QUANTITATIVE ASSESSMENT OF THE IMPACT OF THE PREVIOUS CYCLE'S CORE EXPOSURE ON THE TRANSIENT RESPONSE OF BOILING WATER REACTOR SYSTEMS

By

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A THESIS PRESENTED TO THE GRADUATE SCHOOL
OF THE UNIVERSITY OF FLORIDA IN PARTIAL FULFILLMENT
OF THE REQUIREMENTS FOR THE DEGREE OF
MASTER OF SCIENCE

UNIVERSITY OF FLORIDA

2008

To my family

ACKNOWLEDGMENTS

I would like to thank everyone who assisted in the research and writing of this thesis. At GE Hitachi, the assistance of Randy Jacobs and Pauline Stier was invaluable. At GE I would also like to thank Jennifer Bowie, Brian Triplett, Cindy Fung, and Keith Bentley for their help in navigating through GE and the University of Florida graduate school. At the University of Florida, I would like to thank my advisor, Dr. Samim Anghaie, as well as my professors. I'd also like to thank Mireille Rowe for being my homework partner throughout our undergraduate and graduate studies. Finally, I would like to thank my family: my parents, Ryan, and Emma, and especially my fiancé, Tyler Schweitzer.

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LIST OF ABBREVIATIONS

2SRVOS	Two safety relief valves out of service
6SVOS	Six safety valves out of service
ASME	American Society of Mechanical Engineers
BWR	Boiling water reactor
BWR/N	Boiling water reactor generation N
CLTP	Current licensed thermal power
CPR	Critical power ratio
Cycle N	Current cycle
ΔCPR	Delta critical power ratio
EFPD	Effective full power days
EOC N-1	End of previous cycle
EOC	End of cycle
FWCF	Feedwater control failure
GE	General Electric
GNF	Global Nuclear Fuel
HBB	Hard bottom burn
ICF	Increased core flow
ICPR	Initial critical power ratio
LCF	Low core flow
LHGR	Linear heat generation rate
LRNBP	Load rejection, no bypass
MCPR	Minimum critical power
MELLLA	Maximum extended load line limit analysis
MOC	Middle of cycle

MOP	Mechanical overpower
MSIV	Main steam isolation valve
MSIVF	Main steam isolation valve failure
MWd/ST	Megawatt-day per short ton
NBP	No turbine bypass
NFW	Normal feedwater temperature
NRC	Nuclear Regulatory Commission
OLMCPR	Operating limit critical power ratio
PEOC	Projected end of cycle
PRFDS	Pressure regulator failure, downscale
Psig	Pressure per square inch, gauge
RFW	Reduced feedwater temperature
SLMCPR	Safety limit critical power ratio
TCV	Turbine control valve
TOP	Thermal overpower
TSV	Turbine safety valve
TTNBP	Turbine trip, no bypass

Abstract of Thesis Presented to the Graduate School
of the University of Florida in Partial Fulfillment of the
Requirements for the Degree of Master of Science

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December 2008

Chair: Samim Anghaie
Major: Nuclear Engineering Sciences

We conducted a more intensive study of the effect of previous cycle exposure than is performed in a normal reload licensing analysis. For this research, analyses were performed to determine the validity of the Cycle N licensing limits for MCPR (minimum critical power ratio), LHGR (linear heat generation rate), and peak vessel pressure based on nominal EOC N-1 shutdown compared to the results obtained from early or late EOC N-1 shutdown. Early/late conditions as large as 1000 MWd/ST (as much as plus/minus 30 Effective Full Power Days (EFPD)), as well as intermediate early/late shutdown conditions of plus/minus 500 MWd/ST were examined. The analyses were performed in the same manner as the nominal exposure EOC N-1 reload licensing analysis. Design variations were minimized except for the EOC N-1 exposure effects in order to establish a valid metric for the dependence of results on previous cycle exposure variation. It was found that the plus/minus 500 MWd/ST cases did not challenge the nominal exposure reload licensing limits. The plus/minus 1000 MWd/ST cases were closer to the nominal limits and challenged them in some domains.

CHAPTER 1
INTRODUCTION

The work undertaken in this thesis research was performed at GE Hitachi Nuclear Energy in Wilmington, North Carolina, in the Transient Analysis group. As part of the Nuclear Analysis Center of Excellence, the Transient Analysis group at GE Hitachi performs transient analyses for each new fuel cycle. The potentially limiting transients for each plant are analyzed each cycle. These transients are then analyzed at rated and sometimes off-rated power levels to demonstrate compliance with the fuel thermal margin requirements and American Society of Mechanical Engineers (ASME) reactor vessel overpressure protection criteria.

This study on expanded exposure windows was performed because utilities often desire or request additional flexibility for operation of Cycle N-1 such that it does not affect the licensing of Cycle N. The intent was to show that the current method of performing licensing calculations is valid even if the previous cycle ends thirty days earlier or later than expected. This study focused on a BWR/4 and a BWR/6 plant. The BWR/4 was based on a 560 bundle core at 2536 MWt and the BWR/6 was based on 748 bundle core at 3758 MWt.

The objective of the research was to perform transient analyses based on early/late shutdown conditions at the end of the previous cycle (EOC N-1). Early/late conditions as large as 1000 MWd/ST (± 30 Effective Full Power Days (EFPD)), as well as intermediate early/late conditions of ± 500 MWd/ST were examined. The analyses were performed in the same manner as the nominal exposure EOC N-1 reload licensing analysis. Design variations were minimized except for the EOC N-1 exposure effects in order to establish a valid metric for the dependence of results on previous cycle exposure variation. The early/late shutdown results were compared to nominal licensing results.

The research was a more intensive study of the effect of previous cycle exposure than is performed in a normal reload licensing analysis. Normal reload licensing considers an early shutdown window to assure that cold reactivity and shutdown margin concerns are acceptable (one-rod-out and standby liquid control system) at these potential early EOC N-1 conditions. Previous transient analysis studies have also examined smaller exposure windows, and have found an acceptably small impact on transient performance. For this research, analyses were performed to determine the validity of the Cycle N licensing limits for MCPR (minimum critical power ratio), LHGR (linear heat generation rate), and peak vessel pressure based on nominal EOC N-1 shutdown compared to the results obtained from early or late EOC N-1 shutdown. The expected outcome was to show that the current licensing process is acceptably bounding even for these very large variations in Cycle N-1 shutdown and corresponding variation in exposure distribution.

Boiling Water Reactors

Boiling water reactor technology was developed at Argonne National Laboratory and the Nuclear Energy Division of General Electric (GE) in the 1950s (Lahey, 1993). The first reactor license issued by the Atomic Energy Commission was to Vallecitos 1, which came online near San Jose, California in 1957. This reactor produced 5 MWe and provided power to the Pacific Electric and Gas Company grid. The first commercial plant was the 180 MWe Dresden 1 in Morris, Illinois, in 1961 (BWR/6, 1980).

Figure 1-1 shows the layout of a direct cycle reactor system, such as a BWR. Steam is produced in the vessel itself, where it travels through steam separators and dryers before exiting and traveling to the turbine. Most BWRs have three-stage turbines, of which one is high-pressure and the other two are low pressure. The steam goes through moisture separators and reheaters between each turbine stage. This is crucial because moisture is very damaging to

turbine blades. After exiting the turbine the steam is passed through a condenser to turn it back into water. The water is demineralized and heated, and then again is pumped to the reactor. This water that travels through the reactor and turbine is radioactive due primarily to N^{16}, which has a half-life of seven seconds.

Figure 1-2 shows the steam and recirculation systems for the BWR. Shown on either side are the jet pumps. These pumps recirculate water through the reactor vessel; they generate about two-thirds of the recirculation flow within the reactor vessel. They also allow for load following by the plant through increases or decreases in the coolant flow. The main water/steam path through the vessel is also shown in the figure. Cold water flows down the sides of the vessel in the downcomer region. The recirculation and jet pumps provide the recirculation head to drive the flow through the fuel bundles, where it turns to steam. The steam is then separated from the water in the steam separators and additional moisture is removed when the steam passes through the steam dryer assembly near the top of the vessel. The steam then flows to the turbine.

Figure 1-3 shows a detailed view of the BWR pressure vessel. The steam dryers and separators are shown in more detail. These two systems are static and utilize baffles and turning vanes to remove moisture from the steam. One notable item from the figure is the relatively small size of the core itself – the pressure vessel has many components, of which the fuel itself is a small part.

The control rods enter the vessel from the bottom by hydraulic control rod drives. The control rods themselves are cruciform in shape (when viewed from the top) and fit between sets of four fuel bundles. They are filled with B_4C powder. In addition to the control rods, the BWR power can be controlled by the amount of recirculating water. An increase in the recirculating

water flow rate decreases the void fraction and thus causes power to increase. A decrease in the flow rate will cause power to decrease.

Boiling Water Reactor Evolution

In the 18 years between the first BWR/1 and BWR/6 startup dates, the GE BWR design underwent many changes. The design variations can be classified into two groups – the reactor systems and the containment design. The BWR has six distinct reactor system design generations and three containment designs. The Dresden 1 BWR/1 plant, with its ex-vessel steam generators and dry containment is very different from the Perry BWR/6 plant, with its Mark III containment and lack of steam generators.

The steam generators disappeared in the BWR/2, the first of which was Oyster Creek in 1959. These plants were purchased solely for economic reasons, and were thus much larger than the BWR/1 plants. Five recirculation loops were used to remove heat from the core. This number was reduced to two by the introduction of internal jet pumps in the BWR/3 design, such as in plants like Dresden 2. The BWR/3 also improved the Emergency Core Cooling System (ECCS) in the plants by adding spray and reflood capability. Figure 1-4 shows the evolution of the BWR.

Changes in the next generations of the BWR family were smaller in scale, and focused on improving power density in the BWR/4 (first online in 1972) and improving the ECCS and introducing recirculation valve flow control in the BWR/5 (first online in 1972). The BWR/6 (first online in 1978) improved the nuclear system protection system and reduced the size of the control room.

These changes in the reactor systems generally improved the transient and accident response of BWRs. One major difference between BWR/4 and BWR/6 plants, as seen in this

research, is their neutron flux response during transients. The BWR/4 response is much more severe than the BWR/6 increase due to the faster scram speed introduced with the BWR/6.

The containments for BWR plants changed greatly, as well. The first BWRs used dry, spherical containments. The elimination of the steam generators and reduction of the outside recirculation loops in turn allowed for the reduction of the containment volumes. The Mark I containment, shaped like an inverted light bulb, holds water in its torus at the bottom of containment. The Mark II containment has a conical shape and a large containment drywell. The Mark III containment is universally used in BWR/6 plants. It is an easily-constructed right circular cylinder within a free-standing steel structure. Figure 1-5 shows the evolution of the BWR containment (ABWR, 2006).

Boiling Water Reactor Fuel Types

The two most advanced fuel bundles offered by GNF were considered in this study, GE14 and GNF2 Advantage. Both fuel product lines contain tie plates, spacers, channel boxes, fuel rods, and water rods made up of Zircalloy-2, an alloy of zirconium, tin, iron, chromium, and nickel. The fuel bundles are 10x10 arrays and contain 92 fuel rods, 14 of which are part-length rods. GNF2 Advantage bundles contain eight part-length rods that are approximately two thirds the full length and six shorter part-length rods (GNF2 Advantage, 2007). The part-length rods of the GE14 bundle are all the same length (DiFillipis, 2005). The bundle also contains two large water rods which take the place of eight fuel rods. The rods are all surrounded by a channel box that prevents cross-flow between channels. Tie plates at the top and bottom, as well as spacers spaced axially, hold the rods in place (GNF2 Advantage, 2007). Figure 1-6 shows the GE14 fuel bundle. Figure 1-7 shows the GNF2 Advantage fuel bundle.

The fuel rods are filled with stacked high density, ceramic UO_2 or $(U, Gd)O_2$ pellets. The zircalloy cladding is lined with a thin barrier layer of zirconium and the rod is pressurized with

helium gas. Global Nuclear Fuel (GNF) uses gadolinia as its burnable poison of choice, in the (U, Gd)O_2 pellets.

In reload fuel, GNF2 Advantage and GE14 fuel rods typically have a 6-12 inch blanket at the bottom and typically a six inch blanket at the top. In addition, in the gadolinia rods the enrichment can also vary axially.

Safety Parameters of Interest

The transient analysis process is undertaken to determine the values for several safety factors: MCPR, fuel centerline melting, excessive cladding strain, and peak vessel pressure.

Minimum Critical Power Ratio

The minimum critical power ratio is in essence a measure of how far a fuel rod is from transition boiling; that is, how far the fuel rod is from leaving the highly efficient nucleate boiling behind and moving into transition boiling and film boiling. The value used to determine this margin is the critical power ratio (CPR) (Todrias, 1990).

To determine the CPR, a critical steam quality is defined where the onset of transition boiling would occur. This quality is defined in terms of boiling length, mass flow rate, power level, pressure, local steam quality, bundle geometry, and local peaking power. A critical power is found that would produce the critical quality, and then the critical power ratio is the ratio of the critical power to the operating bundle power. The minimum critical power ratio is the ratio of the critical power to the maximum operating bundle power (Lahey, 1993).

No reactor operator wishes to operate at the CPR, so several layers of conservatism are applied for actual reactor operation. Each plant has a safety limit minimum critical power ratio (SLMCPR), which usually ranges from 1.07 to 1.10, and then operating limit minimum critical power ratios (OLMCPR) are defined for each cycle and operating condition, in an attempt to be certain the MCPR never approaches the value of one for an anticipated operational occurrence

(transient event). To determine the OLMCPR, the ΔCPR for each type of transient, such as 0.12 for a Feedwater Controller Failure (FWCF) or 0.19 for a Load Rejection with No Bypass (LRNBP), is added to the SLMCPR (Ishigai, 1999). The highest, or most limiting, OLMCPR is set as the limit for the cycle.

Linear Heat Generation Rate

The LHGR is a measure of the power produced by a fuel pin divided by the length of active fuel. In the core, a maximum linear heat generation rate is found by finding the fuel rod with the highest surface heat flux. This value is monitored to be sure that fuel thermal and mechanical limits are not exceeded. For operational transients, the fuel thermal and mechanical overpowers are evaluated for margin to thermal-mechanical limits; these values are margins to fuel melt and margin to 1% cladding plastic strain.

Peak Vessel Pressure

The ASME limit for peak vessel pressure is 1375 psig for upset conditions, which is 10% greater than the design pressure. This assures the integrity of the reactor vessel during postulated overpressure events. To prevent the vessel pressure from exceeding this limit, safety / relief valves are set to open if the vessel pressure reaches the opening setpoint for the safety / relief valves; these valves will vent steam to the containment to relieve the system pressure. The most limiting transient for peak vessel pressure is Main Steam Isolation Valve Closure with a Flux Scram (MSIVF).

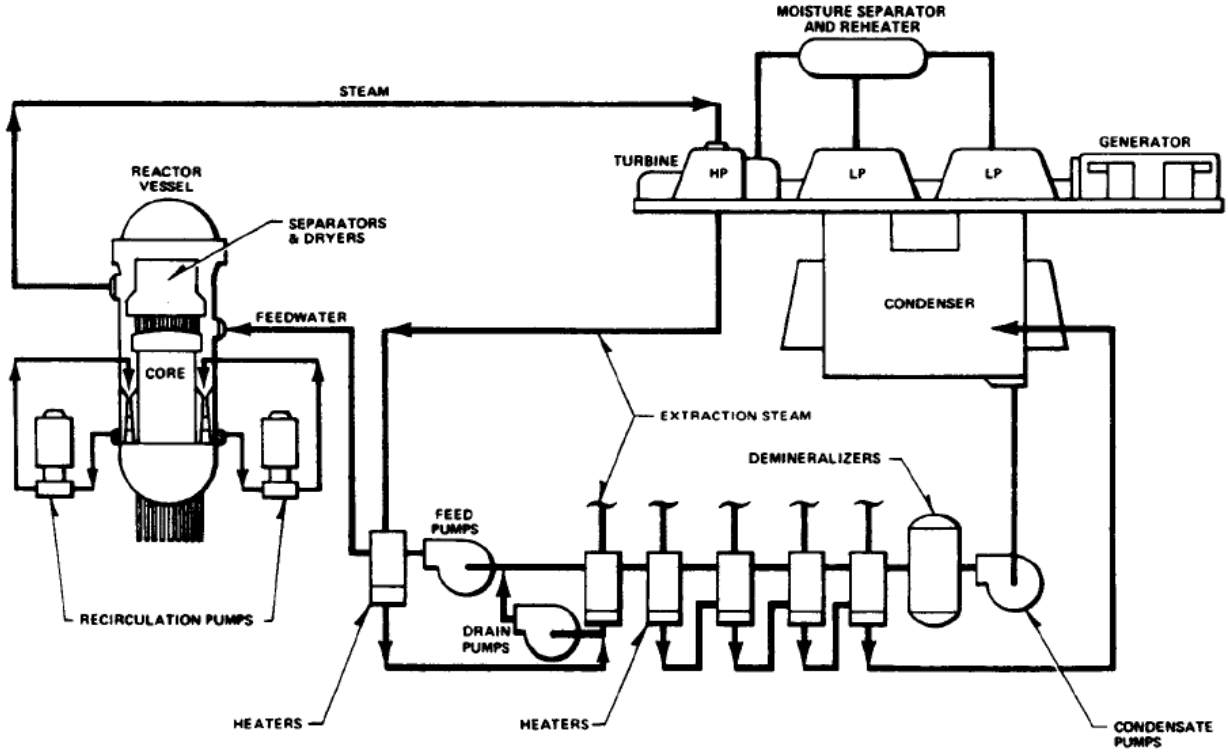

Figure 1-1. Direct cycle reactor system (BWR/6, 1980).

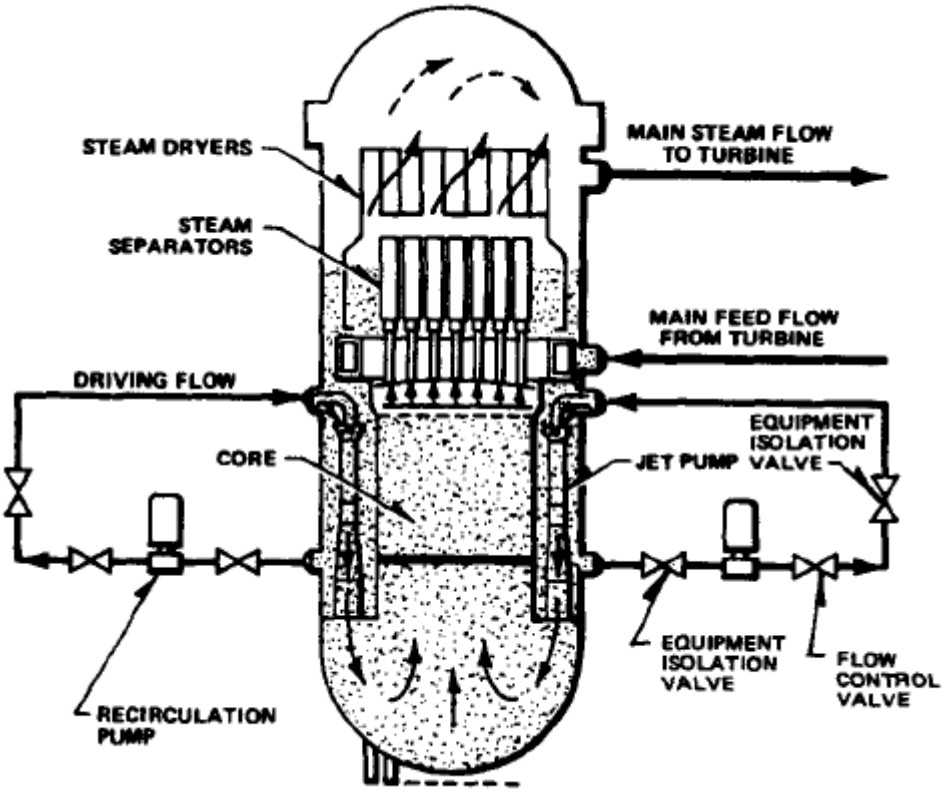

Figure 1-2. BWR/6 steam and recirculation systems (BWR/6, 1980).

