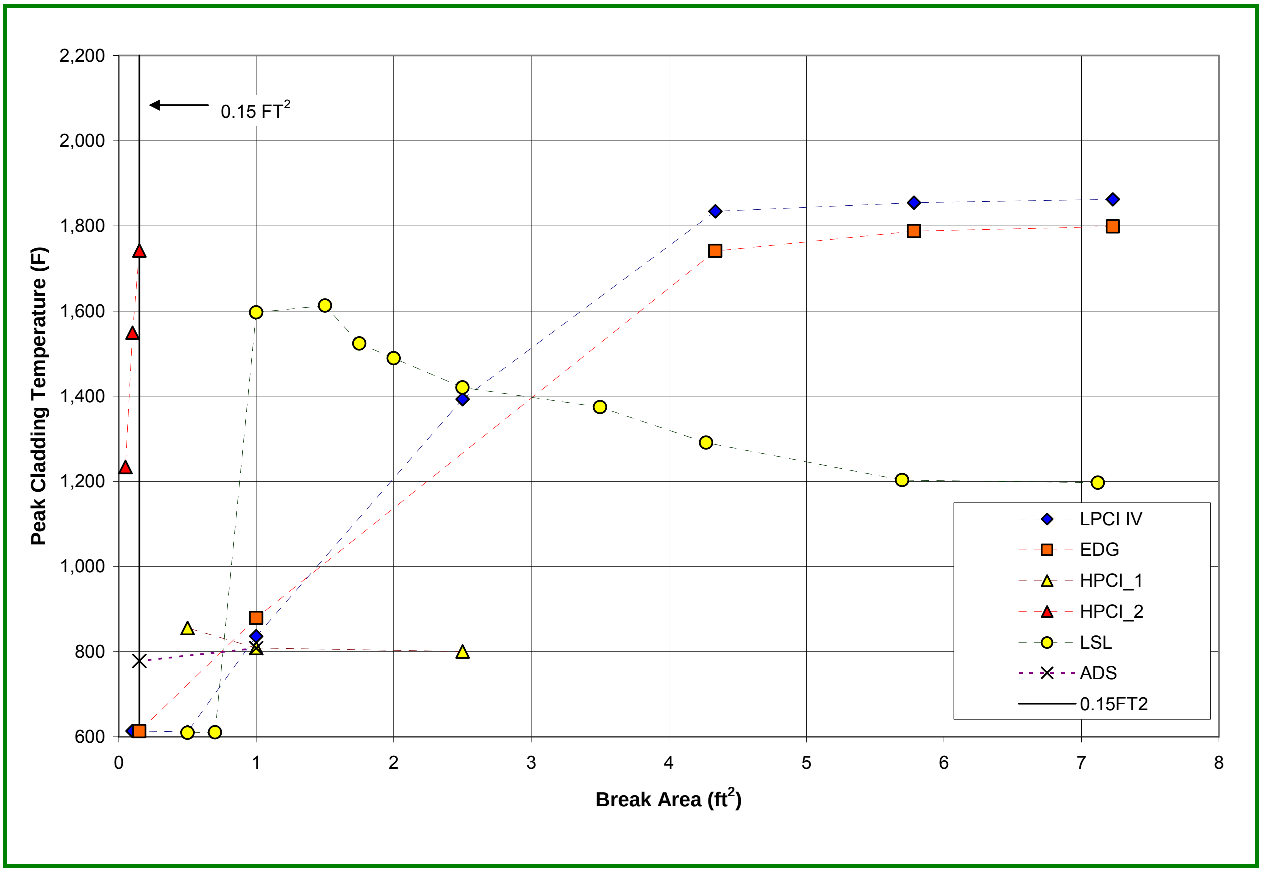

Figure 4-3 Summary of Break Spectrum Results

4.2 Recirculation Line Breaks

Breaks in the recirculation line are usually limiting with respect to impact on ECCS performance. Because of their location, the break mass flow rate is higher than it would be for breaks of the same area, but at higher elevations. The flow rate from a break in the pump suction line is usually limiting compared to breaks in the pump discharge line. However, for breaks smaller than 0.15 ft^2, the loop select logic cannot reliably determine the location of the intact loop. Therefore, it is assumed to fail to detect the intact loop for breaks of this size and smaller. In this case, a break in the pump discharge line is more limiting when LPCI is assumed to function since much of the LPCI water injected into the pump discharge line will spill out the break before reaching the reactor vessel.

The following subsections discuss the analyses performed for the various single failures considered.

4.2.1 Case 1: LPCI Injection Valve Failure

This case investigated the single failure of the LPCI injection valve. In this situation, no coolant injection from the LPCI system occurs. However, two LPCS pumps, one HPCI pump and five ADS valves are assumed to be operable. For large breaks, the system depressurizes very rapidly and neither HPCI nor ADS are actuated.

Several break sizes in the suction line were evaluated. Breaks in the discharge line are generally not as limiting due to the flow restriction of the jet pump nozzles on the vessel side of the break. As shown in Table 4-2, the limiting break was the large double-ended guillotine break in the pump suction line.

Figure 4-4 through Figure 4-6 show the predicted dome pressure, LPCS injection, system mass and predicted peak cladding temperature (PCT) for the limiting full double-ended guillotine break. As shown, the dome pressure increases at the beginning of the event due to the closure of the turbine stop valve. This results in a small power increase due to the void feedback effect. However, reactor trip occurs very rapidly resulting in a decrease in reactor power and a rapid decrease in dome pressure. The pressure decreases below the minimum pressure at which HPCI can operate before the HPCI pump can start. As a result, there is no HPCI injection for this break. At approximately 30 seconds, the pressure reaches the pressure permissive and the LPCS pumps can start injecting. Flow from the LPCS pumps enters the spray spargers in the upper plenum where it can flow downward through the core or the bypass channels. The water that flows into the core provides cooling directly to the fuel rods. The water that flows down the bypass channels refills the lower plenum until the water level reaches the core inlet. After this time, the flow through the core switches from counter-current flow to co-current upward flow.

Table 4-2 shows how the PCT decreases with decreasing break size. For the very small breaks, the core does not uncover. Figure 4-7 through Figure 4-10 show the results for the 1.0 ft^2 break. As shown, HPCI is actuated at approximately 45 seconds. HPCI flow ceases when the system pressure decreases below the required low pressure cutoff. Both LPCS pumps actuate at approximately 100 seconds when the system pressure decreases below the pressure permissive setpoint (300 psig). LPCS actuation is prior to ADS actuation, which occurs at approximately 135 seconds. The result is a very brief uncovery of the core and a small heatup before two-phase cooling conditions are restored.

Table 4-2 Case 1 (LPCI IV Failure): PCT Results for Recirculation Line Breaks

1.0 DEG PS	0.8 DEG PS	0.6 DEG PS	2.5 FT^2 PS	1.0 FT^2 PS	0.5 FT^2 PS	0.1 FT^2 PS
1862 °F	1854 °F	1834 °F	1393 °F	836 °F	612 °F	614 °F

