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ComEd

February 5, 1996
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: LaSalle County Station Units 1 and 2
Response to Notice of Violation
NRC Docket Numbers 50-373 and 50-374.

Reference:
1. Letter from Mr. G.G. Benes NLA to the U.S. Nuclear Regulatory Commission Mr. William T. Russell dated October 17, 1994. LaSalle County Station Units 1 and 2 - Second Ten Year Inspection Interval for the InService Inspection (ISI) and InService Testing (IST) Programs. NRC Docket Nos. 50-373 and 50-374.

2. Letter from Mr. R. Latta NRC to Mr. D. Farrar dated September 1, 1995. LaSalle County Station Units 1 and 2 - Request for Additional Information (TAC Nos. M90704 and M90705).

3. Letter from Mr. Robert E. Querio LaSalle County Station to NRC dated October 4, 1995. Request for Extension to Respond to RAI on Second Ten-Year Interval InService Inspection Program Plan.

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The enclosed attachment provides the LaSalle County Station (LSCS) response to the Request for Additional Information dated September 1, 1995. The format of this LSCS response retains the numbering sequence of the reference 2 letter. The second 10-year ISI Plan document has been revised to incorporate additional information, revise relief requests, and correct Code Category populations.

If there are any questions or comments concerning this letter, please refer them to me at (815) 357-6761, extension 3600.

Respectfully,

R. E. Querio
Site Vice President
LaSalle County Station

cc: H. J. Miller, Regional Administrator, Region III
M. D. Lynch, Project Manager, NRR
H. J. Simons, Acting Senior Resident Inspector, LaSalle
D. L. Farrar, Nuclear Regulatory Services Manager, NORS
Central file

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A. **Provide a list of the components subject to examination during the second 10-year interval; include a list of Code Class 1, 2, and 3 piping and components that have been exempted from examination and the basis for their exemption.**

As requested, LaSalle has included in this response tables by Code Category of the components subject to examination in the second 10-year interval. The tables include notes which describe the components which are exempt from these examinations. Also included are Component Boundary Drawings (CBDs) which have been color coded to reflect Code Class as well as exemption boundaries. Isometric drawings showing the system, location and configuration of the components to be examined are also included. The component tables, CBDs, and isometric drawings are provided to allow the staff to continue their review. These items will continue to be reviewed and updated as the second 10-year interval progresses. Changes to these items will not be formally docketed with the Staff. If future updates or corrections to these items result in changes to the formally docketed ISI Plan, the Staff will be made aware of the changes via the appropriate method in a timely fashion.

B. **Provide a list of the ultrasonic calibration standards being used during the second 10-year interval at LaSalle County Station, Units 1 and 2. The list should include the calibration standard identifications, material specifications, and sizes, as well as a reference to the piping and/or components to which the calibration standards apply.**

The requested information is included as Tab # 2 located directly after the second 10-year ISI Plan document.

ATTACHMENT

C. Address the degree of compliance with augmented examinations that have been established by the NRC when added assurance of structural reliability is deemed necessary. Examples of documents that address augmented examinations are:

1. Branch Technical Position MEB 3-1, High Energy Fluid Systems, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment;

2. Regulatory Guide 1.150, Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations;

3. NUREG-0619, BWR Feedwater Nozzle and CRD Return Line Nozzle Cracking;

4. NUREG-0803, Integrity of BWR Scram System Piping; and

5. Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping" (Reference NUREG-0313).

Discuss these and any other augmented examinations that may have been incorporated in the LaSalle County Station, Units 1 and 2, Second 10-Year Interval Inspection Program Plan.

