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ROBERT C MECREDY
Vice President
Nuclear Operations

January 21, 2003

Mr. Robert L. Clark
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Response to Request for Additional Information Related to Bulletin 2002-01
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

References: (1) Letter from Robert Clark, NRC, to Robert C. Mecredy, RG&E, Subject: *Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60-Day Response for R.E. Ginna Nuclear Power Plant Request for Additional Information (TAC. No. MB4548)*, dated November 22, 2002

Dear Mr. Clark:

In Reference (1). the Nuclear Regulatory Commission (NRC) provided Rochester Gas and Electric Corporation (RG&E) with a Request for Additional Information (RAI) related to Bulletin 2002-01. Attachment B to this letter contains our response to the subject RAI while Attachment C contains several plant specific documents referenced within this response. A table is provided in Attachment A identifying all commitments contained within the response.

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 Mr. Brian Flynn, Manager. Primary / Reactor Systems at (585) 771-3734.

Very truly yours,

Executed on January 21. 2003

Robert C. Mecredy

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Attachment A
List of Regulatory Commitments

The following table identifies those actions committed to by Rochester Gas & Electric (RG&E) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Brian Flynn, Manager, Primary / Reactor Systems at (585) 771-3734.

REGULATORY COMMITMENT	DUE DATE
Conduct an inspection of the metal surface of the lower RPV head in the vicinity of the instrument tube penetrations. This includes an attempt to remove fibrous insulation and use of borescopes and enhanced lighting.	Prior to startup following the 2003 fall refueling outage (RFO).
If the 2003 RFO inspection is not able to evaluate all penetrations, remove and replace insulation on the bottom RPV head as required to conduct the necessary inspections.	Prior to startup following the 2005 RFO.

Attachment B
Response to NRC RAI Dated November 22, 2002

1. **Provide detailed information on, and the technical basis for, the inspection techniques scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head.)**

Locations of Alloy 600 and weld material 82/182 within the Ginna RCPB is limited to the Reactor Pressure Vessel (RPV) and Steam Generators (SGs). Alloy 600 components are the CRDM penetrations, RPV bottom head instrument penetrations, and lower internals radial support lugs. These components were welded to the vessel using Alloy 82/182 weld material. In addition, the SI nozzle on the vessel has a ½ inch Alloy 82/182 weld overlay on the ID surface to form a 'boss' to meet the mating nozzle on the lower internals. With respect to these materials, only the upper and lower RPV head locations have the potential to produce reactor coolant pressure boundary leakage. As discussed in the previous response to Bulletin 2002-01 (Reference 1), Ginna is replacing the RPV closure head at its next refueling outage in the fall of 2003. Details of previous inspections were provided at a meeting between RG&E and the NRC on December 12, 2001 (Reference 2). Since there will be no Alloy 600 or Alloy 82/182 weld material in the replacement RPV closure head, further discussion of this area is not provided here. The tubesheets on the SG's were overlaid with Alloy 82 weld deposit, however this was done prior to stress relief by post-weld heat treatment. Therefore, this material is not subject to high residual stresses.

Given the above, the primary area of interest is the RPV lower head penetrations. As previously noted in NRC Bulletins 2002-01 and 2002-02, susceptibility to stress corrosion cracking is related to "operating conditions (in particular the operating temperature and time)" and the presence of "high residual stresses resulting from initial manufacture and the impact of tube straightening that may have been needed after welding". The RPV bottom head penetrations are much less susceptible to stress corrosion than the CRDM penetrations in that:

1. The operating temperature is much lower (for Ginna Tcold is 533 degF).
2. The penetrations are smaller requiring less weld material (less residual stress).
3. The verticality requirements for the lower head penetrations are less stringent than those for the CRDM penetrations Therefore it is not expected that straightening operations would have been necessary on these penetrations.
4. The lower reactor vessel head was stress relieved after these penetration were installed and welded in place, thereby lowering residual stress levels.

The lower RPV head is insulated with permanently attached reflective metal insulation. Removal of this insulation would require its destruction, and replacement with new insulation. The design drawings for this insulation indicates a minimum gap to the RPV of ½ inch, and a minimum gap of ¼ inch around each of the instrument penetrations. The specification indicates that the ¼ " gap around each penetration is to be filled with fibrous insulation. Due to the permanent nature of the bottom head insulation, and the lack of any industry experience with leakage from these penetrations, RG&E has not attempted to perform any bare metal inspections in this area. RG&E does perform visual inspections of this area with the insulation in place at operating pressure and temperature for signs of leakage during refueling outages. The inspection is performed by VT-2 qualified personnel each refueling outage (18 months). Due to the location, it is expected that any leakage would drip downward away from the head and be detected. However, since there is increased concern over the potential consequences of leakage in this area, RG&E intends to conduct a best effort inspection of the metal surface of the lower head in the vicinity of these penetrations during the next refueling outage. This would include an attempt to remove fibrous insulation, if possible, and use of borescopes and enhanced lighting. If it is concluded that this inspection is not able to evaluate all penetrations, measurements and planning will be made to remove and replace insulation, as required, such that an adequate inspection of the bottom RPV head can be performed. This would occur during the following refueling outage (Spring 2005).

2. **Provide the technical basis for determining whether or not insulation is removed to examine <u>all</u> locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.**

Criteria for removal of insulation when performing visual examinations for leakage are described in Procedure NDE-VT-109 (see Attachment C). This procedure requires systems containing boric acid to have insulation removed for all bolted connections. Additionally, this procedure requires that if any leakage is detected, that the examiner shall determine the actual source of leakage which would typically require insulation removal. As described in the response to Question #1, areas susceptible to primary water stress corrosion cracking include the CRDM penetrations and RPV bottom head instrument penetrations. Insulation removal has not been required for these locations; however, actions to inspect the lower head and replace the upper head are described in the response to Question #1. With the exception of the insulation on the reactor vessel below the flange, all insulation on the reactor coolant system is removable.

3. **Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in <u>inaccessible areas</u>. In addition, describe the**

degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Areas considered as inaccessible or of limited accessibility during power operation include the Containment Building, and certain areas within the Chemical and Volume Control System (CVCS) which cannot be accessed due to high radiation fields. The containment is walked down on a monthly basis while at power. This walkdown does not include entry into the RCS loop areas, nor areas directly around the Reactor vessel due to high radiation fields. However, there are several cameras located within containment, including the loop areas but not within Sump A, that can be viewed from the control room. These cameras have identified leakage in the containment in the past (Reference 3 discusses a RCS leak that was confirmed via use of these cameras). However, the sensitivity of the cameras to radiation does tend to limit their usefulness during extended periods of operation. The frequency of containment walkdowns are in accordance with surveillances required by Ginna Technical Specifications. Additional inspections are triggered based on increased RCS leakage in accordance with our leakage monitoring program. For increased leakage within containment, Procedure S-12.2 requires that a sample of sump water be taken to determine if the source of leakage is from a borated system (i.e., RCS or SI) or not (e.g., Service Water, Feedwater). If the source of leakage is borated, it is assumed to be from the RCS until it can be positively identified. Appropriate actions in accordance with Technical Specification requirements are taken, up to and including, plant shutdown.

4. **Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis that was used to establish the acceptance criteria. In addition,**

   a. **If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or**

   b. **If observed leakage is not determined to be acceptable, describe what actions are taken to address the leakage.**

When leakage from a mechanical joint is identified, a Work Order / Trouble Report is initiated by plant personnel. If this leakage is from systems addressed by Technical Specifications or the Maintenance Rule (which includes all systems containing boric acid), an ACTION Report is initiated in accordance with the Ginna Corrective Action program as described in procedure IP-CAP-1, "ABNORMAL CONDITION TRACKING INITIATION OR NOTIFICATION (ACTION) REPORT." For equipment related issues, this procedure requires that an operability determination be made. While there is no prescribed method nor acceptance criteria for leakage other than RCS leakage, the procedure would require a technical evaluation/operability assessment to be performed by Engineering and concurred by Operations for the leaking component if the operability of the affected component could not be readily determined. This evaluation would require assessment of the design and licensing requirements for the component (e.g., Technical Specifications, accident analysis assumptions, Codes and Standards, etc).

For components that remain acceptable for continued operation, leakage is monitored and trended as described in Attachment 2 of the procedure IP-HSC-3, "Housekeeping Control" (see Attachment C). If observed leakage is not determined to be acceptable, the component is declared inoperable and a work order to repair or replace the component is planned and implemented in accordance with the station work control program. Also, for any condition where boric acid is observed on a carbon steel component, visual examination by a qualified examiner is required to determined operability.

5. **Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through wall cracking in the bottom reactor pressure vessel head incore instrument nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with the loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of leakage of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.**

RG&E has not conducted a detailed analysis to determine the potential leakrate from a crack in a RPV bottom head nozzle. However, it should be noted that the assembly of these lower nozzles does not include an interference fit, as with the CRDM penetrations, and therefore resistance to leakage is lower in this area. Additionally, unlike the upper head, by nature of its location, 'pooling' of leakage on the surface of the bottom head which may cause a more aggressive environment is not likely.

Due in part to its relatively small containment size, Ginna has demonstrated leakage detection capability down to 0.013 gpm within 20 minutes (UFSAR Section 5.2.5.1), utilizing the containment air particulate monitor (R-11) (see Reference 3). RG&E believes that this level of detection, in conjunction with the visual inspections described in response to question 1, is adequate to detect leakage well before a significant amount of accumulation and concentration can occur to cause boric acid corrosion of the RPV base material.

From a leak path standpoint, the RPV bottom head instrument penetrations are located in the Sump A at an approximate elevation of 225'. This is approximately 10' below the containment floor elevation. Therefore, complete loss of integrity of a lower head penetration would result in the eventual filling of the Sump A to the floor elevation provided that the leak rate is sufficient to initiate a SI signal which isolates the Sump A sump pumps. This is a long-term result of any RCS LOCA. It should be noted that leakage from a bottom head penetration in excess of the makeup capability of the CVCS system is highly unlikely, given the small internal diameter of these penetrations, and the presence of the incore thimble within the penetration. With the exception of the non-safety related containment sump pumps and level instrumentation, there are no components in the leakage path from the bottom head of the RPV. Neither of these components is credited for post-accident recovery.

6. **Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.**

As describe in response to Question #5, Ginna has demonstrated leakage detection capability down to 0.013 gpm within 20 minutes (UFSAR Section 5.2.5.1). RG&E believes that this detection capability coupled with periodic visual inspections is sufficient to detect boric acid leakage within the RCPB from small diameter nozzles prior to significant boric acid corrosion.

As described in response to Question #4, evidence of possible leakage from any component, including small diameter nozzles is required to be evaluated in accordance with the Ginna corrective action program. While there is no prescribed evaluation methodology, assessment of impact on the nozzle, and any component in the leak path would be required.

7. **Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.**

Susceptibility models or consequence models are currently not utilized in determining the inspection frequency, nor method of inspection within the ASME Section XI Inservice Inspection Program. We believe that use of such models, including risk-based ISI, would result in less stringent inspections due to inherent conservatisms in our current program. Currently, every effort is made to access areas, remove insulation and address any available Operating Experience. Constraints to inspections are limited to original design conditions, such as the permanently installed reactor vessel insulation described in response to Question #1, and availability of effective inspection technology for given configurations.

8. **Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.**

RG&E is not aware of any recommendations made by the Ginna reactor vendor, Westinghouse on visual inspections of nozzles with Alloy 600/82/182. Through a request from the Westinghouse Owners Group (WOG), Westinghouse has conducted an extensive review of its databases and applicable communications to determine what recommendations Westinghouse had made to the owners of Westinghouse NSSSs on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The detailed review of this information did not identify any Westinghouse recommendations on visual inspections of Alloy 600/82/182 locations in Westinghouse NSSSs.

9. **Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the *Code of Federal Regulations* (10CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.**

In response to NRC Bulletin 2002-01, RG&E conducted a self-assessment of our Boric Acid Leakage Control program (Reference 4). This assessment concluded that the program met the requirements of Generic Letter 88-05. A copy of this assessment is provided in Attachment C. The assessment did identify several areas for program improvement. RG&E is in the process of incorporating these enhancements into its program. These improvements incorporate more explicit guidance for the evaluation of identified leakage for degradation. With respect to ASME Section XI, paragraph IWA-5250 (b), whenever leakage on a system containing boric acid is detected the corrective action program is invoked. Residue is cleaned from the surface so that an accurate assessment of areas of general corrosion can be made. The degree of degradation, and an evaluation of the component's ability to meet acceptance criteria for continued operation is performed. This is consistent with the guidance of the referenced paragraph.

<u>References:</u>

1. Letter from Robert C. Mecredy, RG&E, to Robert L. Clark, NRC, Subject: *Response to NRC Bulletin 2002-01, Subject: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*, dated March 22, 2002.
2. Public Meeting Between RG&E and NRC, December 12, 2001.
3. Letter from Robert C. Mecredy, RG&E, to Guy S. Vissing, NRC, Subject: *Discussion of Leak Detection System in Support of Leak-Before-Break (LBB) Application of Portions Residual Heat Removal (RHR) System (TAC NO. MA0389)*, dated September 16, 1998.
4. Letter from Robert C. Mecredy, RG&E, to Robert L. Clark, NRC, Subject: *60 Day Response to NRC Bulletin 2002-01, Subject: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*, dated May 17, 2002.

Attachment C

Procedure IP-HSC-3, Housekeeping Control, Revision 5

Self Assessment #202-0037, Effectiveness of the Ginna Station Program for Prevention of Boric Acid Corrosion

VT-109, Visual Examination for Leakage, Revision 5

RG&E	NUCLEAR OPERATIONS GROUP INTERFACE PROCEDURE	IP-HSC-3 Revision 5 Page 1 of 7

Controlled Copy #__________

HOUSEKEEPING CONTROL

John F. Smith
Responsible Manager

Robert C. Mecredy
Robert C. Mecredy
Vice President
Nuclear Operations Group

5-18-01
Effective Date

1.0 PURPOSE

The purpose of this procedure is to describe the Housekeeping Control Program at Ginna Station.

2.0 REFERENCES

2.1 Source Documents

2.1.1 Regulatory Guide 1.39, Revision 2, 1977, Housekeeping Requirements for Water-Cooled Nuclear Power Plants

2.1.2 ANSI N45.2.3, 1973, Housekeeping During the Construction Phase of Nuclear Power Plants

2.1.3 ND-HSC, Housekeeping and System Cleanness and Foreign Material Exclusion (FME)

2.2 Development Documents

2.2.1 IP-HSC-1, Foreign Material Exclusion

2.2.2 IP-HSC-2, System Cleanness

2.2.3 SOER 95-1, Reducing Events Resulting From Foreign Material Intrusion

2.3 Use Documents

2.3.1 A-54.7, Fire Protection

2.3.2 A-54.7.1, Safety Coordinator/Inspector Tour

2.3.3 A-1603.3, Work Order Planning

2.3.4 M-1306, Ginna Station Material Condition Inspection Program

2.3.5 IP-DES-3, Temporary Modifications

2.3.6 IP-SEP-3, Ginna Station Tour Program

3.0 INSTRUCTIONS

3.1 Definitions

3.1.1 Housekeeping Zones - areas at Ginna Station are categorized into five (5) zones of housekeeping control. The zones are defined as follows:

<u>NOTE</u> Personnel access control, protective clothing, or material accountability may be required in any of the Housekeeping Zones for Security, Radiation Protection or quality reasons.

<u>NOTE</u> Zone definitions are ANSI (N45.2.3) definitions, modified as per Regulatory Guide 1.39, c(3) for applicability to Ginna Station.

<u>NOTE</u> Attachment 1, classifies each Ginna Station area into its corresponding zone level.

Zone I Highest area of cleanness, requiring clean clothing changes, including use of shoe covers, head covers and gloves to protect equipment from outside contamination. These areas also require material control, filtered air, material pre-cleaning, personnel access control and no use of tobacco or eating. Zone I is not applicable to any area at Ginna Station.

Zone II Cleanness requirements, less restrictive than Zone I, but where foreign matter may have detrimental effects; equivalent to FMEA Zone 1 as defined in procedure IP-HSC-1.

Zone III Areas less restrictive than Zones I and II, but requiring access control over personnel and material.

Zone IV Areas where it is desired to maintain good housekeeping for material and equipment protection or for health and fire hazards.

Zone V Unrestricted areas requiring good housekeeping practices only.

3.1.2 Leakage Monitoring Tag - a tag affixed in the area of leakage from a plant component. The Leakage Monitoring Process is described in Attachment 2, Part 4 of this procedure and an example of the tag is located on Attachment 4 of this procedure.

3.1.3 Material Accountability - tool/material logs and tethering as required by approved procedures used to perform the work in/around the vessel/systems.

3.1.4 Personnel Access Control - Security/Radiation Protection authorization required for entry.

3.1.5 Protective Clothing - the clothing to protect against radiation, contamination and any other physical hazard as specified for the task.

3.1.6 Responsibility Identification Tag - a tag affixed to extension/electrical cords, installed ladders, temporary hose(s) or electrical routings, sample collection containers, miscellaneous required equipment to support ongoing, work/sampling activities to identify the responsible group and reason for the use of the item. An example of this tag is included as Attachment 3 to this procedure.