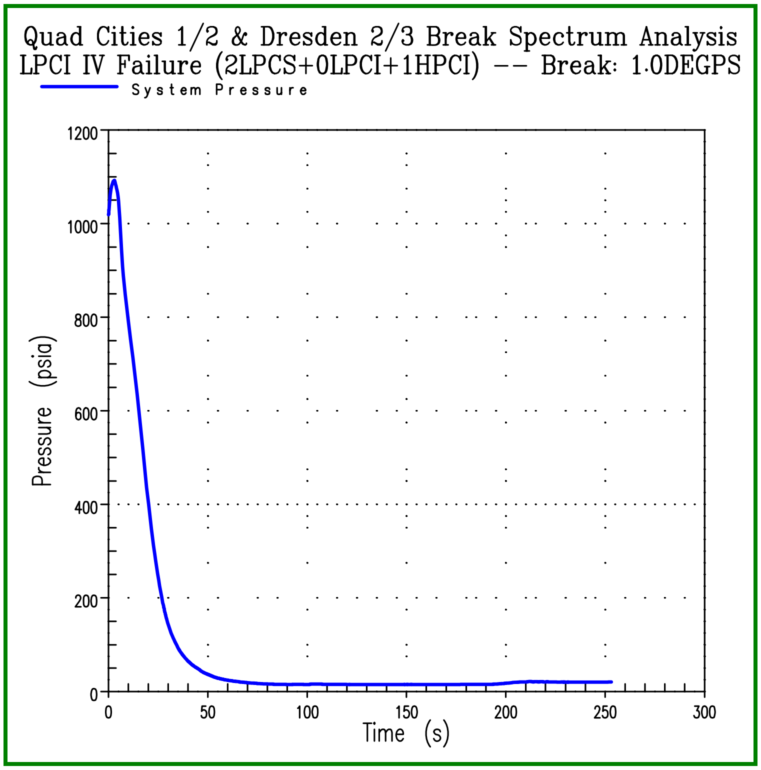

Figure 4-4 Case 1: Dome Pressure for 1.0DEGPS Break

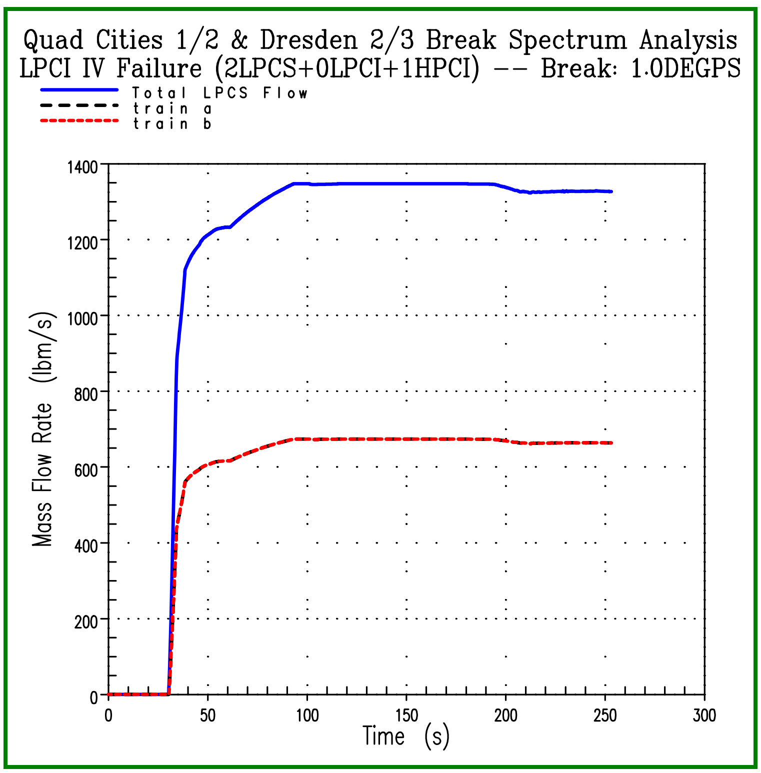

Figure 4-5 Case 1: LPCS Flow Rate for 1.0DEGPS Break

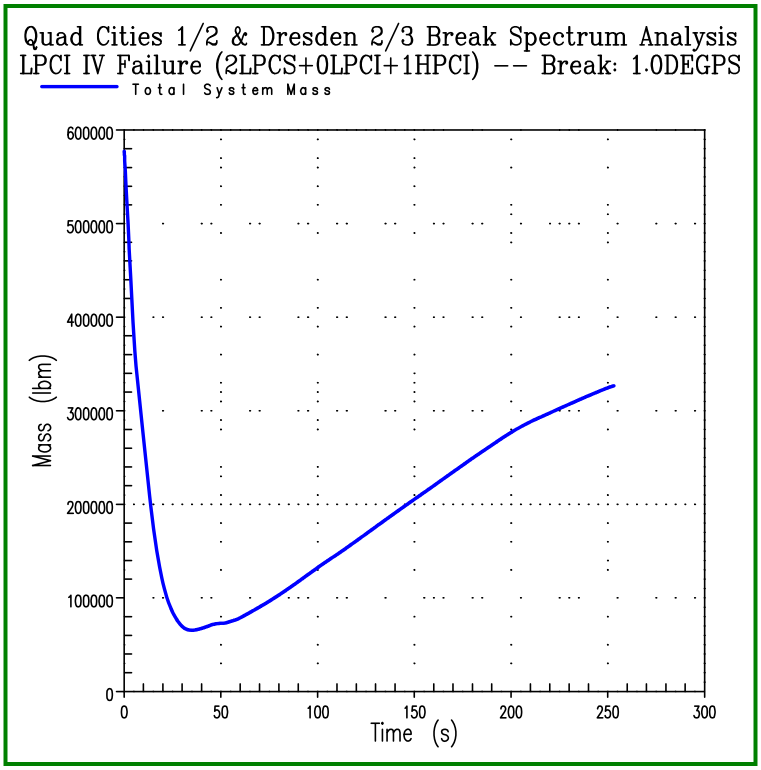

Figure 4-6 Case 1: System Mass for 1.0DEGPS Break

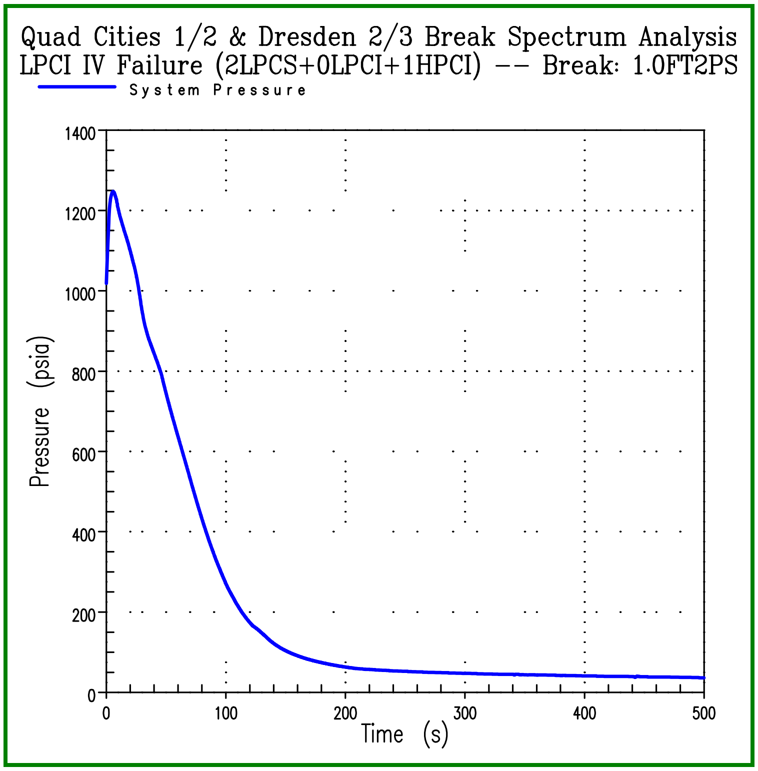

Figure 4-7 Case 1: Dome Pressure for 1.0 ft^2 Pump Suction Break

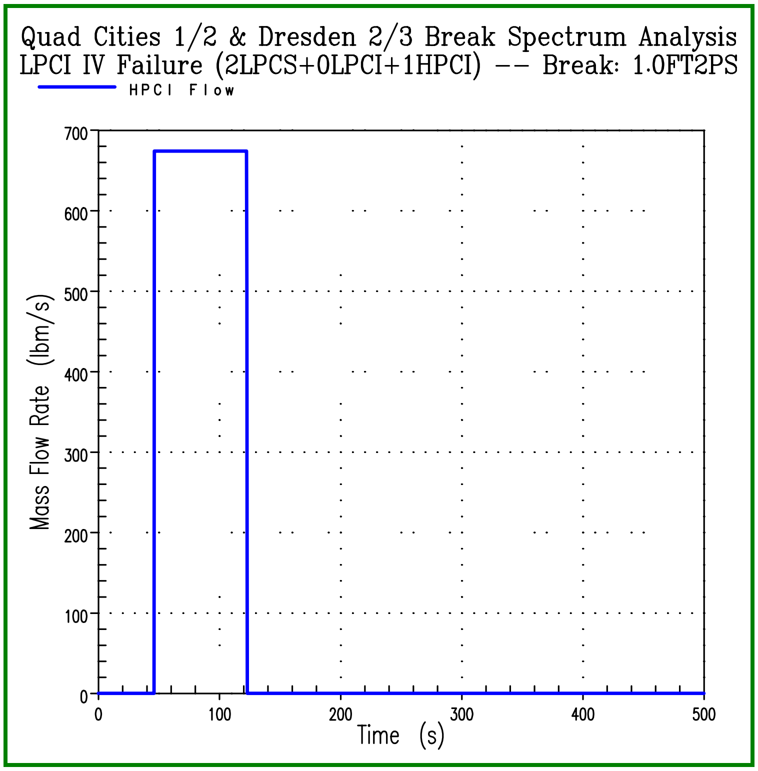

Figure 4-8 Case 1: HPCI Flow Rate for 1.0 ft^2 Pump Suction Break

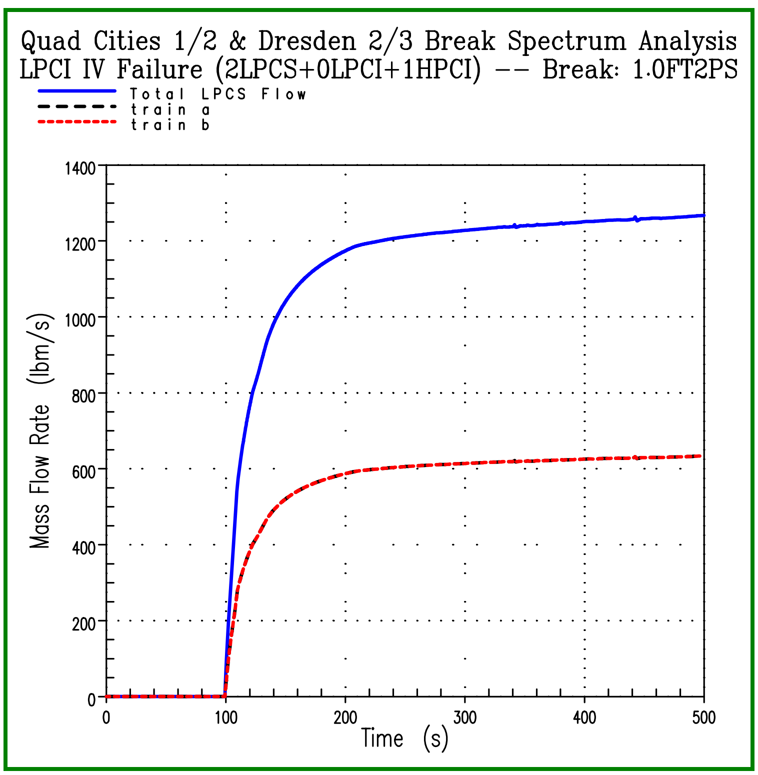

Figure 4-9 Case 1: LPCS Flow Rate for 1.0 ft^2 Pump Suction Break

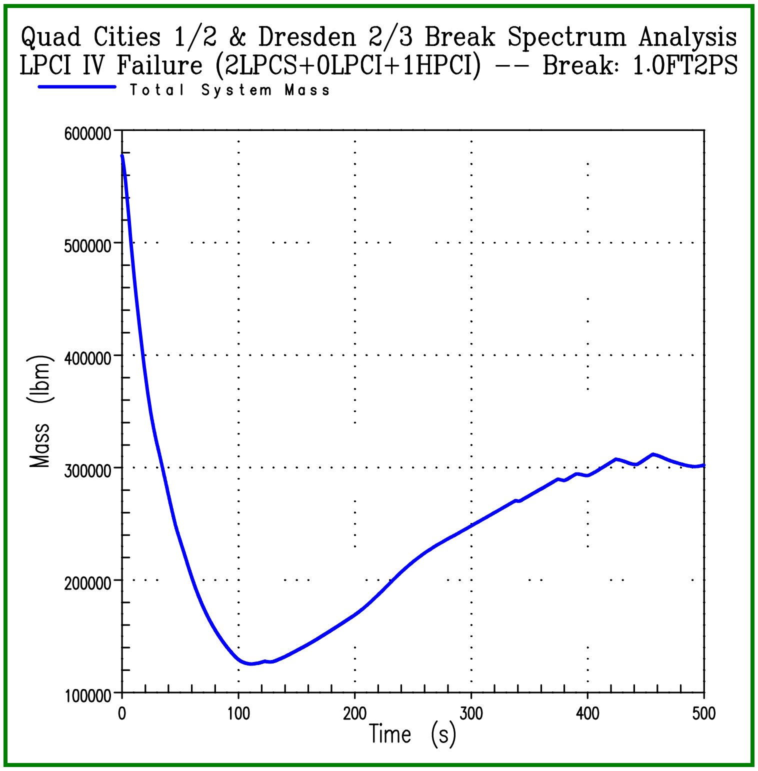

Figure 4-10 Case 1: System Mass for 1.0 ft^2 Pump Suction Break

4.2.2 Case 2: EDG Failure

This case investigates the single failure of one of the emergency diesel generators (EDGs) to start. For this failure, only one train of equipment will be powered. Therefore, one LPCS pump, two LPCI pumps, one HPCI pump and five ADS valves will be operable. For large breaks, the system depressurization is rapid and neither HPCI pump nor ADS actuate.

The loop select logic will select the intact loop and align the LPCI pumps to the intact loop for break sizes greater than 0.15 ft^2. For breaks equal to or smaller than 0.15 ft^2, it is assumed that loop select logic will not reliably select the intact loop. In these situations it is assumed that the break is located in the pump discharge of the selected loop. As a result, much of the LPCI water injected into the pump suction leg will be spilled out the break. For recirculation line break areas greater than 0.15 ft^2, the break is assumed to be located in the pump suction line as it results in the most inventory loss.

As shown in Table 4-3, the large double-ended break in the pump suction line is limiting although not as limiting as the large double-ended break for the LPCI injection valve failure. Figure 4-11 through Figure 4-14 show the graphical results for the large double-ended pump suction break. The LPCS and LPCI pumps begin to inject at approximately 30 and 38 seconds respectively (the LPCS pumps have a higher shutoff head than the LPCI pumps). The system pressure decreases below the low pressure threshold for the HPCI pump before it can start. As shown in Figure 4-14, the system mass begins to recover after the ECCS pumps begin to inject. Two-phase cooling conditions are restored at the core midplane by approximately 175 seconds.

Figure 4-15 through Figure 4-19 show the graphical results for the 1.0 ft2 pump suction break. As shown, HPCI actuates at approximately 48 seconds, the LPCS pump begins to inject at approximately 100 seconds; the LPCI pumps begin to inject at approximately 132 seconds. All pumps begin injecting before ADS actuates at approximately 132 seconds.

The 0.15 ft^2 break simulated a break in the discharge leg into which LPCI injects. This simulates the failure of the loop select logic to identify the broken loop. The graphical results of this case are not shown as there was no core uncovery. HPCI injection maintained reactor inventory until ADS actuated and the system pressure reduced sufficiently for the LPCS and LPCI pumps to inject and recover the inventory. The mission time of the HPCI pump was less than 10 minutes as required for cases involving failure of one of the EDGs.

