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Constellation Energy
R.E. Ginna Nuclear Power Plant, LLC

December 3, 2004

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Request for Additional Information (RAI) dated November 9, 2004, Regarding the Proposed Control Room Emergency Air Treatment System (CREATS) Modification
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

References:
1. Letter from Robert C. Mecredy (RG&E) to Robert L. Clark (NRC) dated May 21, 2003, License Amendment Request Regarding Revision of Ginna Technical Specification Sections 1.1, 3.3.6, 3.4.16, 3.6.6, 3.7.9, 5.5.10, 5.5.16, and 5.6.7 Resulting From Modification of the Control Room Emergency Air Treatment System and Change in Dose Calculation Methodology to Alternate Source Term.

2. Letter from Robert L. Clark (NRC) to Mary G. Korsnick (R.E. Ginna NPP) dated November 9, 2004, Request for Additional Information Regarding R.E. Ginna Nuclear Power Plant License Amendment Request Relating to the Control Room Emergency Air Treatment System Modification (TAC No. MB9123).

3. Letter from Robert C. Mecredy (RG&E) to Robert L. Clark (NRC) dated April 22, 2004, Design Information for the Proposed Control Room Emergency Air Treatment System (CREATS) Modification.

4. Letter from Robert C. Mecredy (RG&E) to Robert L. Clark (NRC) dated December 1, 2003, Addendum to License Amendment Request submitted May 21, 2003.

Dear Mr. Clark:

On May 21, 2003 Ginna submitted Reference 1 related to the Control Room Emergency Air Treatment System (CREATS) modification and conversion to Alternate Source Term (AST). The attachments to this letter provide a response to the Request for Additional Information (RAIs) contained in Reference 2, including an addendum to Reference 4 revising Technical Specification Table 3.3.6-1 to reflect the new setpoint methodology. This information demonstrates that the Dose Analysis, CREATS Equipment Qualification, Cable Separation, and Control Room Radiation Monitor Analytical Limit are being appropriately addressed.

A003

1001201

Very truly yours,

Joseph A. Widay

Joseph A. Widay

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF WAYNE :

I, Joseph A. Widay, being duly sworn, state that I am Acting Vice President – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Joseph A. Widay

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 3rd day of December, 2004.

WITNESS my Hand and Notarial Seal: Sharon L Miller

Notary Public

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 20 06

My Commission Expires:

12-21-~~04~~ 06

Date

Attachments:

1. Response to RAIs
2. Topical Design Basis – Electrical Independence
3. Representative Analysis – New CREATS Cables
4. Representative Analysis – Existing Cables
5. Design Analysis DA-EE-2000-009, Revision 1
6. Revised Submittal – Tech Spec Table 3.3.6-1

Cc: Mr. Robert L. Clark (Mail Stop O-8-C2)
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Attachment 1

Response to RAIs

ATTACHMENT 1

REQUEST FOR ADDITIONAL INFORMATION

R. E. GINNA NUCLEAR POWER PLANT

CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM

R. E. Ginna Nuclear Power Plant, LLC proposed design modifications to the Control Room Emergency Air Treatment System (CREATS), the Control Room Emergency Cooling System (CRECS), and the Containment Post Accident Charcoal Filters are based on the full scope implementation of the alternate source term (AST). The Nuclear Regulatory Commission staff has reviewed the license amendment request and has determined that the following additional information is needed.

1. Dose Analysis

Section 2.2 of Attachment 1 to Constellation Energy's July 14, 2004 letter states that discharges from the atmospheric relief valve (ARV) pathway were used to model a steam line break accident with the steam generator intact. However, a description of the radiological analysis for this accident has not been provided in Attachment 1. Please provide a description of the radiological analysis for the steam line break accident.

Response:

The control room dose analyses use the x/Q for the for the ruptured steam header, for both the steam header and the intact SG ARV locations. The steam header x/Q values are slightly smaller than the ARV values, but the doses for the steam line break are dominated by the releases from the faulted loop steam header. The intact SG ARV accounts for only about 2% of the total activity release. If the activity releases for the steam header and intact SG (ARV) are separated and appropriate control room x/Qs applied to each, the total control room dose is estimated to increase by about 2.5%. For example, the CR dose for the accident initiated spike case will increase from 0.632 rem to approximately 0.65 rem. As such, using the same x/Q for both release points provides acceptable results even though it is a small non-conservatism, and has a minimal impact on the result as compared to the available margin to the 5 rem TEDE limit.

Section 2.6 of Attachment 1 to Constellation Energy's July 14, 2004 letter states that the tornado condition atmospheric dispersion factors at the Exclusion Area Boundary (EAB) were calculated using a distance-to-receptor value of 503 m. Please justify why the shortest EAB distance listed in Ginna Updated Final Safety Analysis Report, Table 2.3-20 (450 m for the SSE, S, and SSW sectors) was not used.

Response:

The EAB distance of 503 meters, as stated in UFSAR Section 2.3.4.2.1, was the distance used by the NRC when calculating x/Q values during the Systematic Evaluation Program (SEP), Topic II-2.C. The shortest EAB distance per the above mentioned UFSAR table is 450 meters.

If recalculated based on 450 meters, the x/Q value increases slightly to 2.17E-6 sec/m3. Using this x/Q, the dose at the EAB increases from 2.16E-2 rem to 2.51E-2 rem, which is less than 1% of the 6.3 rem dose acceptance criteria. Therefore, the results of the analysis are not significantly affected.

In your response, dated July 14, 2004, you assumed no fuel melt for the rod ejection accident analysis. However, Ginna Updated Final Safety Analysis Report, Table 15.4-3, "Parameters Used in the Analysis of the Rod Cluster Control Assembly Ejection Accident," assumes that the fuel melt is less than 10%. Using an appropriate fuel melt assumption for this design bases event, please provide a sensitivity analysis of the radiological consequences.

Response:

Ginna UFSAR Section 15.4.5.6 does not quantify a melted core fraction, but only a failed fuel fraction. The 10% failed fuel and 0% fuel melt assumptions used in the analysis were referenced from the Ginna SEP (topic XV-12). In subsequent phone conversations, the NRC Reviewer suggested that 0.25% melt fraction was a typical and appropriate assumption. If the suggested 0.25% melt fraction is assumed, the calculated doses are:

	EAB (Limit = 6.3)	LPZ (Limit=6.3)	Control Room (Limit=5.0)
W/O Fuel Melt	6.64e-01	2.03e-01	1.06e+00
With 0.25% Melt	7.59e-01	2.31e-01	1.19e+00

As can be seen from the above, the increase in calculated dose is minimal as compared to the established limit and does not significantly affect the results of the analysis.

In your response, dated July 14, 2004, you included the Gas Decay Tank Rupture as a design bases event. Since there is no change in the source term for the Gas Decay Tank Rupture Analysis, the staff believes that this event need not be re-analyze using the AST. In addition, the staff does not consider this accident to be a design bases event. It is not listed or addressed in RG 1.183 or Standard Review Plan 15 and should, therefore, not be included in the license amendment request. Based on the above reasons, the staff recommends that this event be excluded from the license amendment request.

Response:

The analysis was performed primarily to evaluate the control room dose using the new x/Q for the different operating modes of the new CREATS. The source term was not changed from previous analysis. Therefore, this analysis is not within the scope of the alternate source term (AST) conversion, and does not require NRC review for that purpose.

2. CREATS Equipment Qualification

In your response, dated July 8, 2004, to RAI, No. 1, you stated that (a) Dampers and Duct Work in the Relay Room Annex, (b) Dampers and Duct Work in Stairwell, and (c) the Filter Units in the Relay Room Annex were seismically qualified in accordance with IEEE 344-1987, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Plants." Please provide a brief description of the method of analysis and the acceptance criteria used.

Response:

(a) Relay Room Annex Dampers and duct work:

Dampers were qualified by analysis following the recommended practices of IEEE-344/87. The Equivalent Static Coefficient method was used to develop seismic loading and perform the analysis. These analyses were performed by SSM Industries, are documented in seismic qualification report # 153-QS-03-2851, and comply with the requirements of Ginna Station specification ME-326.

Ductwork is qualified by analysis following the recommended practices of IEEE-344/87. A finite element model was developed and an Equivalent Static method applied to develop seismic loading and perform the analysis. These analyses were performed by SSM Industries, are documented in seismic qualification report # 153-QS-03-2851, and comply with the requirements of Ginna Station specification ME-326

(b) Stairwell Dampers and duct work:

Dampers are qualified by analysis following the recommended practices of IEEE-344/87. The Equivalent Static Coefficient method is used to develop seismic loading and perform the analysis. The analysis was performed by SSM Industries, documented in seismic qualification reports SSM # 163-QS-99-1899, and complies with the requirements of Ginna Station specification ME-326.

Duct work, dampers and corresponding damper supports will be added in the stairwell adjacent to the control room. The new equipment will be connected to existing duct work to complete the flow paths. Design analysis CEG DA-ME-2004-034 addresses seismic qualification of added dampers, duct work, and corresponding supports. The Equivalent Static Coefficient Method is used to develop seismic loading and perform the analysis. The analysis complies with the requirements of Ginna Station specification ME-326.

The new equipment is designed such that it does not add load to the existing duct work, and in fact acts to stiffen the existing ducts because of the well-supported dampers

added to the span. Therefore, no additional analysis is being performed for the existing ductwork.

(c) Relay Room Annex Filter Units (components and housings) are qualified by analysis, by AAF International Report # NESE 1110. An ANSYS finite element model was developed and run. Response spectra runs were made using the appropriate response spectra curves. The analysis and acceptance criteria comply with the requirements of Ginna Station specification ME-326.

In your response to RAI No.5, you stated that Ginna Design Specification ME-326 is used only for components located inside the Control Room and Relay Room Annex. Please provide criteria for the design of the air conditioning units mounted on the roof of the Relay Room Annex including seismic and design basis tornado wind loads.

Response:

The air conditioning units mounted on the roof of the relay room annex were seismically tested in accordance with the requirements of IEEE 344-1987 and are documented in TRENTEC report # 4Q007. Load combinations contained in design criteria for PCR 2000-024, Revision 1 (previously submitted on February 16, 2004), were considered for the tornado wind load qualification of these units, as documented in CEG DA-ME-2004, Revision 0.

In addition, please provide the status of the qualification effort including the appropriate 10 CFR 50, Appendix B documentation for all equipment affected by the modified CREATS and CRECS systems.

Response:

The below table has been revised/expanded from that shown in the above-mentioned July 8, 2004 response to show the requested level of detail.

Table of Equipment and Seismic Qualification

Equipment	Equipment Location	Qualification Method	Status of Testing/Analysis
CREATS Fans	Relay Room Annex	Testing IEEE 344-1987	Test performed by TRENTEC report # 3Q028 (Complete)

ATTACHMENT 1

Interconnecting duct/pipe between Control Room and Annex	See Ref. 3	See Ref. 3	DA-CE-2003-067 (Complete) DA-CE-2003-068 (Complete)
Dampers	Control Room above suspended ceiling	Analysis IEEE 344-1987 Spec. ME-326	Analysis Qualification reports: SSM # 163-QS-03-2851 (In Progress)
Dampers	Relay Room Annex	Analysis IEEE 344-1987 Spec. ME-326	Analysis Qualification reports: SSM # 153-QS-03-2851 (In review)
Dampers	Stairwell	Analysis IEEE 344-1987 Spec. ME-326	Analysis Qualification reports: SSM # 163-QS-99-1899 (In review)
Duct	Control Room above suspended ceiling	Analysis EPRI 1007896 Spec. ME-326	Analysis report: CEG DA-ME-2003-064 (Complete) CEG DA-ME-2004-039 (Complete)
Duct	Relay Room Annex	Analysis IEEE 344-1987 Spec. ME-326	Analysis Qualification reports: SSM # 153-QS-03-2851 (In review)
Duct	Stairwell	New - Analysis Spec. ME-326 Existing	CEG DA-ME-2004-034 (Complete) See (b) above
Filter Units	Relay Room Annex	Analysis IEEE 344-1987 Spec. ME-326	AAF International Report # NESE 1110 (In review)
Heaters	Relay Room Annex	Test IEEE 344-1975 (as endorsed by Reg Guide 1.100 rev. 1)	Test performed by Nutherm Qualification report # AAF-9099R (Complete)
Motor Control Center (MCC) Molded case circuit breakers and transformers	Relay Room Annex	Test IEEE 344-1987	Spectrum Technologies report # QTR04P0090 (Complete)

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Thermostats	Control Room	Test IEEE 344-1987	TRENTEC report # 4Q034.1 (Complete)
Relays	Various	Test IEEE 344-1987	AREVA Framatome ANP, Inc. Report (Complete) QR04-05 rev 1 (Complete)
Timers	Relay Room Annex	Test IEEE 344-1987	TRENTEC REPORT # 61393.0 (Complete)
Switches and indicators	Control Room	Test IEEE 344-1987	United Controls NQR-VCI-002, Rev 0 (Complete) NQR-1930, Rev 0 (Complete) CGDS-208, Rev 6 (Complete)
Air conditioning unit	Relay Room Annex	Test IEEE 344-1987	Test performed by TRENTEC report # 4Q007 (Complete)

3. Cable Separation

The following questions are with regards to your letter dated March 8, 2004, concerning cable separation and fault protection.

Are all Train A and B cables for the CREATS modification routed in separate cable trays or conduits? If the same cable tray is designated for both trains, please clarify how physical separation will be maintained in accordance with your licensing bases.

Response:

With the exception of two trays, all cables are routed in separate cable trays or conduits. In these two instances, although the same tray is used for both cables, different portions (lengths) of the tray are used. A detailed description of both instances where the same tray is used for both trains is provided below:

Tray 28N: This tray runs directly under the Auxiliary Benchboard (ABB), (beneath the floor) for the full length of the ABB. The penetrations from the Relay Room up through the floor into the ABB can only be accessed from Tray 28N. Therefore, all cables, from both trains, that enter the ABB must be routed in Tray 28N. The penetrations to be utilized for CREATS cables have been selected so that A train cables enter through Penetration CR-148-P in the west end of the ABB, while B train cables enter through Penetration CR-41-P in the center section, east of CR-149-P. Those penetrations are approximately 12" apart. From there, A train cables are routed directly west and B train cables directly east in Tray 28N, directly away from each other. Therefore, the closest that opposite train cables approach each other is the 12" separation where they enter the penetrations. After that, all of these circuits route back to the annex in separate trays.

Tray 180: Tray 180 contains the two circuits, one per train, that go to the Safety Injection (SI) cabinets to pick up the SI auxiliary contact used to initiate CREATS isolation. SIA1 rack and SIB1 rack are in two rows in the Relay Room, separated by a 36 inch wide aisle. Tray 180 runs across the aisle, directly above both racks. Therefore, the first tray encountered by any cable coming out of either train rack is Tray 180. Therefore, circuits C5663 (A train) and C5666 (B train) enter Tray 180, but three feet (the distance between the racks) apart. From there, the two circuits are routed into separate trays, going in opposite directions. At no point do the two circuits come closer than the initial three feet separation. The circuits stay in separate trays until entering conduit before penetrating into the annex.

Please confirm that the installation of the new power/control cables (480 VAC, 120 VAC, and 125 VDC) associated with the CREATS modification are designed such that:

(a) ***no single fault on any of the new cables can cause failure of both redundant trains of the CREATS or any other safety related systems.***

Response:

All 480 VAC cable is being routed in conduit. Regarding 120 VAC and 125 VDC cables, with the exception of the cables noted above, all control and power cables are routed in separate trays to ensure redundancy. To prevent failure of other safety related systems, the new CREATS cables are protected by isolation devices to ensure a fault in those cables will not propagate to other cables/systems (Attachment 3).

Please confirm that all pre-existing 120 VAC and 125 VDC control and power cables routed in the same cable trays containing CREATS cables have protective devices and are capable of clearing the most limiting fault such that no CREATS cables will be damaged. Discuss the results of the analyses (energy released and cable heat up) that supports this conclusion.

Response:

Existing cables in the trays being utilized for CREATS have protective devices installed (fuses or breakers) to protect the cables from sustaining a cable-damaging fault that would propagate across both CREATS trains. We previously sent to NRC the document titled "Topical Design Basis - Electrical Independence" (provided again as Attachment 2), which includes discussion relating to the original design reviews and original plant design criteria. Plant modification procedures are in place to maintain the design bases. To illustrate this, a representative analysis (Attachment 4) of the 125VDC system cables and 120 VAC Instrument Power system cables has been completed and demonstrates that these cables are protected within their cable damage curves. However, a comprehensive analysis does not exist to analyze every cable in the utilized relay room trays.

4. Control Room Radiation Monitor Analytical Limit

ATTACHMENT 1

In accordance with your letter dated September 10, 2004, you stated in Attachment 1, Design Analysis DA-EE-2001-013, that the limiting analytical limit (AL) for the control room radiation monitor is 0.91 mr/hr. The previous AL for these instruments was 0.96 mr/hr. Since the new AL is less conservative then the previous value, Design Analysis DA-EE-2000-009 for the control room radiation monitor setpoints should be revised to indicate the correct Limiting Safety System Setting (LSSS). Technical Specifications Table 3.3.6-1, "CREATS Actuation Instrumentation," should also be revised to reflect the new LSSS and the performance based operability requirements for the Channel Operational Test.

Per a phone conversation on 12/2/04, Mr. Clark (NRC) requested that the revised analysis (DA-EE-2000-009) pertaining to the new LSSS setting be included in this submittal for NRC review.