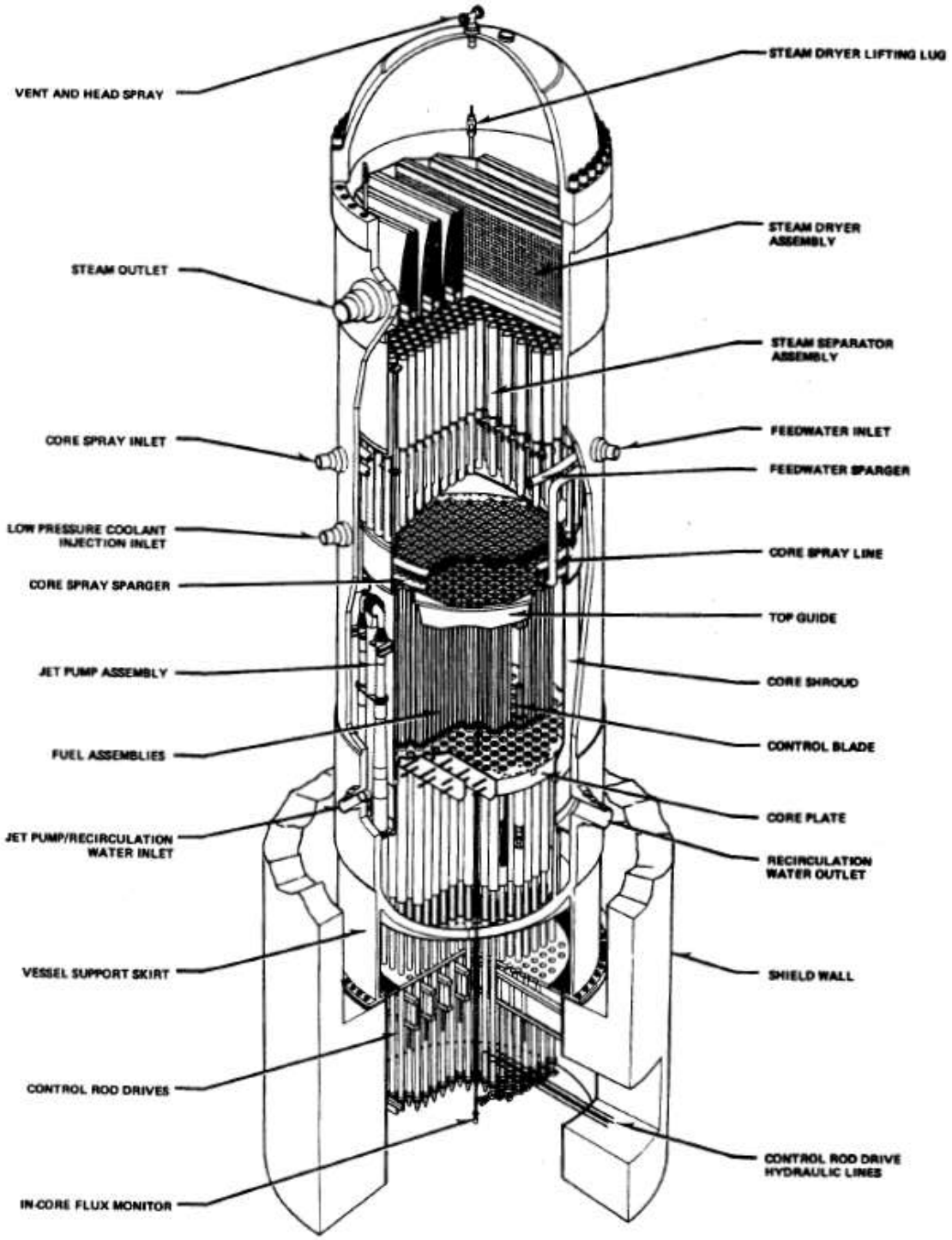

Figure 1-3. The BWR/6 pressure vessel (BWR/6, 1980).

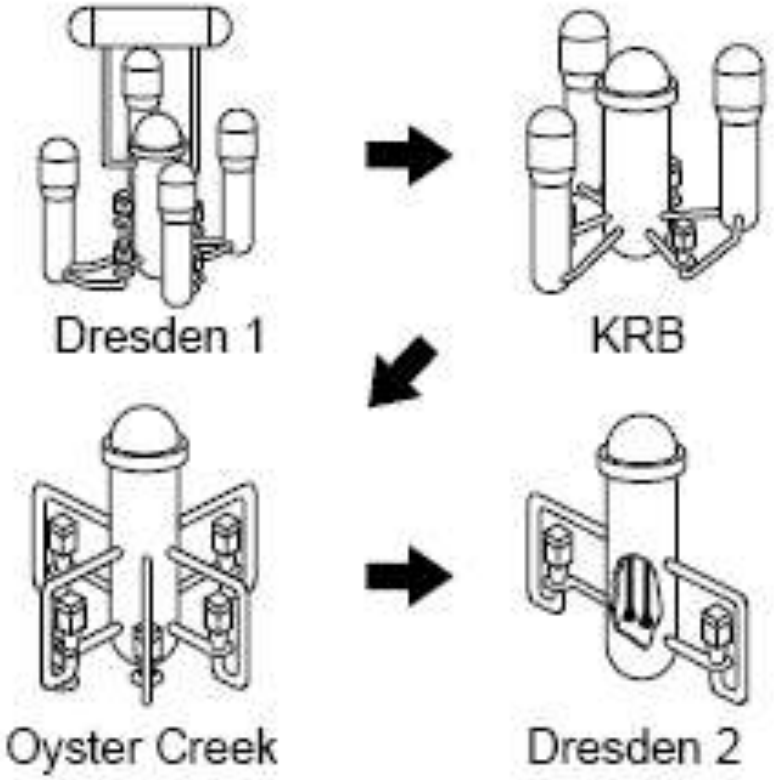

Figure 1-4. Evolution of the BWR reactor system design (ABWR, 2006).

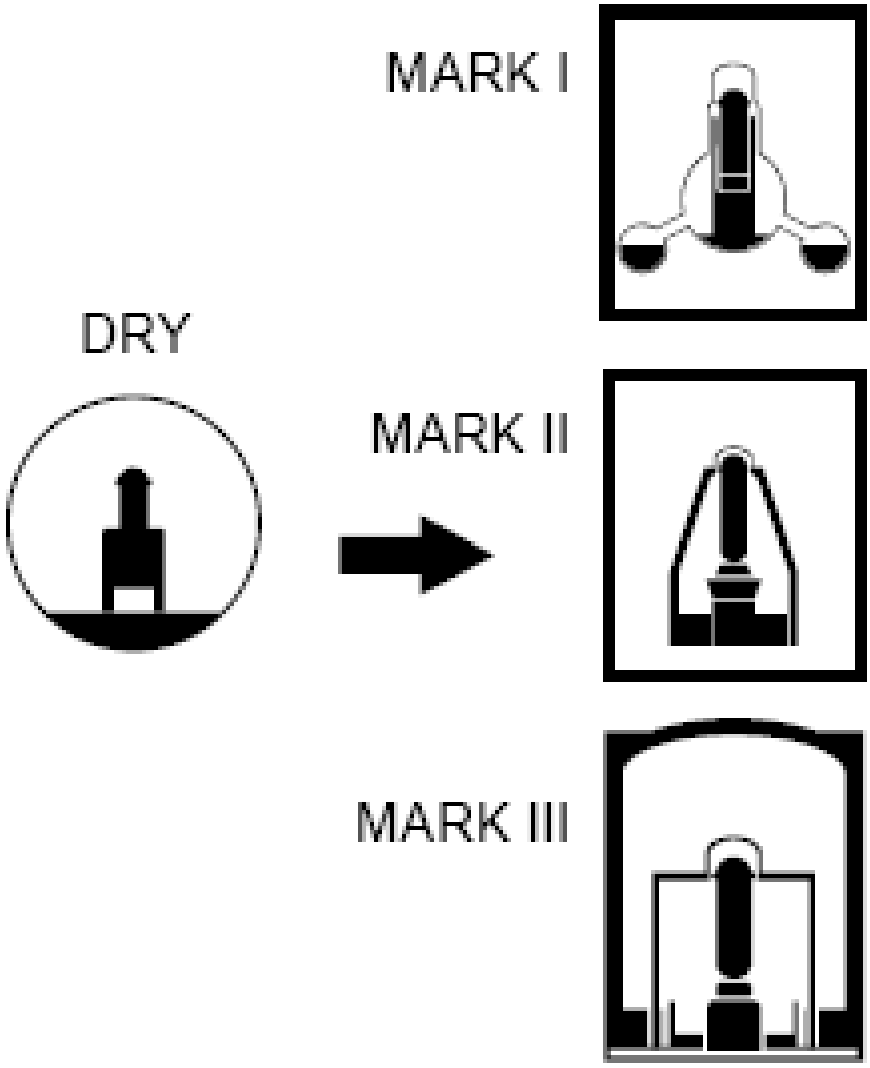

Figure 1-5. Containment evolution of the BWR (ABWR, 2006).

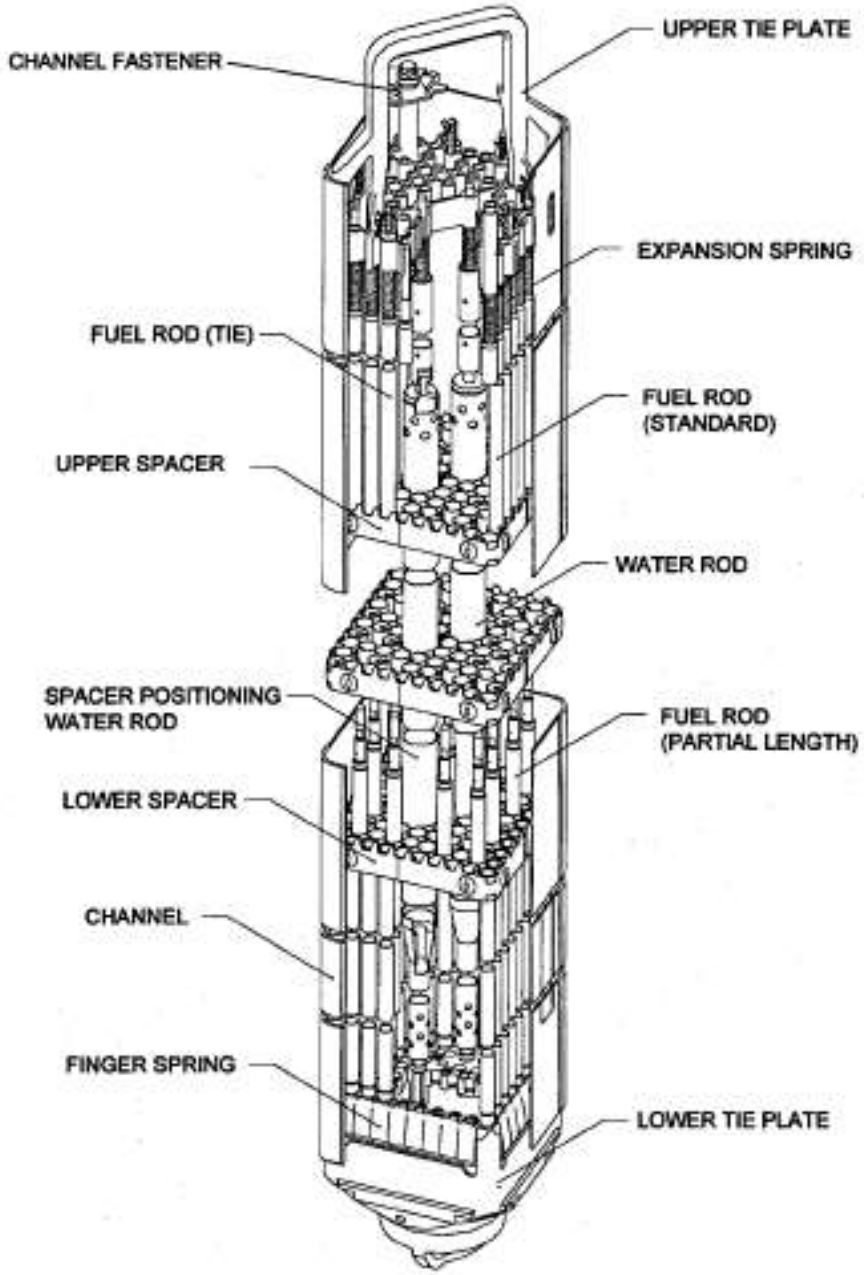

Figure 1-6. The GE14 fuel bundle. (DeFilippis, 2005).

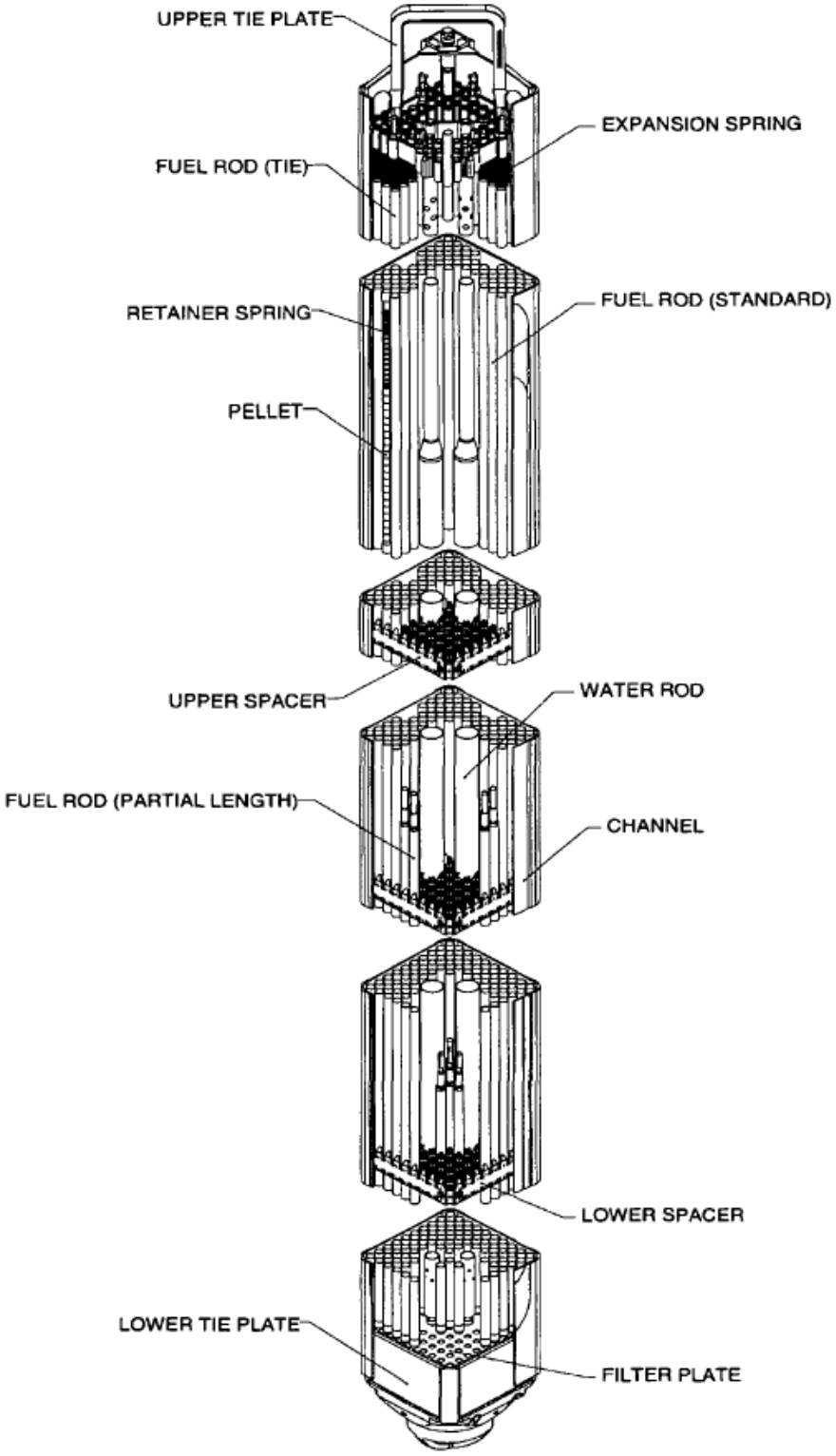

Figure 1-7. The GNF2 Advantage fuel bundle (GNF2 Advantage, 2007).

CHAPTER 2
APPROACH AND METHOD

Exposure Cases

Five exposure cases were analyzed for this research. In addition to the nominal case, state-point nuclear cross sections (wrap-ups) were created for both plus and minus 500 and 1000 MWd/ST. These wrap-ups contain the fuel bundle exposure and control rod patterns at multiple exposure points throughout the cycle. For both the BWR/4 and the BWR/6, end of cycle (EOC) points were investigated, as well as middle of cycle (MOC) points for the BWR/4. Additionally, normal and reduced feedwater temperatures were analyzed, as well as increased core flow and low core flow conditions.

Operating Domains

Boiling water reactors can operate at varied flow and power configurations. To operate safely, the reactor must operate within a certain area of the power versus flow map, shown in Figure 2-1. The reactor must operate between the minimum and maximum core flow at the various power levels shown on Figure 2-1. This area is shown in Figure 2-1 by the quadrangle formed by the minimum pump speed line, minimum rod line, 100% current licensed thermal power (CLTP) rod line that pass through point E, and line EH. In addition, the plant may opt to operate in low core flow or increased core flow conditions. The low core flow region, called the maximum extended load line limit analysis (MELLLA) region, is bound between the line CD and the 100% CLTP rod line. It permits operation at lower core flows. The increased core flow region is bound in the figure by 100% and 105% core flow, and lines EF and HG. Increased core flow provides flexibility at rated power and can be used to extend the operating cycle by the additional reactivity associated with the increased core flow.

In addition, the feedwater temperature may be altered during operation. The plant usually operates at its normal feedwater temperature (NFW), but a reduced feedwater temperature will result in higher reactivity because the reduced temperature feedwater (RFW) is more dense, resulting in more moderator in the core. This may be used by the plant when a feedwater heater is out of service or for use in extending the cycle with the increased moderation and reactivity.

Combinations of ICF, MELLLA, RFW, and NFW are used to produce multiple operating domains for a variety of situations. These combinations form the various initial conditions studied in the transient analysis.

Reactor and Fuel Parameters

The BWR/4 had a licensed thermal power of 2536 MW and a rated core flow of 77 Mlb/hr. The increased core flow domains were analyzed at 105% of rated flow, or 80.85 Mlb/hr. The low flow domains were analyzed at 79.8% of rated flow, or 61.45 Mlb/hr. The increased core flow and low flow values were analyzed to envelope the allowed operating range at the license power. The license (rated) power was used for all MCPR analyses. The normal feedwater temperature was 424 ^{o}F, while the reduced feedwater temperature was analyzed at 344 ^{o}F. The cycle analyzed for BWR/4 contained a mixed core of GNF2 Advantage and GE14 fuel.

The BWR/6 analyzed had a licensed thermal power of 3758 MW and a rated core flow of 104 Mlb/hr. The increased core flow domains were analyzed at 105% of rated flow, or 109.2 Mlb/hr. The low flow domains were analyzed at 81% of rated flow, or 84.24 Mlb/hr. The license power was used for all MCPR analyses. The normal feedwater temperature was 425.5 ^{o}F, while the reduced feedwater temperature was 255.5 ^{o}F. The BWR/6 core contained GE14 fuel bundles.

Transients

Five transients were analyzed to determine required operating limits: Main Steam Isolation Valve Failure – Flux Scram (MSIVF), Load Rejection without Bypass (LRNBP), Turbine Trip without Bypass (TTNBP), Feedwater Control Failure (FWCF), and Pressure Regulator Failure, Downscale (PRFDS). The MSIVF transient was analyzed at a thermal power of 102% of rated to determine the margin to the ASME upset code limit for the peak vessel pressure and the technical specification safety limit for the dome pressure. The LRNBP, TTNBP, FWCF, and PRFDS were analyzed at rated power to determine the MCPR limit and to show compliance with the LHGR overpower limits. Additionally, the FWCF was also analyzed with the turbine bypass valve out of service (NBP).

For the BWR/4, LRNBP, TTNBP, FWCF, and MSIVF were all analyzed; additionally, the analysis considered two safety relief valves out of service (2SRVOS) because that is a Nuclear Regulatory Commission (NRC) -approved operating strategy for this plant and this flexibility causes more limiting overpressure conditions than with the safety relief valves in service. Table 2-1 shows the analysis matrix for the BWR/4, while Table 2-2 shows the transient matrix performed for the BWR/4 -500 MWd/ST case. Similar matrices were performed for all the BWR/4 cases.

