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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards petition for order to show cause why facility OL should not be suspended or why permission to restart reactor should not be withheld until actions have been taken to assure protection of public safety.

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SIERRA CLUB

530 Bush Street San Francisco, California 94108 (415) 981-8634

Please reply to: 278 Washington Blvd.
Oswego, New York 13126

March 11, 1982



Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

Enclosed for filing is a Petition For Order to Show Cause prepared by the Sierra Club. The petition pertains to the Ginna Nuclear Power Plant, Docket No. 50-244, and arises from the January 25, 1982, accident. As staff review of the accident is already in progress, we request prompt response to our petition.

Very truly yours,

Ruth N. Caplan

Ruth N. Caplan, Chair
Sierra Club National Energy
Committee

Enclosure

cc. with petition:

Senator Gary Hart
Senator Alan Simpson
Congressman Morris Udall
Congressman Richard Ottinger
Congressman Edward Markey
Congressman Toby Moffett
Richard Goldsmith, Esq.
Karin Sheldon, Esq.
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Joseph Fontaine, President, Sierra Club
Eugene Coan, Sierra Club
Jesse Riley, Nuclear Subcom, Sierra Club
Richard Lippes, Chair, Atlantic Chapter
Beatrice Anderson, Chair, Rochester Group
Robert Pollard, Union of Concerned Scientists
John E. Maier, Rochester Gas & Electric

Y603
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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

| | | |
|---|---|---|
| In the Matter of<br>Rochester Gas and Electric Corporation<br>R.E. Ginna Nuclear Power Plant | ) | Docket No. 50-244 |

# SIERRA CLUB PETITION
# FOR ORDER TO SHOW CAUSE

## INTRODUCTION

This petition is broughtbefore the Office of Nuclear Reactor Regulation by the Sierra Club. Pursuant to 10 CFR 2.206, 50.54, 50.100 and 50.109, and for reasons set forth below, the Sierra Club requests that Rochester Gas and Electric Company be required to show cause, as provided in 10 CFR 2.202, why the operating license for the Ginna nuclear reactor in Ontario, New York, should not be suspended, or in the alternative, why permission to re-start the reactor should not be withheld, until such time as essential actions have been taken by the licensee and the Commission to assure the protection of public health and safety. The necessity for such actions arises from the accident on January 25, 1982, which was initiated by a steam generator tube break and which triggered a site emergency.

In requesting this action, the Sierra Club wishes to stress our concern regarding the potentially serious safety implications of the Ginna accident, not only to our 500 members living in Rochester, but also to the general public. Further, as a national environmental organization with approximately 225,000 members across the country and 18,000 members in New York State, we are concerned about the

implications of the Ginna accident for the safe operation of other pressurized water reactors in New York and across the country.

Given the clear safety implications of both under- and over-pressurization which can arise subsequent to a steam generator tube break, the Sierra Club concurs with the November 24, 1981, "Information Report--Steam Generator Tube Experience" by NRC staff which states:

> These tubes, like many interface components, affect both [primary and secondary] systems, and their failure is an operational as well as a potential safety concern. Therefore, the steam generator must be viewed as part of the total system in which it operates. Thus, maintaining the integrity of the tubes requires a systems approach that should encompass mechanical, structural, material, and chemical considerations. (page 35, emphasis added)

RELIEF REQUESTED

The Sierra Club requests that the Director of Nuclear Reactor Regulation initiate a full review by staff of matters pertaining to the ability of the licensee to safely operate the reactor so as to protect public health and safety, in light of the January 25th accident. Such review should be made part of the review now in progress by staff and should include, but need not be limited to, the specific areas detailed below. Pending completion of this review by the staff, the Operating License for Ginna should be suspended, or in the alternative, re-start of the reactor should not be permitted.