Response:

Design Analysis DA-EE-2000-09 has been revised to reflect the new AL and appropriate sections included as Attachment 5. The proposed updated revision to Technical Specification Table 3.3.6-1 is included as Attachment 6.

Attachment 2

Topical Design Basis
Electrical Independence

RO827387

TOPICAL DESIGN BASIS - ELECTRICAL INDEPENDENCE

Revision 0
06/19/97

Prepared by: 6.19.97
Reviewed by: 6/21/97
Approved by: 7-7/97

REVISIONS PAGE

Revision Number	Date	Section
0	06/19/97	all

TOPICAL DESIGN BASIS - ELECTRICAL INDEPENDENCE
Table of Contents

1.0 PURPOSE

The purpose of this Topical Design Basis Document is to describe the evolution of the licensing basis with respect to electrical independence for the Ginna Nuclear Power Plant.

Ginna Station was licensed to specific set of approved Atomic Industry Forum design criteria. When changes were made in the criteria, plant change documents were modified, as considered appropriate, and these criteria were used in plant modifications, to the extent practicable. The overall objective in modifying the plant's licensing basis, and the resultant plant changes, was to maintain plant configuration in accordance with current safety and regulatory criteria as much as reasonably possible.

Adherence to the principles described in this document will assist electrical design personnel in assuring the continued compliance with Ginna's design basis, licensing requirements, and related commitments made to the Nuclear Regulatory Commission.

2.0 DESIGN BASIS CRITERION DEVELOPMENT

Ginna Station was designed, constructed, and tested to standards that were selected with the prime objective of ensuring the safe and reliable operation of the plant under all anticipated conditions. Those standards were reviewed by the Atomic Energy Commission and were found adequate with respect to ensuring the design posed no undue risk to the public. Over time the standards applicable to commercial nuclear power units have evolved. Ginna Station does not meet all modern (current) standards for cable separation. There is, however, an original design basis for cable separation and electrical isolation and there are standards which must be achieved to ensure compliance with our current licensing basis.

Functions which are important to safety (essential functions) must be preserved sufficiently such that no credible event places the public at unreasonable risk. With respect to preserving the essential functions of cables and electrical wiring there are three areas of interest; physical separation, electrical isolation, and cable material selection.

When taken together, physical separation and electrical isolation should work to achieve a desired level of electrical independence - for Ginna Station this translates to: the state in which there is no credible mechanism by which any single design basis event can cause a loss of the functions credited with mitigating that event.

To best understand Ginna Station's electrical independence

criteria it is logical to examine the history of Ginna Station from its conceptual design and construction, through the Systematic Evaluation Program, and then through the Appendix R Fire Protection evaluation. Each of these represents a milestone in the history of cable separation and electrical isolation and each added to, or otherwise modified, the cable criteria maintained by Ginna Stations's current licensing basis.

The Updated Final Safety Analysis Report (UFSAR) contains numerous, but widely distributed, references to electrical separation, independence, and circuit protection methods (see also Table 2 - Updated Final Safety Analysis Report / IEEE Standard cross reference). Those descriptions reflect the results of the Ginna Station design basis but not necessarily the basis itself. When reading the UFSAR care should always be given to fully understand the framework for which the text is crafted. Statements which are appropriate to a specific circumstance are not always transportable to another case, even if the subject matter appears to be the same.

2.1 Physical Separation

Physical separation is a term used to describe the use of free space or barriers interposed between redundant functional devices important to safety. The concept is to provide sufficient autonomy between the devices so that there is no single mechanism which can render both simultaneously inoperable.

The conceptualization and design of Ginna Station reflects the design engineers' knowledge that it was necessary to protect the operability of certain pieces of equipment from the effects of random and common-mode failures. That is, certain functions must be performed following a given design basis event to ensure the event is successfully mitigated. Equipment important to safety has to be relied upon to achieve their functions even in the event of the worst case postulated event. As described in early Westinghouse design documents (i.e. WCAP 7486), "the principal defense against random component failures is the employment of redundancy; that is, if a component failure can give rise to problems in a vital circuit, a redundant component provides an identical function so that one component (or train) can fail without impairment of function. Simple redundancy does not, however, provide significant defense against common-mode failures unless coupled with other techniques. To prevent occurrence of such failure, it is necessary to employ such measures as functional diversity, physical separation, testing, and equipment diversity...".

Functional diversity was primarily applied to the reactor protection system. Functional diversity relies upon monitoring and utilizing the maximum number of process variables such that completely eliminating the sensing of one variable would not lead to a failure of the protection system to perform its function.

Equipment diversity refers to the use of multiple equipment of different design or manufacture to perform identical or

equivalent redundant functions. Physical separation offers a measure of defense against common-mode failures EXCEPT against unrecognized functional (design) deficiency and those situations where the causative factor is widespread in its effects. The general premise of physical separation is that a designer can aid in eliminating the effects of some unrecognized system design deficiencies by being alert to possible interdependencies and making conscious efforts to avoid introducing the potential for common-cause failures. [Separation protects redundant equipment from common-cause failure, diversity protects important functions from the effects of design deficiencies.]

Physical separation is an important factor in minimizing the effects of external hazards such as fires or missiles. Separation must be considered a design feature whose function is to increase a system's resistance to common-mode failures.

Implementation of Ginna Station's original electrical separation design philosophy is best described by Westinghouse document E-EPS-1, Issue 3, Electrical Systems Recommended Design Basis, ELECTRICAL CIRCUIT PHYSICAL SEPARATION which states in part:

> [Author's note: These criteria are illustrative. The design and construction of Ginna Station predates present industry standards of physical separation, but have a continuity to these standards. E-EPS-1 is an example that shows the thought and process used by the design engineers at the time of Ginna's design and construction. The link between E-EPS-1 era documents and later standards, i.e. the engineering thought and process, is that the same engineers often developed both. For instance, R. I. Hayford signed the approval of E-EPS-1 as Manager, Electrical Power Systems, Westinghouse; Hayford was also a member of the Nuclear Power Engineering Committee at the time of approval

of IEEE 384 Independence of Class 1E Equipment and Circuits.]

The electrical supply and control conductors for redundant or back-up circuits of a nuclear plant should have such physical separation as is required to assure that no single credible event will prevent operation of the associated function by reason of electrical conductor damage. Critical circuits and functions include power, control and analog instrumentation associated with the operation of reactor protection, engineered safeguards, reactor shutdown and residual heat removal systems. Credible events shall include, but not be limited to, the effects of short circuits, pipe rupture, missiles, etc. Such electrical separation as is required for protection against plant design basis events should be included in the basic plant design.

General

- Cables of redundant or back-up circuits shall be run in separate conduits, cable trays, ducts, penetrations, etc.
- Where it is impractical for reasons of terminal equipment arrangement to provide separate wireways, cables of redundant or back-up circuits shall be isolated by physical barriers, be in separate metallic conduit, or consist of suitable armored cable.
- One foot horizontal or three foot vertical separation shall be maintained between wireways (or armored cable) associated with redundant equipment.
- Power and control conductors rated at 600 volts or below shall not be placed in wireways with conductors rated above 600 volts.
- Low level analog signal conductors shall not be routed in wireways containing power or control cables.

[In the original design of Ginna Station, safety related analog instrumentation circuits used for reactor protection and safeguards actuation were routed in dedicated conduits from the instrument sensor to the Protection Racks in the control room. Display and indication functions were isolated from the safety related portion of these instrument channels, and their signal cables were routed in non-safety related instrument cable trays. At that time, all display and indication functions were considered non-safety related, and to this day, all cable trays designated as instrument trays are still classified as non-safety. After TMI and the issuance of NUREG 0737 and Reg. Guide 1.97, many new instruments were installed with safety related indication and control functions. Since the non-safety related

instrument cable trays could not be utilized for these new circuits, they were routed in safety related control and power trays. These new circuits utilized shielded cables with 600 volt insulation rating. For this reason, several low level analog signal cables are now existing in power and control cable trays (see also section 2.5, Assignment to Designated Trains).]

- In congested areas, such as under or over the control boards, instrument racks, etc., wireways shall be identified using permanent markings. The purpose of such markings is to facilitate cable routing identification for future modifications or additions.
- Positive, permanent identification of cables and/or conductors shall be made at all terminal points.

As previously explained, during the period Ginna Station was designed and constructed many of the codes and standards available today were either non-existent, in draft, or of an early revision. In 1967 plant designers relied upon the proposed Atomic Industrial Forum (AIF) General Design Criteria (GDC) in defining the safety objectives and approaches necessary to incorporate those objectives into Ginna's design. A description of those criteria, along with a comparison to the GDC later contained in the Code of Federal Regulations (CFR), is maintained in the UFSAR, section 3.1, Conformance with NRC General Design Criteria. In 1972, when the plant submitted an application for a full term operating license, the application contained a supplement which examined the adequacy of the design with respect to IEEE standards 279-1971, 308-1971, 317-1971, 323-1971, 334-1971, 336-1971, 338-1971, 334-1971. This examination is discussed in UFSAR section 1.8.3, Conformance to IEEE Criteria.

The acceptance of the adequacy of Ginna Station's original

design and construction by the Atomic Energy Commission (later to become the Nuclear Regulatory Commission) was documented in an AEC Safety Evaluation Report. A provisional operating license was granted to Rochester Gas and Electric Corporation for Ginna Station on September 19, 1969. The AEC Safety Evaluation Report SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING U. S. ATOMIC ENERGY COMMISSION IN THE MATTER OF ROCHESTER GAS AND ELECTRIC CORPORATION ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1, dated 06/19/69 states, in part:

> Section 3.7.3 The applicant's criteria relating to the cable tray loading and separation may be summarized as follows:
>
> (a) cables, whether power, control, or instrumentation of one train or system are not mixed with cables of a redundant train or system;
> (b) physical separation is provided between redundant cables for control and instrument systems within a tray by means of a galvanized sheet metal barrier in cable trays;
> (c) the minimum physical dimensions between redundant power, control, and instrument cable trays are 5 inches vertical separation and 2 inches horizontal separation;
> (d) metal-enclosed 4160-volt buses are used for all major bus runs where large blocks of current are carried; and
> (e) the routing is such as to minimize exposure to mechanical, fire and water damage.
>
> An ambient temperature of 50 degrees C within the reactor containment and an ambient temperature of 40 degrees C in all other plant areas is the design basis for all power cable ratings.
>
> All a.c. circuits within the plant are protected by three-phase circuit breakers.
>
> We have reviewed the criteria and conclude that they reduce the possibility of cable fires, and provide protection against random and systematic failures. Our conclusion with respect to the low probability of cable fires is based on

> the limited cable tray loading, and upon derating factors. There is only one layer of 4160-volt cable in a tray, and a derating factor of 0.81 is used. For the 480-volt cables, a derating factor of 0.6 has been used for size #4 and larger, and 0.5 for size #6 and smaller. Further, the pressurizer heater cables have been given extra spacing, and have been derated by a factor of 0.5.
>
> With respect to systematic and random failures, we conclude that the physical separation of redundant cables, and the metal barrier (where used) within a tray provide adequate protection against the propagation of a fire, and against any lesser single event occurring within a tray. The use of three-phase breakers in lieu of fuses should immediately isolate all three phases of a line from a fault occurring in one phase.

As more standards became codified it became questionable to the nuclear regulatory body (the transition had occurred from the Atomic Energy Commission to the Nuclear Regulatory Commission, or NRC) if plants of Ginna Stations vintage provide sufficient protection when judged against the (then) newer standards. In order to resolve these concerns, a regulatory initiative known as the Systematic Evaluation Program (SEP) was begun at Ginna Station in 1978. This effort, completed in December, 1982, documented the acceptability of Ginna Station's design (after plant modifications) with respect to the intent of the present day design criteria as documented in Appendix A to 10CFR50. That is, the SEP review essentially sought to demonstrate functional equivalency between Ginna's design and a plant designed to newer construction standards.

Included in the SEP were events and causal effects which were not considered in the original licensing basis. One of the major differences between Ginna Station's original design basis and the SEP review criteria is consideration of a series of

mitigative requirements necessary to achieve safe shutdown accounting for the effects of more stringent internal, external, or "special" events. The inclusion of the mitigative features for these increased requirements into the licensing basis reflected a fundamental change in which equipment functions, and hence cables, were required to be protected.

The SEP not only validated the adequacy of the plant design against the original design basis, but also necessitated commitments to maintain a plant design commensurate with new principles. [i.e., Some of these commitments dealt with the physical protection of cables which did not meet the separation standards which would be imposed on newly constructed systems of equal importance.] The list of credible events was expanded and now included different postulates than were explicitly detailed in the original design basis (e.g. high energy line breaks outside containment). RG&E was required to demonstrate the ability to protect the public from the consequences of all the design basis events by demonstrating the ability to shut down the reactor and remove decay heat in a safe and acceptable manner.

The equipment set necessary to mitigate the newly defined events was bound, in most cases, by the set certified to mitigate the original design basis events. In these cases, the NRC closed the review of the SEP topic by providing a Safety Evaluation Report (SER) detailing the acceptance of our compliance methodology (including the acceptance of the adequacy of any

necessary proposed modifications). These reports became integral to the plant current licensing basis.

Those cases where the plant conformance with the then-current criteria was of concern to RG&E and the NRC, an evaluation consistent with the principles of 10CFR50.109 was conducted.

A set of ground rules needed to be agreed upon in order to establish conformance standards. Chief among the issues was the question of whether or not the facility needed to engineer the solutions to the expanded event and hazard set such that the equipment group necessary to achieve safe shutdown must withstand the postulated event while sustaining an active single failure. The mitigation of an event while accounting for the effects of random single failure in the equipment set used to manage that event is one of the reasons why important systems are designed with at least two trains (or divisions).

Many SEP SERs were written acceding to the principle that the consequences (damages) sustained during the newly postulated (beyond original design basis) events embody the single failure the plant needs to be designed to withstand (e.g., tornado). That is, so long as the event did not act as a precursor to a design basis event then the event effects, and the cascaded consequences of those effects, were what was needed to be mitigated. This ideology promulgated two important concepts; functional equivalency and safe shutdown.

The UFSAR, section 7.4, describes the minimum systems

required to take the reactor from operating to shutdown conditions (SEP topic VII-3). These systems, and their functional alternatives, are described in UFSAR Table 7.4-1. ONE OF THE PRIMARY GOALS OF GINNA'S CABLE SEPARATION DESIGN BASIS IS TO ENSURE THAT THE SAFE SHUTDOWN FUNCTIONS ARE SUSTAINED GIVEN THE POSTULATED EFFECTS OF DESIGN BASIS ACCIDENTS, TRANSIENTS AND EXTERNAL EVENTS.

In defining the equipment set referred to above it is important to understand and distinguish between terms used to describe an equipment set's attributes. Often equipment is "binned" in broad quality group categories such as "safety related", "safe shutdown", "class 1E" and the like. Typically, specific codes and standards apply to the procurement, fabrication, installation and testing of equipment commensurate with their importance to safety. Likewise, these quality labels each evoke a specific level of design control. CAUTION MUST BE EXERCISED WHEN EXAMINING THE ROUTING AND SEPARATION CRITERIA THAT IS APPLIED TO THE ELECTRICAL EQUIPMENT AT GINNA. As described in the various SEP SERs, circuits which provide alternative functional equivalency do not always meet the separation criteria associated with their respective quality group classification. They must, however, always meet the criteria associated with their licensed function. An easy to understand example of this issue can be shown by examining the auxiliary and standby auxiliary feed water systems.

The auxiliary feedwater pump motors are located next to each

other in the Intermediate Building. If we judged the adequacy of their separation using IEEE 384-1974 (Independence of class 1E equipment and circuits) we would find them not in compliance. They are susceptible to a common-mode failure caused by a steam line break in the area. In order to achieve safe shutdown a diverse system, standby auxiliary feedwater, was installed. This system is protected against the effects of a steam line break, yet individually, it too does not meet all of the IEEE hazards separation requirements. Yet together, they provide sufficient independence to achieve the licensed objective of providing cooling water for decay heat removal during all events feedwater is required for, even while accounting for the effects of an additional single active failure (e.g. a diesel failing to start, a pump motor failure, etc.). THUS, IT IS POSSIBLE THAT MORE SPECIFIC SEPARATION CRITERIA IS APPLICABLE TO A PARTICULAR FUNCTION THAN IS TYPICALLY IMPLEMENTED BASED ON QUALITY GROUP CLASSIFICATION.

During the period the SEP program was underway one other regulatory initiative was undertaken that had significant impact on Ginna's cable separation design basis: 10CFR50 Appendix R - Fire Protection for plants operating prior to January 1, 1979.

Although Ginna was licensed to the fire protection requirement of Branch Technical Position BTP 9.5-1 the unit committed to sections G, J and O of 10CFR50, Appendix R. Part G, Fire Protection of safe shutdown capability, contains specific cable protection and separation criteria which must be achieved

thus ensuring one train of the systems necessary to achieve and maintain hot shutdown are maintained free from fire damage. For RG&E this includes the alternative shutdown system components. Ginna's strategy is described in the UFSAR section 7.4.4, Alternative Shutdown System and section 9.5.1.4, Safe Shutdown Capability.