Table 4-3 Case 2 (EDG Failure): PCT Results for Recirculation Line Breaks

1.0 DEG PS	0.8 DEG PS	0.6 DEG PS	1.0 FT^2 PS	0.15 FT^2 PD
1799 °F	1788 °F	1741 °F	879 °F	613 °F

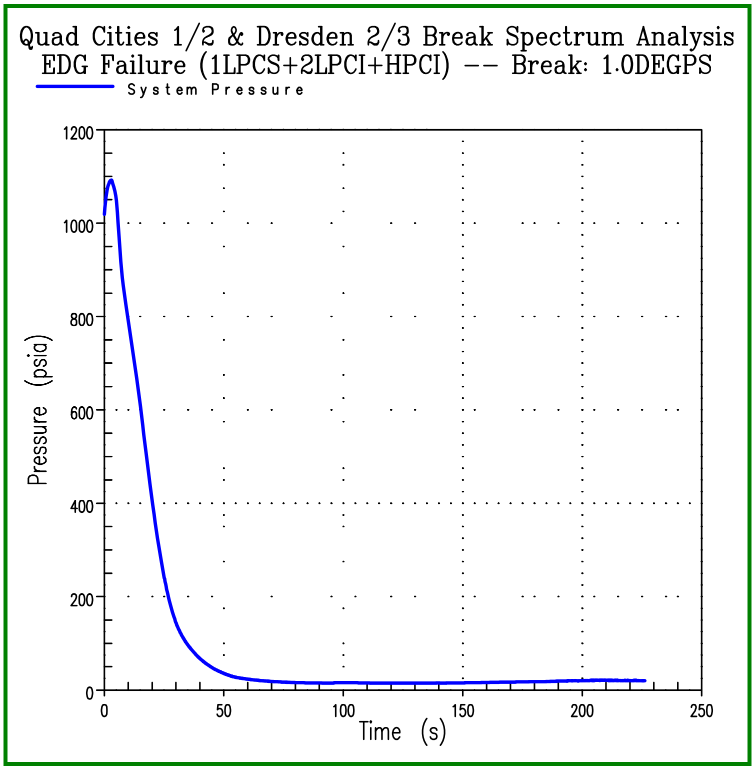

Figure 4-11 Case 2: Dome Pressure for 1.0DEGPS Break

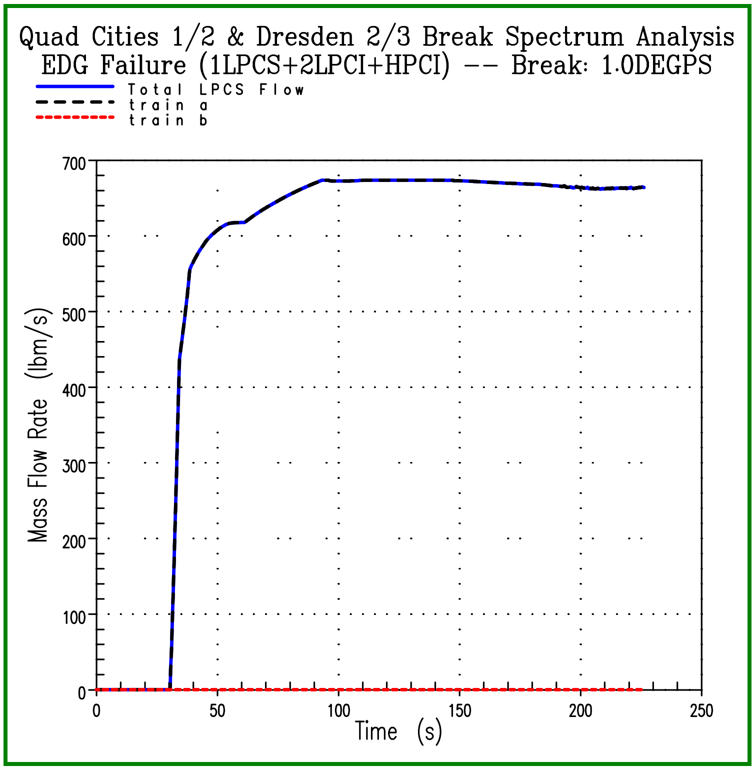

Figure 4-12 Case 2: LPCS Flow Rate for 1.0DEGPS Break

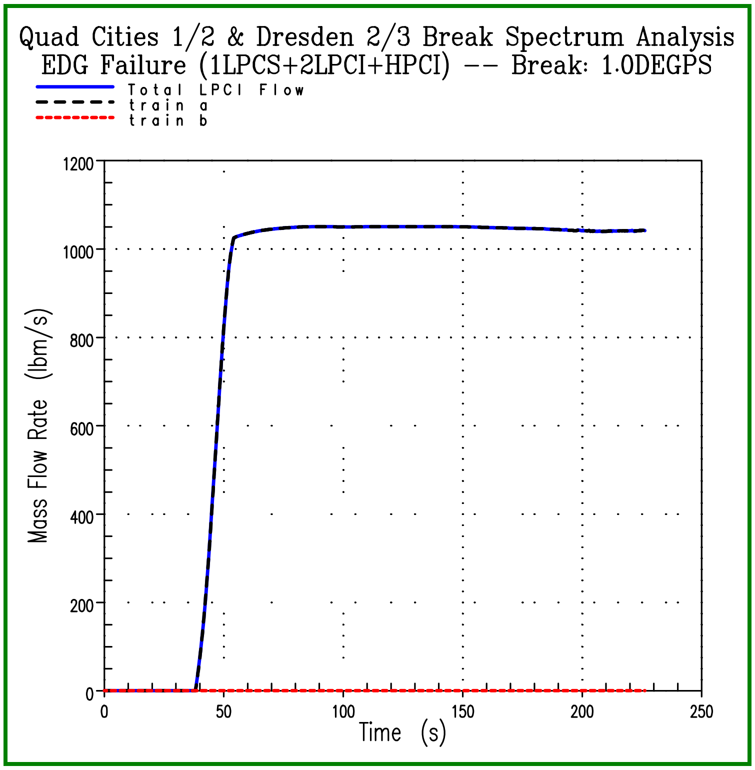

Figure 4-13 Case 2: LPCI Flow Rate for 1.0DEGPS Break

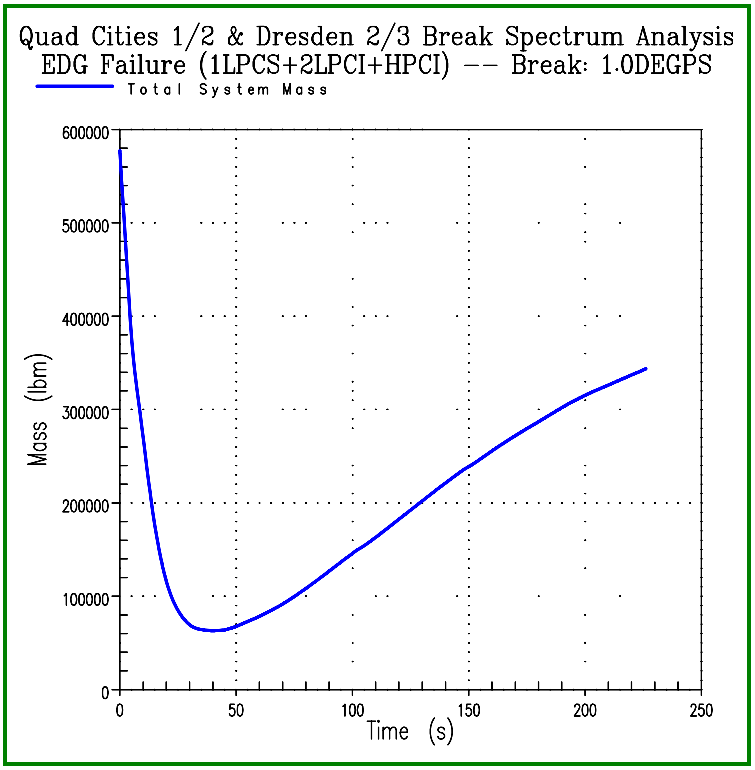

Figure 4-14 Case 2: System Mass for 1.0DEGPS Break

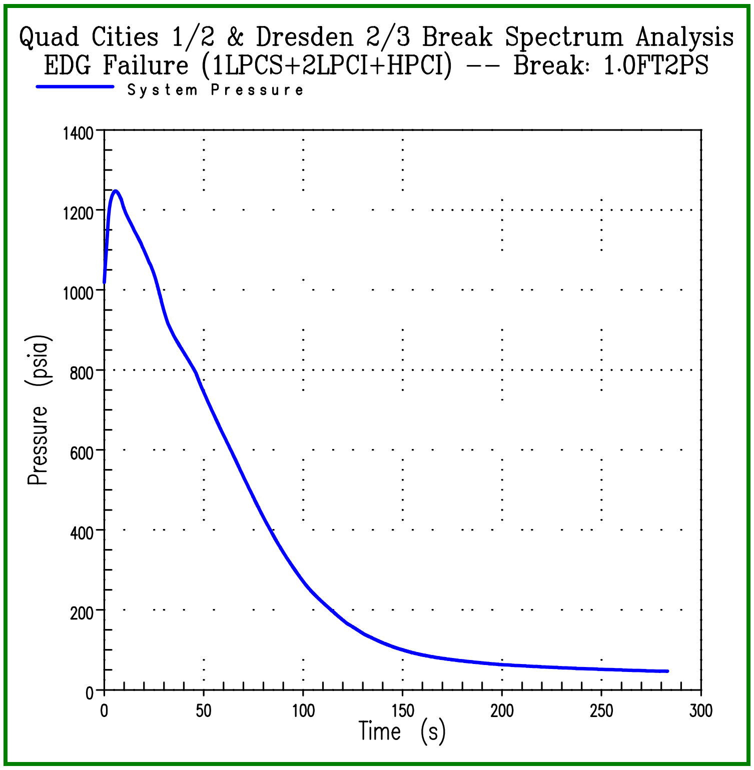

Figure 4-15 Case 2: Dome Pressure for 1.0 Ft^2 Pump Suction

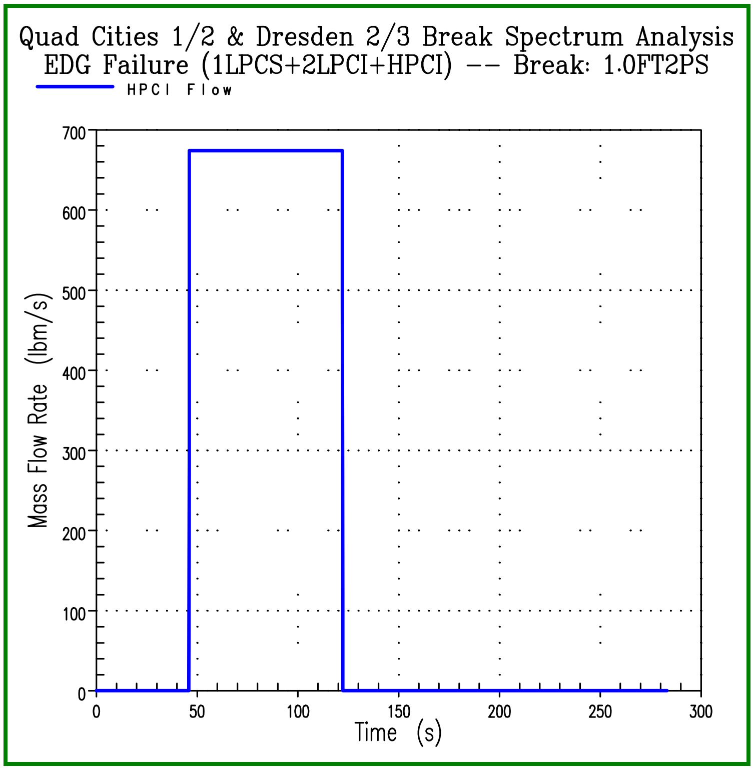

Figure 4-16 Case 2: HPCI Flow Rate for 1.0 Ft^2 Pump Suction Break

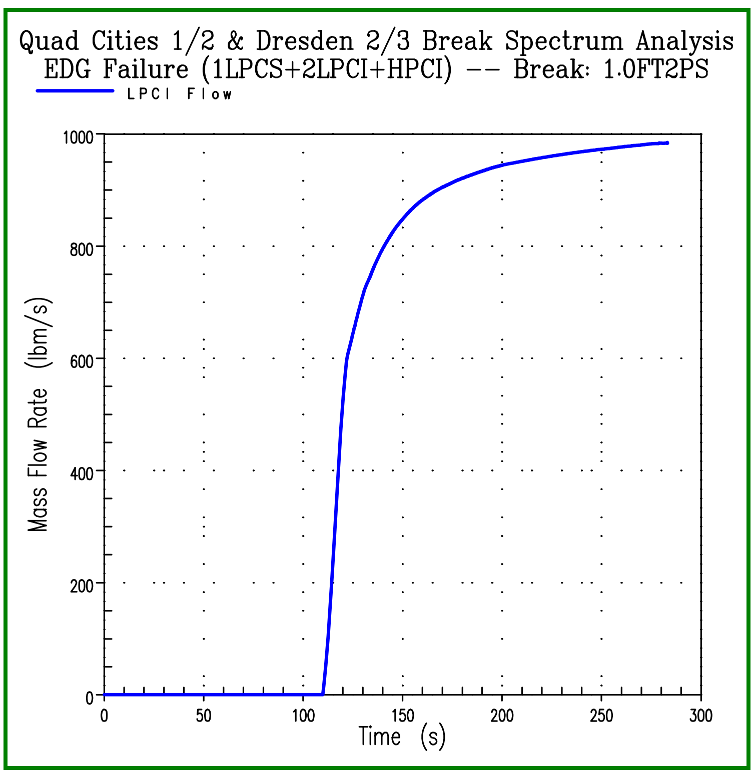

Figure 4-17 Case 2: LPCI Flow Rate for 1.0 Ft^2 Pump Suction Break

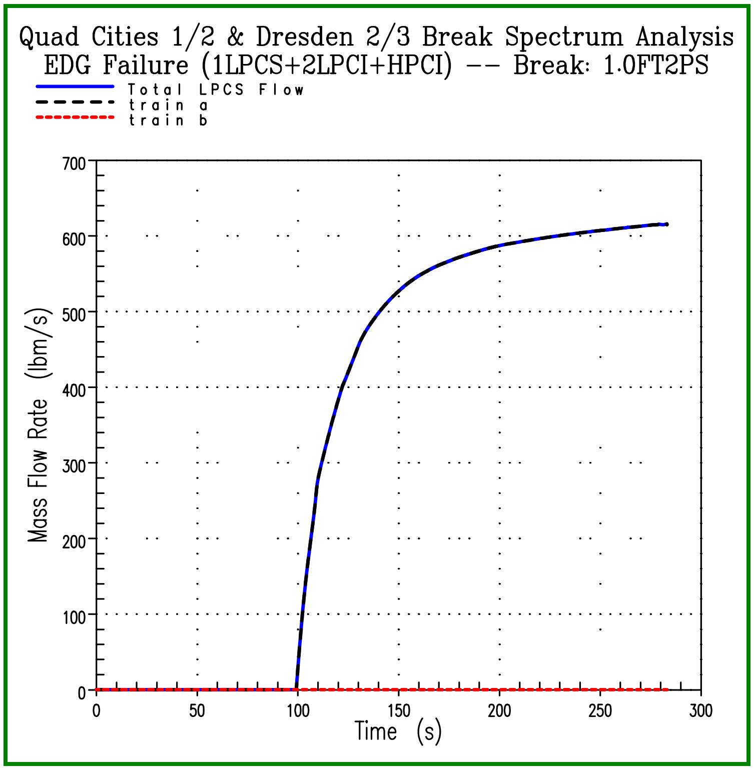

Figure 4-18 Case 2: LPCS Flow Rate for 1.0 Ft^2 Pump Suction Break

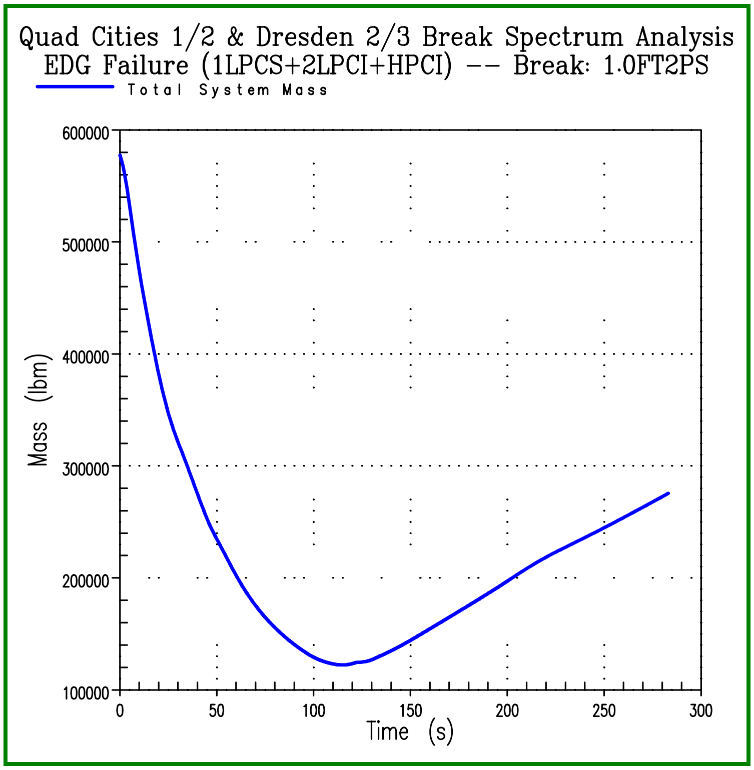

Figure 4-19 Case 2: System Mass for 1.0 Ft^2 Pump Suction Break

4.2.3 Case 3: HPCI Failure

This case considers the failure of the HPCI system. Since HPCI provides flow over a restricted range of system pressure, it does not have time to actuate for large double-ended breaks. Therefore, this investigation evaluates intermediate to small breaks only. In this situation it is assumed that two LPCS pumps, four LPCI pumps (two trains) and five ADS valves will be operable.

The results of the study showed that the uncovery time was very short for the 2.5 ft^2, 1.0 ft^2 and 0.5 ft^2 break. Figure 4-20 through Figure 4-23 show the graphical results for the 1.0 ft^2 break. In this case, the system pressure decreases below the pressure permissive for the low pressure pumps before ADS actuated. However, for break areas less than or equal to 0.15 ft^2, the loop select logic is assumed to fail. In this case the break is placed in the discharge line of the selected loop, which results in much of the injected LPCI water spilling out the break. As shown in Table 4-4, the 0.15 ft^2 break is the limiting break in this series. Figure 4-24 through Figure 4-27 show the graphical results for the 0.15 ft^2 break. As shown in Figure 4-24, the safety valves control system pressure for the first part of the transient until ADS actuates at approximately 160 sec. After ADS actuation, the system depressurizes to the permissive, which initiates the ECCS pumps. As shown in Figure 4-27, the system mass begins to recover shortly after ECCS actuation. The peak cladding temperature occurs at approximately 365 seconds.

