BWR OWNERS' GROUP

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Project Number 691

BWROG-06036
November 16, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated March 21, 2005

REFERENCE: Letter TSTF-06-13 from the Technical Specifications Task Force to the NRC Document Control Desk, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," dated July 3, 2006

Dear Sir or Madam:

In the response to RAI #3 in the subject letter, the Technical Specifications Task Force stated that the requested GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station, GE-NE-0000-0024-9858 R0, February 2004," would be provided by the BWROG to the NRC after receiving the appropriate approvals from the licensee and the BWROG. Approval from the licensee has been obtained and accordingly, we are providing a non-proprietary version of this report in the enclosure for your consideration.

Please contact me or Fred Emerson (BWROG Project Manager, 910-675-5615) if you desire further information.

Sincerely,

R C Bunt

R. C. Bunt
BWR Owners' Group Chair

Enclosures

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**Exelon Nuclear**

# CRD Notching Surveillance Testing for Limerick Generating Station
## (Report originally prepared in 2004)

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# IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ATWS | Anticipated Transient Without Scram |
| CRD | Control Rod Drive |
| HCU | Hydraulic Control Unit |
| IGSCC | Intergranular Stress Corrosion Cracking |
| CRT | Collet Retainer Tube |
| CT&F | Cylinder, Tube & Flange Assembly |
| PRA | Probabilistic Risk Assessment |
| SIL | Service Information Letter |
| UFSAR | Updated Final Safety Analysis Report |

## 1. INTRODUCTION

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod at least one notch) performed at a frequency of 7 days (weekly) and 31 days (monthly) for fully withdrawn control rods and partially withdrawn control rods, respectively. During power operation, most control rods in the core are fully withdrawn and subjected to the weekly notch testing which routinely pre-occupies the reactor operators. SIL 139 also credits the weekly notch testing as a means to demonstrate the Collet Retainer Tube (CRT) integrity. This report provides the technical justification to decrease the weekly notch testing recommended in SIL 139 for all fully withdrawn control rods from weekly to monthly. Review of plant licensing commitments with regards to the proposed change to the notch test surveillance frequency is not within the scope of this report. This report was originally prepared in 2004.

## 2. SUMMARY

CRT cracking is a well understood phenomenon and is observed in the original design (919D258G001) and interim design (919D258G002) CRTs still in current operation. Continued circumferential cracking could lead to failure of the CRT that would render a Control Rod Drive (CRD) immovable. Notch testing is a viable method of demonstrating CRT integrity and it should be continued. Based on the absence of any known CRT failure and the slow crack growth rate, it is acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

The recommended CRD inspections related to nitride corrosion and extent of the CRT cracking for CRDs that operated for an extended period are discussed in Section 5, RECOMMENDATIONS.

## 3. DISCUSSION

During a planned inspection of CRDs in 1975, cracks were observed on the CRT surfaces of the Cylinder, Tube & Flange Assembly, 919D258G001 (Fig. 1). Although the cracks did not affect the CRD functionality, design modifications of the CRT (919D258G002, 919D258G003) and CRD system operation changes (Ref. 2) were implemented. SIL No. 139, and its revisions and supplements (Ref. 1) provided the inspection and operating recommendations. Changes to the Technical Specifications were also recommended in Reference 3 which addressed actions to be taken if a rod would not insert. Also, it was recognized that the weekly notch testing provides adequate demonstration of CRT integrity.

### 3.1 Original Collet Retainer Tube Design Description

The BWR CRDs are hydraulically operated stepping mechanisms mounted in the CRD housings that extend down from the reactor vessel bottom head. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a special hydraulic signal to unlock for rod withdrawal. The CRT is a short tube welded to the upper end of the CRD which houses the locking collet and its supporting piston, collet return spring and an unlocking cam. The CRT has three primary functions: a) to carry the hydraulic unlocking pressure to the collet piston, b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

The CRT of the original design Cylinder, Tube & Flange Assembly (919D258G001) was manufactured from wrought 304 stainless steel and the inner surface hardfaced by nitriding. The CRT is provided with three flow holes to allow displaced water from inside the CRD to flow out into the annulus surrounding the CRD. The wall thickness below the flow hole elevation is increased by 0.2 inch. The CRT is then welded to the top of the outer tube.

