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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Supplemental Information Concerning License Amendment to Allow Ganged Rod Drive Capability of the Rod Control Management System

Reference: Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Allow Ganged Rod Drive Capability of the Rod Control Management System," dated August 14, 2007

In the referenced letter, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises the licensing basis to allow ganged rod drive capability of the Rod Control Management System (RCMS). This letter provides supplemental information concerning this license amendment request (LAR) in response to Requests for Additional Information (RAIs) that were transmitted by NRC emails dated January 2, January 24, and January 31, 2008, and clarified during teleconferences between EGC and the NRC on January 15, 2008 and April 10, 2008. The supplemental information is provided in the attachments to this letter, as follows:

- Attachment 1 provides a response to the NRC RAIs.
- Attachment 2 provides a Framatome ANP, Inc. evaluation supporting the proposed LAR.

There are no regulatory commitments contained in this letter.

If you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803 or Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of May 2008.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager - Licensing

Attachments:

1. Supplemental Information Concerning License Amendment to Allow Ganged Rod Drive Capability of the Rod Control Management System

2. Framatome ANP, Inc., "Low Power CRWE Evaluation for LaSalle," October 31, 2005

ATTACHMENT 1
Supplemental Information Concerning License Amendment to Allow Ganged Rod Drive Capability of the Rod Control Management System

Background

In e-mails dated January 2, January 24, and January 31, 2008, the NRC transmitted Requests for Additional Information (RAIs) to Exelon Generation Company, LLC (EGC) concerning a license amendment request (LAR) for LaSalle County Station (i.e., letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Allow Ganged Rod Drive Capability of the Rod Control Management System," dated August 14, 2007, hereafter described as the "referenced LAR"). These RAIs were discussed, and the questions were clarified during teleconferences between EGC and the NRC on January 15, 2008 and April 10, 2008.

The clarified NRC questions and the EGC response to each question are described below, as well as in an engineering evaluation that is provided in Attachment 2.

NRC Request 1

Part of the basis for ARTS-style credit of a reactivity management system is that the accident, if unmitigated by the reactivity management system, would result in very few, if any, fuel failures. Reactor Systems would like to discuss the evaluation provided.

Provide information that the Rod Worth Minimizer (RWM) and Rod Block Monitor (RBM) are not credited in the mitigation of an accident or anticipated operational occurrence (AOO) below the automatic bypass setpoint of the RBM (i.e., the applicable power level for enabling of ganged rod withdrawal capability).

EGC Response

Section 15.4, "Reactivity and Power Distribution Anomalies," of the LaSalle County Station (LSCS) Updated Final Safety Analysis Report (UFSAR) describes accidents, events, and AOOs that impact reactivity. In the description of accidents and AOOs, the LSCS Rod Worth Minimizer (RWM) and Rod Block Monitor (RBM) systems are not credited in the mitigation of an accident or AOO, below the automatic bypass setpoint of the RBM (i.e., the applicable power level for enabling ganged rod withdrawal capability).

The UFSAR description for the following accident and AOOs credit the RWM and RBM systems with prevention of an accident below the automatic bypass setpoint of the RBM as a "what if" in the event operator action/inaction fails to prevent the accident or AOO.

- **Continuous Rod Withdrawal During Reactor Startup (UFSAR 15.4.1.2)**

  The RWM is credited with preventing an operator from selecting and withdrawing an out-of-sequence control rod. The probability of initiating causes (or multiple errors) for this event alone is considered low enough to warrant its categorization as an "infrequent incident." The probability of further development of this event is extremely low because it is contingent upon failure of the RWM system, concurrent with high rod worth, out-of-sequence rod selection contrary to procedures, plus operator non-acknowledgment of continuous alarm annunciations prior to safety system actuation.

- **Control Rod Drop Accident (UFSAR 15.4.9)**

  The RWM is credited with minimizing the maximum rod worth prior to occurrence of the Control Rod Drop Accident (CRDA). However, the RWM cannot and does not perform any function to mitigate the CRDA once it occurs.

Similar to these events, the RWM, as part of the new Rod Control Management System (RCMS) will also be credited with prevention of the new "Multiple Rod Withdrawal Error on Startup" event.

The Continuous Rod Withdrawal during Reactor Startup event is described in UFSAR Section 15.4.1.2 as an infrequent incident, based solely upon the low probability of initiating causes for the event. However, UFSAR Section 15.4.1.2 does not address ganged rod motion as part of that accident. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," addresses the evaluation of this event for ganged rod motion in Section 15.4.1, Revision 3, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition."

In Section 15.4.1 of NUREG-0800, for BWR/6 designs, the NRC reviewed the possibilities for single failures of the reactor control system that could result in uncontrolled withdrawal of control rods under low power startup conditions. The NRC concluded that the requirements of GDC 10, 17, 20, and 25 had been met, based upon the inclusion in the plant design of a Rod Pattern Control System (i.e., "Rod Block Instrumentation").

As described in NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Volume 2, "Bases," Section 3.3.2.1, "Control Rod Block Instrumentation," the rod pattern controller, along with operator actions, ensures that "during start-up conditions, only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to [10]% RTP." The NRC reviewed this system and found it acceptable because it precluded single failures in the Reactor Control System that could result in uncontrolled withdrawal of control rods under low-power conditions. The scope of the NRC review included the design features that act to prevent such withdrawals. The review also demonstrated that no single failure would permit an uncontrolled rod withdrawal that could lead to reactivity insertions greater than those routinely encountered during operation.

The EGC analysis in Section 3.3 of the referenced LAR, "Analysis of Potential Single Failures," demonstrated that the RCMS is also designed such that no single failure can cause an uncontrolled ganged rod withdrawal, and thus the NRC evaluation in Section 15.4.1 of NUREG-0800 for ganged rod motion at a BWR/6 is also applicable to ganged rod motion at LSCS.

Nevertheless, EGC has conducted a one-time evaluation of a postulated ganged rod withdrawal error at low power to evaluate the potential impact of this accident upon specified fuel design limits, relative to the acceptance criteria in NUREG-0800, Section 15.4.1, Items 2.a and 2.c. This evaluation is provided in Attachment 2. This evaluation did not credit either the RWM or RBM to mitigate or stop the response of the reactor core. The evaluation was performed by evaluating ganged rod withdrawal scenarios using initial power levels of 5%, 10%, 15%, 20%, 25%, and 30%.

The evaluation indicates that in the event that a control rod gang is erroneously withdrawn out-of-sequence, the result will not challenge fuel integrity. As such, the impact of the infrequent incident, with respect to fuel design limits (i.e., minimum critical power ratio, linear heat generation rate, uniform cladding strain, and peak pin enthalpy), is within the acceptance criteria of NUREG-0800, Section 15.4.1, and is bounded by the spectrum of other analyzed accidents for LSCS (e.g., UFSAR 15.4.1.2, "Continuous Rod Withdrawal During Reactor Startup").

NRC Request 2

Provide an evaluation that demonstrates that an accident or AOO at low power levels (i.e., below the automatic bypass setpoint of the RBM) that is not mitigated by the reactivity management system, will not challenge the fuel integrity (i.e., the fuel design limits including Minimum Critical Power Ratio, Linear Heat Generation Rate, uniform cladding strain, and peak pin enthalpy, relative to the Acceptance Criteria of NUREG-0800, Chapter 15.4.1), and is bounded by the spectrum of other analyzed accidents for LSCS.

EGC Response

Attachment 2 provides the requested evaluation (i.e., Framatome ANP, Inc., "Low Power CRWE Evaluation for LaSalle," dated October 31, 2005), the conclusion of which was described in the EGC response to NRC Request 1.

NRC Request 3

We accept ATWS compliance based in part on acceptable operator actions. How is Exelon monitoring operator response to the ganged rod insertion capability?

EGC Response

Licensed operator response to an anticipated transient without scram (ATWS) event is monitored during CR simulator drills as part of periodic licensed operator requalification training. This training, including the Control Room (CR) simulator drills, is developed in accordance with EGC Training and Qualification (TQ) procedures. As part of RCMS implementation, including ganged rod drive capability, drills will be developed for low, mid, and high power ATWS scenarios, with and without balanced partial rod insertion across the rod gangs.

In response to the simulator ATWS event, the operator will insert control rods using either Gang or Single mode, based on the resulting rod pattern. In the event an operator leaves Gang mode selected for a gang that already has 3 of the 4 rods full in, the single rod will insert as fast as if Single mode were selected. This is due to the proper operation of the CRD system drive water pressure stabilizing valves. Operator response to the transient, including the use of ganged rod control, will be evaluated utilizing the EGC training performance monitoring and evaluation process.

<u>NRC Request 4</u>

What Accidents and AOOs use the cram rod capability? How does cram rod insertion affect the core?

EGC Response

- **Accidents**

  Cram rod capability is not credited to mitigate any UFSAR Chapter 15 accident. Although not credited, cram rod capability may be used to provide a more rapid negative reactivity insertion during an ATWS event (i.e., UFSAR Section 15.8, "Anticipated Transients Without Scram (ATWS)").

- **Anticipated Operational Occurrences (AOOs)**

  Cram rod capability can be used in response to several power excursion AOOs:

  - Increased Reactor Recirculation flow
  - Reduced Feedwater (FW) temperature
  - Imminent equipment failures (e.g., Main Power Transformer loss-of-cooling, Loss of Main Generator Stator Water Cooling, Reactor Recirculation Pump Loss, Condenser Vacuum degradation)
  - Oscillating Power Range Monitoring System High Alarms (i.e., Core Instabilities)

  From an operational perspective, the insertion of a cram rod gang would replace the current procedure of four discrete rod insertions performed sequentially over an approximately four-minute period. The new procedure would enable an operator to insert the same four rods over a one-minute period. This will result in the same change to core power distribution. Although the change in core power due to a cram rod insertion would occur over a shorter time period, relative to single rod insertion, the time frame for the change in core power distribution would still be significantly greater than a reactor scram.

<u>NRC Request 5</u>

What changes in fuel design could result from the enhanced ramp-up capability? Is the ramp rate changing appreciably? How will increases in ramp rate affect the fuel?

EGC Response

- **What changes in fuel design could result from the enhanced ramp-up capability?**

  There are no changes in fuel design. Current EGC fuel conditioning limits will still be applicable and will continue to limit the ramp rates with reactor power above 30%. Below 30% power, ramp rates are limited to less than 300 MWe per hour by balance-of-plant (BOP) requirements and procedures. This low power ramp rate limitation will not change with ganged rod withdrawal.

- **Is the ramp rate changing appreciably?**

  At the low power levels where ganged rod withdrawal will be used (i.e., 0% to 30% power), the fuel conditioning limits do not restrict the ramp rate. As such, ramp rates are limited by balance-of-plant concerns to 300 MWe per hour or less.

  The current procedure of selecting and moving individual rods limits the ramp rate to an average range between approximately 50 MWe per hour and 100 MWe per hour. The use of ganged rod withdrawal would result in periods of time where the ramp rate may average closer to the 300 MWe per hour. In this case, the use of ganged rod withdrawal would require less discrete rod pulls, and hence less chance for selection or motion direction start errors.

- **How will increases in ramp rate affect the fuel?**

  As stated above, ramp rates below 30% power are limited by BOP concerns, as opposed to fuel conditioning limits. Therefore, any increase in ramp rates through the use of ganged rod withdrawal will not affect the fuel.

<u>**NRC Request 6**</u>

LaSalle is requested to provide: Software QA Plan/Procedures, System Description to Block Diagram Level, QA Plan for digital HW and SW, Software Configuration management Plan, Software Design Specification, Software Integration Plan, Software Maintenance Plan, Software Management Implementing Procedures, Software Test Procedures, Requirements Traceability Matrix, V&V Procedure, V&V Reports (9 months after LAR submittal), Test and V&V Problem Reports (Available for audit), Vendor as-built documentation (Available for audit), Site Acceptance Test Report (Available for review by the Region), Summary of test results and problem reports with resolution (Available for audit prior to operation).

EGC Response

As part of the referenced LAR, EGC described the processes, specifications, and procedures that are listed in NRC Request 6, and provided the applicable documents in Attachment 2 of the LAR, with the exception of test, verification, and audit report documents that are not yet available (i.e., Verification and Validation (V&V) Reports, Test and V&V Problem Reports, Vendor as-built documentation, Site Acceptance Test Report, and the Summary of test results and problem reports with resolution identified).

