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ROBERT C. MECREDY
Vice President
Nuclear Operations

March 22, 2002

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Response to NRC Bulletin 2002-01, Subject: *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Clark:

On March 18, 2002, the Nuclear Regulatory Commission (NRC) issued the referenced Bulletin requesting that all addressees provide to the NRC a written response in accordance with the provisions of 10 CFR 50.54(f). Information requested relates to the integrity of the reactor coolant pressure boundary (including the reactor pressure vessel head) and the basis for concluding that applicable regulatory requirements are being met. The Bulletin requires that specific information be submitted within 15 and 60 days of the date of the Bulletin, and within 30 days after plant restart following the next inspection of the reactor pressure vessel head. The purpose of this letter is to provide the 15 day response (see enclosure). Since Ginna Station is currently shutdown and performing a refueling outage, RG&E requests that the NRC provide a response to this letter as soon as possible so that RG&E can most efficiently utilize its resources to deal with this issue. In addition, since the EPRI Materials Reliability Program (MRP) has committed to continue to provide information to the NRC concerning this Bulletin, RG&E will review the relevant information and update its response as appropriate within the Bulletin specified response time of 15 days. The remaining two responses will be provided at a later date.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by RG&E to make this submittal and that the foregoing is true and correct.

Any questions concerning this issue should be directed to Brian Flynn, Scheduling Manager at (585) 771-3734.

Very truly yours,

Executed on March 22, 2002

Robert C. Mecredy

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Enclosure

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ENCLOSURE
NRC BULLETIN 2002-01 REQUESTED INFORMATION

The following provides RG&E's response to the information required within 15 days of the date of Bulletin 2002-01. The responses required within 60 days of the date of the Bulletin and within 30 days after plant restart following the next inspection of the reactor pressure vessel head will be provided under separate cover. The items in bold italics represent the requested information as documented in the Bulletin. RG&E's response follows each requested item.

(1) Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

A. a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

RESPONSE:

Rochester Gas and Electric Corporation (RG&E) performs several inspections of the reactor pressure vessel head as summarized below. Since RG&E is currently in a refueling outage (RFO), these described items are with respect to previous outages. The response to item 1.D below provides a discussion of the inspections to be performed during the current RFO.

1. The ASME Section XI code requires VT-2 leakage examination on Class 1 components, including the reactor pressure vessel head each RFO. This inspection is performed in accordance with procedure PT-7, *ISI System Leakage Test, Reactor Coolant System* with the primary system (including the reactor pressure vessel head) fully assembled.

2. In 1993, following the discovery of leakage at a French pressurized water reactor in the early 1990's, RG&E performed visual inspections of the upper surface of the insulation located on the reactor pressure vessel head. The Ginna Station reactor pressure vessel head configuration is such that access to the upper head surface is restricted to existing CRDM Cooling Shroud HVAC duct connection ports. There are three such ports equidistant around the circumference of the CRDM Cooling Shroud. The duct openings are nominally 16 inches at the connection to the HVAC duct work. The purpose of these 1993 inspections was to ensure that no large deposits of boric acid existed on top of the insulation (i.e., other than minor flaking). The inspections continued to be performed each refueling outage as an enhanced Generic Letter (GL) 88-05 inspection performed by the Program Engineer and a VT-3 examiner with photographs taken to support trending. The only exception was during the 2000 RFO when the

visual inspection was performed by the refueling SRO with no photographs being taken.

3. RG&E also performed a video inspection in 1993 beneath the reactor pressure vessel head at the penetration to head interface (J-groove welds) confirming that there were no gross distortions of the welds.

4. RG&E performed an Eddy Current (EC) inspection of the inside wall of all reactor pressure vessel head penetrations in 1999. Extensive under the head work adjacent to J-grove welds again revealed no gross distortions. Our responses to Bulletin 2001-01 provides additional details (References 1 and 2).