Compliance with the document of C.1, above has been addressed in the LaSalle responses to NRC Questions 111.72 and 111.81 in the Final Safety Analysis Report which has been inserted verbatim, as follows:

Response to Question 111.72

The augmented inservice inspection program described in this response will be implemented on LSCS. In order to clarify those requirements that are above and beyond those of Section XI, the following summary is presented:

ATTACHMENT

a. Table IWB-2500 Category B-J of Section XI requires that only 25% of the circumferential joints and pipe branch connection joints be examined during each inspection interval. The augmented program requires 100% examination of all weld joints in the break exclusion area during each inspection interval.

b. Table IWB-2500, Item numbers B4.7 and B4.8 (Summer of 1975 Addenda), for Category B-J, only require surface examinations of branch pipe connection welds 6-inch diameter and smaller. The augmented program requires volumetric examinations of these areas. Ultrasonic inspection of socket welds doesn't yield definitive results. Since surface penetrant examinations positively indicate flaws for socket welds, the LaSalle inspection of socket welds will utilize this appropriate technique.

c. Article IWC-2000 of Section XI requires that Categories C-F and C-G receive 100% volumetric examination of all weld joints only once during the plant life, with partial examinations being performed at each inspection interval to assure all welds are examined by the end of the plant life. The augmented program requires that all weld joints in the exclusion area be examined during each inspection interval.

The preceding summary represents the full extent of the augmented inservice inspection program for break exclusion boundary piping above and beyond that required by ASME Section XI.

Response to NRC Question 111.81

The following summary is provided to place this inquiry in proper perspective:

a. The LSCS design basis for postulating and evaluating pipe breaks inside and outside containment is Regulatory Guide 1.46 and the A. Giambusso letter of December 15, 1972, respectively, as stated in Section 3.6 and Appendix C. However, since theses NRC documents provided no guidance for a break exclusion region, that stated in J. F. O'Leary letter of July 12, 1973 was utilized. This criteria required only that the piping be "conservatively reinforced and restrained beyond the valve such that loads will neither impair the operability of the valve nor the integrity of the piping or the containment penetration."

b. Subsequently, additional NRC requirements for the break exclusion area were obtained via the NRC Questions of April 3, 1975 on PSAR Special Report Number 3. These were implemented as described in response to Question 111.45.

c. In Question 111.72, the NRC attempted to impose the additional requirements of Branch Technical Position APCSB 3-1 via the reference to Standard Review Plan 3.6.2 which expanded on the augmented inservice inspection requirements for the break exclusion area. The response to Question 111.72 demonstrated compliance to BTP APCSB 3-1 and BTP MEB 3-1, which required only that the requirements of Section XI be extended to provide 100% volumetric examination during each inspection interval.

d. The current question attempts to impose additional requirements on the break exclusion boundary piping, beyond those of Section XI or the BTP's, by requiring that 3-inch and smaller and Class 1 piping be subjected to volumetric examination as part of an ever increasing augmented inservice inspection program. Although Section XI is consistently referenced throughout the NRC SRP's and BTP's, this question states that the augmented inservice inspection program is not based on Section XI requirements.

Thus, from a design basis commitment that required simply a conservative design on the break exclusion boundary piping, the requirements have proliferated during the licensing review to a point beyond the NRC's own Standard Review Plans and Branch Technical Positions. Such racheting indicates incomplete or poorly defined NRC treatment of the new subject and a penchant for trivia of no significance to public health and safety.

ATTACHMENT

Only two Class 1 lines on LSCS could be affected by this new requirement, a 3-inch main steam drain line (Penetration M-22) and the 1 1/2 inch standby liquid control discharge line (Penetration M-34).

The standby liquid control line is a moderate energy line for the portion that penetrates primary containment; and therefore, circumferential breaks need not be considered. The maximum postulated through-line leakage crack which could be assumed for this line is comparable in area to the area of the restricting orifices in the reactor coolant pressure boundary instrument lines. Therefore, the analysis to establish whether the plant could be safely shutdown within 10CFR 100 guidelines for this standby liquid control line is unnecessary because it is bounded by the Instrument Line Failure Analysis already presented in Subsection 15.6.2 where the offside thyroid dose prediction in less than 3x10-4 rem for an instrument line failure. In addition, this piping is socket welded which cannot be examined by ultrasonic methods.