For the BWR/6, LRNBP, TTNBP, FWCF, and PRFDS were all run with two relief valves out of service (2RVOS) because it is an NRC-approved operating strategy and more limiting than with the valves in service. The MSIVF transient was run with six safety valves out of service (6SVOS). Table 2-3 shows the case matrix for the BWR/6 analyses, while Table 2-4 shows the transient matrix performed for the BWR/6 –500MWd/ST case. Similar matrices were performed for all the BWR/6 cases.

During LRNBP, the generator experienced a loss of electrical load. To prevent a turbine/generator over-speed condition, the turbine control valves must close as quickly as possible. This cut off the flow of steam through the turbine, which removed the working fluid and causes the turbine to slow to a stop. This created an increase in pressure as the steam headed to the turbine is stopped, resulting in a pressure wave that transmitted through the system back to the reactor vessel. Positive reactivity was introduced into the core as voids collapsed under the pressure wave, resulting in an increase in neutron flux. However, this also increased the heat flux, which, combined with control rod scram, inserted negative reactivity to bring the system under control. If needed, safety relief valves could also open, depressurizing the system by piping steam into the suppression pool.

Figure 2-2 shows the typical system flux and core flow response after LRNBP. After the turbine control valves were completely shut at 0.152 sec, the neutron flux spiked to several times its rated value. This increase in neutron flux was quickly turned around by the start of control blade motion at 0.281 sec. There was also a moderate increase in the average surface heat flux, which follows the neutron flux, but this was turned around, as well, by the insertion of the control rods. The data for the figure was from the BWR/4, which has a much more severe neutron flux response than BWR/6 plants.

Figure 2-3 shows the increase in vessel pressure and the flow of the relief valves, as well as the turbine bypass flow during LRNBP. The vessel pressure rose to above 175 psi above the initial condition before the relief valves opened at 1.975 sec. The relief valves brought the pressure down to the closing setpoints associated with the safety relief valves. It should be noted that the bypass valve flow was constant at zero because this transient did not credit the turbine bypass valves.

Figure 2-4 shows vessel steam flow, turbine steam flow, and feedwater flow in the core during LRNBP. The turbine steam flow cut off at 0.15 sec and stayed at zero for the rest of the transient. The vessel steam flow showed oscillatory behavior; this is a result of the pressure wave propagating between the vessel and the turbine control valves. Once the safety relief valves opened, the magnitude of the vessel steam flow oscillations dampened to the safety relief valve flow rate.

During TTNBP, the turbine tripped off-line and the turbine stop valves closed. This trip could have been caused by several events, including large vibrations, high water level in the vessel, and low condenser vacuum. A similar course of events occured as with LRNBP. Figure 2-5 shows the neutron and average surface heat flux, as well as the core inlet flow during the TTNBP. The figure is very similar to Figure 2-2 because after the initiation, the TTNBP had a very similar course of events as the LRNBP. Figures 2-6 and 2-7 are also similar to their counterparts for LRNBP.

During the MSIVF, the reactor was assumed to be operating at 102% of rated power. A main steam isolation valve (MSIV) closed in the minimum time and the first available scram on MSIV position was not credited. The pressure increased in the core, collapsing voids, causing an increase in power. The neutron flux increased and reached the scram setpoint, which triggered the control rod scram and decreased the power to bring the system back under control. The safety relief valves could also open to reduce the vessel pressure.

Figure 2-8 shows the neutron and average surface heat flux, as well as the core inlet flow, during MSIVF. The initial pressure of the vessel was 1094 psig. At time zero the MSIVs began closing and were fully closed at 3.00 sec. When the MSIVs began to have a significant effect on the steam flow (at approximately 0.6 seconds) there was a spike in neutron flux that resulted in a

scram. The control rods began motion at 1.81 sec and turned the neutron flux increase around. The heat flux again followed the neutron flux, but at much smaller values.

Figure 2-9 shows the vessel pressure rise, relief valve flow, and bypass valve flow for the MSIVF. Just like in the LRNBP and TTNBP, the vessel pressure increased until the relief valves opened, where they vented the steam and allowed the pressure to decrease.

Figure 2-10 shows the vessel steam flow and turbine steam flow during MSIVF. The turbine steam flow dropped off throughout the transient to zero. The vessel steam flow once again showed oscillatory behavior, with a spike in steam flow at about the same time as the second spike in neutron flux. Also noteworthy is that the oscillations were more frequent with the MSIV closure because the main steam isolation valves were much closer to the reactor vessel.

During a FWCF, a postulated failure occurred in the feedwater control system, which demanded the maximum feedwater flow. This caused the water level to rise until it reached the high-water set point, at which time the turbine and feedwater pumps tripped, the turbine stop valve and turbine control valves were demanded to close, and the control rods scrammed. Normally, the turbine bypass system would be demanded to open, directing steam to the condenser. However, if the turbine bypass valves were inoperable the pressure at the turbine inlet and the vessel would have a larger pressure increase due to the buildup of steam. The safety relief valves could open if needed to relieve the pressure.

Figure 2-11 shows the neutron flux, average surface heat flux, core inlet flow, and core inlet subcooling for FWCF with bypass. One factor that is immediately apparent is how long it took for the transient to take effect – FWCF is a relatively slow transient and the effects are not seen until several seconds after the increase in feedwater flow. At zero seconds the feedwater

flow was increased to the maximum demand, about 145% of rated feedwater flow. Feedwater flooded the vessel until the level reached the high water level trip setpoint at 11.5 sec. To protect the turbine, the turbine stop valves and turbine control valves began closing at 11.5 sec and the turbine was tripped. Once the turbine was tripped the FWCF behavior was much like a turbine trip except the turbine bypass system is normally credited in this event. The control blade motion began at 11.82 sec. The characteristic spikes in neutron flux and average surface heat flux are shown in the figure. Also, the core inlet subcooling rose because of the influx of cold water in the core. The large change in inlet subcooling at approximately 12 second was due to the vessel pressurization.

Figure 2-12 shows the vessel pressure rise, relief valve flow, and bypass valve flow for the FWCF. Note that because this transient takes credit for the turbine bypass system, the bypass valves began to open at 11.6 sec. The relief valves began to open at 13.8 sec and turn the pressure was turned around.

Figure 2-13 shows the vessel steam flow, turbine steam flow, and feedwater flow for FWCF. The feedwater flow was a constant 145% until the vessel reached the vessel high level setpoint, and the increase in vessel water can be seen in the vessel level plot in the figure. Once the turbine tripped and the turbine steam flow dropped to zero, the pressure wave propagated from the turbine to the vessel and produced the oscillatory behavior of the vessel steam flow, which was very similar to a TTNBP event.

A transient unique to the BWR/6 licensing is PRFDS. During this transient, the pressure regulators failed, which demanded the turbine control valves to close. The turbine control valves fully closed and the bypass flow was rendered inoperable due to the pressure regulator failure, causing an increase in reactor power and pressure. This event resulted in a slow closure of the

turbine control valves and thus no direct reactor scram occurred. The reactor was scrammed once the high neutron flux setpoint is reached.

Figure 2-14 shows the typical neutron flux, average surface heat flux, core inlet flow, and core inlet subcooling responses during PRFDS. The neutron flux spike was much smaller than that seen during the other transients, and was mostly due to the slower closure of the turbine control valves. The core inlet subcooling rose to over 200% of its rated value because of the sudden pressure increase.

Figure 2-15 shows the typical system pressure rise and valve flow response after PRFDS. As with the other transients, the vessel pressure rose until the relief valves open and vent the steam.

Figure 2-16 shows the typical vessel and turbine steam flow after PRFDS. The oscillatory behavior of the steam flow seen in other transients was not seen during PRFDS. This is because during PRFDS the ~2.5 second TCV closure time was much slower than the ~0.1 second TCV/TSV closure time of transients like LRNBP. Because there was a gradual reduction in steam flow, a large pressure wave is not created at the turbine inlet (Watford, 2000).

Method

Several codes were utilized to perform the transient analysis. PANACEA was a three-dimensional, nodal diffusion BWR simulator. It coupled the neutronics and thermal hydraulics and determined both the steady state conditions in the core and the nuclear parameters such as k-effective, cross sections for the statepoint of interest, and the delayed neutron fraction. CRNC collapsed the three-dimensional parameters to a one-dimensional form of averaged cross sections so that they could be used by OYDN, the one-dimensional transient simulator. ISCOR and TASC calculated the thermal hydraulic response and hot-channel analysis. TACLE was an automation code that is used to run the codes.

ODYN was the transient simulator and was the code that produced the transient results described in Figures 2-1 through 2-16. Coupling the neutron kinetics with the thermal hydraulics and heat transfer in the core, ODYN was a best-estimate one-dimensional core model. It modeled a multi-node core consistent with the axial noding used in PANACEA to determine the core average response.

The neutron kinetics in ODYN were assumed to be one-dimensional and varied axially with time. Parameters from PANACEA, collapsed by CRNC, were used for the one-energy group diffusion and six delayed neutron groups. To collapse the cross-sections, radial spatial weighting factors were applied to the nuclear parameters. These preserved the dynamic response of the core during core-wide abnormal operating events.

ODYN modeled the average core response and therefore the parameters such as the flow area and gap conductance are represented by averaged values of the fuel types modeled. The model was initialized to the core pressure drop calculated by the BWR steady state simulator. The core model was tuned to be sure that calculated parameters accurately reflect physical parameters in the core, recirculation system and steam line.

The steady-state axial power distribution was calculated in ODYN with its one dimensional kinetics model. The cross section collapsing process assured that it is a close match to the PANACEA results. Once the power distribution was calculated, the steady-state fuel temperature distribution was calculated. These distributions provided a starting point for the transient calculations.

The transient analysis model simulated the reactor core model and the balance of plant. The reactor core model calculated the pressure, flow, neutron flux, heat flux, fuel temperatures,

reactor core exit quality, and core pressure drop. The plant model calculated steam line dynamic response along with the recirculation loop, feedwater, and pressure control systems.

During transients, pressure, flow, neutron flux, and heat flux were dependent on time and are determined at each time step. These factors provided boundary conditions for the next time step, and a new neutron flux, fuel temperatures, pressure, heat flux, and other parameters are determined again for the new time. To calculate the thermal hydraulic behavior of the core, a combination of mass and energy conservation for liquid and vapor plus a momentum conservation equation for the mixture was used in two channels – a heated one representing the average core conditions and one simulating the bypass region (Supplemental Safety Evaluation, 1981).

ISCOR and TASC took the core pressure, core flow, inlet subcooling, power generation, and axial power shape, all time-dependent, as inputs from ODYN. TASC calculated the change in CPR during the transient, using the core pressure, axial power shape and inlet subcooling from ODYN. To determine the hot channel power and flow, ISCOR calculated the hot channel inlet flow from the core pressure, core inlet flow, core inlet enthalpy, and core power. The hot channel flow and power were input to TASC and TASC determined the MCPR. This process iterated until the calculated MCPR was equal to the SLMCPR.

Figure 2-17 shows the relationship between ODYN, ISCOR, and TASC, and the inputs and outputs associated with the codes. The one-dimensional nuclear data and core geometry were fed into ODYN, which in turn provided the core inlet flow to ISCOR. The core power and power shape, core pressure, and core inlet enthalpy from ODYN, plus the hot channel power and hot channel inlet flow from ISCOR, were fed into TASC, which calculated the critical power ratio and maximum rod temperature versus time.

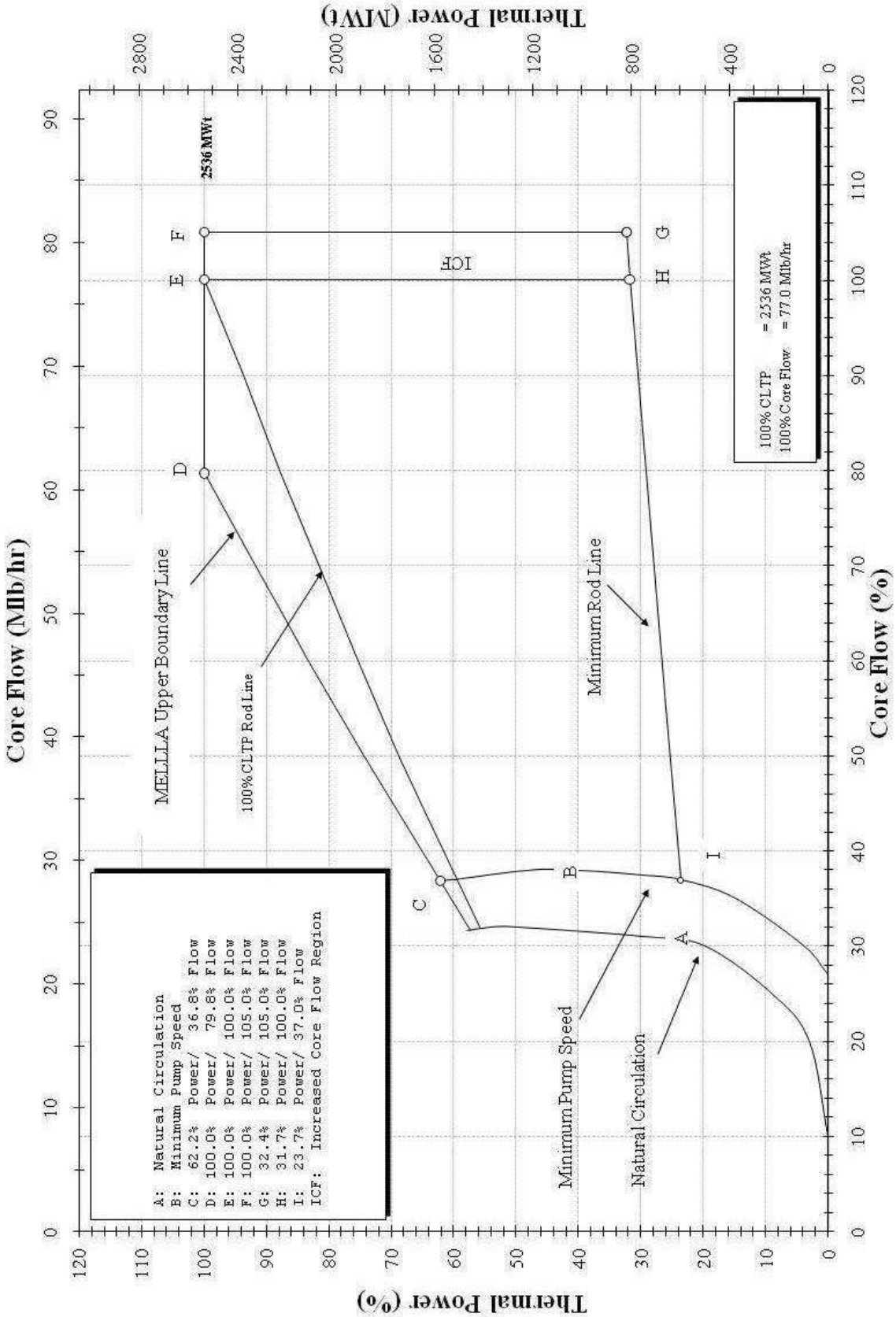

Figure 2-1: Power/Flow map for the BWR/4.

Table 2-1. Case matrix performed for the BWR/4 analyses.

Core Flow	Feedwater Temperature	Nominal	+1000 MWd/ST	+500 MWd/ST	-500 MWd/ST	-1000 MWd/ST
Increased	Normal	ICF___TN	IF_P1ETN	IF_P5ETN	IF_-5ETN	IF_-1ETN
Increased	Normal	ICF___TN (MOC)	IF_P1MTN	IF_P5MTN	IF_-5MTN	IF_-1MTN
Low	Normal	MEL___TN	LF_P1ETN	LF_P5ETN	LF_-5ETN	LF_-1ETN
Low	Normal	MEL___TN (MOC)	LF_P1MTN	LF_P5MTN	LF_-5MTN	LF_-1MTN
Increased	Reduced	ICF___TR	IF_P1ETR	IF_P5ETR	IF_-5ETR	IF_-1ETR
Low	Reduced	MEL___TR	LF_P1ETR	LF_P5ETR	LF_-5ETR	LF_-1ETR
Increased	Normal	ICF__BTN*				
Low	Normal	MEL__BTN*				
Increased	Reduced	ICF__BTR*				
Low	Reduced	MEL__BTR*				

Note: The eight-digit names in the table are the domain identifiers for the analysis.
The FWCF/NBP event was included in the domains for the varied exposure cases instead of being in its own domain.
*Domain only analyzes FWCF/NBP event.

Table 2-2. Transient matrix performed for the BWR/4 –500MWd/ST case.

Domain	Flow (% of rated)	Exposure	Feedwater temperature (^oF)	Transient	Equipment out of service
IF_E5LTN	105	EOC	424	MSIVF	2SROVS
IF_E5LTN	105	EOC	424	LRNBP	2SROVS
IF_E5LTN	105	EOC	424	TTNBP	2SROVS
IF_E5LTN	105	EOC	424	FWCF	2SRVOS
IF_E5LTN	105	EOC	424	FWCF	NBP/2SRVOS
LF_E5LTN	79.8	EOC	424	MSIVF	2SROVS
LF_E5LTN	79.8	EOC	424	LRNBP	2SROVS
LF_E5LTN	79.8	EOC	424	TTNBP	2SROVS
LF_E5LTN	79.8	EOC	424	FWCF	2SRVOS
LF_E5LTN	79.8	EOC	424	FWCF	NBP/2SRVOS
IF_E5LTR	105	EOC	344	FWCF	2SRVOS
IF_E5LTR	105	EOC	344	FWCF	NBP/2SRVOS
LF_E5LTR	79.8	EOC	344	FWCF	2SRVOS
LF_E5LTR	79.8	EOC	344	FWCF	NBP/2SRVOS
IF_M5LTN	105	MOC	424	MSIVF	2SROVS
IF_M5LTN	105	MOC	424	LRNBP	2SROVS
IF_M5LTN	105	MOC	424	TTNBP	2SROVS
IF_M5LTN	105	MOC	424	FWCF	2SRVOS
LF_M5LTN	79.8	MOC	424	MSIVF	2SROVS
LF_M5LTN	79.8	MOC	424	LRNBP	2SROVS
LF_M5LTN	79.8	MOC	424	TTNBP	2SROVS
LF_M5LTN	79.8	MOC	424	FWCF	2SRVOS

Table 2-3. Case matrix performed for the BWR/6 analyses.

Core flow	Feedwater temperature	Nominal	+1000 MWd/ST	+500 MWd/ST	-500 MWd/ST	-1000 MWd/ST
Increased	Normal	ICF___TN	IF_P1ETN	IF_P5ETN	IF_-5ETN	IF_-1ETN
Low	Normal	MEO___TN	LF_P1ETN	LF_P5ETN	LF_-5ETN	LF_-1ETN
Increased	Reduced	ICF___TR	IF_P1ETR	IF_P5ETR	IF_-5ETR	IF_-1ETR
Increased	Normal	ICF__PTN*				
Increased	Reduced	ICF__PTR*				

Note: The eight-digit names in the table are the domain identifiers for the analysis.
The PRFDS event was included in the domains for the varied exposure cases instead of being in its own domain.
*Domain only analyzed the PRFDS event.

Table 2-4. Transient matrix performed for the BWR/6 –500MWd/ST case.