The mechanical and pressure loads on the CRT are very small, and the primary design consideration is rigidity and ability to withstand handling damage. During scrams at reactor operating conditions, the temperature distribution in the CRT changes substantially. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

### 3.2 CRT Operating History

SIL 139 (Ref. 1) provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular (see Section 3.6), were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360° severance of the CRT that would render the CRD inoperable (i.e., prevent insertion, withdrawal or scram). Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the Technical Specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CT&F (919D258G002) but no cracks haven been observed in the final design CT&F (919D258G003) [SIL 139, Supplement 5, Revision 1]. To date, operating experience data shows no reports of a severed CRT at any BWR.

Limerick Units 1&2 notch test results from the previous six years have not reported any control rod repositioning (i.e., notching) failures that could be attributed to a failed CRT. However, approximately 40% of Limericks' CRTs have been rejected when subjected to the inspection criteria recommended in SIL 139. The criteria specify that CRTs with relevant penetrant indications should be rejected with the intent to detect cracking before it becomes a problem. However, there has been no reported failure (e.g., 360° severance) of CRTs throughout Limericks' operating history. It should be noted that many CRDs have been replaced with an improved CRT design that is less susceptible to cracking.

### 3.3 Control Rod Notch Testing

The primary purpose of the notch testing is to verify that the control rods are movable in response to a scram signal. This is accomplished by inserting a control rod at least one notch and observing that the control rod moves with normal drive water pressures. Occasionally, elevated drive water pressures are required to insert control rods due to other causes not related to the scram capability of the control rods. These factors are commonly attributed to excessive CRD seal leakages or HCU directional control valve operational anomalies. Additional diagnostic testing or system operating parameters can provide confirmation of these factors. Notch testing is a viable method to identify any control rod with excessive mechanical binding that could prevent scram. It should be noted that routine notch testing can identify most postulated causes for excessive mechanical binding but notch testing at the fully withdrawn position is not intended to provide reliable detection of channel bow. Therefore, extending the frequency of notch testing for the fully withdrawn rods has no impact on any interim testing currently being performed to monitor channel bow.

Numerous BWRs have reported difficulties notching out from the fully inserted position following a plant scram. A primary cause has been attributed to increased internal seal leakages resulting from crud entering the CRD sealing areas as indicated by higher withdrawal stall flows. Plant data have shown that the average stall flows exhibit steady decreasing trends following a scram event. Also, changes in CRD performance (e.g., notching) have been observed when the rods were moved to the fully withdrawn position after residing at a deep position (i.e., exercised monthly) for a sequence. The weekly notching exercises may effectively flush the sealing areas and subsequently improve the sealing function. This crud induced anomaly is not a permanent mechanical degradation and does not adversely affect the scram function.

Another added benefit of notch testing is the flushing of the CRD internal creviced surfaces. The creviced nitrided parts are potentially vulnerable to corrosion degradation (pitting); three creviced areas are formed by: (a) the drive piston seals/bushings and the piston tube OD, (b) the collet fingers and the index tube notch, and (c) the collet piston rings and the collet retainer tube ID (for 919D258G001). The notch-in and notch-out movements permit the drive water flow in the under-piston and over-piston annuli, and at the same time actuate the collet assembly. This notching exercise is deemed the most effective means of flushing the nitrided creviced areas during power operation.