The Appendix R Alternative Shutdown System Report is the analysis which describes in detail the modifications that were necessary to demonstrate the ability to achieve safe shutdown (including physical protection of safe shutdown cables and cable re-routs) following the effects of a fire. Basically, all the circuits required to achieve (single train) safe shutdown either meet IEEE 384 standards for physical protection with respect to the hazard in a particular fire zone, or an alterative way of achieving the associated function was found which was not susceptible to that same fire hazard. The specific separation criteria employed to ensure compliance with this strategy is complex and is implemented by Ginna's Fire Protection/Appendix R conformance verifications. All wiring and cable modifications made at Ginna must be screened through the Appendix R conformance verification process. Like the equipment credited in the SEP the licensing basis for the Appendix R equipment is complicated. The alternative shutdown process utilizes both safety and non-safety quality group equipment. Additionally, the equipment set used to achieve safe shutdown given a fire in a specific area may not be the set required if the fire is in a different area. GINNA'S

CABLE SEPARATION DESIGN BASIS MANDATES THAT CAREFUL CONSIDERATION MUST ALWAYS BE GIVEN WHEN ADDING OR RE-ROUTING BOTH SAFETY AND NON-SAFETY CABLES, EVEN IF THE CHANGE DOES NOT INVOLVE A FUNCTIONAL CHANGE FOR THE AFFECTED EQUIPMENT. Because of the results of changing requirements all the functions credited to a cable, cable routing, cable tray, or conduit are not always readily apparent. MAINTAINING GINNA'S CABLE SEPARATION DESIGN BASIS INCLUDES ACCOUNTING FOR FUNCTIONAL AND PHYSICAL DIVISION. SYSTEM EQUIPMENT MAY BE DIVERSE AND SEEM PHYSICALLY UNRELATED YET BE CREDITED TO ACHIEVE THE SAME SAFE SHUTDOWN FUNCTION GIVEN THE EFFECTS OF A POSTULATED FAILURE. For example; Safety Injection is used to provide primary system makeup in the event of a fire in the charging pump room. Therefore during plant modifications consideration must be made of all the cables and equipment associated with that SI function so that it is not possible to fail both the SI and the charging function because of one fire. [Examining the Appendix R success path matrix contained in the safe shutdown report will show further detail of equipment inter-dependencies and functional equivalencies.] CARE MUST ALWAYS BE EXERCISED TO ENSURE THAT COMMON-MODE FAILURES ARE NOT INTRODUCED BETWEEN FUNCTIONALLY REDUNDANT EQUIPMENT SETS CREDITED WITH MITIGATING THE CONSEQUENCES OF AN EVENT.

2.2 Electrical Isolation

Along with the spatial analysis necessary to achieve physical separation, Ginna's separation design basis also

accounts for electrical isolation. Electrical isolation provides a means of ensuring that associated circuits do not introduce a failure mechanism in a circuit that performs an important function. Electrical isolation is also the means used to ensure that a fault in a load will be interrupted prior to its causing a failure of the cable supplying power to the load. In vital applications electrical isolation devices are also typically coordinated such that the device closest to the fault provides circuit interruption prior to failing the sources' protection devices (breaker selective tripping and DC fuse coordination). The UFSAR section 8.3.1.1, The 480V System, provides a description of breaker coordination for vital buses. Sections 8.3.1.2.4.6 and 8.3.2.3 provide descriptions of fuse coordination.

With respect to cabling, the consequence of an uninterrupted load fault is typically a fire. UFSAR section 8.3.3, Fire Protection for Cable Systems, summarizes the methods by which AC breakers and DC fuses are credited with protecting vital cables.

As discussed in the Appendix R submittal, non-vital cables are routed in trays containing vital cables (associated circuits). The acceptability of this practice is predicated on the proper use of isolation devices. If the cable has appropriate electrical isolation it will not create a secondary fire in a different area than where the source fire introduced the fault.

Having isolation devices does not, however, give liberty to

randomly mix divisions or trains within conduits or cable trays. As discussed in section 2.1, whenever redundant equipment or systems are at risk from the consequences of an single, cascaded failure (event), an alternative means must be available to achieve the at risk function, or the primary provider must be protected.

Ginna Station's licensing basis does contain exceptions to the generally accepted electrical separation and isolation criteria. Ginna Station has committed to the NRC to provide certain post-accident instrument displays (UFSAR section 7.5.2, Safety Parameter Display). Some portions of the instrument loops which drive the displays are associated with non-vital control functions. This is because Ginna Station's original design isolated the protection portion of the loop from the control portion. This configuration ensured that a fault in a non-vital process control circuit would not affect the protection channel. In this arrangement a specific train's control and display signals are not electrically independent. Additionally, because of the control room design (see section 2.4), redundant channels may not have complete physical separation. In the SER reviewing Ginna's conformance to Reg. Guide 1.97 (2/24/93) the NRC examined Ginna's configuration and found the redundancy and separation acceptable for those Category 1 variables where instrumentation is not upgraded to meet the Reg. Guide 1.97 recommendations. As part of that acceptance RG&E committed to perform safety analysis of future modifications where separation of Category 1

instrument channels can not be maintained to IEEE standard 384-1981.

2.3 Relating the electrical independence design criteria to the separation and independence criteria described in the Updated Final Safety Analysis Report

The UFSAR includes information that describes the facility, establishes the design basis and the limits of operation, presents a safety analysis of the structures, systems and components as a whole, and other information detailed in 10CFR50.34. The information is intended to provide the NRC staff with a basis for determining that the plant can be operated without undue risk to public health and safety. Ginna Station's UFSAR is not a compendium of commitments with respect to equipment engineering and design details.

All modifications to the station require a safety review or safety analysis. As part of these reviews the UFSAR is examined. The examination for topics related to electrical separation, isolation and independence will yield instances where configuration descriptions and design statements are softened with; "generally", "to the extent practicable", "where possible", etc..

The phraseology is crafted to demonstrate the understanding that Ginna Station's licensing basis contains deviations to some specific codes, standards and construction criteria (as described in sections 2.1 and 2.2 above). Those "generally" and "to the

extent practicable" cases are specifically defined in the licensing basis; the wording is not to be viewed as authorization to add additional nonconforming cabling simply because it is acknowledged that exceptions do exist. As compelled by GDC 1, "... Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function."

2.4 Special Case: Cable Convergence in Racks, Panels, and Control Boards

As described in the plant design basis, every attempt has been made to maintain cable separation within the applicable codes and standards. The size and configuration of the Main Control Board (MCB), and system instrument racks, panels, and enclosures in the Control and Relay Rooms of the Control Building require that cables converge in these areas. Within some of the above enclosures, cable termination point physical limitations have made it impractical to maintain minimum separation distances or to install the fire barriers or cable shielding necessary for train separation. [Plants designed and constructed after Ginna also grappled with defining separation criteria for these areas and enclosures.] As noted in Regulatory Guide 1.75, Physical Independence of Electrical Systems, "the degree of separation required varies with the potential hazards in a particular area". The above referenced areas are special in that control of the

space manages hazards and thus limits damage precursors to fires and the consequences of random equipment (in this context; cable) failures. The consequences of a fire have been discussed in the Appendix R analysis. The risks associated with a random cable failure at one of the convergence points during a design basis event must be managed.

As detailed in R.G. 1.75, section 5.1.1.2, "In those areas where the damage potential is limited to failures of faults internal to the electrical equipment or circuits, the minimum separation distance can be established by analysis of the proposed cable installation". Per the guide, this applies to both the Main Control Board and Instrument Cabinets. The design basis separation is therefore the offset distances or physical shielding described by the applicable code <u>or</u> the analysis of the installation, [including fault isolation] and the suitability of the flame-retardant characteristics of the cable material.

2.5 Assignment to Designated Trains

It must be acknowledged that the system of cable trays (raceway system) at Ginna Station was not designed to accommodate segregated routing of designated trains and channels on a comprehensive basis. [This is a designation or labeling issue; as discussed throughout this document, functional separation was a routing objective.]

Where a choice of several cable trays was available, the routing assignments were straight-forward. In certain locations

the structural configuration precludes redundant raceways, in these cases other schemes such as barriered trays or shielded cable were used (see section 2.1). [In order to abate the {at the time of construction} emerging issue, certain areas also had enhanced fire detection and suppression capability installed. The design philosophy was to engineer an equivalent level of protection as would be afforded by a redundant raceway system.] In addition, while certain statements in the FSAR refer to separating "all redundant" circuits by train (or channel) it is clear that this was used in a sense that permitted routing of cable with one train designation in raceway with the redundant train designation if the redundant cable was not present in that tray or that section of tray. [i.e. There are cables for redundant components that enter the same tray, then the cables are routed in opposite directions such that when they enter a hazards area they have achieved separation.] Hence, a tray of either train designation may have cable from redundant systems occurring at specific locations along its length.

There was a significant effort to maintain physical separation consistent with train designation for documentation purposes but this was not considered as important as functional separation. If routing a cable in a raceway of the redundant train achieved physical separation with its redundant function, that routing was permitted. When there was an ambiguous train assignment, such as with power to the "swing" SI pump, or control for diesel generators, which have provision for automatic

transfer between A and B trains, separation was also not always consistent with train designation.

Additionally, during construction, when routing redundant circuits in areas with no exposure to significant internal or external hazards, and where routing paths had restricted raceway availability, the rigorous separation of redundant circuits was not always of prime concern.

The design strategy of objectively separating all circuits on the basis of train designation alone was not standardized at the time of Ginna Station's construction. There was no consensus standard for acceptable "arbitrary" physical separation on this basis until IEEE-384 was published in 1974 (IEEE Standard 384-1974, Trial-Use Standard Criteria for Separation of Class 1E equipment and Circuits). All statements made in the FSAR regarding routing, train designation and physical separation of circuits must be interpreted in the context of these issues.

2.6 Criteria Development General Summary

In summary, Ginna Station's design basis for electrical independence incorporates the principle of functional separation. For equipment used in mitigating UFSAR Chapter 15 accidents and design basis earthquakes, cable separation and isolation must support reaching Hot Shutdown conditions assuming the initiating event, one single credible failure, and all cascading effects. For the equipment used to mitigate the remaining external events, cable separation and isolation must support reaching Hot Shutdown conditions assuming the initiating event and its cascading effects only. For equipment used in mitigating fire events, cable separation and isolation must support reaching Cold Shutdown within 72 hours assuming the initiating event and cascading effects only.

Table 1

Historical Milestones

Oct 1965	Rochester Gas and Electric Corporation applies to the Atomic Energy Commission (AEC) for a Construction/Operating License for Ginna Nuclear Power Plant.
Nov 1965	The AEC issues 27 proposed General Design Criteria (GDC) which, in part, required electrical separation to be considered in the design of the Reactor Protection System and Engineered Safety Features Systems.
Apr 1966	Construction Permit issued by the AEC.
Oct 1967	Cable tray installation begins in Ginna Containment.
1968	Review, discussion, and rerouting of cable installation and separation to address AEC concerns (RG&E, Westinghouse and Gilbert Associates).
1968	Development of Institute of Electrical and Electronics Engineers (IEEE) CRITERIA FOR NUCLEAR POWER PLANT PROTECTION.
Jan 1968	Submittal of Final Safety Analysis Report to AEC.
Jul 1969	Westinghouse document E-EPS-1 Electrical Systems Recommended Design Basis, ELECTRICAL CIRCUIT PHYSICAL SEPARATION.
Jun 1969	SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING U. S. ATOMIC ENERGY COMMISSION IN THE MATTER OF ROCHESTER GAS AND ELECTRIC CORPORATION ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1.
Sep 1969	RG&E receives Provisional Operating License.
Nov 1969	Initial criticality of Ginna reactor.
Mar 1970	Reactor achieves 100% power.
1971	Final Safety Analysis Report reviewed against IEEE 279 STANDARD FOR PROTECTION SYSTEMS.
1972	RG&E applies for full term license.

3.0 DEFINITIONS

The definitions that follow include both those in use during the design/construction phase of Ginna Station as well as those in present use by industry. Definitions provided are for use in this document only.

<u>Active Safe Shutdown Components</u>

Components that must actively operate (i.e., run) to achieve safe shutdown, or change their operating state or position from the normal position or operating state (i.e., stop, open, close, etc.).

<u>Alternative Shutdown</u>

Safe shutdown activities requiring utilization of operational practices including:

(a) Other than normal safe shutdown activities from the Control Room

(b) Operations from designated alternative control systems location

(c) Manual operations at various equipment locations

<u>Appendix R Fire</u>

Achieve cold shutdown within 72 hours f· lowing a single fire with no additional failures.

<u>Associated Circuits</u>

Non-Class 1E circuits that share power supplies, signal sources, enclosures, or raceways with Class 1E circuits and are not physically separated or electrically isolated from Class 1E circuits by acceptable separation distance, barriers, or isolation devices. Also, circuits not considered as part of safe shutdown circuits, but whose fire-induced failure could prevent the proper performance of safe shutdown system functions. Associated circuits of concern are categorized by common power supply, common enclosure, and spurious operation.

<u>Associated Circuits By Common Power Supply</u>

Cables not required for safe shutdown whose fire-induced failure could cause the loss of a power source (power, control or instrument bus) that is necessary to support safe shutdown.

Associated Circuits By Common Enclosure

(a) Cables not required for safe shutdown whose fire--induced failure could cause circuit faults in electrically unprotected cables such that secondary fires may occur in enclosures (raceways, panels, etc.) outside the fire area of concern and damage safe shutdown equipment or cables.

(b) Cables that would allow fire to spread by burning beyond the immediate area of concern, ultimately affecting redundant safe shutdown equipment or cables.

Associated Circuits Due To Spurious Operation

Those circuits that could, by fire-induced failures, cause safe shutdown or equipment not required for safe shutdown to mal-operate in a way that defeats the function of safe shutdown systems or equipment, including high/low pressure interfaces and circuit interlocks from instruments and control circuits not required for safe shutdown.

The concept of associated circuits, as it relates to the plant safe shutdown in the event of a fire, is included in the Rochester Gas and Electric Corporation's Appendix R Report.

Auxiliary Supporting Features

Systems or components which provide services (such as cooling, lubrication, and energy supply) which are required for the safe shutdown system to accomplish its functions.

Cable Failures

Open circuits, short circuits, shorts to ground and cable to cable hot shorts.

Channel

An arrangement of components and modules as necessary to generate a single protective signal within the plant protection system when required by a plant condition. A channel's identity is lost after passing through an isolation device. A plant protection system channel's identity undergoes transition to a safety system train at the logic relay coil/relay contact interface.

Class 1E

The Institute of Electrical and Electronics Engineers (IEEE) safety classification of the electric equipment and systems that

are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive materials to the environment.

Design Bases

That information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system or component must meet its functional goals. (10CFR50.2)

Design Basis Accident

Accidents addressed within Chapter 15 of the UFSAR. These events require the consideration of a credible single failure in addition to the accident initiator.

Design Basis Event

Per 10CFR50.49(b)(1) includes:

a. Conditions of normal operation
b. Anticipated operational occurrences
c. Design basis accidents
d. External events
e. Natural phenomena

Items a, b, and c require consideration of a credible single failure in addition to the accident initiator. Items d and e are essentially the same category of events (i.e., an initiating event external to the plant systems) and are referred to as external events hereafter. Included within external events are earthquakes, tornadoes, hurricanes, floods, and tsunami (GDC 2). Hurricanes and tsunami are not credible design basis events given Ginna Station's location. Mitigation of an earthquake requires consideration of a credible single failure while tornadoes and floods do not.

Division

The designation applied to a given system or set of components that enable the establishment and maintenance of physical, electrical and functional independence from other redundant sets of components. This terminology envelopes both train and channel.

Essential Function

A function which assists in managing the ability to remove heat from the reactor core. Essential functions include:

- Control of reactivity
- Control of reactor coolant system pressure
- Control of reactor coolant system inventory
- Heat removal from the reactor coolant system

Equipment that is safety related or important to safety perform essential functions.

High/Low Pressure Interface Components

Components that have the potential of causing uncontrolled or unrecoverable depressurization and loss of primary coolant.

Important to Safety

Systems, structures and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. (See also 10CFR50, appendix A, General Design Criteria for Nuclear Power Plants, Introduction and 10CFR50, Appendix R, Fire Protection Program for Nuclear Power Facilities, Introduction and Scope.)

Independence

The state in which there is no mechanism by which any single design basis event, such as a loss of coolant accident (LOCA), can cause redundant equipment to be inoperable.

Interaction

A direct or indirect effect of one device or system upon another.

Internal Tray Divider

A strip of metal used in trays in order to separate cables due to service class requirements.

Isolation Device

A device in a circuit that prevents malfunctions in one section of an electrical circuit from causing unacceptable effects in other sections of the circuit or in other circuits. Acceptable isolation devices for power circuits are single isolation devices actuated by fault currents (breakers and fuses). For low energy control and instrumentation circuits, acceptable isolation devices are those actuated by fault currents (e.g., fuses or circuit breakers), relays, control switches, transducers, isolation amplifiers, current transformers, diodes, and fiber optic couplers.

Licensing Basis

All changes to the *original design basis* as reflected in commitments made to the NRC and/or as documented in the UFSAR. The original licensing basis therefore is equivalent to the original design basis.

Local Controls and Indications

Components located outside of the Control Room near the equipment or at local control panels.

Missile Producing or Missile Hazard Area

The following areas are considered missile producing or missile hazard areas due to the fact that they are in direct line with a potential projectile generated by a malfunction of machinery or equipment:

Physical Barrier

A partition, cover, wall, floor, or ceiling which provides protection between redundant safety system components and/or cabling when adequate physical separation is not provided.