1. The cause of the tube break initiating the January 25, 1982, accident should be thoroughly explained and corrective action taken to prevent such breaks in the future. The mechanical damage arising from loose pieces of metal should be studied in the context of the generic corrosion problems at Ginna. Specifically, corrosion arising from AVT (all volatile treatment) control of secondary water chemistry should be addressed in relation to denting of tubes, stress

corrosion, and intergranular attack. This should include corrosion in the feedwater system and corrosive impurities introduced by condenser leaks.

2. The adequacy of the steam generator tube testing program should be evaluated and a determination made regarding the following issues:

a. Is the routine multi-frequency eddy current testing method being employed at Ginna the best available given current state-of-the-art? If not, what justification is there for not employing the best available technology, in light of chronic tube degredation problems at Ginna and at other PWR's and the existence of techniques such as fiber optic examination?

c. Does the current testing program, which only tests a sample of tubes and which does not test their full length, provide sufficient information to prevent tube failure?

3. The technical specifications defining the extent of allowable tube degradation for steam generator tube rejections should be reviewed in light of the Ginna accident to determine whether they are sufficiently stringent to prevent a tube break.

4. The increased risk of steam generator tube breaks/leaks, if RG&E operates the reactor without having proceeded with the preventative sleeving program originally scheduled for the Spring, 1982, refueling outage, should be assessed and a determination made as to whether the original schedule should be adhered to.

5. The safety implications of current and proposed plugging and sleeving of steam generator tubes and of further repairs such as insertion of stabilizing cables should be examined in order to assess additional stress, such as from changes in fluid dynamics, which may

be induced in tubes remaining in use.

6. An evaluation should be completed to determine the safety implications of operator action currently required to re-establish the instrument air system and to open the PORV manually.

7. The safety implications of the failure of the PORV to close should be assessed in light of the problems which developed during the Ginna accident, particularly with regard to the creation of a steam bubble in the reactor vessel as a result of depressurization. The potential for uncovering the core, due to a steam bubble in the reactor vessel or elsewhere in the primary system should be addressed. A determination should be made as to whether safety functions performed by the PORV require that it be designated as safety grade and be required to meet all NRC regulations applicable to such safety grade designation, in order to assure safe operation of the reactor.

8. A determination should be made, given the demonstrated unreliability of the PORV, as to whether a reliable method exists for removing decay heat by means of the secondary system, without providing, at the very minimum, one pathway for removing decay heat which consists of safety grade equipment. Such determination should also include an assessment of the reliability of essential auxiliary support systems such as instrument air, and should consider the consequences of loss of off-site power to determine whether General Design Criteria #17 of 10 CFR Part 50 Appendix A is met.

9. A determination should be made as to whether the emergency operator procedures set forth in "Westinghouse Emergency Operator Guidelines for Steam Generator Tube Rupture Events" are adequate to protect the public health and safety. Operator delay, or apparent hesitancy, in terminating the HPI (high pressure injection) is of particular concern in relation to the risk of over-pressurization

[illegible]

of the reactor pressure vessel as reported in the Speis memorandum (see infra #11) and to the increased reliance on proper functioning of steam generator safety valves. Further, the Ginna emergency procedures should be conformed to the Westinghouse guidelines.

10. The conditions under which the reactor vessel can become over-pressurized in the course of operator action to control an accident should be clearly specified and a determination made as to whether an automatic response system would decrease the chance of over-pressurization problems from developing and whether the installation of such a system at Ginna is an action that "will provide substantial, additional protection which is required for the public health and safety...." as provided in 10 CFR 50.109.

11. The concerns raised in the Speis memorandum (Themis Speis to Roger Mattson, "Preliminary Evaluation of Operator Action for Ginna SG Tube Rupture Event" dated January 28, 1982, see infra Attachment E) regarding problems and potential problems in cooling the reactor following the tube break should be addressed; a determination made as to their safety significance; and necessary corrective action taken. These include the following problems:

a. the apparent stratification in the B steam generator and its effect on slowing depressurization of the faulted steam generator;

b. the consequence of an additional coolant system failure, including a leak in the A steam generator or "a secondary side safety/relief valve" sticking open;

c. the necessity to remove decay heat from the A steam generator by steaming to the atmosphere due to improper functioning of the condensor;

d. the problems associated with the use of the PORV for coolant discharge during "feed and bleed" cooling.