In conclusion, because the main steam drain line is the only break exclusion boundary line at LaSalle which is actually affected by the imposition of this new criteria, and in order to expedite the licensing review, it is agreed that the exemption referenced in the response to Question 111.72 will be deleted. This exemption will not be sought on break exclusion boundary piping. A revised response to Question 111.72 is included, deleting the reference to the exemption stated in paragraph IWB-1220(b) (1) of Section XI.

The specific welds augmented into the Inservice Inspection Plan for the initial 10-year interval will continue to be augmented in the second 10-year interval. The specific welds involved are designated in the tables supplied with this response as Type 1A Augmented Inspections.

ATTACHMENT

Compliance with the document of C.2, above,

Ultrasonic examinations of the LaSalle County Station, units 1 and 2 Reactor Pressure Vessel welds have been completed in accordance with procedures that incorporate the recommendations contained in Regulatory Guide 1.150 Revision 1. ComEd Special Processes Procedure Manual Volume IV, "InService NDE" procedure NDT-C-30 for manual examinations, and complementary contractor NDE procedures for automated examinations have been and will continue to be used to perform ultrasonic examinations of the Reactor Pressure Vessels.

Compliance with the document of C.3, above,

LaSalle County Station, Unit 1 and 2 is committed to NUREG-0619 for the second 10-year inspection interval. The extent of these augmented examinations includes ultrasonic examination of the Inner Radius areas and bores of the Feedwater nozzles at every other refueling outage. LaSalle also performs VT-1 examinations of all Feedwater Spargers at each refueling outage. Cladding has been removed from all Feedwater nozzle inner radius areas at LaSalle, units 1 and 2.

The CRD Return Line Nozzles at LaSalle County Station, Units 1 and 2, have been cut and capped since original construction. The nozzles continue to be examined in accordance with ASME Section XI requirements. The CRD nozzle butters are also examined for indications associated with IGSCC in accordance with the requirements of Generic Letter 88-01.

Compliance with the document of C.4, above,

NUREG-0803 (3.1.2, 5.1) requires that SDV piping, because of its importance in achieving the scram function, should, as a minimum, be subjected to the ISI requirements of ASME Section XI for Class 2 piping. The LaSalle County Station response to NUREG-0803, LaSalle Report Regarding Integrity of Scram System Piping, dated January 21, 1982, stated that the inservice inspection of the scram discharge volume piping would be addressed the Inservice Inspection program.

ATTACHMENT

The LaSalle County Station, Units 1 and 2 - Second 10-Year Inspection Interval for Inservice Inspection Program, Section 3, isometric drawings, and Control Rod Drive Hydraulic CBD Drawings ISI-M-100 and ISI-M-146 (included) identify Class 2 inspections per Section XI, IWC-2500-1, Category C-H, Note 7, for both Category B (piping-SDV lines, valves-SDV lines and insert & withdraw lines from the drive flange up to and including the first valve on the hydraulic control unit) and D (all others) components. Therefore, the augmented examination would consist of the Category C-H components identified as being examined in accordance with IWC-2500-1, Category C-H, Note 7.

Compliance with the document of C.5, above,

In conjunction with NUREG-0313, Rev.2, Generic Letter 88-01 requires an Inservice Inspection (ISI) Program to be implemented for austenitic stainless steel piping covered under the scope of this letter that conforms to the staff position on inspection schedules, methods and personnel, and sample included in this letter.

In response to Generic Letter 88-01 dated January 25, 1988, Commonwealth submitted a letter to the U.S. Nuclear Regulatory Commission on July 27, 1988 for Dresden, LaSalle, and Quad Cities Stations. On page 8 of this letter, LaSalle Unit 1 made the following commitment:

Beginning with the next refueling outage for Lasalle Unit 1 (December, 1989), ASME Class 1, 2, and 3 piping made of stainless steel that is four (4) inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation will be subject to an augmented inspection program. This augmented inspection program will conform to the NRC staff positions on inspection schedules, methods, personnel and sample expansion delineated in Generic Letter 88-01. The

Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods and personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

D. Define the systems or portions of systems that provide Residual Heat Removal (RHR), Emergency Core Cooling (ECC), and Containment Heat Removal (CHR) functions at LaSalle County Station, Units 1 and 2, and provide a list of the subject welds that have been excluded from selection based on wall thickness as allowed by Table IWC-2500-1. From this list, identify those welds that will be scheduled for examination to provide an appropriate sampling of excluded examination areas.