3.2 General Housekeeping

<u>NOTE</u> Housekeeping standards identified in Attachment 2, Part I deal with acceptable plant conditions and Part II deals with acceptable work area conditions.

3.2.1 Housekeeping practices shall recognize requirements for control of radiation zones and work activities, conditions, and environments that can affect nuclear safety and protection of personnel and equipment.

3.2.2 Housekeeping shall include all activities related to the cleanness and non-radiological contamination of facilities, materials, and equipment. Housekeeping practices shall recognize and support the requirements of other Directives, including:

A. Conduct of maintenance activities in accordance with ND-MAI, "Maintenance."

(Step 3.2.2 contd)

B. Fire prevention and protection including disposal of combustible material and debris in accordance with ND-FPP, "Fire Protection Program."

C. Radioactive contamination control in accordance with ND-RPP, "Radiation Protection Program."

D. Storage of solid radioactive waste in accordance with ND-RMP, "Radiological Material Processing, Transport and Disposal."

E. Use and control of organic materials in accordance with ND-SNM, "Special Nuclear Material Accountability."

F. ND-HSC, Housekeeping and System Cleanness and Foreign Material Exclusion (FME)

3.2.3 Closure inspections are always required immediately prior to final closure of systems and components by the responsible group performing the closure. Furthermore, the completion of the closure inspection for safety-related and important to safety equipment/components shall be performed by Quality Control (QC) and documented.

3.3 Control of Site Area, Facilities, Materials, and Equipment

3.3.1 The control of tools, equipment, and materials used in cleanness zone Levels I, II, and III shall be maintained to prevent the inadvertent inclusion of deleterious materials or objects in systems consequential to safety. Such items shall be controlled through use of log sheets, tethers, and lanyards where appropriate. Specifically, tools and other items used during cleaning and flushing which may fall or drop into the system or component shall be attached to either the worker or the outside of the system with tethers or lanyards.

3.3.2 Work areas shall be adequately lighted, ventilated, and accessible for work being performed. Temporary lighting may be utilized. Ventilation shall prevent dust accumulation, noxious fumes, and temperature extremes. Barriers, screens, shields, and access restrictions shall be provided for high noise areas, as necessary.

3.3.3 Work areas shall be kept sufficiently clean and orderly so that work activities can proceed in an efficient manner that will produce and maintain quality and promote safety. Work groups are responsible to maintain cleanness of the work area when performing work, and shall return work areas to pre-work conditions on completion of work.

3.3.4 Electric control panels/cabinets shall not be utilized for storage purposes.

3.3.5 Any work within 15 ft. of a smoke detector which could cause airborne particles such as dust, vapors or other contaminates shall require Fire Protection Engineer evaluation of the area before work commences.

3.3.6 Where large accumulations of materials occur on a non-routine basis, the material shall be promptly removed or stored neatly.

A. Garbage, trash, scrap, litter, and other excess materials shall be collected and removed from the work area at the completion of each job or at the end of the day for jobs longer than one (1) day.

B. Excess material shall not be allowed to accumulate and create conditions that will adversely affect quality, plant or personnel safety.

3.3.7 Disposal of cleaning materials shall be accomplished in accordance with the requirements of ND-ENV, "Environmental Protection," so additional hazards are not created.

3.3.8 Materials and equipment delivered to the work area shall be placed in an accessible location such that it will not hinder or be damaged by the work in progress.

3.3.9 The use, location, and deployment of tools, supplies, and equipment shall be regulated to keep access and work areas clear and prevent conditions that will be adverse to quality. Such items shall be identified with the work in progress.

3.3.10 Periodic surveillances of work areas such as shops, laboratories, storage areas, and plant equipment areas shall be performed to verify adequate implementation of housekeeping requirements.

(Step 3.3.10 contd)

A. Supervisory personnel shall be assigned responsibility for periodic inspection tours of specific plant areas for compliance with the requirements of this directive.

B. Quality Assurance (QA) shall schedule surveillances of housekeeping activities in accordance with ND-ASU, "Assessments and Surveillances."

3.3.11 Surveillances of installed items shall ensure, as appropriate, the adequacy of maintenance of protection, preservation of precautionary signs, preservation of item identity, and protection from fire, weather, movement of materials and equipment, and other factors that may result in damage to items.

3.3.12 A leakage monitoring process has been established to ensure that leakage on plant components is monitored and the components as well as the surrounding areas are kept clean, until maintenance can be performed. The leakage monitoring process is described in Attachment 2, Part 4 of this procedure.

4.0 RECORDS

None.

5.0 ATTACHMENTS

5.1 Attachment 1, Housekeeping Zones

5.2 Attachment 2, Housekeeping Standards
- Part 1 - Acceptable Plant Conditions
- Part 2 - Acceptable Work Area Conditions
- Part 3 - Guidance of Use of Responsibility Identification Tag
- Part 4 - Leakage Monitoring

5.3 Attachment 3, Responsibility Identification Tag

5.4 Attachment 4, Leakage Monitoring Tag

Attachment 1

Housekeeping Zones

Zone I — Not applicable to any area at Ginna Station

NOTE — Zone II is applicable to FMEA Zone 1 as defined in procedure IP-HSC-1

Zone II — Steam Generator Tent and Primary Side
Steam Generator Secondary Side
Reactor Coolant Pump
Pressurizer
Refueling area during refuel operations
Spent Fuel Pool (inside fenced area)
Turbine Electrical Generator - generator end
Any other primary system equipment where tools will be used inside

Zone III — Vital areas or radiation controlled areas within the plant protected area; and the old Steam Generator Storage Facility

Zone IV — Non-vital and non-radiation controlled plant areas inside the plant protected area; and main warehouse

Zone V — Station facilities or grounds outside the plant protected area, except main warehouse

Attachment 2

Housekeeping Standards
Part 1 - Acceptable Plant Conditions

Site Roads, Parking Lots and Grounds

Site roads, parking lots and grounds are to be kept free of debris and non-permanent objects. These areas reflect care and pride in the appearance of the site. Roads and drainways are maintained as necessary for optimum function. All road signs, lights, pavement markings, and other visual aids are to be kept clear and usable according to established standards.

In order to maintain these standards:

1. Areas are to be kept clean of litter including paper, cans, bottles, cigarette butts and any other debris.

2. Grass areas are to be mowed according to established schedules.

3. Any materials for temporary storage within these areas shall be approved prior to storage by the responsible group and by the Safety and Security Sections.

Structures and Building Exteriors

Structures and building exteriors (steel, wood, masonry, etc.) retain their original surface integrity, appearance and utility.

In order to maintain this standard:

1. Leaky roofs shall be written up in a trouble report and promptly repaired.

2. Visible holes in exterior or facade walls shall have a work order written for timely repair.

3. Cracks or possible structural problems in buildings shall be investigated and repaired as necessary.

Office Areas

All office areas are kept neat and orderly. Floors, walls, ceilings, and office furnishings are maintained as installed or finished, free from foreign material, blemishes or abusive damage. Walls display only orderly items such as photographs, posters, or calendars that are socially acceptable.

In order to maintain these standards:

1. Work materials such as files, books, tools, drawings, or paperwork are to be properly stored following use.

2. Work related consumable items such as paper, tape, lubricant, detergent, or scrap material are to be disposed of only in appropriate waste containers

3. Personal consumable items such as disposable food or drink containers, or clothing are to be disposed of only in appropriate waste containers.

4. Contract cleaning services shall regularly clean floors, walls, and furnishings, and empty waste containers

Housekeeping Standards
Part 1 - Acceptable Plant Conditions (contd)

5. Temporary signs or notes are to be placed only in appropriate locations. General plant areas are not appropriate.

6. Any permanent signs shall be approved for use prior to display, and keep consistent with other permanent signs in the general area.

7. Personal pictures, posters or calendars shall be in good taste and not offensive.

Lunchroom, Coffee Areas, Locker Rooms

Areas provided for rest, food preparation or sanitary purposes reflect cleanliness. All surfaces are kept free of unauthorized markings or debris.

In order to maintain this standard:

1. Personal articles are to be stored in assigned lockers.

2. Towels are to be removed from the shower area after use.

3. Litter including paper, cans, bottles, and any other debris is to be picked up and disposed of properly before leaving the area.

4. Debris on tables is to be properly disposed of after use.

5. Spilled food or drink is to be cleaned by the person responsible for the spill.

Material Storage

Warehouse and yard storage areas are properly delineated (marked) with fixtures in place to accept specified material inventories. Prior to storage, the material or equipment is identified (labeled) for easy access and retrieval As necessary, material or equipment is prepared for inclement weather and acclimated to the storage environment

Inventory Control personnel maintain material access, material condition, and the housekeeping of material storage areas. Any work groups or individuals responsible for the warehouse or yard storage of materials must meet the same standards as station material control personnel.

In order to maintain these standards:

1. Hazardous chemicals, oils, greases, or other liquids shall be properly segregated, labeled, and stored in approved containers.

2. Shelves, cabinets, racks, drawers, or other stacked storage areas shall be kept orderly, neat, and labeled.

Attachment 2 (contd)

Housekeeping Standards
Part 2 - Acceptable Work Area Conditions

Common Work Areas

Common work areas remain clean and orderly during the job, and are maintained by the personnel performing work in the area. Tools and materials are properly stored immediately following completion of operation, maintenance, or testing by the people performing the work. Any debris created during the course of work is properly disposed of before or at completion of the job.

In order to maintain these standards:

1. Work area conditions such as painting, insulation, or labeling are to be provided by supplemental support staff as necessary through the use of maintenance work orders or label requests.

2. Special cleaning conditions such as final touch-up or decontamination are to be provided by supplemental housekeeping support staff as necessary through the use of maintenance work orders.

3. Permanent signs are to be approved and consistent with other permanent signs in the work area

4. Temporary signs, markers, tapes or labels shall be approved for use in plant areas.

5. Good housekeeping and safety practices are to be maintained throughout the work area These include (but are not limited to):

    a. Fastening or securing electrical covers on terminal boxes, cabinets, panels, or other equipment with the proper closures, bolts or screws.

    b. Routing temporary electrical cables, wires, hoses or pipes to protect people from tripping, or being shocked or burned. All temporary routing shall be marked with Temporary Modification tags if the routing is within the scope of procedure IP-DES-3. If the routing is not within the scope of the -IP-DES-3 procedure or installation and removal are not covered by a procedure or written instructions, and the routing will not be removed by the end of the work shift, then the routing should be marked with a Responsibility Identification Tag.

    c. Securing compressed gas bottles properly.

    d. Maintaining lights, switches and plugs.

    e. Ensuring that all floors and horizontal surfaces are free of dust and oil.

Work areas are set-up and cordoned off (if necessary) to assure the safety of other workers in the area, or to control materials or tools on the job. All work areas are kept in a safe, orderly manner. All tools, parts or equipment are stored and retrieved in a manner to reflect the professionalism of those who work in the area.

In order to maintain these standards:

When planning the work area:

1. Wait until set-up of the work area, as necessary.

2. In seismic areas, wait to set-up scaffolding until it is needed.

Housekeeping Standards
Part 2 - Acceptable Work Area Conditions (contd)

3. Prevent blocking walkways or exits.

4. Be aware of safety at all times.

At the end of each work shift, leave the area in a secure condition that includes:

1. Proper storage of all ladders.

   NOTE It is not the intent of this procedure that ladders be returned to ladder storage locations at the end of each shift. Ladders may be stored at the work location, however, they should be stored in such a manner as to not affect the operation of safety-related or safety-significant equipment and have a responsibility tag affixed.

2. Removing all portable tools, equipment and tool boxes from the worksite if they are in a seismic area. A control table may be set-up in non-seismic areas for storage of tools, material and equipment. For jobs in progress in radiologically contaminated areas, the area should be picked up but the removal of tools and equipment is not required as decontamination would not be practical.

3. Picking up and properly disposing of all dust and debris from the area - including floors and horizontal surfaces. This includes removal of items such as paper, cans, tape or pens.

4. Removing or properly identifying extension cords and temporary routings, with a Responsibility Identification Tag.

5. Unplugging all tools when not in use.

6. In cases where the potential exists for misplacement or inadvertent disposal of parts; bagging, boxing, constraining, and/or labeling is advised.

7. Removing all clothing, safety gear, tools or parts from plant equipment.

Attachment 2 (contd)

Housekeeping Standards
Part 3 - Guidance for Use of Responsibility Identification Tags

NOTE This guidance does not apply to items that fall under the scope of procedure IP-DES-3 or to items that are installed and removed under guidance of procedures or written instructions.

Responsibility Identification Tags should be installed in the following locations:

1. Temporary routings of hoses, cables or electrical extension cords that will remain deployed after the completion of the work shift. The tag should be installed on the hose, cable or cord near the source of energization or fluid.

2. Sample locations in which catch containers, funnels and/or hoses are used to collect or direct sample and flushing liquids. The tag should be located on or near the source of the sample fluid or on the catch container.

Responsibility Identification Tags may be installed on other items excluded from the scope of IP-DES-3.

The following information should be entered on the Responsibility Identification Tag: (Attachment 3 of this procedure displays an example)

- Clearly state the reason for use of the extension cord, cable, hose or sample, catch container.
- Installation date
- Responsible contact person, phone number extension, responsible group such as Radiation Protection, Mechanical Maintenance, I&C, etc.

Attachment 2

Housekeeping Standards
Part 4 - Leakage Monitoring

Leakage on plant components (except those in which leakage is designed and properly routed) should be promptly addressed and corrected.

A Leakage Monitoring process has been established to ensure that leakage on plant components is monitored and the components as well as surrounding areas are kept clean, until maintenance can be performed.

The following actions are associated with the leakage monitoring process:

1. Any observer noting leakage on a plant component initiates a Work Request/Trouble Report in accordance with procedure A-1603.1.

2. The Lead Planner decides if leakage monitoring is warranted during a walkdown and review of the Work Order during Planning as per procedure A-1603.3.

   <u>NOTE</u> There may be some leakage conditions in which the Planner (with the concurrence of the System Engineer) may prescribe that the leakage be monitored and cleaned without pursuing corrective action. This approach may be utilized when the leakage condition is not indicative of an equipment Operability or reliability concern, some examples include:

   - Slight dry boric acid buildup in valve packing areas.
   - Slight seepage on pump seals.

3. The Lead Planner contacts the System Engineer for the following:

   - Notification of leakage location and description
   - Notification that leakage will be monitored
   - Concurrence whether to pursue corrective maintenance or not

   <u>NOTE</u> Leakage Monitoring Tags are not to be utilized in Containment or on non-plant system related equipment.

4. The Lead Planner enters the following information on the Leakage Monitoring Tag (see attachment 4 for an example of a typical tag)

   - Component/EIN
   - System Number
   - Plant Location
   - Leakage Description
   - Work Order Number
   - Responsible Group name that will be performing the inspections/cleaning
   - Frequency the leakage location should be inspected/cleaned (i.e. daily, weekly, monthly etc.)

5. The Lead Planner contacts the Responsible group that will perform the monitoring and cleanup (i.e. Radiation Protection for radiation controlled areas and Maintenance for other plant locations), and provides the following information:

   - Leakage location and description
   - Frequency the leakage location should be inspected/cleaned (i e. daily, weekly, monthly

etc.)

6. The Lead Planner performs the following if corrective maintenance will be pursued in conjunction with leakage monitoring:

   - Ensures both Leakage Monitoring and Maintenance ID tags are hung at the leakage location and referenced in the Work Order.
   - Plans the Work Order in accordance with procedure A-1603.3 (Work Order Planning).
   - Enter in the Problem/Work Description field of the Work Order the following information in addition to the Problem/Work Description:

     - Leakage monitoring is being performed
     - Responsible Group for the Leakage Monitoring/Cleaning
     - Plant location

7. The Lead Planner performs the following if corrective maintenance will not be pursued in conjunction with leakage monitoring:

   - Ensures Leakage Monitoring tag is hung at leakage location and referenced in the Work Order.
   - Ensures Maintenance ID tag is removed from leakage location and Work Order references.
   - Enters the following Work Order information:

     - Status "C1"
     - Work Type "L"
     - Sent to applicable Planning Group, i.e "Pipe Planning"
     - Enter in the Problem/Work Description field of the Work Order the following information in addition to the Problem/Work

     Description·

     - Leakage monitoring is being performed. Responsible Group for the Leakage Monitoring/Cleaning
     - Plant location

   - The applicable Planning Group keeps the work order until further action (or cancellation) is warranted.

     <u>NOTE</u> The inspection frequency noted on the tag is a recommendation only and strict adherence is not required, (i.e. daily inspections are not required on weekends or holidays). If there is a housekeeping concern or the inspection frequency is noted by any observer to not have been performed and within the following tolerances, the observer should contact the Responsible Group or submit an ACTION Report:

     - Daily inspection not performed within a week
     - Weekly inspection not performed within 2 weeks
     - Monthly inspection not performed within 2 months
     - Other frequency specified not performed within twice that

frequency.

8. The Responsible group is responsible for the following:

   NOTE The Radiation Protection Group is responsible to ensure leakage containers/absorbent pads do not present concerns with contamination, radioactive waste and/or potential mixing of hazardous waste (such as chromates) in Controlled areas.

   - Installation of any required leakage containers or absorbent pads.
   - Performance of the inspection and noting the date on the Leakage Monitoring Tag and whether (Y/N) cleaning was performed.