Domain	Flow (% of rated)	Exposure	Feedwater temperature (^oF)	Transient	Equipment out of service
IF_-5ETN	105	EOC	425.5	LRNBP	2RVOS
IF_-5ETN	105	EOC	425.5	TTNBP	2RVOS
IF_-5ETN	105	EOC	425.5	FWCF	2RVOS
IF_-5ETN	105	EOC	425.5	MSIFV	6SVOS
IF_-5ETN	105	EOC	425.5	PRFDS	2RVOS
LF_-5ETN	84.24	EOC	425.5	LRNBP	2RVOS
LF_-5ETN	84.24	EOC	425.5	TTNBP	2RVOS
LF_-5ETN	84.24	EOC	425.5	FWCF	2RVOS
LF_-5ETN	84.24	EOC	425.5	MSIFV	6SVOS
IF_-5ETR	105	EOC	255.5	FWCF	2RVOS
IF_-5ETR	105	EOC	255.5	PRFDS	2RVOS

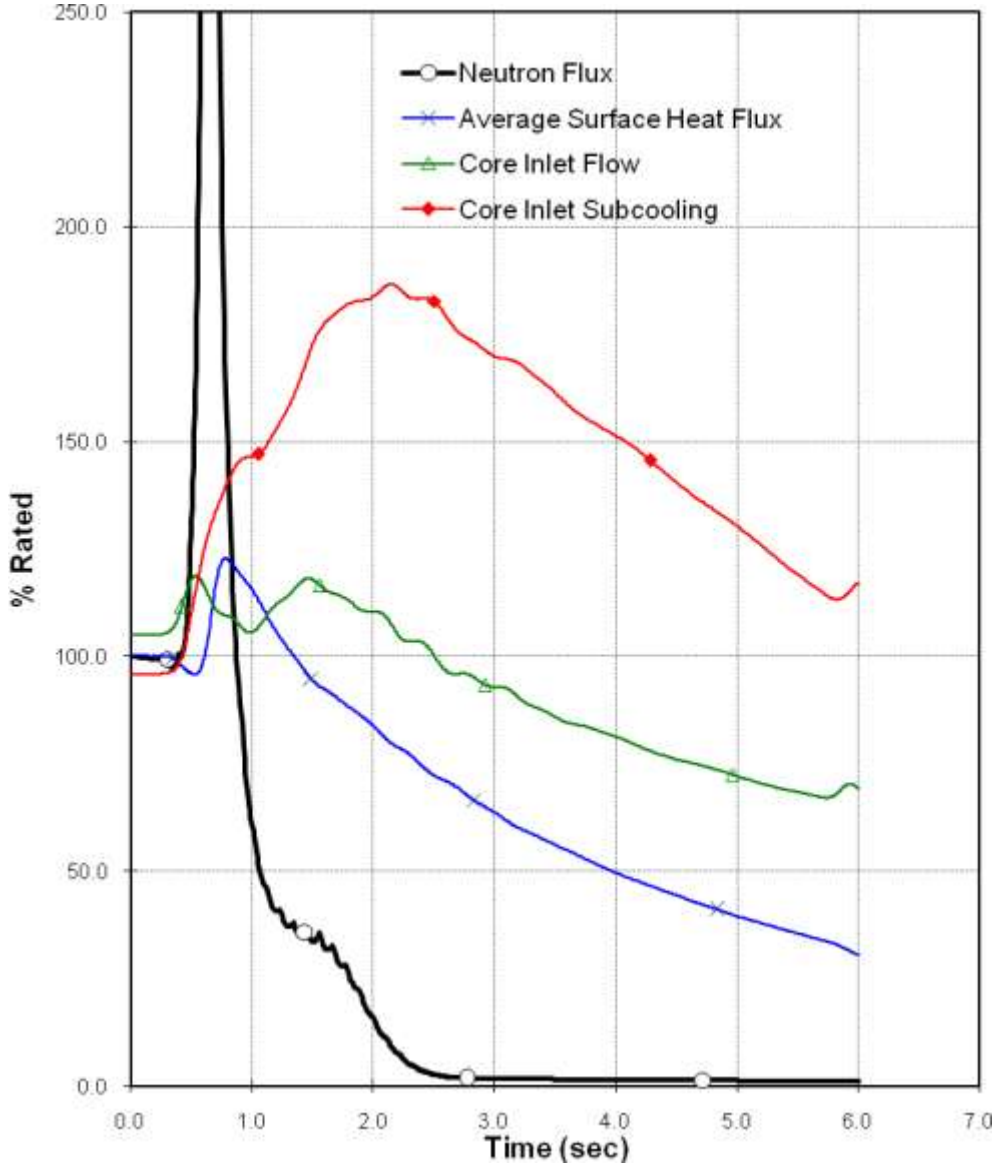

Figure 2-2. Response to LRNBP of system flux and core flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

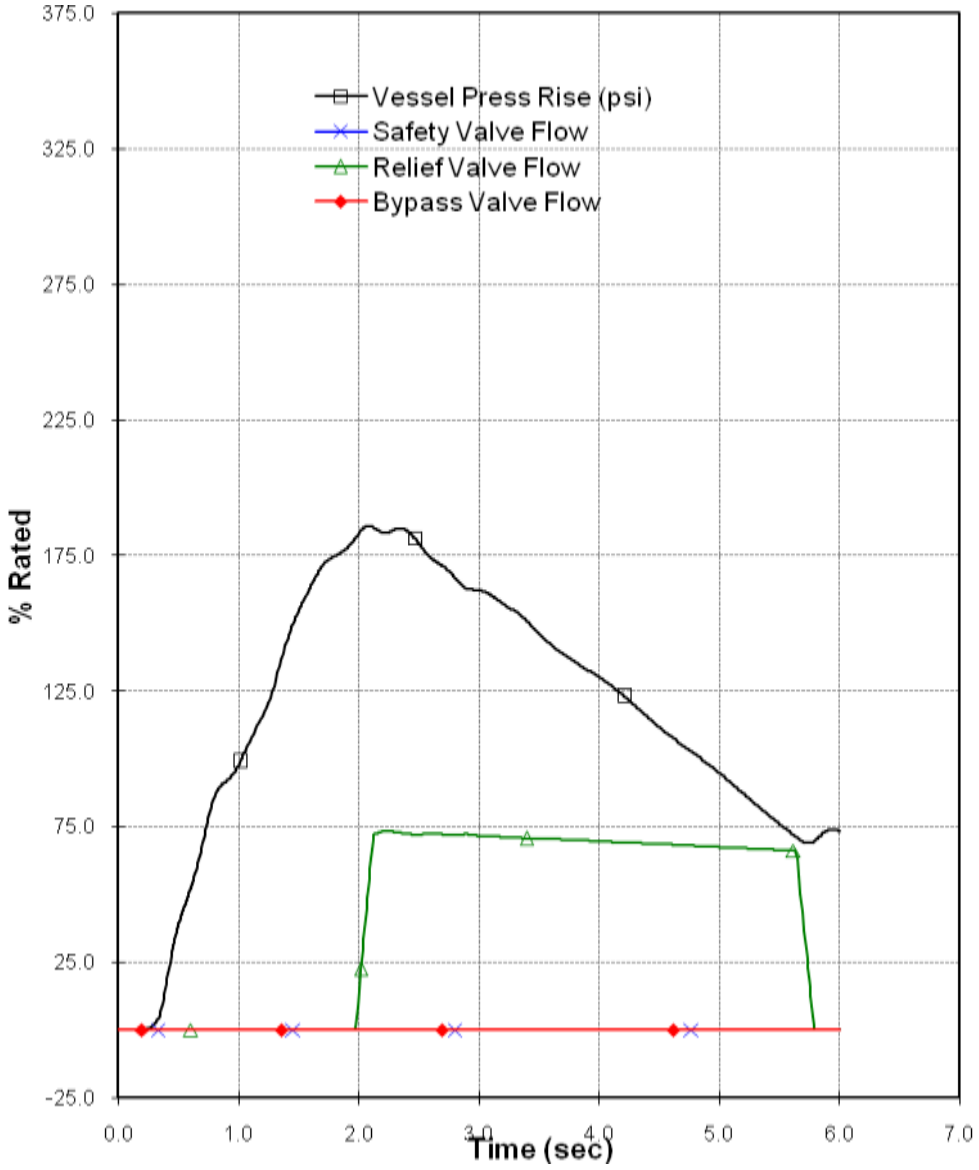

Figure 2-3. Response to LRNBP of system pressure rise and valve flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

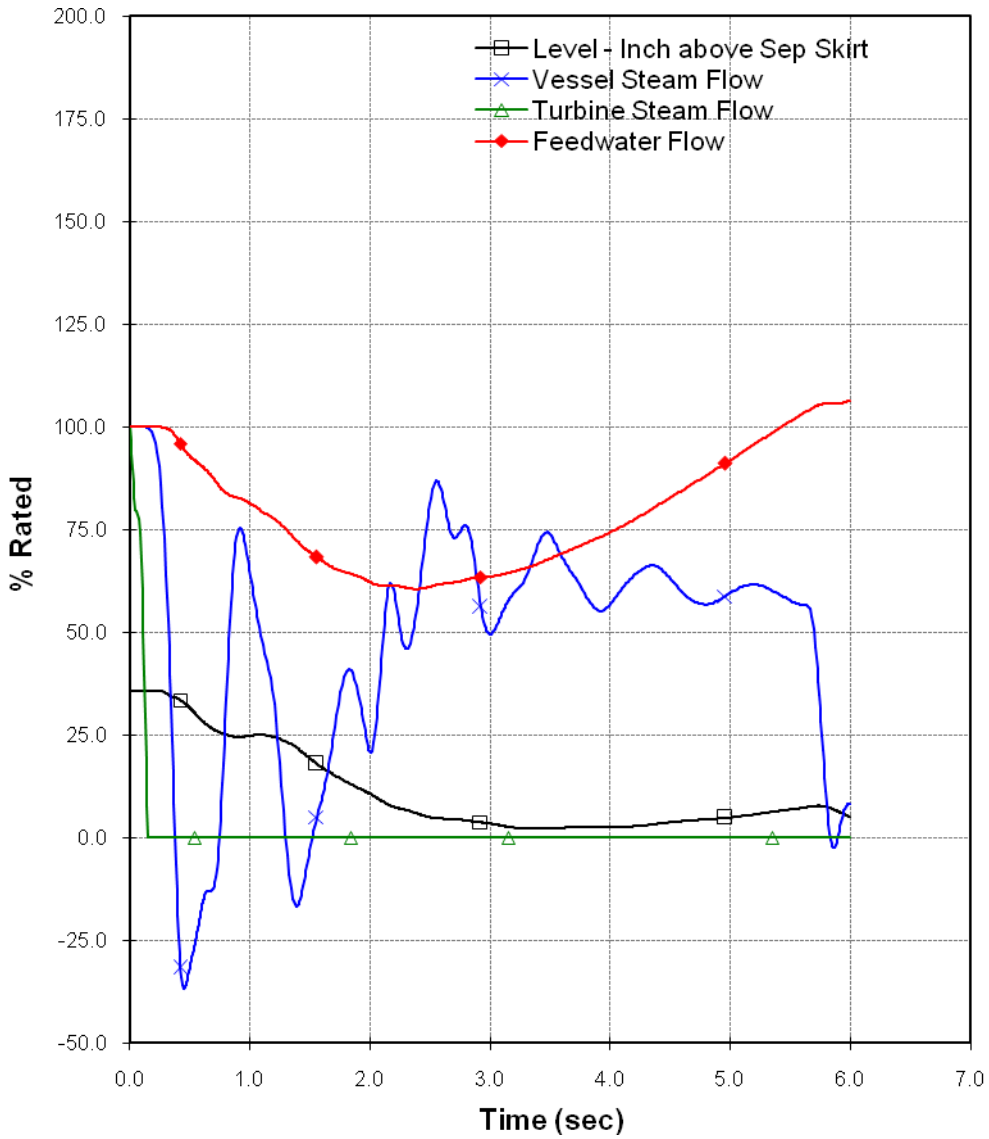

Figure 2-4. Response to LRNBP of typical system flow response. Data are from the BWR/4 - 500 MWd/ST, ICF, NFW case.

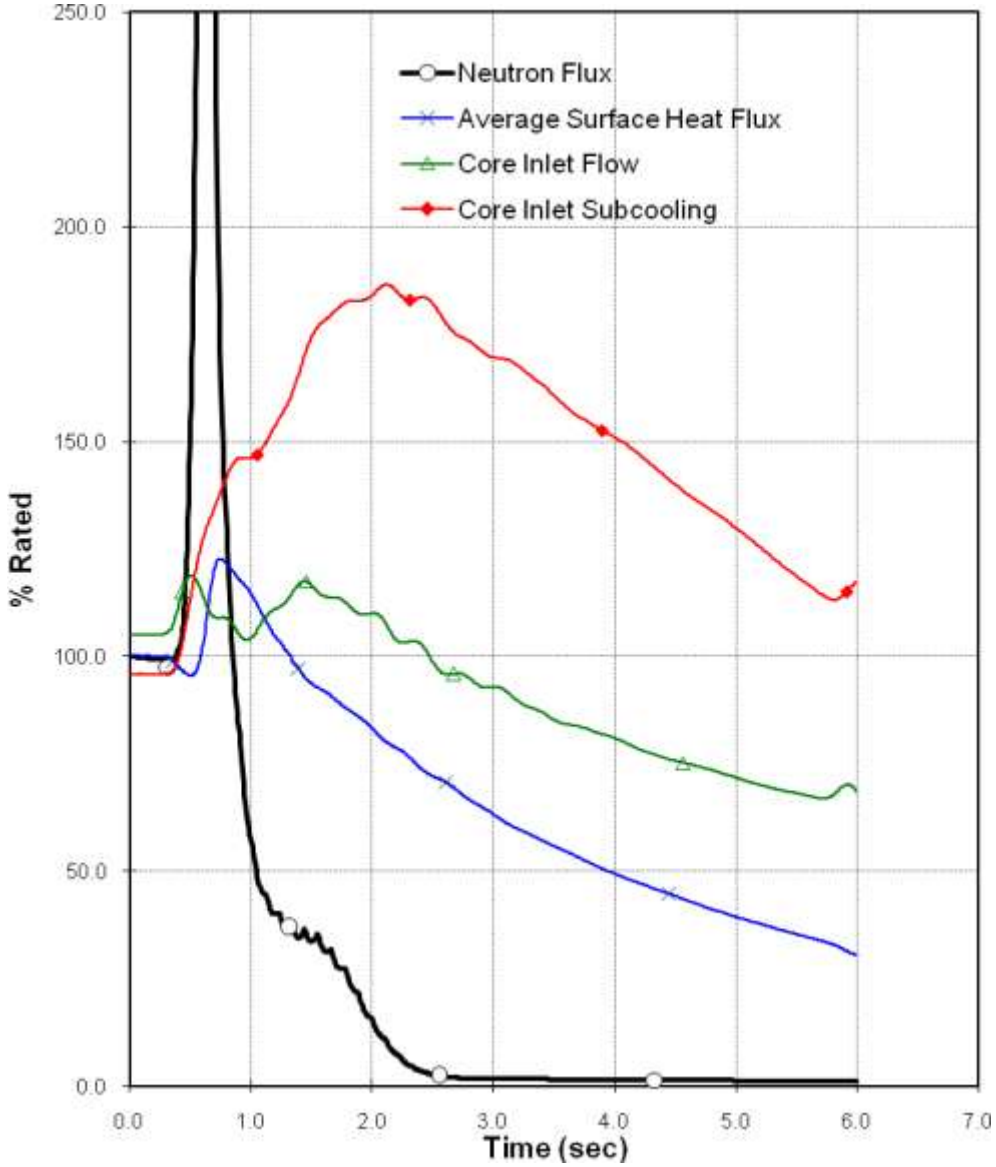

Figure 2-5. Response to TTNBP of system flux and core flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

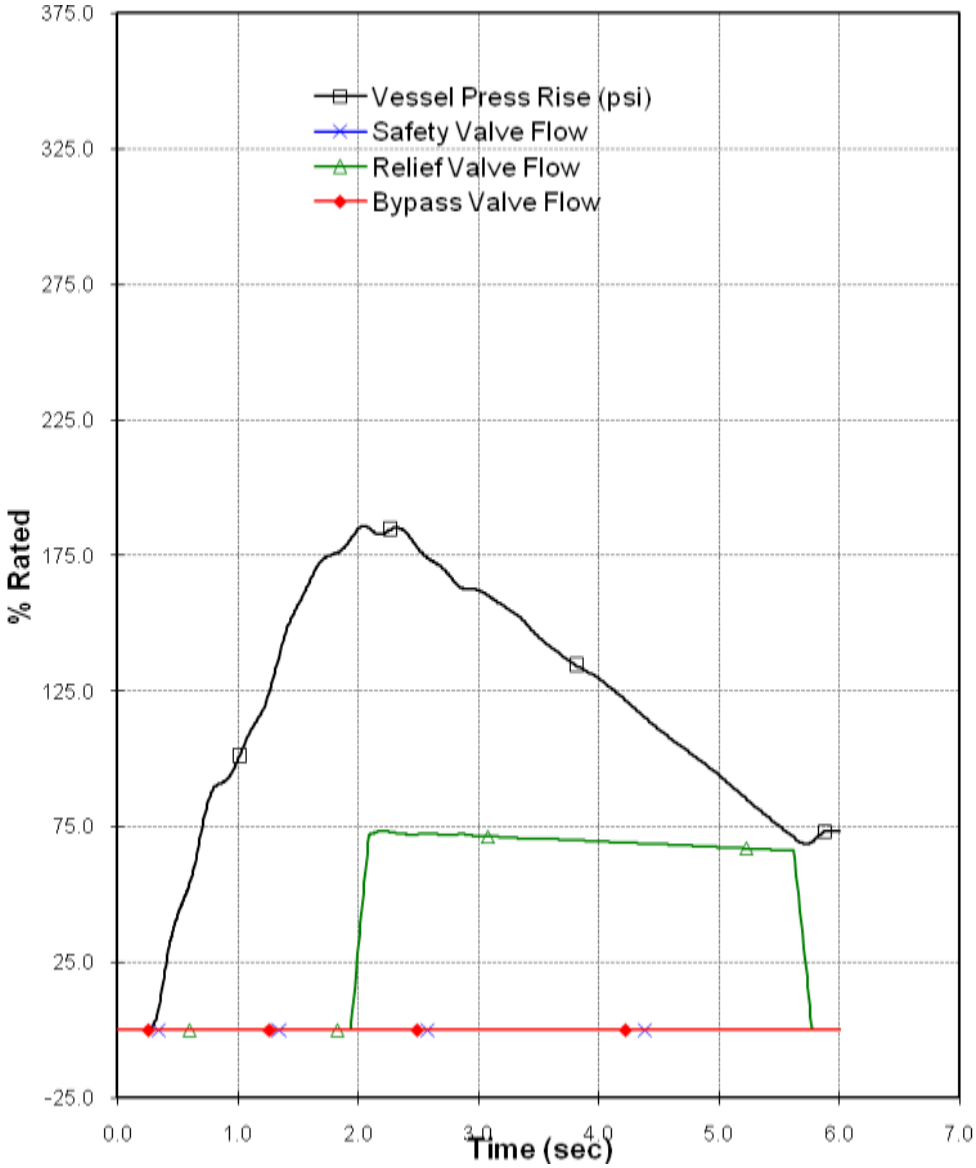

Figure 2-6. Response to TTNBP of system pressure rise and valve flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

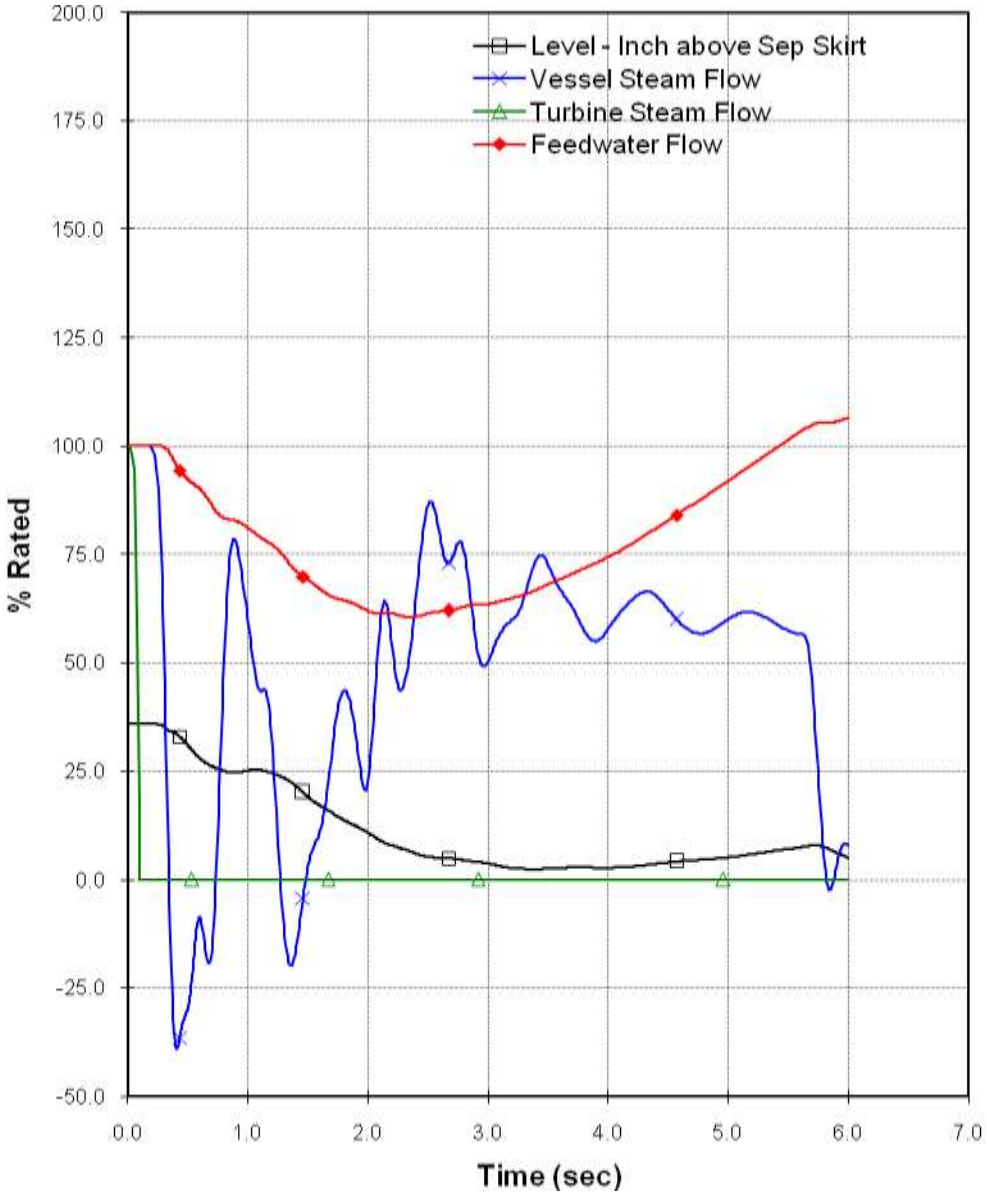

Figure 2-7. Response to TTNBP of system flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