12. A determination should be made as to the extent to which failure to implement the TMI Action Plan requirement for instrumentation to allow direct measurement of the water level in the reactor vessel contributed to operator problems in determining proper timing for operating the ECCS pumps and in determining the size of the steam bubble.

13. A full investigation should be made to determine the state of embrittlement of the Ginna reactor pressure vessel to determine the likelihood that over-pressurization will lead to vessel rupture as a consequence of pressurized thermal shock.

14. The NRC should determine whether the reactor can operate safely without replacement of the steam generator and associated parts of the nuclear steam supply system and whether the newest Westinghouse steam generator design will ameliorate the problems, given the recent problems which have developed with this design at McGuire and at European reactors.

15. The total projected worker exposure should be calculated in advance of NRC approval of RG&E's repairs and a specific plan developed to keep worker exposure as low as reasonably achievable (ALARA). This should include a determination as to whether time should be allowed for radioactive decay, particularly of Cobalt 58, in the steam generator prior to repairs, in order to prevent unnecessary worker exposure and still allow all necessary repairs to be made.

16. An overall safety assessment should be performed before the reactor is allowed to re-start in order that the combined risk of potential failure modes can be determined, in relation to the protection of public health and safety. At a minimum such an assessment should

address the following:

a. the degredation of the Ginna steam generators, including the plugging, sleeving and other repairs required to date and planned;

b. the on-going contribution to tube degredation of corrosion arising from AVT control, from condenser leakage, and from the feedwater system (as opposed to the suspected damage from loose pieces of metal in the B steam generator);

c. the lack of a safety grade pathway in the secondary system to remove decay heat;

d. the chance that operator error will lead to over- or under-pressurization of the reactor vessel;

e. the state of reactor vessel embrittlement.

The facts which constitute the basis for our request are set forth in Attachments A, B, C, D and E.

We respectfully request that a decision on our petition be rendered forthwith.

On behalf of the Sierra Club,

Respectfully submitted by,

Ruth N. Caplan

Ruth N. Caplan, Chair
Sierra Club National Energy Committee

278 Washington Blvd.
Oswego, New York 13126

315-343-2412

I hereby affirm that the facts alleged herein are true and correct to the best of my knowledge and belief.

Ruth N. Caplan

DATED: March 11, 1982

Ruth N. Caplan

AFFIDAVIT OF BEATRICE ANDERSON

1. My name is Beatrice Anderson. I live at 12 Spinet Drive, Rochester, New York 14625, which is about 20 miles from the Ginna reactor owned by Rochester Gas and Electric.

2. I am a member of the Sierra Club and I chair the Rochester Group of the Sierra Club which has 500 members in the Rochester area.

3. On behalf of myself and the Rochester Group, I authorize the Sierra Club to represent my interests in the request for show cause action before the U.S. Nuclear Regulatory Commission. These interests include the potential danger to my health and safety if the Ginna reactor is allowed to restart prior to such actions as are called for in the Sierra Club show cause request.

Beatrice Andersen

Sworn and subscribed to before me this 23rd day of February, 1982.

Edwin R Jeffries

Notary Public

EDWIN R. JEFFRIES JR.
Notary Public in the State of New York
MONROE COUNTY, NEW YORK
Commission Expires March 30, 1982

My commission expires ____________.

ATTACHMENT A. FACTUAL BASIS FOR SHOW CAUSE PETITION

1. On January 25, 1982, a steam generator tube rupture at the Ginna nuclear plant in Ontario, New York, occurred. The rupture occurred in a tube which was last inspected in May, 1981, at which time the tube showed less than 20% wasting of the tube wall, according to "Weekly Information Report, February 18, 1982, from T.A. Rehn, Assistant for Operations Office of the EDO to the Commissioners", included herein as Attachment B.