The documents that were submitted in the referenced LAR are proprietary to General Electric (GE) - Hitachi Nuclear Energy Americas LLC (GEH). As such, EGC requested that the information be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), and 10 CFR 9.17, "Agency records exempt from public disclosure," paragraph (a)(4). An affidavit attesting to the proprietary nature of this information was included in Attachment 3 of the referenced LAR.

The following description provides a cross-referenced accounting for the requested documents in the referenced LAR:

- **Software QA Plan/Procedures; QA Plan for digital HW and SW; Software Configuration Management Plan; Software Maintenance Plan; Software Management Implementing Procedures; Software Test Procedures; Requirements Traceability Matrix; V&V Procedure**

  Section 3.2.7 of Attachment 1 in the referenced LAR provided a description of software development, testing, and configuration control for the RCMS system. The software for the RCMS equipment was developed using the General Electric (GE) Nuclear Measurement Analysis and Control (NUMAC) Software Configuration Management Plan (SCMP), the GE NUMAC Software Management Plan (SMP), and the GE NUMAC Software Verification and Validation Plan (SVVP) process, as modified in GE Nuclear Energy DRF 0000-0038-3006, Revision 2, "Rod Control Management System (RCMS) Software Development Plan," dated March 9, 2007.

  The GE NUMAC software process is GE's standard safety-related software development process. This process has been submitted to the NRC multiple times and has been accepted for use in safety-related software. The most recent examples were license and TS amendments approving the implementation of a NUMAC Power Monitoring system at Susquehanna station. The NRC approved these applications of the NUMAC software process in Safety Evaluations dated March 3, 2006 (i.e., ML Accession # ML060540429) and March 23, 2007 (i.e., ML Accession # ML070720675).

  Documents that describe the software design, verification, and control for the RCMS were included in Attachment 2 of the referenced LAR, as described below:

  - Item 1 of Attachment 2 in the referenced LAR provided GE Nuclear Energy DRF 0000-0038-3006, Revision 2, "Rod Control Management System (RCMS) Software Development Plan." The RCMS Software Development Plan specifically addresses software design control, change control, documentation, record keeping, independent verification, and software development requirements, as described in RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," Revision 2.
  - Item 2 provided the "GE Nuclear Energy NUMAC Software Management Plan," (i.e., 23A5162, Revision 3). This document describes the process for the design, development, and maintenance of NUMAC product software.
  - Item 1, Section 2.5.3, "Traceability of Software Functional Testing to Software Requirements," and Section 2.7 describe the use of a "requirements traceability matrix" to help identify software requirements and provide traceability of all software testing to the software requirements and specifications. This includes software testing in both a "white box" test environment (i.e., module testing, integration testing, and code reviews as part of the software development process) and a "black box" test environment. The requirements traceability matrix is a database tool that is used, in concert with the software requirements and specifications, to confirm that all of the software testing requirements are satisfied.
  - Item 3 provided the "GE Nuclear Energy NUMAC Software Verification and Validation Plan," (i.e., 23A5163, Revision 3). This document describes the Software Verification and Validation Plan (SVVP) process used for all NUMAC product software.

- **Software Design Specification**

  Section 3.3.2 of Attachment 1 in the referenced LAR provided a description of RCMS software design features and requirements. This description is based upon specification documents that were included in Attachment 2 of the referenced LAR, as described below:

  - Item 5 provided the "RCMS/MCR Controller Performance Specification," (i.e., 26A6515, Revision 6). This document describes the software performance capabilities of the RCMS Controllers and Main Control Room (MCR) Controllers, including block diagrams of hardware configuration and a processing signal flow diagram.
  - Item 10 provided the "Rod Control Management System (RCMS) Displays," specification (i.e., 26A6609). This document specifies software performance requirements for microprocessor-based RCMS display units, and addresses operator actions and display-related functions at all of the displays.
  - Item 11 provided the "File Control Processor Performance Specification," (i.e., 26A6616). This document describes the performance of the File Control Processor (FCP) card with respect to its use as a Rod Position Information System (RPIS) subassembly in the RCMS. The FCP serves as the master control for all the Probe Multiplexer cards in a Multiplexer file (i.e., the mechanism for transmitting position information and commands to and from the RCMS to the individual control rod drives).

- **System Description to Block Diagram Level**

  Section 3.2.3 of Attachment 1 in the referenced LAR provided a description of the RCMS components, specifications, and interfaces.

  Section 3.2.6 of Attachment 1 in the referenced LAR described the qualification of RCMS components and equipment. As stated in Section 3.2.6, the RCMS and components have been built to the GE Nuclear Energy Quality Assurance Program, NEDO-11209-04A, Revision 8, which has been reviewed and accepted by the NRC. This program meets the requirements of ANSI N45.2, and 10 CFR 50, Appendix B.

  The descriptions in Sections 3.2.3 and 3.2.6 of the referenced LAR for RCMS components, interfaces, and qualifications are based upon specification documents that were included in Attachment 2 of the referenced LAR, as described below:

  - Item 4 provided the "Rod Control Management System Requirements Specification," (i.e., 26A6356, Revision 4). This document defines the design and performance requirements for the design and manufacture of the RCMS system, including detailed descriptions of the RCMS components and block diagrams for the system.
  - Item 6 provided the "Rod Control Management System (RCMS) External Interface Specification," (i.e., 26A6517, Revision 3). This document describes the external interfaces of the RCMS with the existing plant computers.
  - Item 7 provided the "RCMS Human System Interface Specification," (i.e., 26A6518, Revision 4). This document defines all graphical user interface screens for the RCMS, including the basis for the Human Factors Engineering analysis that was performed for the RCMS system.

- Item 8 provided the "RCMS and MCR Interface Performance Specification," (i.e., 26A6530, Revision 4). This document defines the performance requirements for the RCMS and MCR interface. Although the system contains two channels of both RCMS and MCR interfaces, the specification has been developed from the perspective of hardware/software residing on a single channel of each interface.
- Item 9 provided the "Rod Control Management System (RCMS) Internal Communication Protocol Specification," (i.e., 26A6582, Revision 2). This document defines all internal interfaces of the microprocessor-based RCMS.
- Item 10 provided the "Rod Control Management System (RCMS) Displays," specification (i.e., 26A6609). This document specifies software performance requirements for microprocessor-based RCMS display units, and addresses operator actions and display-related functions at all of the displays.

NRC Request 7

The touch-screen VDU is used for RCMS controls. The VDU is not seismically qualified even though it is mounted seismically (where required) to account for II/I concerns. Please justify how LaSalle intends to handle multiple spurious actuations in case of a seismic event.

EGC Response

The RCMS components are not safety-related or seismic, but are seismically installed in the cabinets and panels to satisfy seismic II/I concerns, where required. The components have similar chassis and construction as NUMAC hardware that is qualified to meet the seismic test criteria for safety-related applications. Additional information concerning the conformance of this equipment to seismic test criteria is provided in Item 4 of Attachment 2 to the referenced LAR.

In the main control room (MCR), the operator video display interface is comprised of three LCD display screens – one is the 40-inch full core map and the other two are 20-inch touch activated screens for monitoring and control. The 40-inch full core map graphically displays control rod position information, and does not provide any control or selection function.

The 40-inch display and the two 20-inch touch screens are controlled by the two MCR Controllers (i.e., one for the full core display and one 20-inch display, and one for the other 20-inch display). On the MCR 20-inch rod select screen, the operator can select a rod by touching the rod button on the screen. Once a rod is selected, the operator then can use the pushbuttons on the MCR Rod Select Panel (i.e., hard-wired physical, "Hall-Effect" switches, not software switches) to start rod motion.

The RCMS utilizes the same operating scheme for rod motion pushbuttons as currently installed for the Reactor Manual Control System. The RCMS pushbuttons are mechanically recessed into the Rod Select Console requiring the depressing slightly below the surface of the console, similar to the surrounding collar of the original design. In addition, as with the current rod control design, continuous rod withdrawal with RCMS requires the simultaneous pressing of the two pushbuttons on opposite sides of the Rod Select Console (i.e., requiring two-handed operation). The RCMS pushbuttons are slightly smaller than the current pushbuttons, providing a smaller target for a seismic event. Therefore, even if a rod is already selected with the new

RCMS during a seismic event, the probability of an inadvertent notch withdraw is approximately the same or lower as with the current design. Similarly, the recessed buttons and two-handed operation makes inadvertent rod withdraw not credible during a seismic event.

Rod movement requires two actions by the operator. Selection of the desired rod is accomplished from the 20-inch touch screen (in control) and rod movement is requested by the pushbuttons on the Rod Select Console. The 20-inch touch screens that are used in the RCMS design are capacitive touch, (i.e., based on the typical capacitance from a finger). In the event of a seismic event, these displays are adequately mounted to the H13-P603 panel and are not sensitive to falling objects or debris from other systems. In addition, a seismic event of enough magnitude to cause erratic indication or display on the 20-inch MCR touch screen would provide sufficient physical indication to the control room operator to preclude operation of the two RCMS pushbuttons, and/or terminate the use of these pushbuttons. Thus any erratic behavior in the VDU would have no adverse affect on rod motion.

NRC Request 8

Cyber security has been addressed in Section 3.2.5 of the submittal. Per Figure 6.1, Defensive Model of NEI 04-04, Control and Safety Systems are categorized as Level 4 with no two-way communication. LaSalle Figure 3, RCMS External Network Diagram shows two-way communications. Please provide analysis or justification explaining how the network diagram meets the NEI 04-04 guidance.

EGC Response

EGC has completed a cyber security risk assessment at LSCS in accordance with Nuclear Energy Institute (NEI) 04-04, Revision 1, "Cyber Security Program for Power Reactors," and NUREG/CR-6847, "Cyber Security Self-Assessment Method of U. S. Nuclear Power Plants," for site Critical Digital Assets, including the proposed design for RCMS. The proposed RCMS design was ranked as Very Well Protected, with a susceptibility level of 2, a risk category of B-4, and was deemed an Acceptable Risk with no design-related mitigation required. As such, the proposed RCMS design, as an LSCS Critical Digital Asset, satisfies the intent and purpose of NEI 04-04.

Section 3.2.5 of the referenced LAR states that all RCMS components, with the exception of the RCMS Controllers, are connected to each other via a dedicated private network, with no direct connection to the higher level networks. Each RCMS Controller is connected to both the Plant Process Computer (PPC) and the RWM Sequence Computer via dedicated connections. These connections only provide bi-directional messaging capabilities between the RCMS Controllers and both the PPC and RWM systems; no access to RCMS control functions is directly available via these links.

Item 6 of Attachment 2 to the referenced LAR provided the "Rod Control Management System (RCMS) External Interface Specification," (i.e., 26A6517, Revision 3). This document describes the interfaces of the RCMS Controllers with the two PPCs (i.e., A and B) and the RWM Sequence Computer. The PPCs and RWM connect with the RCMS Controller on a TCP IP link. Up to two TCP connections are active at a time between each PPC and an RCMS Controller (i.e., a Data Connection and a Status Connection). One TCP connection is active between the RWM and an RCMS Controller.

- **Messages sent to RCMS Controller by the PPC Over the Data Connection**

  Utilizing the Data Connection, the PPC will send a "Rod Scram Timing Data" request to the RCMS Controller. This data is normally available and stored in the RCMS Controller cyclic buffer. If no scram timing data exists in the RCMS Controller, a corresponding message is returned to the PPC. The format for the request is the only format that the RCMS Controller will acknowledge. This transfer is a low priority in the RCMS and thus will not override higher priority programming in the RCMS.

- **Messages sent to RCMS Controller by the PPC Over the Status Connection**

  Utilizing the Signal Connection, the PPC will send a PPC status message to the RCMS Controller every 1000 milliseconds. This message contains current PPC status and time information.

  The timestamp in the message is used to update the timestamp of the RCMS Controller. The RCMS Controller time clock is synchronized with the "Synchronization Time" that is provided in the PPC Status Message. The status and timestamp message from the PPC is a fixed format message that is the only acceptable format that the RCMS Controller will use for time synchronization. Synchronization is performed at midnight each night, as well as when the following criteria are satisfied:

  - No rod motion occurring, and
  - A time difference between the RCMS Controller timestamp and PPC timestamp greater than 100 milliseconds.

  Data and signal transfer from the external PPC to the RCMS Controller, including the timestamp, provides no control activity, and must be in a specified format for use by the RCMS Controller. Therefore, there is no dependency relationship between the RCMS Controller (i.e., an NEI 04-04 "Control & Safety System") and the PPC (i.e., an NEI 04-04 "Data Acquisition System").