B. an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse,

RESPONSE:

The Bulletin identifies that evaluations are on-going with respect to the cause of the degradation of the reactor pressure vessel head as observed at Davis-Besse. However, the Bulletin discusses several observations with respect to the degradation. These observations, and the abilities of the reactor pressure vessel head inspections at Ginna Station to effectively deal with them, are discussed below:

1. The base metal of the reactor pressure vessel head degraded near leaking penetration nozzles.

The 1999 EC inspection of the internal diameter of the Alloy 600 penetrations was not capable of examining the pressure vessel head material itself, nor the J-grove welds located on the underside of the head. However, if it were conservatively assumed that a through-wall CRDM crack existed during power operation, pressurized primary water containing boric acid would escape from the crack upward into the annular space between the reactor pressure vessel head and penetration. As the pressure decreases, some of the water would flash to steam, which would produce a high velocity steam jet. During this process, boric acid would remain in the liquid phase. However, the liquid would then boil because it is exposed to a high temperature environment, i.e., it is in contact with hot (> 212°F) metal or insulation. As the liquid escapes and boils, only non-volatile species in the water will remain (i.e., boric acid). Over a period of time, this process would be expected to lead to an accumulation of boric acid and voluminous corrosion product (significantly greater than the volume of wasted carbon steel) at the head/insulation interface, in the annulus between the insulation and penetration tube, or above the insulation. It would be expected that this accumulation would eventually exert sufficient force on the insulation to create: (1) a localized bulge, crack, displacement, etc., of the insulation blocks which should be visually detectable, and/or (2) crystalline deposits around the penetration or on top of the insulation which would also be visible. This displacement is due to the fact that the insulation consists of tight fitting block insulation that is covered by a layer of FiberFrax cement with a final coating of silicone resin (waterproofing). The insulation blocks are dense, inflexible (as demonstrated by minor surface cracking), and would therefore be expected to crack and be displaced upward. The escaping steam, if the leak were large enough, would also be expected to erode the insulation and thus produce visible evidence of leakage. It would also be detectable by RCS leakage detection systems. During cooldown, cool liquid under pressure could reach the top of the head when the metal temperature is below 212°F. Some evaporation will still occur as the temperature drops to ambient. The insulation around the leak location could soak up water during this period, but as soon as the head heats up again, the water will evaporate. Consequently, the leakage would be expected to produce visible evidence of insulation distortion and crystalline deposit.

RG&E has performed visual inspections of the external configuration of the insulation located on top of the reactor pressure vessel head inside of the CRDM Cooling Shroud as described in the response to item 1.A. No evidence of any significant distortion consistent with the accumulation of boric acid and corrosion products caused by wastage of the reactor vessel pressure head has ever been observed. Preliminary inspections during the current RFO also do not show any evidence of gross wastage or boric acid accumulation (note - these inspections are continuing). Furthermore, as discussed in the response to item 1.E, RG&E has reasonable assurance that there is no known through-wall leakage of the reactor coolant system.

2. **The reactor pressure vessel head had boric acid deposits in the vicinity of the degraded areas for several years.**

Since RG&E is planning to replace the reactor pressure vessel head during the Fall 2003 RFO, coupled with our 1999 EC inspection results, RG&E has not performed a bare metal inspection of the reactor pressure vessel head due to the insulation configuration as documented in Reference 1. However, the visual inspections of the insulation on top of the reactor pressure vessel head would be expected to identify boric acid deposits or corrosion products sufficient to cause accelerated wastage of the carbon steel was located beneath the insulation. Past visual inspections do not show these type of distortions.

3. **Some of the boric acid deposits on the top of the reactor pressure vessel head came from leaking CRDM flanges.**

   The Ginna Station CRDM design is different from that at Davis-Besse. Specifically, the comparable joint at Ginna Station utilizes a threaded connection that is subsequently seal welded. The insulation inspections performed since 1993 show no evidence of any large deposits of boric acid crystals located on top of the insulation. In addition, the response to item 1.C discusses how RG&E has addressed previous boric acid leakage from above the reactor pressure vessel head insulation. Finally, it is important to note that the reactor pressure vessel head insulation specification calls for a waterproof emulsifier to be coated on the top of the tight fitting insulation. The insulation is specified as block insulation with joints sealed by a layer of FiberFrax cement with a final coating of silicone resin (waterproofing). Although some cracks and minor exposed sections have developed in the cement coating, the insulation and coating are largely intact. This configuration would be expected to limit exposure of boric acid leaks onto the reactor pressure vessel head from above.