2. It is our understanding that RG&E has not yet been able to provide a satisfactory explanation for the rupture of the steam generator tube. Upon information and belief, a clear relationship has not been established between loose pieces of metal discovered in the steam generator, the damaged peripheral tubes, and the ruptured tube. An alternate explanation linking the rupture to stress corrosion has been advanced by RG&E. (See Rehm memo, page 2 of Enclosure B)

3. Upon information and belief, the Ginna tube testing program has been based on multi-frequency eddy current testing at the time of refueling. Such testing has included only a sample of tubes and only part of the tube length has been examined. According to Nuclear Safety, "most tubes were tested to the first support plate, some to the sixth support plate, and a few over the U-bend." (Nuclear Safety, V. 22, N. 5, Sept.-Oct., 1981. Included infra as Attachment C.)

4. Upon information and belief, the "Quality Assurance Manual, Ginna Station--Inservice Inspection Program for the 1980-1989 Interval" allows the tube inspection interval to be extended to once every 40 months under certain conditions. Section 2.5 of this document states:

> The inservice inspection intervals for the examination of steam generator tubes shall not be more than 24 months. However, if over a nominal two year period (e.g., two normal fuel cycles) at least two examinations of the separate legs result in less than 10% of the tubes with detectable wall penetration (> than 20%) and no significant (> than 10%) further penetration of tubes with previous indications, the inspection interval of the individual legs may be extended to once every 40 months. (page 5 of 22)

5. Upon information and belief, RG&E reported to the NRC staff on February 10, 1982, that tests after the accident did not reveal serious problems with the steam generator tubes which would prevent RG&E from re-starting the reactor. Yet After fiber optic examination was required by staff, serious problems were found in tubes previously plugged. John Maier, RG&E Vice-president for Electric and Steam Generation, commented to the press the next day: "The pictures are very dramatic.... It looks like somebody went in with a hacksaw. Some of the tubes show severe denting and external degredation." (AP quoted in Palladium-Times, Feb. 12, 1982) Further examination revealed two pieces of metal weighing "'a couple of pounds'...with one of them as large as 6.5 x 4 inches and seven-sixteenths inches thick." (Nucleonics Week, Feb. 18, 1982) As reported in Nucleonics Week, Feb. 25, 1872, one RG&E source stated: "'Some are corroded, some are imploded, some are just sheared.'"

6. Upon information and belief, RG&E was planning an extensive sleeving program to remedy corrosion problems regarding the steam generator tubes. In a letter from John Maier to Dennis Crutchfield, January 15, 1982, RG&E requested permission to "delete the 25 sleeve limitation" so that more sleeves could be installed during each steam generator inspection. (See infra, Attachment D.)

7. As recently as September 21, 1981, Ginna was not listed as one of the 11 units with the most serious steam generator problems (New York Times, Sept. 21, 1981, B-10). It is our opinion that this fact emphasizes the unpredictable nature of the rupture and reinforces the need for much more stringent test procedures.

8. Upon information and belief, the introduction of AVT control of secondary water chemistry at Ginna has led to problems of intergranular attack and tube corrosion, requiring the plugging of steam generator tubes. (Nuclear Safety, Ibid.)