### 3.4 Technical Specification Surveillance Testing

At the time of the initial CRT crack discovery (1975), each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Subsequently, many BWRs have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

### 3.5 Reliability Assessment

BWR plants are designed to scram on demand with an extremely high reliability. The scram function requires successful generation of a scram or a reactor trip signal, successful operation of the hydraulic control units (HCU) and successful insertion of the control rods. In response to the ATWS Rule, a number of improvements have been made to the original design to further improve the scram function reliability. In the unlikely event the scram system fails, the operator can inject liquid boron solution into the reactor to shut the plant down.

In a recent NRC-sponsored study, NUREG/CR-5500, Volume 3, (Reference 4), the BWR scram system reliability was evaluated to be 5.8E-6 per demand. Because of this low failure probability, the contribution of ATWS events in BWR plant PRAs is very low. Limerick PRA shows an ATWS contribution of less than 5% to total plant core damage frequency.

From a scram function reliability point of view, the failure of individual rods to insert randomly is not a concern. According to the PRA success criterion in Reference 4, it is necessary for one third of the total number control rods to fail randomly to cause a scram function failure. For Limerick, with 185 control rods, more than sixty rods have to fail randomly at the time of the scram event, which is judged to be highly unlikely. From a scram function reliability point of view, the common cause failure of the control rods to insert is more of a concern than the failure of the individual rods to insert.

The common cause failures that result in multiple rod insertion failure are excessive mechanical binding due to either control blade deformation or multiple CRT failures. A number of surveillance tests, such as the notch testing described in Section 3.3 and the scram time testing (described below), are performed to verify that control rods are movable in response to a scram signal and thus ensure a high reliability for the scram function. As part of the scram time tests, the scram times for a sample of the control rods are tested periodically. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components (scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator), as well as operation of the control rods. Finally, the HCUs and control rods are also tested during refueling outages, approximately every 18-24 months.

The notch test can identify if a CRT is totally severed resulting from a 360-degree IGSCC-initiated crack. However, it is extremely unlikely that more than 60 CRTs would sever as a result of IGSCC in a relatively short period of time of one month. Similarly, the notch testing can identify most postulated causes of mechanical binding. The Limerick notch testing results for the last six years indicate that all control rods were successfully repositioned in both Limerick Units 1 and 2. However, notch testing at fully withdrawn position may not be reliable to detect conditions such as channel bow and special tests may be needed to identify them. The notch test for such cases only provides limited assurance that the scram system is highly reliable. The scram tests done with the 10% of the control rods provide a greater assurance that control rods are not stuck, since the scram test checks for the complete insertion of the rod. Therefore, decreasing the notch test frequency from weekly to monthly for the fully withdrawn rods is expected to have insignificant impact on scram system reliability.

In 614 groups of weekly tests conducted during the past six years at Limerick Units 1 and 2, with over 166 rods (estimated 90% of the 185 rods are fully withdrawn at any time) tested in each group of tests, no failures have been detected. Even though these tests are not able to detect all failure modes that could result in scram failure, for the failures that can be detected, these tests show a very low failure rate: 6.8E-6 per demand for the single rods (The failure rate is obtained through statistical calculations using Chi-squared distribution). This low failure rate provides additional basis for extending the tests from weekly to monthly intervals. In summary, from a scram system reliability point of view, it would be acceptable to extend the CRD notch test interval from weekly to monthly frequency.

### 3.6 IGSCC Evaluation

As described in SIL 139, it was found that CRT cracking was attributed to IGSCC. To address the SIL 139 concerns, GE recommended that the plants use surveillance notch testing to establish that circumferential IGSCC sufficient to sever the CRT is not present. Therefore, control rod insertion capability and the accompanying CRT integrity verification has been substantiated by the notch test which is performed at a weekly frequency for fully withdrawn control rods.