C. ***a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,***

<u>RESPONSE:</u>

There have been three instances of fluid containing boric acid potentially leaking from above the reactor pressure vessel head insulation as summarized below (see Attachment 1 for the relative locations of these leaks):

1. In 1971, there was a CRDM vent pin hole leak at the seismic restraint area. This area is located at the top of the control rod travel housing approximately 15 feet above the head. Pictures taken at the time of discovery show that the leakage was very localized. The area of stainless steel was cleaned at time of discovery such that no boric acid reached the reactor vessel pressure head.

2. On March 16, 1985, there was an instance where conoseal leaks occurred during refueling operations. Several gallons of primary water were

emitted due to three instrument port conoseals not being sufficiently torqued prior to RCS fill. The four instrument port penetrations are shown as penetrations #34, #35, #36, and #37 on Attachment 1 which are the outer most head penetrations and not near the center of the head as was the case at Davis-Besse. The affected exposed areas were cleaned and wiped down.

3. In 1991, seepage occurred at a lower instrument port conoseal for penetration #34. The 1991 Refueling Engineer log entry notes the removal of the plate around the conoseal and describes that there was no boric acid on the reactor pressure vessel head. The affected area was cleaned of all boric acid residue. Again, the conoseal is located outside the CRDM Cooling Shroud as described above.

No subsequent wetting of the insulation area has occurred. There have been no known instances of leakage from CRDM to CRDM adaptor seal welds at Ginna Station. Previous insulation visual inspections show no significant deposits of boric acid crystals on the insulation (i.e., other than minor flaking).

D. your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and

RESPONSE:

Ginna Station is currently in its planned 18 month RFO and does not plan to perform a bare metal examination of the reactor pressure vessel head. In response to the conditions reported in NRC Bulletin 2002-01 RG&E is planning the following inspection approach:

1. A visual inspection of the insulation on top of the reactor pressure vessel head will be performed inside the CRDM Cooling Shroud as follows:

    a. The inspection will look for any signs of boric acid leakage from above which could lead to boric acid accumulation on the carbon steel surface of the vessel. This enhanced GL 88-05 type inspection will be performed by qualified VT examiners and will specifically look for any boric acid leakage which may have originated from overhead areas such as the CRDM head adaptor to CRDM seal weld, or from other overhead areas of the CRDM housing.
    b. Photographs of the interface of the reactor pressure vessel head

penetrations and insulation will be taken to identify boric acid crystals and deformation of the insulation. The intention is to do 100% of this interface. These photographs will also be compared to previous photographs of the head region inside the CRDM Cooling Shroud in order to identify any changes from previous inspections.

c. A video tape of the region will be made for future reference.

2. Based on the visual and photographic inspection results of (1), RG&E will identify any suspect areas which require further investigation. Any penetrations which show signs of insulation distortion as described in the response to item 1.B above, or indications of a through-wall leak, will be further investigated (i.e., insulation will be removed).

3. Ultrasonic Testing (UT) will be performed to verify the thickness of the reactor pressure vessel head for the center penetration (#1) from beneath the head. The penetration will be UT inspected using two concentric inspection paths around the penetration. Additionally, four areas on the outside of the CRDM Cooling Shroud support ring on the downhill side of the four instrumentation ports will be UT inspected from the exterior surface of the reactor pressure vessel head. Any areas which are identified by ultrasonic examinations to be significantly thinner than the design wall thickness will be subject to additional examinations up to, and including, insulation removal.

Finally, RG&E plans to replace the reactor pressure vessel head during the Fall 2003 RFO. This activity is currently on schedule, with several templating activities ongoing during the current RFO. The replacement head will utilize numerous design improvements in materials, welding, and configuration to minimize the potential for the problems identified in the recent NRC Bulletins, and to facilitate future inspection of the head. Since RG&E is replacing the reactor vessel pressure head in the first RFO following the current outage, the next inspection would not occur until the Spring 2005 RFO. RG&E will determine the appropriate scope of these future inspections based on industry experience to ensure continuing reactor pressure vessel head integrity.