9. The Responsible (monitoring/cleaning) group, System Engineer, or any other observer should notify the Maintenance Planning Group (with the Work Order or Leakage Monitoring Tag number, if possible) if there are concerns associated with the leakage, such as:

   - Leakage is increasing or not evident (i.e. after 2 periodic inspections)
   - Change to inspection frequency is requested
   - Request for a new Leakage Monitoring tag (i.e. if the tag is worn, damaged or inspection spaces on the back of the tag are all filled in.
   - ALARA or operational concerns associated with extended monitoring and cleaning of the leaking component.

10. The number of Work Orders associated with Leakage Monitoring Tags will be identified on the Maintenance Monthly Performance Indicator Report. The two categories listed will be:

    - Leakage Monitoring until maintenance
    - Leakage Monitoring only

11. The Lead Planner will review the Leakage Monitoring associated work orders each month and perform the following:

    - Provide a list of open Work Orders with Leakage Monitoring tags to each Responsible (monitoring/cleaning) Group.

    - Review open Leakage Monitoring only Work Orders and request concurrence from the System Engineer and Maintenance Management if the leakage monitoring is expected last more than 18 months. The Planner should document this concurrence in the "Action Taken" section of these work type "L" work orders.

Attachment 3

Responsibility Identification Tag
(typical tag content)

RESPONSIBILITY IDENTIFICATION TAG
(refer to IP-HSC-3)

Purpose: ______________________________

____

____

____

____

Installed Date: ______________________________

Responsible Person: ______________________ Ext ________

Responsible Group: ______________________________

Leakage Monitoring Tag
(typical tag content)

Front

Leakage Monitoring Tag
(refer to IP-HSC-3)
Tag Number XXXXX

Component/EIN ________________

System Number ________________

Plant Location ________________

Leakage Description ________________

____

Work Order Number ________________

Responsible Group·
(circle or fill-in) Radiation Protection
Mechanical Maintenance
Other ____________

Inspection Frequency:
(circle or fill-in) Daily
Weekly
Monthly
Other ____________

Back

Leakage Monitoring Tag
(refer to IP-HSC-3)
Tag Number XXXXX

Date/Cleaned	Date/Cleaned	Date/Cleaned
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N
_____/Y/N	_____/Y/N	_____/Y/N

Rochester Gas and Electric Corporation

Inter-Office Correspondence

Date: April 5, 2002

To: Tom Marlow, Department Manager, Nuclear Engineering Services
Mark Flaherty, Manager Nuclear Safety and Licensing

Subject: Self-Assessment #2002-0037, Ginna Station, March 21-28, 2002
Effectiveness of the Ginna Station Program for Prevention of Boric Acid Corrosion

The subject Self-Assessment was performed March 21-28, 2002, by the following Assessment Team:

Jack St.Martin, Nuclear Safety and Licensing, Team Leader
Ralph Davis, Nuclear Assurance
Steve Carter, Nuclear Training
Bruce Goranowski, Nuclear Assurance

This assessment was conducted at your request and consisted of a review of Ginna Station's current conformance to the requirements of NRC Generic Letter (GL) 88-05. This assessment include reviews of Nuclear Operations Group and Ginna Station procedures, review of NRC correspondence and commitments, interviews with stakeholders, and review of documents and records, to ensure continued conformance to the requirements of NRC GL 88-05. The attached Assessment Report documents the conclusions and recommendations of the Assessment Team.

Please extend our appreciation for the support and cooperation provided to the Assessment Team by personnel contacted during the performance of this assessment, and for those groups who supported the Team needs for records and other documents on such short notice.

John T. St.Martin

John T. St.Martin
Assessment Team Leader

xc: Richard Marchionda, Department Manager, Nuclear Assessment
Robert Mecredy, Vice President Nuclear Operations
Self-Assessment #2002-0037 File

SELF-ASSESSMENT #2002-0037
GINNA STATION
March 21-28, 2002

EFFECTIVENESS OF THE GINNA STATION PROGRAM FOR PREVENTION OF BORIC ACID CORROSION

I. INTRODUCTION

The assessment consisted of reviews of Nuclear Operations Group and Ginna Station procedures, review of NRC correspondence and commitments, interviews with stakeholders, and review of documents and records, to ensure continued conformance to the requirements of NRC Generic Letter (GL) 88-05. The purpose of this assessment was to support the NOG activities needed to respond to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity".

The Assessment Team:

Jack St.Martin, Nuclear Safety and Licensing, Team Leader
Ralph Davis, Nuclear Assurance
Steve Carter, Nuclear Training
Bruce Goranowski, Nuclear Assurance

II. SCOPE, OBJECTIVES, CRITERIA

NRC Bulletin 2002-001 states:

> Within 60 days of the date of this bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:
>
> > the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

Therefore, the scope of this self-assessment was to evaluate the basis for concluding that the Ginna Station Program for Prevention of Boric Acid Corrosion (or simply, the Boric Acid Program) is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter (GL) 88-05 and Bulletin 2002-01. Additional scope, outside of regulatory requirements, included the adequacy of the Boric Acid Program for other areas potentially susceptible to boric acid corrosion and not within the RCS pressure boundary. This narrowly-focused self-assessment was driven by the Process Owner in response to an emergent industry issue as documented in NRC Bulletin 2002-01.

This assessment did NOT evaluate issues related to the reactor pressure head itself. Vessel head issues are being separately addressed to support submittal of the Bulletin 2002-01 15 day response, which includes a summary of the reactor pressure vessel head inspection and maintenance programs, an evaluation of the ability of these 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, and the conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met.

The desired objective is to conclude that the Ginna Station Program for Prevention of Boric Acid Corrosion is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and Bulletin 2002-01. If found that a documented basis does not exist as a result of this self-assessment, then recommendations would be provided to provide our plans to the NRC, if any, for a review of the Boric Acid Program.

Therefore, the scope of this self-assessment is divided into two parts: (1) review of GL 88-05 requirements and the current Boric Acid Program, and (2) review specific program attributes as identified by the Process Owner. With respect to part (1), GL 88-05 states, in part:

> In light of the above experience, the NRC believes that boric acid leakage potentially affecting the integrity of the reactor coolant pressure boundary should be procedurally controlled to ensure continued compliance with the licensing basis. We therefore request that you provide assurances that a program has been implemented consisting of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program should include the following:
>
> (1) A determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion. Particular consideration should be given to identifying those locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces.
>
> (2) Procedures for locating small coolant leaks (i.e.. leakage rates at less than technical specification limits). It is important to establish the potential path of the leaking coolant and the reactor pressure boundary components it is likely to contact. This information is important in determining the interaction between the leaking coolant and reactor coolant pressure boundary materials.
>
> (3) Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.

(4) Corrective actions to prevent recurrences of this type of corrosion. This should include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at the locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings/claddings.

These four program requirements were all reviewed. For part (2) of this assessment, additional criteria were provided by the Process Owner (which were also reviewed, note that these criteria do NOT include issues related to the reactor pressure vessel head):

(1) Review all exemptions submitted to the NRC related to the Boric Acid Program

(2) Evaluate the current Boric Acid Program and compare it to the audit of the program conducted by the NRC in October 1989

(3) Solicit insights from the License Renewal Program, and recommend enhancements to the Boric Acid program, as appropriate, to support License Renewal

(4) Perform an Effectiveness Review of related NRC generic communications, as listed in NRC Bulletin 2002-01

(5) Review aspects of the Ginna Station operating history for past "opportunities" to identify, assess, and correct occurrences of boric acid leaks that could lead to corrosion (including ACTION Reports, Work Requests, Trouble Reports, LIS documents, pressure tests including PT-7)

(6) Status and effectiveness of training on this issue

(7) Evaluate the quality and content of training, especially for training provided to Operations, Maintenance, and System Engineers

(8) Outside of regulatory requirements, assess the adequacy of the Boric Acid Program for other areas potentially susceptible to boric acid corrosion and not within the RCS pressure boundary, specifically Class 2 piping such as RHR

(9) Consider whether the Boric Acid Program, currently an A-14XX series plant procedure, should be a NOG Interface Procedure

III. ASSESSMENT SUMMARY

The program established to comply with NRC Generic Letter (GL) 88-05 exists. This program meets all four requirements as stated in GL 88-05 (also listed in Section II of this report). Thus, there are no deficiencies associated with establishment of the Boric Acid Program (Part 1 of the scope of this self-assessment). With respect to Part 2 of the scope, there is one deficiency identified. This deficiency is listed in Section III.B below. However, given the insights from recent industry events and NRC generic communications subsequent to GL 88-05, there are also several recommendations and concerns that were identified during the course of this self-assessment. These recommendations and concerns are listed in Section III.C below.

Three ACTION Reports were initiated as a result of this self-assessment as listed below:

1. ACTION Report 2002-0713, "Documentation of Leaks in Containment (Initial Inspection)"
2. ACTION Report 2002-0819, "Proposed Enhancements to PT-7 Inspections"
3. ACTION Report 2002-0834, "Proposed Improvements to the Ginna Station Program for Prevention of Boric Acid Corrosion - A-1407".

There are also "Strengths" listed in Section III.A below.

An overall characterization can be made that the Boric Acid Program has stagnated at 1988-era industry standards and should be enhanced to more closely match 2002-era industry standards and elevated management expectations. It should be noted that the System Engineer for the Reactor Coolant System and CVCS was unavailable during the initial week of this self-assessment. However, CATS Items-# E07627, 07628, and 07629 were issued a few years ago to address INPO SEN 190. These internal reviews were performed by the RCS / CVCS System Engineer (and by Operations, and Maintenance). These CATS items are summarized as Attachment II of this report.

The Technical Performance and Field Inspection (TPFI) group plays a central role in implementation of the Boric Acid Program. In procedures and historical documents, the forerunner groups to TPFI included LIS and MEIS. Therefore, the acronyms TPFI, LIS and MEIS are used interchangeably in this report.

A. STRENGTHS

1. The threshold for implementing S-12.2, OPERATOR ACTION IN THE EVENT OF INDICATION OF SIGNIFICANT INCREASE IN LEAKAGE, is at a very conservative level. Since 1996 the procedure has been performed 67 times, most of which determined there was no increase RCS leakage. (Refer to Attachment V for supporting documentation.)

2. Operations Training has had two sessions of industry events training on the subject of

leak detection and boric acid corrosion, one in 1999 and one in 2001. (Refer to Attachment V for supporting documentation.)

3. The RCS Hydro (PT-7) is performed in a thorough and conservative manner at the end of each refueling outage. (Refer to Attachment VI for supporting documentation.)

4. Buildup of boric acid is minimized on all components, not just those containing carbon steel. ACTION Reports are generated to document this buildup. Recent examples from the 2002 outage include AR 2002-0698, 2002-0699, 2002-0703, 2002-0717, and 2002-0720. (Refer to Attachment VI for supporting documentation.)

5. Personnel who perform PT-7 and other NDE examinations are qualified and experienced at Ginna Station.

6. The list of pressure-retaining components that contain carbon steel has been accurately maintained as Attachment I and Attachment II to A-1407. Comparison of A-1407 to DA-ME-2001-038, "Mechanical-Systems-Material V & V for PSSL: 02," demonstrated that the list in A-1407 is complete.

B. DEFICIENCY

1. There are no deficiencies associated with establishment of the Boric Acid Program (Part 1 of the scope of this self-assessment)

2. With respect to Part 2 of the scope of Self-Assessment 2002-0037, there is one deficiency. An inspection of the containment including looking for and addressing Boric Acid deposits is included as part of the process of preparing the containment for entry at the start of an outage. The location of Boric Acid deposits also identifies areas needing decontamination. Current policy (for the 2002 Refueling outage) was to use RP technicians and deconners for this activity. These personnel are neither trained in A-1407 nor complied with Step 3.5.4 of A-1407. Initial inspection of Containment at the start of the 2002 Refueling outage may have included removal of boric acid from selected components. Supporting information for this deficiency is provided as part of Finding #3 in Attachment VIII. This was addressed by ACTION Report 2002-0713.

C. OTHER RECOMMENDATIONS/CONCERNS

There are five areas associated with the Boric Acid Program that could be improved. These areas are:

(1) Awareness of and Attention to the Boric Acid Program and Supporting Procedures and Processes

(2) PT-7 Enhancements

(3) Opportunities for Enhanced Inspections

(4) Training

(5) Use of A-1407

These areas are being tracked by ACTION Report 2002-0819 (Area 2, items #1 and #2 listed below) and ACTION Report 2002-0834 (all remaining items listed below).

AREA 1: AWARENESS OF AND ATTENTION TO THE BORIC ACID PROGRAM AND SUPPORTING PROCEDURES AND PROCESSES

There are two Strengths in this area. Refer to Strength #1 and #4 above.

1. The Boric Acid Program and its governing document (A-1407) have suffered from a lack of focus over time. Refer to Attachment VIII, Finding #1 for supporting information. Suggested recommendation:

   A requirement for having a Boric Acid Program as required by NRC GL 88-05 should be included in a suitable Nuclear Directive. It is recommended that A-1407 be converted to an IP Level Document and updated. Other procedures and methods used to implement the program should be evaluated and updated accordingly.

2. Procedure IP-CAP-1, which is used to report and disposition degradation of components caused by reactor coolant system leakage, could be enhanced to ensure the appropriate root cause determination is performed in accordance with NRC Bulletin 2002-01. Refer to Attachment VIII, Finding #2 for supporting information. Suggested recommendation:

   IP-CAP-1 should be revised to include provisions which implement Bulletin 2002-01 for root cause determination. Additional details are suggested in Attachment VIII, Recommendation #2.

3. Several initiatives such as a leakage monitoring program outside containment (IP-HSC-3) and reducing the priority of packing leakage ACTION Reports may detract from the effectiveness of the GL 88-05 intent. Refer to Attachment VIII, Finding #5 for supporting information. Suggested recommendation:

   It is recommended that the listed policies be reviewed with respect to NRC GL 88-05 and credit taken for the "zero tolerance" good practice with respect to cleanup of Boric Acid when found.

4. There are more recent issued technical documents to consider for inclusion in A-1407. Refer to Attachment IV, Data Sheet for NP-5985, for supporting information. Suggested recommendation:

   A process should be established for ensuring that the engineer in charge of the Boric Acid Program is made aware of significant new technical documents relevant to the enhancement of the Boric Acid Program. The overall procedural controls for the Boric Acid Program (currently located in A-1407), should be periodically updated to include more recent industry technical guidance. Additional details are suggested in Attachment IV.

5. While IP-HSC-3 is not used for leakage inside containment, this should be clearly stated within the procedure. In addition, if IP-HSC-3 is to be used to implement the recommendations from the License Renewal Program (see Attachment IX), it should be upgraded to include the requirements of the Boric Acid Program, as initially discussed in GL 88-05 and implemented via A-1407. Refer to Attachment IV, Data Sheet for IP-HSC-3, for supporting information.

6. Consider whether to continue limiting IP-HSC-3, Attachment 2, to only areas outside of Containment. If so, consider the need for a new program to track and monitor boric acid leaks inside Containment. Refer to Attachment IV, Data Sheet for IP-HSC-3, for supporting information.

7. If A-1407 is upgraded to a NOG Interface Procedure, consider including NRC Bulletin 2002-01 as both a reference and as a source of information. Include "Lessons Learned" from other utilities. especially Davis-Besse. This includes consideration of periodically checking of Containment Ventilation system filters for boric acid and iron oxide residues.

8. Ensure there is adequate focus on the company-wide implications of the Boric Acid Program, versus only for Ginna Station employees (see current steps 3.5.2 and 3.5.3 of A-1407):

   3.5.2 Ginna Station plant personnel are responsible for investigating the leak source and leak-path.

   3.5.3 Once the leakage source and leak-path have been identified, plant personnel shall determine if RCPB carbon steel components may have been in contact with boric acid.)

9. Ensure A-1603.3, Step 3.1.12 maintains compliance by Maintenance Planning with the requirements of A-1407, Step 3.5.5 and GL 88-05, (3):

   3.5.5 The disposition of the Trouble Card will ensure that RCPB carbon steel components affected shall be inspected by assigned personnel, and evaluated for any evidence of possible component degradation.

GL-88-05, (3) Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.

10. Address the impact of boric acid corrosion of materials in future plant changes. Include an check for this impact in EP-3-S-0306, "Change Impact Evaluation Form".

AREA 2: PT-7 ENHANCEMENTS

There are two Strengths in this area. Refer to Strengths #3 and #5 above.

1. The documentation of and directions to personnel performing PT-7 procedure and leakage examinations of the Reactor Coolant System and adjoining systems are in need of improvement. Refer to Attachment VI and Attachment VIII, Finding #4 for supporting information.

   It is recommended that PT-7 procedure, documentation methods and conduct process be examined and strengthened prior to the PT-7 test for the 2002 outage. Therefore, conduct a meeting prior to performing the PT-7 system leakage test of the reactor coolant system.

2. A review of the impact of prior leakage around and above the pressurizer should be considered for impact on carbon steel shell since this component is at the highest temperature within the RCS. This recommendation does not propose to remove pressurizer insulation; instead it is recommended that the insulation be reviewed for telltale signs of leakage that warrant further investigation. Refer to Attachment VIII, Finding #8 for supporting information.