To assess the request to extend the time between notching tests, GE's PLEDGE model was used to evaluate the additional crack lengthening that could occur in 31 days versus the current 7 days between tests. This model, which is based on fundamental principles of stress corrosion cracking, can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. The assessment evaluated the postulated bounding crack growth using the current water chemistry recommendations put forth in SIL 148 (Ref. 2) and furnace sensitized material condition. It was found that the additional time (~24 days) only represented an additional 10 mils of growth in total crack length. This small difference would have very little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective it is acceptable to decrease the frequency of the notching test given the low crack growth rates associated with the Limerick Water Chemistry specification for the CRD that are in alignment with the objectives of SIL 148 recommendations. It is assumed that the Limerick Generating Station has been following these specifications.

## 4. CONCLUSIONS

**4.1** The material condition (sensitized stainless steel), geometric configuration (step change, crevice), thermally induced operating stresses and residual stresses are the primary attributes contributing to the potential for intergranular stress corrosion cracking in the original design CRTs. The stress characteristics in the CRT make the IGSCC both cycle and time dependent.

**4.2** The notch testing, that is accomplished by inserting a control rod at least one notch and observing that the control rod moves, is a viable method to confirm the integrity of the CRT.

**4.3** A primary cause for CRD notching difficulties has been attributed to increased internal seal leakages resulting from crud entering the CRD sealing areas as indicated by higher CRD stall flow measurements. The weekly notching exercises may effectively flush the sealing areas and subsequently improve the sealing function.

**4.4** The added benefit of notch testing is the flushing of the CRD internal creviced surfaces. The notching exercise is deemed the most effective means of flushing the nitrided creviced areas to minimize the risk of corrosion degradation.

**4.5** There has been no reported failure (e.g., $360^{\circ}$ severance) of CRTs throughout Limericks' operating history.

**4.6** The BWR scram system has extremely high reliability. One of the potential contributors to scram system unreliability is the common cause failure of control rods to insert. While both the notch test and the scram testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding, the primary assurance of scram system reliability is provided by the scram testing. Hence, extending the CRD notch test interval to monthly is not expected to impact the reliability of the scram system.

**4.7** Based on modeling of IGSCC crack growth rates, there is very little predicted difference in the crack length between the weekly and monthly notch test. This analysis supports the decision to change the frequency.

**4.8** Based on the above, it is acceptable to decrease the SIL 139 recommended notch testing surveillance frequency of fully withdrawn control rod from weekly to monthly.

## 5. RECOMMENDATIONS

**5.1** Since the creviced nitrided parts are potentially vulnerable to corrosion degradation, a limited sampling such as four CRDs, removed for maintenance, should be inspected for evidence of discernable corrosion that is different compared to corrosion observed in the past when weekly notching was performed. This inspection should be conducted twice with the first inspection no later than two cycles of reactor operation with the reduced frequency. This recommendation is made to address the low, but possible risk of unexpected degradation due to the reduced flushing through the revised notch-testing schedule. The corrosion level should be confirmed to be unaffected by the change from weekly to monthly notch testing.

**5.2** Although there has been no reported failure of the original design CRTs (919D258G001), a few CRTs have been noted to be severely cracked. An inspection sampling plan of CRDs with the original CRT design based on CRD water chemistry, years of continuous reactor service, and evaluation of CRT maintenance data is recommended to assess the actual extent of CRT cracking and to assess the need for replacement of the original design CRT.

## 6. REFERENCES

1. SIL No. 139, Control Rod Drive Collet Retainer Tube Cracking, July 18, 1975 [including Revised Supplement 1 (April 29, 1977), Supplement 2 October 24, 1975), Supplement 3 (January 30, 1976), Supplement 4 (April 29, 1977), Supplement 5 Revision 1 (February 6, 1986)].

2. SIL No. 148, Water Quality For the Control Rod System, September 15, 1975.

3. NUREG-0479, Report on BWR Control Rod Drive Mechanical Failures, Technical Activity B-48, January 1979, USNRC.

4. NUREG/CR-5500, Vol. 3, "Reliability Study: General Electric Reactor protection System, 1984-1995" USNRC, May 1999

**Figure 1: CRT Configurations**