Table 4-4 Case 3 (HPCI Failure): PCT Results for Recirculation Line Breaks

2.5 FT^2 PS	1.0 FT^2 PS	0.50 FT^2 PS	0.15 FT^2 PD	0.10 FT^2 PD	0.05 FT^2 PD
800 °F	809 °F	856 °F	1742 °F	1549 °F	1234 °F

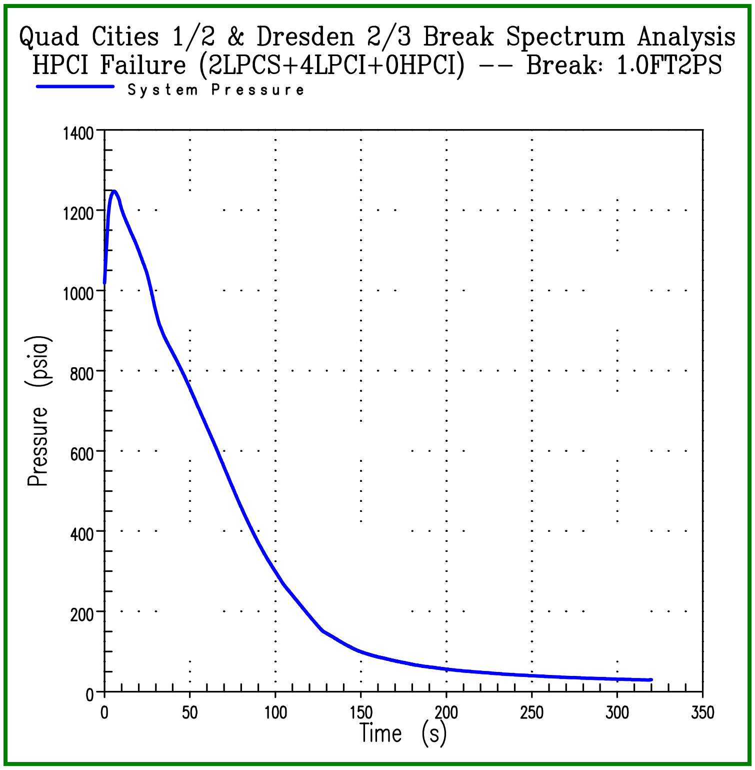

Figure 4-20 Case 3: Dome Pressure for 1.0 Ft^2 Pump Suction Break

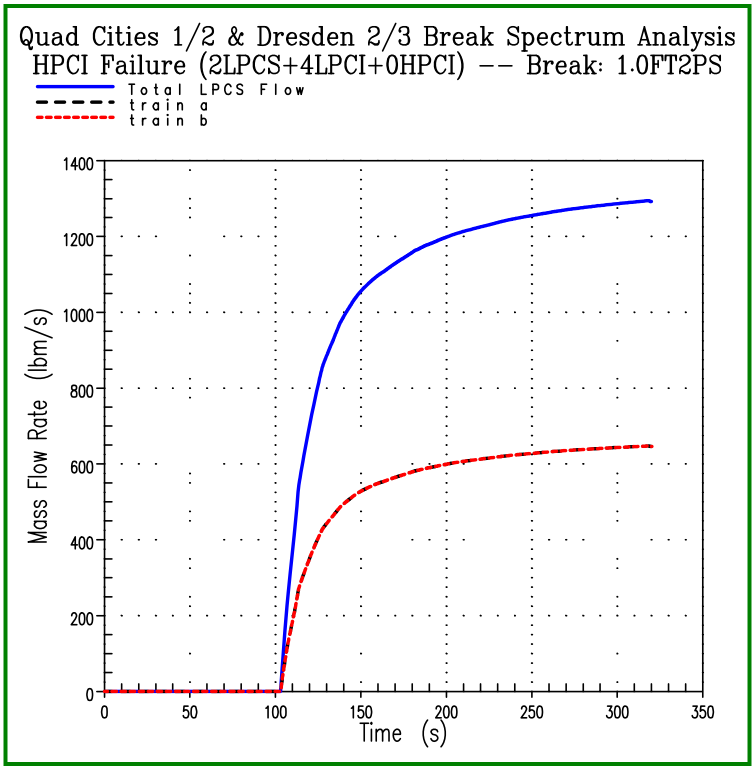

Figure 4-21 Case 3: LPCS Flow Rate for 1.0 Ft^2 Pump Suction Break

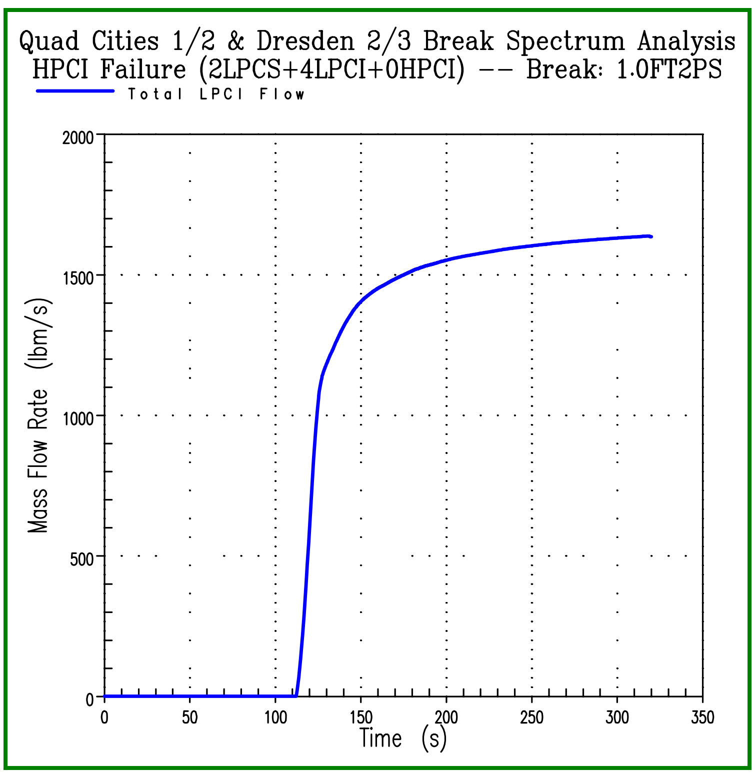

Figure 4-22 Case 3: LPCI Flow Rate for 1.0 Ft^2 Pump Suction Break

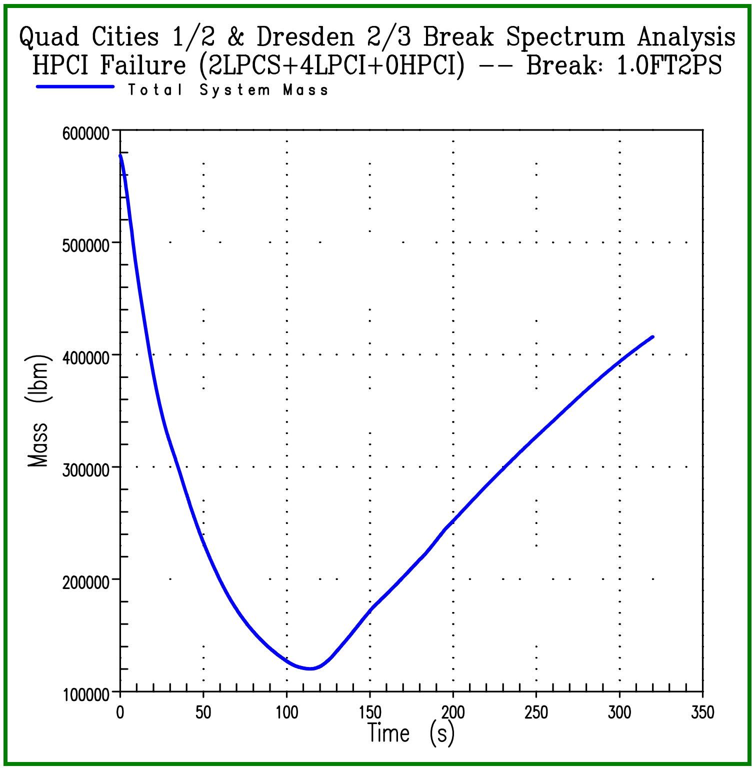

Figure 4-23 Case 3: System Mass for 1.0 Ft^2 Pump Suction Break

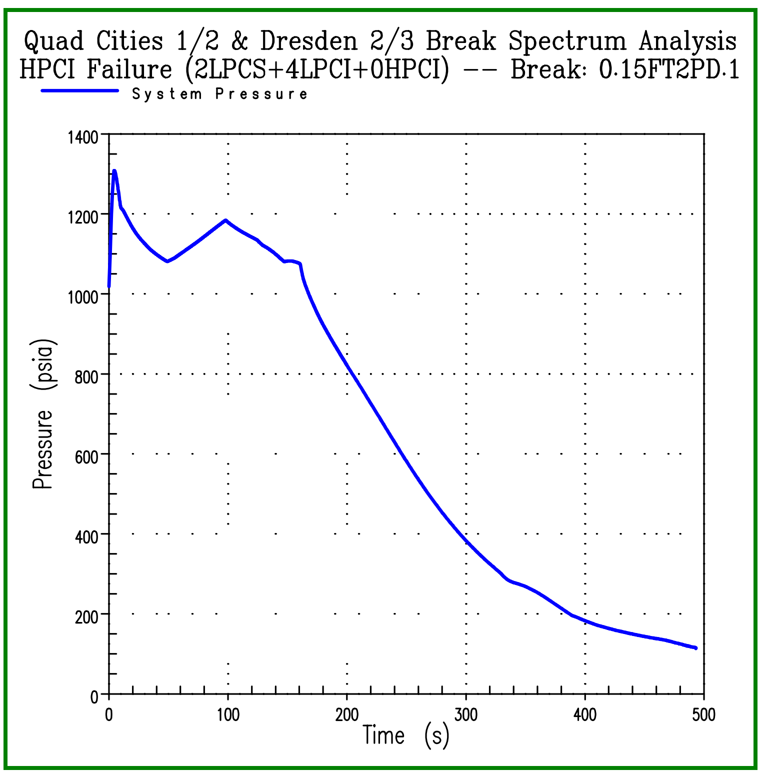

Figure 4-24 Case 3: Dome Pressure for 0.15 Ft^2 Pump Discharge Break

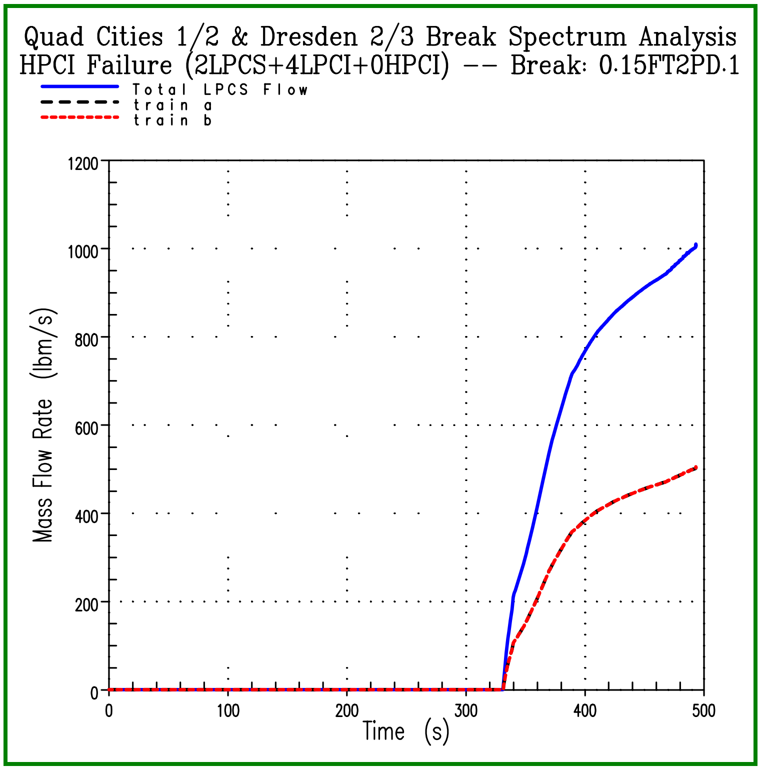

Figure 4-25 Case 3: LPCS Flow Rate for 0.15 Ft^2 Pump Discharge Break

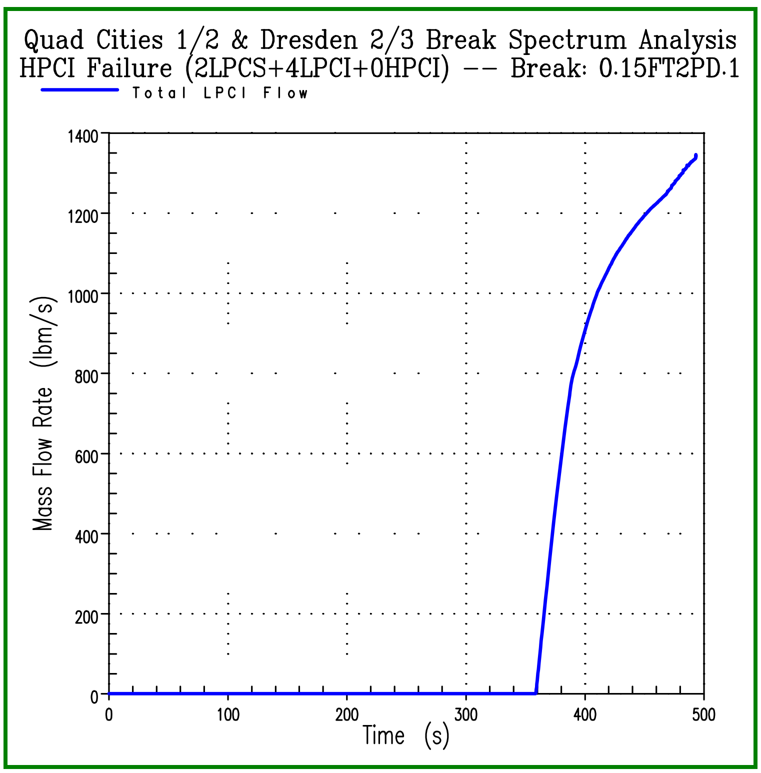

Figure 4-26 Case 3: LPCI Flow Rate for 0.15 Ft^2 Pump Discharge Break

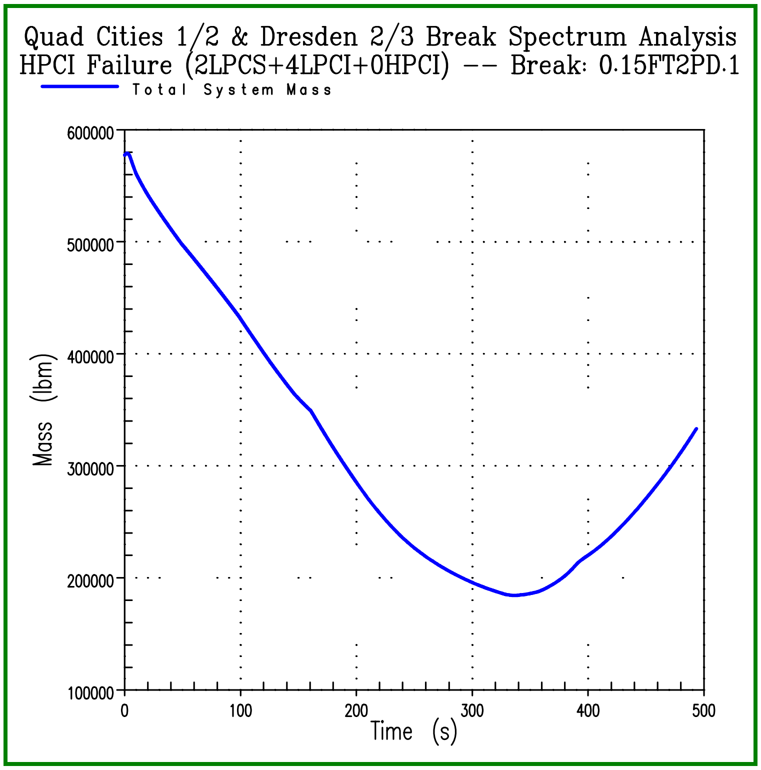

Figure 4-27 Case 3: System Mass for 0.15 Ft^2 Pump Discharge Break

4.2.4 <u>Case 4: Loop Select Logic Failure</u>

This case considers the failure of the loop select logic (LSL) to detect and select the intact loop in the event of a break in the recirculation line. For these cases, it was assumed that the break was in the discharge leg so that the water injected by the LPCI system would be lost out the break.

Due to the location of the break, the break flow rate is smaller than for the same sized suction leg break. As a result, the system pressure decreases more slowly and the HPCI system has time to actuate even for the largest break size. When the system pressure decreases below the permissive setpoint, the two LPCS pumps and four LPCI pumps begin to inject and the system mass begins to increase shortly afterward. The trend of PCT with break size shown in Table 4-5 indicates that small breaks would be limiting for this single failure. Figure 4-28 through Figure 4-32 show the graphical results for the limiting 1.5 ft^2 pump discharge break. A comparison of the break flow rate to the LPCI flow rate indicates that the break flow increases after LPCI actuation and that all of the water injected by LPCI is lost out the break.

Table 4-5 Case 4 (LSL Failure): PCT Results for Recirculation Line Breaks

7.1 FT^2 PD	**5.7 FT^2 PD**	**4.3 FT^2 PD**	**3.5 FT^2 PD**	**2.5 FT^2 PD**	**2.0 FT^2 PD**	**1.75 FT^2 PD**	**1.5 FT^2 PD**
1196 °F	1203 °F	1291 °F	1374 °F	1420 °F	1489 °F	1524 °F	1613 °F
1.0 FT^2 PD	**0.7 FT^2 PD**	**0.5 FT^2 PD**					
1597 °F	610 °F	610 °F					

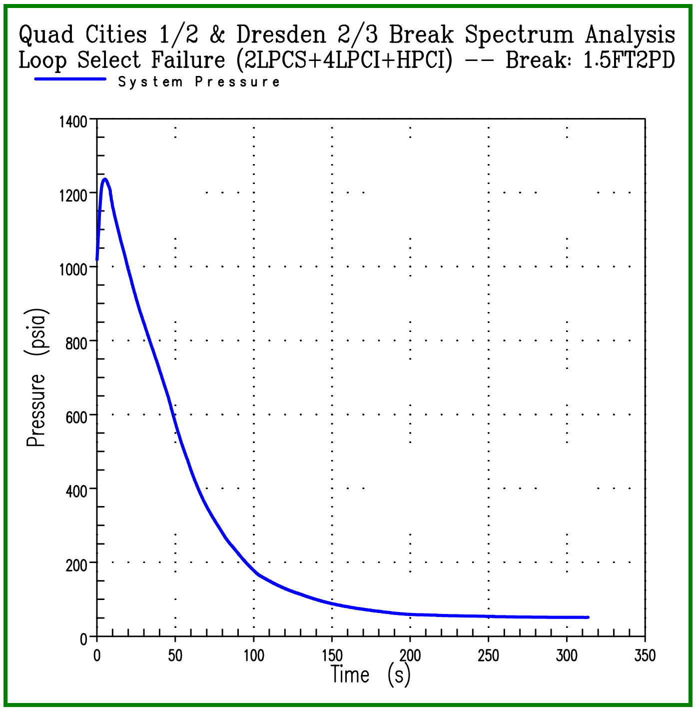

Figure 4-28 Case 4: Dome Pressure for 1.5 Ft^2 Pump Discharge Break

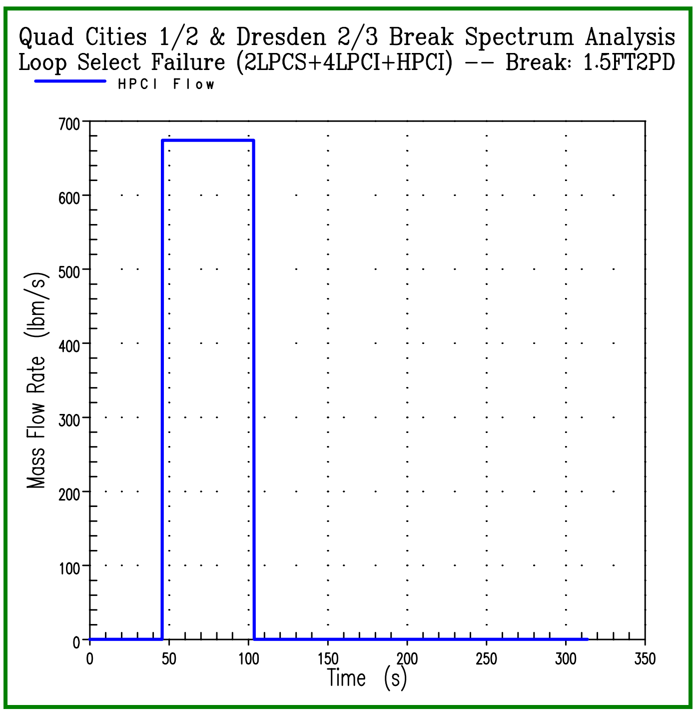

Figure 4-29 Case 4: HPCI Flow for 1.5 Ft^2 Pump Discharge Break

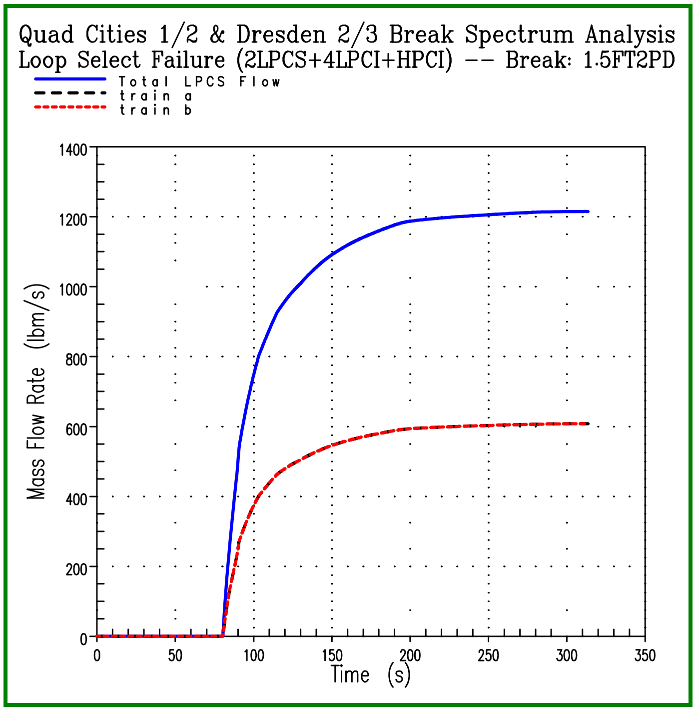

Figure 4-30 Case 4: LPCS Flow for 1.5 Ft^2 Pump Discharge Break

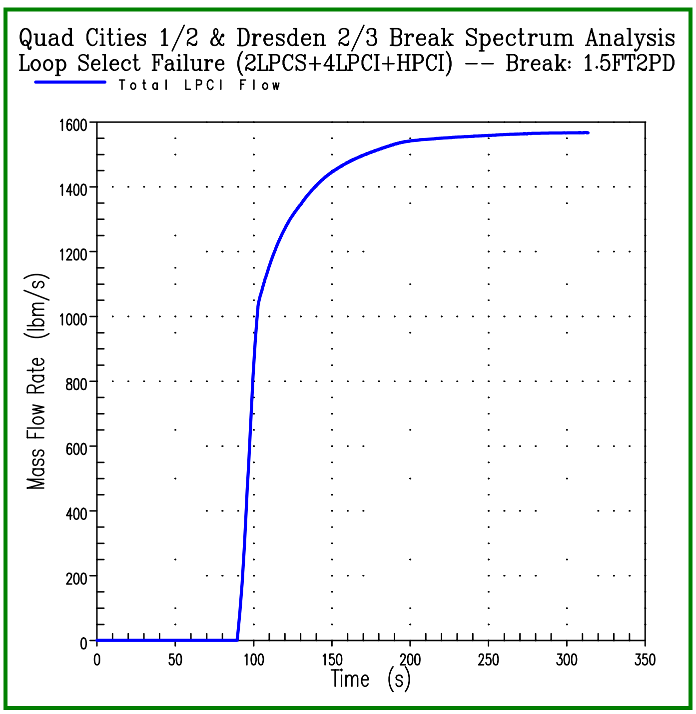

Figure 4-31 Case 4: LPCI Flow for 1.5 Ft^2 Pump Discharge Break

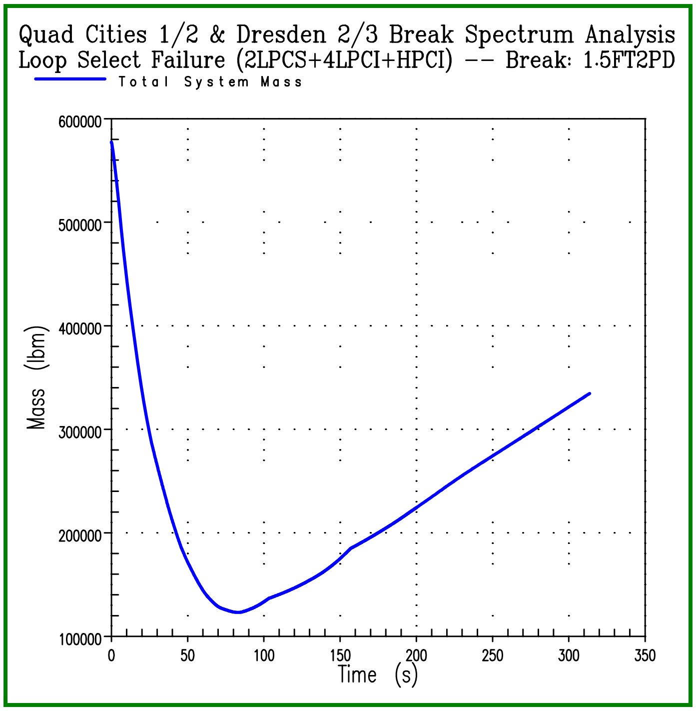

Figure 4-32 Case 4: System Mass for 1.5 Ft^2 Pump Discharge Break

4.2.5 Case 5: ADS Failure

This case considers a mechanical failure of one of the five ADS valves. The valve with the largest capacity was assumed to fail in the closed position. Since ADS valves are important only for small breaks, large breaks were not evaluated in this study. One HPCI pump, two LPCS pumps, four LPCI pumps and four ADS valves were assumed to function. Table 4-6 shows the PCTs for the various break sizes considered for this single failure. As shown, there was no significant heatup for any of these cases. The 1.0 ft^2 break in the pump discharge line depressurized quickly enough to allow the low pressure pumps to start before ADS actuated.