Note: Paragraph 10 CFR 50.55a(b) (2) (iv) requires that appropriate ASME Class 2 piping welds in the RHR, ECC, and CHR systems shall be examined. Portions of these systems should not be completely omitted from inservice volumetric examination based on Section XI criteria (piping wall thickness) specified in Table IWC-2500-1. The staff has previously determined that a 7.5 percent augmented volumetric sample of thin-walled welds constitutes an acceptable resolution at similar plants.

The **Residual Heat Removal System (RHR)** restores and maintains coolant inventory to adequately cool the core after a LOCA. It provides containment cooling to assure condensation of the steam blowdown of a LOCA. The RHR system also supplements Fuel Pool Cooling and with RCIC removes decay heat from the reactor when isolated from the main condenser. **Reference paragraph 5.4.7 of the Final Safety Analysis Report (FSAR).**

The **Emergency Core Cooling System (ECC)** in conjunction with the containment, is to limit the release of radioactive materials following a loss-of-coolant-accident so that resulting radiation exposures are within the guideline values given in published regulations. The ECC system consists of a **high-pressure core spray (HPCS) system, a low-pressure core spray (LPCS) system, a**

low-pressure coolant injection (LPCI) system (three loops), and an automatic depressurization system (ADS). Reference paragraph 6.3 of the FSAR.

The **Containment Heat Removal System** function is accomplished by the containment cooling mode of the RHR system. The system is also equipped with spray headers in the drywell and suppression chamber areas. However, no credit was taken for these spray headers for either heat removal or fission product control following a LOCA. **Reference paragraph 6.2.2 of the FSAR.** NUREG-0519, LaSalle Safety Evaluation Report, Supplement No. 3 identifies the following systems as performing containment heat removal functions.

Residual Heat Removal
(Reference FSAR 5.4.7)

CSCS-ECWS
(Reference FSAR 9.2.1)

Auxiliary Power System *
(Reference FSAR 8.1.2.1)

Diesel Fuel Oil System
(Reference FSAR 9.5.4)

Diesel Generator System
(Reference FSAR 9.5.4)

Control Room HVAC *
(Reference FSAR 6.5.1)

Diesel Generator Room Vent *
(Reference FSAR 9.4.5.1.2)

Auxiliary Electrical Equipment Room Vent *
(Reference FSAR 9.4)

Switchgear Heat Removal *
(Reference FSAR 9.4.5)

CSCS-ECWS Vent System *
(Reference FSAR 9.2.1)

* These systems are not classified as Category A, B, C, D, or D+. LSCS UFSAR, Table 3.2-1 indicates "NA"

The **Primary Containment Chilled Water System (VP)** provides chilled water to the primary containment fan-coil units to meet the cooling load requirements in each drywell. A separate system is used for each unit drywell. Primary Containment Chilled Water is not safety related. **Reference FSAR 9.2.9.**

The tables by code category included with this response identify the category C-F-1 and C-F-2 weld populations. These listings include welds excluded from selection based on piping wall thickness per Table IWC-2500-1. The excluded components are noted in the remarks column of each respective table. The excluded components are exempted from the examination requirements of IWC-2500-1 by the rules of IWC-1221(f). Therefore, they do not contribute to a sample examination of any percentage.

E. Provide the staff with the status of the augmented reactor pressure vessel (RPV) examinations required by new regulations issued September 8, 1992, and provide a technical discussion describing how the regulation was/will be implemented for these welds at LaSalle County Station. Include in the discussion a description of the approach and any specialized techniques or equipment that was/will be used to complete the required examination.