3. Prepare standard references for the recurring need to inspect bolted connections and to remove insulation from selected areas. Refer to Attachment VI for supporting information.

4. Prepare a standard procedure for performance of the "10 year ISI", for which additional inspections (beyond the normal PT-7) must be performed. Refer to Attachment VI for supporting information.

AREA 3: OPPORTUNITIES FOR ENHANCED INSPECTIONS

1. Refueling procedures should be strengthened to include more formal checks for and documentation of evidence of Boric Acid deposits during disassembly operations similar to those required for conoseals.

2. The issue of Inconel 600 in areas not on the reactor vessel head should continue to be monitored and evaluated by NES. CATS ID # 10369 and NRC Information Notice 2000-17 are related to this issue. Refer to Attachment IV, Data Sheet for NSAL 95-010 for supporting information.

3. Ensure the "Lessons Learned" from Aging Mechanism Reviews (AMR) of components are reviewed for possible enhancement of the Boric Acid Program. This will ultimately involve expansion of the program before the end of the current Operating License, and would then include ALL systems which contain boric acid (not just RCS), ALL materials in the vicinity of these systems that could be susceptible to corrosion (not just carbon steel), and components and structures that could be affected (not just pressure-retaining components). Refer to Attachment IX for supporting information. It is also recommended that these enhancements be implemented as soon as resources become available versus waiting until September 2009.

4. System Engineers for RCS (02), RHR (03), SI (05), CVCS (07), and Steam Generators should have guidelines to document compliance (and ensure that adequate documentation of that compliance is produced) with all applicable requirements of NRC GL 88-05, GL 97-01, Bulletin 2001-01, and Bulletin 2002-02, and with A-1407.

5. When insulation is removed from RCS components during the performance of maintenance and other inspection activities during an outage, evaluate the benefits of documenting inspections of the condition of these components at that time.

AREA 4: TRAINING

There is one Strength in this area. Refer to Strength #2 above.

1. Not all personnel responsible for locating leaks in the Reactor Coolant Pressure Boundary have had training in the use of A-1407. Refer to Attachment V for supporting information.

2. All personnel who have access to areas containing piping with borated water should be made aware of the requirements of A-1407.

3. Evaluate the need for refresher/continuing training on A-1407 in all applicable programs. Refer to Attachment V for supporting information.

4. If RP Techs are going to perform the initial containment entry tour, they should be trained in the use of A-1407. Refer to Attachment V for supporting information.

AREA 5: USE OF A-1407

There is one Strength in this area. Refer to Strength #6 above.

1. A-1407 is not consistently used as guidance for finding leaks in the reactor coolant pressure boundary per S-12.2. Refer to Attachment V for supporting information. Suggested recommendation:

    Reinforce management expectations to ensure A-1407 is used as the guiding document for investigating reactor coolant pressure boundary leakage inside containment. Upgrade S-12.2 if necessary. Refer to Attachment V for supporting information.

2. Re-evaluate the use of RP technicians and/or deconners for cleanup of boric acid deposits during an initial containment entry at the start of an outage. If they continue to be used, ensure compliance with A-1407, Step 3.5.4.

3. Provide guidance to deconners and housekeeping personnel that ensures compliance with A-1407 when removing boric acid deposits.

IV. CONCLUSION

The overall conclusion of the Assessment Team is that a documented basis exists to conclude that the Boric Acid Program is providing reasonable assurance of compliance with the applicable requirements discussed in NRC Generic Letter 88-05 and NRC Bulletin 2002-01. Correction of the deficiency identified during this Self-Assessment will bring Ginna Station into full compliance and help in the License Renewal Program.

V. ATTACHMENTS

(1) Attachment I provides a list of generic communications that were reviewed

(2) Attachment II summarizes an earlier RG&E review of INPO SEN 190

(3) Attachment III summarizes the status of management criteria for this self-assessment

(4) Attachment IV contains 21 data sheets providing details of Effectiveness Reviews

(5) Attachment V contains 2 data sheets providing details of Training issues

(6) Attachment VI is an assessment of PT-7

(7) Attachment VII provides a table to demonstrate compliance between NRC GL 88-05 requirements and implementing procedures or processes

(8) Attachment VIII provides supporting information for several Findings and associated Recommendations

(9) Attachment IX describes the probable License Renewal Program commitment for the Boric Acid Corrosion Program

ATTACHMENT I

List of Related Generic Communications (as listed in NRC Bulletin 2002-01)

This section of Self-Assessment #2002-0037 lists related NRC generic communications as listed in NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity. Effectiveness Reviews were performed for these communications. Refer to associated Self-Assessment Data Sheets in Attachment IV of this report for the results of the individual reviews.

Related Generic Communications (as listed in NRC Bulletin 2002-01):

1. NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982

2. NRC Bulletin 2001-01: "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001. [ADAMS Accession No. ML012080284]

3. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988

4. Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997

5. Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," June 11, 1980

6. Information Notice 82-06, "Failure of Steam Generator Primary Side Manway Closure Studs," March 12, 1982

7. Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986 and Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," April 20, 1987 and Information Notice 86-108, Supplement 2, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 19, 1987

8. Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," January 5, 1995

9. Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990

10. Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by

Cladding Cracks," August 30, 1994

11. Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996

12. Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001. [ADAMS Accession No. ML011160588]

13. Information Notice 2002-11: "Recent Experience with Degradation of Reactor Pressure Vessel Head," March 12, 2002. [ADAMS Accession No. ML020700556]

14. NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994

Other generic or plant-specific communications identified during Self-Assessment #2002-0037 were also reviewed:

15. NRC Audit of the Ginna Station Program for the Prevention of Boric Acid Corrosion at Ginna Nuclear Power Plant (documented in a letter from NRC to RG&E, dated August 20, 1990).

16. NRC Generic Letter 91-017, "Generic Safety Issue (GSI) 29, Bolting Degradation or Failure in Nuclear Power Plants", October 17, 1991

17. INPO SOER 84-5, Bolt Degradation or Failure in Nuclear Power Plants

18. EPRI Report NP-5985

19. Westinghouse NSAL 95-010, Pressurizer Upper Level Instrument Tap Nozzle Cracks, November 30, 1995

20. INPO SER 93-20 and Westinghouse "Info Gram 93-009, "Update on Ringhals Unit 2 Vessel Head Penetration Attachment Weld Indication, November 15, 1993

21. IP-HSC-3, Housekeeping Control

Attachment II

INPO SEN (Significant Event Notification) 190, "Pressurizer Spray Valve Nuts Dissolved by Boric Acid"

Executive Summary:

On September 18, 1998, with Davis-Besse at 100 percent power, engineering personnel determined that a pressurizer spray valve was not capable of maintaining the reactor coolant system pressure boundary under all accident conditions because two body-to-bonnet nuts were severely damaged by boric acid corrosion. The pressurizer spray valve had been leaking at the outer diameter of the packing gland since startup following the May 1998 refueling outage. One nut was completely dissolved, and a second nut was more than 90 percent dissolved. After discovering the missing nuts, station personnel concluded that three of the eight required stainless steel nuts had been inadvertently replaced with carbon steel nuts some time in the past.

CATS ID # E07627, E07628, and E07629 were assigned to, respectively, the System Engineer, Operations, and Mechanical Maintenance, for each of them to review and evaluate SEN 190 regarding pressurizer spray valve bonnet nuts dissolved by boric acid leaks, to ensure programs, procedures, and training are in place to prevent this event from occurring at Ginna. Here are the results of those reviews, as documented in the CATS files:

Responses Provided by the System Engineer:

1. A. How do we verify materials exposed to leakage from borated systems are not susceptible to boric acid corrosion?

   RESPONSE: The carbon steel components of all the RCS pressure boundary components are identified in A-1407 Attachment I and II.

   B. Under what circumstances would we field-verify materials when plant prints indicate all stainless steel construction?

   RESPONSE: Procedure A-1407 steps 3.5.1 and 3.5.7 detail the initial investigation and evaluation of a boric acid leak by maintenance, systems engineering and LIS personnel. This evaluation would include verification of the materials exposed to the boric acid leak.

2. A. Who is notified of reactor coolant system leaks?

RESPONSE: During the reactor plant startup, procedure PT-7 is used to initially assess the condition of the RCS boundary at three different pressure plateaus. If a leak were to occur during startup then PT-7 is used to identify and resolve the leak. Maintenance, systems engineering and LIS personnel are notified.

Operations procedure S-12.4 is used to monitor the RCS pressure boundary prior to the reactor coolant temperature increasing above 200 degrees F and during normal at power RCS operation. If a leak is noted, procedure S-12.2 is used in an attempt to identify the leak using all available means outside the containment. If the location of the leak still cannot be identified, then procedure A-1407 is referenced and a containment entry is made. Maintenance, systems engineering and LIS personnel are notified.

B. How often do we initially inspect a leaking component?

RESPONSE: During the reactor plant startup, procedure PT-7 is used to initially assess the condition of the RCS boundary at three different pressure plateaus. If a leak were to occur during startup then PT-7 is used to identify and resolve the leak. Maintenance, systems engineering and LIS personnel are notified. --

C. If boric acid is present, how is the inspection frequency affected?

RESPONSE: Per procedure A-1407, step 3.5.6, if any degradation is noted an ACTION Report is to be written, then LIS personnel will evaluate the situation and report to the systems engineering group. Then per step 3.5.7 the primary systems engineering group will evaluate different inspection techniques. frequencies. consequences. and possible methods of repair.

D. Based on observed conditions, how do we decide to change the inspection frequency?

RESPONSE: Per procedure A-1407, step 3.5.7, the primary systems engineering group will evaluate different inspection techniques, frequencies, consequences, and possible methods of repair.

E. What approvals are necessary when an inspection interval is to be changed?

RESPONSE: Per procedure A-1407, step 3.5.7, the primary systems engineering group will evaluate different inspection techniques, frequencies, consequences, and possible methods of repair.

3. What techniques do we use to maintain plant cleanliness when removing boric acid residue to permit inspection?

RESPONSE: The area is taped and bags are present for collection of the boric acid residue. The residue is removed by several methods from vacuuming to lightly using a small wire brush and collecting the boric acid residue for disposal. This item is a skill-of-the-trade type of issue.

Responses Provided by Operations Supervision:

4. When reactor coolant leaks are identified, what methods are used to promptly identify changes in leak rate? How would the presence of a wet borated water leak affect inspection frequency and repair priority?

5. When containment inspections identify a reactor coolant system valve body-to-bonnet or packing leak, what means are used to verify that bonnet material and bolting is not susceptible to boric acid corrosion? What inspection criteria do we use to identify all equipment damage resulting from borated water leaks? What techniques do we use to maintain plant cleanliness when removing boric acid residue to permit inspection?

RESPONSE: Ginna Station has procedures and programs in place to ensure this does not happen at Ginna. Procedure S-12.2, Operator Action in the Event of Indication of Significant Increase in Leakage", directs you to A-1407 "Program to Prevent Degradation of Reactor Coolant Pressure Boundary Components from Boric Acid Corrosion".

Responses Provided by Mechanical Maintenance:

6. How do you know when stainless steel parts are required? How do you verify that stainless steel nuts and bolts are actually stainless steel prior to installing them?

   RESPONSE: Parts used in maintenance such as bolts and nuts, material traceability is maintained through administrative controls (procedure A-801) that requires a documented traceability (a green tag containing Material ID and serial numbered material requisition) from receipt from the stockroom through installation. Stainless steel nuts and bolts are verified by visual appearance and by the documentation.

7. What is our policy for reusing equipment such as nuts and bolts? When reusing materials, how do we know that the materials being reused were not inadvertently mixed with improper materials?

   RESPONSE: Our policy is to procedurally require in the Limits and Precautions section of procedures, to bag and tag parts during disassembly. This prevents inadvertently mixing material. (Also included in the CATS file are documentation excerpts providing examples of material traceability from purchase order to installation of stainless/steel bolts and nuts on the Ginna Station pressurizer spray valves 431A and 431B.)

ATTACHMENT III
SUMMARY OF MANAGEMENT CRITERIA, PART 2 OF THE SCOPE

(1) Review all exemptions submitted to the NRC related to the Boric Acid Program

RESPONSE: In the context of NRC Generic Letter (GL) 88-05 and the Boric Acid Program, any exemptions to NRC requirements would have initially been requested in RG&E's response to GL 88-05. No exemptions were requested in that response (dated May 31, 1988). In subsequent communications from the NRC to RG&E (letters dated January 30, 1990 and August 7, 1990), NRC concurred that the RG&E program fully complies with GL 88-05. It should be noted that GL 88-05 only addresses boric acid leakage potentially affecting the integrity of the reactor coolant pressure boundary, not all carbon steel components. As documented within Attachments VII and VIII, a detailed comparison of GL 88-05 requirements and RG&E's conformance was performed. No RCS pressure containing carbon steel components were found to be missing from the program. This was also confirmed by the License Renewal Program (see (3) below).

(2) Evaluate the current Boric Acid Program and compare it to the audit of the program conducted by the NRC in October 1989

RESPONSE: An Effectiveness Review of the NRC audit was performed as documented in Attachment IV. The conclusions of the NRC after performing this audit were that the program meets the intent of Generic Letter 88-05, that the Maintenance Program could be improved if a priority system for repair is incorporated, and that the use of a mock-up for leaks used by the materials engineering group should be used for the site people (Aux operators, SROs/ROs) to familiarize themselves with what to look for in the field. CATS ID # R00598 and R00599 were initiated to address the two NRC recommendations, and documentation of the RG&E resolution of these recommendations is in the respective CATS files.

In addition, a review of the GL 88-05 requirements was performed as documented in Attachments VII and VIII. The results of this review concluded that Ginna Station is meeting the requirements of GL 88-05, but there are several enhancements that could be made. ACTION Reports were generated to track these enhancements.

(3) Solicit insights from the License Renewal Program, and recommend enhancements to the Boric Acid Program, as appropriate, to support License Renewal

RESPONSE: A meeting was held with Gerry Geiken from the License Renewal Program (LRP). The LRP is recommending implementation of an enhanced Boric Acid Corrosion Program that will:

a. include all borated water systems at Ginna Station,

b. include examination of all systems that are in proximity to the borated water systems that leak

Attachment IX contains additional details of the proposed program. While this program does not have to be implemented until September 2009, it is being recommended as part of this self-assessment that the program be implemented as soon as resources are made available.

(4) *Perform an Effectiveness Review of related NRC generic communications, as listed in NRC Bulletin 2002-01*

RESPONSE: These reviews are documented in Attachment IV. Several enhancements were identified as summarized in Section III, Assessment Summary. These enhancements are being tracked by ACTION Reports.

(5) *Review aspects of the Ginna Station operating history for past "opportunities" to identify, assess, and correct occurrences of boric acid leaks that could lead to corrosion (including ACTION Reports, Work Requests, Trouble Reports, LIS documents, pressure tests including PT-7)*

RESPONSE: During the course of this self-assessment, Steve Carter reviewed a total of sixty-seven S-12.2 reports, as summarized in Attachment V. Ralph Davis reviewed a significant cross-section of ACTION Reports, Work Requests and Trouble Reports. Bruce Goranowski reviewed the two most recent PT-7 inspections (documented in Attachment VI). The conclusion is the Ginna Station process to identify occurrences or boric acid leaks is a Strength. The processes to assess and correct leaks is adequate but could be enhanced (e.g., documentation and training weaknesses) as stated in Section III, Assessment Summary. These enhancements are being tracked by ACTION Reports.

(6) *Status and effectiveness of training on this issue*

RESPONSE: The results are summarized in Attachment V. Several recommendations and concerns were identified with respect to training of plant personnel as stated in Section III, Assessment Summary which are being tracked by ACTION Reports.

(7) *Evaluate the quality and content of training, especially for training provided to Operations, Maintenance, and System Engineers*

RESPONSE: The results are summarized in Attachment V. Several recommendations and concerns were identified with respect to training of plant personnel as stated in Section III, Assessment Summary which are being tracked by ACTION Reports.

(8) *Outside of regulatory requirements, assess the adequacy of the Boric Acid Program for other areas potentially susceptible to boric acid corrosion and not within the RCS pressure boundary, specifically Class 2 piping such as RHR*

RESPONSE: The Boric Acid Program was established to comply with GL 88-05. Thus, areas that are not within the RCS pressure boundary were not required to be in the scope of the program. This was a deliberate decision made in 1988, and in 2002 it can be stated that the program is inadequate for Class 2 piping and areas outside the RCS pressure boundary. As described in Attachment IX, the License Renewal Program is proposing to expand the scope of the program. This is being tracked both by the License Renewal Program and the recommendations contained in Section III, Assessment Summary.

(9) *Consider whether the Boric Acid Program, currently an A-14XX series plant procedure, should be a NOG Interface Procedure*

RESPONSE: The results of this self-assessment conclude that the Boric Acid Program should not be procedurally treated as a plant administrative document. Therefore, it should be upgraded and converted to a NOG Interface Procedure. This is also addressed by Finding (1) in the Assessment Summary and is being tracked by an ACTION Report.