Raceway

Any device that is designed and used expressly for supporting or enclosing wires, cables, or bus bars. Raceways consist primarily of, but are not restricted to, cable trays, conduit, and cable risers.

Redundant Equipment or System

An equipment or system that duplicates the essential function of another equipment or system to the extent that either may perform the required function regardless of the state of operation or failure of the other.

Safe Shutdown

A condition that exists when the plant achieves and maintains:

(a) the reactor subcritical;

(b) the reactor coolant inventory such that pressurizer level is within the indicating range;

(c) the reactor heat removal function such that it is capable of removing the decay heat being generated; and

(d) the process monitoring function such that it is capable of providing direct readings of the process variables necessary to perform and control the above functions.

Safe Shutdown Equipment

Equipment (i.e., components, support components, cables, piping, raceways) that may be used for achieving and maintaining safe shutdown in the event of a fire in a plant area.

Safe Shutdown Functions

The safe shutdown functions are reactivity control, reactor coolant make-up, reactor coolant pressure control, reactor heat removal, process monitoring, and support functions.

Safety Function

One of the processes or conditions (for example, emergency negative reactivity insertion, post-accident heat removal, emergency core cooling, post-accident radioactivity removal, and containment isolation) essential to maintain plant parameters within acceptable limits established for a design basis event.

Safety Related

That term used to call attention to safety classifications incorporated in the design, installation and documentation of the system. Safety related systems, structures and components are those relied upon during or following design basis to assure:

- the integrity of the reactor coolant pressure boundary
- the capability to shutdown the reactor and maintain it in a safe shutdown condition, or
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.

Safety System

Those systems (the reactor trip system, an engineered safety feature, or both, including all their auxiliary supporting features and other auxiliary features) which provide a safety function. A safety system is comprised of more than one safety group of which any one safety group can provide the safety function.

Train

One set of equipment required to achieve safe shutdown.

Vital Circuit

Circuit which serves equipment important to safety.

Table 2

IEEE References in FSAR

IEEE Standard	FSAR Section			
279-1971	7.1.2 8.3.1.2.7.1	7.2.1.1	7.2.5.1 10.6.2.9	7.3.3.1
308	8.3.2.3			
308-1971	1.8.3.2	8.1.4.3		
308-1974	8.3.1.2.7.1			
317	1.8.3.3	8.3.1.3		
317-1971	8.1.4.3			
323	6.2.1.5.3			
323-1971	1.8.3.4	8.1.4.3		
323-1974	6.2.1.5.2			
323-1983	8.3.2.1.2			
334-1971	1.8.3.5			
336-1971	1.8.3.6	8.1.4.3		
338-1971	1.8.3.7			
344-1971	1.8.3.8	8.1.4.3		
344-1975	3.10.1.2	6.2.1.5.2	8.3.1.1.4	8.3.2.1.2
379-1072	8.3.1.2.5			
383	9.5.1.2.4.8			
384-1974	8.3.1.2.5	9.5.1.2.4.8		
384-1981	8.3.2.3			
450	8.3.2.1.1			

4.0 DESIGN BASIS

4.1 Design Basis Summary

The columns in Table 3 are shown chronologically, left to right, from design and construction to present day.

One of the primary goals of Ginna Station's cable separation design basis is to ensure that the safe shutdown functions are sustained, given the postulated effects of design basis accidents, transients and external events. Therefore, caution must be exercised when examining both the routing and separation of cable and electrical equipment, and the criteria that is applied to the routing and separation. Maintaining Ginna Station's cable separation design basis includes accounting for functional and physical division. System equipment may be diverse and seem physically unrelated yet be credited to achieve the same safe shutdown function given the effects of a postulated failure. Examining the Appendix R success path matrix contained in the safe shutdown report will show further detail of equipment inter-dependencies and functional equivalencies. Care must always be exercised to ensure that common-mode failures are not introduced between functionally redundant equipment sets credited with mitigating the consequences of an event.

Ginna Station does not meet modern (current) standards for cable separation. There are, however,

a) Original design basis for cable separation and electrical isolation

and

b) Standards which must be achieved to ensure compliance with our current licensing basis.

When evaluating the routing and separation of existing cables, the cables must meet the standards invoked at the time of installation; i.e. original installation or modification. Originally installed cable would be evaluated to criteria of a, above (the columns headed Design Standard, Documentation Submitted for Design Basis Review, and After Construction Reviewed Against IEEE Standards (UFSAR Section) in Table 3).

The routing and separation of existing cables installed after the submittal of the original FSAR (Green Book) or the impact and effects of new cable installation would be evaluated to criteria b, above (the columns headed Current Licensing Basis and Preservation Process in Table 3) as specified in the Engineering Evaluation that implemented the plant change.

Table 3

DESIGN BASIS SUMMARY

Topical Area	Design Standard (DESIGN BASIS)	Documentation Submitted for Design Basis Review[3]	After Construction Reviewed Against IEEE Standards (UFSAR Section)	CURRENT LICENSING BASIS: Current Licensing Basis Described in UFSAR Section	PRESERVATION PROCESS: Ginna Station Procedure Which Maintains	Industry Standards[2]
4160VAC	E-EPS-1 (Westinghouse)	FSAR 8.2.2	1.8.3.2 1.8.3.4	8.1.4 8.2.2 8.3.1.1.3 8.3.1.4.1 a,b	EE-29 EE-80 Design Criteria EWR-10275	IEEE 308 IEEE 384 IEEE 422
480VAC	E-EPS-1 (Westinghouse)	FSAR 8.2.2	1.8.3.2.2 1.8.3.4	8.1.4 8.2.2.2 8.3.1.1.4 8.3.1.1.6 8.3.1.4.1 a,c,d	EE-29 EE-80 Design Criteria EWR-10275	IEEE 308 IEEE 384 IEEE 422
120VAC Control Power	E-EPS-1 (Westinghouse)	FSAR 8.2.2	1.8.3.2 1.8.3.4	8.1.4 8.2.2.2 8.3.1.4	EE-29 EE-80 Design Criteria EWR-10275	IEEE 308 IEEE 384 IEEE 422
120VAC Instrument Power	E-EPS-1 (Westinghouse)	FSAR 8.2.2	1.8.3.2 1.8.3.4	8.1.4 8.2.2.2 8.3.1.4 8.3.1.5	EE-29 EE-80 Design Criteria EWR-10275	IEEE 308 IEEE 384 IEEE 422

Table 3

DESIGN BASIS SUMMARY (Cont)

Topical Area	Design Standard	Documentation Submitted for Design Basis Review[3]	After Construction Reviewed Against IEEE Standards (UFSAR Section)	Current Licensing Basis Described in UFSAR Section	Ginna Station Procedure Which Maintains	Industry Standards[2]
	<<<<<<<<<<<<<<<<<<<< DESIGN BASIS >>		>>>>>>>>>>>>>>>	CURRENT LICENSING BASIS	<<<<<<< PRESERVATION PROCESS >>>>>>>	
120VDC	E-EPS-1 (Westinghouse)	FSAR 8.2.2	1.8.3.2 1.8.3.4	8.1.4 8.3.1.4 8.3.2.1	EE-29 EE-80 Design Criteria EWR-10275	IEEE 308 IEEE 384 IEEE 422
Instrument and Control Signal Loops	E-EPS-1 (Westinghouse)	FSAR 7.2.2 8.2.2	1.8.3.4	7.1.2 7.2.1 7.3.1 7.4.1.1 8.3.1.4	EE-29 EE-80 Design Criteria EWR-10275	IEEE 279 IEEE 384 IEEE 422
Cable Selection (Materials)	IPCEA[1]	FSAR 8.2.2	1.8.3.2 1.8.3.4	8.3.1.4.1 9.5.1.2.4.8	EE-29 EE-80 Design Criteria EWR-10275	IEEE 422 IEEE 383

Table 3

DESIGN BASIS SUMMARY (Cont)

	<<<<<<<<<<<<<<<<<<<< DESIGN BASIS	>>	>>>>>>>>>>>>>>	CURRENT LICENSING BASIS	PRESERVATION <<<<<<< PROCESS >>>>>>>	
Topical Area	Design Standard	Documentation Submitted for Design Basis Review[3]	After Construction Reviewed Against IEEE Standards (UFSAR Section)	Current Licensing Basis Described in UFSAR Section	Ginna Station Procedure Which Maintains	Industry Standards[2]
Circuit Isolation	E-EPS-1 (Westinghouse)	FSAR 7.2.2 8.2.2	1.8.3.2	8.1.4 8.3.1.1.4.2 8.3.1.4 8.3.2.1	DA-EE-93-107-07 DA-EE-93-104-07 EEA-09003 DA-EE-96-005-07 EWR 4773-I	IEEE 279 IEEE 308 IEEE 384 IEE 422

NOTES:

1) IPCEA - Insulated Power Cable Engineers Association
2) Standards enabled after construction are described in the Mod package on a job basis
3) Final Facility Description and Safety Analysis Report

Attachment 3

Representative Analysis
New CREATS Cables

Response to NRC Inquiry Concerning Electrical Separations for CREATS Cables in Tray

This purpose of this correspondence is to describe the approach taken by Ginna to ensure that the installation of the cables required for the CREATS modification would maintain the plant design basis for separation of redundant systems' cables installed in trays. The routing of the CREATS cables was evaluated from two different perspectives. First, the new cables to be installed in cable trays for the CREATS modification were reviewed to demonstrate that they will not be susceptible to failure of both redundant trains due to a single cable fault. Secondly, the cable installation and protection was reviewed to ensure that no single fault on any of the new cables can cause loss of any other safety related redundant equipment functions. Within those two approaches, the key characteristics of the installation that were reviewed were 1) physical separation between the two new trains of CREATS cables, 2) the fault protection of the cables to clear faults before cable damage could occur, propagating faults or fires, and 3) cable ratings with respect to voltage levels for CREATS cables and other cables in the trays being used.

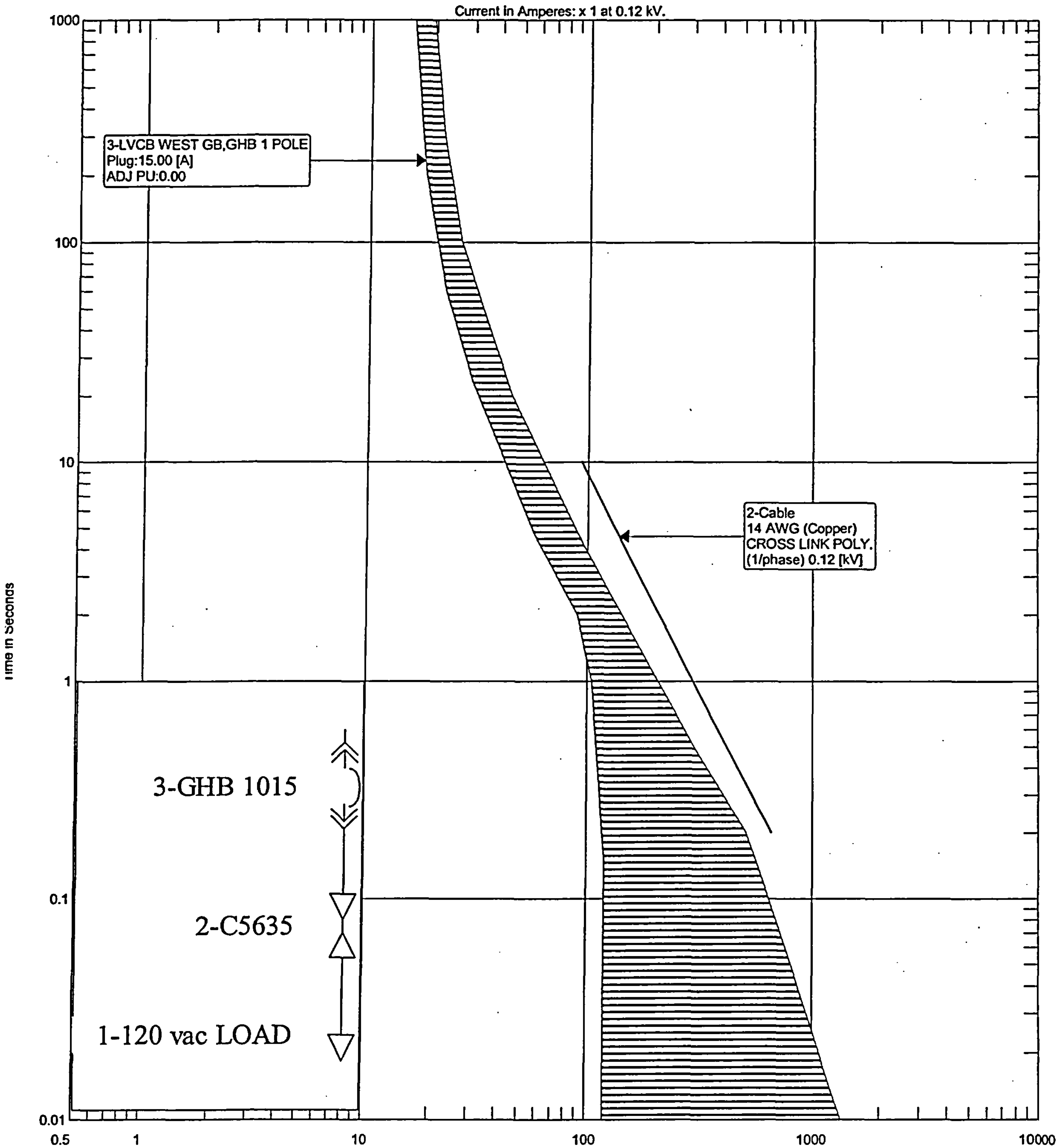
Circuit schedules issued for PCR 2003-0037, "CREATS Electrical Scope", for construction have the cable routing details representing the cables being installed in trays in the Relay Room. All of these cables are 120 volts AC or 125 volts DC. There are no 480 volt or higher power cables being installed for the CREATS modification in tray. All 480 volt cables being installed for CREATS are in independent conduits.

The Ginna design philosophy, since original plant design and during all plant modifications since, has been to implement protection schemes that assure that the fuses and breakers selected for a circuit are sized to protect the cable from damage. The fuse/breaker clearing curves are evaluated to ensure a fault or overload clears before the cable damage curve is reached. The fault protection of all cables being installed for this modification has been analyzed to show that the protective device for each cable will clear any fault before the cable damage curve is reached. Breaker and fuse protection of these circuits has been reviewed. The devices installed for this modification will all be procured and installed Safety Related. This feature ensures that if a cable installed for CREATS faults, that fault will clear before cable damage could cause a tray fire that could spread to the redundant train on any "associated circuit" that may transverse between the two redundant cables. Attached are two representative curves created to evaluate the cable protection of new cables. These represent the circuits with the highest voltage rating and smallest cable size for the circuit in each category installed in trays: a 120 VAC control cable (Circuit Schedule C5635) and a 125 VDC control cable (C5692). Both plots show that the protective device (breaker or fuse) will protect the limiting cable.

Following is a table that lists the trays to be utilized by the CREATS project. A walkdown of these trays in the relay room was performed to verify that the utilized portion of each of these trays is separate. Where the same tray designation is utilized for both trains, the cables for each train will maintain physical separation in the tray so the A and B cables will not be routed together, but utilize separate sections of the trays.