Note: Effective September 8, 1992, new regulations were issued regarding augmented examination of reactor vessels. As a result of these regulations, all licensees must augment their reactor vessel examinations by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A of the 1989 Code. In addition, all previously granted relief for

Item B1.10, Examination Category B-A, for the interval in effect on September 8, 1992, is revoked by the new regulation. For liscensee's with fewer than 40 months remaining in the interval on the effective date, deferral of the augmented examination is permissible with the conditions stated in the regulation.

During the initial 10-year inspection intervals at LaSalle County Station, Units 1 and 2, examinations of the RPV shell welds of Item B1.10, Category B-A were completed in accordance with the 1980 Edition with Winter of 1980 Addendum of Section XI of the ASME Boiler and Pressure Vessel Code. The examinations were completed prior to the effective date of the new regulation in the Spring of 1991 for Unit 1, and the Spring of 1990 for Unit 2. During this, the 2nd 10-year inspection interval the examinations will be completed in accordance with the 1989 Edition of Section XI.

The examinations were/will be completed utilizing automated scanning and data analysis techniques from the outside surface (O.D.) of the RPV. Due to the size of the annular space between the RPV shell O.D and the Sacrificial Shield wall, the automated scanner is able to interrogate a large percentage of the length of each weld. The balance of each weld is then completed using manual techniques. This is normally only necessary in areas where circumferential and longitudinal welds are in close proximity to RPV nozzles. Since this automated system utilizes no track or pole system, it results in high quality examinations, and large reductions in personnel exposure.

F. Regarding Request for Relief No. CR-12, which deals with reactor vessel closure stud examination requirements, address LaSalle County Station's compliance with Appendix VI and provide assurance that the enhanced volumetric technique provides an equivalent sensitivity to that of the Code-required surface examination.

(Although a discussion on the use of an enhanced ultrasonic examination technique was provided, it appears that the enhanced technique is being used in lieu of removal of additional studs

when surface examinations of removed studs reveal flaws that exceed acceptance standards. It should be noted that the applicable requirement for volumetric examination of studs is in Appendix VI of the 1989 Edition of Section XI. This Appendix requires that the volumetric technique and personnel be qualified for examination of studs.)

It should be noted at this point that the Staff's RAI contains a typographical error regarding this Request for Relief. The RAI requests information for Request for Relief No. CR-12. The subject matter for CR-12 (inaccessible component supports) is not consistent with the additional information requested. LSCS will respond based on the assumption that the staff is pursuing additional information on Request for Relief No. CR-07.

It is the intention of LSCS to comply with the requirements of the 1989 Edition of Section XI including Appendix VI for the enhanced UT of the RPV closure studs in Units 1 and 2. The enhanced end-shot UT will be completed at each refueling outage of Unit 1 and 2 in accordance with Volume IV of the ComEd Special Process Procedures Manual, procedure NDT-C-50, Ultrasonic Examination of Reactor Head Studs for Dresden, LaSalle County, and Quad Cities Stations. This procedure incorporates both the previously described enhanced end-shot technique, and the requirements of the 1989 Edition of Section XI including Appendix The subject volumetric examinations using the enhanced end-shot technique and incorporating the 1989 Section XI requirements have most recently been completed during the Spring of 1996 seventh refueling outage of LaSalle Station Unit 1. This was the first refueling outage, of the first inspection period, in the second 10-year interval of Unit 1. The examinations were completed using personnel, a calibration standard, and examination technique which incorporates the enhanced end-shot qualities and Section XI Appendix VI requirements.

The intent of Request for Relief No. CR-07 is to alleviate the requirement to remove additional studs for surface examination if surface examinations of the 1/6 original sample of removed studs reveals flaws that exceed acceptance standards. The methodology that will be employed is as follows:

- 100% of the studs will be UT'ed each outage using the techniques and procedures described above.
- If practical, a 1/6 sample of the studs will be removed and subjected to surface examination.
- If a flaw is revealed during the surface exam, the flaw will be sized utilizing a bore probe.
- If bore probe sizing of the flaw indicates that the flaw is deeper than the documented 0.3" detection limit of the enhanced end-shot, then the required sample expansion will be completed using the bore probe as opposed to the Code required surface exam.