- **Messages sent to RCMS Controller by the RWM Sequence Computer**

  The RWM Sequence Computer will send the following messages to the RCMS Controllers:

  - Operating Sequence Upload Request,
  - Operating Sequence Download Data, and
  - RWM Status Request.

  Data and signal transfer from the external RWM Sequence Computer to the RCMS Controller provides no control activity. Therefore, there is no dependency relationship between the RCMS Controller (i.e., an NEI 04-04 "Control & Safety System") and the RWM Sequence Computer (i.e., an NEI 04-04 "Data Acquisition System").

EGC has determined that the dedicated connections between the RCMS Controller and both the PPC and the RWM Sequence Computer (i.e., the "PPC Private Network Switch" that is depicted in Figure 3 of the referenced LAR), in concert with the type of data transferred to the RCMS Controller from the PPC and RWM Sequence Computer, establish an acceptable variation from the Defensive Model that is depicted in Figure 6.1 of NEI 04-04.

NRC Request 9

Provide information concerning operator actions (added, deleted, or changed) and their impact on EOPs, DBAs, and PRAs.

EGC Response

- **Impact of Revised Operator Actions on EOPs**

  Section 3.2.9 of Attachment 1 in the referenced LAR describes the impact of RCMS upon the Emergency Operations Procedures (EOPs). RCMS will provide all the current expected functions for the EOP process. Other than the revised operator interface in the control room, there will be no difference in the displays for the EOP required indications. The new RCMS will enhance the information available to the control room for EOP events with improved shutdown confirmation displays.

  With respect to ganged rod drive capability, the only applicable EOP is LGA-010, "ATWS." The change to this EOP will include the use of ganged rod insertion, which will help to achieve a shutdown state sooner. The actual change will likely be similar to a footnote directing the operator to use ganged rod insertion, if available.

- **Impact of Revised Operator Actions on DBAs**

  Both the existing Reactor Manual Control System (RMCS) and the new RCMS are not safety-related, are not used for plant shutdown resulting from accident or nonstandard operational conditions, and are not assumed to function during the events analyzed in Chapter 15 of the LSCS Updated Final Safety Analysis Report, including the Control Rod Drop Accident, Control Rod Removal Error During Refueling, Rod Withdrawal Error - at Power, and Continuous Rod Withdrawal during Reactor Startup.

  The new RCMS has no interaction with the safety-related portion of the reactivity control system. Therefore, any revised operator actions associated with the implementation of RCMS and ganged rod drive capability will not impact any Design Basis Accidents.

- **Impact of Revised Operator Actions on PRA**

  As described above, RCMS has no interaction with the safety-related portion of the reactivity control system, nor is it credited to function during the events analyzed in Chapter 15 of the LSCS UFSAR. Therefore, current and revised operator actions associated with RCMS are not modeled in the LSCS PRA, and consequently have no effect.

NRC Request 10

Provide information concerning training development and implementation.

EGC Response

Development and implementation of licensed operator training at EGC, including any required training on the RCMS and ganged rod drive capability is defined and directed by a series of standardized procedures, as delineated below.

- TQ-AA-1, Training and Qualification
- TQ-AA-10, Training System Development Process Description
- TQ-AA-150, Operator Training Programs
- TQ-AA-221, Exelon Nuclear Training - Analysis Phase
- TQ-AA-222, Exelon Nuclear Training - Design Phase
- TQ-AA-223, Exelon Nuclear Training - Development Phase
- TQ-AA-224, Exelon Nuclear Training - Implementation Phase
- TQ-AA-225, Exelon Nuclear Training - Evaluation Phase

Although this training has not been completely developed and implemented for RCMS and ganged rod drive capability, this training will include CR simulator drills for low, mid, and high-power ATWS scenarios, with and without balanced rod insertion across the rod gangs. In these scenarios, licensed operators will then insert control rods, using either gang or single-rod control, based on the resulting rod pattern. Operator performance during these drills will be evaluated in accordance with the normal training performance evaluation process.

NRC Request 11

Provide information concerning simulator impact and planned mods.

EGC Response

Prior to completion of RCMS installation, EGC will update the LSCS simulator to reflect the RCMS modification, and utilize the updated simulator to train operators, as described in the response to NRC Request 10.

NRC Request 12

Provide information concerning control room layout, displays, alarms, controls (added, deleted, or changed).

EGC Response

Section 3.2.4 of the referenced LAR, "Internal Communication and Data Interfaces," provides a summary of the equipment, controls, and operator processes for the LSCS RCMS. Detailed descriptions of the changes to the LSCS MCR layout, including new equipment, alarms, and controls is described in multiple specifications that were provided in Attachment 2 of the referenced LAR.

Specifically, Item 4 provided the "Rod Control Management System Requirements Specification," (i.e., 26A6356, Revision 4). This document provided detailed descriptions of the RCMS components and block diagrams for the system. In particular, Section 3.0 describes the hardware requirements; Section 4.4 describes the operator control and display system; and Section 8.0 describes the user interface with respect to controls and displays.

Item 7 provided the "RCMS Human System Interface Specification," (i.e., 26A6518, Revision 4). This document defines all graphical user interface screens for the RCMS, including the basis for the Human Factors Engineering analysis that was performed for the RCMS system.

Item 10 provided the "Rod Control Management System (RCMS) Displays," specification (i.e., 26A6609). This document addresses operator actions and display-related functions at all of the RCMS displays.

In addition to these documents, during a public meeting with the NRC on September 19, 2006, EGC provided a presentation on the design and functionality (i.e., ADAMS Accession Numbers ML062920419 and ML063330288). This presentation included a description and graphical representation of the new RCMS equipment in the control room.

NRC Request 13

Provide information concerning operating experience and how it was factored into the design.

EGC Response

During the design specification phase of RCMS development, EGC reviewed all GE Service Information Letters (SILs) concerning control rod drive and reactivity controls. Applicable operating experience from these SILs was addressed in the RCMS design. Primarily, the operating experience associated with ganged rod drive capability that was reviewed addressed ATWS response, shutdown confirmation, and a faster RPIS scan rate.

The final RCMS design is leveraged heavily from experience derived from the existing GE-Advanced Boiling Water Reactor (ABWR) design, as well as existing NUMAC hardware, software, and displays. The RCMS design inputs include the existing system design specification and system elementary diagram.

NUREG-0700, "Human-System Interface Design Review Guidelines," formed the basis for development of the RCMS operating displays. This design was based on the ABWR human factors engineering (HFE) design, as well as, extensive reviews from LSCS operations personnel (i.e., as described in the response to NRC Request 14 below). Existing RCMS color schemes and Display Primitives were selected by LSCS operations personnel to provide similar appearance as the existing Rod Control System with enhanced surveillance and operating capabilities.

NRC Request 14

Provide information concerning human factors V&V, including any planned "man-in-the-loop" testing.

EGC Response

Throughout the RCMS design process, EGC utilized a dedicated Reactor Operator (RO) and Senior Reactor Operator (SRO) for input and guidance. These NRC-licensed operators, in conjunction with GE experts, played a primary role in the human factors design of RCMS.

Additional LSCS SROs and ROs were also consulted, and evaluations were performed, resulting in the development of a design mock-up that enabled the SROs and ROs to test different designs for the rod movement control switches in the new RCMS (i.e., the push buttons that are used by the operator to withdraw and insert a control rod). In addition, the actual RCMS CR equipment has been assembled in a dedicated, non-production area at LSCS, and has been manipulated by SROs and ROs, utilizing V&V procedures. Thus, significant real-time evaluation and mock-up testing of the RCMS by licensed operators has been a significant input into an integrated "man-in-the-loop" evaluation.

With respect to a standard human factors review of the RCMS, the graphical user interface screens were developed in accordance with an HFE program that defined the information, controls, and alarms for controlling and monitoring the RCMS. Although the RCMS has no safety-related functions, the following codes, standards, and guidelines were used as the basis for the HFE analysis that was performed for this system.

- Institute of Electrical and Electronics Engineers (IEEE) Standard 1023 -1988, "IEEE Guide to the Application of Human Factors Engineering to Systems, Equipments and Facilities of Nuclear Power Generating Stations."
- International Electrotechnical Commission (IEC) Publication 964, "Design for Control Rooms of Nuclear Power Plants, 1989."
- NUREG-0700, Revision 2, "Human-System Interface Design Review Guideline."
- NUREG-0711, "Human Factor Engineering, Program Review Level."
- NUREG-0800, Revision1, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18, "Human Factor Engineering."
- NUREG/CR-3331, "A Methodology For Allocating Nuclear Power Plant Control Functions to Human and Automated Control."

ATTACHMENT 2
Framatome ANP, Inc., "Low Power CRWE Evaluation for LaSalle," October 31, 2005

AREVA

SWJ:05:014

Date: October 31, 2005

To: C. M. Powers

From: S. W. Jones

Subject: **Low Power CRWE Evaluation for LaSalle**

Ref.: 1. E-mail, C. M. Powers to S. W. Jones, "TDS NF-B0279," July 8, 2005

Distribution

D. G. Carr
R. J. DeMartino
P. D. Wimpy

Attached is the final letter report for the evaluation of low power CRWE with ganged control rod withdrawal that has been prepared for LaSalle. The report is non-proprietary as specified in the Reference work authorization.

The results have been documented in accordance with FANP QA program. Modifications to the report were made to address the comments provided by Exelon on the preliminary report. The most significant change was the discussion of the fuel enthalpy. NUREG 1433 was added as a reference for the 170 cal/g threshold for fuel failure protection for the low power rod withdrawal event.

Please forward this report to Exelon.

Approved: PDW ______________________ 31 OCT 05
O. C. Brown, Manager — Date
BWR Neutronics

Approved: ______________________ 3 NOV 05
D. W. Pruitt, Manager — Date
Methods Development

sjp
Attachment

Framatome ANP, Inc. An AREVA and Siemens Company

2101 Horn Rapids Road
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Tel: 509 375 8100
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SWJ:05:014
Revision 0

Low Power CRWE Evaluation for LaSalle Rod Control Management System

October 2005

Framatome ANP, Inc.

SWJ:05:014
Revision 0

Low Power CRWE Evaluation for
LaSalle Rod Control Management System

sjp

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

Nomenclature

BOC	beginning of cycle
BPWS	banked position withdrawal sequence
CPR	critical power ratio
CRWE	control rod withdrawal error
DMCPR	change in minimum critical power ratio
EOC	end of cycle
FANP	Framatome ANP (Advanced Nuclear Power), Inc..
GNF	Global Nuclear Fuels
IMCPR	initial minimum critical power ratio
IRM	intermediate range monitor
LHGR	linear heat generation rate
MCPR	minimum critical power ratio
PHE	peak hot excess reactivity
RBM	rod block monitor
UFSAR	updated final safety analysis report
ΔCPR	change in critical power ratio

1.0 Introduction and Summary

Implementation of the rod control management system (RCMS) at LaSalle allows for ganged control rod pulls below the rod block monitor operability. The current updated final safety analysis report (UFSAR) (Reference 1) identifies the low power rod withdrawal as not a credible event. Although it is believed that the low power rod withdrawal error will not be a credible event with the implementation of the RCMS and gang withdrawal, an evaluation has been performed to demonstrate that the ganged withdrawal below the Rod Block Monitor (RBM) operation is bounded by other events.

The evaluation was performed using the CASMO4/MICROBURN-B2 approved methodology (Reference 2) and the RAMONA5-FA code (not approved References 3, and 4.) The evaluations were performed in a manner consistent with Standard Review Plan (SRP) 15.4.1-3 (Reference 5) to the extent possible. For power levels above 5% and up to the power level of RBM operability, the CASMO4/MICROBURN-B2 methodology was used to evaluate the impact of the ganged control rod withdrawal error (CRWE) on critical power ratio (CPR) and linear heat generation rate (LHGR). The evaluation was completed at various exposures with the LaSalle Unit 1 Cycle 12 core design.

The control rod patterns evaluated were typical startup sequences with the exception of the actual error rod or gang of rods. Until the time of the actual rod withdrawal error, the pattern is maintained with the Rod Worth Minimizer and is in accordance with BPWS.

The CASMO4/MICROBURN-B2 methodology was also used to identify the strongest gang RWE events below 5% power. The RAMONA5-FA code (a three dimensional, coupled neutron kinetics-thermal hydraulics model) was used to evaluate the intermediate range monitor (IRM) instrument response at power levels below 5% and the peak pin fuel enthalpy.