E. ***your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:***

RESPONSE:

RG&E has concluded there is reasonable assurance that regulatory requirements are currently being met, and will continue to be met, given the inspections described in the response to item 1.D. Currently, Ginna Station is in a refueling outage and has committed to performing the inspections described in the response to item 1.D prior to startup. Following startup, Ginna Station will continue to meet the associated regulatory requirements. The Applicable Regulatory Requirements section of Bulletin 2002-01 describes several requirements with respect to reactor coolant pressure boundary integrity, including technical specifications, general design criteria (GC), 10 CFR 50.55a, and Appendix B to 10 CFR Part 50. Each of these are described below:

a. Technical Specifications - The Ginna Station Improved Technical Specifications (ITS) LCO 3.4.13 requires that reactor coolant system (RCS) leakage be limited to 10 gpm identified leakage, 1 gpm unidentified leakage, and no pressure boundary leakage. RG&E has performed a review of the leakage logs since December 1999 and determined that there has been no increased trend of unidentified leakage that would be indicative of through-wall leakage within the RCS. As shown in Attachment 2, the average unidentified leakage rate is 0.0647 gpm during at-power conditions. This value tends to increase over a cycle but resets upon startup. This is further supported by a review of containment gaseous and particulate radioactivity monitor data which shows no leakage trend (see Attachment 2). It should be noted that while ITS only requires RCS inventory balances every 72 hours, RG&E procedurally performs these once per shift. The RCS leakage surveillances also include a review of containment sump actuations, containment fan cooler condensate collection dumps, and containment gaseous and particulate radioactivity monitor status. These values are evaluated each shift and signed off by the shift supervisor. Any significant increase in value (e.g., RCS makeup rate increases by 0.25 gpm over the normal rate) requires further evaluation.

   It should be noted that for significant, accelerated wastage of the reactor vessel pressure head to occur similar to Davis-Besse, the boric acid must be exposed to an oxygenated environment. The primary system oxygen levels are restricted in accordance with the Ginna Station Technical Requirements Manual. Consequently, the only source of significant oxygen would be from the containment atmosphere. However, the containment atmosphere is monitored for radionuclides as described above, and any increase in RCS leakage would be detectable.

b. GDC (10 CFR 50, Appendix A) - Ginna Station was designed, built, and licensed during the 1960s which was prior to the codification of the GDC in 1971. Instead, Ginna Station was designed to proposed Atomic Industrial Forum (AIF) versions of the GDC that were issued for comment by the Atomic Energy Commission (AEC) on July 10, 1967. The AIF-GDC have similar requirements to the Bulletin specified GDC 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). These are AIF-GDC 9 (Reactor Coolant Pressure Boundary), AIF-GDC 34 (Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention), and AIF-GDC 36 (Reactor Coolant Pressure Boundary Surveillance). RG&E believes that these AIF-GDC continue to be met given the reasons described within this letter.

c. <u>10 CFR 50.55a</u> - The Bulletin identifies that "10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall degradation of the reactor pressure vessel head penetration nozzles." It also references IWB-3522.1(c) and (d) with respect to conditions that require corrective action "including the detection of leakage from insulated components, and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage." The visual inspection of the insulation each RFO would provide indication of discoloration or accumulated residues. Plant walkdowns during startup following each RFO (procedure PT-7) look for evidence of leakage on floors and through insulation. Also, the review of unidentified leakage records and containment gaseous and particulate radioactivity monitor data indicates that there is no known through-wall leakage of RCS components.

d. <u>10 CFR 50, Appendix B (Criterion V, IX, and XVI)</u> - The consideration of each one of these criterion is described below:

   i. Criterion V - The inspections of the reactor pressure vessel head described in item (1).D above will be documented in procedures and retained as quality records.
   ii. Criterion IX - The inspections described in item (1).D above will be performed with qualified personnel using plant procedures.
   iii. Criterion XVI - Any degradation of the reactor coolant boundary identified by the described inspections will be placed within the Ginna Station corrective action program. Significant changes in RCS leakage while at-power are procedurally required to be further evaluated. If warranted, these increases would also be entered into the corrective action process.

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met,

discuss your plans for plant shutdown and inspection.

RESPONSE:

Since RG&E has determined that there is reasonable assurance that regulatory requirements are being met, there are no future plans for plant shutdown and inspection per this Bulletin.

(2) ***If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.***

RESPONSE:

Currently, Ginna Station is in a refueling outage and has committed to performing the inspections described in the response to item 1.D prior to startup.