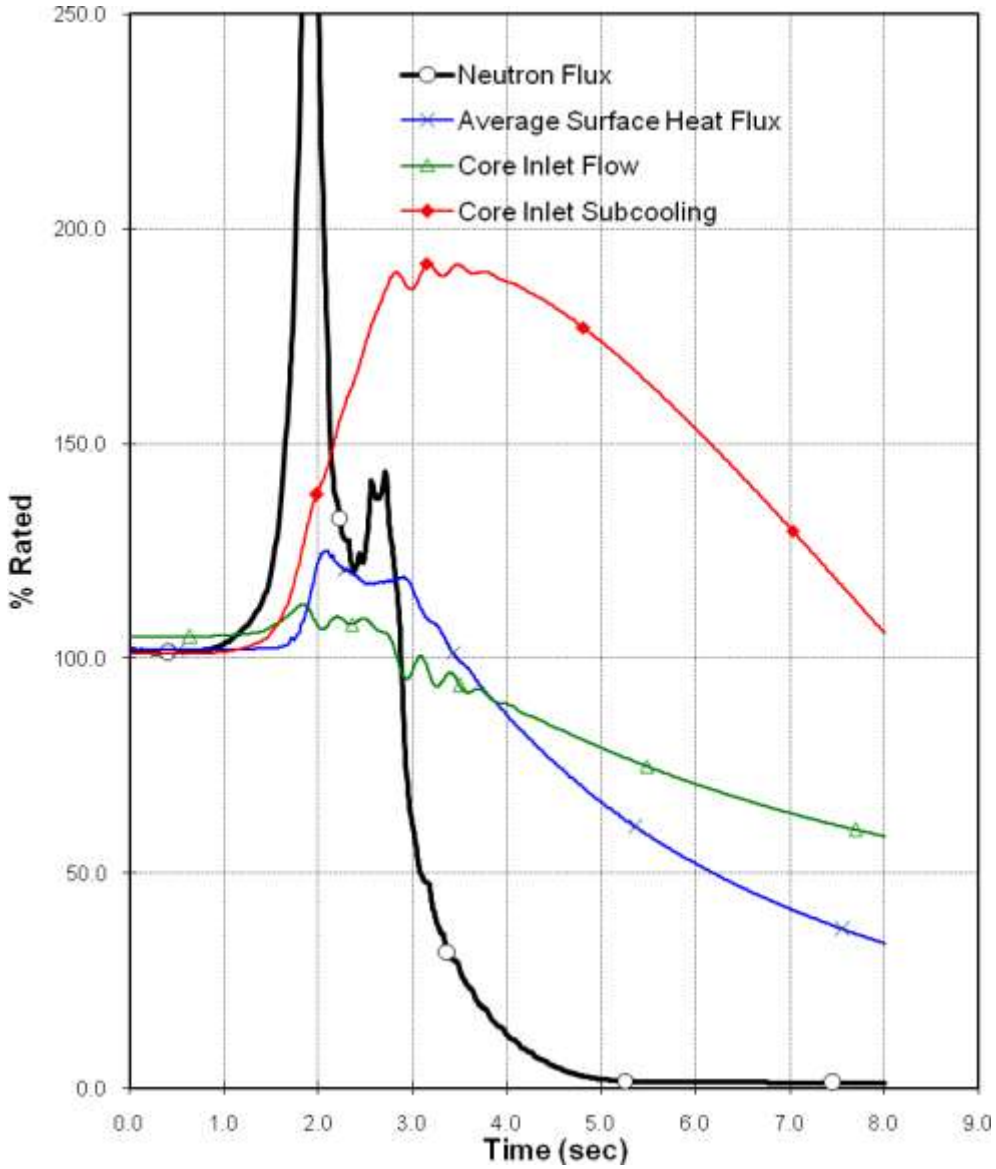

Figure 2-8. Response to MSIVF of system flux and core flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

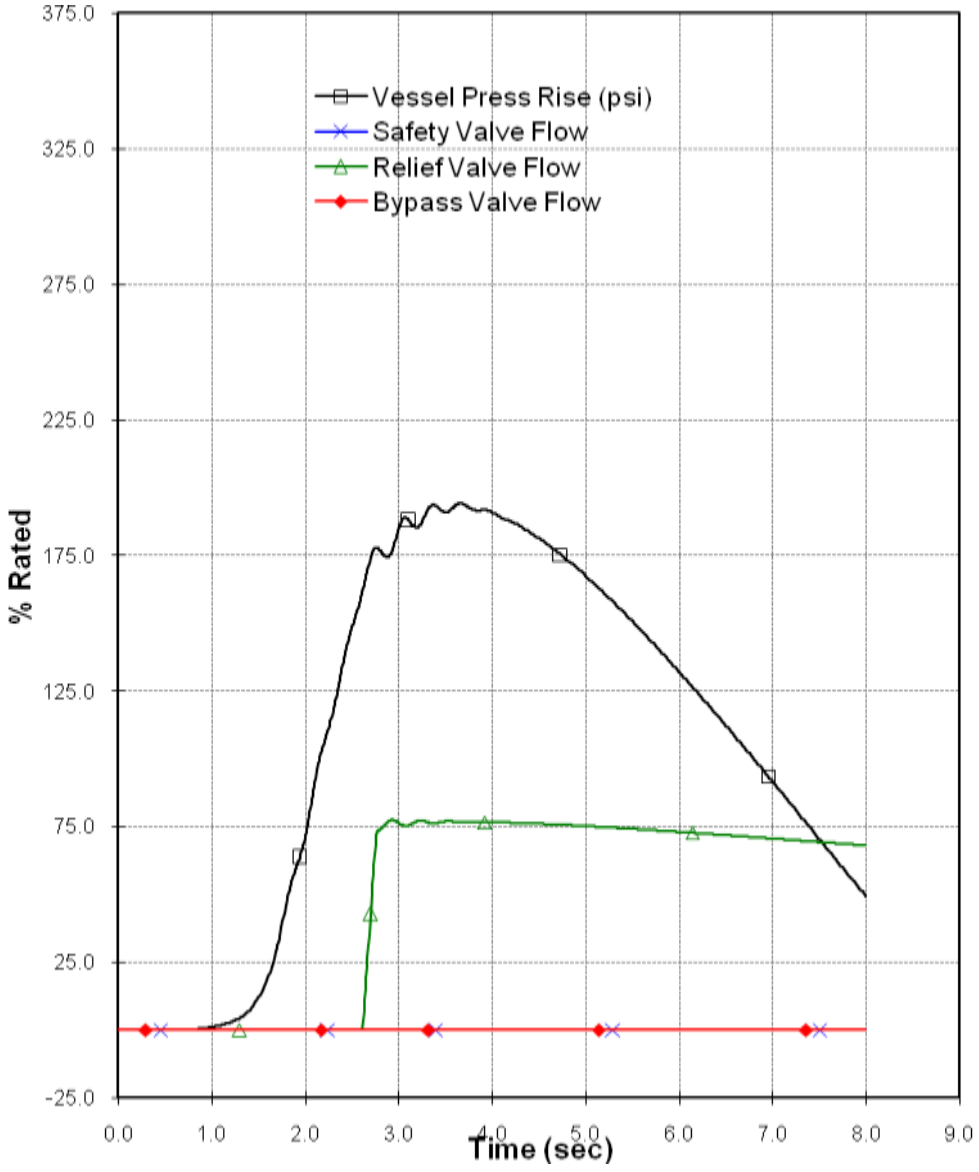

Figure 2-9. Response to MSIVF of system pressure rise and valve flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

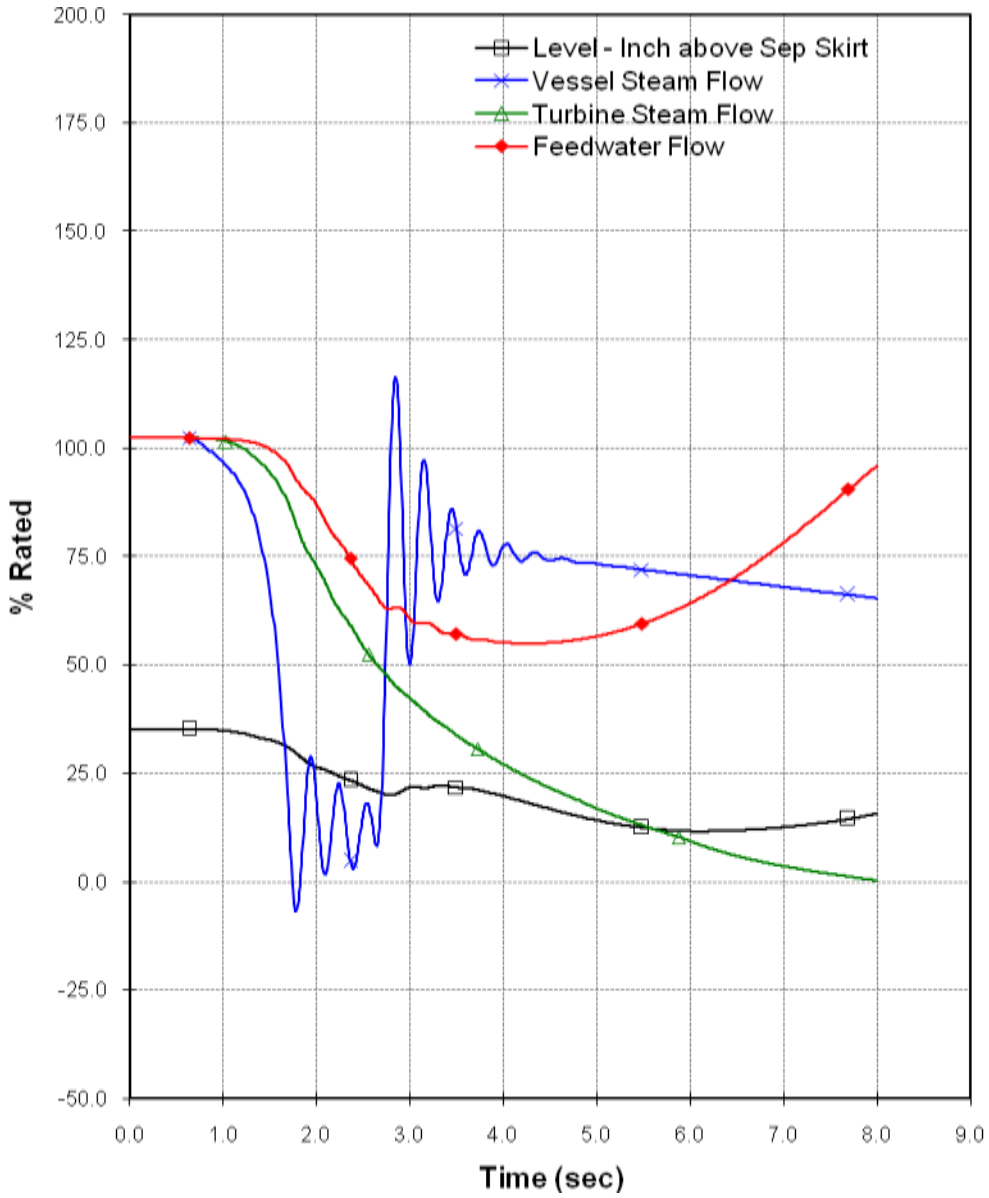

Figure 2-10. Response to MSIVF of system flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

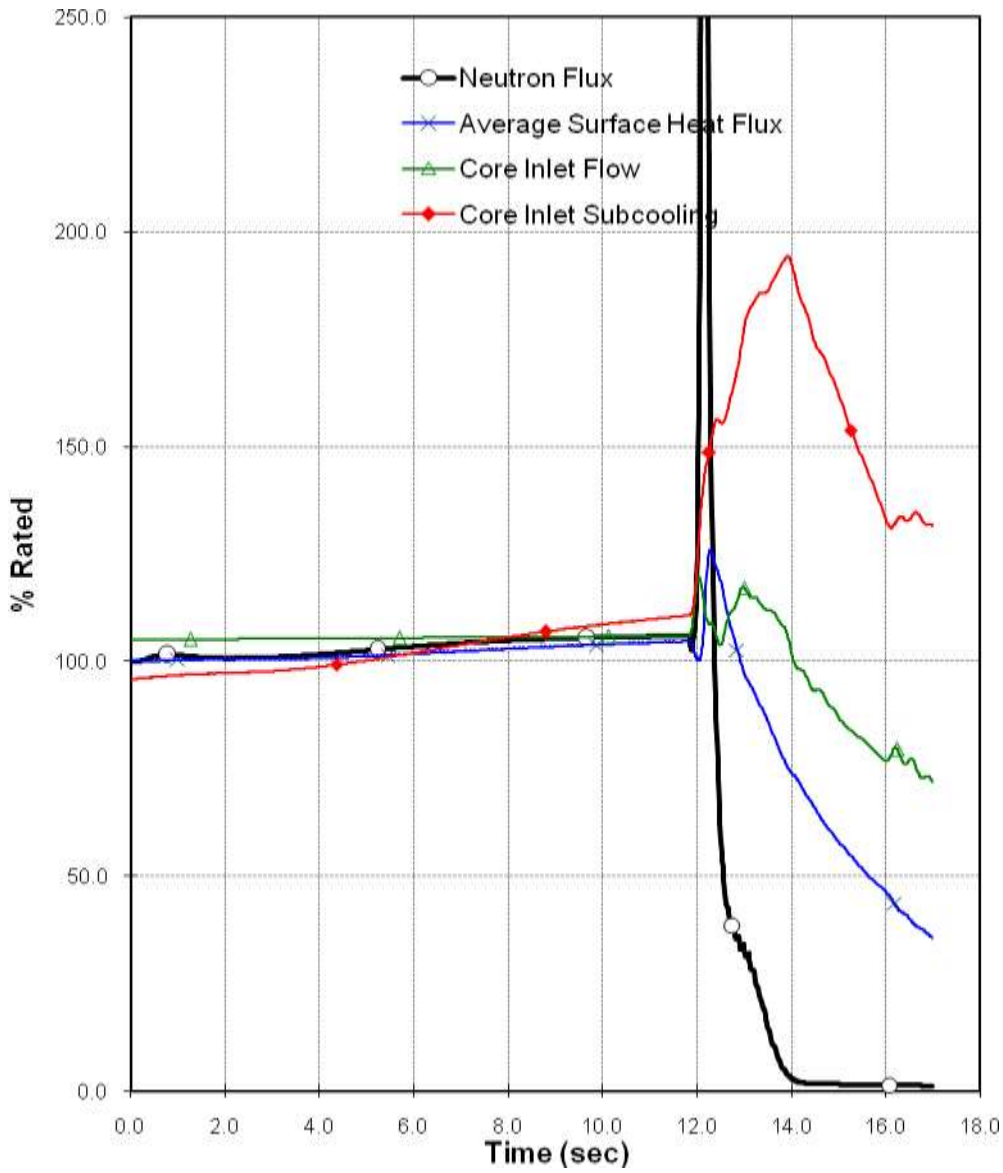

Figure 2-11. Response to FWCF with bypass of system flux, core flow, and core inlet subcooling. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

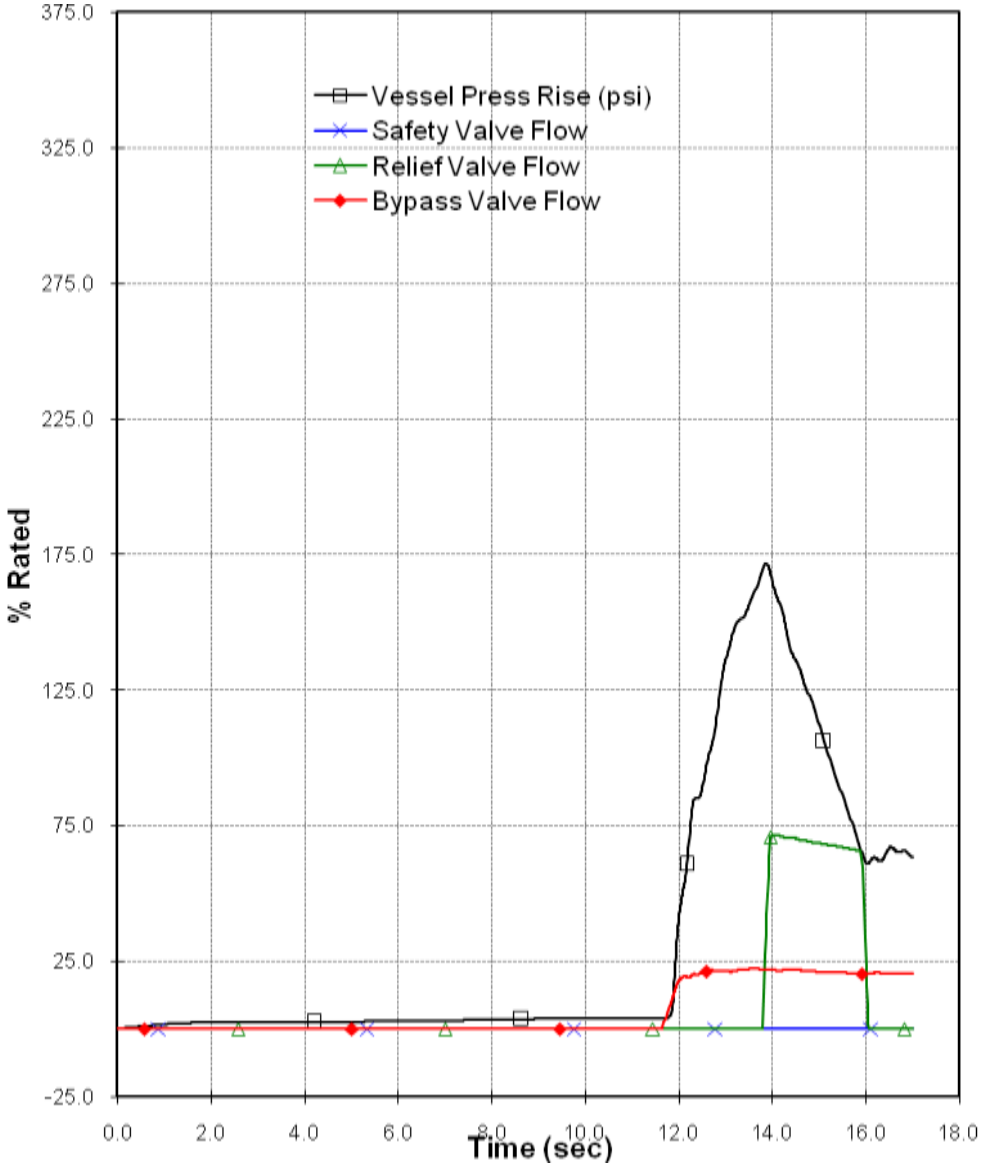

Figure 2-12. Response to FWCF with bypass of system pressure rise and valve flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