There was a very short uncovery of the midplane before two-phase conditions were restored. The 0.15 ft^2 break was placed in the pump discharge line as it is assumed that loop select logic will not detect the recirculation line containing the break for breaks of this size and smaller. In this case, the reactor depressurized more slowly and ADS actuation was required to depressurize the reactor vessel below the permissive pressure of the low pressure ECCS pumps. However, HPCI actuated earlier and there was a very short uncovery at the midplane before conditions were restored.

There was no uncovery for the 0.1 ft^2 pump discharge break. Figure 4-33 through Figure 4-37 show the graphical results for the 0.15 ft^2 break.

Table 4-6 Case 5 (ADS Failure): PCT Results for Recirculation Line Breaks

1.0 FT^2 PS	0.15 FT^2 PD	0.10 FT^2 PD
808 °F	778 °F	N

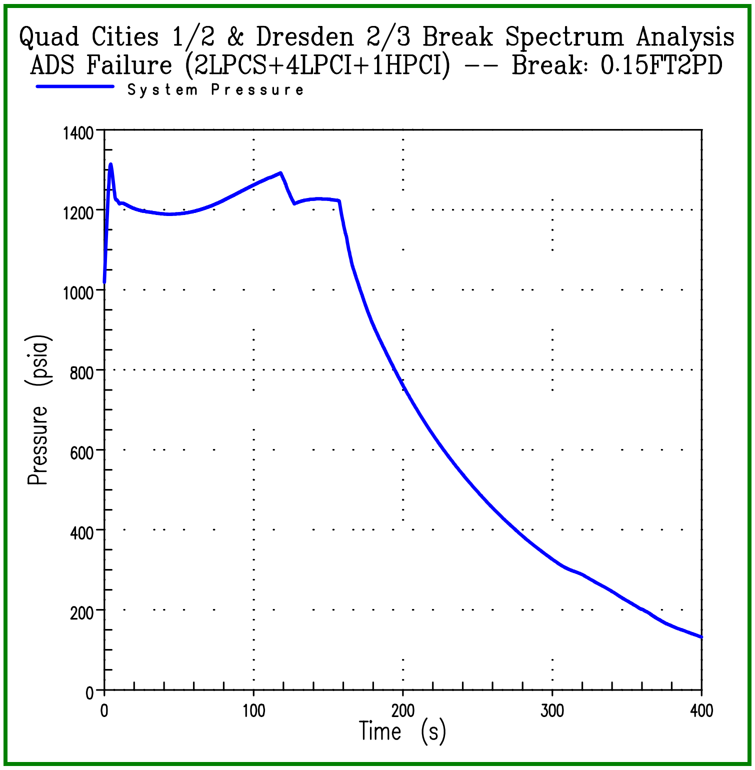

Figure 4-33 Case 5: Dome Pressure for 0.15 Ft^2 Pump Discharge Break

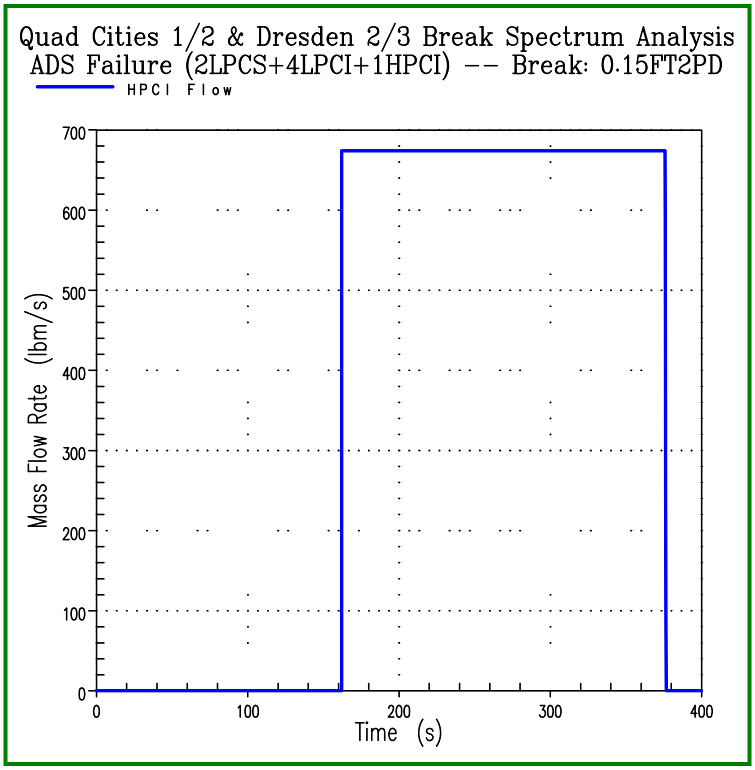

Figure 4-34 Case 5: HPCI Flow for 0.15 Ft^2 Pump Discharge Break

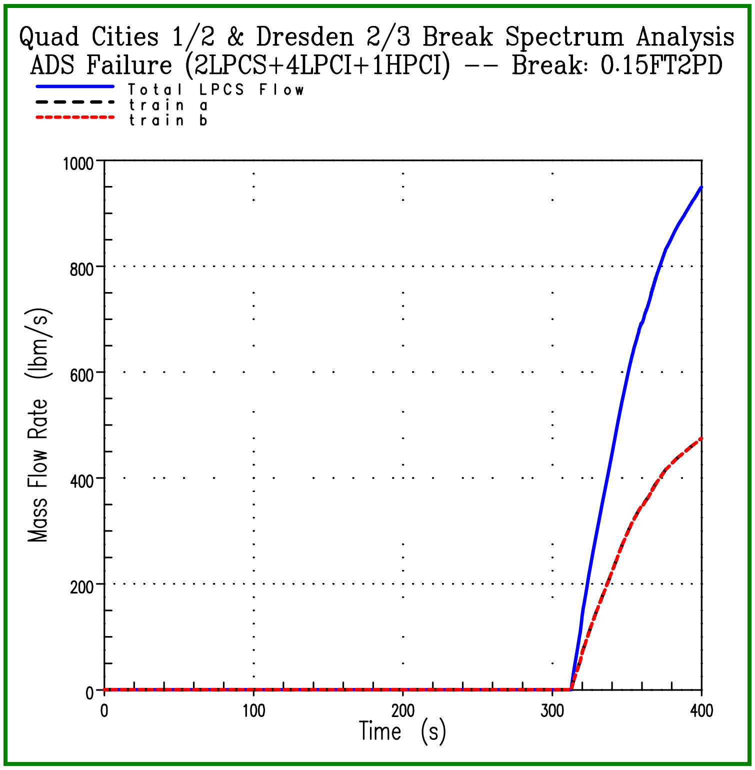

Figure 4-35 Case 5: LPCS Flow for 0.15 Ft^2 Pump Discharge Break

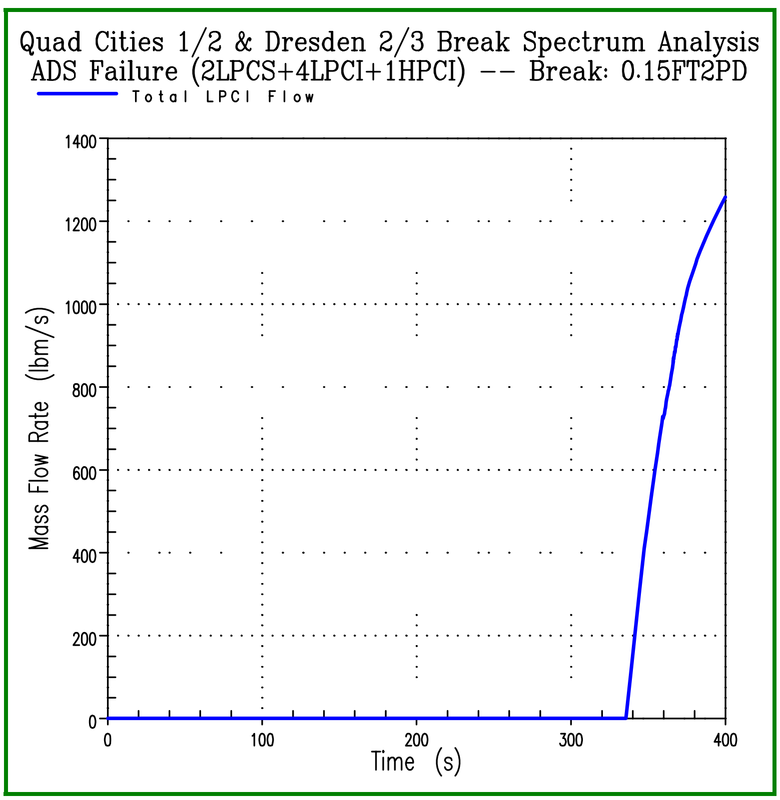

Figure 4-36 Case 5: LPCI Flow for 0.15 Ft^2 Pump Discharge Break

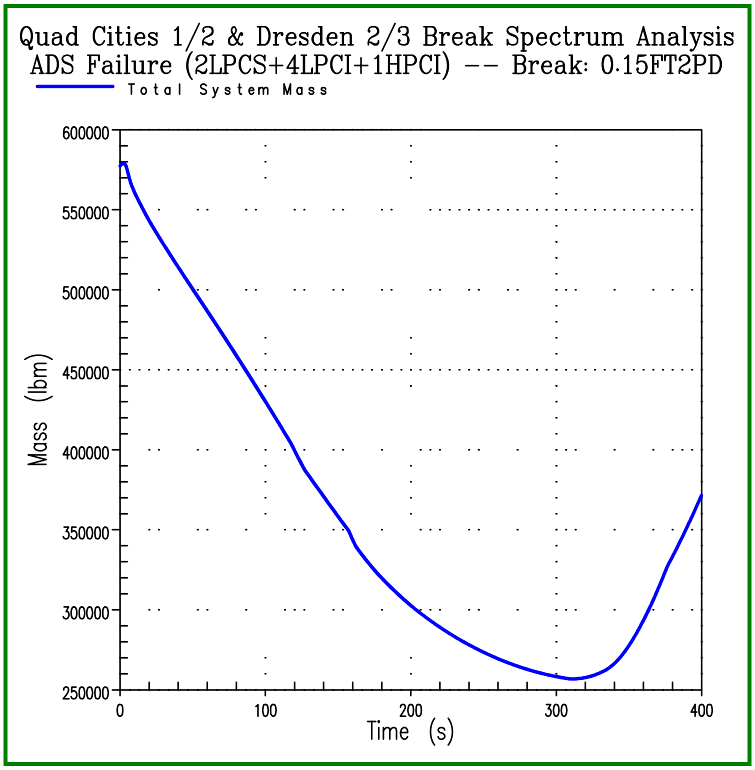

Figure 4-37 Case 5: System Mass for 0.15 Ft^2 Pump Discharge Break

4.3 Steam Line Breaks

Steam line breaks were assumed to occur between the reactor vessel and the flow restrictor upstream of the MSIV. As HPCI is steam driven, it was not credited for any steam line breaks upstream of the MSIVs.

Large double-ended ruptures of the steam line in this location result in reverse flow from the other steam lines until the MSIVs close. Since the LOCA model does not have a detailed steam line model, this effect was modeled conservatively by increasing the break area by a factor of 1.6 until MSIV closure occurs. The factor was chosen after reviewing the pressure losses encountered when the steam flows from the vessel through the three intact steam lines to the header and then from the header back to the break through the single line. The break flow that would result from a detailed model would be less than the flow that would occur with an area 1.6 times the single line area. After MSIV closure (six seconds), a single line area is used.

Unlike the breaks in the recirculation line, large steam line breaks do not exhibit a pressure increase when the turbine stop valves close. As a result, there is no initial power increase caused by void collapse. The depressurization causes voiding to occur in the downcomer and the two-phase mixture to swell. The depressurization rate decreases when the two-phase mixture reaches the elevation of the break.

Note that case numbers identified below are in accordance with Table 4-1.

4.3.1 Case 1: LPCI Injection Valve Failure

This case investigated the single failure of the LPCI injection valve. In this situation, no coolant injection from the LPCI system occurs. However, two LPCS pumps and five ADS valves are assumed to be operable. The HPCI pump was assumed to fail due to the location of the break. For large breaks, the system depressurizes very rapidly and ADS is not actuated.

Two large breaks in the steam line were evaluated – a full area break and a half area break. Figure 4-38 through Figure 4-40 show some of the graphical results for the full area break. As shown in Figure 4-39, the LPCS pumps started injecting at approximately 68 seconds. Figure 4-40 shows the system mass beginning to recover shortly afterward. The core remained covered with a two-phase mixture throughout the event and there was no cladding heatup. The change in the rate of mass recovery that occurs after 150 seconds is due to an increase in the amount of liquid out the break as the downcomer fills to with a two-phase mixture to the elevation of the steam lines.

Since neither break resulted in any core uncovery, it was concluded that this break location was not limiting for this single failure.