ATTACHMENT IV

Results of Effectiveness Review

The following related generic communications (and other relevant documents) were reviewed for this Effectiveness Review:

(1) NRC Bulletin 82-02
(2) NRC Bulletin 2001-01
(3) NRC Generic Letter 88-05
(4) NRC Generic Letter 97-01
(5) NRC Information Notice 80-27
(6) NRC Information Notice 82-06
(7) NRC Information Notice 86-108
(8) NRC Information Notice 86-108, Supplement 1
(9) NRC Information Notice 86-108, Supplement 2
(10) NRC Information Notice 86-108, Supplement 3
(11) NRC Information Notice 90-10
(12) NRC Information Notice 94-63
(13) NRC Information Notice 96-11
(14) NRC Information Notice 2001-05
(15) NRC Information Notice 2002-11
(16) NUREG/CR-6245
(17) NRC Audit (letter from NRC to RG&E, dated August 7, 1990)
(18) NRC Generic Letter 91-017
(19) INPO SOER 84-5, in Nuclear Power Plants
(20) EPRI Report NP-5985
(21) Westinghouse NSAL 95-010
(22) INPO SER 93-20
(23) IP-HSC-3

Self-Assessment Data Sheet

Objective: Effectiveness Review of NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982

Brief Summary of Observation

This generic communication summarized the issues previously disseminated by the NRC in NRC Information Notice 80-27 and Information Notice 82-06. This bulletin resulted in two RG&E responses to the NRC, stating that our program is acceptable, based on no known significant degradation at Ginna Station. Thus, it was stated that the Ginna Program is effective in identifying minimal degradation and preventing such degradation from becoming significant.

Supporting Information

Subsequent concerns were also raised by NRC inspectors in NRC Inspection Report (IR) 91-013. RG&E responded to these concerns in a letter dated October 23, 1991.

This issue was later assessed by RG&E (refer to CATS ID# R01642). This issue was adequately addressed in the initial responses to the bulletin, and by subsequent activities. Refer to the Effectiveness Review of NRC Generic Letter 91-017 for additional supporting information.

Recommendation

None

Evaluated By	Jack St.Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of NRC Bulletin 2001-01: "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001. [ADAMS Accession No. ML012080284]

Brief Summary of Observation

This issue is outside the scope of Self-Assessment #2002-0037, and has resulted in several meetings and letters between RG&E and the NRC. This issue would not benefit from an Effectiveness Review at this time, and also is addressed as part of the 15 Day response to NRC Bulletin 2002-01.

Supporting Information

No Effectiveness Review was performed for this generic communication.

Recommendation

Continue to work with NRC and Industry Groups to resolve this issue.

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988

Brief Summary of Observation
NRC assessed the Ginna Station Program for the Prevention of Boric Acid Corrosion at Ginna Nuclear Power Plant in a special NRC audit, conducted in October 1989. The results of this audit are documented in a letter from NRC to RG&E dated August 20, 1990.
Supporting Information
Refer to Attachment III of Self-Assessment #2002-0037 for a detailed Effectiveness Review of this generic communication. Refer to the Effectiveness Review of the "audit report" (letter from NRC to RG&E, dated August 20, 1990) for additional supporting information.
Recommendation
Refer to other sections of Self-Assessment #2002-0037 for relevant recommendations.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997

Brief Summary of Observation

This issue is outside the scope of Self-Assessment #2002-0037, and was initially characterized by the NRC as not an immediate safety concern, thus the issuance of a GL. Additional occurrences of this type of degradation caused the NRC to re-evaluate the safety significance, which resulted in the issuance of NRC Bulletin 2001-01.

Supporting Information

No Effectiveness Review was performed for this generic communication.

Recommendation

Continue to work with NRC and Industry Groups to resolve this issue.

Evaluated By	Jack St Martin	Criterion Met	Yes No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," June 11, 1980

Brief Summary of Observation

This issue was one example of boric acid corrosion, and was discovered in May 1980. Although not confirmed by metallurgical analysis at that time, the cause of the stud wastage was thought to be corrosive attack by hot boric acid from the primary coolant. The condition of the studs raised concerns that such severe corrosion, if undetected, could lead to stud failures which could result in loss of integrity of the reactor coolant pressure boundary. The lack of effectiveness of 1980-era ultrasonic examinations in revealing wastage emphasized the need for supplemental visual inspections and use of instrumented leak detection systems to preclude unacceptable stud degradation going undetected. Licensees were reminded to consider that the potential for undetected wastage of carbon steel bolting by a similar mechanism could exist in other components such as valves.

Supporting Information

This issue was subsequently listed as one of the examples in NRC Bulletin 82-02. It was later listed in NRC GL 88-05, and is adequately addressed by the Effectiveness Review of GL 88-05.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 82-06, "Failure of Steam Generator Primary Side Manway Closure Studs," March 12, 1982

Brief Summary of Observation

This issue was another example of boric acid corrosion, this time on a steam generator primary side manway, where eleven 1 1/2 x 10 inch studs of SA 540 grade B 24 alloy steel had been exposed to boric acid from a small primary coolant leak and to Furmanite sealing compound (primary grade) applied in an attempt to seal this leak. The studs exhibited evidence of surface corrosion attack possibly as result of an interaction associated with stud preload, lubricant, Furmanite and primary coolant leakage environment. The failures described were attributed to stress corrosion cracking and corrosion wastage of high strength studs that are difficult to detect.

Supporting Information

This issue is adequately addressed by the Effectiveness Review of GL 88-05.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986, and Supplements 1 and 2

Brief Summary of Observation

Effectiveness Review of Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," December 29, 1986, and
Information Notice 86-108, Supplement 1. "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," April 20, 1987 and
Information Notice 86-108, Supplement 2. "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 19, 1987

Each of these generic communications discusses one example of boric acid corrosion. These three occurrences are separately listed in GL 88-05, and probably were the "last straw" for the NRC, resulting in the issuance of GL 88-05.

Supporting Information

This issue is adequately addressed by the Effectiveness Review of GL 88-05.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 86-108, Supplement 3, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," January 5, 1995

Brief Summary of Observation

This generic communication discusses the "latest" two incidents (which occurred in 1994) of boric acid-induced corrosion, and indicated that there may still be a lack of awareness of the conditions that can lead to boric acid attack. The NRC reminded licensees of the wide range of ambient conditions around reactor primary coolant leak sites with the resulting wide variation in boric acid corrosion rates, which makes it difficult to predict the likelihood of corrosion damage when a leak is present. This is particularly true of components such as insulated flanges and valve bonnets that are somewhat isolated from the areas of heat input from the reactor coolant and may experience large temperature variations. The NRC reminded licensees that the primary defense against boric acid corrosion remains the same; i.e., minimize leakage, detect and stop leaks soon after they start, and promptly clean up any boric acid residue.

Supporting Information

This was previously assessed by RG&E System Engineering personnel. This assessment is documented in the file for CATS ID# R04739. The information confirms that the actions implemented due to GL 88-05 provide adequate guidelines at Ginna Station. The information was adequately assessed in 1995 and no further assessment or actions are required.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 90-10, "Primary Water Stress Corrosion Cracking of INCONEL 600," February 23, 1990

Brief Summary of Observation

This generic communication restated that intergranular stress corrosion cracking (IGSCC) requires the presence of the following three key elements: an aggressive environment, susceptible material, and sufficient tensile stresses for crack initiation and propagation. Primary water stress corrosion cracking (PWSCC) refers to IGSCC in the primary water environment of PWRs. PWSCC of Inconel 600 is not a new phenomenon (even in 1990). However, very little special attention had been given to the inspection for PWSCC in Inconel 600 applications other than that associated with the steam generator tubing. As a result of the reported instances of PWSCC in the pressurizer heater thermal sleeves and instrument nozzles in several domestic and foreign PWRs, the NRC advised that it may be prudent for licensees of all PWRs to review their Inconel 600 applications in the primary coolant pressure boundary, and when necessary, to implement an augmented inspection program.

Supporting Information

This was previously assessed by RG&E Maintenance Engineering personnel. This assessment is documented in the file for CATS ID# R00340. Subsequent NRC generic communications (GL 97-01) required substantial efforts concerning this issue. The information was effectively assessed in 1990 and re-addressed under GL 97-01, and no further assessment or actions are required.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks," August 30, 1994

Brief Summary of Observation

This generic communication discussed the issue of cracking, which resulted in boric acid attack of the carbon steel base metal. Rust, visible on the surface of the stainless steel cladding of a charging pump, indicated that base metal corrosion attack had occurred. Corrosive attack by boric acid coolant resulting from small cracks in charging pump cladding generally proceeds relatively slowly due, apparently, to the low temperature of the charging coolant. However, such attack can eventually lead to significant thinning of the pump casing and possibly substantial leakage. This experience also shows that the corrosion of the base metal due to cladding cracks is usually relatively easy to identify through visual inspection.

Supporting Information

This was previously assessed by RG&E Mechanical Engineering personnel. This assessment is documented in the file for CATS ID# R04408.

The information was effectively assessed in 1994 and no further assessment or actions are required.

Recommendation

None

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996

Brief Summary of Observation

This issue is outside the scope of Self-Assessment #2002-0037, and was previously assessed as documented in the CATS ID # R05188 file. Subsequent NRC generic communications (GL 97-01) required a specific response to this issue, which was provided in a letter from RG&E to NRC dated July 25, 1997.

Supporting Information

Since the information was re-addressed under GL 97-01, no further assessment or actions are required.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 2001-05, "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," April 30, 2001. [ADAMS Accession No. ML011160588]

Brief Summary of Observation
This issue is outside the scope of Self-Assessment #2002-0037, and was subsequently addressed in detail as part of NRC Bulletin 2002-01.
Supporting Information
No Effectiveness Review was performed for this generic communication.
Recommendation
Continue to work with NRC and Industry Groups to resolve this issue.

Evaluated By	Jack St Martin	**Criterion Met**	Yes No NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes No NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Information Notice 2002-11: "Recent Experience with Degradation of Reactor Pressure Vessel Head," March 12, 2002. [ADAMS Accession No. ML020700556]

Brief Summary of Observation

This issue is outside the scope of Self-Assessment #2002-0037, and will be addressed in detail as part of NRC Bulletin 2002-01.

Supporting Information

No Effectiveness Review was performed for this generic communication.

Recommendation

Continue to work with NRC and Industry Groups to resolve this issue.

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994

Brief Summary of Observation
This NUREG is normally available at Ginna Station in the Engineering Library of NUREGs. However, this particular NUREG had been checked out to Al Butcavage. Both Mr. Butcavage and Mr. G. Geiken have recently reviewed this NUREG.
Supporting Information
No Effectiveness Review was performed for this generic communication.
Recommendation
Continue to utilize this and other NRC technical documents to assist in resolving this issue.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of the NRC Audit of the Ginna Station Program for the Prevention of Boric Acid Corrosion at Ginna Nuclear Power Plant (documented in a letter from NRC to RG&E, dated August 7, 1990)

Brief Summary of Observation

The conclusions of the NRC after performing this audit were:

1. The program for boric acid corrosion prevention at R.E. Ginna meets the intent of Generic Letter 88-05.
2. The Maintenance Program could be improved if a priority system for repair is incorporated.
3. The training modules examined were very good. The use of a mock-up for leaks used by the materials engineering group should be used for the site people (Aux operators, SROs/ROs) to familiarize themselves with what to look for in the field.

CATS ID # R00598 and R00599 were initiated to address the two NRC recommendations, and documentation of the RG&E resolution of these recommendations is in the respective CATS file.

Supporting Information

The NRC conclusions were addressed promptly, and the information was effectively assessed in 1991. No further assessment or actions are required.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of NRC Generic Letter 91-017, "Generic Safety Issue (GSI) 29, Bolting Degradation or Failure in Nuclear Power Plants", October 17, 1991

Brief Summary of Observation

This generic communication was issued by the NRC to provide information on the NRC's resolution of GSI 29. The NRC has resolved this GSI, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area. The NRC gives credit to the same joint task group (AIF/MPC Joint Task Group on Bolting) that was discussed seven years earlier in INPO SOER 84-5. This GL was assessed by RG&E Mechanical Engineering personnel in 1993. This assessment is documented in the file for CATS ID# R01889. The information contained in the CATS file is comprehensive and the delineates that the actions discussed in GL 91-017 continue to be implemented, and are sufficient to assure the integrity of safety-related bolting at Ginna Station.

Supporting Information

This issue did not require any response from utilities, so there is no docketed communications from RG&E concerning GL 91-017. However, the assessment conducted by Mechanical Engineering comprehensively discusses how RG&E conforms to and supports the conclusion of the NRC that this issue is resolved, as stated in GL 91-017.

Recommendation

None

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of INPO SOER 84-5, Bolt Degradation or Failure in Nuclear Power Plants

Brief Summary of Observation

INPO SOER 84-5 was issued by INPO on September 20, 1984, which predated the NRC issuance of GL 91-017 (Generic Safety Issue (GSI) 29, Bolting Degradation or Failure in Nuclear Power Plants). All issues raised in the INPO SOER were adequately addressed in hte next seven years as documented in NRC GL 91-017.

Supporting Information

INPO SOER 84-5 referenced many of the same NRC generic communications that were also referenced in NRC Bulletin 82-02, and also several other industry events that had been previously disseminated to INPO members as INPO SERs. All issues have been resolved to the satisfaction of the NRC, as documented in NRC GL 91-017.

Recommendation

None

Evaluated By	Jack St Martin	Criterion Met	Yes	No	NA
Date	March 27, 2002	Opportunity for Improvement	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of EPRI Report NP-5985

Brief Summary of Observation

EPRI Report NP-5985 is one of the suggested sources of information that could be used to resolve issues of boric acid corrosion, as stated in A-1407. This EPRI report dates back to 1988, and it probably was the most current technical guidance available when A-1407 was first written in 1988. However, there have been several technical documents issued by both the NRC and EPRI in the intervening years.

Supporting Information

There are more recent issued technical documents to consider for inclusion in A-1407. For example, EPRI TR-101108, "Boric Acid Corrosion Evaluation (BACE) Program, Phase I - Task 1 Report", was issued in December 1993. EPRI TR-104748, "Boric Acid Corrosion Guidebook", was issued in April 1995. Both of these technical documents, as well as several others, are in the Ginna Station Engineering Library, and were actually in the possession of Al Butcavage during the conduct of this self-assessment.

Recommendation

1. A process should be followed for ensuring that the engineer in charge of the Boric Acid Program is made aware of significant new technical documents relevant to the enhancement of the Boric Acid Program.

2. The overall procedural controls for the Boric Acid Program (currently located in A-1407), should be periodically updated to include more recent industry technical guidance.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of Westinghouse NSAL 95-010, Pressurizer Upper Level Instrument Tap Nozzle Cracks, November 30, 1995

Brief Summary of Observation

Westinghouse NSAL 95-010 was issued to discuss an event, for the purpose of determining that it is not a 10 CFR 21 issue. However, it is relevant to consider it in light of GL 97-01 and Bulletin 2001-01.

At Surry (Virginia Power), through wall circumferential cracks were discovered in two upper pressurizer instrument level tap nozzles just outboard of the inner structural weld between the stainless steel nozzles and the vessel shell stainless steel cladding. The cracks were 1/2 to 3/4 inches from the inner diameter of nozzle end. Rust stains and boric acid crystals were identified on the two instrument connections during visual inspection of the four upper and five lower instrument connections. No other leakage was found on the remaining instrument tap nozzles.

Westinghouse has determined that the pressurizer upper level instrument tap nozzle cracks will not lead to a catastrophic failure of the instrument tap and that even if such a failure is postulated to occur, the consequences are bounded by previously completed accident analyses for small break LOCA resulting from a failed open PORV which would result in an equivalent break diameter of 2.1 inches. Therefore, this failure mode does not represent a substantial safety hazard nor a failure to comply pursuant to 10 CFR Part 21.

Supporting Information

CATS ID # E05088 was issued by the Ginna Operating Experience Group for this NSAL. The CATS summary stated that cracked pressurizer instrument nozzles were previously found at Calvert Cliffs and San Onofre and two foreign Combustion Engineering plants. In response to the event at Calvert Cliffs, INPO issued SER 90-2, "Pressurizer Heater Sleeve Cracking," and NRC issued IN 90-10, "primary water stress corrosion cracking (PWSCC) of Inconel 600." The failure mode was identified as primary water stress corrosion cracking (PWSCC) of Inconel 600 nozzles found in pressurizers at CE plants. There had been no evidence of cracking of stainless steel nozzles in stainless steel pressurizers in Westinghouse plants. The root cause is still being investigated by Westinghouse and Virginia Power.

Westinghouse has issued this NSAL to increase awareness of pressurizer instrument tap nozzle cracks as a followup to Virginia Power's issuance of OE 7490 on 09/12/95. (This ends the CATS discussion.) (The issue of Inconel 600 is a concern that is being addressed by NES.)