Based on the enhanced UT technique, and adherence to the requirements of Section XI Appendix VI provided by this methodology, it is believed that the imposition of any Code required sample expansion for further surface examinations would not provide any added assurance of the structural integrity of the studs, and would only increase personnel radiation exposure to those individuals who would be required to remove, prepare, and examine the additional studs.

G. Regarding Request for Relief No. CR-10, which addresses limited ultrasonic and surface examination of Examination Category B-J, C-F-1, and C-F-2 welds, provide a list of the applicable welds, drawings depicting the joint designs, and ultrasonic coverage plots where Code requirements are not satisfied.

Note: This request for relief appears to be generic in nature as specific welds have not been identified. For this request for relief to be ccnsidered, the licensee must submit information specific to each weld where Code-required examinations have not been satisfied.

Request for Relief No. CR-10 has been withdrawn. This is reflected in the applicable table contained in Revision 1 to the ISI plan document included in this response.

H. Regarding Request for Relief No. CR-14, which addresses examination of the RPV support weld, provide a technical discussion of the possibility of a volumetric examination of the C-D region of Figure IWB-2500-13. Based on the review of Figure IWB-2500-13, the required surface examination, and the licensee's submittal, it appears that access to examination area C-D is restricted due to the RPV lower head and support skirt configuration. The license has not, however, proposed any alternative examination for the C-D region.

Request for Relief No. CR-14 has been withdrawn. This is reflected in the applicable table contained in Revision 1 to the ISI plan document included in this response.

I. Regarding Request for Relief No. PR-04, which addresses alternative testing for the RHR heat exchanger tubes, discuss how the proposed alternative provides an acceptable level of quality and safety when it is essentially based on the loss of system integrity. (The licensee's proposed alternative is to monitor radiation levels across the pressure boundary during shell-side pressure tests.) Other utilities have proposed eddy current testing of the heat exchanger tubing and a VT-2 visual examination when the channel head cover is removed for maintenance activities.

Regarding Relief Request PR-02, it should be noted at this point that the Staff's RAI contains a typographical error regarding this Request for Relief. The RAI requests information for Request for Relief No. PR-04. The subject matter for PR-04 (RPV head flange seal leak detection) is not consistent with the additional information requested. LSCS will respond based on the assumption that the staff is pursuing additional information on Request for Relief No. PR-02.

It should be noted that LaSalle Station is not requesting relief from the requirement to perform a VT-2 visual inspection of the RHR heat exchanger shell side tubing while at nominal operating pressure. LaSalle intends to perform the required V-2 visual examination of the tubing with the bottom plate removed and the tube side drained. The tubing will be inspected for leaks with the RHR Service Water Cooling pumps running at nominal operating pressure as required. LaSalle Station is requesting relief from the Code required frequency of the VT-2 examination such that it will coincide with a normal period of planned scheduled maintenance on the RHR Heat Exchanger(s). This is not unlike the Code required visual examinations of Class 1 pump and valve internal surfaces in accordance with IWB-2500-1. The examination of these components located within the primary coolant pressure boundary is not required unless the components are opened for other reasons. It should also be noted that the VT-2 method in and of itself is in all cases dependent on detection of the loss of, or evidence of the loss of the structural integrity of the component being examined. In light of these facts it is clear that imposition of the Code requirements, would result in undue hardship and personnel radiation exposure without any commensurate increase in the level of safety to the public. An identical Relief Request (RI-27 Revision 1) was approved by the staff for use during the initial 10-year ISI interval in an SER dated May 23, 1994.

J. Verify that there no relief requests in addition to those submitted. If additional relief requests are required, the licensee should submit them for staff review.

One additional relief request (PR-07) is being submitted for review and is included in revision 1 to the second 10-year ISI plan document.