The criteria used in this evaluation are those presented as Acceptance Criteria items 2.a and 2.c of SRP 15.4.1 (Reference 5).

2. The requirements of GDC 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

    a. The thermal margin limits (DNBR for PWRs and MCPR for BWRs) as specified in SRP Section 4.4 are met.

b. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 do not exceed the melting point.

c. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%.

Although not identified in Reference 5, an additional failure criterion of 170 cal/g deposited enthalpy is provided in Section B3.3.1.1 of Reference 7 for fuel protection by the IRM for low power rod withdrawal events.

The impact on CPR was evaluated and determined not to challenge CPR limits in the 5 to 10 percent power range. At low powers the minimum critical power ratio (MCPR) is high and requires a substantial change to approach the MCPR safety limit. In this analysis, the maximum DCPR/ICPR (delta CPR over initial CPR) was demonstrated to be less than that of other events.

The CPR could not be calculated for all conditions below 5% power in that the conditions were outside of the SPCB critical power correlation bounds. Therefore, comparisons of fuel enthalpy were made between the cases below 5% to that from the 30% power case. The peak pin fuel enthalpy of the cases below 5% power remained below the maximum value of the hot channel average enthalpy for the 30% power case. In both conditions the flow is essentially at natural circulation and an increase in power results in an increase in flow. As such, the enthalpy comparison between the low power peak pin enthalpy and the hot channel average enthalpy at 30% power demonstrates that the critical power is not exceeded below 5% and criteria 2a is met below 5% power for the ganged RWE.

An evaluation of linear heat generation rate (LHGR) was completed at the beginning of cycle. At 30 % power, the largest fraction of LHGR limit was 0.544 with a total peaking factor of 5.02. A total peaking factor would have to be greater than 9 to approach the LHGR limit at 30% power. As the power decreases, the total peaking factor would have to be even greater to approach the steady state LHGR limit.

An evaluation of the change in LHGR relative to the fuel design limit demonstrated that the greatest increase in nodal LHGR was less than 30% of the steady state LHGR limit. (Both Global Nuclear Fuels (GNF) and Framatome ANP, Inc.* (FANP) have proprietary limits for transients on the allowed increase in LHGR which are greater than this.) Therefore, the 1 %

* Framatome ANP, Inc. is an AREVA and Siemens company.

cladding strain limit is not exceeded and criteria 2c is met for operating power levels. The enthalpy deposition criteria identified in Reference 7 was also evaluated and it is shown that the IRM scram prevents enthalpies from exceeding the 170 cal/g criterion for ganged rod withdrawal.

2.0 Background

Control rod withdrawal with the original reactor manual control system (RMCS) at the LaSalle Units was limited to a single control rod at a time. Implementation of the new rod control management system (RCMS) may allow the gang pull of control rods during startup. The control rod gangs are composed of groups of one to four rods which may be withdrawn simultaneously. Gangs are defined for both A and B sequence rod patterns. The gang definitions for the LaSalle Units and used in this evaluation are given in Figures 2.1 and 2.2. The withdrawal of multiple control rods, i.e. gangs, results in more positive reactivity insertion into the core than a single control rod. However, the gangs are configured such that reactivity insertion is minimized. Gang withdrawal is standard for BWR-6 reactors which have a rod control system which enforces banked position withdrawal sequence (BPWS) during startup such that the low power rod withdrawal is not a credible event. The capability of the RCMS is similar in that BPWS will be enforced during reactor startup as well. The assumed BPWS withdrawal sequence used in this evaluation is provided in Table 2.1. This evaluation is performed to demonstrate that in the event a gang is erroneously withdrawn out of sequence the result will not challenge the fuel integrity. A failure mechanism is not identified as to how a gang could be erroneously withdrawn.

At low powers in the reactor startup mode, the IRM instruments provide a trip signal. The IRMs are configured into two groups for the purpose of generating trip signals. At any time one IRM per trip system group may be out of service. Therefore, the spatial dependence of the nearest operating IRM must be considered in the evaluation of the single rod withdrawal error. However, when control rods are pulled in gang mode, the spatial dependence on functioning IRMs with respect to generating a trip signal is essentially eliminated. Figure 2.1 also provides a schematic of the location of the IRM instruments. The gangs are configured such that a control rod being withdrawn will be close to a working IRM. In addition to the evaluation of the LHGR and CPR response to the RWE, an evaluation of IRM responses to the simulated rod withdrawal error was included in the analysis. Although there are gangs defined which contain less than four rods, these gangs are located near the four central IRMs.

Order	Group	End Position		Order	Group	End Position
1	G07	4		12	G06	48
2	G08	4		13	G07	48
3	G07	6		14	G08	48
4	G08	6		15	G09	4
5	G07	8		16	G09	12
6	G08	8		17	G09	48
7	G07	12		18	G10	8
8	G08	12		19	G10G	48
9	G05	12		20	G10E	12
10	G05	48		21	G10F	12
11	G06	12				

Table 2.1 Assumed Pull Sequence Following Group 4

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					G06B 1	G03G 2	G05D 3	G04F 4	G05C 5	G03G 6	G06C 7				
55				G05A 8	G01F 9	G09E 10	G02E 11	G10G 12	G02D 13	G09F 14	G01F 15	G05B 16			
51			G06A 17	G03E 18	G08C 19	G04D 20	G07F 21	G03C 22	G07G 23	G04D 24	G08D 25	G03E 26	G06A 27		
47		G05B 28	G01E 29	G09D 30	G02C 31	G10E 32	G01C 33	G09C 34	G01D 35	G10F 36	G02C 37	G09D 38	G01E 39	G05A 40	
43	G06C 41	G03F 42	G08D 43	G04C 44	G07E 45	G03B 46	G08A 47	G04A 48	G08B 49	G03B 50	G07E 51	G04C 52	G08C 53	G03F 54	G06B 55
39	G01G 56	G09F 57	G02D 58	G10F 59	G01B 60	G09B 61	G02A 62	G10B 63	G02B 64	G09B 65	G01A 66	G10E 67	G02E 68	G09E 69	G01G 70
35	G05C 71	G04E 72	G07G 73	G03D 74	G08B 75	G04B 76	G07A 77	G03A 78	G07B 79	G04B 80	G08A 81	G03D 82	G07F 83	G04E 84	G05D 85
31	G02F 86	G10G 87	G01D 88	G09C 89	G02B 90	G10A 91	G01A 92	G09A 93	G01B 94	G10C 95	G02A 96	G09C 97	G01C 98	G10G 99	G02F 100
27	G05D 101	G04D 102	G07F 103	G03C 104	G08A 105	G04A 106	G07D 107	G03B 108	G07C 109	G04A 110	G08B 111	G03C 112	G07G 113	G04D 114	G05C 115
23	G01G 116	G09E 117	G02D 118	G10E 119	G01B 120	G09B 121	G02A 122	G10D 123	G02B 124	G09B 125	G01A 126	G10F 127	G02E 128	G09F 129	G01G 130
19	G06B 131	G03F 132	G08C 133	G04C 134	G07E 135	G03A 136	G08B 137	G04B 138	G08A 139	G03A 140	G07E 141	G04C 142	G08D 143	G03F 144	G06C 145
15		G05A 146	G01E 147	G09D 148	G02C 149	G10F 150	G01C 151	G09C 152	G01D 153	G10E 154	G02C 155	G09D 156	G01E 157	G05B 158	
11			G06A 159	G03E 160	G08D 161	G04E 162	G07G 163	G03D 164	G07F 165	G04E 166	G08C 167	G03E 168	G06A 169		
7				G05B 170	G01F 171	G09F 172	G02E 173	G10G 174	G02D 175	G09E 176	G01F 177	G05A 178			
3					G06C 179	G03G 180	G05C 181	G04F 182	G05D 183	G03G 184	G06B 185				

IRM

Gang ID

Rod #

Figure 2.1 LaSalle A-Sequence Gangs, Control Rod Numbers, and IRM Locations

	2	6	10	14	18	22	26	30	34	38	42	46	50	54	58
59					G03J 1	G06B 2	G04E 3	G05A 4	G04F 5	G06A 6	G03I 7				
55				G02I 8	G07H 9	G01E 10	G08G 11	G02G 12	G08F 13	G01F 14	G07H 15	G02H 16			
51			G03H 17	G09C 18	G04C 19	G10E 20	G03F 21	G09F 22	G03G 23	G10E 24	G04D 25	G09G 26	G03H 27		
47		G02H 28	G07G 29	G01D 30	G08D 31	G02E 32	G07E 33	G01C 34	G07D 35	G02F 36	G08D 37	G01D 38	G07G 39	G02I 40	
43	G03I 41	G09H 42	G04D 43	G10D 44	G03E 45	G09C 46	G04A 47	G10C 48	G04B 49	G09C 50	G03E 51	G10D 52	G04C 53	G09H 54	G03J 55
39	G05B 56	G01F 57	G08E 58	G02F 59	G07C 60	G01B 61	G08B 62	G02C 63	G08A 64	G01B 65	G07C 66	G02E 67	G08E 68	G01E 69	G05A 70
35	G04F 71	G10G 72	G03G 73	G09E 74	G04B 75	G10B 76	G03A 77	G09A 78	G03C 79	G10A 80	G04A 81	G09D 82	G03F 83	G10F 84	G04E 85
31	G06A 86	G02G 87	G07F 88	G01C 89	G08C 90	G02A 91	G07A 92	G01A 93	G07B 94	G02D 95	G08C 96	G01C 97	G07F 98	G02G 99	G06A 100
27	G04E 101	G10F 102	G03F 103	G09D 104	G04A 105	G10A 106	G03D 107	G09B 108	G03B 109	G10B 110	G04B 111	G09E 112	G03G 113	G10G 114	G04F 115
23	G05A 116	G01E 117	G08E 118	G02E 119	G07C 120	G01B 121	G08A 122	G02B 123	G08B 124	G01B 125	G07C 126	G02F 127	G08E 128	G01F 129	G05B 130
19	G03J 131	G09H 132	G04C 133	G10D 134	G03E 135	G09C 136	G04B 137	G10C 138	G04A 139	G09C 140	G03E 141	G10D 142	G04D 143	G09H 144	G03I 145
15		G02I 146	G07G 147	G01D 148	G08D 149	G02F 150	G07D 151	G01C 152	G07E 153	G02E 154	G08D 155	G01D 156	G07G 157	G02H 158	
11			G03H 159	G09G 160	G04D 161	G10E 162	G03G 163	G09F 164	G03F 165	G10E 166	G04C 167	G09G 168	G03H 169		
7				G02H 170	G07H 171	G01F 172	G08F 173	G02G 174	G08G 175	G01E 176	G07H 177	G02I 178			
3					G03I 179	G06A 180	G04F 181	G05A 182	G04E 183	G06B 184	G03J 185				

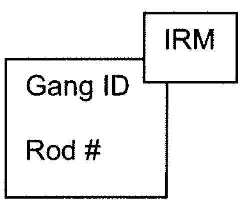

Figure 2.2 LaSalle B-Sequence Gang Definitions

3.0 **MCPR Evaluation**

The evaluation of MCPR was performed by establishing a critical rod pattern at 5% power intervals. Starting from the initial condition, all inserted gangs (excluding Groups 5 and 6) were then pulled simulating an error withdrawal. Evaluations were performed at beginning of cycle (BOC), peak hot excess reactivity (PHE), and end of cycle (EOC) for A-sequence and BOC for a B sequence rod pattern. The results, presented as a function of initial power, are provided in Tables 3.1 through 3.4 showing the gang withdrawal which resulted in the largest ΔCPR. In all cases the final CPR remained well above the MCPR safety limit. Graphical presentation of the results for all gang withdrawals for the A sequence are presented in Figures 3.1 through 3.3. The B sequence results are shown in Figure 3.4. The conclusion remains unchanged in that the MCPR remains well above the safety limit for the ganged CRWE event.

For the BOC-A sequence, the single rod Gang G09A had the largest decrease in MCPR. This same trend is demonstrated for the BOC-B sequence withdrawals in that fewer rods in a gang results in larger change in MCPR. However, this change in CPR in response to the RWE is less than applicable plant transient limits. The maximum ΔCPR over initial CPR was less than 0.39 at 10% power. An operating limit of 1.82 is determined for a safety limit of 1.11 using the formulation of:

$$\text{MCPROL} = \text{MCPRSL} \,/\left(1-\frac{\Delta CPR}{IMCPR}\right)$$

This is bounded by the 25% power operating limit of 1.98 reported in Reference 6.