A Train Trays		B Train Trays
20		25
21		27N
23		28N
27S		29N
28N		29
28S		30
31		180
32		195
126		401
127		
157		
163		
180		
196		
376		

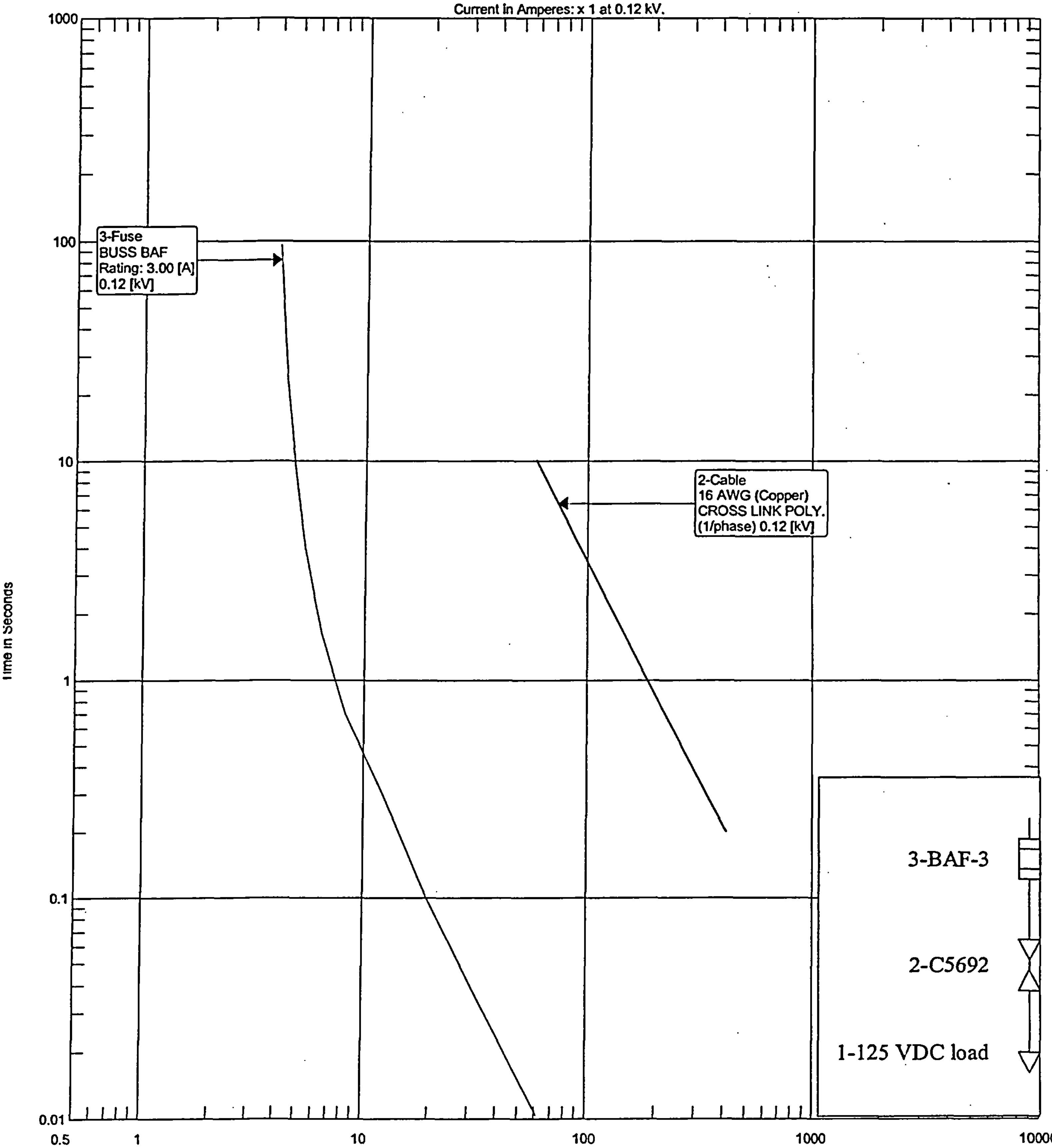
The voltage level of the cables installed in the trays being utilized for CREATS routing was reviewed. There is no 480 volt or higher power cables installed in the utilized trays. Safety related 120 VAC and 125 VDC instrumentation and control cables have been previously analyzed to demonstrate that the protective device will protect the installed cable. It is a design assumption that original plant design and subsequent design changes were performed to applicable standards such that all unanalyzed cables have appropriate protection. This design characteristic will clear a cable fault before the cable could damage CREATS cables in the same tray.



C5635 A Train 120 VAC	TIME CURRENT CHARACTERISTIC CURVES	
	BY:	
PLOTTING VOLTAGE: 0.12 kV	NO:	
	DATE: 9-24-2004	

Study: untitled.dat

Att 3 3 of 6



C5692 B Train 125VDC	TIME CURRENT CHARACTERISTIC CURVES BY:	
PLOTTING VOLTAGE: 0.12 kV		NO: DATE: 9-24-2004

Study: S:\Prod\MOD\PCR\2003\2003-0037 Rev 1\#16 awg vs BAF-3 fuse.tcc

Att3 4 of 6

ROCHESTER GAS AND ELECTRIC CORP. - GINNA STATION CIRCUIT SCHEDULE

Page: 1 of 2 Date: 01/29/04 Rev.: A Circuit No.: C5635

Prepared: Karen Coke Reviewed: [signature] Approved: [signature]

Project : CREATS I & C / ELECTRICAL SCOPE
Job No.: PCR 2003-0037 Rev. 1
Other :
Misc. :

Date	Status	Engr.
9/3/04	CONSTRUCTION	[initials]

Conductors								Conduit(s)		Cable Routing
No. of Cables	No. of Cond.	Cable Size	Shield	Operating Voltage	Cable Rating	BOM / PO#	Cable Length	Size	Length	
1	3/C	14 AWG		120	600	MID5005788	140	1 1	10 20	CR-148-P, 28N, 28S, 27S, 23, 32, 31, C5635, RR-129A-P, C5635_1

Work Group / QC Verification and Validation

Reel No.	Cut Length	QC Cable	QC Pull	QC Conduit	QC Tag

Nature of Circuit: *C 120 VAC CONTROL* Division: A - TRAIN A

3 Phase: NO
Maintained Spacing: NO
Appendix R Safe Shutdown: NO
Safety Classification: SAFETY CLASS-3

From Equip : ABB
From Desc : AUXILIARY BENCHBOARD
From Drawing: ATTACHMENT 18, SH 6

To Equip : CJB-3A
To Desc : CREATS JUNCTION BOX 3A
To Drawing : ATTACHMENT 17, SH 1

Term Block	Term No.	Wire Mark	Base / Trace Color	Cont. WG	Cont. QC	Megger COND	Megger SHD	Megger QC	From WG	From QC	To WG	To QC	Term Block	Term No.
CB	8	1103-3											TB1	1
CB	9	1103-4											TB1	4
CB	10	1103-N											TB1	2

NOTES:

1. THIS IS A NEW CIRCUIT.
2. INITIATE A BREECH PERMIT IN ACCORDANCE WITH FPS-1.
3. INSTALL THROUGH NEW PENETRATION RR-129A-P AND EXISTING PENETRATION CR-148-P.

ROCHESTER GAS AND ELECTRIC CORP. - GINNA STATION CIRCUIT SCHEDULE

Page: 1 of 2 Date: 02/24/04 Rev.: A Circuit No.: C5692

Prepared: Karen Cota Reviewed: Paul Swift Approved: Paul Swift

Project : CREATS I & C ELECTRICAL SCOPE
Job No. : PCR 2003-0037 Rev. 1
Other :
Misc. :

Date	Status	Engr.
9/3/04	CONSTRUCTION	PS

Conductors

No. of Cables	No. of Cond.	Cable Size	Shield	Operating Voltage	Cable Rating	BOM / PO#	Cable Length
2	2/C	16 AWG	YES	125	600	MID5000196	120

Conduit(s)

Size	Length
2	70
2	10

Cable Routing

C5631, RR-804-P, C5631_1, 25, 27N, 28N, CR-41-P

Work Group / QC Verification and Validation

Reel No.	Cut Length	QC Cable	QC Pull	QC Conduit	QC Tag

Nature of Circuit: *C 125 VDC CONTROL/POWER*

Division: D - TRAIN B Associated

3 Phase: NO
Maintained Spacing: NO

Appendix R Safe Shutdown: NO
Safety Classification: SAFETY CLASS-3

From Equip : CJB-1B
From Desc : CREATS JUNCTION BOX 1B
From Drawing: ATTACHMENT 16, SH 4

To Equip : ABB
To Desc : AUXILIARY JUNCTION BOX
To Drawing : ATTACHMENT 18, SH 6

Term Block	Term No.	Wire Mark	Base / Trace Color	Cont. WG	Cont. QC	Megger COND	Megger SHD	Megger QC	From WG	From QC	To WG	To QC	Term Block	Term No.
TB9	1	ANN-E08											CRHD	4
TB9	4	EAP											CRHD	8
TB9	8	ANN-K31											CRHD	12
TB9	9	KAP											CRHD	11

NOTES:

1. THIS IS A NEW CIRCUIT.
2. INITIATE A BREECH PERMIT IN ACCORDANCE WITH FPS-1.
3. INSTALL THROUGH NEW PENETRATION RR-804-P AND EXISTING PENETRATION CR-41-P.

Attachment 4

Representative Analysis
Existing Cables

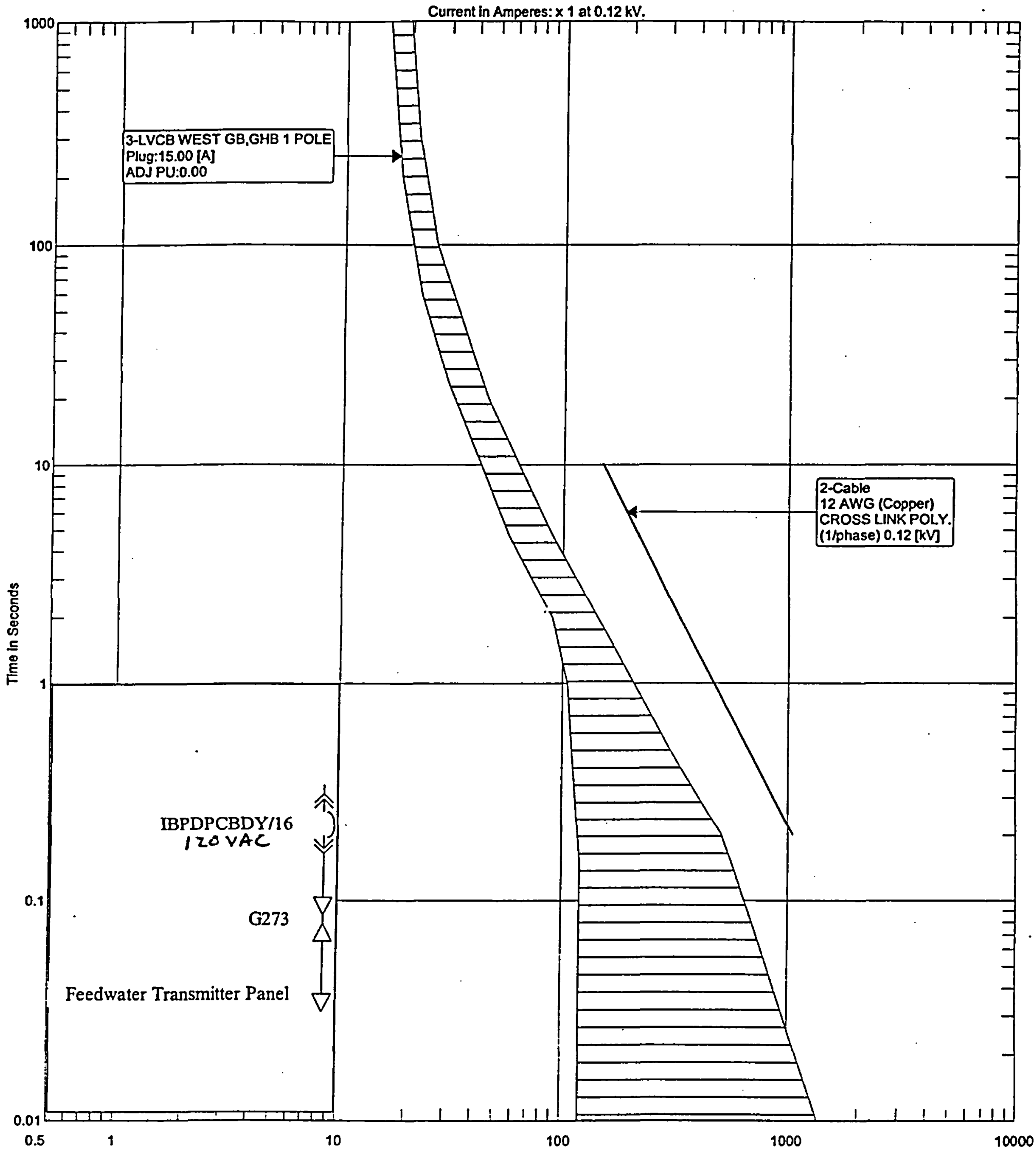
EXISTING CABLE ANALYSIS

As discussed with NRC during several conference calls, and demonstrated to the NRC representatives who participated in the August 19, 2004 walk-down of the relay room, the trays in question contain a large number of cables. However, very few are marked due to construction standards used at the time of installation.

To provide a reasonable assurance that the existing cables are properly protected, representative pre-existing cables from the trays used for the CREATS modification have been sampled and reviewed to determine the protective device used for fault protection of that cable. Cable damage curves were then created to demonstrate that the protective device is appropriately selected to protect the cable, clearing any fault or overload before cable damage can occur.

The cables in these trays fall into three categories: 125 VDC control power, 120 VAC control power, and 120 volt and lower instrument power. No 480 volt or higher power cables are in these trays. Addressing each category, the 125 VDC cables are protected by fuses. A DC fuse coordination analysis for the safety related DC system has been performed that would envelope the protection of DC circuits that may be routed in these trays. A representative plot is attached demonstrating that the fusing is designed to protect the cable below the cable damage curve. 120 VAC control power cables in these trays are fed by breakers. Most 120 VAC control power cables in the relay room trays originate from instrument bus or associated sub-panels. These circuits are all protected by safety related breakers, and an analysis has been performed that demonstrates that the breakers are correctly sized to protect the cables. A representative curve for 120 VAC control power cables damage curve versus breaker protection is attached. The last category, for 120 volt and lower instrumentation cables, was reviewed. The currents carried by these cables are driven by instrumentation that has current driving capability limited by power supply size or instrument fusing. Non-Safety related portions of these instrument loops are normally isolated from safety related portions by isolation devices (optical isolators, transformers, or fuses).

As demonstrated from the above discussion and attached curves, there is reasonable assurance that a fault in existing cables will not result in cable damage sufficient to propagate to adjacent CREATS cables.

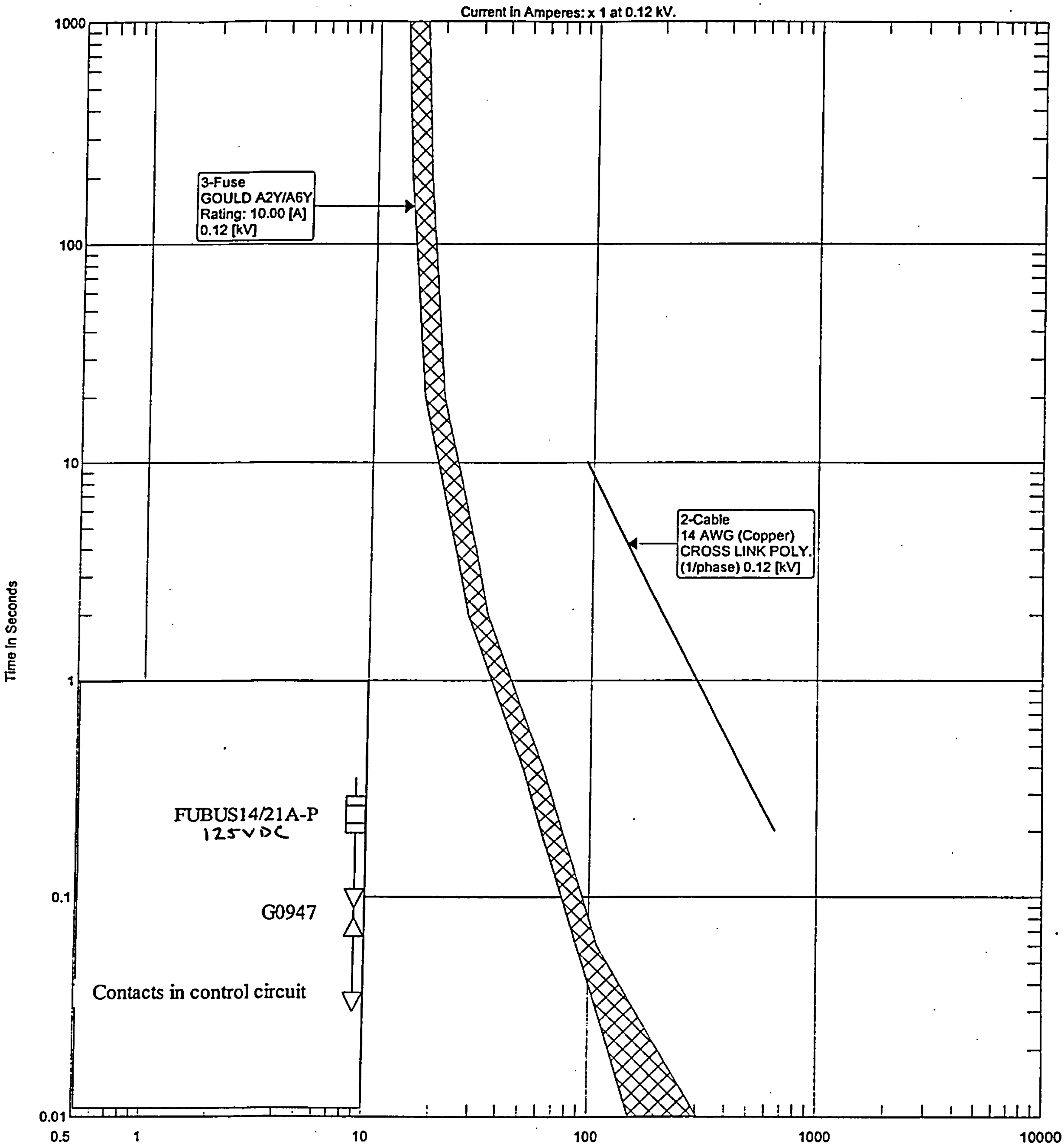


Coordination between Feedwater Transmitter Panel and Instrument Bus D distribution panel. Circuit G273 routed in tray 28	PLOTTING VOLTAGE:0.12 kV 120 VAC BY: D.Martin	NO: DATE: 11-19-2004

Study: S:\Prod\MOD\PCR\2003\2003-0037 Rev 1\cable loading for NRC g273.tcc

1/2

Att 4 2 of 5



Coordination between load and control circuit fuse (ref dwg 21946-0112,1)
Circuit routed in cable tray 28

PLOTTING VOLTAGE:0.12 kV 125 VDC
BY: D.Martin

NO:
DATE: 11-19-2004

Study: S:\Prod\MOD\PCR\2003\2003-0037 Rev 1\cable loading for NRC dc circuit.tcc

2/2.