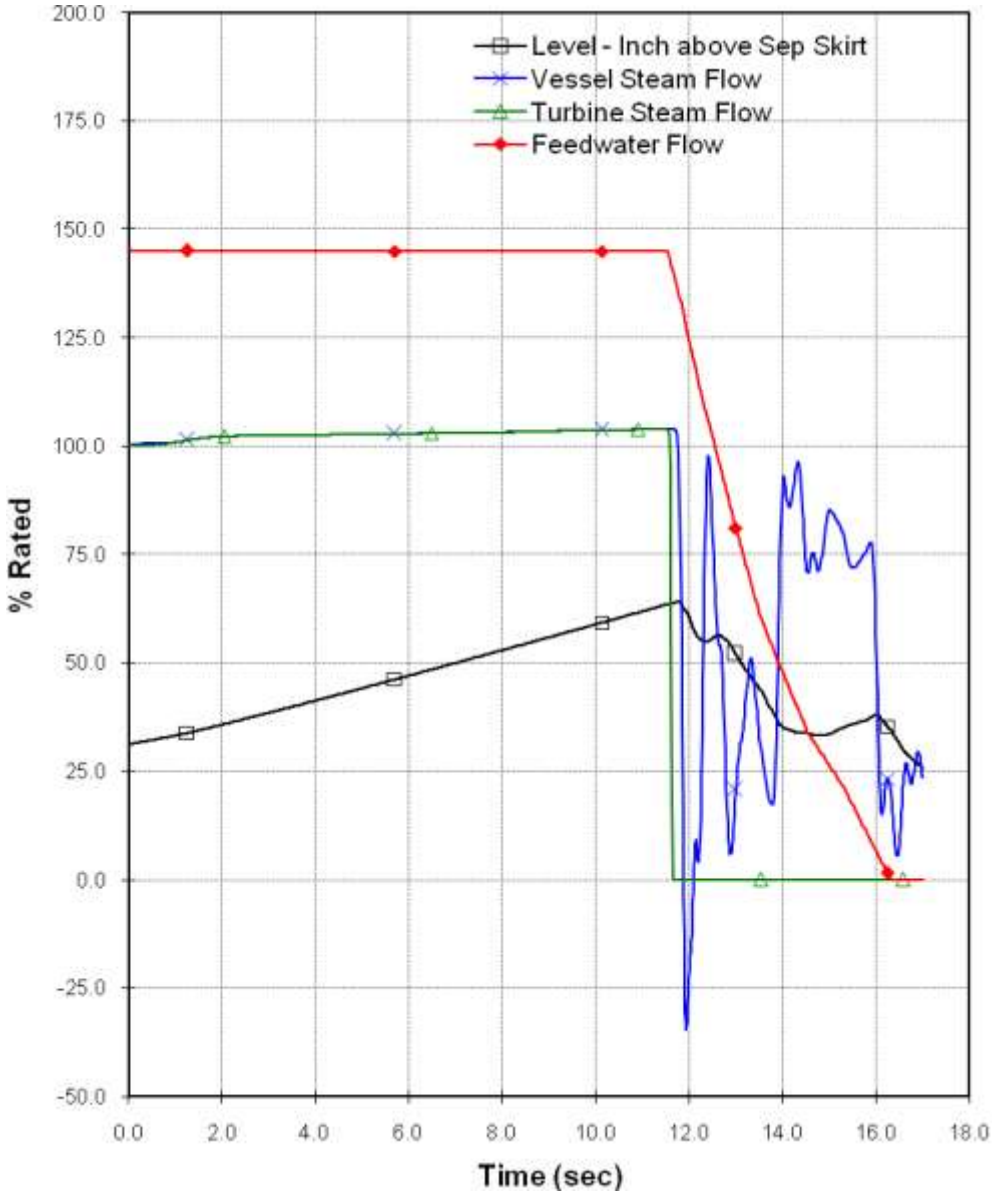

Figure 2-13. Response to FWCF with bypass of system flow. Data are from the BWR/4 -500 MWd/ST, ICF, NFW case.

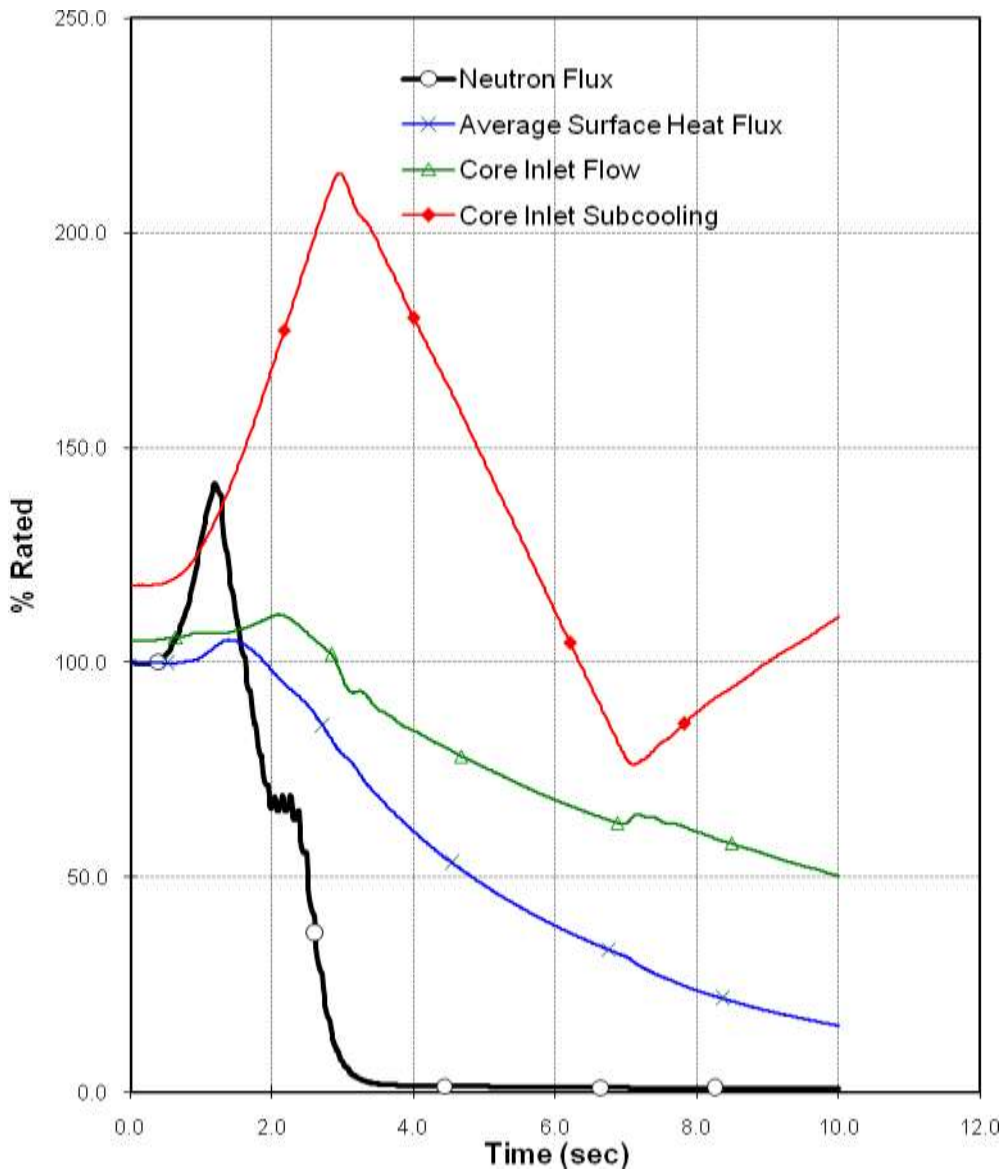

Figure 2-14. Response to PRFDS of system flux and core flow. Data are from the BWR/6 -500 MWd/ST, ICF, NFW case.

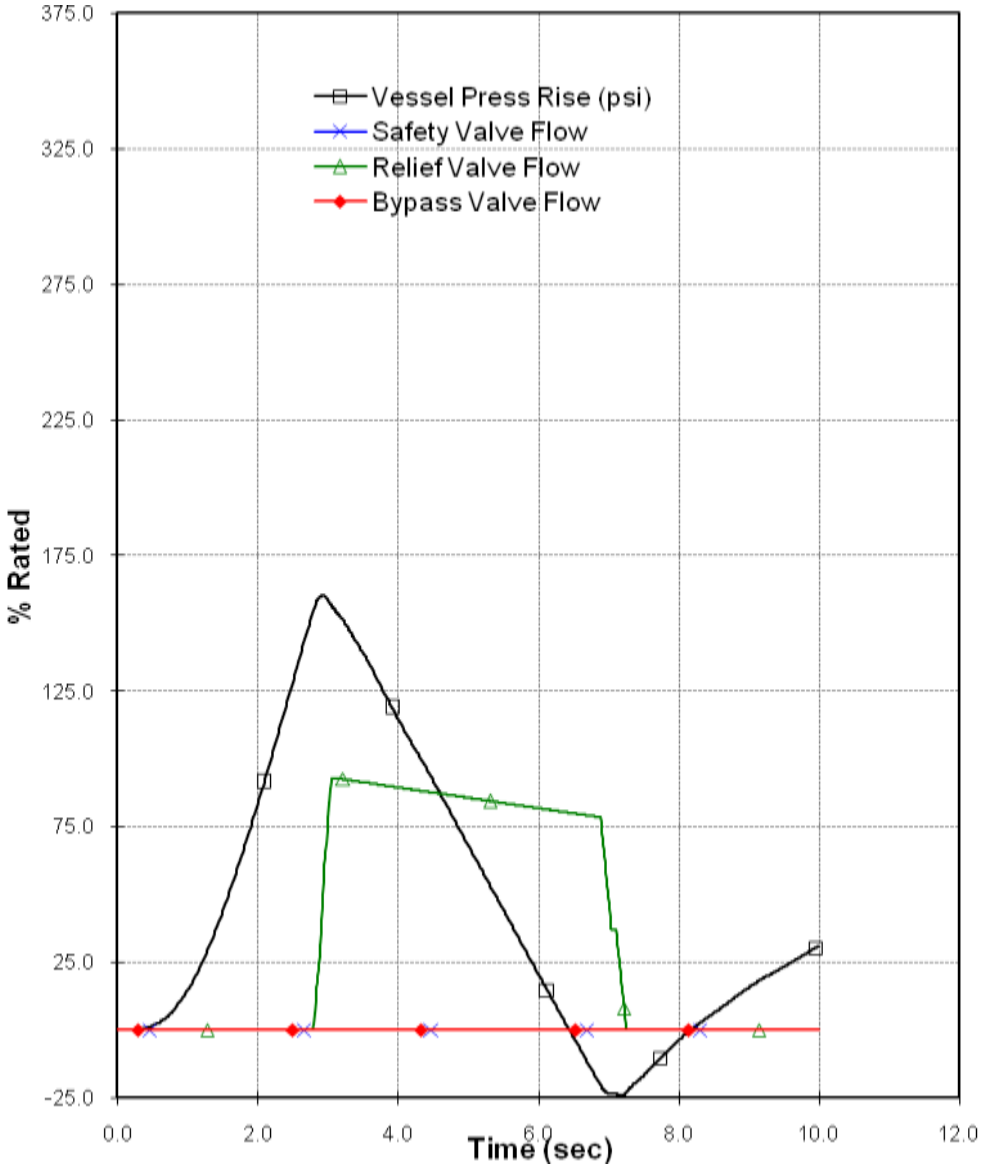

Figure 2-15. Response to PRFDS of system pressure rise and valve flow. Data are from the BWR/6 -500 MWd/ST, ICF, NFW case.

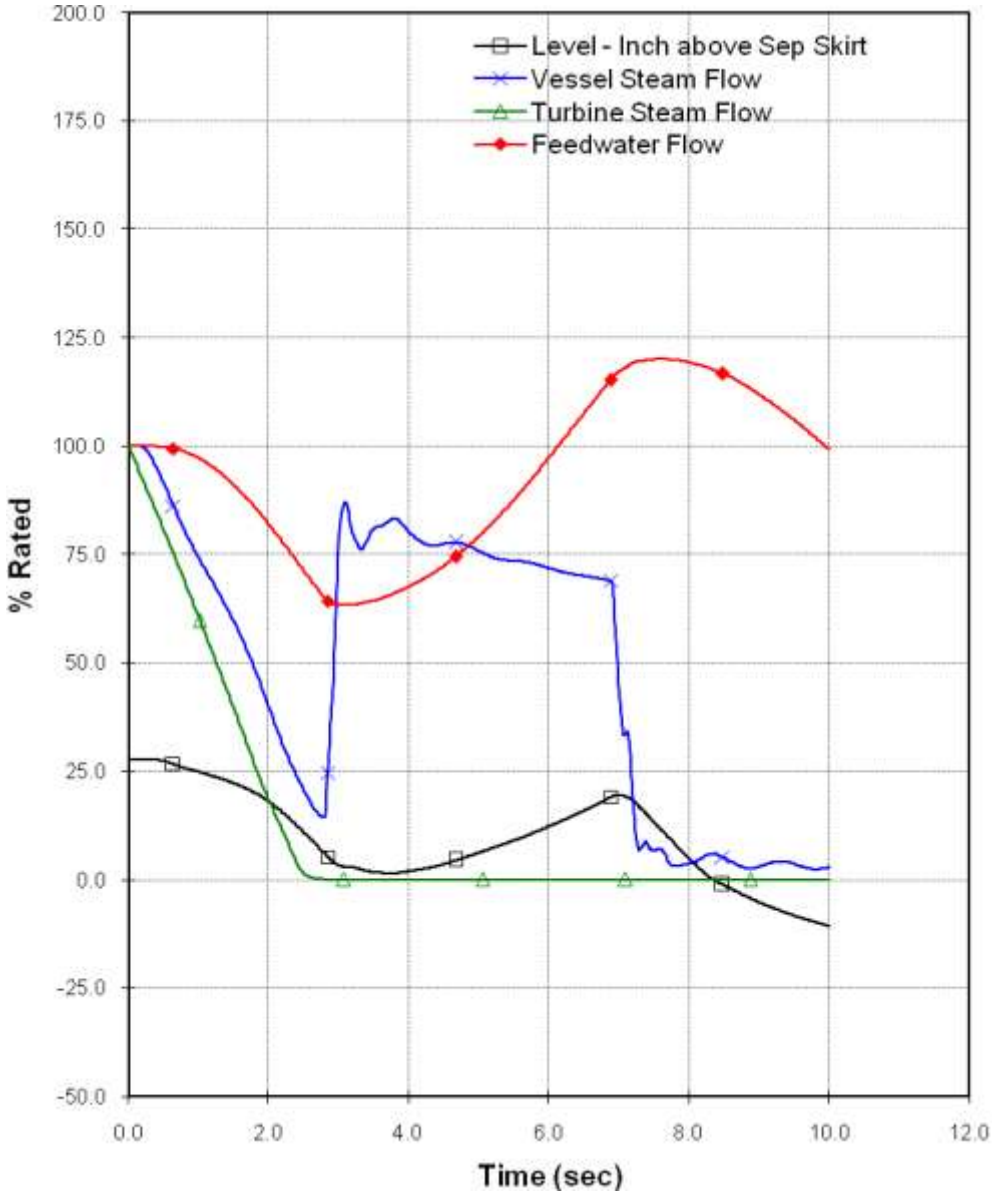

Figure 2-16. Response to PRFDS of system flow. Data are from the BWR/6 -500 MWd/ST, ICF, NFW case.

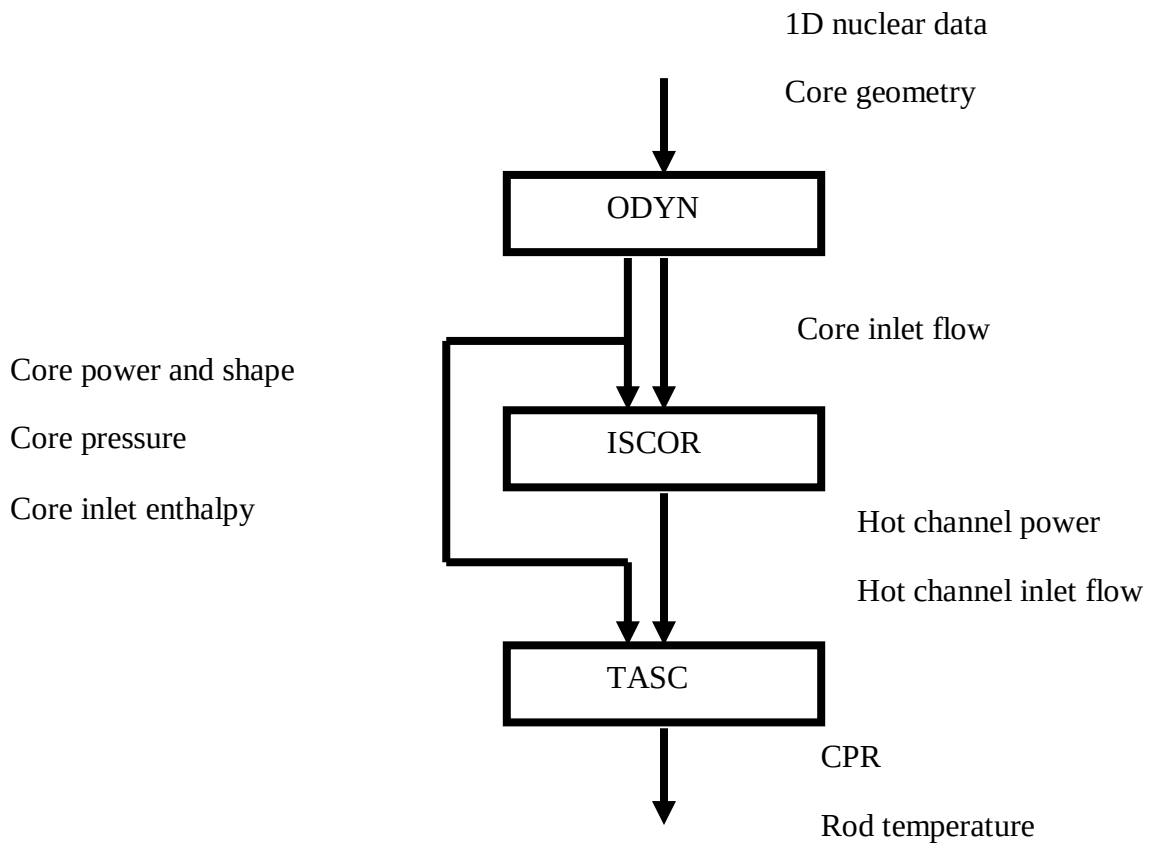

Figure 2-17. Transient methodology. Inputs and outputs associated with ODYN, ISCOR, and TASC (Lamb, 2007)

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CHAPTER 3
RESULTS

The domains analyzed could not be compared directly; it was not an apples-to-apples comparison because the projected end of cycle (PEOC) , or nominal shutdown exposure, cases were bounding hard bottom burns (HBB), while the varied exposure cases were nominal burns. The intent was to show that the HBB analyzed in the original/licensing transient analysis, using the projected end of cycle, would bound any nominal burns that arise from exposure shortfalls or extensions.

Safety Results of Interest

Delta Critical Power Ratio

Delta critical power ratio (ΔCPR) was the value measured during the transients; it was a measure of the amount the critical power ratio drops between the initial CPR and the Minimum CPR (MCPR) of the transient. Because each transient in a domain generally begins at the same initial CPR (ICPR), ΔCPR was an easy way to measure which transient was most severe in terms of MCPR.

The transients analyzed to determine ΔCPR were LRNBP, TTNBP, and FWCF (both with and without bypass). Figure 3-1 shows a comparison of critical power ratio versus time for the BWR/6 LRNBP. All of the domains had similar trend for CPR versus time. Figure 3-2 is a comparison of the BWR/6's ΔCPR values by transient and domain. It can be seen that the increased core flow region had higher ΔCPR values than the low core flow region, and that PRFDS and FWCF had lower ΔCPR values than LRNBP and TTNBP.

Thermal and Mechanical Over-Power

Thermal over power and MOP were values that expressed the percentage above the steady state fuel thermal and mechanical limits. These values used were screening criteria for determining margins to the overpower limits.

Vessel Overpressure

To prevent vessel overpressure, the peak vessel pressure was found and checked to see if it exceeded the ASME limit for vessel pressure. This was important because an overpressure event could cause failure of the reactor vessel.

The BWR/4

Though the BWR/4 core contained both GE14 and GNF2 Advantage fuel bundles, only the GE14 results are reported below. The GNF2 Advantage results required further sensitivity studies to determine the impact of shutdown exposure variation on the safety factors.