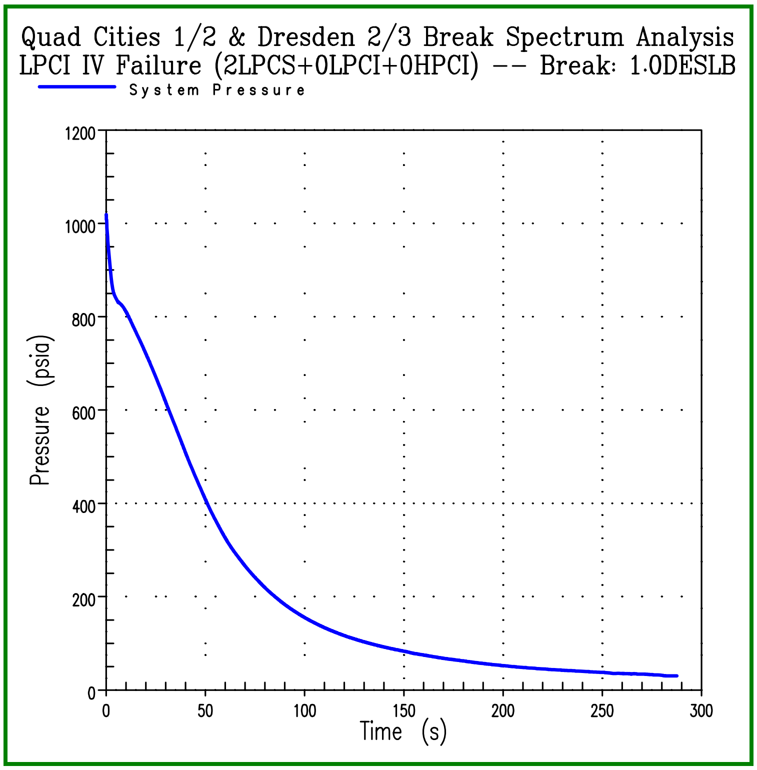

Figure 4-38 Case 1: Dome Pressure for 1.0 DEG Steam Line Break

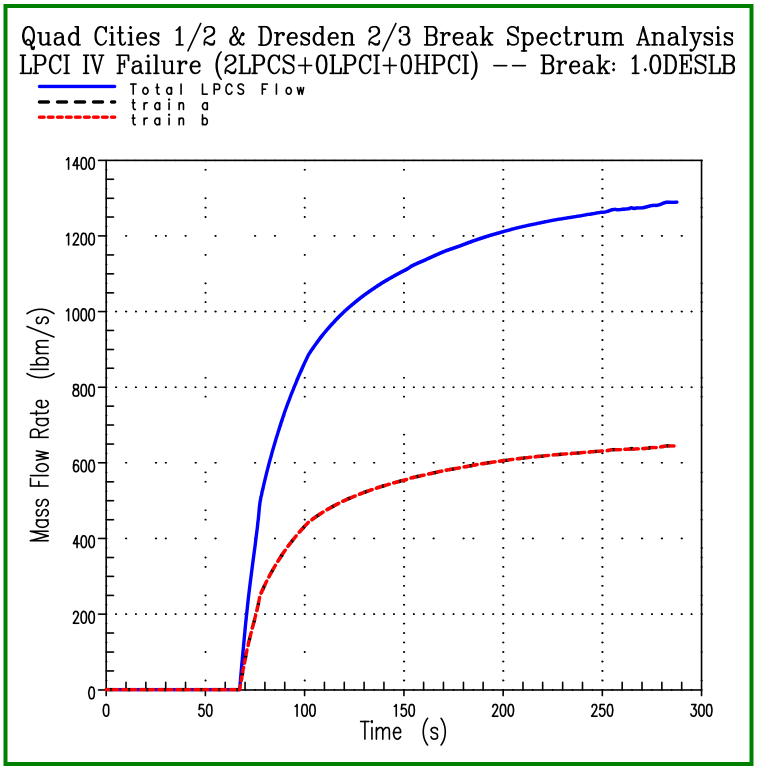

Figure 4-39 Case 1: LPCS Injection for 1.0 DEG Steam Line Break

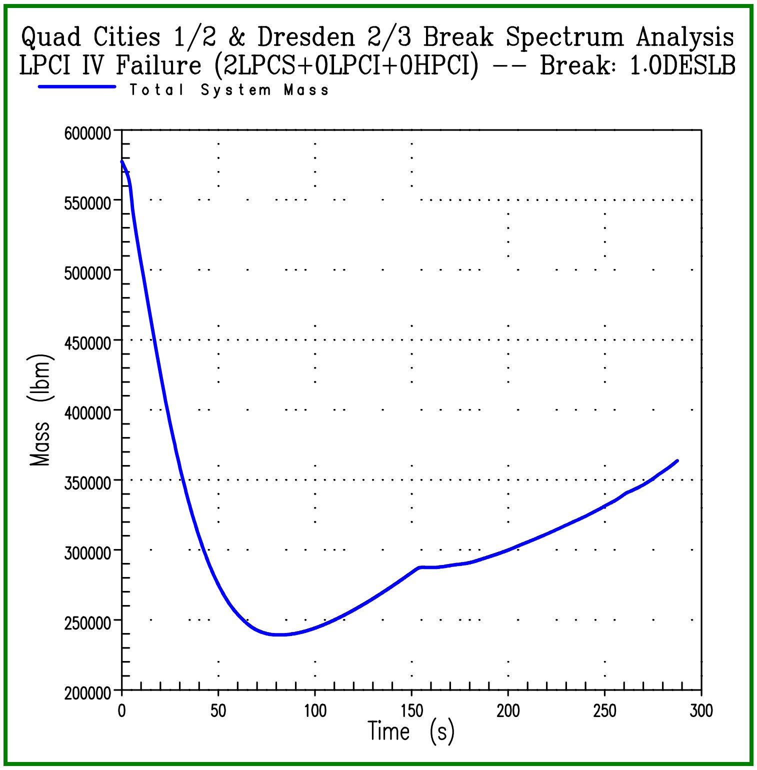

Figure 4-40 Case 1: Total System Mass for 1.0 DEG Steam Line Break

4.3.2 Case 2: EDG Failure

This single failure was found to be less limiting than the single failure of the LPCI injection valve for breaks in the recirculation line. Since the combination of one LPCS pump and two LPCI pumps provides more makeup than two LPCS pumps, it is judged that Case 1 described above would bound this failure. Since Case 1 resulted in no core uncovery, this break location is considered not limiting with respect to ECCS performance.

4.3.3 Case 3: HPCI Failure

Since the HPCI pump is steam driven, HPCI was assumed to not function for breaks in the steam line.

4.3.4 Case 4: Loop Select Logic Failure

Consideration of loop select failure is an important consideration for breaks in the recirculation line. However, the failure of the logic for steam line breaks has no effect on the breaks in the steam line.

4.3.5 Case 5: ADS Failure

As described in Section 4.2.5, this failure did not result in significant cladding heatup for the LPCI injection valve failure. Since the LPCI injection valve failure case is considered to be more challenging from a core cooling perspective, it was judged that this failure would not be limiting for breaks in the steam line.

4.4 LPCS Line Breaks

A complete severance of one of the LPCS lines was analyzed. The flow area of the spray line is 0.51 ft^2. This break results in a blowdown of the reactor vessel through one of the core spray spargers and prevents the delivery of coolant from one of the LPCS pumps. The single failures that result in the least coolant injection for breaks in this location is the failure of the LPCI injection valve and the failure of one of the EDGs to start.

4.4.1 Case 6: LPCI Injection Valve Failure

This case investigated the single failure of the LPCI injection valve. In this situation, no coolant injection from the LPCI system occurs. Due to the location of the break, one of the core spray pumps is prevented from delivering water. However, one LPCS pump, one HPCI pump and five ADS valves are assumed to be operable.

Figure 4-41 through Figure 4-44 present the graphical results for this case. As shown in Figure 4-41, the dome pressure increases rapidly at the beginning of the event due to the closure of the turbine stop valves. After reactor trip, the pressure decreases rapidly as the system blows down through the spray line. HPCI begins to inject at approximately 45 seconds. HPCI injection continues until the pressure decreases below low pressure cutoff at approximately 225 seconds. ADS actuates shortly after 120 seconds, which increases the system depressurization rate. The pressure permissive for the LPCS pumps is met at approximately 170 seconds and the LPCS pumps begin to provide makeup. Figure 4-44 shows the effect of ECCS injection on system mass. The system mass begins to recover at approximately 250 seconds when the injected flow exceeds the break flow. There is no cladding heatup during the event as the core remains covered with a two-phase mixture throughout the event.

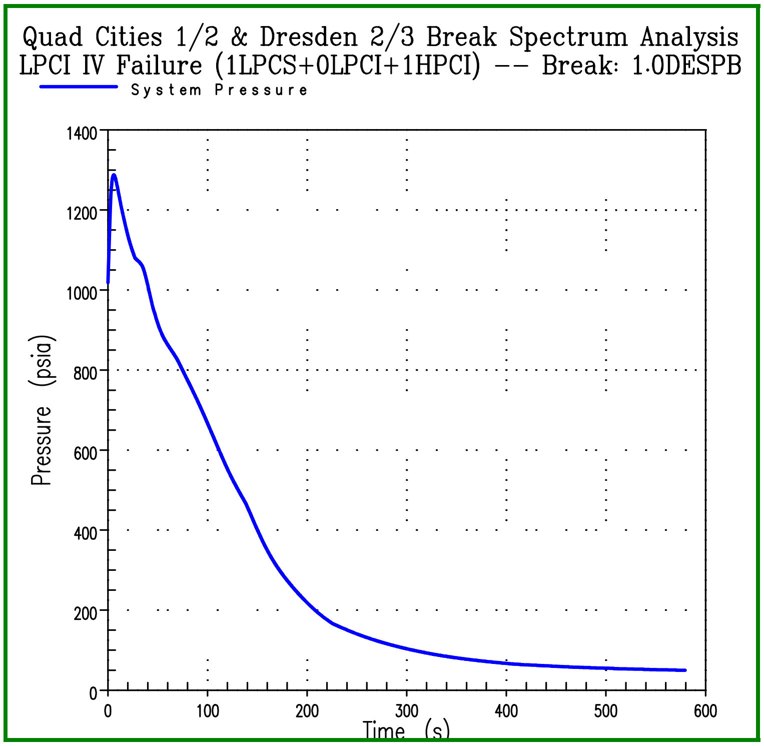

Figure 4-41 Case 6: Dome Pressure for Spray Line Break

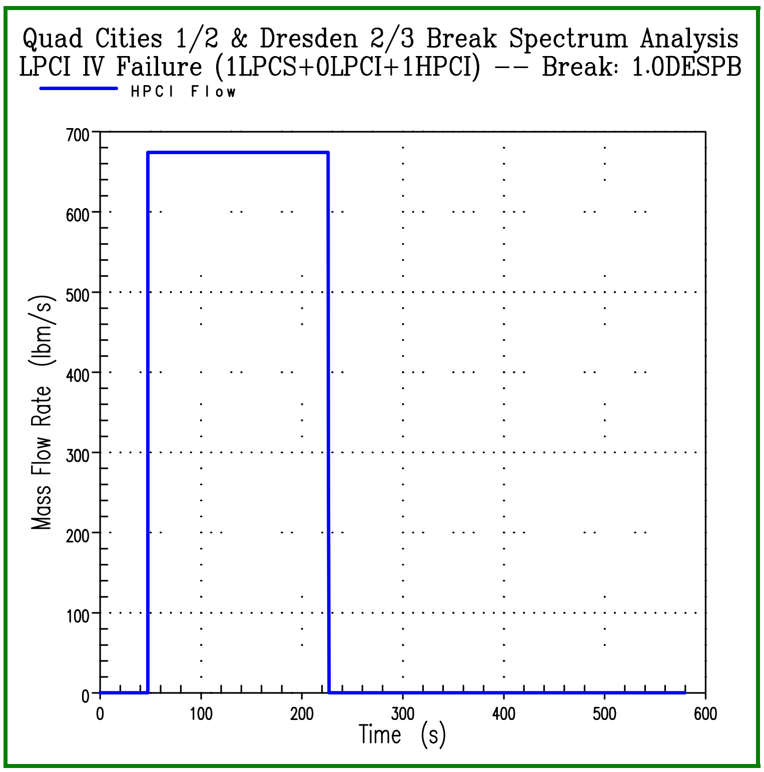

Figure 4-42 Case 6: HPCI Injection for Spray Line Break

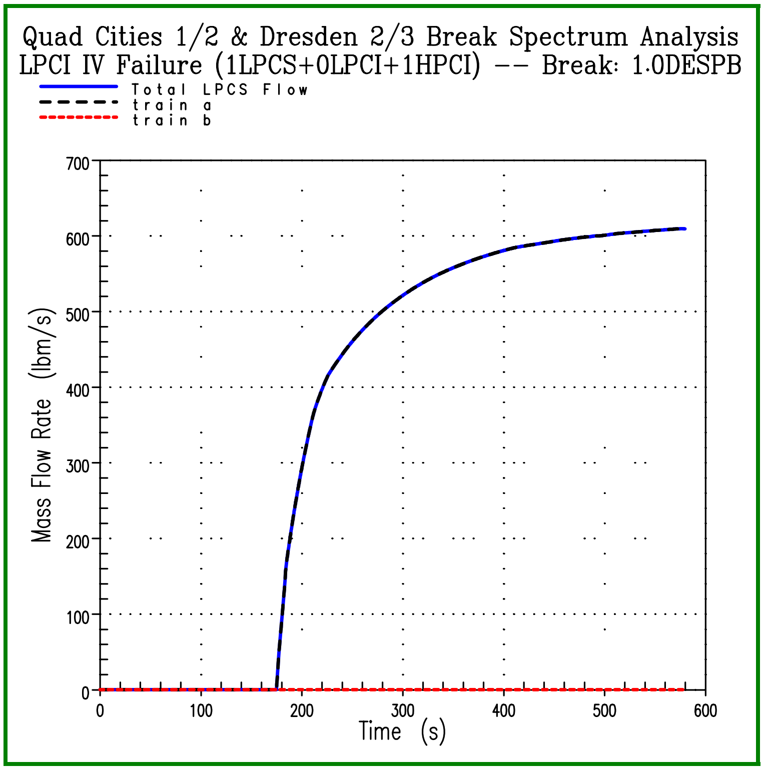

Figure 4-43 Case 6: LPCS Injection for Spray Line Break

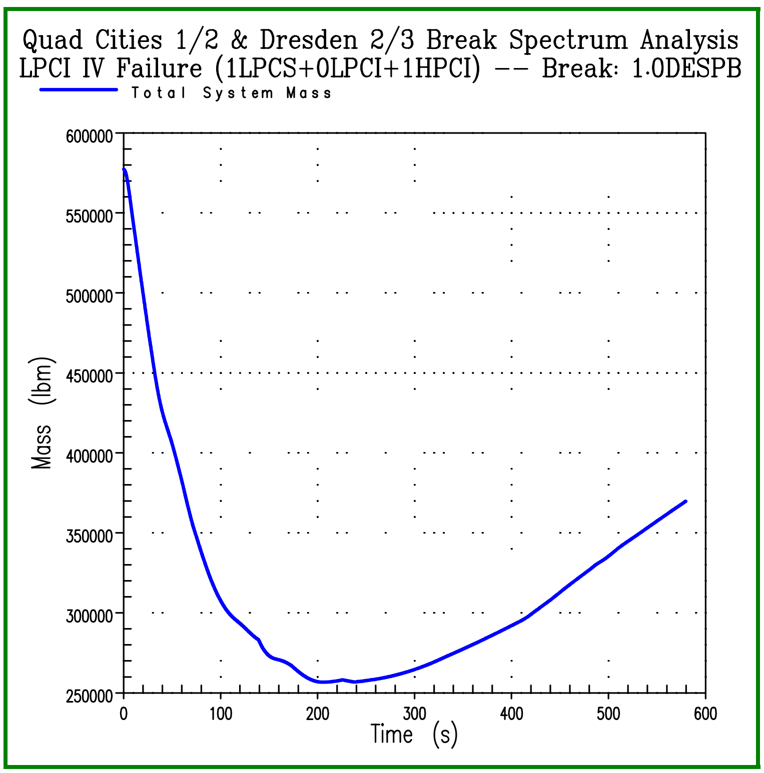

Figure 4-44 Case 6: Total Mass for Spray Line Break

4.4.2 Case 7: EDG Failure

This single failure and break location was found to be less limiting than the single failure of the LPCI injection valve for breaks in the recirculation line. With the failure of the EDG and a break in one of the spray lines, it was assumed that the failed EDG supplied power to the intact LPCS train. Therefore, only two LPCI pumps, the HPCI pump and the five ADS valves are assumed to be operable. The analysis of this case confirmed that there was no core uncovery. HPCI actuated at approximately 48 seconds, followed by ADS at approximately 140 seconds and LPCI at approximately 190 seconds. The total system mass began to recover when LPCI actuated and there was no core uncovery.

4.4.3 Case 8: HPCI Failure

This single failure was one of the limiting failures for small breaks in the recirculation line (see Section 4.2.3). Since the spray line break is a small break, this single failure was analyzed for this break location.

The analysis showed that, although the break was large enough to result in system depressurization, ADS actuated at approximately 130 seconds. The operable LPCS pump started to inject at approximately 200 seconds, followed by the four LPCI pumps. The injection flow exceeded the break flow at approximately 225 seconds after which the system mass recovered.

Due to the location of the break and the actuation of the ECCS, there was no core uncovery prior to the inventory being restored.

4.4.4 Case 9: Loop Select Logic Failure

This case is important for breaks in the recirculation line. The failure of the logic for breaks in the core spray line would have no effect on the transient.

4.4.5 Case 10: ADS Failure

As described in 4.2.5, this failure did not result in significant cladding heatup for the LPCI injection valve failure. Since the LPCI injection valve failure case is considered to be more challenging from a core cooling perspective, it was judged that this failure would not be limiting for a break of the LPCS injection line.

4.5 Feedwater Line Breaks

There are two pairs of feedwater nozzles attached to the reactor vessel. Each pair is connected to a common header, which is fed by a feedwater train. A double-ended break of one of the feed lines is assumed to occur coincident with a loss of offsite power. Although feedwater would continue to supply one side of the break as the feedwater pumps coast down, it was conservatively ignored as this increases the break flow from the reactor vessel. Further, since HPCI is supplied to one of the feedwater lines, it is assumed that the break occurs in that feedwater line. To account for break flow from both feedwater lines, the break was simulated as a single break having twice the area of one of the feedwater lines. Hence the total break area assumed in the analysis was 1.31 ft^2.

Several ECCS failure combinations are possible. However, the failure of the LPCI injection valve to open is limiting with respect to the amount of coolant that be supplied and is discussed below.

4.5.1 Case 11: LPCI Injection Valve Failure

This case investigated the single failure of the LPCI injection valve. In this situation, no coolant injection from the LPCI system occurs. Due to the location of the break, the HPCI pump is prevented from delivering water. However, two LPCS pump and five ADS valves are assumed to be operable.

Figure 4-45 through Figure 4-47 present the graphical results for this case. As shown in Figure 4-45, there is an initial increase in pressure due to the closure of the turbine stop valves. Dome pressure decreases rapidly after reactor trip. The pressure permissive setpoint is satisfied after approximately 90 seconds and the LPCS pumps begin to inject. Injection flow exceeds break flow after approximately 100 seconds and the system mass begins to recover. There was no core uncovery prior to the system mass recovery.