Table 3.1 BOC A Sequence MCPR for Low Power CRWE

% Power	IMCPR	MinCPR	DMCPR	Gang	DMCPR/IMCPR
5	9.99	5.53	NA	G09A	
10	6.11	3.79	2.32	G09A	0.38
15	4.7	3.14	1.56	G09A	0.33
20	4.51	3.09	1.42	G09A	0.32
25	2.93	2.04	0.89	G09A	0.30
30	2.73	1.84	0.89	G09A	0.33

Table 3.2 PHE A Sequence MCPR for Low Power CRWE

% Power	IMCPR	MCPR	DMCPR	Gang	DMCPR/IMCPR
5	9.99	5.61	NA	G09B	
10	6.78	4.52	2.26	G09B	0.33
15	4.70	3.34	1.36	G09B	0.29
20	3.50	2.54	0.96	G09B	0.27
25	2.93	2.46	0.47	G09B	0.16
30	2.77	1.91	0.86	G09B	0.31

Table 3.3 EOC A Sequence MCPR for Low Power CRWE

% Power	IMCPR	MCPR	DMCPR	Gang	DMCPR/IMCPR
5	9.99	5.49	NA		
10	6.2	4.16	2.04	G09B	0.33
15	4.35	3.14	1.21	G09B	0.28
20	3.43	2.50	0.93	G09B/G09C	0.27
25	3.02	2.24	0.78	G09B/G09C	0.26
30	2.61	1.98	0.63	G10F	0.24

Table 3.4 BOC B Sequence MCPR for Low Power CRWE

% Power	IMCPR	MCPR	DMCPR	Gang	DMCPR/IMCPR
10	6.44	4.08	2.36	G09A	0.37
15	5.16	3.47	1.69	G09A	0.33
20	4.20	2.93	1.27	G09A	0.30
25	3.00	2.29	0.71	G09A	0.24
30	2.82	2.07	0.75	G09A	0.27