Delta Critical Power Ratio

The most limiting domain for ΔCPR was the increased core flow, normal feedwater temperature domain. The results are broken down by transient below. The results were split into EOC and MOC results. The EOC results were further split into with and without bypass. Table 3-1 shows the corrected EOC ΔCPR results for GE14 fuel and Table 3-2 shows the corrected MOC ΔCPR results.

Load Rejection, No Bypass

The GE14 ΔCPR values for the extended N-1 cycles were larger than the nominal cycle; however, these differences were less than 0.01 and thus not significant. The sensitivity was very small, on the order of 0.003 and is considered an equivalent result.

Figure 3-3 shows the ΔCPR values by domain and case for LRNBP. The increased core flow domain was the most limiting domain for all cases.

Turbine Trip, No Bypass

The TTNBP ΔCPR results were less limiting than the LRNBP results. The same trends were seen in the TTNBP results as were seen in the LRNBP results – the increased core flow domain was most limiting and the extended exposure cases had larger values for ΔCPR than the exposure cases that fell short of the nominal exposure. The GE14 values of the varied exposure cases were not significantly larger than those of the nominal case, and were smaller than the LRNBP ΔCPR nominal limiting values. The sensitivity was very small, on the order of 0.003 and is considered an equivalent result. Figure 3-4 shows the ΔCPR values by domain and case for TTNBP.

Feedwater Control Failure

The FWCF transients in the nominal case had several challenges by the varied exposure cases for the most limiting ΔCPR values. Once again, the ICF, NFW domain had the most limiting values. The ΔCPR values of the –1000 MWd/ST, +500 MWd/ST, and +1000 MWd/ST cases were all within 0.01of that of the nominal, a trend that was also apparent in the ICF, RFW domain. The sensitivity was very small, on the order of 0.003 and is considered an equivalent result. The FWCF/NBP cases in general had larger ΔCPR values than those with bypass operable. In both the ICF, NFW and ICF, RFW domains, the three exposure cases mentioned previously were very close to the limiting values set by the nominal case.

Thermal and Mechanical Over-Power

In terms of overpowers, the ICF, RFW domain was more limiting for GE14 thermal overpower (TOP), while the ICF, NFW was more limiting for mechanical overpower (MOP). Table 3-3 shows the margin to the screening criterion for thermal overpower for GE14 and Table 3-4 shows the margin to the screening criterion for mechanical overpower for GE14. The results are broken down by transient below.

Load Rejection, No Bypass

For the LRNBP transient, the nominal, increased core flow domain had the smallest margins to the TOP and MOP limits. For all transients, the nominal exposure case was the most limiting for both TOP and MOP. For TOP, the EOC nominal exposure case had a margin of 13.5% to the screening criterion. The closest TOP value was 14.5% in the EOC +1000 MWd/ST exposure case. For MOP, the EOC nominal shutdown exposure domain had the smallest margin of 13.5%, and was the most limiting for all transients. The next smallest margin was 14.5% in the EOC +1000 MWd/ST case.

Turbine Trip, No Bypass

The TTNBP transient TOP and MOP results were similar to those of LRNBP. Once again, the EOC nominal, increased core flow domain had the smallest margins to the TOP and MOP screening criteria. For TOP, the nominal exposure case had the smallest margin of 14.1%; the next smallest was in the +1000 MWd/ST case, which had a margin of 15.0% to the screening criterion. For MOP, the nominal exposure domain had a margin of 14.1% to the screening criterion. The next closest case was the EOC +1000 MWd/ST case, which had a margin of 15.0%.

Feedwater Control Failure

For the FWCF transient, much smaller margins were observed for TOP and MOP. The EOC nominal exposure case had the smallest margins to the screen criterion for both TOP and MOP, with values of 11.7% for both for FWCF and 6.4% for both for FWCF/NBP.

Vessel Over-Pressure

Analyzed at end of cycle, the nominal case had the most limiting values in terms of vessel over-pressure. The MSIVF transient traditionally has the smallest margin to the ASME vessel pressure limit. For increased core flow, the peak dome pressure was 1301 psig, less than the

limit of 1325 psig. The peak vessel pressure at increased core flow was 1337 psig, less than the limit of 1375 psig. For low core flow operation, the peak dome pressure was 1298 and the peak vessel pressure was 1327. Therefore, the varied exposure cases did not challenge the ASME pressure limit. Table 3-5 shows the peak vessel pressure results for the BWR/4.

The BWR/6

Delta Critical Power Ratio

The most limiting domain for ΔCPR was the increased core flow, normal feedwater temperature domain. The results are broken down by transient below. Table 3-6 shows the corrected GE14 ΔCPR values.

Load Rejection, No Bypass

Only one of the varied exposure cases challenged the ΔCPR set by the nominal case during a LRNBP; this was again in the increased core flow domain. However, the ΔCPR of the –1000 MWd/ST case was only 0.003 greater than that of the nominal, which in normal transient analysis was not considered a significant difference. The other varied exposure cases showed improvement over the nominal ΔCPR; the most improvement was shown by the +1000 MWd/ST case, which was 0.046 less than that of the nominal. Figure 3-6 shows the ΔCPR values by case and domain for LRNBP. It can be seen that the ICF, NFW domain had higher overall ΔCPR values.

In the low core flow domain, the +1000 MWd/ST case showed a 0.101 improvement over the nominal case. All of the varied exposure cases showed improvement over the nominal in the low core flow domain.

Turbine Trip, No Bypass

Because the TTNBP transient is very similar to the LRNBP, it was expected that those results would follow the same trends as the LRNBP ones. This proved true; only the –1000

MWd/ST case came close to challenging the ΔCPR set by the nominal case, but did not exceed it. The increased core flow domain was the more limiting of the two domains, and as the N-1 EOC shutdown exposure decreased, the ΔCPR increased.

Feedwater Control Failure

The FWCF transient in the nominal case was the most limiting for the ΔCPR compared to the other burn strategies. Once again the increased core flow, normal feedwater temperature case was the most limiting, followed by the increased core flow, reduced feedwater temperature case. The –1000 MWd/ST case came close to challenging the nominal case in both the ICF, NFW and ICF, RFW domains, but did not exceed the ΔCPR. The low core flow, normal feedwater temperature ΔCPR values had comfortable margin in the varied exposure cases to the nominal case ΔCPR.

Pressure Regulator Failure, Downstream

The PRFDS transient only occurred in two domains: increased core flow, normal feedwater temperature, and increased core flow, reduced feedwater temperature. Both of these domains caused the varied exposure cases to challenge the nominal case during other transients and this trend continued during the PRFDS transients. For the increased core flow, reduced feedwater temperature, the –1000 MWd/ST case was 0.005 greater than the nominal case. It was not a significant difference. In the increased core flow, reduced feedwater temperature domain, both of the smaller exposure cases challenged the nominal case. While the differences were not significant (0.005, 0.002), the difference was shown. In both domains, the cases not noted had large margins to the nominal ΔCPR. Note that the PRFDS is not close to the limiting TTNBP event.

Thermal and Mechanical Over-Power

The increased core flow, reduced feedwater temperature domain was the most limiting for both thermal and mechanical overpower. Table 3-7 shows the margin to the screening criterion for thermal overpower for GE14 and Table 3-8 shows the margin to the screening criterion for mechanical overpower. The results are broken down by transient below.

Load Rejection, No Bypass

For LRNBP, the increased core flow domain for the nominal case was the most limiting. For TOP, there was a margin of 34.5%, and for MOP, a margin of 34.2%. The case with the closest margins to the nominal case was the increased core flow, normal feedwater temperature, -1000 MWd/ST case, which had a TOP margin of 34.7% and a MOP margin of 34.7%.

Turbine Trip, No Bypass

The TTNBP transient showed the same trends as the LRNBP. The TOP margin for the increased core flow, nominal domain was 33.6% and the MOP margin was 33.3%. The case with the closet margins was the increased core flow, -1000 MWd/ST case, which had margins about 1% greater.

Feedwater Control Failure

The case with the smallest margin during a FWCF transient was the nominal case, increased core flow and reduced feedwater temperature. The TOP margin was 31.8% and the MOP margin was 27.8%. The closest margins were of the +500 MWd/ST case, increased core flow and reduced feedwater temperature domain. The TOP margin for that domain was 32.1% and the MOP margin was 29.5%. .

Pressure Regulator Failure, Downstream

The most limiting case in terms of TOP and MOP for the PRFDS transient was the –1000 MWd/ST, increased core flow and reduced feedwater temperature, however there are still large

margins for this event. The TOP margin was 17.2% and the MOP margin was 38.7%. The closest margins were those of the –500 MWd/ST, increased core flow and reduced feedwater temperature; the TOP margin was 20.0% and the MOP margin was 41.2%. The nominal case, increased core flow and reduced feedwater temperature domain had a TOP margin of 22.9% and a MOP margin of 44.3%.

Vessel Over-Pressure

For the increased core flow, normal feedwater temperature and the low core flow, normal feedwater temperature domains, the nominal case had the most limiting peak dome and vessel pressures. For the ICF, NFW domain, the nominal case had a peak dome pressure of 1271 psig and a peak vessel pressure of 1299. For the LCF, NFW domain, the nominal case had a peak dome pressure of 1269 psig and a peak vessel pressure of 1290 psig. All of those values are comfortably below the limits of 1325 psig for peak dome pressure and 1375 psig for peak vessel pressure. The case that came closest to the nominal case for both domains was the -500 MWd/ST case, which was still had peak pressures several psi below the nominal values reported above. Table 3-9 shows the peak vessel pressures for the BWR/6.

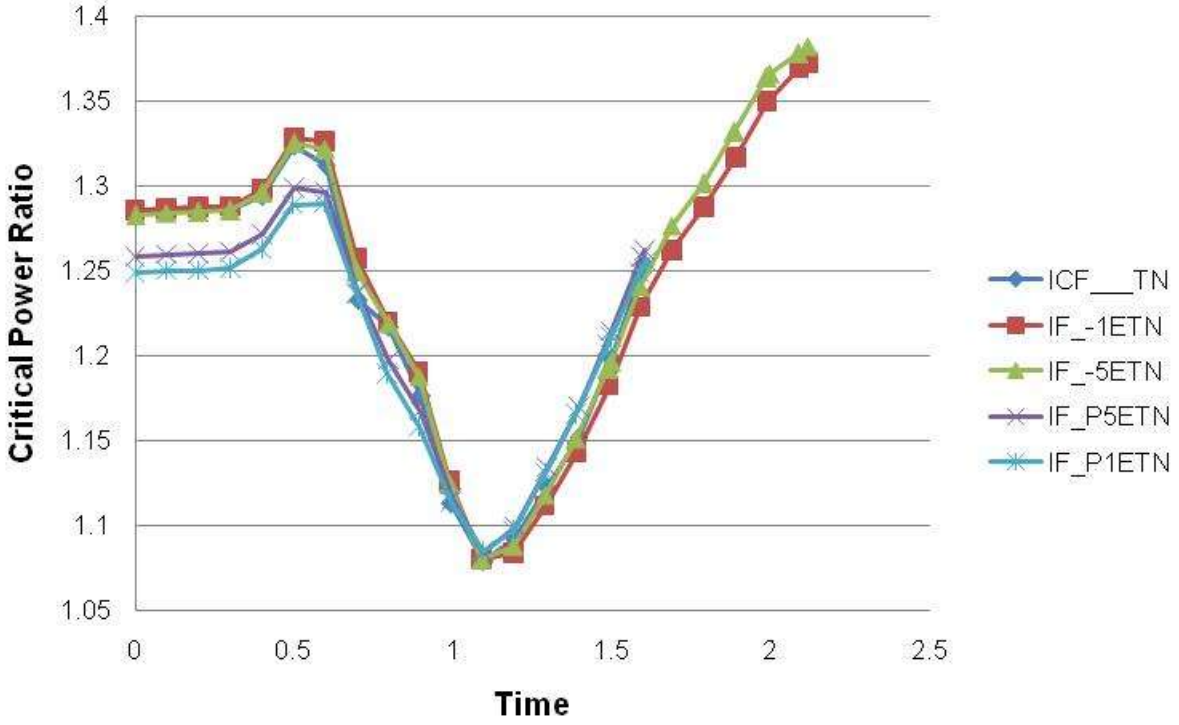

Figure 3-1. Critical power ratio versus time for BWR/6 LRNBP.

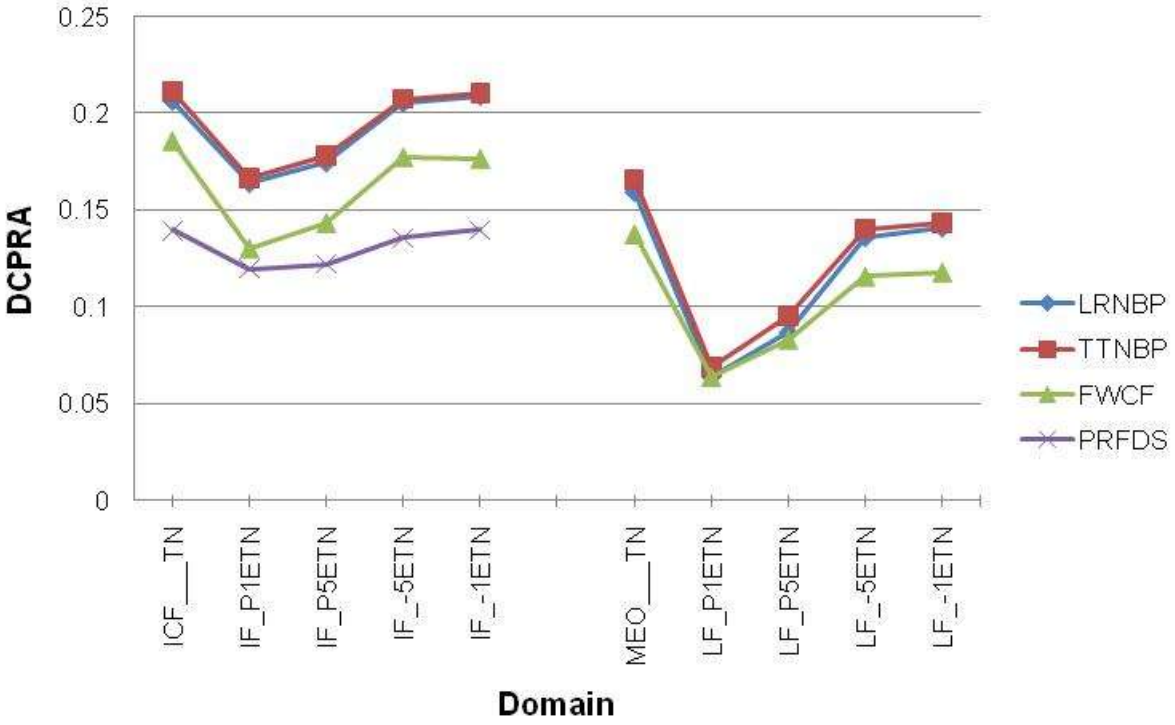

Figure 3-2. Comparison of BWR/6 ΔCPR values by transient and domain.

Note: The following describes the domain labels for the chart above.

Domain	Description
ICF___TN	Increased Core Flow, Hard Bottom Burn Licensing power shape
IF_P1ETN	Increased Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
IF_P5ETN	Increased Core Flow, N-1 Exposure +500, EOC, normal FW temp.
IF_-5ETN	Increased Core Flow, N-1 Exposure -500, EOC, normal FW temp.
IF_-1ETN	Increased Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
MEO___TN	MELLLA (Low) Core Flow, Hard Bottom Burn Licensing power shape
LF_P1ETN	Low Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
LF_P5ETN	Low Core Flow, N-1 Exposure +500, EOC, normal FW temp.
LF_-5ETN	Low Core Flow, N-1 Exposure -500, EOC, normal FW temp.
LF_-1ETN	Low Core Flow, N-1 Exposure -1000, EOC, normal FW temp.

Table 3-1. The BWR/4 EOC GE14 corrected ΔCPR results.

GE14 corrected ΔCPR		EOC18 -1000 MWd/ST C19 nominal	EOC18 -500 MWd/ST C19 nominal	EOC18 PEOC C19 HBB	EOC18 +500 MWd/ST C19 nominal	EOC18 +1000 MWd/ST C19 nominal
LRNBP	ICF	0.359	0.327	0.358	0.360	0.361
TTNBP	ICF	0.355	0.323	0.353	0.354	0.356
FWCF	ICF	0.356	0.327	0.353	0.354	0.354
FWCF/NBP	ICF	**0.395**	0.372	0.392	0.394	0.393
LRNBP	LCF	0.306	0.238	0.317	0.308	0.306
TTNBP	LCF	0.302	0.237	0.313	0.300	0.303
FWCF	LCF	0.298	0.239	0.309	0.296	0.298
FWCF/NBP	LCF	0.343	0.281	0.357	0.340	0.343
FWCF	ICF, RFW	0.343	0.336	0.340	0.344	0.342
FWCF/NBP	ICF, RFW	0.382	0.375	0.381	0.384	0.383
FWCF	LCF, RFW	0.298	0.262	0.303	0.298	0.299
FWCF/NBP	LCF, RFW	0.339	0.318	0.345	0.339	0.339

Table 3-2. The BWR/4 MOC GE14 corrected ΔCPR results.

GE14 corrected ΔCPR		EOC18 -1000 MWd/ST C19 nominal	EOC18 -500 MWd/ST C19 nominal	EOC18 PEOC C19 HBB	EOC18 +500 MWd/ST C19 nominal	EOC18 +1000 MWd/ST C19 nominal
LRNBP	ICF, MOC	0.309	0.253	**0.321**	0.310	0.308
TTNBP	ICF, MOC	0.306	0.250	0.315	0.306	0.304
FWCF	ICF, MOC	0.308	0.253	0.319	0.308	0.306
LRNBP	LCF, MOC	0.251	0.187	0.270	0.250	0.251
TTNBP	LCF, MOC	0.248	0.186	0.266	0.249	0.249
FWCF	LCF, MOC	0.247	0.187	0.262	0.241	0.243

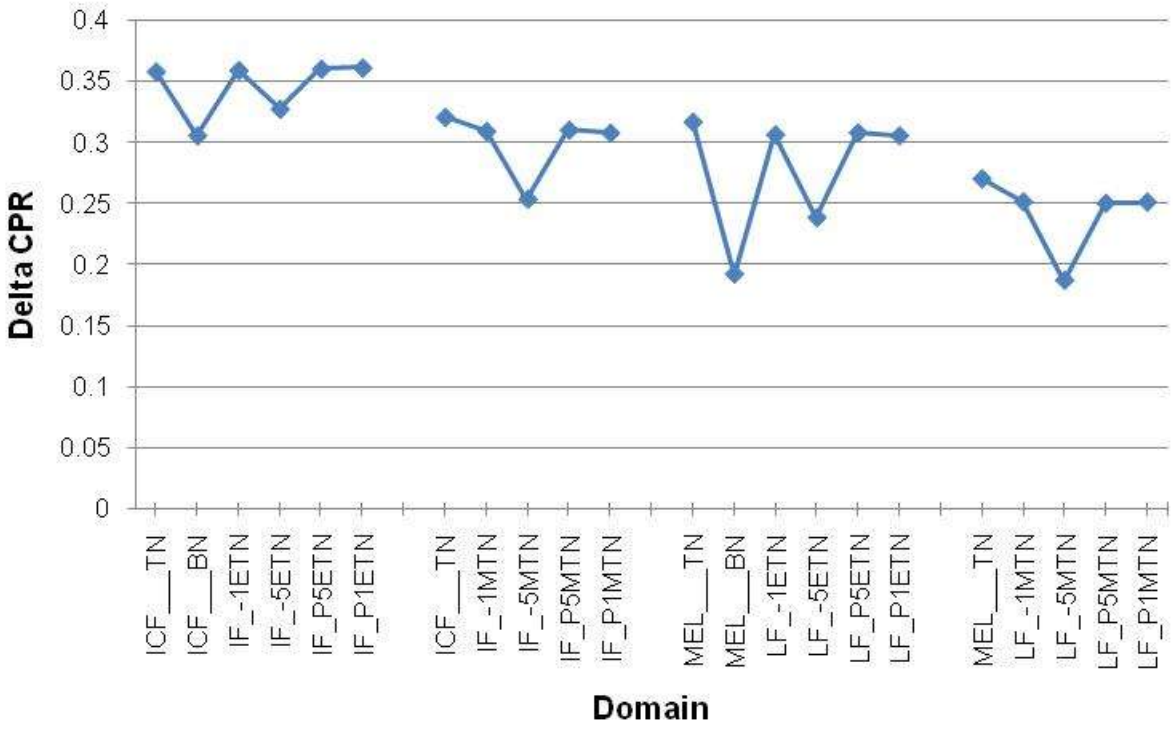

Figure 3-3. The LRNBP ΔCPR by domain and case for the BWR/4. The following describes the domain labels for the chart above.