Att 4 3 of 5

ROCHESTER GAS AND ELECTRIC CORP. - GINNA STATION CIRCUIT SCHEDULE

 DATE: 5/10/85 REV: 2 CIRCUIT NO. G273

Project: MFWP NPSH | Prepared: Richard A. Ba JUN 6 198

EWR No: 4115 | Reviewed: Richard A. Ba JUN 6 1

FCR No: | Approved: J. W. Daniels

ECN No:

DATE	STATUS	ENGR
6-6-85	AS BUILT	RAB

CONDUCTORS				CONDUIT		CABLE ROUTING
NO & SIZE	LENGTH	VOLTS	B/M	SIZE	LENGTH	Maintained Spacing ( )
1-2-12			EK-7c	3/4	5	(12+10)31,32,23(12+16)27,28,29,151,56,53,48,49(A+3)20

REEL NO.	CUT L'TH	QC VERIFICATION			
		CUT	PULL	CONDUIT	TAGGING

NATURE OF CIRCUIT: Instrument Power ( ) 3ø

FROM: Inst. Distribution Panel 1D | TO: Feedwater Transmitter Panel

REFERENCE DRAWING FROM: W S00B447 (3) | REFERENCE DRAWING TO: 33013-1627

TERM BLOCK	TERM NO.	*	WIRE MARK	COLOR BA/TR	CONT. WG	CONT. QC	MEGGER COND	MEGGER SHD	MEGGER QC	FROM WG	FROM QC	TO WG	TO QC	TERM BLOCK	TERM NO.	*
	16		L1											A	L1	
	N		L2											A	L2	

RECEIVED
JUL 10 1985
CENTRAL RECORDS

NOTES: * Indicates determination. ** Indicates revised items.

CONTROLLED COPY
NO. 5 D:108

Inst. I.D. Numbers: ______

Tool I.D. Numbers: ______

Approvals: ______ ______ ______ ______

WG Supervisor(s) Date(s) QC Supervisor(s) Date(s)

WG/QC Remarks:

Att 4 4 of 5

ROCHESTER GAS AND ELECTRIC CORP. - GINNA STATION CIRCUIT SCHEDULE

PAGE 1 OF 1 DATE: 01/12/94 REV: 6 CIRCUIT NO. G0947

PROJECT: FIRE RELAY PANEL MODIFICATION

Prepared: [signature]
Reviewed: [signature]
Approved: [signature]

EWR No: 4668
FCR No:
ECN No:

DATE	STATUS	ENGR
1/26/94	As-Built	MSG

CONDUCTORS				CONDUIT		CABLE ROUTING
NO & SIZE	LGTH	V:OP,BOM	BOM#,PO#S	SIZE	LENGTH	Maintained Spacing (N)
1 4/C 14(AWG)	0	125 600	EK-7C	1	30.00	197,28,29,151,56,59,167, 165,170,171,62,G0947

REEL NO.	CUT L'TH	QC VERIFICATION			
		CUT	PULL	CONDUIT	TAGGING

NATURE OF CIRCUIT: C 125 VDC Control/Power
3 PHASE (N)
SAFE SHDN: NO

SAFETY CLASSIFICATION: Safety Signif./Non 1
DIVISION:

FROM EQUIP: PZ42
DESC: TERMINAL BOX PZ42
WIRING DIAG: 33013-0993

TO EQUIP: CFLCG
DESC: CHARCOAL FILTER LOGIC CAB G
WIRING DIAG: 10909-0067

TERM BLOCK	TERM NO.	WIRE MARK	COLOR BA/TR/TR	CONT. WG	CONT. QC	MEGGER C,SHD	MEGGER QC	FORM WG	FORM QC	TO WG	TO QC	TERM BLOCK	TERM NO.
A	7	FP-P5	BLACK									FUSE	P5
A	8	FP-N5	WHITE									FUSE	N5
A	9	21A-P	RED									TC	22
A	10	21A-T	GREEN									TC	23

NOTES:

1. RELAY AR93.
2. CIRCUIT CONTINUES VIA G947A.

ADDITIONAL REFERENCE DRAWINGS:

Elementary: 10905-0439
Cond Dwg 1:

Other Dwg: 21946-0112S2
Cond Dwg 2:

Inst I.D. Numbers: ______

Tool I.D. Numbers: ______

Approvals

WG Supervisor(s)	Date(s)	QC Supervisor(s)	Date(s)

WG/QC Remarks:

CCD 38

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NO. 5:1%

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Attachment 5

Design Analysis DA-EE-2000-009
Revision 1

Instrument Loop Performance Evaluation
and Setpoint Verification

Instrument Loop Number RMS R45/R46

Constellation Generation Group
R.E. Ginna Nuclear Power Plant
1503 Lake Road
Ontario, NY 14519

DA-EE-2000-009
Revision 1
October 26, 2004

Prepared by:	John R Guirlen Design Engineer	11/18/2004 Date
Reviewed by:	L. Clift Po Primary Reactor Systems	11/29/2004 Date
Reviewed by:	[illegible] Nuclear Safety & Licensing	11-18-2004 Date
Approved by:	[illegible] Independent Review Engineer	12/2/04 Date

REVISION STATUS SHEET

REVISION	AFFECTED SECTIONS	DESCRIPTION OF REVISIONS
0	All	Initial issuance to support PCR 99-004.
1	1.0, 2.0, 3.2, 3.5, 3.7, 3.8, 3.9, 3.11, 4.2, 4.3, 5.7, 5.13, 7.1.1, 7.1.3, 7.3.1.1, 7.3.2, 7.7, 7.7.2, 7.8.1, 7.9, 8.1, 8.2	Removed reference to RG&E. Methodology change to remove the term Allowable Value, consistent with Technical Specification amendment 85. Incorporated revised Analytical Limit.

INSTRUMENT PERFORMANCE EVALUATION AND SETPOINT VERIFICATION

TABLE OF CONTENTS

INSTRUMENT PERFORMANCE EVALUATION AND SETPOINT VERIFICATION

1.0 **Purpose:**

The purpose of this calculation is to document the overall loop uncertainty associated with control room ventilation system air intake radiation monitors R-45 and R-46, and to ensure that sufficient margin exists at the alarm setpoint. Since R-45 and R-46 are identical in design, R-45 will be discussed; however, this calculation applies to both loops.

Revision 1 of this analysis is for the following purposes:

- Update the analysis based on a revised Analytical Limit.
- Remove the section associated with Allowable Value determination.
- Provide the Channel Operational Test (COT) acceptance criteria.

2.0 **Conclusions:**

The calculated uncertainties of ±36.8% for the indication and 37.4% for the alarm is acceptable to ensure proper indication and alarm functions and to ensure that control room ventilation isolation occurs when necessary.

3.0 **Design Inputs:**

3.1 Victoreen (Inovision), Installation Operation, and Maintenance Instruction Manual, Model 955A, Part No. 955A-1, published 5/96 by Victoreen (Inovision) Inc.

3.2 R.E. Ginna Drawing 33013-1867, Control Room HVAC.

3.3 PCR 99-004, Control Room Radiation Monitor Skid Replacement.

3.4 EE-171, Control Room Radiation Monitor Specification, 12/6/99.

3.5 R.E. Ginna drawing 33013-0721, Control Building Ventilation Duct New Outside supply.

3.6 Procedure P-9, Radiation Monitoring System.

3.7 R.E. Ginna sketch 33013-2656-1, RMS1, RMS2 and RMS3 Rack Layout.

3.8 R.E. Ginna sketch 33013-2787-1, Control Room Ventilation Instrument Locations, Instrument Panels and Conduit Layout.

3.9 Design Analysis, DA-EE-2001-013, Revision 1, Control Room Radiation Monitors Analytical Limit Calculation.

3.10 Victoreen (Inovision) procedure CAL848AD, Calibration Procedure for Model 848-8A 10 mCi Field Calibrator -- Digital, dated 3/2/01.

3.11 CPI-MON-R45, Calibration of Control Room Ventilation Radiation Monitor R-45.

4.0 Referenced Documents:

4.1 Regulatory Guide 1.97, "Instrumentation for Light Water-Cooled Nuclear Plants to Assess Plant and Environs Conditions During and Following an Accident", (Rev. 3, Dated 5/83).

4.2 R.E. Ginna Drawing 10905-384, Elementary Wiring Diagram Annunciator Panel E.

4.3 R.E. Ginna UFSAR, Sections 6.4.2, 6.4.5, 11.5.2, and Table 3.11-1 Environmental Service Conditions for Equipment Designed to Mitigate Design Basis Events.

4.4 Improved Technical Specifications, R.E. Ginna Nuclear Power Plant, Section 3.3.6 and Table 3.3.6-1.

4.5 Procedure CH-RETS-RMS, RMS Monitor Setpoint Determination.

4.6 AR-E-11, Alarm Response Procedure, Control Room HVAC.

4.7 ANSI NB.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.

4.8 ODCM, Offsite Dose Calculation Manual.

4.9 Procedure CH-RETS-CAL-SPEC, Calibration of Victoreen (Inovision) RMS Detectors to Establish Alarm Setpoints.

4.10 Procedure EP-3-S-0505, Instrument Setpoint/Loop Accuracy Calculation Methodology.

4.11 ANSI N13.10-1974, Specification and Performance Of On-Site Instrumentation For Continuously Monitoring Radioactivity In Effluents.

4.12 ANSI N42.22-1995, American National Standard Traceability of Radioactive Sources.

5.0 Assumptions:

5.1 Victoreen (Inovision) reference accuracy specification for the detector is ±20% of actual dose (reading), $\underline{+}1$ digit ($\underline{+}1$% of reading) for the digital display and ±3% of reading for the alarm setting.

Basis:

Reference Victoreen (Inovision) Model 897A series detector and Model 956A-201 Universal Digital Ratemeter specification sheets in Design Input 3.1.

5.2 The following inaccuracies were assumed:

Detector power supply effect is negligible.

Detector and monitor drift effects are assumed to $\underline{+}10$% of reading.

Basis:

The equipment operates within stated limits of performance specifications for the variations in external power supply voltage. The high voltage setting for optimal detector response is set at the mid point of the voltage plateau. This allows for variations in the detector dc supply voltage of at least ± 50 vdc with no significant change in the detector response. Normal variations in the regulated 120 VAC ±2% supply (MQ400E) will result in insignificant changes to the detector and UDR dc voltage levels. Therefore, the external power supply effect is considered negligible.

The detector is a GM tube which has negligible drift over a 30 month period and the monitor digital signal processing is inherently stable. Therefore, although the drift may be considered negligible, a 30 month drift value of $\underline{+}10$% of reading will conservatively be used.

5.3 The monitor (indication) M&TE error is conservatively estimated to be $\underline{+}5.0$%.

Basis:

M&TE equipment should be more accurate than the device being calibrated by a ratio of 4 to 1. The calibration of the system will be performed during normal conditions and the calibration tolerance for the displayed indication is $\underline{+}$ 20% of the Cs 137 source strength. Plant calibration procedures normally require that the accuracy of the test equipment is four times greater than the accuracy of the equipment being calibrated (reference IP-MTE-1). It is assumed that the M&TE equipment used to calibrate the monitor has an accuracy equal to 1/4th of the calibration accuracy. Therefore, it is considered that the M&TE error of +5% of the reading for the meter is conservative.

5.4 The detector and monitor (ratemeter) temperature and radiation effects are negligible.

Basis:

The detector will be located in the control room ventilation duct, which per design input 3.2 has temperature limits of 2°F to 91°F. The vendor specification of -10°F to 122°F envelopes these limits. The monitor (ratemeter) will be located in the main control room where the normal temperature limits are 50°F to 104°F (normally 70-78°F). The vendor specification of 32°F to 122°F envelopes these limits also.

For a steam line break accident, the turbine building temperature may reach as high as 220°F (for 30 minutes), then is reduced to100°F within 3 hours (see section 7.3.3.2). Because of the high flow rate (2000 cfm) in the 42 inch ventilation duct and the relatively short duration that the turbine building temperature is greater than 100°F, there will be no significant increase in the internal duct temperature. Therefore, the detector will continue to operate below the vendor specified upper limit of 122°F.

It is assumed that control ventilation isolation will occur prior to any significant increase in radiation levels in the control room and therefore there will be no radiation effect on the monitor.

5.5 Insulation resistance error (cable leakage effect) is not applicable..

Basis

During harsh temperature and humidity conditions associated with a LOCA or HELB design basis accident, insulation resistance (current leakage) effects may induce signal current leakage. The cable that connects the detector to the monitor will not be exposed to a harsh environment during accident conditions. Therefore, abnormal environmental conditions due to an accident are not applicable to this function.

5.6 Indicator resolution is assumed to be negligible.

Basis

Indicator resolution is assumed to be ± 1 digit (least significant digit) per draft ISA-dTR67.04.03, "Indication Uncertainties and Their Relationship With Indicated Values". The least significant digit of the Victoreen (Inovision) model 956A-201 UDR is 1/100 mr/hr, and therefore will have a negligible effect on the overall uncertainty value of the indicated reading.

5.7 Indicator calibration tolerance is assumed to be ± 20% of reading, the alarm/control room HVAC isolation actuation tolerance is assumed to be ±4% of setting.

Basis

The calibration procedure (CPI-MON-R45) requirement for the indicator tolerance band is +/-20% of the corrected Cs 137 source value, and for the alarm/control room HVAC isolation actuation the tolerance is ±4% of the alarm setting.

5.8 M&TE accuracy and drift uncertainty for the alarm are assumed to be ± 3% of setting.

Basis

The vendor does not specify a drift setting for the alarm setpoint. The alarm setpoint circuitry is part of the UDR (Universal Digital Ratemeter) and as such is expected to be inherently stable. The drift term will therefore be conservatively set equal to the vendor alarm reference accuracy of 3% of setting. The M&TE error effect is conservatively assumed to be equal to the reference accuracy for the alarm setting. This is considered reasonable because it is necessary for the M&TE equipment to be more accurate than the device being calibrated in order for the calibration to be effective. In addition, M&TE equipment is required to be more accurate than the device being calibrated by a factor of 4:1 unless otherwise justified.

5.9 The statistical accuracy of the indication is ±10% of reading.

Basis

During normal operation the "statistics" switch may be maintained in the 10% position. This causes the monitor to base the displayed radiation level on the (last) minimum number of counts that will ensure a "precision" of ±10% at 95% confidence. Therefore, the displayed radiation level will move after each update within an approximately 10% band around the "true" radiation level.

The total response time of the system to a step change in the radiation value is 60 seconds, due to the operation of the pulse counting algorithms. The detector radiation value displayed is the result of a rolling average of the latest 60, 1 second values, and is updated once per second. An alarm will be initiated within one second after the current rolling one minute average exceeds the alarm setpoint.

For a large radiation source term, as during a postulated Design Basis Accident, the effect of the statistical error and time averaging circuitry become negligible. The Control Room intake air gamma activity rises well above the level of the control room ventilation isolation setpoint in a very short period of time (Design Input 3.9).

5.10 Cs-137 reference source field calibrator has an uncertainty of ±10%.

Basis

Victoreen (Inovision) calibration procedure CAL848-8AD "Calibration Procedure for Model 848-8A 10 mCi Field Calibrator -- Digital" specifies an uncertainty of ±10% relative to actual dose rate for the Model 848-8A field calibrator.

5.11 The process measurement effects due to location of the detector will not negatively impact the calculation.

Basis

The detector location in the air duct will be such that there is adequate mixing of the sample. The measurement of the activity inside the duct is a representative sample of the actual activity such that isolation will occur before significant exposure can occur. The more centralized in the duct the detector is, the more accurate the reading. The input to this calculation assumes the most offset mounting location, at 28" to the end of the duct. Any location more centralized will result in a more conservative result. There is also a conservative assumption that only the activity within the air duct is being measured and no credit is taken for any activity outside the duct (at the detector location in the turbine building).

5.12 The energy dependence is ± 15% of the reading.

Basis

The response of a radiation detector is sensitive to the specific radionuclide present. Changes in the level of radiation flux incident upon the detector and the environment will cause changes in the detector response. Each type of detector has a different response to radiation of varying energies. The vendor specification sheet for this detector shows an energy dependence value of ± 15% of the reading.

5.13 The Analytical Limit for Control Room Ventilation isolation is 0.91 mr/hr as read at the in-duct monitoring locations.

Basis

The Analytical Limit of the setpoint for the monitors is based on the limit in the Control Room specified in 10 CFR 50, Appendix A, GDC 19 and the guidance provided by the NRC in NUREG-0737 Clarification of TMI Action Plan Requirements section II.B.2, Dose Rate Criteria, and NUREG-0800 Standard Review Plan section 6.4, Control Room Habitability Program. Areas that require continuous occupancy are to be designed for a maximum of 5 rem whole body dose, or its equivalent to any part of the body for the duration of the accident. This is further defined as a 30 day weighted average dose rate of less than 15 mrem/hr. Due to the reduced effective volume within the air duct as compared to the Control Room the Analytical Limit at the detector location must be lowered. Per reference 3.9 Design Analysis, this results in an Analytical Limit of 0.91 mr/hr.

5.14 The high alarm/control room ventilation isolation actuation setpoint shall be set high enough (greater than 10 times background) to prevent spurious alarms and undesired actuations of control room isolation.

Background radiation levels were measured at the control room air intake duct (at the detector location) and also at the roof air intake to the control room. All readings were less than 0.01 mr/hr, which is less than 1/10th of the high alarm/control room ventilation isolation actuation setpoint of 0.25 mr/hr.