9. As indicated in the Point Beach proceedings, AVT control does not function to precipitate out solid impurities that leak into the generator and does not prevent build-up of hardness scale on the heat transfer surfaces. Both conditions degrade steam generator tubes. (Docket 6630, ER-10, Exhibit 16E at 14-15)

10. As observed by NRC staff, "denting" of steam generator tubes occurred in several PWR facilities, including Turkey Point, Units 3 and 4, and Surry, Units 1 and 2, after 4 to 14 months of operation, following the conversion from a sodium phophate treatment to AVT chemistry for the steam generator secondary coolant. ("Information Report--Steam Generator Tube Experience, November 24, 1981, SECY 81-664," Appendix B, page 3.) We note hte report's observation that: "Tube denting is most severe in the rigid regions or so-called 'hard spots' in the tube support plates. These hard spots are located...around the peripheral locations of the support plate where the plate is wedged to the wrapper and shell." (Ibid., page 3) Upon information and belief, the staff has already requested that RG&E have Westinghouse prepare a report regarding this matter.

11. The NRC "Information Report--Steam Generator Tube Experience" concludes: "copper alloys should be eliminated from all areas of the condensate/feedwater/steam condensation cycle. Substantial evidence exists that copper oxides in the steam generators are an important catalyst in accelerating the rate of corrosion processes within the steam generators." (Ibid., p. 42)

12. Condenser leakage is also relevant to the action at hand. Staff states: "With the exception of a few reactors which are sited where no acid producing species exists in the condenser cooling water, all currently operating plants are susceptible to denting, if sufficient condenser leakage occurs. Because copper oxide has been demonstrated to be a catalyst, those plants with copper in their secondary cycles are even more susceptible." (Ibid., Appendix A, page 6)

13. Steam generator problems are not automatically solved by installing new steam generators as evidenced by the problems faced by Prairie Island 2 and by North Anna 1. Brookhaven National Laboratory commented

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

last year as follows:

> It seems ironical that Prairie Island 2, which has no copper in the system, stainless steel condensers, and meticulous monitoring of water chemistry, should be the one unit to have suffered from this particular phenomenon (of tube corrosion): the Prairie Island Units have to date been a shining example of what we thought was the proper way to avoid corrosion problems.
>
> (Docket 6630, CE-20, Exhibit 40, p.3)

Such experiences make it all the more imperative to have a stringent testing schedule for tubes and strict standards for removing tubes from service.

14. Upon information and belief, the sequence of events during the January 25 accident clearly indicate the interdependency of the nuclear steam supply system and the reactor safety system. Reactor trip in response to the tube break initiated containment isolation which resulted in loss of instrument air. This required operator action to open the PORV manually, when the valve was required to relieve over-pressurization. The reactor vessel became under-pressurized when the PORV stuck open and the block valve had to be closed. Lowered pressure produced a steam bubble in the top of the reactor vessel when water flashed to steam. A second drop in pressure about 30 minutes later again led to water in the reactor vessel flashing to steam. (Source: "Preliminary Evaluation of Operator Actions for Ginna SG Tube Rupture Event" by Themis Speis. See infra Attachment E.)

15. Upon information and belief, the Speis memo also indicates that over-pressurization of the reactor vessel was of concern during the sequence of events during which operators tried to stabilize the reactor. First, charging pumps were restarted before the B steam generator was isolated, leading to a build-up of reactor pressure. Second, the SI pump was restarted without apparent need to do so, which has elicited concern regarding operator hesitance to terminate HPI and the consequence for pressurized thermal shock.

16. According to the "Information Report--Steam Generator Tube Experience," the total man-rems exposure can be quite significant. The report states: "Where major repair or replacement efforts are required, dose expenditures may range from 2000 to 3500 man-rems." (Ibid, page 51) The largest dosage reported results from steam generator repair at San Onofre Unit 1, where 3493 man-rems exposure is reported for the 273-day outage during 1980-1981. (Ibid, Table 6) This is more than the 1759 man-rems for steam generator replacement at Surry, Unit 1 or the 2140 man-rems for Surry, Unit 2 replacement. (Ibid., Appendix B, page 13 and Table 6) It is our belief that these dose levels point to the need to evaluate total man-rems exposure in determining the best course of action to be followed at Ginna.