Recommendation

The issue of Inconel 600 in areas not on the reactor vessel head should continue to be monitored and evaluated by NES. CATS ID # 10369 and NRC Information Notice 2000-17 are related to this issue.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 28, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of INPO SER 93-20 and Westinghouse "Info Gram 93-009, "Update on Ringhals Unit 2 Vessel Head Penetration Attachment Weld Indication, November 15, 1993

Brief Summary of Observation

The Ginna Operating Experience issued CATS ID # E03778 to disseminate information about this issue. The CATS information (which was prepared in the mid-1990's) is as follows:

Westinghouse is providing an update to results of the inspection of the Ringhals reactor vessel head penetration weld indication for information only. Westinghouse and Vattenfall are currently evaluating the safety significance of the indication and will provide additional information as information becomes available. As part of the program implemented to examine the Ringhals Unit 2 head penetrations for primary water stress corrosion cracking (PWSCC), Vattenfall decided to perform an examination on several penetration attachment welds that secure the penetration to the reactor vessel head. A total of six penetration positions were examined. These attachment welds form part of the structural and pressure boundary of the reactor vessel head/ penetration assembly and are subject to periodic visual inspection per ASME Section XI. Penetration testing performed revealed a linear indication at the periphery of the attachment weld that is approximately 180 mm long and extends for 110 degrees around the circumference of the weld to the vessel boundary.

Vattenfall has continued to pursue a program of resolution and cause identification. Currently larger boat samples at the vessel weld interface have been taken and analyzed, ECT and UT inspections have been performed on all penetration weld surfaces and UT examination has been performed from the penetration inside diameter to establish the integrity of the weld to penetration tube interface. Results indicate a significant lack of fusion with a physical gap at the weld to vessel interface. This lack of fusion zone extends for 180 degrees of the circumference. Also, 18 out of 28 inspected from the inside diameter of the penetrations have indicated a lack of fusion at the weld to penetration interface. Vattenfall plans to inspect all remaining penetration welds on the vessel head and on the bottom head. These weld issues are considered to be fabrication related. Vattenfall and Westinghouse has been unable to isolate these indications to Ringhals 2 and are currently performing an assessment of the technical and safety significance of this issue for Ringhals.

Supporting Information

Additional information from the CATS # E03778 file:
RG&E has been involved in initial presentations to the NRC by Westinghouse Owners Group and NUMARC. The Ginna vessel and Ringhals vessel are of similar design. However, the manufacturer of the Ginna vessel was Babcock and Wilcox Corporation. The Ringhals vessel manufacturer was the Rotterdam Shipyard. The failure mechanism has been determined to be related to the manufacturing process, and since the Ginna vessel was provided by a different vendor, it is not anticipated at this time that a basis exists to impact Ginna operations. The analysis of the Ringhals data on the weld of the penetration of the vessel is presently incomplete. Mechanical Engineering has developed an action plan to address the concerns of Westinghouse Info Gram 93-009 and INPO SER 93-20 that includes:

1. No immediate action is required since the two vessels were manufactured by different vendors.

2. Include this item in EWR 10028 "RV Head Adapter Tube"

    A. Support the Westinghouse Owners Group (WOG) and NUMARC in formulating a consolidate industry approach in addressing NRC concerns on CRDM issue.

    B. Monitor, through WOG and NUMARC, the results of the three sample inspection on U.S. plants which are proposed for 1994.

    C. Prepare a Ginna specific risk assessment to serve as basis for future actions.

3. In addition, complete a final report for the visual inspection that was performed at Ginna.

4. Use the 1993 visual inspection report as a documented basis for providing a reasonable degree of confidence that a boric acid buildup, which per WCAP 13565 would give cause for concern, does not exist at Ginna Station.

SCHEDULE:
Since Westinghouse Owners Group and NUMARC activities may impact the actions taken on the new weld issue; and the fact that this issue will be added to the scope of work of EWR 10028, a proposed action date for this accomplishment of a response to SER 93-20 is December 30, 1994 This date is to provide additional revised actions to address the concern based on results of further analysis and inspection work at U.S. plants.

Recommendation

None. As stated in closeout of this CATS item: Westinghouse Info Gram 93-009 and Westinghouse Info Gram 93-009a are being administratively closed and all further actions and closure information can be found in INPO SER 93-20. No further assessment of Info Gram 93-009 is necessary.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 28, 2002	**Opportunity for Improvement**	Yes	No	NA

Self-Assessment Data Sheet

Objective: Effectiveness Review of IP-HSC-3, Housekeeping Control

Brief Summary of Observation

Attachment 2 of IP-HSC-3 is titled, "Housekeeping Standards, Part 4 - Leakage Monitoring". There is a perception that this IP procedure is a part of the Boric Acid Program. In reviewing Attachment 2, it is clear that the Housekeeping Standards do not address or implement the requirements of NRC Generic Letter 88-05 or those of A-1407.

Supporting Information

To implement an INPO Finding in Section EQ.1-3 of the 1999 INPO report, CATS ID# M07964 was initiated, to "Develop a Program to track and monitor boric acid leaks". The commitment was: "a program to track and monitor boric acid leaks will be developed". The resolution of CATS ID # M07964 was "IP-HSC-3, Attachment 2, Part 4". In discussion with Tom Plantz, he cautioned me that this is strictly a housekeeping issue, and is not intended to include anything in Containment. However, the only statement addressing Containment states: "Leakage Monitoring Tags are not to be utilized in Containment or on non-plant system related equipment." This can be interpreted to apply (or not to apply) to all the other controls of IP-HSC-3. If Containment leaks are completely not applicable, then the Attachment 2 requirements are completely acceptable, since no part of the reactor coolant system pressure boundary is outside of Containment.

Recommendation

1. Consider whether to limit IP-HSC-3, Attachment 2, to only areas outside of Containment. If so, consider the need for a new program to track and monitor boric acid leaks inside Containment.

2. Revise IP-HSC-3 to either clearly exclude leaks in Containment, or upgrade to include the requirements of the Boric Acid Program, and initially discussed in GL 88-05 and implemented via A-1407.

Evaluated By	Jack St Martin	**Criterion Met**	Yes	No	NA
Date	March 28, 2002	**Opportunity for Improvement**	Yes	No	NA

ATTACHMENT V

Summary of Training Self-Assessment

1. Data Sheet for S-12.2

2. Data Sheet for Training Content

Self-Assessment Data Sheet

Objective: Evaluate the adequacy of Procedure S-12.2, OPERATOR ACTION IN THE EVENT OF INDICATION OF SIGNIFICANT INCREASE IN LEAKAGE, with regards to A-1407, PROGRAM TO PREVENT DEGRADATION OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS FROM BORIC ACID CORROSION.

Brief Summary of Observation

Reviewed 67 completed S-12.2, OPERATOR ACTION IN THE EVENT OF INDICATION OF SIGNIFICANT INCREASE IN LEAKAGE, procedures from June 1996 to present. The procedure is a checklist to determine if there is an actual leak or if something else is affecting the leakage indications. Conclusion: A-1407 is not consistently used as guidance for finding leaks in the reactor coolant pressure boundary.

Supporting Information

A majority of the 67 completed procedures determined that there was no increase in leakage. Changes in lake temperature, Recirc. Fan changes, and sampling were the main causes of the increased leakage indication.

A-1407 was performed and initialed in the procedure five times. There were however, several occurrences when A-1407 was performed but N/A'd in the procedure. There were also several instances when the procedure indicated that a containment entry was made to investigate the leak but A-1407 was not used.

Once the leak location has been determined, the actions taken to correct the leak are timely and appropriate, however, there is no documentation that the material was evaluated for degradation.

Recommendation

1. Ensure A-1407 is used as the guiding document for investigating reactor coolant pressure boundary leakage inside containment.

2. Reinforce management expectations to ensure A-1407 is used as the guiding document for investigating reactor coolant pressure boundary leakage inside containment. Upgrade S-12.2 if necessary.

Evaluated By	Steve Carter	Criterion Met	Yes No NA
Date	March 25, 2002	Opportunity for Improvement	Yes No NA

Self-Assessment Data Sheet

Objective: Evaluate the quality and content of training provided to implement the Boric Acid Program

Brief Summary of Observation

Reviewed the training programs and materials for information pertinent to the Boric Acid Program. The programs reviewed included Operations, Maintenance, and ESP.

Supporting Information

ESP – Lesson BGT08C "EROSION / CORROSION PROGRAM" provides information on the corrosion of carbon steel, however, it does not reference A-1407 or Generic Letter 88-05.

Maintenance – Lessons MM926C "PRACTICAL SHOP METALLURGY" and MM927C "BASIC CORROSION" provide adequate information for implementation for implementing the Boric Acid Program including referencing A-1407 and the events included in Generic Letter 88-05. These lessons are for mechanical maintenance only. There are no lessons for Radiation Protection, Chemistry, Electrical, or I&C that provide training on the Boric Acid program.

Operations – Lesson RAD15C "PROGRAM TO PREVENT DEGRADATION OF REACTOR COOLANT PRESSURE BOUNDARY" provides adequate information for implementation for implementing the Boric Acid Program including referencing A-1407 and the events included in Generic Letter 88-05. It also includes references to S-12.2, OPERATOR ACTION IN THE EVENT OF INDICATION OF SIGNIFICANT INCREASE IN LEAKAGE and SOER 84-5, Bolt Degradation or Failure in Nuclear Power Plants. There have also been two training sessions based on industry events concerning leak detection and boric acid corrosion (SEN182 and SEN 216). These training sessions were held in Cycles 99-2 and 01-1.

Recommendation
1. Submit a TWR to evaluate the need for training on A-1407 in the ESP program.
2. If RP Techs are going to perform the initial containment entry tour, they should be trained in the use of A-1407.
3. All personnel, who have access to areas containing piping with borated water, should be made aware of the requirements of A-1407.
4. Submit a TWR for evaluating the need for refresher/continuing training on A-1407 in all applicable programs.

Evaluated By	Steve Carter	Criterion Met	Yes	No	NA
Date	March 26, 2002	Opportunity for Improvement	Yes	No	NA

ATTACHMENT VI

Summary of PT-7 Issues

1. Data Sheet for PT-7 Issues

2. Recommendation and Justification for PT-7 Meeting

3. Summary of Items that Require Attention Prior to Next PT-7

Self-Assessment Data Sheet

Objective: This letter contains a recommendation for the PT-7 system leakage test to be performed during the scheduled 2002 refueling outage.

Brief Summary of Observation

Enhance the controls over and documentation of performance of PT-7.

Supporting Information
JUSTIFICATION FOR MEETING

1. NRC Bulletin 2002-01 requires all licensed pressurized water nuclear power plant owners to submit a basis for concluding their boric acid inspection program is providing assurance of compliance with Generic Letter 88-05. This basis must be submitted to the NRC by May 18, 2002. A well executed PT-7 test of the reactor coolant system, providing clear and concise procedure direction and documentation, would greatly reinforce our commitment to the prevention of boric acid corrosion at Ginna Station.
2. Procedure A-1407 was written to comply with the requirements of NRC Generic Letter 88-05. Procedure A-1407 has not been given adequate consideration in the past when performing the PT-7 test.
3. Better coordination and direction of personnel while performing the PT-7 leakage test would increase cohesion of work activities between work groups.
4. A review of past PT-7 paperwork indicates a need for improvement. The paperwork is hard to decipher and would not be easy to audit.
5. A review of any impact to the schedule due to PT-7 test enhancements could be discussed.

Recommendation

Conduct a meeting prior to performance of PT-7 to discuss the work scope of the PT-7 system leakage test of the reactor coolant system, and how that relates to our program for the prevention of boric acid corrosion. All work groups involved in the PT-7 test should be invited to this meeting.

Evaluated By	Bruce Goranowski	**Criterion Met**	Yes	No	NA
Date	March 27, 2002	**Opportunity for Improvement**	Yes	No	NA

Ginna Nuclear Power Plant
Ginna Station Program for Prevention of Boric Acid Corrosion
Self Assessment 2002-0037
March 27, 2002

This letter contains a recommendation for the PT-7 system leakage test to be performed during the scheduled 2002 refueling outage.

RECOMMENDATION

Conduct a meeting by April 5, 2002 to discuss the work scope of the PT-7 system leakage test of the reactor coolant system, and how that relates to our program for the prevention of boric acid corrosion. All work groups involved in the PT-7 test should be invited to this meeting.

JUSTIFICATION FOR MEETING

- NRC Bulletin 2002-01 requires all licensed pressurized water nuclear power plant owners to submit a basis for concluding their boric acid inspection program is providing assurance of compliance with Generic Letter 88-05. This basis must be submitted to the NRC by May 18, 2002. A well executed PT-7 test of the reactor coolant system, providing clear and concise procedure direction and documentation, would greatly reinforce our commitment to the prevention of boric acid corrosion at Ginna Station.
- Procedure A-1407 was written to comply with the requirements of NRC Generic Letter 88-05. Procedure A-1407 has not been given adequate consideration in the past when performing the PT-7 test.
- Better coordination and direction of personnel while performing the PT-7 leakage test would increase cohesion of work activities between work groups.
- A review of past PT-7 paperwork indicates a need for improvement. The paperwork is hard to decipher and would not be easy to audit.
- A review of any impact to the schedule due to PT-7 test enhancements could be discussed.

Ginna Nuclear Power Plant
Ginna Station Program for Prevention of Boric Acid Corrosion
Self Assessment 2002-0037
March 29,2002

This document identifies items that require special attention and/or improvement for performance of the upcoming PT-7 examination. These improvements were derived from reviewing previous PT-7 test data, and by suggestions from Self Assessment 2002-0037 team members. Also, in light of current industry events (Davis-Besse reactor head degradation) and NRC requirements, our boric acid corrosion program should be aligned with the PT-7 test.

1. Generic Letter 88-05 and NRC Bulletin 2002-01

- Assure we have complied with these two documents.
- Generic Letter 88-05 is a request from the NRC to assure we have a procedurally controlled program to ensure that boric acid corrosion does not lead to degradation of the reactor coolant system. Procedure A-1407 was written to comply with NRC Generic Letter 88-05.
- NRC Bulletin 2002-01 requires all licensed pressurized water nuclear power plant owners to submit a basis for concluding their boric acid inspection program is providing assurance of compliance with Generic Letter 88-05. A well executed PT-7 test of the reactor coolant system, providing clear and concise procedure direction and documentation, would greatly reinforce our commitment to the prevention of boric acid corrosion at Ginna Station.

2. A-1407 Procedure Items

- Assure procedure A-1407 is aligned with the PT-7leakage examination.
- Assure components in A-1407 on Attachment I and II are inspected.
- (3.5.6) If any component degradation is noted, initiate an ACTION Report.
- (3.5.7) If degradation is recorded, NRC Generic Letter 88-05 shall be reviewed.

3. PT-7 Procedure Items

- (4.1) Assure the ISI Engineer, or designee, have reviewed the VT-2 examination test boundary drawings.
- (4.2) Assure qualified VT-2 test personnel are assigned to perform the leakage examination.
- (4.3) Assure maintenance has provided the Maintenance Leakage Checklist.
- (4.4) Assure a work order has been initiated to document and control maintenance activities.

4. Applicable Work Order Package for Maintenance PT-7 Support

- Is the work order package ready for PT-7
- Designate the plateau on the PT-7 Data Sheets.
- Capture the activity performed in more detail on the PT-7 Data Sheet.
- Completely fill out the Decon List, i.e. action taken, status.
- Permanently record and document the area, and amount of boron cleaned.

5. VT-109 Procedure Items

- Capture PT-7 activities in more detail on the "Visual Examination For Leakage" data sheet.
- Generate and permanently record an insulation removal list (8.3.5.4).
- Adhere to the recording criteria and permanently document it. Examples of VT-109 recording criteria is listed below.
  a. (9.2) Attach working copy data to the final copy.
  b. (9.3) All supporting material shall be attached to the report.
  c. (9.4) Define the leak for recordable and insignificant indications.
  d. (9.6) Record the color and dimensions of boric acid buildup.
  e. (9.6.1) Exact location of leakage/boric acid residue/corrosion shall be described by referencing proximity to welds, supports, and components.
  f. (9.6.1.2) Record the depth and area of boric acid corrosion on components.
  g. (9.9) Document examination boundaries, and include them in the permanent record.

6. General Items of Interest

- Who is the PT-7 Coordinator?
- Who is on the Outage Planning Group PT-7 Committee?
- Is the PT-7 Committee and Coordinator ready for PT-7?
- Conduct a pre and post job brief (IP-HPE-4 and IP-HPE-5).
- Are all applicable procedures ready for the PT-7?
- Complete data sheets in a timely manner with accurate dates.

ATTACHMENT VII

Table of GL 88-05 Requirements and Implementing Documents

	GL 88-05	Implemented By	Assessment
1A	<u>**PURPOSE OF GL**</u> To assess safe operation of pressurized water reactors (PWRs) when reactor coolant leaks below technical specification limits develop and the coolant containing dissolved boric acid comes in contact with and degrades low alloy carbon steel components	Not evaluated	
1B	The principal concern is whether the affected plants continue to meet the requirements of General Design Criteria 14, 30, and 31 of Appendix A to10CFR Part 50 when the concentrated boric acid solution or boric acid crystals, formed by evaporation of water from the leaking reactor coolant, corrode the reactor coolant pressure boundary.	Not evaluated	

2A	**<u>GL - WHY'S</u>** Our concerns regarding this issue were prompted by incidents in PWR plants where leaking reactor coolant caused significant corrosion problems. At Turkey Point Unit 4, leakage of reactor coolant from the lower instrument tube seal on one of the incore instrument tubes resulted in corrosion of various components on the reactor vessel head including three reactor vessel bolts. At Salem Unit 2, leakage occurred from the seal weld on one of the instrument penetrations in the reactor vessel head, and the leaking coolant corroded the head surface. At San Onofre Unit 2, boric acid solution corroded nearly through the bolts holding the valve packing follow plate in the shutdown cooling system isolation valve. During an attempt to operate the valve, the bolts failed and the valve packing follow plate became dislodged causing leakage of approximately 18,000 gallons At Arkansas Nuclear One Unit 1. leakage from a high pressure injection valve dripped onto the high pressure injection nozzle. At Fort Calhoun, seven reactor coolant pump studs were reduced by boric acid corrosion from	Not directly evaluated. Examples were considered when looking at how GL 88-05 is implemented at Ginna.	