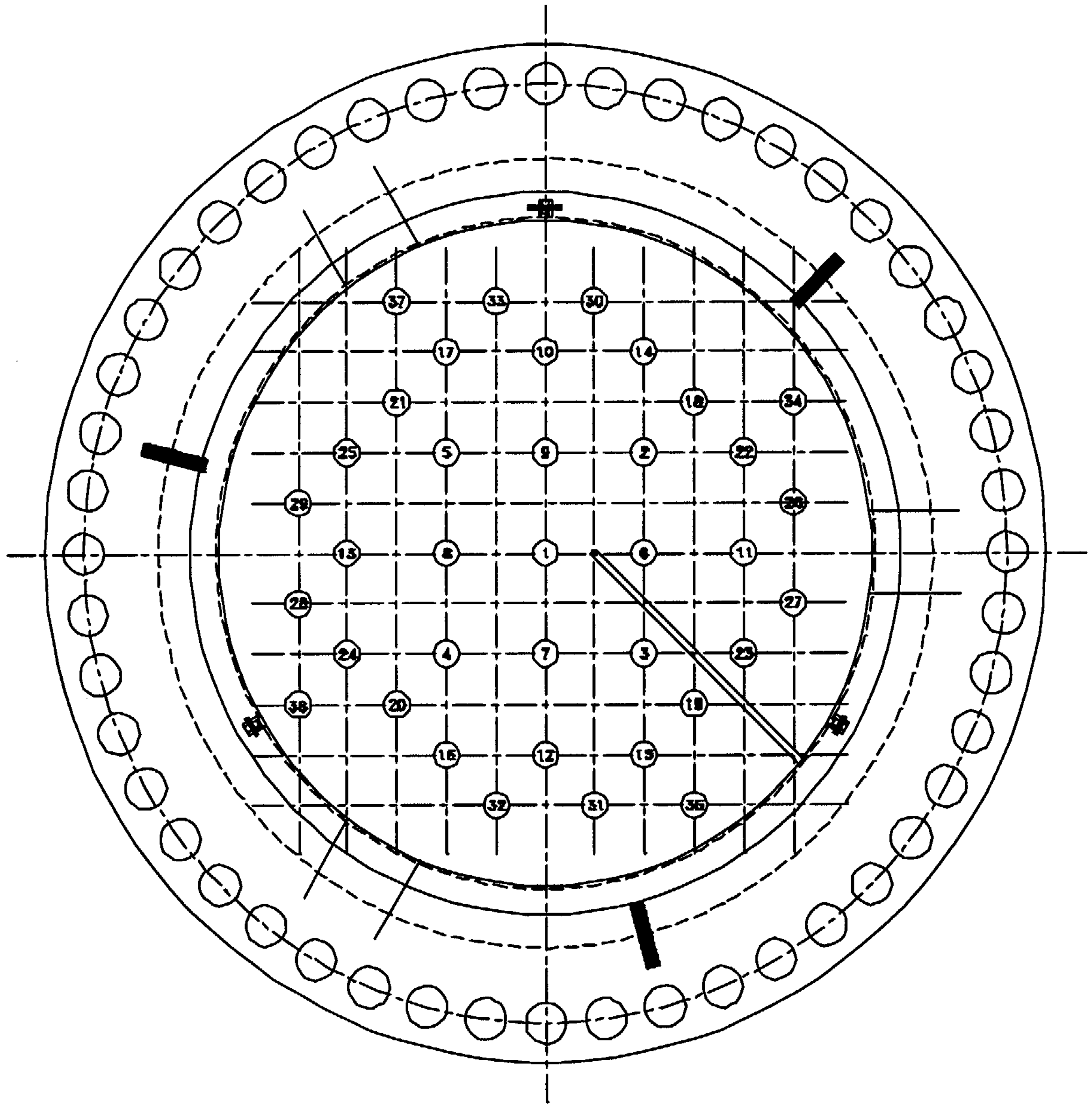
References:

1. Letter from R.C. Mecredy, RG&E, to R.L. Clark, NRC, Subject: *Response to NRC Bulletin 2001-01*, dated September 4, 2001.