DDPP ATTACHMENT B

February 18, 1982

For: The Commissioners

From: T. A. Rehm, Assistant for Operations, Office of the EDO

Subject: WEEKLY INFORMATION REPORT - WEEK ENDING FEBRUARY 12, 1982

A summary of key events is included as a convenience to those Commissioners who may prefer a condensed version of this report.


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T. A. Rehm, Assistant for Operations
Office of the Executive Director
for Operations

*No input this week.

Contact:
T. A. Rehm, OEDO
49-27781

FOR SUBSCRIBERS ONLY

R. E. GINNA

1. The ruptured tube in the Ginna steam generator was inspected in May 1981. The ECT results showed that there was 20% penetration (an OD signal) 3 to 6 in. above the tubesheet. The failed tube is in row 42, column 55 which is near the periphery of steam generator. It is located in the "wedge area" of the steam generator. This is the section of the support plates that does not have flow holes. Three of the six previous small leaks that have been experienced have been in that area. There is no sludge pile in that area.

2. The ruptured tube in the Ginna steam generator has been inspected using fiber optics. The rupture has been determined to start approximately 2 to 3 inches above the tube sheet and is approximately 5 inches long. The rupture is kite-shaped with a maximum width of 3/4 to 7/8 inch. RG&E postulates that the rupture was due to stress corrosion linked to differential expansion between the tube and the tube wrapper in the wedge region (a region where the tube support plate is fastened to the wrapper). Profilometry, to determine bulging or unusual shape of the tubes, showed some bowing of the tubes in the area of the rupture, thus adding credance to this theory. The ruptured tube is being plugged and removal of the tube is not anticipated due to its location in the tube bundle. RG&E is planning to use fiber optics to inspect the ruptured tube from the secondary side.

Eddy current testing (ECT) of the "B" steam generator has been completed. In addition to the ruptured tube, twenty other hot leg tubes are scheduled to be plugged. Three of the tubes are adjacent to the ruptured tube while the others are tubes nonrelated to the accident that indicate intergranular attack (IGA) or >40% degradation. No plugging, other than the ruptured tube, is planned for the cold leg of the "B" steam generator. RG&E has commited to ECT 100% of the "A" steam generator hot leg tubes plus all periphery tubes and a random sample of the sludge area tubes in the cold leg of the "A" steam generator.

R. E. GINNA (Cont'd)

3. On Wednesday, February 10, 1982, members of the NRC staff met with representatives of Rochester Gas and Electric Corporation (RG&E) to discuss the requirements to be met prior to restart of the R. E. Ginna Nuclear Power Plant. RG&E had scheduled the startup of Ginna for Monday, February 15, 1982, and proposed operation until the scheduled May 15, 1982 refueling outage, at which time the plant would be shut down and eddy current tests (ECT) of the steam generators (S/G) would be performed. RG&E presented information on the cause and corrective action for the tube located in the wedge area that ruptured. In addition, there was a description of Power Operated Relief Valve (PORV) modifications and discussion of emergency procedures.

   RG&E has performed extensive ECT of both S/Gs. The ruptured tube has been inspected using fiber optic equipment and a videotape of the rupture was shown at the meeting. Fiber optic inspection of the secondary side of the "B" S/G is in progress. In addition to the failed tube, the licensee has plugged 20 additional tubes in the "B" S/G because of intergranular attack or wastage indications.

   The staff has concluded that there was not sufficient technical basis presented at this time to permit the Ginna plant to return to operation. Specifically, the staff felt that prior to restart RG&E should:

   1. Finish the fiber optic inspection of the secondary side of "B" S/G including inspection for loose parts;

   2. Obtain the S/G designer's opinion on the effects of plugging in the wedge area; and

   3. Provide a more complete basis for operating for the proposed 3 months.

   A meeting to discuss the remaining areas will be set up when RG&E has prepared their response.

ENCLOSURE B

ATTACHMENT C

### Farley 1, USA

One leaking tube was plugged at Farley 1. The defect was located at the U-bend, but the cause of failure was not determined.