2B	In many of these cases, although the licensees had detected the existence of leaks, they had not evaluated their significance relative to the safety of the plant nor had they promptly taken appropriate corrective actions.	Not directly evaluated. Examples were considered when looking at how GL 88-05 is implemented at Ginna.	See item 4C. Instances were found where action taken was not well documented such as initial inspection of containment this outage.	
2C	In December 1984, the Electric Power Research Institute issued a summary report on the corrosion of low alloy steel fasteners which, among other things, discussed boric acid-induced corrosion. The information contained in these documents clearly indicated that boric acid solution leaking from the reactor coolant system can cause significant corrosion damage to carbon steel reactor coolant pressure boundaries.	Step 3.5.7.4 in procedure A-1407 recommends that the recommendations of this report, NP 5985, be considered.	The report has been updated by EPRI. The later version of the report should be evaluated and recommendations included in a more user friendly fashion in applicable procedures.	

3A	**GL -PROGRAM PURPOSE** Office of Inspection and Enforcement (IE) Bulletin 82-02 requested licensees to identify all of the bolted closures in the reactor coolant pressure boundary that had experienced leakages and to inform the NRC about the inspections to be made and the corrective actions to be taken to eliminate that problem. However, the bulletin did not require the licensees to institute a systematic program for monitoring small primary coolant leakages and to perform maintenance before the leakages could cause significant corrosion damage.	Bolted closures are included in the listings in A-1407. Bolted closures are inspected during PT-7.	Acceptable
3B	In light of the above experience, the NRC believes that boric acid leakage potentially affecting the integrity of the reactor coolant pressure boundary should be procedurally controlled to ensure continued compliance with the licensing basis.	A-1407 is our program.	Procedure is not always linked in other procedures used to implement the program. It is not a procedure which most plant personnel use or refer to when performing activities relevant to its contents.
3C	We therefore request that you provide assurances that a program has been implemented consisting of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture	A-1407 is our program.	Procedures used to perform diagnostic checks, evaluation of leaks and corrective actions, e.g.: PT-7, A1603.3, IP-CAP-1 and decontamination instructions do not have required systematic measures. Ginna relies on experience and knowledge of key indivduals who may not always be a part of the process.

4A	**GL PROGRAM REQUIREMENTS** A determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion. Particular consideration should be given to identifying those locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces.	A-1407 has two lists, one by component and a parallel list by location. List was originally based on guidance in item in 5B below and our initial response to NRC.	Focus is on removing items from list rather than adding to it if a reverse change occurs. Interviews with License Renewal confirmed that list is reasonably current. Reviews of our responses to leaks indicate we do not rely on it to determine corrective actions. The locations with carbon steel do not receive special attention when looking for leaks in PT-7 and S-12.2.

4B	Procedures for locating small coolant leaks (i.e., leakage rates at less than technical specification limits). It is important to establish the potential path of the leaking coolant and the reactor pressure boundary components it is likely to contact. This information is important in determining the interaction between the leaking coolant and reactor coolant pressure boundary materials.	S-12.2 requires operations to investigate indications of small RCS leaks. Requires documentation on ACTION Report or Trouble Report for evaluation once located. PT-7 used prior to initial startup after each outage to inspect RCS and other systems in containment after each RFO at operating pressure. Procedure RF-65.1 inspects conoseals during refueling and documents any evidence of BA crystals. A line item in outage schedule performs inspections and cleanup of Boric Acid deposits in containment.	S-12.2 was frequently used, however the relationship to A-1407 when leak is located or a containment entry is needed needs to be strengthened. PT-7 and related activities need to be strengthened to improve the documentation and formality of the directions for the inspection and maintenance portions of test. The refueling procedures should be revised to provide better visibility and formal direction of inspections performed for evidence of Boric Acid deposits. The initial inspection of containment is an important diagnostic and preventative tool which should be controlled by procedure which implements the engineering evaluation process and documents the results in a manner suitable for trending and evaluation. See 4C below for related information

4C	Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.	The methods are left to technical expertise of planner or engineer. Program relies heavily on people knowing contents of A-1407 which implements this provision	Information is not always gathered as recommended before evidence is removed. Procedural direction in corrective action procedures such as A-1603.3 and IP-CAP-1 do not directly implement the requirement.	
4D	Corrective actions to prevent recurrences of this type of corrosion. This should include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at the locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings/claddings.	Some carbon steel bolts and parts have been replaced with corrosion resistant materials. Valve packing improvement program has helped reduce leakage probability. Corrective action program trending and Maintenance Rule program used to determine repetitive leaks.	Documentation of leaks observed and action taken is not always available or required for some leaks especially those noted on containment entry after a power run. Specific efforts to trend indications of reactor coolant leaks was not found.	

4E	You shall maintain, in auditable form, records of the program and results obtained from implementation of the program and shall make such records available to NRC inspectors upon request.	The various procedures and their revisions which implement aspects of the program are retained as a record. Documentation of inspections, tests, work performed and corrective actions are maintained when required by procedure. Of note, A-1407 does not require any specific records.	Records are retained by record type and are as difficult to audit as most other items at Ginna when a functional area is examined rather than trying to find a single activity record. Records examined exhibited a lack of attention to detail and a lack of clarity as to what was accomplished. Recourse to the originator was needed in some cases. Color photos documenting as found conditions are copied and photographed using black and white microfilm. The color image, which may also be digital, is shipped for archival storage. Colors of deposits and components are important attributes in evaluating Reactor Coolant leakage impact and history. As noted above documentation of leakage is not always generated in a manner in which it will be maintained as a record. The level of detail documented as to extent of an engineering evaluation is highly dependent on individuals involved. Details of what was looked at and what was done to correct items found are important.	

5A	**GL-IMPLEMENTATION GUIDANCE** In many cases, however, coolant that leaks out of the reactor coolant system loses a substantial volume of its water by evaporation, resulting in the formation of highly concentrated boric acid solutions or deposits of boric acid crystals. These concentrated solutions of boric acid may be very corrosive for carbon steel. This is illustrated by recent test data, tabulated below, which were referenced in NRC Information Notice No. 86-108, Supplement 2.(See Bulletin for Table of Corrosion Rates) If all of the water evaporates and boric acid crystals are formed, the corrosion is less severe. However, boric acid crystals are not completely benign toward carbon steel, and at a temperature of 500F, corrosion rates of 0.8 to 1.6 mils/month were obtained in the Westinghouse tests referenced in the generic letter	Our practice appears to be to clean up residues when detected. This is background information which should be used by personnel trained in performing engineering evaluations.	This should be included in procedures which perform engineering evaluations of a leak location.

5B	The most effective way to prevent boric acid corrosion is to minimize reactor coolant leakages. This can be achieved by frequent monitoring of the locations where potential leakages could occur and repairing the leaky components as soon as possible. Review of the locations where leakages have occurred in the past indicates that the most likely locations are (1) valves; (2) flanged connections in steam generator manways, reactor head closure, etc.; (3) primary coolant pumps where leakages occur at cover to-casing connections as a result of defective gaskets; and (4) defective welds. In many of these locations the components exposed to boric acid solution are covered by insulation and the leaks may be difficult to detect. If leak detection systems have been installed in the components (e.g., reactor coolant pumps from certain vendors), they should be used to monitor for leakage.	List is in Attachments I and II of A-1407. Our response to GL 88-05 indicated that we used these types of components as the basis for our reviews of RCS to determine locations with carbon steel. These types of joints are examined during periodic pressure tests per PT-7.	The term "frequent monitoring of the locations where potential leakages could occur" can be interpreted with much latitude. Do we keep an eye on known problem areas or those where carbon steel is located other than once an outage. We have a leak detection system on reactor vessel flange joint. S-12.2 investigates other symptoms. We do potentially have direct monitors, e.g.: video cameras but unsure if they consider 88-05 issues.	
5C	It is important to determine not only the source of the leakage but also the path taken by the leaking fluid by evaluating the mechanism by which leaking boric acid is transported. In some cases boric acid may be entrained in the steam emerging from the opening in the pressure boundary that subsequently condenses inside the installation thus carrying boric acid to locations that are remote from the source of leakage.	Procedure A-1407 partially implements this requirement by requiring documenting the extent prior to removal in a work order. This can proceed evaluating the full impact and may remove trail of where boric acid went.	Not always done. No effective procedural or process hooks to trigger use of A-1407 by personnel who typically find it or process work orders.	

5D	Boric acid corrosion can be classified into two distinct types: (1) corrosion that actually increases the rate of leakage and (2) corrosion that occurs some distance from the source of leakage and hence does not significantly affect the rate of leakage. An example of the first type is the corrosion of fasteners in the reactor coolant pressure boundary, for example, in reactor coolant pumps. This type of corrosion can lead to excessive corrosion of studs. The second type of corrosion can contribute significantly to the degradation of the reactor coolant pressure boundary.	This guidance is not included in our program.	We do not try to characterize corrosion in this manner. Deposits of both types are noted.	
5E	Because of the nature of the corrosion produced by boric acid, the most reliable method of inspection of components is by visual examination. Ultrasonic testing performed in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code may not be sensitive enough to detect the wastage.	VT(visual method) series procedures used in leakage exams.	Visual is our primary method. We also use other measurements such as leak rates and airborne radioactivity to trigger a visual inspection.	

	New Items In Bulletin 2002-01		
6A	New Section ASME Section XI paragraphs referenced in Bulletin 2002-01.	Items not directly referenced in applicable program procedures, e.g.: A-1407 & PT-7. ISI program and procedures reference and implement inspection provisions.	Section XI references should be reviewed and included and referenced in applicable procedures and documents implementing the GL 88-05 Program.
6B	10 CFR 50 App B Criterion V applies to visual and volumetric examinations of reactor coolant pressure boundary especially with respect to acceptance criteria and determination of satisfactory accomplishment.	Visual and volumetric exams by TPFI (LIS) or QC are covered by procedures with acceptance criteria.	Reports of inspections can be improved as to details of what was inspected.
6C	10 CFR 50 App B Criterion IX applies to visual and volumetric examinations of reactor coolant pressure boundary especially with respect to process qualification and a plant specific analysis demonstrating that methods reliably detect and accurately characterize flaws or degradation.	Visual and volumetric exams procedures used by TPFI have been process qualified and personnel are qualified in their use. A plant specific analysis demonstrating reliable detection and accurate characterization of flaws was included with 15 day response for the head.	Examination techniques are qualified on a generic basis. Except for the details in the 15 day response for the head, we do not appear to have a plant specific analysis for the remainder of the reactor coolant system with respect to GL 88-05 concerns for reliable flaw detection. Subsequent characterization would be flaw specific.

6D	10 CFR 50 App B Criterion XVI applies. It also indicates that for degradation of the reactor coolant pressure boundary, root cause determination is important for understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the boundary	Procedure A-1407 requires use of an ACTION Report or work order for Boric Boric Acid degradation and are not specifically listed as reporting threshold in IP-CAP-1. A root cause is required only if a significant condition adverse to quality is noted per IP-CAP-1. It is not clear that the degradation of this nature would trigger a root cause determination.	IP-CAP-1 could be revised to include boric acid deposits as a reporting threshold and a root cause determination requirement for leakage which results in reactor coolant boundary degradation.	
6E	10 CFR 50 App A Criterion 14 – *Reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.	Not specifically evaluated. This is an initial design consideration and goal of GL 88-05 program		
6F	10 CFR 50 App A Criterion 30 - *Quality of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	Not specifically evaluated. This is an initial design consideration and goal of GL 88-05 program		

6G	10 CFR 50 App A Criterion 31 – *Fracture prevention of reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws	Not specifically evaluated. This is an initial design consideration and goal of GL 88-05 program		
6H	10 CFR 50 App A Criterion 32 – *Inspection of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.	Not specifically evaluated. This is an initial design consideration.	Some design changes may be needed. If additional access is needed for any new inspections, material changes may be needed to implement this bulletin from the plant specific analysis. See item 6C above.	

ATTACHMENT VIII

Supporting Information for Findings and Recommendations

SUPPORTING INFORMATION FOR FINDINGS AND RECOMMENDATIONS

Finding #1

The program and its governing document have suffered from a lack of focus over time. Examples are noted below.

a. Procedure A-1407 has not been converted to an IP or ND in a manner similar to other comparable programs.

b. Interviews noted that procedure A-1407 is not part of the training people responsible for implementing it receive.

c. A-1407 is applicable to but not referenced and incorporated in current procedures which now implement its requirements, e.g.: IP-CAP-1 and A-1603.3.

d. Procedure M-37.91, which implemented Generic Letter 88-05, was deleted without a suitable replacement when the responsibility for implementation changed.

e. Current revisions of the ASME Section XI which include relevant requirements have not been incorporated and referenced.

f. EPRI Report 5985 is referenced in A-1407 and GL 88-05. EPRI TR-104748 is the latest and should be factored into and referenced in the program.

Recommendation #1

A requirement for having a GL 88-05 Program should be included in a suitable Nuclear Directive. It is recommended that A-1407 be converted to an IP Level Document and updated. Other procedures and methods used to implement the program should be evaluated and updated accordingly.

Finding #2

Procedure IP-CAP-1, which is used to report and disposition degradation of components caused by reactor coolant system leakage, could be enhanced to ensure the appropriate root cause determination is performed in accordance with NRC Bulletin 2002-01. A review of IP-CAP-1 and related ACTION Reports:

a. Reactor Coolant Leakage or evidence thereof is implied but not specifically identified as a threshold for Action Report initiation in Attachment 1.6.

b. An engineering evaluation of the impact of reactor coolant leakage on surrounding items by the system engineer could be performed as part of a TE/past operability evaluation. It is not clear what level of reactor coolant leakage triggers a TE.

c. An ACTION Report describing degradation from Reactor Coolant Leakage does not meet the thresholds or wording in Section 3.2 requiring a root cause determination. Bulletin 2002-01 indicates the that degradation from reactor coolant leakage is a significant condition warranting a root cause determination and correction.

d. Guidance to leave boric acid deposits in place until an engineering evaluation is completed as recommended by GL 88-05 is not included in IP-CAP-1.

e. Several examples of ACTION reports were reviewed. When they were assigned to the Reactor Coolant System Engineer. the provisions of GL 88-05 were addressed and an in-depth cause investigation performed without designating it as a root cause on the ACTION report. This is attributed to knowledge of the system engineer and the fact that he is currently the Responsible Person for A-1407.

Recommendation #2

It is recommended that IP-CAP-1 be revised to include provisions which implement GL 88-05 and Bulletin 2002-01.

Steps would be as follows:

1)Ensure that the criteria of A-1407, Step 3.5.6, is adequately captured in IP-CAP-1, Attachment 1-6, Section 4. Step 3.5.6 states:

> 3.5.6 If any degradation is noted, an ACTION Report shall be initiated in accordance with IP-CAP-1, and Laboratory Inspection Services (LIS) shall inspect and determine the extent of degradation, and forward this information to Primary Systems Engineering.

2)Document condition.

3)Evaluate and document impact and operability. (Where was its source? Where did it go? What type of materials are involved? What is impact? What is the reason for the leak? Is the leak dependent on current plant conditions which are different from normal operations?)

4)If no carbon steel or wastage evident initiate cleanup, decon and correction of source. Consider nature of leak. Should cleanup wait for different plant conditions?

5)If carbon steel or other damage possible initiate step 3 plus and inspection for damage and acceptable cleanup.

6)Initiate action to correct any damage which is not otherwise acceptable. Determine a root cause of the leak and damage and evaluate actions needed to prevent recurrence.

Finding #3

An inspection of the containment including looking for and documenting Boric Acid deposits is included as part of the process of preparing the containment for entry at the start of an outage. The location of Boric Acid deposits also identifies areas needing decontamination. Several items relating to this important activity as discussed below should be improved.

a. A procedural or program requirement for performing and documenting results of the effort does not exist. It is currently directed by an outage activity line item.

b. The effort in the past has been performed by several different groups such as TPFI, QC, Mechanical Maintenance, RP etc. with different perspectives and results.

c. This year the effort was performed by the RP group using the decontamination perspective. They were not given any training in or direction to use GL 88-05 program or A-1407. The places where boric acid deposits were found were well documented and cleaned up without an engineering evaluation. It can not be determined if these actions contradicted the guidance in Generic Letter 88-05, "measure" #(3):

> (3) Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup. Actions had to be initiated by the assessment team to have the locations inspected in accordance with GL 88-05. (ACTION Report 2002-0713 was initiated to address this issue.)

d. Starting in 1989, the initial inspection and cleanup in Containment was performed by technicians who were trained in and/or complied with the requirements of A-1407, Step 3.5.4:

3.5.4 If boric acid has been in contact with RCPB carbon steel components, the extent of contact shall be documented on a Maintenance Work Request (Trouble Card) (A-1603), before removal of any boric acid crystal build-up.

e. The maintenance planning group was given a copy of the list. Work orders were being planned to correct the apparent cause of the deposits. Discussions with the planner indicated that issuing an AR/TR for each noted leak is specified in A-1407 would not be efficient.

f. Procedure A-1603.3 which would be used to process the AR/TR as a work order was reviewed. This procedure did not include the need to perform an engineering evaluation prior to cleanup as required. Discussions with maintenance planning supervision confirmed the lack of investigation before cleanup or disturbing the deposit.

g. Based on additional items noted during evaluating c. above, the extent, methods and training for this evolution should be further evaluated. Indications are that Boric Acid deposits which existed at time of walk down by RP were not detected until later in the outage.