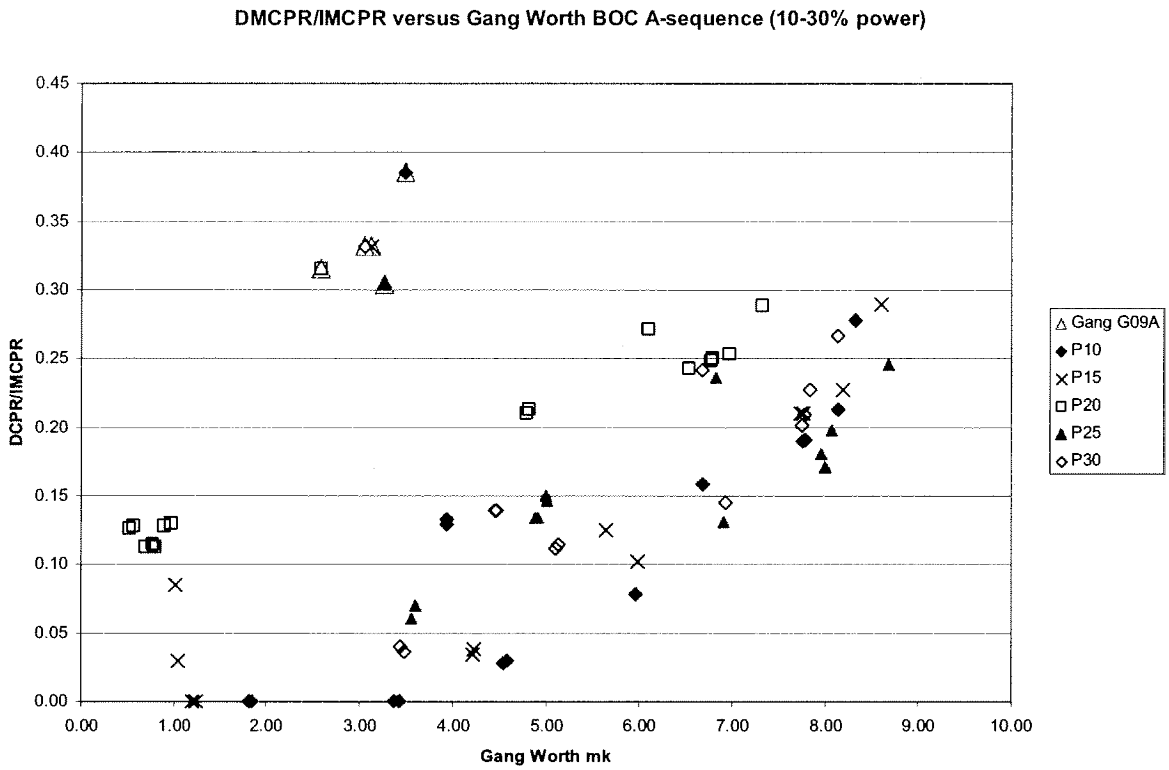

Figure 3.1 DMCPR/IMCPR versus Gang Worth BOC A-Sequence

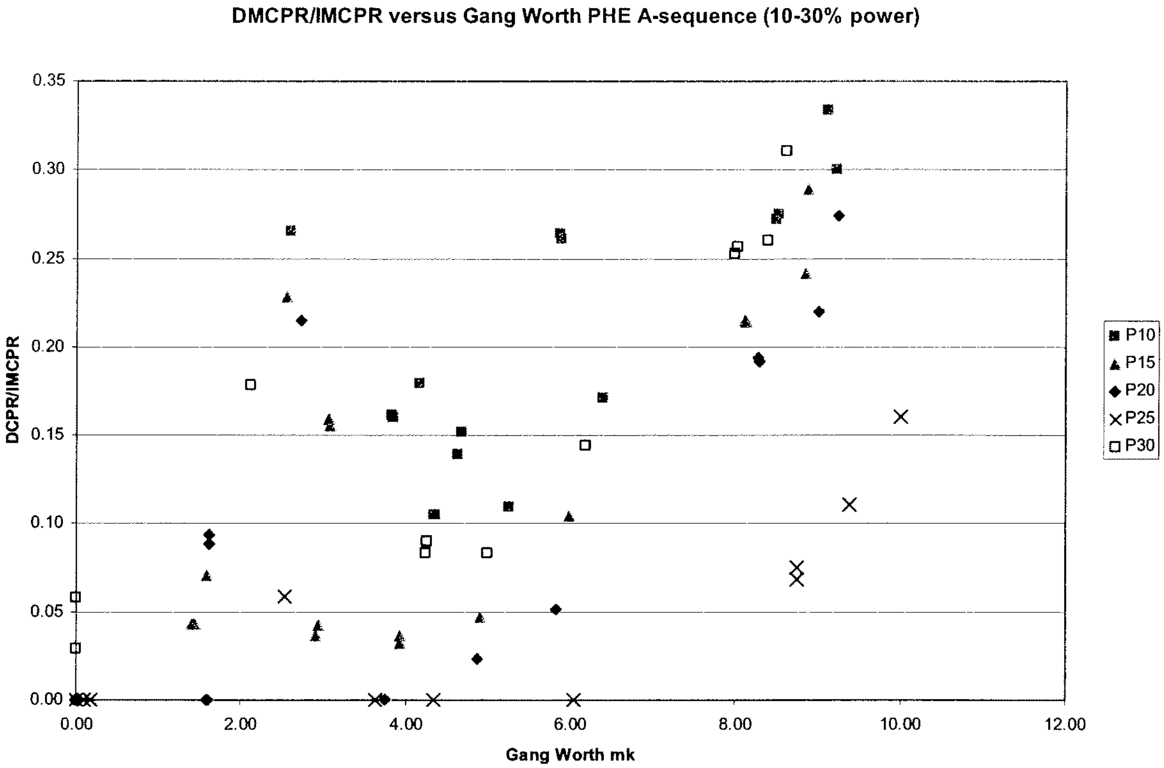

Figure 3.2 DMCPR/IMCPR versus Gang Worth PHE A-Sequence

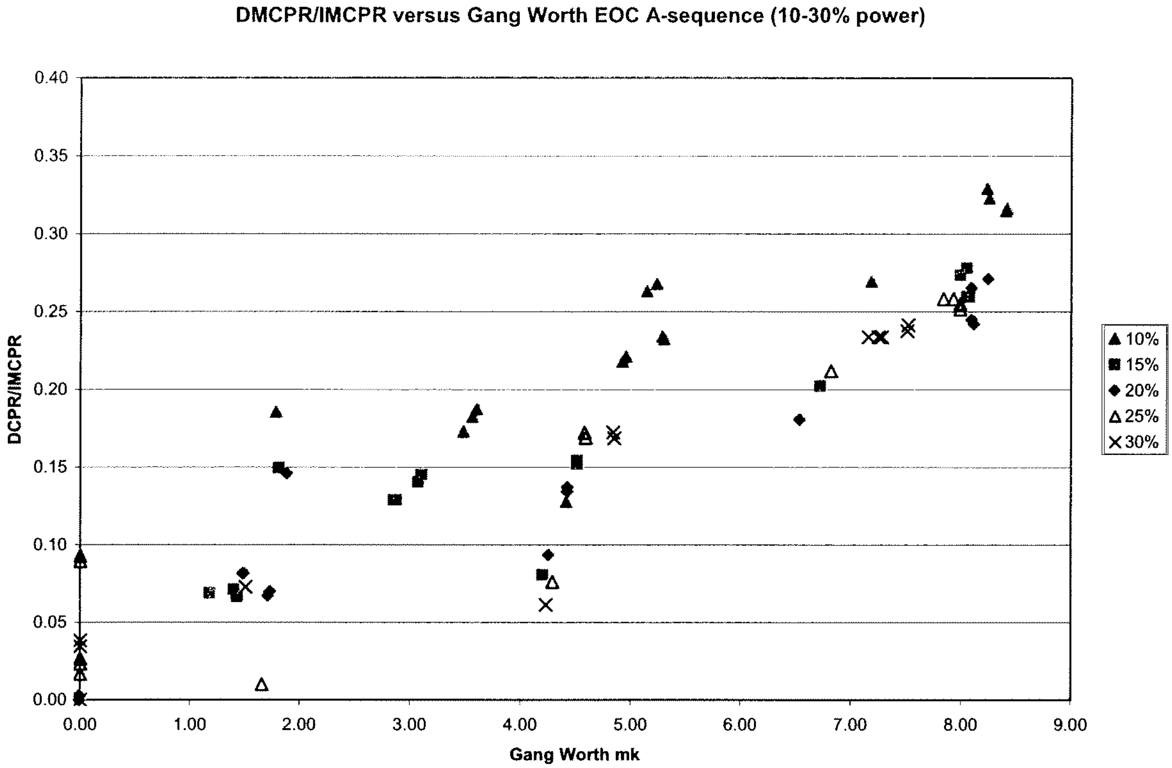

Figure 3.3 DMCPR/IMCPR versus Gang Worth EOC A-Sequence

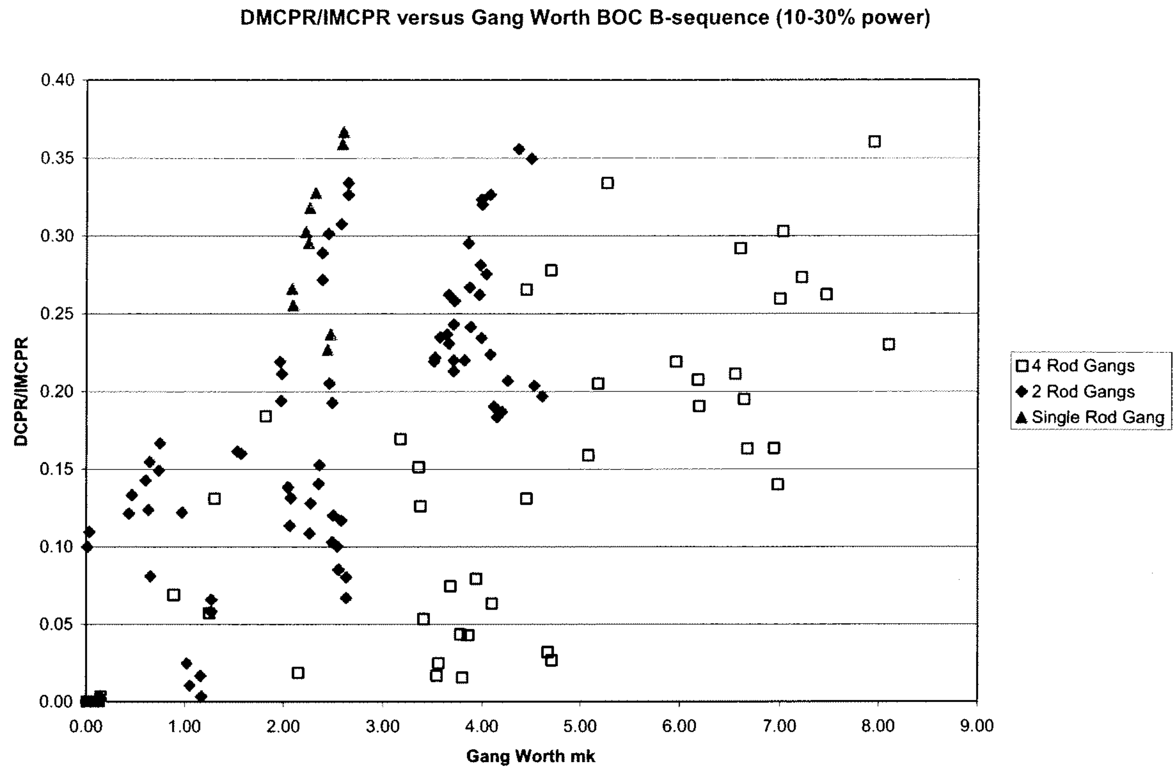

Figure 3.4 DMCPR/IMCPR versus Gang Worth BOC B-Sequence

4.0 LHGR Evaluation

The change in LHGR was evaluated to verify the 1% strain criterion is not exceeded. The evaluation of LHGR was performed at the beginning of cycle. At 30% power, the largest fraction of LHGR limit was 0.544 with a total peaking factor of 5.02. A total peaking factor would have to be greater than 9 to approach the LHGR limit at 30% power. As the power decreases, the total peaking factor would have to be even greater to approach the steady state LHGR limit.

An evaluation of the change in LHGR relative to the fuel design limit demonstrated that the greatest increase in nodal LHGR was less than 30% of the steady state LHGR limit. (Both GNF and FANP have proprietary limits for transients on the allowed increase in LHGR which are greater than 30% for rod withdrawal.) Therefore, the 1 % cladding strain limit is not exceeded and criteria 2c is met. Figure 4.1 identifies the maximum fraction of the steady state LHGR limit for the various gang withdrawals at various power levels. Figure 4.2 identifies the maximum value of the nodal change in LHGR divided by the LHGR limit. As Figure 4.2 indicates, the maximum change in LHGR is less than 0.30 of the LHGR design limit.

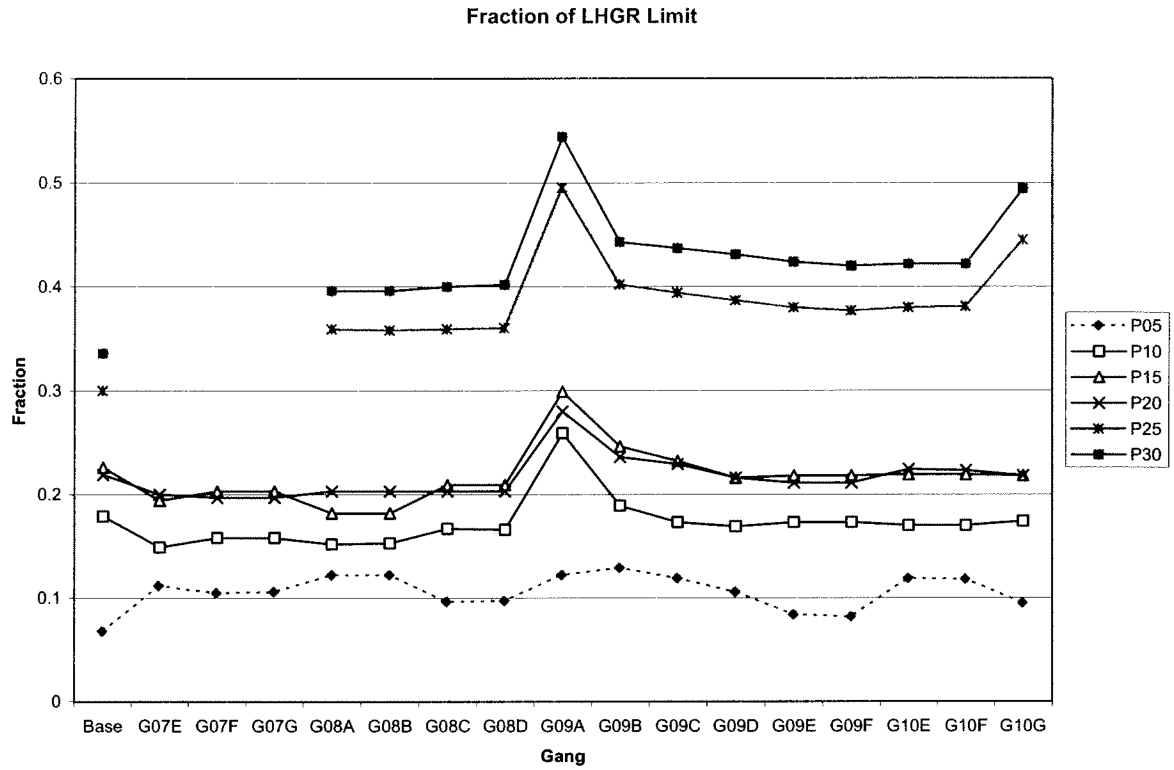

Figure 4.1 Maximum Fraction of Limit for LHGR

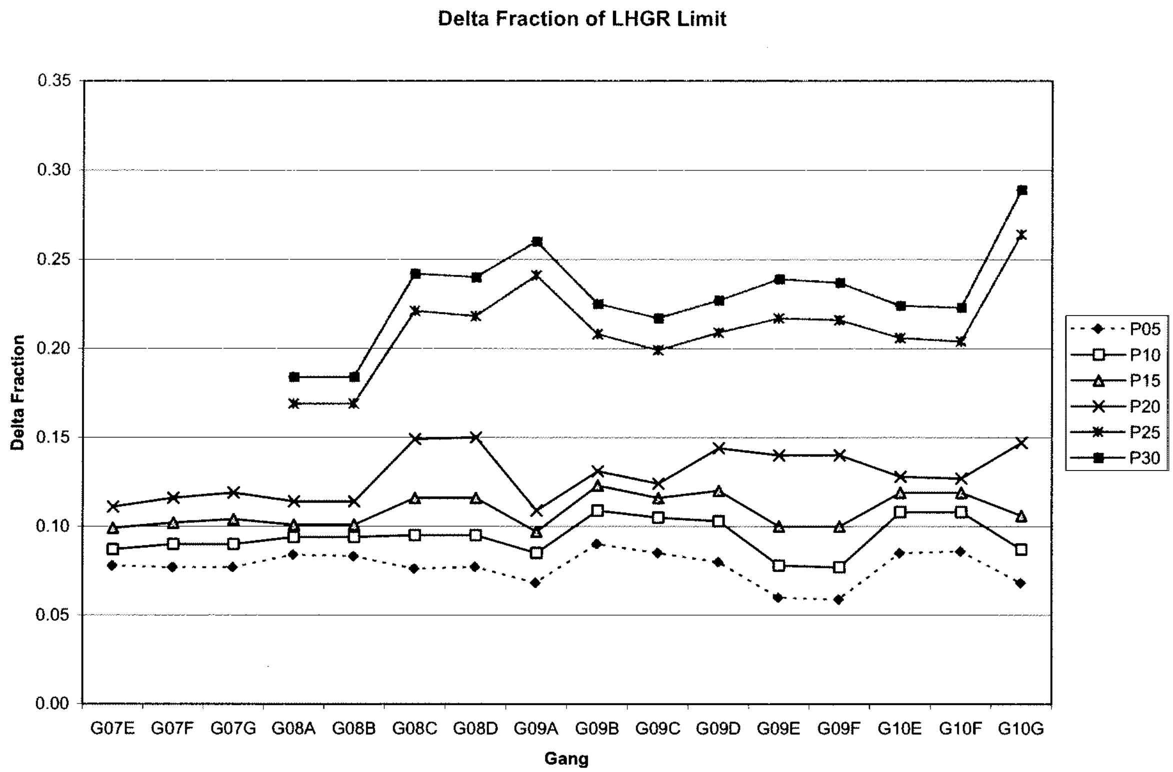

Figure 4.2 Maximum Delta LHGR Divided by LHGR Limit

5.0 Enthlapy Evaluations

The RAMONA5-FA code is a transient code used to evaluate the time-dependence of the RWE. The code was first used to repeat the RWE cases performed with MICROBURN-B2. It was then used to evaluate fuel enthalpy and IRM responses at power levels below 5% power. The gangs selected for evaluation below 5% power are given in Table 5.1.

The hot channel average enthalpy was determined for the withdrawals at 30% power and is shown in Figure 5.1. The enthalpy values for 30% power cases are used for relative comparisons with low power RAMONA results.

The highest gang worths were selected from each inserted group. Groups 5 and 6 and those Gangs comprised of a single control rod were excluded from this selection process with the exception of Gang G09A. At 5% power, only Gangs G09A and G09B were evaluated.

A local peaking factor of 2.0 was used for the evaluation of peak pin enthalpy. This value conservatively bounds the actual local peaking and results in a higher pin temperature and enthalpy. The peak pin and channel enthalpies for the rod withdrawal at BOC A sequence are given in Table 5.2. All enthalpies are less than the maximum hot channel average enthalpy shown in Figure 5.1. The fuel enthalpies for PHE and EOC for the A sequence are given in Tables 5.3 and 5.4. The peak pin channel enthalpies during the transients are shown in Figures 5.2 through 5.4 for the BOC A sequence rod withdrawals. The maximum hot channel average enthalpy from Figure 5.1 is also included in Figures 5.2 through 5.5. These figures show that the enthalpy for the hottest pin for the low power withdrawal was less than the average hot channel enthalpy of the 30% power case.

An evaluation of the IRM response was included in the evaluation to demonstrate that the IRM system would preclude the adiabatic enthalpy from exceeding 170 cal/g as indicated in Reference 7. The evaluation of the IRM response assumed that the IRMs were at 10% of scale prior to the RWE and a trip would occur at 120% of scale or an increase in the flux level by a factor of 12.

Gangs G09C, G09D and G10G were selected for evaluation of the IRM response for a single rod withdrawal versus a gang withdrawal. Control rod 87 (R087) was selected for single rod withdrawal evaluation from Gang G10C, control rod 89 from Gang G09C, and control rod 156 (R156) from Gang G09D. These calculations were initiated from 0.2% power to demonstrate the

IRM response and trip function. The peak pin enthalpies for these withdrawals are provided in Figure 5.5. The IRM responses during the pull of R156 from Gang G09D are given in Figure 5.6. The IRM responses during the pull of Gang G09D are given in Figure 5.7.

The peak pin enthalpy of the withdrawal of Gang G09C (four rods) is only slightly higher than that of the single rod 89. The enthalpy is well below challenging fuel integrity limits compared to the enthalpy of the 30% power case (Figure 5.1).

For Gang G09D, the peak pin enthalpy of the four rod case is consistent with the single rod case R156. Withdrawal of Gang G09D would result in an IRM initiated trip since several IRMs increase by a factor of 12. An IRM trip cannot be assumed for the single rod R156 in that single IRM which detected the flux increase by a factor of 12 could be bypassed. For this example the resultant peak pin enthalpy is greater for single rod than for the gang withdrawal since the gang RWE would be terminated by the IRM trip.

A scram would be initiated for the four rod Gang G10G and the peak pin enthalpy remains below that of withdrawal of R087 which would not initiate a scram.

The adiabatic enthalpies for the cases which scrammed are provided in Table 5.5. The enthalpy results reported are considerably less than the 170 cal/g criterion identified in Reference 7.