Eddy-current inspection was performed on 153 tubes in steam generator A and 306 tubes in steam generator C, where the leaking tube was located. Remote television inspection was used to augment eddy-current testing.

### Ginna, USA

Nineteen tubes, all in steam generator B, were plugged at Ginna during 1979. Thirteen of the tubes had indications of intergranular attack in the tube sheet crevice; two tubes showed wall thinning just above the tube sheet. Tube corrosion by intergranular caustic SCC is typical of steam generators with a long tube sheet crevice. At Ginna, these failures have occurred every year since 1975, the year after introduction of AVT control of secondary-water chemistry. The wall thinning at support plates 1 and 2 was thought to be caused by water flashing to steam in the annulus during the early years of operation. These annuli are now packed with corrosion products. Other tubes have this type of defect, but the thinning is <20% of the tube wall thickness. The wastage defects are thought to be caused by a hydraulic–mechanical mechanism rather than corrosion because all affected tubes are in the periphery of the bundle where sludge does not normally accumulate.

Tube inspection was performed by multifrequency eddy-current testing, as in 1977 and 1978. The inspection pattern was similar to that of 1978: most tubes were tested to the first support plate, some to the sixth support plate, and a few over the U-bend. About 2000 tubes were tested in each steam generator, with a 5 : 1 ratio between the hot and cold legs.

Ginna was the first PWR station with recirculating steam generators to use full-flow deep-bed condensate demineralization in the United States.[9] Very good experience has been reported with steam generator water chemistry control and with the operation of the demineralizer system.

### Indian Point 2 and 3, USA

Twenty-six steam generator tubes were plugged at Indian Point 2 because of reduced tube diameter at the support plates. These defects were found by eddy-current inspection of 1519 hot-leg tubes.

Denting, a phenomenon caused by ingress of chloride leading to acid-forming conditions, results in nonprotective corrosion product deposition in tube-to-tube-support annuli in steam generators with drilled-hole carbon steel support plates. It has been postulated that the addition of boric acid to secondary water mitigates denting by forming stable, protective iron borates. This treatment is now being used at Indian Point 2.

Of the 437 tubes plugged in the four steam generators at Indian Point 3, denting defects were observed in 69 tubes at support plate intersections. Because denting causes inward distortion at the support plate, giving rise to the potential for SCC at the small-radius U-bends, all tubes in row 1 were plugged (368 tubes).

The steam generators at Indian Point 3 were inspected by techniques commonly used at plants with significant denting. This includes using eddy-current probes of different diameter and photographing the secondary side to measure distortion of flow slots. The sludge deposit on the tube sheet was found to be soft, and it was estimated that ~92% could be removed by lancing with water. Boric acid is added to steam generators during condenser leakage.

### José Cabrera, Spain

Three tubes were plugged because of fretting at the antivibration bars, and one was plugged because of phosphate wastage just above the tube sheet. Only seven tubes have been plugged in the José Cabrera steam generator in 2915 EFPD of operation with phosphate treatment of secondary water, and six of these failures were caused by fretting at the antivibration bars.

Multifrequency eddy-current testing was used to inspect 80 tubes at the U-bend and almost all tubes to the first support plate. Phosphate wastage of 40 to 49% of the tube wall was detected in six tubes (including 1 that was plugged), and wastage of 30 to 39% was detected in 46 tubes. This is the first reported instance of phosphate wastage at José Cabrera.

### KKS Stade, Federal Republic of Germany

Eddy-current inspection of 574 tubes in steam generator 1 and 1262 tubes in steam generator 2 showed that three tubes in steam generator 1 and 56 in steam generator 2 had phosphate wastage of <25% of the tube wall. Two tubes were removed for metallurgical examination. Stade, like Borssele, has Incoloy 800 tubes and has used low-phosphate treatment (2 to 6 mg