Recommendation #3

It is recommended that the containment inspection process prior to an outage be reviewed and strengthened as discussed above.

Finding #4

The documentation of and directions to personnel performing a pressure test and leakage examinations of the Reactor Coolant System and adjoining systems are in need of improvement. Reviews of PT-7, referenced procedures, ASME Section XI, related documentation and related records found that:

- Instructions defining what insulation must be removed to assure bolted connections are visually inspected per Section XI are not provided.
- A maintenance list is developed in PT-7, however, the criteria for developing it and individuals responsible for preparing and reviewing it are not provided.
- Disposition of leakage noted by inspection personnel is not always evident.
- A PT-7 committee is required to review results at an intermediate pressure but not at the final test pressure. The makeup of this committee is not defined.
- The emphasis required by Generic Letter 88-05 on susceptible areas is not evident in PT-7.
- The attention to detail and the descriptions of what was examined during a test can be improved per Bruce Goranowski's reviews.

Recommendation #4

It is recommended that PT-7 procedure. documentation methods and conduct process be examined and strengthened to remedy the above prior to the PT-7 test for the 2002 outage.

Finding & Recommendation #5

Several initiatives such as a leakage monitoring program outside containment and reducing the priority of packing leakage ACTION Reports may detract from the effectiveness of the GL 88-05 intent. See IP HSC-3. See IOC from G Hermes & J Zapetis to Rudy Forgensi dated 1/9/02. It is recommended that these policies be reviewed with respect to GL 88-05 and credit taken for "zero tolerance" good practice with respect to cleanup of Boric Acid when found.

Finding and Recommendation #6

A review of impact of prior leakage around and above the pressurizer should be considered for impact on carbon steel shell.

A review of the construction details of the pressurizer found several areas which may warrant additional attention in light of Bulletin 2002-01:

- Heater penetrations
- Nozzles areas where stainless steel inserts are welded to cladding
- Nozzles with where thermal sleeves or other items are welded to the cladding

ATTACHMENT IX

Boric Acid Corrosion Program - License Renewal Issues

88-05 Boric Acid Corrosion Program - License Renewal Issues

Scope of the program must be broadened to include all borated water systems and all systems that are in proximity to borated water systems in all areas of the plant (i.e., Aux Bldg., Intermed Bldg., etc), not just restricted to RCS in Containment.

Program must include examination of structures and components in all areas and spaces which could potentially be wetted and damaged by leaking boric acid. This includes external surfaces of carbon/low alloy steel bolting, carbon/low alloy steel piping and equipment, structural members and bolting, electrical cables, cable trays, cable connectors, conduits, racks, panels and enclosures, etc.

The program must be enhanced so that these locations, areas, spaces, and components are specifically addressed.

The program should provide the following:

- Locations for surveillance and detection,
- Procedures for investigation of leaks, which includes determination of the source(s),
- Methods for engineering evaluation, including assessment of the condition of the surface(s) affected and measurements, data, etc, to verify serviceability of the structure or component affected,
- Removal and cleanup,
- Corrective action.

The Boric Acid Corrosion Program is credited as a License Renewal Aging Management Program for managing boric acid corrosion (wastage) of external surfaces of structures and components in all borated water systems and all systems that are in proximity to borated water systems. The Systems Monitoring Program and Structures Monitoring Program will be enhanced to reference the Boric Acid Corrosion Program.

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VISUAL EXAMINATION FOR LEAKAGE

Effective Date: 08/29/2000

Expiration Date:

Prepared By:	Kenneth Kemp		**Date:** 05/11/2000
Reviewed By:	Paul Lewis	**Status:** Approved	**Date:** 08/23/2000
Approved By:	Michael Saporito	**Status:** Approved	**Date:** 08/29/2000

1.0 PURPOSE AND SCOPE

1.1 The purpose of this procedure is to provide the minimum requirements necessary to accomplish Visual Examination for Leakage (VT-2) from Pressure Retaining Components at R. E. Ginna Station.

1.2 The Visual Examination will be performed to the requirements specified in the Fourth Interval Inservice Inspection Program Document.

1.3 This procedure complies with the requirements of NDE-201.

2.0 REFERENCES

2.1 ASME BPV Code, **Section XI**, "Rules for Inservice," Inspection of Nuclear Plant Components", 1995 Edition. 1996 Addenda

2.2 ASME BPV Code, **Section XI**, "Rules for Inservice," Inspection of Nuclear Plant Components", 1992 Edition. 1992 Addenda

2.3 ASME BPV Code, **Section XI, Code Case N-416-1** , "Alternate Pressure Test Requirements for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2 and 3."

2.4 ASME BPV Code, **Section XI, Code Case N-498-1** , "Alternate Rules for 10-Year System Hydrostatic Testing For Class, 1, 2 and 3 Systems."

2.5 **ASME NQA-1** - "Quality Program Requirements for Nuclear Facilities," 1989 Edition

2.6 **IP-IIT-1**, Interface Procedure, "ASME Section XI Repair and Replacement Process for Class 1, 2 and 3."

2.7 **NDE-102** - "Qualification of Visual Examination Personnel"

2.8 NDE-203 - "NDE Recording Forms"

2.9 NDE-806 - "Leakage Test Boundaries"

2.10 VT-101 - "Visual Examination Acceptance Criteria"

3.0 **RESPONSIBILITY**

3.1 Certified VT-2 examination personnel shall be responsible for complying with the requirements of this procedure.

4.0 **CONTENTS** **Section**

5.0 **PERSONNEL QUALIFICATIONS**

Personnel performing Visual Examinations in accordance with this procedure shall be qualified in accordance with the requirements of NDE-102.

6.0 **EQUIPMENT**

6.1 Commercially available equipment shall be used as necessary for the performance of the bolting examination.

6.2 General Electric Model 214 or equivalent light meter.

6.3 A near distance vision test chart (Character Card) containing lower case text no greater than .158" in height for VT -2 examinations.

6.4 Borescopes, fiberscopes, telescopes, closed circuit television, cameras, mirrors or other such equipment may be used for the remote method of visual examination.

7.0 **CALIBRATION**

7.1 All measuring equipment requiring calibration shall have a label affixed to it, indicating date of calibration and date of recalibration.

7.2 Equipment that is out of calibration shall not be used.

7.3 All examination data obtained after the calibration expiration date shall be invalid

7.4 Light meters shall be calibrated annually and verified as follows

- Nuclear work - daily verification
- All Other Work (e.g. fossil) - bi-weekly verification

7.5 Character cards containing lower case text without ascenders or descenders shall be verified only once, dated and signed by the Level III Visual examiner.

7.6 Battery powered lighting shall be verified after each examination, series of examinations but not to exceed 4 hours.

8.0 EXAMINATION

8.1 **Visual Examination Methods**

8.1.1 Direct Visual Examination Method

8.1.1.1 The Visual Examination shall be performed by placing the eye within 6 (six) feet of the surface to be examined.

8.1.1.2 The Visual Examination shall be performed with the eye at an angle not less than 30 degrees.

8.1.1.3 The Visual Examination shall be performed using a character card containing lower case text without ascenders or descenders to demonstrate resolution.

8.1.1.4 The Visual Examination may be performed using visual aids to improve the angle of view or to enlarge the area of interest (e.g. mirrors, magnifying glass).

8.1.1.5 The illumination requirements shall be verified in the least discernible location on the area to be examined.

8.1.1.5.1 A GE model 214 light meter or equivalent shall be used to determine that the examiner has a minimum of 15 foot candles illumination to perform the direct visual examination.

8.1.1.5.2 It is not necessary to measure illumination levels on each examination surface when the same portable light source or similar installed lighting is demonstrated to provide the specified illumination at the maximum examination distance.

8.1.1.5 3 When illumination aids are used, they shall be positioned in such a way that leakage or evidence of leakage will not be obscured by excessive or insufficient illumination.

8.1.2 Remote Visual Examination Method

8.1.2.1 Remote Visual Examination shall be used at any time that the requirements for Direct Visual Examination (Paragraph 8.1.1) cannot be met.

8.1.2.2 The examiner shall demonstrate that the remote method used has the ability to provide resolution at least equal to that obtainable by the direct method.

8.2 **Hold times**

8.2.1 The examiner shall verify and document that the hold times, after attaining test pressure, and temperature (where required), are as follows:

Test Type	Hold Time
System Leakage	No holding time required after attaining test pressure an temperature; except for tests required by IWA-4540, (or code case N-416-1 is invoked), a 10 min. holding time fc noninsulated systems or components, or 4 hr. for insulat systems or components, is required after attaining test pressure and temperature.
System Hydrostatic	4 hr. holding time required after attaining the test pressu temperature conditions for insulated systems, and 10 mi noninsulated systems or components.
System Pneumatic	10 min. holding time required after attaining the test pre:

8.2.2 Gages

8.2.2.1 When conducting system hydrostatic or, system pneumatic tests the examiner shall verify and document the gage serial number, range, date of calibration and test pressure of the gage used for the test.

8.2.2.2 Any pressure measuring instrument or sensor including the pressure measuring instrumentation of the normal operating system may be used as set forth in the applicable testing procedure.

8.2.3 The examiner shall verify and document valve lineup by verifying sign-offs in the applicable testing procedure.

8.3 **Examination Requirements**

8.3.1 Parts, surfaces, components and joints shall be examined while under pressure to identify any leakage.

8.3.2 Definitions

8.3.2.1 Examination Boundary

8.3.2.1.1 Examination boundaries are components and piping which shall have VT-2 examinations in accordance with this procedure.

8.3.2.2 Test Boundary

8.3.2.2.1 Test boundaries are components and piping which extend beyond the examination boundary and are not subject to this procedure.

8.3.2.3 Examination Types

8.3.2.3.1 System Leakage - A system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature.

8.3.2.3.2 System hydrostatic - A system hydrostatic test conducted during a plant shutdown at a pressure above nominal operating pressure or system pressure.

8.3.2.3.3 System pneumatic - A system pneumatic test conducted in lieu of either of the above system pressure testes for Class 2 or 3 components as permitted by IWC-5000 or IWD-5000. The requirements for system leakage and hydrostatic tests are applicable to pneumatic tests.

8.3.3 The components to be examined are identified as the "examination boundary" on the P&ID drawing that is part of the system test procedure.

8.3.3.1 The VT-2 Examiner is required to examine only the "exam boundary".

8.3.4 General examination for leakage, corrosion and boric acid residue in noninsulated components

8.3.4.1 The visual examination VT-2 shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage, corrosion or boric acid residue.

8.3.4.2 For components whose external surface are inaccessible for visual examination, only the examination of surrounding area, including floor areas or equipment surfaces located underneath the components, need to be examined for evidence of leakage.

8.3.5 Examination of Insulated Components for Leakage

8.3.5.1 Visually examine accessible exposed surfaces and insulation joints without removal of the insulation, except as noted in 8.3.5.4.

NOTE: Vertical joints of insulation need only be visually examined at the lowest point, whereas, along horizontal joints the entire joint must be examined. Vertical is defined as $\geq$ 60 degrees where as horizontal is $\leq$ 60 degrees.

8.3.5.2 If leakage is detected, the examiner shall identify the actual source of leakage, which may include insulation removal.

8.3.5.3 For components whose external insulation surfaces are inaccessible to direct visual examination, only an examination of the immediate surrounding area, i.e., leakage may be channeled, shall be required.

8.3.5.4 Systems borated for control of reactivity shall have insulation removed from pressure retaining bolted connections prior to performing the Visual Examination.

8.3.6 Examination of Components with Leakage Detection Systems

8.3.6.1 For components where leakage is expected, i.e., valve stems, pump seals, etc., visual examinations shall be conducted by verifying that the leakage collection system is operative, and that the collection system is capable of channeling the leakage as observed.

8.3.7 Examination of Buried Components for Evidence of Leakage

8.3.7.1 Nonredundant systems shall be examined for leakage, after having isolated the system by means of valves in accordance with an approved test procedure.

8.3.7.2 For buried components surrounded by an annulus, the VT-2 visual examination shall consist of an examination for evidence of leakage at each end of the annulus and at low point drains.

8.3.7.3 For buried components where VT-2 visual examination cannot be performed, the examination requirement is satisfied by the following:

a The system pressure test for buried components that isolable by means of valves shall consist of a test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components. The acceptable rate of pressure loss or flow shall be established by the owner.

b. The system pressure test for nonisolable buried components shall consist of a test to confirm that flow during operation is not impaired.

c. Test personnel need not be qualified for VT-2 visual examination.

8.3.8 Examination of Pneumatic Components for Evidence of Leakage

8.3.8.1 When performing examinations on pneumatic systems or components for leakage, a direct and/or remote examination shall be performed.

8.3.8.2 A supplemental examination using the "UE" Ultrasonics Leak Detection Unit shall also be performed.

8.3.8.3 Any leak(s) found by either the visual or the ultrasonic examination shall be verified using "SNOOP" Leak Detection Fluid.

9.0 RECORDING CRITERIA

9.1. Visual Examiners shall record the results of the examination on "Visual Examination for Leakage" (Figure 1 or similar).

9.1.1 If additional sheets are required to complete the examination record, a Supplemental Report Form as shown in NDE-203 shall be used.

9.2 If the examiner transfers the results from a working copy of the report form to a final copy of the report form, the working copy shall be attached to the final copy.

9.3 All supporting material,.(i.e. photographs, sketches, etc.) shall be attached to the report forms as required.

9.4 When recording a "Recordable Indication" or "Insignificant Indication," the examiner shall define the leak using definitive terminology.

9.4.1 Examples of definitive terminology are as follows:

a. Flowing water on pipe surface

b. Puddles on the floor

c. Intermittent dripping

d. Water collecting on underside of pipe

e. Water spraying from pipe surface

f. Extensive Bubbling of "SNOOP"

9.5 When possible, the examiner shall measure rate of leakage using a flow rate per minute.

9.5.1 Examples of flow rates are as follows:

a. cc per minute

b. drops per minute

c. quarts per minute

9.6 On systems that are borated, the examiner shall record boric acid crystal build-up, the color of the crystals that have built-up, and the overall dimensions of the crystal build-up, (i.e., "X" times "Y").

9.6.1 The exact location of leakage/boric acid residue/corrosion shall be described by referencing proximity to welds, valves, pipe supports or other components.

9.6.1.1 If the component is insulated the insulation shall be removed to determine the source of the leakage.

9.6.1.2 For corrosion, the depth and area shall be identified.

9.7 For pressure decay tests, the pressure from one gauge shall be recorded at least 15 minutes apart, in lieu of a visual examination.

9.8 For flow impairment, downstream flow rate or upstream and downstream flow rates shall be measured utilizing flow meters or other devices capable of measuring flow.

9.9 The volume or examination boundary shall also be identified on the examination record.

10.0 EVALUATION

10.1 The acceptance criteria, for each code related examination is located in

VT-101.

10.2 All completed examination reports shall be reviewed by a Level III or his designee.

11.0 RECORDS

11.1 Following review by a Level III, the RG&E NDE group shall notify cognizant individuals of examination results.

11.2 **Nuclear Work**

11.2.1 Following the review and approval by the Level III, completed originals of examination reports shall be submitted to Records Management, Ginna Station for retention under Record Category 15.

Visual Examination for Leakage

Report No ________

Site		Procedure:		Page	____ of ____
Summary No :		Procedure Revision		Date	
Examination For:		Work Order No.:		Time	
Applicable Code		ISO Drawing No.:		Location	

Description ________

Component ID ________

System ID ________

Limitations ________

Gray Card. 1/32" Direct ☐ 1/64" Remote ☐

Visual Equipment/Aids: ________

Examination Conditions: Non-Insulated ☐ Insulated ☐ Buried ☐ Inaccessible ☐

(These conditions shall be identified on the attached drawing)

Test Method/Hold Time:

System Leakage/Not Required ☐ System Functional/10 Minute ☐ System Inservice/4 Hours ☐

System Hydrostatic/10 Minute Non-Insulated 4 Hour Insulated ☐ System Pneumatic/10 Minute ☐

Conditions to be Verified

Pressure Gage No :		Cal Date:		Range		Pressure:	
Temp. Gage No		Cal Date:		Temp		°F	
Relief Valve No		Cal Date		Proper Valve Line Up. Yes ☐ N/A ☐			
Stop Watch No :		(If Used)					

Time Test Pressure Achieved ________ Exam Start Time ________ Exam End Time ________

Flowmeter Rate Upstream ________ Downstream ________

Pressure Decay Exam Hold Time ________ Pressure ________

Visual Examination

Observed Condition R I N I Insig N/A Insignificant/Comments/Leak Rate