Table 5.1 BOC A Sequence Rod Withdrawal Below 5% Power

Gang Pulls at 75% rod density		Gang pulls at 63% rod dentisy	
Gang	Worth(mk)	Gang	Worth(mk)
G03B	12.45	G04A	6.66
G04A	11.69	G07G	13.56
G07G	17.03	G08A	14.96
G08B	17.84	G09C	19.50
G10G	18.17	G10G	15.48

Table 5.2 Relative Peak Fuel Enthalpies from Rod Withdrawal Error

	Peak Pin	Peak Channel
Case	cal/g	cal/g
bocA_G02_G03B	24.37	20.40
bocA_G02_G04A	23.34	19.97
bocA_G02_G07G	24.37	20.47
bocA_G02_G08B	24.84	20.74
bocA_G02_G10G	26.76	21.57
bocA_G03_G04A	22.60	19.59
bocA_G03_G07G	25.80	21.17
bocA_G03_G08A	26.52	21.48
bocA_G03_G09C	28.19	22.22
bocA_G03_G10G	27.47	22.00
bocA_P05_G09A	35.12	25.56
bocA_P05_G09B	29.38	22.89

Table 5.3 Relative Peak Fuel Enthalpies and Rod Worths for PHE A Sequence Rod Withdrawal Error

				Peak pin	Peak Channel
			Worth	Enthalpy	Enthalpy
Case	Initial Group	Gang	mk	cal/g	cal/g
pheA_G02_G03B	1 and 2 at 48	G03B	11.12	25.32	20.86
pheA_G02_G04D	1 and 2 at 48	G04D	11.35	24.61	20.62
pheA_G02_G07F	1 and 2 at 48	G07F	18.18	27.23	21.74
pheA_G02_G07G	1 and 2 at 48	G07G	18.31	27.23	21.74
pheA_G02_G08A	1 and 2 at 48	G08A	16.90	27.23	21.64
pheA_G02_G08B	1 and 2 at 48	G08B	17.12	27.23	21.72
pheA_G02_G09B	1 and 2 at 48	G09B	17.45	27.47	21.76
pheA_G02_G09C	1 and 2 at 48	G09C	19.10	27.47	21.98
pheA_G02_G10F	1 and 2 at 48	G10F	15.39	26.52	21.40
pheA_G03_G07F	1, 2 and 3 at 48	G07F	15.86	28.19	22.12
pheA_G03_G07G	1, 2 and 3 at 48	G07G	16.24	28.19	22.15
pheA_G03_G08A	1, 2 and 3 at 48	G08A	15.57	28.19	22.07
pheA_G03_G08B	1, 2 and 3 at 48	G08B	15.59	28.19	22.12
pheA_G03_G09B	1, 2 and 3 at 48	G09B	15.61	28.43	22.12
pheA_G03_G09C	1, 2 and 3 at 48	G09C	23.22	30.58	23.58
pheA_G03_G10F	1, 2 and 3 at 48	G10F	13.93	27.47	21.72
pheA_P05_G08B	1, 2, 3 and 4 at 48	G08B	11.57	30.10	23.36
pheA_P05_G09B	1, 2, 3 and 4 at 48	G09B	11.66	30.58	23.58

Table 5.4 Relative Peak Fuel Enthalpies and Rod Worths for EOC A Sequence Rod Withdrawal Error

Case	Initial Group	Gang	Worth mk	Peak pin Enthalpy cal/g	Peak Channel Enthalpy cal/g
eocA_G02_G08C	1 and 2 at 48	G08C	27.60	30.10	23.20
eocA_G02_G08D	1 and 2 at 48	G08D	27.71	29.86	23.20
eocA_G02_G09E	1 and 2 at 48	G09E	23.69	27.95	22.22
eocA_G02_G09F	1 and 2 at 48	G09F	22.65	27.47	22.00
eocA_G03_G08C	1, 2 and 3 at 48	G08C	26.35	33.92	24.84
eocA_G03_G08D	1, 2 and 3 at 48	G08D	25.99	33.92	24.84
eocA_G03_G09C	1, 2 and 3 at 48	G09C	19.76	32.73	24.13
eocA_G03_G09D	1, 2 and 3 at 48	G09D	19.74	31.77	23.53
eocA_P05_G08B	1 -4 at 48 G7 at 4	G08B	9.06	36.79	26.28
eocA_P05_G08C	1 -4 at 48 G7 at 4	G08C	8.39	37.74	26.99