Domain	Description
ICF___TN	Increased Core Flow, Hard Bottom Burn Licensing power shape
ICF___BN	Increased Core Flow, Under-Burn Licensing power shape
IF_P1ETN	Increased Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
IF_P5ETN	Increased Core Flow, N-1 Exposure +500, EOC, normal FW temp.
IF_-5ETN	Increased Core Flow, N-1 Exposure -500, EOC, normal FW temp.
IF_-1ETN	Increased Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
IF_P1MTN	Increased Core Flow, N-1 Exposure +1000, MOC, normal FW temp.
IF_P5MTN	Increased Core Flow, N-1 Exposure +500, MOC, normal FW temp.
IF_-5MTN	Increased Core Flow, N-1 Exposure -500, MOC, normal FW temp.
IF_-1MTN	Increased Core Flow, N-1 Exposure -1000, MOC, normal FW temp.
MEL___TN	MELLLA (Low) Core Flow, Hard Bottom Burn Licensing power shape
MEL___BN	MELLLA (Low) Core Flow, Under-Burn Licensing power shape
LF_P1ETN	Low Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
LF_P5ETN	Low Core Flow, N-1 Exposure +500, EOC, normal FW temp.
LF_-5ETN	Low Core Flow, N-1 Exposure -500, EOC, normal FW temp.
LF_-1ETN	Low Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
LF_P1MTN	Low Core Flow, N-1 Exposure +1000, MOC, normal FW temp.
LF_P5MTN	Low Core Flow, N-1 Exposure +500, MOC, normal FW temp.
LF_-5MTN	Low Core Flow, N-1 Exposure -500, MOC, normal FW temp.
LF_-1MTN	Low Core Flow, N-1 Exposure -1000, MOC, normal FW temp.

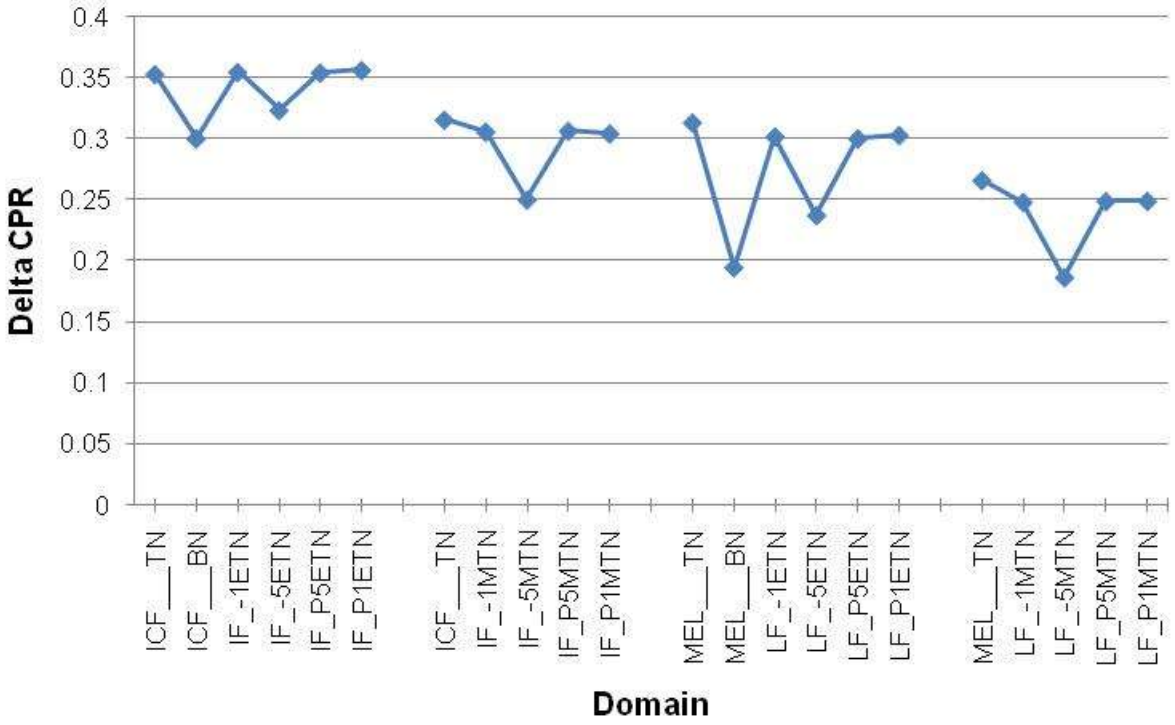

Figure 3-4. The TTNBB ΔCPR by domain and case for the BWR/4. The following describes the domain labels for the chart above.

Domain	Description
ICF___TN	Increased Core Flow, Hard Bottom Burn Licensing power shape
ICF___BN	Increased Core Flow, Under-Burn Licensing power shape
IF_P1ETN	Increased Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
IF_P5ETN	Increased Core Flow, N-1 Exposure +500, EOC, normal FW temp.
IF_-5ETN	Increased Core Flow, N-1 Exposure -500, EOC, normal FW temp.
IF_-1ETN	Increased Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
IF_P1MTN	Increased Core Flow, N-1 Exposure +1000, MOC, normal FW temp.
IF_P5MTN	Increased Core Flow, N-1 Exposure +500, MOC, normal FW temp.
IF_-5MTN	Increased Core Flow, N-1 Exposure -500, MOC, normal FW temp.
IF_-1MTN	Increased Core Flow, N-1 Exposure -1000, MOC, normal FW temp.
MEL___TN	MELLLA (Low) Core Flow, Hard Bottom Burn Licensing power shape
MEL___BN	MELLLA (Low) Core Flow, Under-Burn Licensing power shape
LF_P1ETN	Low Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
LF_P5ETN	Low Core Flow, N-1 Exposure +500, EOC, normal FW temp.
LF_-5ETN	Low Core Flow, N-1 Exposure -500, EOC, normal FW temp.
LF_-1ETN	Low Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
LF_P1MTN	Low Core Flow, N-1 Exposure +1000, MOC, normal FW temp.
LF_P5MTN	Low Core Flow, N-1 Exposure +500, MOC, normal FW temp.
LF_-5MTN	Low Core Flow, N-1 Exposure -500, MOC, normal FW temp.
LF_-1MTN	Low Core Flow, N-1 Exposure -1000, MOC, normal FW temp.

Table 3-3. The BWR/4 GE14 thermal overpower results: margin to screening criteria.

GE14 TOP margin (%)		EOC18 -1000 MWd/ST C19 nominal	EOC18 -500 MWd/ST C19 nominal	EOC18 PEOC C19 HBB	EOC18 +500 MWd/ST C19 nominal	EOC18 +1000 MWd/ST C19 nominal
LRNBP	ICF	14.8	19.1	13.5	14.6	14.5
TTNBP	ICF	15.3	19.7	14.1	15.5	15
FWCF	ICF	13.4	16.6	11.7	13.4	13.4
FWCF/NBP	ICF	8.1	11.1	**6.4**	8.1	8.1
LRNBP	LCF	23.4	29.5	21.5	23.7	24
TTNBP	LCF	24.2	29.8	21.9	24.9	24.9
FWCF	LCF	22.2	27.5	19.7	23.1	22.9
FWCF/NBP	LCF	17.2	22.5	13.9	18.1	17.9
FWCF	ICF, RFW	13.7	14.7	12	13.5	13.4
FWCF/NBP	ICF, RFW	8.6	9.6	6.8	8.5	8.3
FWCF	LCF, RFW	19.8	22.4	17.4	20.1	19.7
FWCF/NBP	LCF, RFW	14.7	15.5	12.3	15.1	14.9
LRNBP	ICF, MOC	15.9	23.6	15.1	16.7	16.9
TTNBP	ICF, MOC	16	24.3	15.5	16.9	17.4
FWCF	ICF, MOC	13.2	21.6	13.3	14.3	14.3
LRNBP	LCF, MOC	25.4	31.8	23.3	26.7	25.9
TTNBP	LCF, MOC	26	32.1	24	27.1	26.1
FWCF	LCF, MOC	23.1	30.5	22.1	24.7	23.7

Table 3-4. The BWR/4 GE14 mechanical overpower results: margin to screening criteria.

GE14 MOP margin (%)		EOC18 -1000 MWd/ST C19 nominal	EOC18 -500 MWd/ST C19 nominal	EOC18 PEOC C19 HBB	EOC18 +500 MWd/ST C19 nominal	EOC18 +1000 MWd/ST C19 nominal
LRNBP	ICF	14.8	19.1	13.5	14.6	14.5
TTNBP	ICF	15.3	19.7	14.1	15.5	15
FWCF	ICF	13.4	16.6	11.7	13.4	13.4
FWCF/NBP	ICF	8.1	11.1	**6.4**	8.1	8.1
LRNBP	LCF	23.2	29.5	21.5	23.4	23.6
TTNBP	LCF	23.8	29.8	21.9	24.4	24.2
FWCF	LCF	20.6	27.5	19.7	21.4	20.9
FWCF/NBP	LCF	15.4	22.5	13.9	16.2	15.7
FWCF	ICF, RFW	13.5	14.7	12	13	12.8
FWCF/NBP	ICF, RFW	8.6	9.6	6.8	8.1	7.8
FWCF	LCF, RFW	17.6	22.4	17.4	18.1	17.6
FWCF/NBP	LCF, RFW	12.3	15.5	12.3	12.9	12.5
LRNBP	ICF, MOC	15.9	23.6	15.1	16.3	16.7
TTNBP	ICF, MOC	16	24.3	15.5	16.5	17.2
FWCF	ICF, MOC	13	21.6	11.9	13.8	14.3
LRNBP	LCF, MOC	24.2	31.7	22.9	25.2	25
TTNBP	LCF, MOC	24.8	31.8	23.3	25.6	25.3
FWCF	LCF, MOC	21.8	30.4	20.2	23.2	22.8

Table 3-5. The BWR/4 peak vessel pressure results.

Peak vessel pressure (psig)		EOC18 -1000 MWd/ST C19 nominal	EOC18 -500 MWd/ST C19 nominal	EOC18 PEOC C19 HBB	EOC18 +500 MWd/ST C19 nominal	EOC18 +1000 MWd/ST C19 nominal
MSIVF	ICF	1333.3	1325.7	**1336.8**	1333.7	1334.2
MSIVF	LCF	1322.2	1312.7	1327.3	1322.2	1322.3

Table 3-6. The BWR/6 GE14 corrected ΔCPR results.

GE14 corrected ΔCPR		EOC12 -1000 MWd/ST C13 nominal	EOC12 -500 MWd/ST C13 nominal	EOC12 PEOC C13 HBB	EOC12 +500 MWd/ST C13 nominal	EOC12 +1000 MWd/ST C13 nominal
LRNBP	ICF	0.246	0.242	0.243	0.209	0.197
TTNBP	ICF	0.247	0.244	**0.248**	0.212	0.200
FWCF	ICF	0.211	0.212	0.220	0.176	0.163
PRFDS	ICF	0.140	0.136	0.140	0.122	0.120
LRNBP	LCF	0.174	0.169	0.194	0.117	0.093
TTNBP	LCF	0.176	0.173	0.201	0.125	0.098
FWCF	LCF	0.149	0.147	0.169	0.113	0.093
FWCF	ICF/RFW	0.241	0.238	0.243	0.236	0.238
PRFDS	ICF/RFW	0.140	0.137	0.135	0.124	0.129

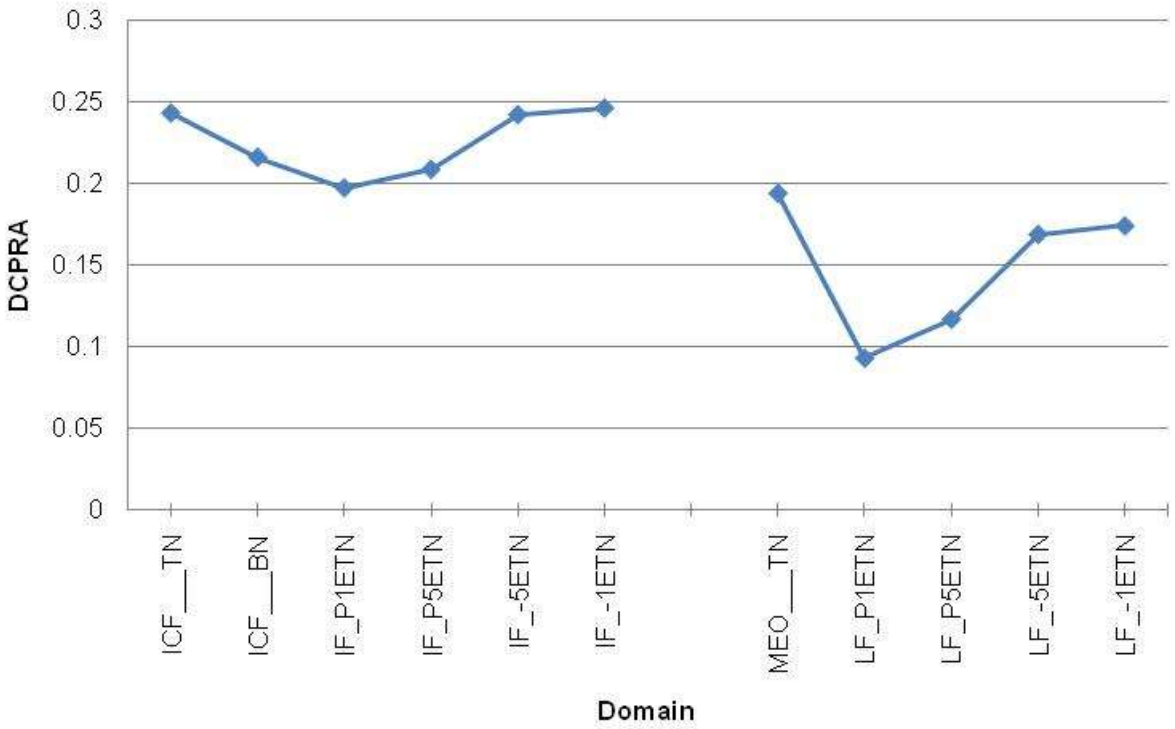

Figure 3-6. The LRNBP ΔCPR by domain and case for the BWR/6. The following describes the domain labels for the chart above.

Domain	Description
ICF___TN	Increased Core Flow, Hard Bottom Burn Licensing power shape
IF_P1ETN	Increased Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
IF_P5ETN	Increased Core Flow, N-1 Exposure +500, EOC, normal FW temp.
IF_-5ETN	Increased Core Flow, N-1 Exposure -500, EOC, normal FW temp.
IF_-1ETN	Increased Core Flow, N-1 Exposure -1000, EOC, normal FW temp.
MEO___TN	MELLLA (Low) Core Flow, Hard Bottom Burn Licensing power shape
LF_P1ETN	Low Core Flow, N-1 Exposure +1000, EOC, normal FW temp.
LF_P5ETN	Low Core Flow, N-1 Exposure +500, EOC, normal FW temp.
LF_-5ETN	Low Core Flow, N-1 Exposure -500, EOC, normal FW temp.
LF_-1ETN	Low Core Flow, N-1 Exposure -1000, EOC, normal FW temp.

Table 3-7. The BWR/6 thermal overpower results: margin to screening criteria.

GE14 TOP margin (%)		EOC12 -1000 MWd/ST C13 nominal	EOC12 -500 MWd/ST C13 nominal	EOC12 PEOC C13 HBB	EOC12 +500 MWd/ST C13 nominal	EOC12 +1000 MWd/ST C13 nominal
LRNBP	ICF	34.7	35.5	34.5	38	37.3
TTNBP	ICF	34.6	35.4	33.6	37.7	37
FWCF	ICF	38.2	38.5	35	41	41.2
PRFDS	ICF	20.9	23.7	22.6	24.4	24.8
LRNBP	LCF	42.4	43.2	41.7	50	52
TTNBP	LCF	41.5	42.4	40.8	48.9	52
FWCF	LCF	44.4	44.8	43.1	42.6	42.5
FWCF	ICF/RFW	36.2	34.4	31.8	32.1	37
PRFDS	ICF/RFW	**17.2**	20	22.9	22.1	21.5

Table 3-8. The BWR/6 mechanical overpower results: margin to screening criteria.

GE14 MOP Margin (%)		EOC12 -1000 MWd/ST C13 nominal	EOC12 -500 MWd/ST C13 nominal	EOC12 PEOC C13 HBB	EOC12 +500 MWd/ST C13 nominal	EOC12 +1000 MWd/ST C13 nominal
LRNBP	ICF	34.7	35.5	34.2	37.8	37
TTNBP	ICF	34.6	35.4	33.3	37.5	36.7
FWCF	ICF	38.2	38.5	34.7	40.9	41
PRFDS	ICF	42.1	44.8	43.9	45.3	45.7
LRNBP	MEL	42.2	43	41.7	47.4	49.1
TTNBP	MEL	41.3	42.2	40.8	46.4	48.8
FWCF	MEL	43.9	43.9	43.1	42.4	42.5
FWCF	ICF/RFW	32	30.1	**27.8**	29.5	32.2
PRFDS	ICF/RFW	38.7	41.2	44.3	42.4	42.1

Table 3-9. The BWR/6 peak vessel pressure results.

Peak vessel pressure (psig)		EOC12 -1000 MWd/ST C13 nominal	EOC12 -500 MWd/ST C13 nominal	EOC12 PEOC C13 HBB	EOC12 +500 MWd/ST C13 nominal	EOC12 +1000 MWd/ST C13 nominal
MSIVF	ICF	1295.3	1295.7	**1298.6**	1291.4	1290.4
MSIVF	LCF	1285.4	1285.9	1289.5	1279.9	1276.4

CHAPTER 4
CONCLUSIONS

This research, a more intensive study of the effect of previous cycle exposure than is performed in a normal reload licensing analysis , found that the current licensing process at GE Hitachi Nuclear for transient analysis is acceptably bounding. This result agreed with the initial hypothesis and the results from previous studies of smaller exposure windows.

For the BWR/4, the nominal exposure case had the most limiting over-pressure results. In addition, while some other cases had more limiting ΔCPR results than the projected HBB case, the differences were not significant. Likewise, the overpower results were not significantly affected and all cases showed acceptable margins to the screening criteria. In addition, the screening criteria are known to be conservative and there is additional margin to fuel melt and cladding strain criteria.

For the BWR/6, the variation in shutdown exposure did not seem to have a large impact on the operating limits determined using the projected HBB. Interestingly, for the BWR/6 the exposure cases that fell short of the projected end of cycle exposure were more limiting than those from the lengthened cycle exposure cases.

Previous studies have supported the ±500 MWd/ST exposure window. Overall, this study confirms that the ±500 MWd/ST exposure steps did not significantly challenge the limits, and therefore a change of fifteen days in the shutdown will be acceptably bounded by the transient analyses done for the reload core using the projected end of cycle data for plants using GE14 fuel. In addition, these results also support that the ± 1000 MWd/ST sensitivities are sufficiently small that the licensing results are acceptable. The licensee would continue exposure monitoring through Cycle N to assure compliance with the licensing basis power shape assumptions.

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BIOGRAPHICAL SKETCH

Lauren Elizabeth Nalepa was born in 1985 in Biloxi, Mississippi, to Robert and Lisa Nalepa. She has two younger siblings, Ryan and Emma. Lauren attended elementary school in Little Rock, Arkansas, and O'Fallon, Illinois, before moving to Okinawa, Japan. She is a 2003 graduate of Kadena High School on Kadena Air Base, Okinawa, Japan. A second-generation Gator, Lauren received her B.S. in nuclear engineering in May 2007 and began graduate school the next fall.

In school, Lauren was active in Phi Sigma Rho, Tau Chapter. She served as Vice President of Standards her junior and senior years. She was also secretary of the UF student chapter of the American Nuclear Society, and presented a paper at the 2006 ANS Student Conference at Rensselaer Polytechnic Institute and a poster at the 2007 ANS Winter Conference in Washington, D.C.

Lauren interned for six semesters while in school, at both Progress Energy Florida's Crystal River nuclear plant in Crystal River, Florida, and GE Hitachi Nuclear Energy in Wilmington, North Carolina. Upon completion of her degree, she will join GE Hitachi's Edison Engineering Development Program as a nuclear engineer in Wilmington, North Carolina.

Lauren enjoys traveling, reading, baking, tennis, and Disney World.