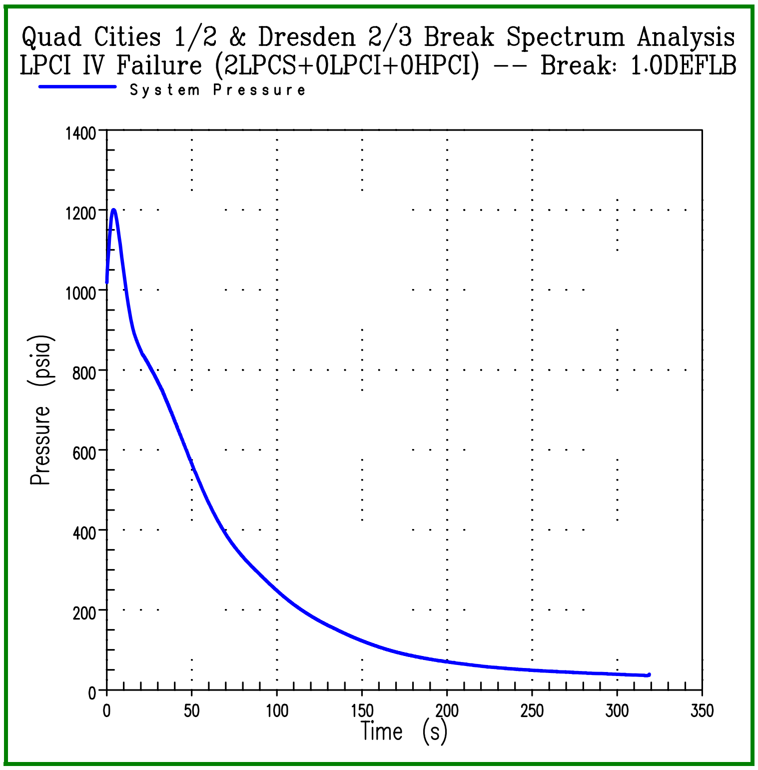

Figure 4-45 Case 11: Dome Pressure for Large Feedwater Line Break

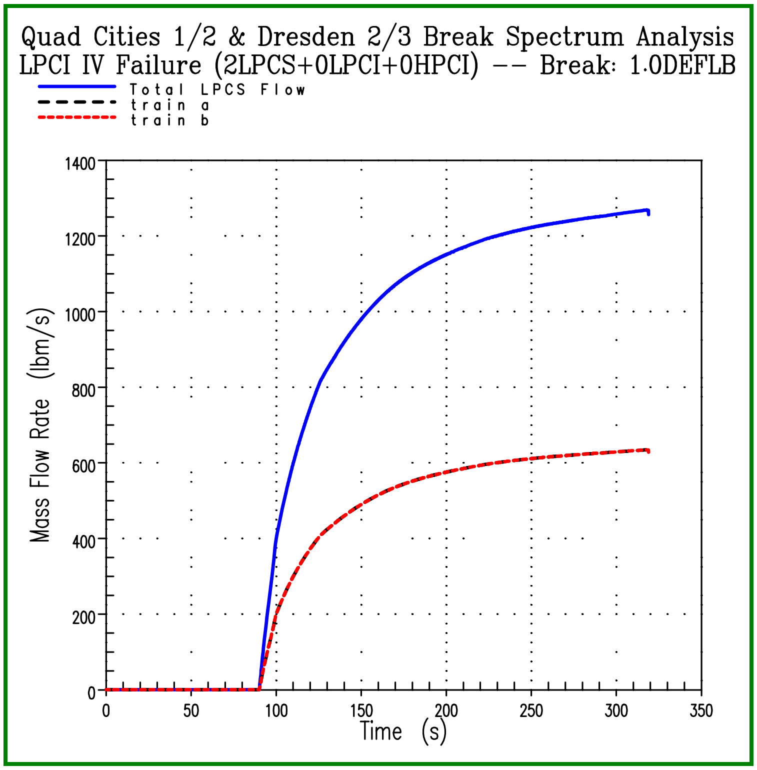

Figure 4-46 Case 11: LPCS Flow for Large Feedwater Line Break

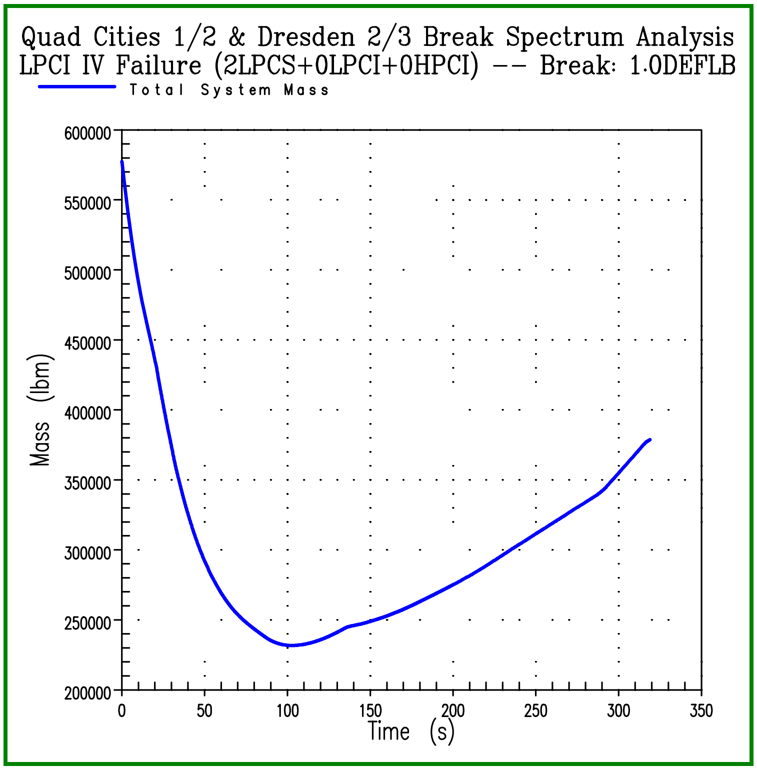

Figure 4-47 Case 11: Total System Mass for Large Feedwater Line Break

4.5.2 Case 12: EDG Failure

This single failure was found to be less limiting than the single failure of the LPCI injection valve for breaks in the recirculation line. Since the combination of one LPCS pump and two LPCI pumps provides more makeup than two LPCS pumps, it is judged that LPCI injection valve failure case described above would bound this failure. Since it resulted in no core uncovery, this break location is considered not limiting with respect to ECCS performance.

4.5.3 Case 13: HPCI Failure

HPCI delivers coolant to one of the feedwater lines. Since the break is assumed to be in that line, HPCI will not be available for feedwater line breaks. Therefore, there was no need to evaluate this single failure for this break location.

4.5.4 Case 14: Loop Select Logic Failure

This case is important for breaks in the recirculation line. The failure of the logic for feedwater line breaks would have no effect on breaks in the feedwater line.

4.5.5 Case 15: ADS Failure

As described in 4.2.5, this failure did not result in significant cladding heatup for the LPCI injection valve failure. Since the LPCI injection valve failure case is considered to be more challenging from a core cooling perspective, it was judged that this failure would not be limiting for a break of the feedwater line.

5.0 Sensitivity Studies

The break spectrum analysis described in Section 4.0 was performed with a model that was initialized at the increased core flow (ICF) point of 108% of rated flow with a core comprised of 100% SVEA-96 Optima2 fuel. Sensitivity studies were performed after completion of the break spectrum to determine if a) a different core configuration or b) a different initial flow condition could result in a more adverse response for the limiting break single failure combination. These sensitivity studies are summarized in this section.

Section 5.1 presents a comparison of the results for a full core of SVEA-96 Optima2 fuel to a full core of GE14 fuel and a transition core comprised of approximately 30% SVEA-96 Optima2 fuel and 70% GE14 fuel.

Section 5.2 presents a comparison of an analysis initialized at low flow maximum extended load line limit analysis (MELLLA) point (95.3% of rated core flow) to the limiting case initialized at 108% flow.

The conclusions of these sensitivity studies are:

(1) There is very little difference in performance for the different core configurations studied. The SVEA-96 Optima2 equilibrium core is adequate for establishing core operating limits.

(2) The introduction of SVEA-96 Optima2 fuel does not adversely affect the response of the resident GE14 fuel compared to an equilibrium core comprised entirely of GE14 fuel.

(3) The analysis at the ICF point (108%) is slightly more conservative than at the low flow MELLLA point (95.3%).

5.1 Transition Core Study

This study is in compliance with condition two (2) placed on the acceptance of Reference 6. When a new fuel design is introduced in an operating reactor, there is concern that differences between the fuel designs (e.g., different core pressure drop) may affect the response of the either the new fuel or the resident fuel to a LOCA relative to the performance assuming an equilibrium core of either design. Transition core studies have been performed in References 2 and 3. Those studies concluded that the transient response of a core comprised of an entire core of the new design agrees very well with the response of the new design in a core containing a different fuel design.

Since the fuel designs considered in pervious studies differ from the current situation, an evaluation of the performance of each fuel design was performed using the GOBLIN code to determine if there is a limiting core configuration. In the current situation, SVEA-96 Optima2 fuel is being inserted into a core containing GE14 fuel. The objectives of this study are to determine 1) is the resident fuel impacted from a LOCA perspective by the insertion of the new fuel design, and 2) is the performance of the new fuel adversely affected by the presence of the resident fuel. [

]a,c

Figure 4-1 shows the GOBLIN nodalization that was used for Case (1). Figure 5-1 shows the core region of the GOBLIN nodalization that was used for Case (2). [

]a,c

[

]a,c

Considering the overall conservatism in the Appendix K evaluation model and as a result of this study, it is concluded that the configuration of the core does not have a significant impact on the ECCS effectiveness. Therefore, the full core SVEA-96 Optima2 model is adequate for evaluating the performance of the ECCS and establishing MAPLHGR limits.

Table 5-1 Target Thermal Hydraulic Parameter Values for Transition Core Study

Parameter	**Units**	**Optima2**	**GE14**
Total core flow rate	Mlb/hr	[	
Active core flow rate	Mlb/hr		
Interassembly bypass flow rates[1]	Mlb/hr		
[ ]a,c	Mlb/hr		
[ ]a,c	Mlb/hr		
[ ]a,c	Mlb/hr		
Assembly bypass flow rates	Mlb/hr		
[ ]a,c	Mlb/hr		
[ ]a,c	Mlb/hr		
Core pressure drop	psid		]a,c

Table 5-2 LOCA System Response for Various Core Configurations

Event	**Equilibrium Core**		**Mixed Core**	
	Optima2	**GE14**	**Optima2**	**GE14**
Jet Pump Flashing	[			
Lower Plenum Flashing				
Initiation of Spray Flow				
Average Channel Midplane Dryout				
Average Channel Midplane Turnaround				]a,c

[1] Numbers in parentheses refer to flow paths shown in Figure 4-1 and Figure 5-1 respectively.

a,c

Figure 5-1 GOBLIN Core Nodalization for GE14 Fuel Equilibrium Configuration

a,c

Figure 5-2 GOBLIN Core Nodalization for Transition Core

a,c

Figure 5-3 Transition Core Study: Comparison of Dome Pressure Response

a,c

Figure 5-4 Transition Core Study: Comparison of System Mass Response

a,c

Figure 5-5 Transition Core Study: Comparison of Midplane Void (GE14)

a,c

Figure 5-6 Transition Core Study: Comparison of Midplane Void (Optima2)

5.2 Initial Core Flow Rate Study

The Quad Cities and Dresden units are licensed to operate over a range of core flow rates while at rated power. Relative to the rated core flow (98 Mlb/hr), the core flow rate can be as high as 108% and as low as 95.3% of rated core flow. The break spectrum results described in Section 4.0 were performed at the high flow ICF point (108%). The effect of lower initial core flow rate was evaluated for the limiting break / single failure combination (1.0 double-ended guillotine break in the pump suction line with failure of the LPCI injection valve to open).

As shown in Table 5-3, this evaluation indicated that the effect of initial flow rate is very small, although the 108% flow case slightly bounds the case performed at the low flow MELLLA point (95.3%).

The 108% flow condition will be used to establish operating limits at rated power for two loop operation. Figure 5-7 through Figure 5-10 show graphical comparisons of the two cases. The figures confirm that the initial flow rate has a very small impact on the transient.

Table 5-3 System and Hot Assembly Responses for Different Initial Core Flow Rates

Event	95.3% Flow	108% Flow
Jet Pump Flashing	[	
Lower Plenum Flashing		
Initiation of Spray Flow		
Hot Assembly Midplane Uncovery		
Hot Assembly Midplane Cladding Turnaround		][a,c]

a,c

Figure 5-7 Initial Core Flow Study: Comparison of Dome Pressure

a,c

Figure 5-8 Initial Core Flow Study: Comparison of Total System Mass

a,c

Figure 5-9 Initial Core Flow Study: Comparison of Midplane Void Fraction

a,c

Figure 5-10 Initial Core Flow Study: Comparison of Midplane Cladding Temperatures[2]

[2] The cladding temperatures shown are those calculated by DRAGON in the hot assembly analysis.

6.0 Limiting Large Break

As shown in Figure 4-3, the limiting break is a full double-ended guillotine break in the pump suction line with the single failure of the LPCI injection valve. This case is used in the calculation of the peak cladding temperature, local and core-wide oxidation, and in the determination of the MAPLHGR core operating limit. As in the break spectrum analysis, the hot assembly power is established at a conservative CPR operating limit of 1.41.

6.1 Sequence of Events

The sequence of events for the limiting large break is summarized in Table 7-1.

Table 6-1 Sequence of Events for Limiting Large Break

Event	Time (seconds)
Break / loss of offsite power occurs	0.0
Turbine stop valve closes on loss of offsite power	0.1
High drywell pressure occurs	0.2
Reactor scram signal on high drywell pressure	1.2
Top of jet pumps uncover	3.2
Suction line uncovers	4.8
Reactor low-low water level (L2) reached	5.4
Lower plenum flashes	6.0
Diesel generators at rated speed and bus powered	17.3
Peak plane uncovers	21.4
LPCS pressure permissive reached	22.6
LPCS pumps start	29.3
LPCS injection occurs	32.2
LPCS pumps at full speed	34.3
LPCS pumps deliver rated flow	58.2
LPCS injection valves full open	75.6
Peak clad temperature occurs	195.3

6.2 Peak Cladding Temperature

The limiting break with the most limiting single failure assumption is analyzed at different exposure steps covering the entire life of the fuel in the core. Based on these calculations, the limiting nodal exposure is identified as 12500 MWD/MTU. At this limiting exposure, the ECCS performance is analyzed for the design basis LOCA. The limiting PCT that bounds both assembly types is 2150°F, which is lower than the regulatory limit set by 10CFR50.46.

Figure 6-1 through Figure 6-6 show some of the graphical results for the limiting large break. As shown, the dome pressure increases rapidly after the closure of the turbine stop valves. After reactor trip, the dome pressure decreases rapidly below the pressure permissive well before ADS can actuate. The peak plane of the hot assembly uncovers at 21.4 seconds at approximately the same time. The LPCS pumps begin injecting at 32.2 seconds and the flow rate delivered to the spray spargers reaches rated flow at 58.2 seconds. The total system mass begins to recover shortly after LPCS actuation and two-phase conditions are restored to the peak plane at 194.4 seconds.