Table 5.5 Adiabatic Enthalpy Results

Gang	Worth mk	Enthalpy cal/g
G10G	18.17	25.4
G09D	17.21	40.37

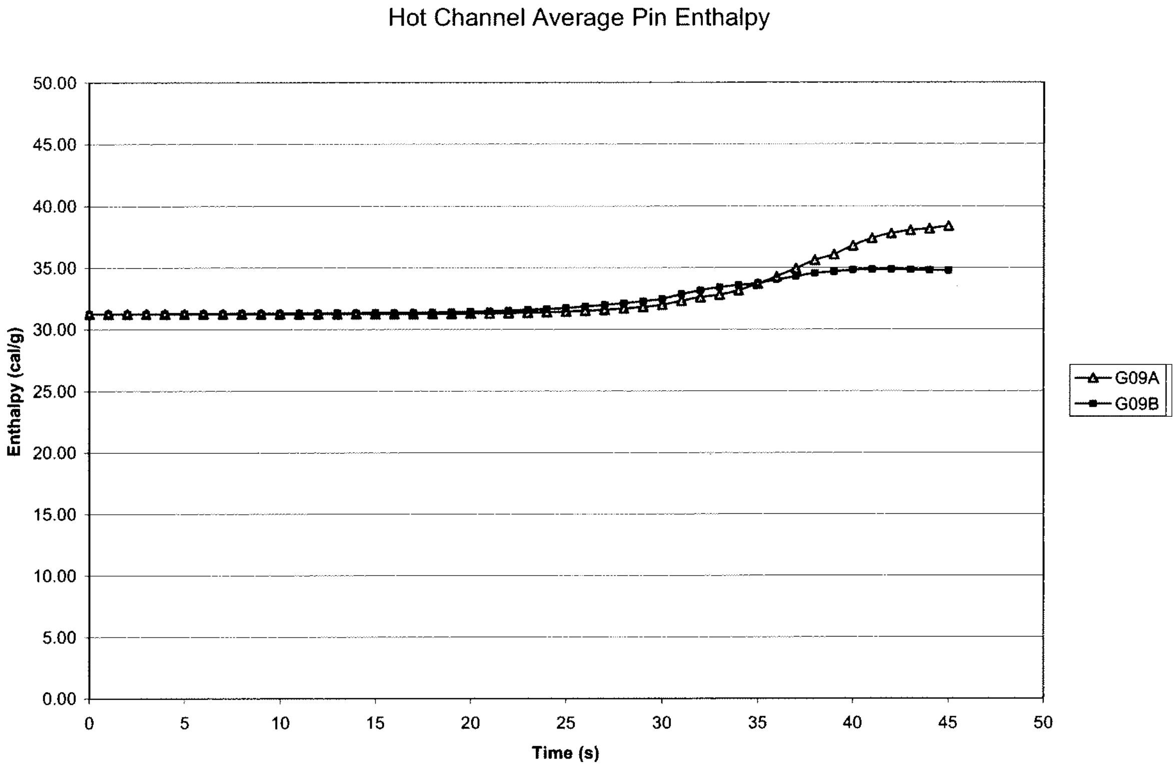

Figure 5.1 Hot Channel Average Enthalpy for Rod Withdrawal at 30% Power

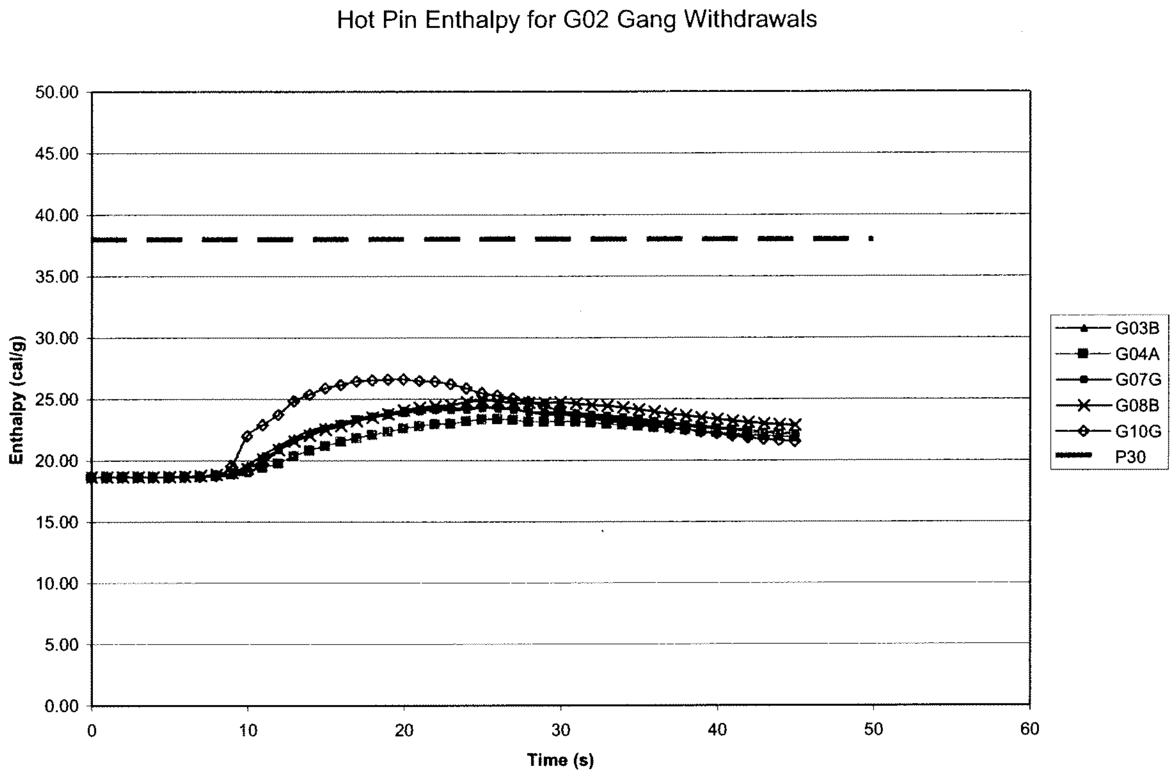

Figure 5.2 Hot Pin Enthalpy for Gang Withdrawals Starting from 75% Rod Density

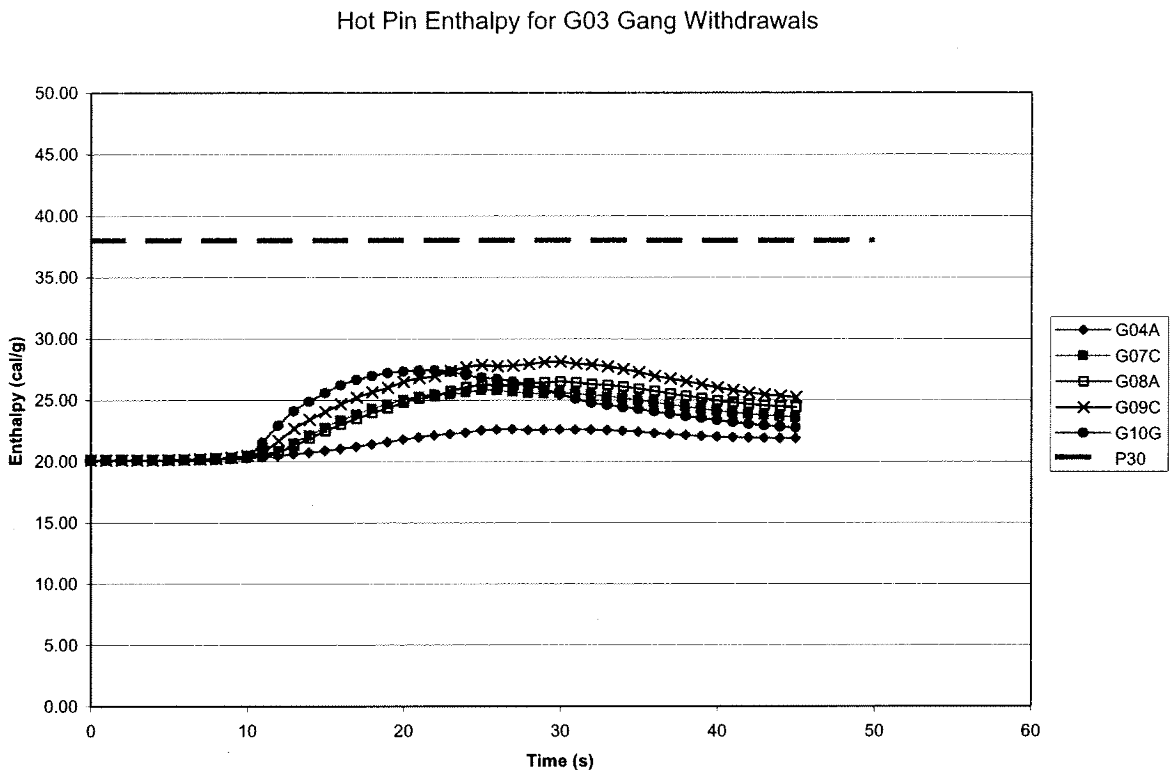

Figure 5.3 Hot Pin Enthalpy for Gang Withdrawals Starting from 63% Rod Density

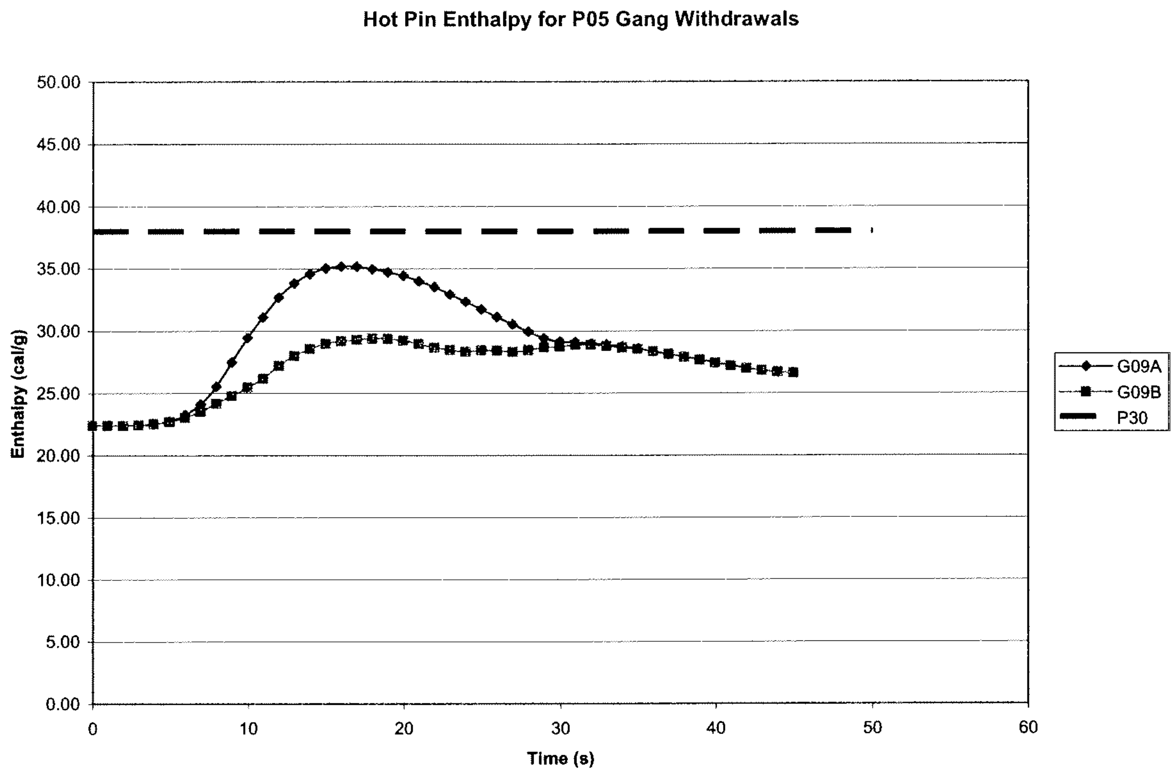

Figure 5.4 Hot Pin Enthalpy for Gang Withdrawals Initiated at 5% Power

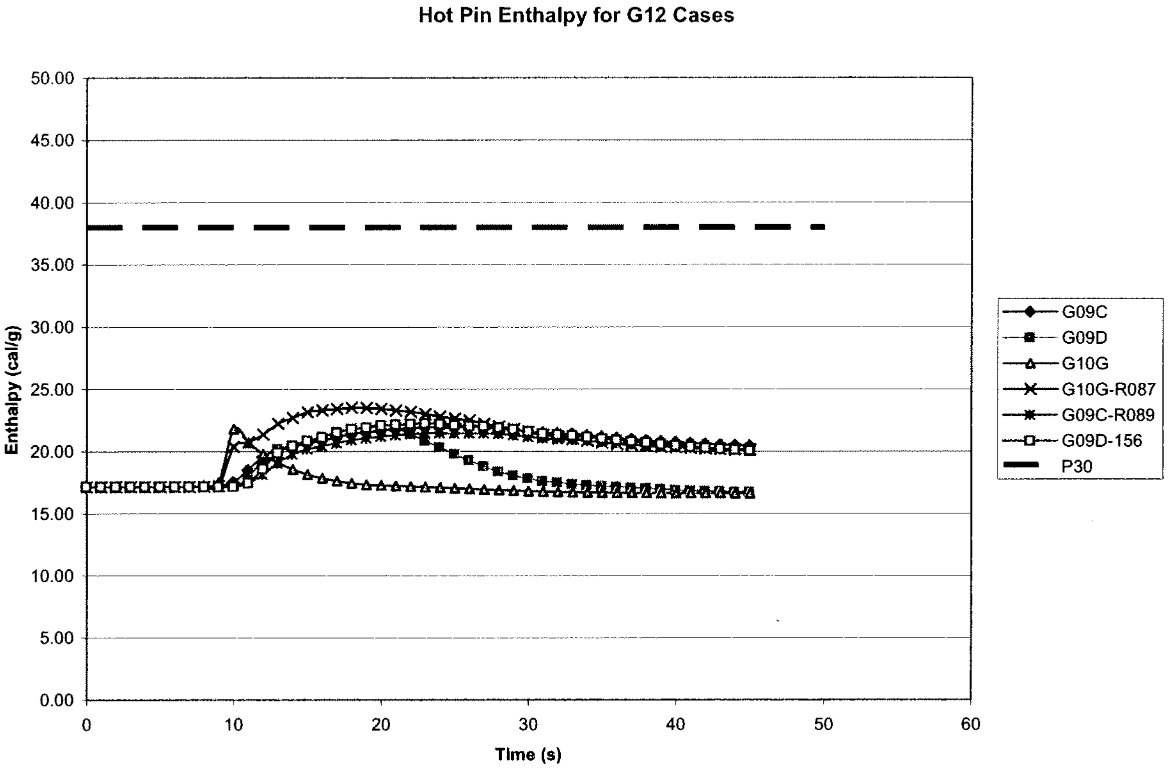

Figure 5.5 Hot Pin Enthalpy for Withdrawals Initiated at 0.2% Power

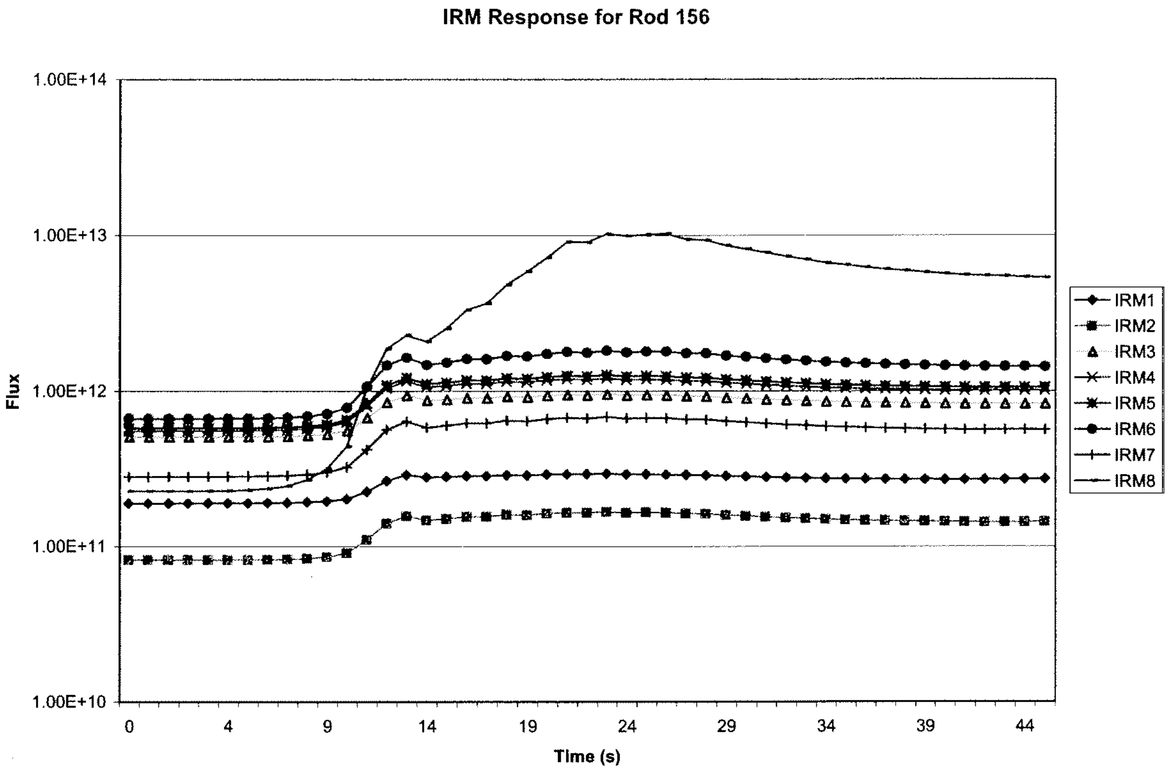

Figure 5.6 IRM Signal Responses for Withdrawal of Rod 156

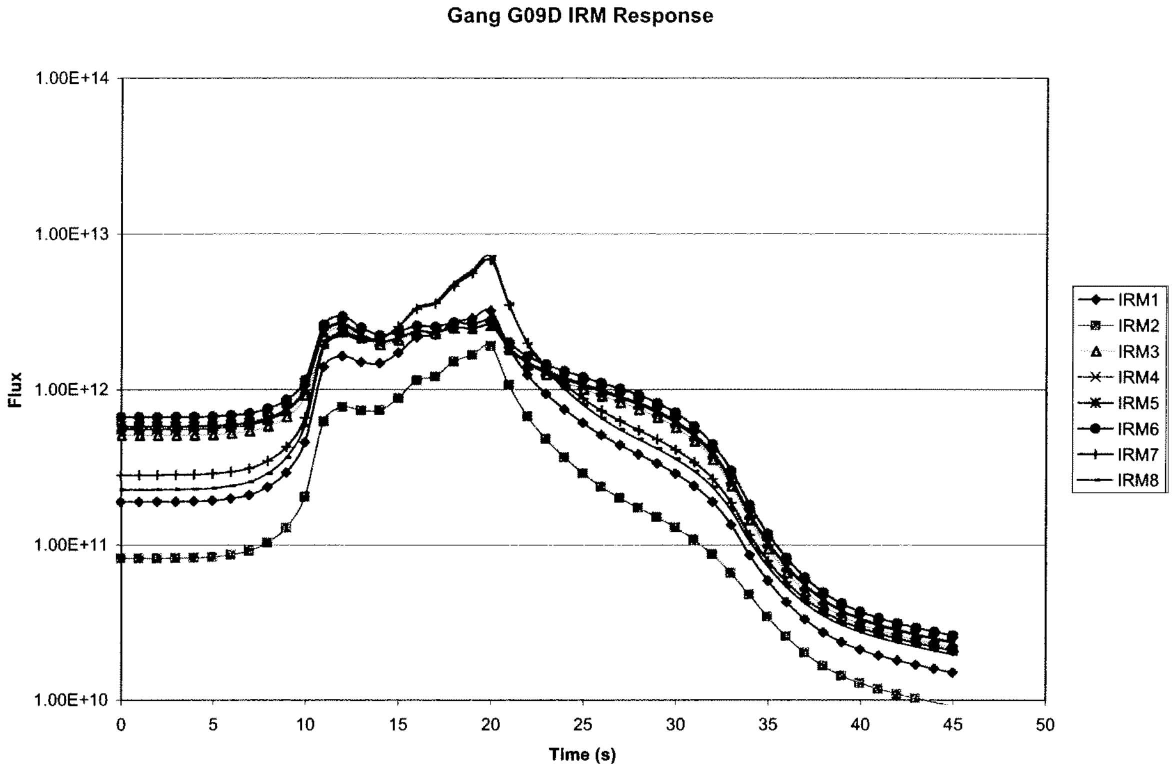

Figure 5.7 IRM Signal Responses for Withdrawal of Gang G09D

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