R-45 Block Diagram

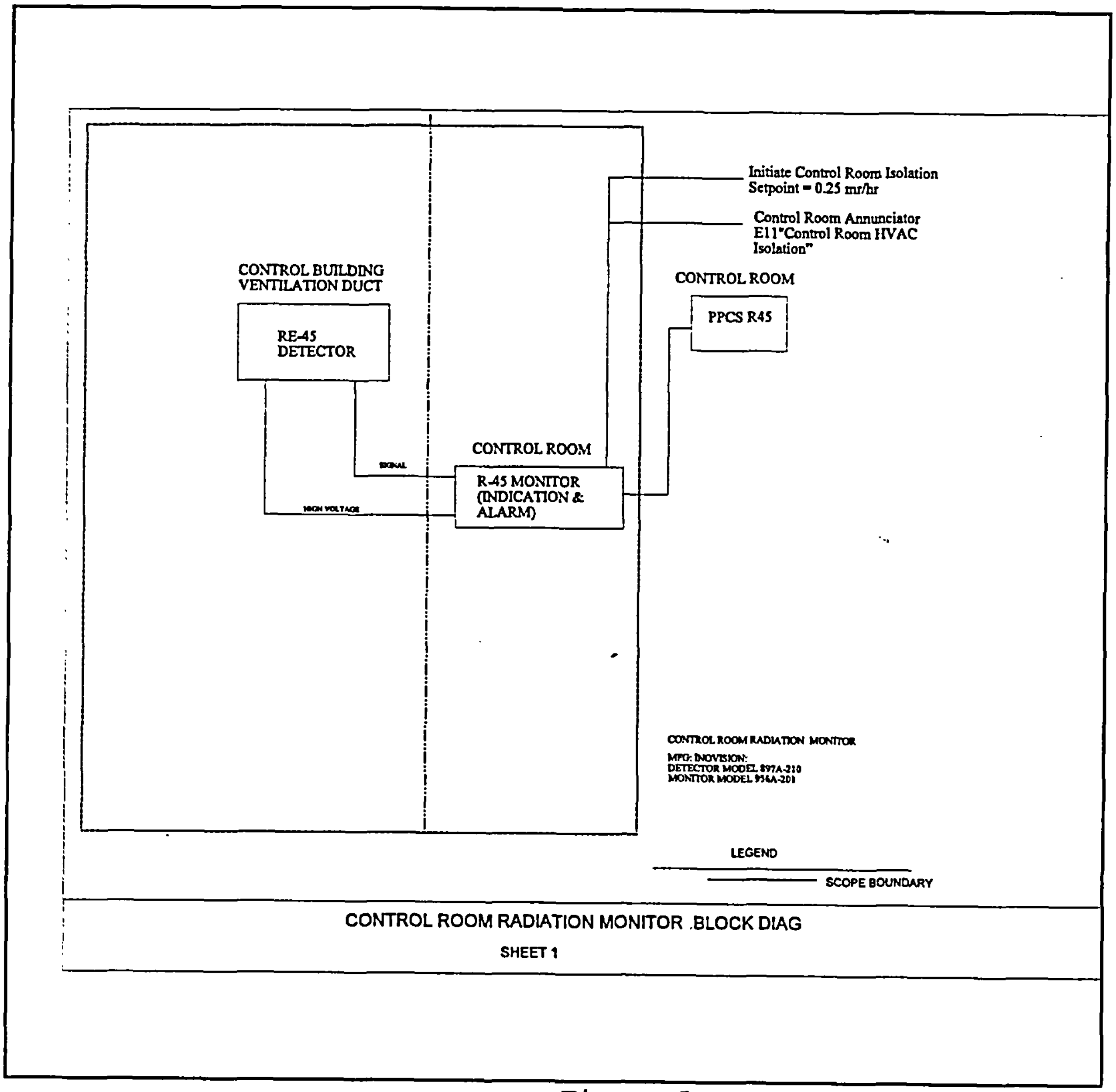

Figure 1

6.0 Computer Codes:

N/A

7.0 Analysis:

7.1 Instrument Channel and Scope of Analysis - Refer to Figure 1

7.1.1 Description of Functions

The Control Room Ventilation Radiation Monitoring System consists of two redundant monitoring channels R-45 and R-46, which provide information concerning the gamma radiation intensity within the ventilation stream of the control room supply air. Each channel consists of a model 897A-210 Geiger-Muller (GM) tube detector with an integral preamplifier, and a model 956A-201 Universal Digital Ratemeter (UDR). The G-M tube is a halogen quenched aluminum encased device, sensitive to beta-gamma activity in the gas sample. The Victoreen (Inovision) monitor (ratemeter) converts the detector input (counts) into a mr/hr display. The ratemeter outputs a 1-5 vdc signal representative of the radiation level in mr/hr to the PPCS. It also has a contact output that feeds the control room HVAC isolation and MCB Annunciator E11 circuitry. Alarm and control room ventilation isolation functions will occur when the high alarm/isolation setpoint is reached. The two channels are identical, therefore, further discussion will be on R-45, but will also be applicable to R-46.

7.1.2 Protection

This channel does not perform any (reactor) protective functions.

7.1.3 Control

This channel will generate an automatic control room ventilation isolation signal (isolating the Control Room HVAC) when the sensed radiation signal reaches the high alarm setpoint. Also, the charcoal filter fan will be started and the control room ventilation system dampers aligned to recirculate the control room air through the HEPA and charcoal filters.

7.1.4 Indication

RMS Channel R-45 provides control room personnel with digital indication of the radiation level of the incoming air to the control room and a high radiation/control room HVAC isolation alarm (control room annunciator E11).

Summary of Instrument Channel Functions

Function	Description of Function	Safety Classification	Within this scope?
Reactor Protection	None	N/A	N/A
Eng. Safety Features	None	N/A	N/A
Control	Control Room Ventilation Isolation	Safety Related	YES
Alarm	Control Room Radiation	Safety-Related	YES
Indication	Control Room Radiation	Safety-Related	YES

7.2 Documenting the Components of Sensor Accident Uncertainty (AEUp and AEUs)

7.2.1 Pipe Breaks

N/A

7.2.2 Seismic Event

N/A

7.2.3 Documenting the Components of the Accident Current Leakage Effect (CLU)

During harsh temperature and humidity conditions associated with LOCA or HELB design basis accidents, insulation resistance effects induce signal current leakage. Abnormal environmental conditions due to an accident are not applicable to this function.

CLU = N/A

7.2.4 Determining the Components of Process Measurement Uncertainty (PMU)

The random nature of the radioactive decay process causes a statistical uncertainty in the detection process. These inherent fluctuations are random and approximately normally distributed. They represent an unavoidable source of uncertainty in all nuclear measurements, often resulting in the predominant source of imprecision of error. The statistical accuracy is $\pm$ 10% (Assumption 5.9).

$$Pma_1 = \pm 10.0\%$$

The energy dependence of the reading is $\pm$ 15% (Assumption 5.12).

$$Pma_2 = \pm 15.0\%$$

Victoreen (Inovision) calibration procedure CAL848-8AD "Calibration Procedure for Model 848-8A 10 mCi Field Calibrator -- Digital" specifies an uncertainty of ±10% relative to actual dose rate for the Model 848-8A field calibrator. (Assumption 5.10).

$Pma_3 = \pm 10.0\%$

7.3 Instrument Loop Performance Requirements

7.3.1 Documenting the Design Requirements for Monitoring the Process Parameter

7.3.1.1 Identify Performance Related Design Bases Associated With the Instrument Loop:

SR — **Safety Classification (SR/SS/NS).**

NO — **NUREG 0737/RG 1.97 as documented in Table 7.5-1, of the Ginna UFSAR**

Per a review of UFSAR Table 7.5-1, this instrument channel is not identified in RG 1.97 as being required as a post-accident monitoring instrument.

NO — **EQ (per the 10 CFR 50.49 list)**

This instrument loop is not identified as requiring Environmental Qualification.

S1 — **Seismic Category (Seismic Class I/ Structural Integrity Only / NS)**

YES — **Technical Specifications**

The limits applicable to this instrument loop are addressed in Section 3.3.6 of the Ginna Improved Technical Specifications.

YES — **UFSAR**

This instrument is described in Sections 6.4.2.2.2, 6.4.2.2.3, 6.4.5, and 11.5.2.2.16 of the Ginna UFSAR.

NO — **EOP**

Per a review of the EOP Setpoint database, there are no EOP related setpoints covered by this analysis.

7.3.2 Description of Limits

Function	Analytical Limit	Reference
Control Room Isolation	.91 mr/hr	Design Input 3.9

7.3.3 **Documenting the Environmental Conditions Associated With the Process Parameter**

7.3.3.1 **Identification of the Sensor Location:**

Control room 42 inch diameter air intake duct, turbine building operating floor

7.3.3.2 **Description of Environmental Service Conditions for the Sensor:**

Normal Operation, Turbine Building

Reference 4.3, UFSAR Table 3.11-1.

Temperature: 50°F to 104°F
Pressure: Atmospheric
Humidity: 60% Nominal
Radiation: Negligible

The detector is located in the Control room air intake duct, which per PCR 99-004, section 3.2.2 has the following ambient environmental limits: temperature limit of 2°F to 91°F, pressure at 0 psig, and 100% humidity.

During Calibration

Same as Normal Operation above.

Accident, Turbine Building

Reference 4.3, UFSAR Table 3.11-1.

Temperature: 220°F for 30 minutes, reduce to 100°F within 3 hours.
Pressure: 1.14 psig on mezzanine and basement levels, 0.7 psig on operating floor for 30 minutes, reduce to ambient 3 hours.
Humidity: 100%
Radiation: Negligible
Flooding: 18 inches in basement

7.3.3.3 **Identification of Other Components Locations:**

Instrument	Location
R-45 Monitor (Indicator) and alarm.	Control Room, Rack RMS2

7.3.3.4 **Description of Environmental Service Conditions for Other Components:**

Normal Operation, Main Control Room

Reference 4.3, UFSAR Table 3.11-1.

Temperature: 50°F to 104°F (usually 70 - 78°F)
Pressure: Atmospheric
Humidity: 60% Nominal
Radiation: Negligible

During Calibration

Same as Normal Operation Above.

Accident, Main Control Room

Reference 4.3, UFSAR Table 3.11-1.

Temperature: Less than 104°F
Pressure: Atmospheric
Humidity: 60% Nominal
Radiation: Negligible
Flooding N/A

7.4 Instrument Channel Component Specifications:

Identify and summarize the specifications associated with each instrument within the scope of this analysis. Complete one Instrument Specification Table for each instrument.

EIN: Readout Module R-45

Specification	Data	Source
Manufacturer/Model No	Victoreen (Inovision)/956A-201	Design Input 3.1
Input Range	digital pulse corresponding to 10E-2 to 10E+3 mr/hr	Design Input 3.1
Output Range	10E-2 to 10E+3 mr/hr indication	Design Input 3.1
Safety Classification	Safety-Related	Proposed
Setpoints	0.25 mr/hr	Proposed
Location	Control Room	Proposed
Accuracy	monitor ±1 % of reading detector ±20 % of reading	Assumption 5.1
Drift	± 10.0% of reading	Assumption 5.2
Calibration Uncertainty (M&TE)	± 5.0% of reading	Assumption 5.3
Calibration Tolerance	± 20.0% of source value	Assumption 5.7
Temperature Effect	Negligible	Assumption 5.4

7.5 Determining the Indication Uncertainties

No logarithmic conversions take place between the detector signal and the ratemeter digital display or alarm circuit. Therefore, no calculations for converting from % of reading to equivalent linear full scale (ELFS) are required.

7.5.1 Process Measurement Uncertainty (PMU) (reference section 7.2.4)

$PMU = (Pma_1^2 + Pma_2^2 + Pma_3^2)^{1/2}$

$PMU = (10.0^2 + 15.0^2 + 10.0^2)^{1/2} = 20.6\%$

7.5.2 Accident Environmental Uncertainties (AEU)

$AEU = 0\%$

7.5.3 Accident Current Leakage Effect (CLU)

$CLU = 0\%$

7.5.4 Calibration Uncertainties (M&TEU):

Per assumption 5.3, Measurement and Test Equipment Effect (M&TEU) is 5%.

$Ice_1 = \underline{+}5\%$

$M\&TEU = \pm[(Ice_1)^2]^{1/2}$

$M\&TEU = \pm[(5.0)^2]^{1/2}$

$M\&TEU = \pm 5.0\%$

7.5.5 Tolerance Uncertainty (TU):

The as left tolerance of the indicator is

$Ice_2 = \underline{+}20\%$

$TU = \pm[(Ice_2)^2]^{1/2}$

$TU = \pm[(20.0)^2]^{1/2}$

$TU = \underline{+}20.0\%$

7.5.6 Sensor Uncertainty (SU):

The sensor uncertainty is $\underline{+}20\%$ (Assumption 5.1)

$Sa = \underline{+}20\%$

$SU = \pm[(Sa)^2]^{1/2}$

$SU = \pm [(20.0)^2]^{1/2}$

$SU = \underline{+} 20.0\%$

7.5.7 Drift Uncertainty (DU):

The drift uncertainty is assumed to be 10% (Assumption 5.2).

$Red_1 = \underline{+}10.0\%$

$DU = \pm[(Red_1)^2]^{1/2}$

$DU = \pm[(10.0)^2]^{1/2}$

$DU = 10.0\%$

7.5.8 Rack Equipment Uncertainty (REU):

Rack equipment uncertainty Rea_I is 1.0% (Assumption 5.1)

$Rea_I = \pm 1.0\%$

$REU = [(Rea_I)^2]^{1/2}$

$REU = \pm [(1.0)^2]^{1/2}$

$REU = \pm 1.0\%$

7.6 Calculating the Total Loop Uncertainties (Indication)

Provide the total loop uncertainty (TLU) for each end device for normal, seismic and accident conditions as applicable.

7.6.1 TLU Indicator

$TLU_I = \pm (PMU^2 + M\&TEU^2 + SU^2 + DU^2 + REU^2 + TU^2)^{1/2}$

$TLU_I = \pm (20.6^2 + 5.0^2 + 20.0^2 + 10.0^2 + 1.0^2 + 20.0^2)^{1/2}$

$TLU_I(+) = +36.8\%$ of reading

$TLU_I(-) = -36.8\%$ of reading

7.7 Determining Alarm Uncertainties

The alarm circuits are integral to the monitor. The additional parameters that must be taken into consideration for the determination of the alarm setting uncertainty are listed in the following table.

EIN: Readout Module R-45 Alarm Function

Specification	Data	Source
Accuracy	$\pm$ 3% of Setting	Design Input 3.1
Calibration Tolerance	$\pm$ 4% of Setting	Assumption 5.7
M&TE Accuracy	$\pm$ 3% of Setting	Assumption 5.8
Drift Uncertainty	$\pm$ 3% of Setting	Assumption 5.8

7.7.1 Calibration Uncertainties (M&TEU):

Alarm Measurement and Test Equipment Uncertainty

The M&TE error effect for calibrating the alarm is considered equal to the reference accuracy.

$$Amte = \pm 3.0\%$$

$$M\&TEU = \pm[(Amte)^2]^{1/2}$$

$$M\&TEU = \pm(3.0^2)^{1/2}$$

$$M\&TEU = \pm 3.0\%$$

7.7.2 Tolerance Uncertainty (TU):

Alarm Calibration accuracy

The as left tolerance for the high alarm setpoint is $\pm$ 4.0% of setting.

$$Ace = \pm 4.0\%$$

$$TU = \pm [(Ace)^2]^{1/2}$$

$$TU = \pm [(4.0\%)^2]^{1/2}$$

$$TU = \pm 4.0\%$$

7.7.3 Rack Equipment Uncertainty (REU):

Alarm reference accuracy = $\pm$3% of setting

$$Are = \pm 3.0\%$$

$$REU = [(Are)^2]^{1/2}$$

$$REU = \pm [(3.0)^2]^{1/2}$$

$$REU = \pm 3.0\%$$

7.7.4 Determining Drift Uncertainty (DU)

$$Red_2 = \pm 3.0\%$$

$$DU = \pm [(Red_2)^2]^{1/2}$$

$$DU = \pm [(3.0)^2]^{1/2}$$

$$DU = \pm 3.0\%$$

7.8 Calculating the Total Loop Uncertainties (Alarm)

7.8.1 TLU Alarm

$TLU_A = \pm (TLU_I + M\&TEU^2 + + DU^2 + REU^2 + TU^2)^{1/2}$

$TLU_A = \pm (36.8^2 + 3.0^2 + 3.0^2 + 3.0^2 + 4.0^2)^{1/2}$

$TLU_A(+) = +37.4\%$ of setting

$TLU_A(-) = -37.4\%$ of setting

Where:
TLU_I = The Total Loop Uncertainty Indication
TLU_A = The Total Loop Uncertainty Alarm
CLU = Current Leakage Uncertainty
AEUs = Accident Environmental Uncertainty (Seismic)
PMU = Process Measurement Uncertainty
REU = Rack Equipment Uncertainty
SU = Sensor Uncertainty
DU = Drift Uncertainty
TU = Tolerance Uncertainty
IU = Indication Uncertainty
M&TEU= Measurement and Test Equipment Uncertainty

No distinction is made between Normal and Accident conditions for the Total Loop Uncertainty.

End Device	TLU
R-45 (Ind)	+/-36.8% of reading
R-45 (Alm)	+/-37.4% of setting

7.9 Comparing the Reference Accuracy vs. the Calibration Tolerance

Identify the calibration tolerance associated with each component. Next, obtain the reference accuracy associated with each component. Translate both effects into the equivalent units.