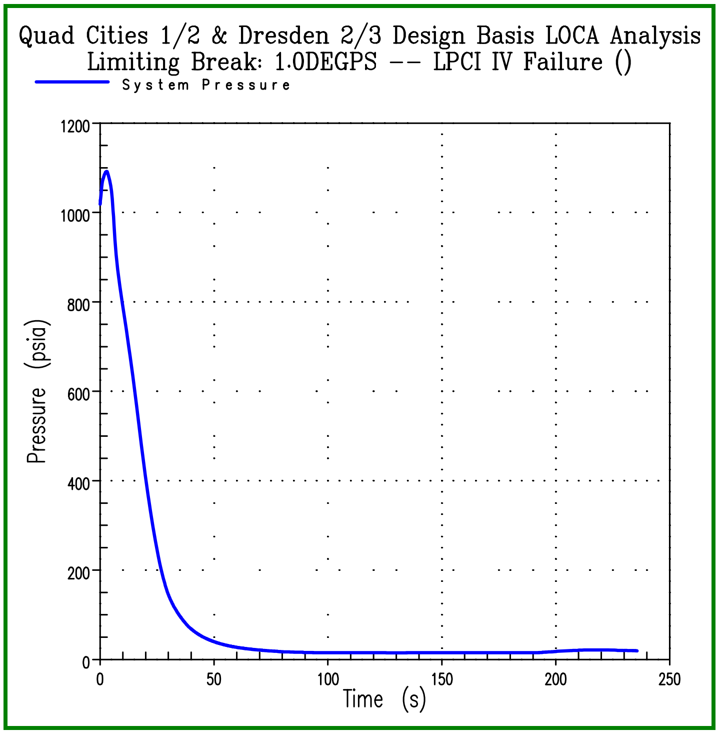

Figure 6-1 Dome Pressure Response for Limiting Large Break

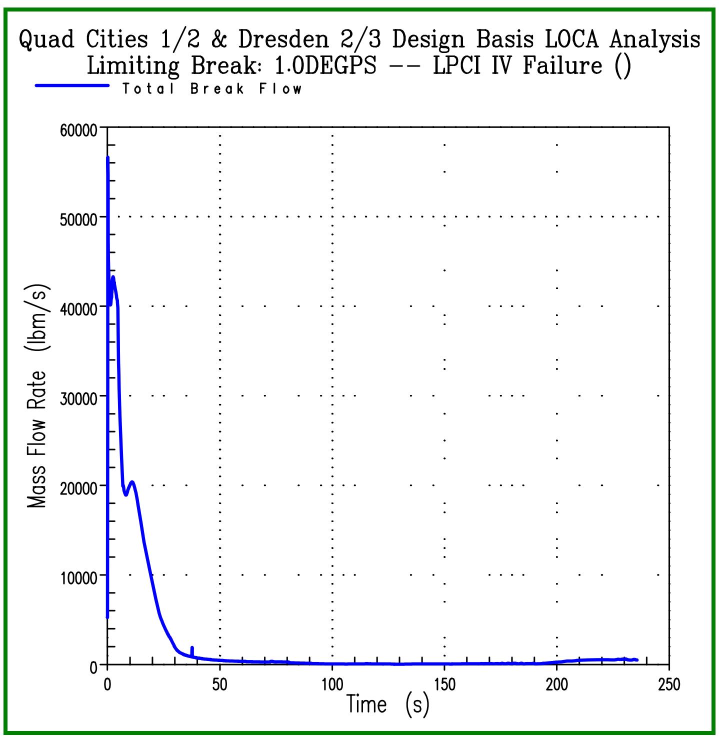

Figure 6-2 Break Flow Rate for Limiting Large Break

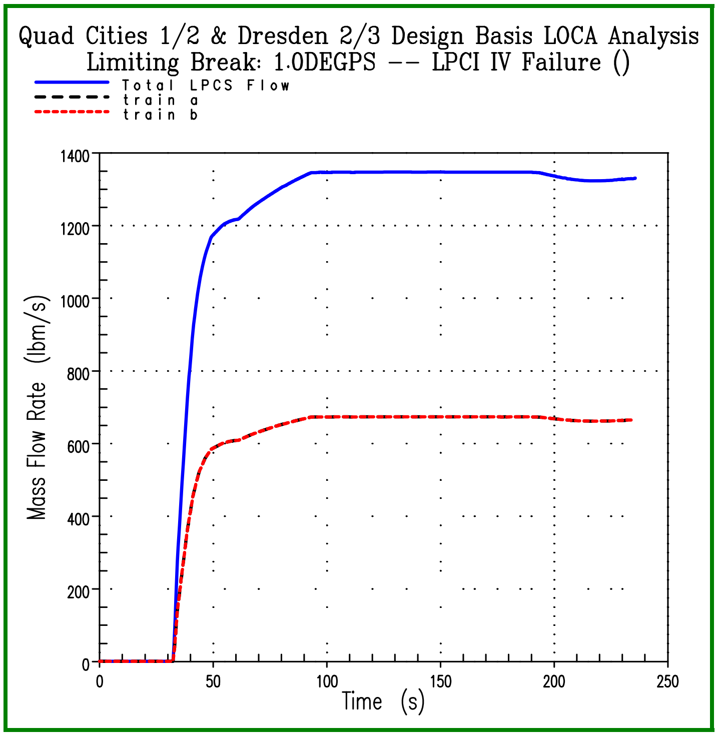

Figure 6-3 LPCS Injection for Limiting Large Break

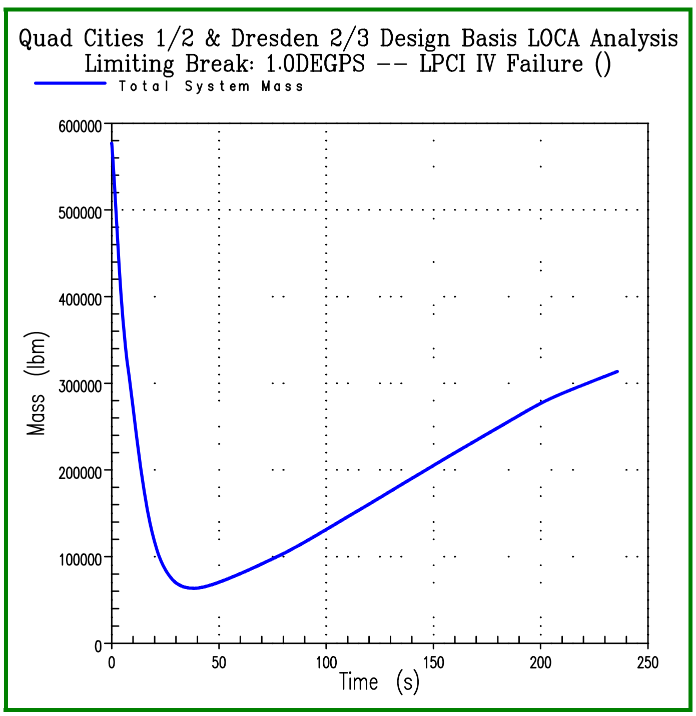

Figure 6-4 Total System Mass for Limiting Large Break

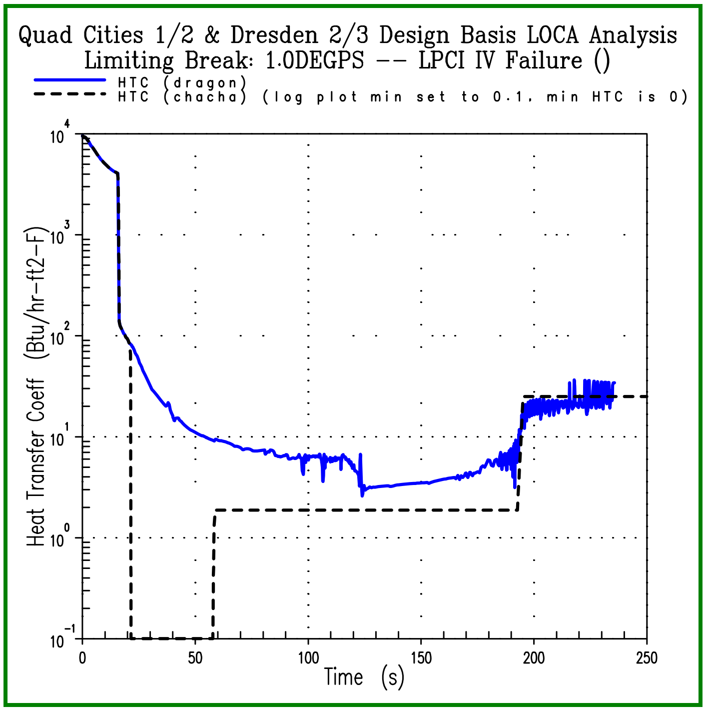

Figure 6-5 Heat Transfer Coefficient at Peak Plane for Limiting Large Break

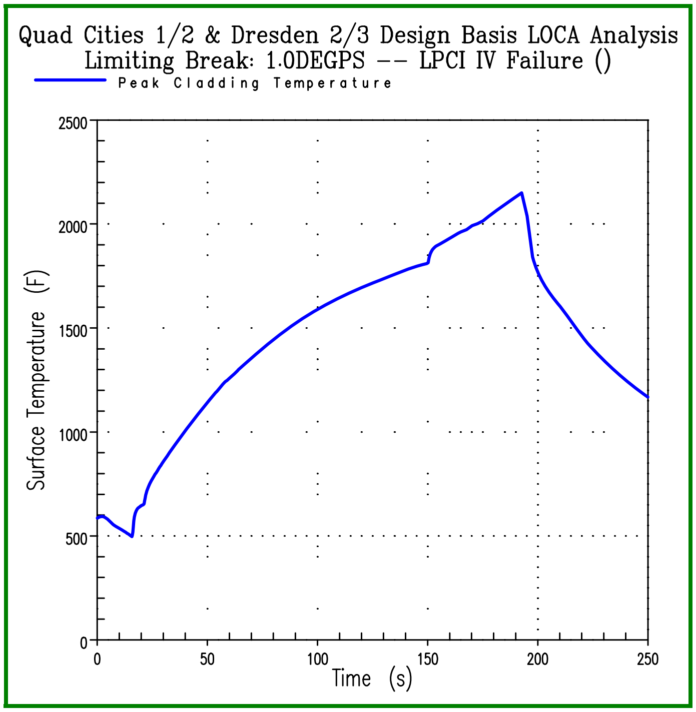

Figure 6-6 Peak Cladding Temperature for Limiting Large Break

6.3 Maximum Cladding Oxidation

The maximum cladding oxidation values that were obtained are shown in Table 6-2 and are substantially less than the 17% 10CFR50.46 acceptance limit. The analysis results include the initial (pre-transient) oxidation in the totals.

The maximum cladding oxidation is most limiting at 10000 MWD/MTU nodal exposure. The maximum cladding oxidation is calculated to be less than 7.1% for all assembly types at any nodal exposure for SVEA-96 Optima2 fuel.

Table 6-2 Peak Cladding Temperature and Local Oxidation vs. Nodal Exposure

Nodal Exposure	Peak Cladding Temperature	Maximum Local Oxidation
(MWD/MTU)	(°F)	(%)
0	2095	6.1%
2500	2054	5.3%
5000	2035	5.2%
7500	2059	5.8%
10000	2120	7.1%
12500	2150	6.8%
15000	2134	5.9%
17500	2080	5.0%
20000	2061	4.8%
22000	2061	4.9%
24000	2078	5.2%
30000	2076	5.5%
36000	2074	5.7%
42000	2059	5.8%
50000	2032	5.8%
58000	2035	6.3%
62000	1933	5.5%
70000	1775	5.0%

6.4 Maximum Hydrogen Generation

It is required to demonstrate that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume were to react. For most fuel designs, the peak cladding temperature and local maximum oxidation acceptance limits, restrict the potential total core hydrogen generation significantly below the 0.01 limit.

The methodology used to calculate the total amount of hydrogen generated during a postulated LOCA is described Section 8.2.3.4 of Reference 11. The limiting case was evaluated using this methodology.

A maximum hydrogen generation calculation was performed for the limiting break at a limiting exposure with the highest transient oxidation. A bounding calculation was performed which assumed a disproportionate number of assemblies in the core at a higher than average relative radial bundle peaking. This assumption yields a conservative total core hydrogen generation rate, which is verified by comparison of power distributions on a cycle-to-cycle basis. The total cladding volume over the active fuel length throughout the cycle that reacted was calculated to be less than 0.57%. This shows substantial margin to the 10CFR50.46 acceptance criterion of one percent (1.0%).

7.0 Limiting Small Break

As shown in Figure 4-3, the limiting small break is a 0.15 ft^2 break in the pump discharge line with the single failure of the HPCI system. The graphical results for this break are presented in Section 4.2.3. Two LPCS pumps, four LPCI pumps and five ADS valves are assumed to be operable. Due to the size of the break, the loop select logic is not assumed to identify the intact loop. As a result, the break is placed downstream of the LPCI injection point.

For breaks larger than this, the loop select logic is assumed to function as designed and all of the water injected by the LPCI system would be injected into the reactor vessel. For breaks smaller than this, less of the water injected by LPCI would be lost out the break.

This study was performed assuming that the initial core flow rate was 108% of rated core flow. As concluded in Section 5.2, the initial core flow rate had no impact on the large break analysis. Since small breaks are impacted even less than large breaks by the initial core flow rate, it was judged that performing the analysis at an initial core flow of 108% is justified.

7.1 Sequence of Events

Table 7-1 shows the sequence of events for the limiting small break. In this case the reactor trip was initiated by high dome pressure following the closure of the turbine stop valves. ECCS actuation was initiated by high drywell pressure well before the emergency diesel generators started. As the break is quite small, the system pressure was at first controlled by the cycling of the safety valves. When the low-low level setpoint was reached, the ADS timer was actuated and 120 seconds later, the ADS valves opened and the system began to depressurize. The depressurization caused flashing in the lower plenum and an insurgence of water into the core. As the reactor vessel continued to lose inventory via the break and the ADS valves without any makeup, the void content in the core increased. Eventually there was core uncovery and the fuel started to heat up. When the system depressurized below the pressure permissive setpoint, the two LPCS pumps began to inject and the system mass stabilized and then began to increase. Shortly afterward, the four LPCI pumps began to inject water into the pump discharge leg that contained the break. This increased the break flow rate, but not all of the injected LPCI water was lost to the break. The inventory in the reactor vessel continued to increase and two-phase cooling conditions were restored.

Table 7-1 Sequence of Events for Limiting Small Break

Event	Time (seconds)
Break / loss of offsite power occurs	0.0
Turbine stop valve closes on lose of offsite power	0.1
High dome pressure occurs	0.6
Reactor scram on high dome pressure	1.6
High drywell pressure occurs	4.3
Diesel generators at rated speed and bus powered	17.3
LPCI pumps start	24.3
Swing bus transfer complete	26.0
LPCS pumps start	29.3
LPCI pumps at full speed	31.3
LPCS pumps at full speed	34.3

Event	Time (seconds)
Reactor low-low water level (L2) reached	40
Recirculation discharge valve closed	74
Top of jet pumps uncover	125
ADS valves open	160
Lower plenum flashes	161
Peak plane uncovers	238
LPCS/LPCI pressure permissive reached	324
LPCS injection occurs	329
LPCI injection valves full open	354
LPCI injection occurs	359
Peak clad temperature occurs	365
LPCS injection valves full open	377
LPCS pumps deliver rated flow	> 377

8.0 Long-Term Cooling

It is required to demonstrate after any calculated successful operation of the ECCS, that the calculated core temperature can be maintained at an acceptably low value and decay heat can be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The long term cooling requirements are the core reflooded to the top of active fuel, or, with the core reflooded to 2/3rd core height, at least one core spray pump delivering spray flow to the top of the core.

8.1 Discussion

Following quenching of the fuel cladding, it is necessary to maintain the cladding temperature sufficiently low to assure that the cladding continues to maintain its function. The criterion of maintaining the core coolable for an extended period of time following a postulated LOCA is achieved by ensuring a continuous source of ECCS water. Since there are some break locations that will preclude completely refilling the reactor vessel (e.g., recirculation loop pump suction breaks), the upper part of the core for these breaks must be cooled either by two-phase media and steam generated by the boiling of coolant in the lower part of the core or by spray flow from above.

In the case of small pipe breaks, the reactor pressure vessel and the complete reactor core is flooded with water and the heat is transferred to the coolant in a safe way. In case of a large recirculation loop pump suction break, the core region is flooded to the height of the jet pumps. ECCS water is added by the core spray to the region inside the main shroud. This water flows down through the core, into the lower plenum and out into the downcomer through the top of the jet pumps. In the downcomer a shallow pool is formed and the water leaves the reactor through the break at the same rate as it is injected. In addition to the jet pumps there are other leakage paths from the region inside the main shroud to the downcomer and recirculation lines.

If the reactor core will fill up to the elevation of the jet pump inlets, the weight of the fluid in a fuel bundle is that of water in the lower 2/3rd of the bundle. With power added to the fluid in the bundle, a two-phase mixture will be formed that will cool the fuel to a higher elevation. A bundle power in order of 10 kW is enough for this mechanism (at atmospheric pressure) to completely cover the bundle and keep it cooled without the aid of any flow from above. However, at lower bundle powers, which will occur later in time, especially in the peripheral positions of the core, the upper part of the bundle will be uncovered. To keep these low power bundles quenched it is necessary to have spray flow from above.

8.2 Spray Flow Needed for Low Power Bundles

The amount of spray flow needed to keep the fuel rods quenched can be estimated conservatively from spray cooling tests that have been performed for the SVEA watercross fuel bundle design. These tests are described in part in Reference 3. These tests have shown that a spray flow of approximately [ ]^{a,c} per bundle is more than sufficient to provide adequate spray cooling, even when the spray distribution to the individual sub-bundles is reduced to a factor of approximately [ ]^{a,c} relative to the uniform distribution case. The tests also show that a fuel rod is quenched from above by a water film that is formed on the rod. The quenching occurs at a high power generation rate that far exceeds the power (also in high power bundles) that shall be cooled during the long-term cooling period.

8.3 Spray Flow Available

It is assumed that only the core spray system is injecting water to the reactor. During the long term cooling period of a large break LOCA, there is no pressure difference between the reactor and the vapor space above the wetwell pool. In these conditions, the total flow to the reactor vessel is 5650 gpm. However, due to leakage at the penetration of the reactor vessel, 250 gpm is spilled into the downcomer. The rest (5400 gpm) is supplied to the region inside the main shroud through the spray nozzles and through leaks inside the main shroud. The leak flow inside the shroud is 403.5 gpm. The average flow per bundle, excluding the leak inside the shroud, is (5400-403.5)/724=6.9 gpm. Spray distribution test data for the plant indicate a spray distribution factor of 0.6. Therefore a minimum of 4.1 gpm is available to each fuel assembly, which is considerably higher than the required flow rate of [ ][a,c] required to maintain coolability, discussed above. Even if the total spray flow delivered to the reactor vessel were reduced to 4500 gpm, which is the rated condition that is the basis for the 0.6 spray distribution factor, there will be adequate cooling water supplied to each fuel bundle.

The SVEA watercross fuel bundle design effectively divides a fuel assembly into four sub-assemblies which all must have a supply of spray water. The ring sparger spray system will produce a smooth variation with position of the spray flow density at the core grid. Therefore there will be only small variations in the amount of spray flow to each sub-assembly. The flow to the sub-assembly with lowest flow will clearly be higher than [ ][a,c] the average of the sub-assemblies and adequate cooling is thus assured also for sub-assemblies.

9.0 References

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5. "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application," WCAP-15682-P-A, April 2003.
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9. "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENPD-287," WCAP-15942-P, October 2004.
10. "LOCA Input Parameters for Quad Cities 1 & 2 and Dresden 2 & 3," NF-BEX-05-2, Rev. 2, February 2006.
11. "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A, July 1996.