2. Letter from R.C. Mecredy, RG&E, to R.L. Clark, NRC, Subject: *Supplemental Response to NRC Bulletin 2001-01*, dated December 31, 2001.

Attachment 1

Top View of Ginna Station Reactor Pressure Vessel Head



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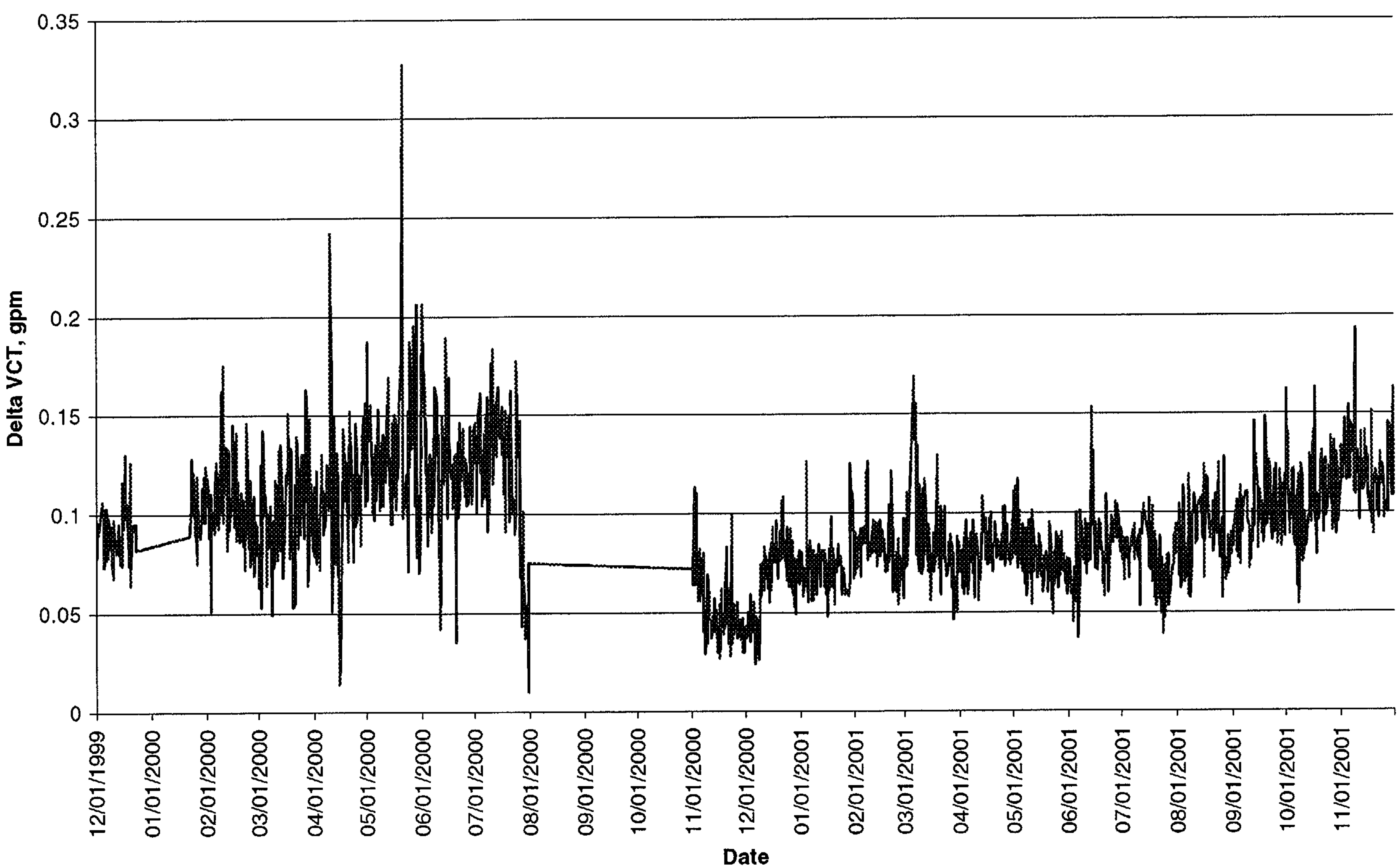
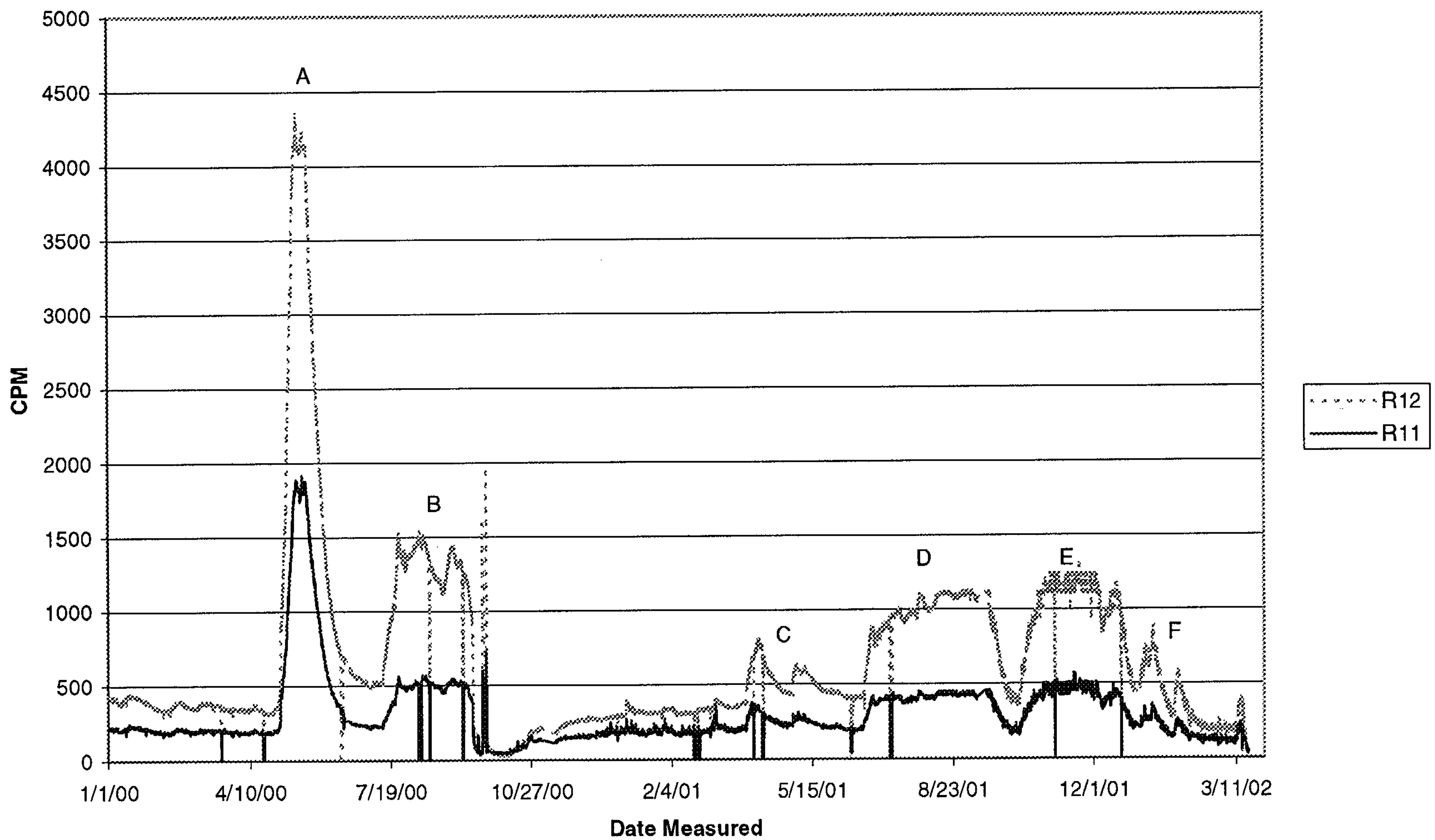
Attachment 2

Recent Ginna Station RCS Leakage History

Ginna Station Unidentified Leakage



R-11 (Particulate) & R-12 (Gaseous)



Key:

A	AOV 950 packing adjusted
B	RPs lined up to sample B loop
C	AOV 951 and 953 gland leakage
D	AR 2001-1695 determined no cause
E	PT-39 (stroked AOV 951)
F	Determined no cause