Tag No.	Reference Accuracy	Calibration Tolerance
R-45 (mon)	± 20% (Reading)	± 20 % of (Reading)
R-45 (Alm)	± 3% (Setting)	± 4.0% (Setting)

Reference accuracy is based on Assumption 5.1. The tolerance is appropriate.

8.0 Results:

8.1 Maximum Calculated Setpoint and COT Uncertainty

Maximum Calculated Setpoint = Analytical Limit - TLU

= 0.91 mr/hr - (0.374 x 0.91 mr/hr) = **0.57 mr/hr**

The alarm setpoints should be set such that they actuate control room isolation before the control room ventilation air duct radiation level exceeds 0.91 mr/hr. Therefore, the maximum calculated setpoint will be 0.57 mr/hr.

COT Uncertainty $= \pm (M\&TEU^2 + DU^2 + REU^2)^{1/2}$

$$= \pm (3.0^2 + 3.0^2 + 3.0^2)^{1/2}$$

$$= \pm 5.2\% \text{ of setting}$$

8.2 Conclusion

A review of the instrument loop performance requirements against the proposed loop configuration for RMS R-45 was conducted by this evaluation. The results of this review determined that the proposed safety related control room radiation monitor will initiate control room ventilation isolation prior to exceeding the 0.91 mr/hr limit. For conservatism it is recommended that the alarm (control room isolation) nominal setpoint will be set at **0.25 mr/hr**, which is in excess of the minimum value of 0.1mr/hr discussed in assumption 5.14.

Attachment 6

Revised Submittal
Tech Spec Table 3.3.6-1

Marked-up Technical Specification Sections

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1
~~MODES 1, 2, 3, and 4,~~
~~During movement of irradiated fuel assemblies,~~
~~During CORE ALTERATIONS.~~

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 ~~- NOTE - The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition.~~ Place ~~CREATS in Mode F.~~ one CREATS train in emergency mode	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1 ~~- NOTE - The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition.~~ 1 Place ~~CREATS in Mode F.~~ one CREATS train in emergency mode	Immediately
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS~~.	~~D.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ D.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform COT.	92 days
SR 3.3.6.3	- NOTE - Verification of setpoint is not required. Perform TADOT.	24 months
SR 3.3.6.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.5	Perform ACTUATION LOGIC TEST.	24 months

①

AND

B.1.2	Enter applicable Conditions and Required Actions for one ~~CREFS~~ CREATS train made inoperable by inoperable ~~CREFS~~ CREATS actuation instrumentation.	Immediately
OR		
B.2	Place both CREATS trains in emergency ~~[radiation protection]~~ mode.	Immediately

Table 3.3.6-1
CREATS Actuation Instrumentation

FUNCTION	Applicable modes or other specified conditions	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE LIMITING SAFETY SYSTEM SETTINGS(a)
1. Manual Initiation	1, 2, 3, 4, (a) (b)	2 trains	SR 3.3.6.3	NA
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, (a) (b)	2 trains	SR 3.3.6.5	NA
3. Control Room Radiation Intake Monitors	1, 2, 3, 4, (a) (b)	2	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.4	.57 ≤ .5 mR/hr
4. Safety Injection	Refer to LCO 3.3.2 "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) attached

(a) During movement of irradiated fuel assemblies

(b) ..

(a) A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the COT Acceptance Criteria. The COT Acceptance Criteria is defined as:

   |as-found TSP - previous as-left TSP| ≤ COT uncertainty

   The COT uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the Limiting Safety System Setting (LSSS). The LSSS and the established , COT uncertainty, calibration tolerance band are defined in accordance with the Ginna Instrument Setpoint Methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration tolerance band.

Typed Technical Specification Sections

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Place one CREATS train in emergency mode.	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1 Place one CREATS train in emergency mode.	Immediately
	AND	
	B.1.2 Enter applicable Conditions and Required Actions for one CREATS train made inoperable by inoperable CREATS actuation instrumentation.	Immediately
	OR	
	B.2 Place both CREATS trains in emergency mode.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform COT.	92 days
SR 3.3.6.3	- NOTE - Verification of setpoint is not required. Perform TADOT.	24 months
SR 3.3.6.4	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.5	Perform ACTUATION LOGIC TEST.	24 months

Table 3.3.6-1
CREATS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	LIMITING SAFETY SYSTEM SETTINGS[a]
1.	Manual Initiation	1, 2, 3, 4, (b)	2 trains	SR 3.3.6.3	NA
2.	Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, (b)	2 trains	SR 3.3.6.5	NA
3.	Control Room Radiation Intake Monitors	1, 2, 3, 4, (b)	2	SR 3.3.6.1 SR 3.3.6.2 SR 3.3.6.4	≤ .57 mR/hr
4.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) A channel is OPERABLE when both of the following conditions are met:

1. The absolute difference between the as-found Trip Setpoint (TSP) and the previous as-left TSP is within the COT Acceptance Criteria. The COT Acceptance Criteria is defined as:

   |as-found TSP - previous as-left TSP| ≤ COT uncertainty

   The COT uncertainty shall not include the calibration tolerance.

2. The as-left TSP is within the established calibration tolerance band about the nominal TSP. The nominal TSP is the desired setting and shall not exceed the Limiting Safety System Setting (LSSS). The LSSS, COT uncertainty, and the established calibration tolerance band are defined in accordance with the Ginna instrument setpoint methodology. The channel is considered operable even if the as-left TSP is non-conservative with respect to the LSSS provided that the as-left TSP is within the established calibration toleranceband.

(b) During movement of irradiated fuel assemblies

Marked-up Bases Sections

Note: These bases pages are being provided for information only to show the changes that Ginna intends to make following approval of the LAR. The bases are under Ginna control for all changes in accordance with Specification 5.5.13. Ginna requests that the NRC document acceptance of these bases changes in the SER.

B 3.3 INSTRUMENTATION

B 3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

BASES

BACKGROUND

CREATS

with respect to

The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This system is described in the Bases for LCO 3.7.9, "Control Room Emergency Air Treatment System (CREATS)." This LCO only addresses the actuation instrumentation ~~for~~ the high radiation state ~~CREATS Mode F~~.

or a Safety Injection (SI) signal

The high radiation ~~state CREATS Mode F~~ actuation instrumentation consists of two GM probe radiation monitors installed in the outside air intake for the control room ventilation system. A high radiation signal from either of these detectors will ~~initiate the CREATS filtration train and isolate each air supply path with two dampers~~. The control room operator can also ~~initiate the CREATS filtration train and isolate the air supply paths~~ by using ~~either of two~~ manual pushbuttons in the control room.

place the CREATS in the emergency mode

, or a SI signal,

the

INSERT A

APPLICABLE SAFETY ANALYSES

(2)

The location of components and CREATS related ducting within the control room emergency zone envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 1). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

Insert A

Technical specifications are required by 10 CFR 50.36 to contain limiting safety system settings (LSSS). The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis. However, in practice, the actual settings for automatic protective devices must be chosen to be more conservative than the Analytic Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

The Calculated Trip Setpoint is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytic Limit. As such, the Calculated Trip Setpoint accounts for uncertainties in setting the device (e.g. calibration), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). As such, the Calculated Trip Setpoint meets the definition of an LSSS and they are contained in the technical specifications.

Technical specifications contain requirements related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in technical specifications as "...being capable of performing its safety functions(s)." For automatic protective devices, the required safety function is to ensure that a SL is not exceeded and therefore the LSSS as defined by 10 CFR 50.36 serves as the OPERABILITY limit for the nominal trip setpoint. However, use of the LSSS (Calculated Trip Setpoint) to define OPERABILITY in technical specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the as-found value of a protective device setting during a surveillance. This would result in technical specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the Calculated Trip Setpoint due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for determining the Calculated Trip Setpoint and thus the automatic protective action would still have been ensured with the as-found setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to within the tolerance band assumed in the determination of the Calculated Trip Setpoint to account for further drift during the next surveillance interval.

The Nominal Trip Setpoint is the desired setting specified within established plant procedures, and may be more conservative than the Calculated Trip Setpoint. The Nominal Trip Setpoint therefore may include additional margin to ensure that the SL would not be exceeded. Use of the Calculated Trip Setpoint or Nominal Trip Setpoint to define as-found OPERABILITY, under the expected circumstances described above, would result in actions required by both the rule and technical specifications that are clearly not warranted. However, there is also some point beyond which the OPERABILITY of the device would be called into question, for example, greater than expected drift. This requirement needs to be specified in the technical specifications in order to define the OPERABILITY limit for the as-found trip setpoint and is designated as the Channel Operational Test (COT) Acceptance Criteria.

The COT Acceptance Criteria described in SR Table 3.3.6-1 serves as a confirmation of OPERABILITY, such that a channel is OPERABLE if the absolute difference between the as-found trip setpoint and the previously as-left trip setpoint does not exceed the assumed uncertainty during the performance of the COT. The assumed uncertainty is primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition, as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established Nominal Trip Setpoint calibration tolerance band, in accordance with the uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the COT Acceptance Criteria the device would be considered inoperable from a technical specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

2

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CREATS acts to terminate the supply of unfiltered outside air to the control room, and to initiate filtration. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel. One train of filtration in conjunction with isolation is sufficient to maintain control room doses within established limits.

Control Room doses were analyzed per Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Ref. 6). Per reference 7, Safety Injection is credited with initiating the CREATS emergency mode within the time assumed in the dose analysis for LOCA, SGTR and MSLB accidents. For other analyzed accidents (Rod Ejection, Locked RCP Rotor, Fuel Handling Accident, SFP Tornado Missile), the high radiation signal is the primary protection. CREATS actuation is not required for GDT Rupture, although the analysis demonstrates that actuation will occur from the radiation monitors for this event.

Subsequent to the installation of the present rad monitors, the control room accident doses were recalculated using the alternate source term methodology (Ref 6)

LSSS

The ~~Allowable Value~~ for the Control Room Radiation Intake Monitors is based on a correlation to the limit specified in 10 CFR 50, Appendix A, GDC 19 (Ref. 3) and the guidance provided by the NRC in NUREG-0737 section II.B.2 (Ref. 4), Dose Rate Criteria, and NUREG-0800 section 6.4, Control Room Habitability Program (Ref. 5). This is a maximum of 5 rem body dose, with a 30 day weighted average dose rate of less than 15 mR/hr. This ~~allowable value~~ LSSS is calculated in accordance with the Ginna Station Setpoint Verification Program and will provide for isolation of the control room ventilation system which will prevent exceeding these limits. The current control room accident dose calculations conservatively assume that the cloud released during the accident enters the control room envelope for ~~30~~ 60 seconds (360 seconds for SGTR) prior to ventilation system isolation. The response time of the Control Room Radiation Intake Monitors to an actual release is bounded by the time used in the analyses (Ref. 7) for those accidents crediting the Radiation monitors.

During movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS,~~ the CREATS ensures control room habitability in the event of a fuel handling accident. It has been demonstrated that the CREATS is not required in the event of a waste gas decay tank rupture (Ref. 2).

The CREATS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREATS is OPERABLE.

1. Manual Initiation

The LCO requires two trains to be OPERABLE. A train consists of one pushbutton and the interconnecting wiring to the actuation logic. The operator can initiate the CREATS ~~Filtration train~~ emergency mode at any time by using ~~either~~ the pushbutton in the control room. ~~This action will cause actuation of all components in the same manner as any of the automatic actuation signals required by this LCO.~~ Each pushbutton will actuate both trains of isolation dampers and the respective fan train.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Actuation Logic and Actuation Relays to be OPERABLE. Actuation logic consists of all circuitry associated with manual initiation, Safety Injection and Control Room Radiation Intake Monitors within the actuation system, including the initiation relay contacts responsible for actuating the CREATS emergency mode.

3

(3)

The Automatic SI Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2. The applicable MODES and specified conditions for the CREATS portion of these functions are different and less restrictive than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the CREATS function is affected, the Conditions applicable to their SI function need not be entered. The less restrictive Actions specified for inoperability of the CREATS Functions specify sufficient compensatory measures for this case.

3. Control Room Radiation Intake Monitor

The LCO specifies two channels of Control Room Radiation Intake Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREATS filtration train and isolation dampers remains OPERABLE.

S

The Nominal Trip Setpoint used in the Control Room Radiation Intake Monitors is based on the ~~Allowable Value~~ LSSS specified in Table 3.3.6-1. The selection of this trip setpoint is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Nominal Trip Setpoint specified in plant procedures is therefore conservatively adjusted with respect to the Analytical Limit. ~~If the measured setpoint exceeds the procedural tolerances of the Nominal Trip Setpoint value, the setpoint is considered OPERABLE unless the Allowable Value as specified in Table 3.3.6-1 is exceeded. The Nominal Trip Setpoint specified in the plant procedures bounds the Allowable Value.~~

INSERT B

4

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS actuation instrumentation must be OPERABLE to control operator exposure during and following a Design Basis Accident.

During movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS,~~ the CREATS actuation instrumentation must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

The most common cause of channel inoperability is failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. ~~The "as left" Nominal Trip Setpoint must be within the tolerance specified by the calibration procedure. If the "as found" setpoint exceeds the Allowable Value limit specified in Table 3.3.6-1, the channel must be declared inoperable immediately and the appropriate Condition entered.~~

Insert B

A channel is considered OPERABLE when:

a. The nominal trip setpoint is equal to or conservative with respect to the LSSS;
b. The absolute difference between the as-found trip setpoint and the previous as-left trip setpoint does not exceed the COT Acceptance Criteria; and
c. The as-left trip setpoint is within the established calibration tolerance band about the nominal trip setpoint.

The channel is still operable even if the as-left trip setpoint is non-conservative with respect to the LSSS provided that the as-left trip setpoint is within the established calibration tolerance band as specified in the Ginna Instrument Setpoint Methodology.

4

4. <u>Safety Injection</u>

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

The CREATS emergency mode is also initiated by all Functions that automatically initiate SI. The CREATS emergency mode requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.6-1. Instead, Function 1, SI, is referenced for all applicable initiating Functions and requirements.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel/train of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

~~A.1~~

5 → Condition A applies to one or more Functions with one channel of the CREATS actuation instrumentation inoperable.

If one radiation monitor channel, one manual initiation train, or one automatic actuation logic train is inoperable, 7 days are permitted to restore it to OPERABLE status. In this Condition the remaining redundant OPERABLE channel/train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE channel/train could result in a loss of function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining channel/train to provide the required capability. If the channel/train cannot be restored to OPERABLE status, the CREATS must be placed in Mode F. This accomplishes the actuation instrumentation function and places the system in a conservative mode of operation.

The Required Action for Condition A is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

6 → B.1

Condition B applies to the failure of two radiation monitor channels, two manual initiation trains, or two automatic actuation logic trains. In this Condition the CREATS actuation instrumentation is not capable of performing its intended automatic function. This is considered a loss of safety function. The Required Action is to place the CREATS in Mode F immediately. This accomplishes the actuation instrumentation function that may have been lost and places the system in a conservative mode of operation.

The Required Action for Condition B is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

(5)

A.1

Condition A applies to the actuation logic train Function of the CREATS, the radiation monitor channel Functions, the manual channel Functions and the SI logic Functions. If one train is inoperable, or one radiation monitor channel is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.9. If the channel/train cannot be restored to OPERABLE status, one CREATS train must be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation Function and places the plant in a conservative mode of operation.

(6)

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two CREATS actuation trains, two radiation monitor channels, two manual channels, or two SI actuation trains. The first Required Action is to place one CREATS train in the emergency mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the plant in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.9 must also be entered for the CREATS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.9.

Alternatively, both trains may be placed in the emergency mode. This ensures the CREATS function is performed even in the presence of a single failure.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time of Condition A or B has not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE that minimizes accident risk. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time of Condition A or B has not been met during movement of irradiated fuel assemblies~~, or during CORE ALTERATIONS~~. Movement of irradiated fuel assemblies ~~and CORE ALTERATIONS~~ must be suspended immediately to reduce the risk of accidents that would require CREATS actuation. This places the plant in a condition that minimizes risk. This does not preclude movement of fuel or other components to a safe position.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which CREATS Actuation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

CHANNEL CHECK acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

This SR is the performance of a COT once every 92 days on each required channel to ensure the ~~entire~~ channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.6.3

This SR is the performance of a TADOT of the Manual Initiation Function every 24 months. The Manual Initiation Function is tested up to, and including, the master relay coils.

The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints.

SR 3.3.6.4

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.5

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations are tested for the CREATS actuation instrumentation. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is acceptable based on instrument reliability and operating experience.

REFERENCES

1. UFSAR, Section 6.4.

2. ~~Letter from Robert C. Mecredy, RG&E, to Guy S. Vissing, NRC, Subject: Application for Amendment to Facility Operating License Control Room Emergency Air Treatment System CREATS Applicability Change (LCO 3.3.6 and LCO 3.7.9), dated July 21, 2000.~~

DA-NS-2000-057, Revision 2, Gas Decay Tank Rupture Offsite and Control Room Doses

3. 10 CFR 50, Appendix A, GDC 19

4. NUREG-0737, Section II.B.2, Dose Rate Criteria.

5. NUREG-0800, Section 6.4, Control Room Habitability Program

6. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

7. RG&E UFSAR Chapter 15 Transient Analysis Calculation Sheet 10.26, Rev 1, Control Room Isolation Time.