GPU NUCLEAR

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November 25, 1998
1920-98-20653

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1, (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Technical Specification Change Request (TSCR) No. 277

In accordance with 10 CFR 50.4(b)(1), enclosed is Technical Specification Change Request (TSCR) No. 277. The purpose of this TSCR is to request changes to the Surveillance Specifications for Once Through Steam Generator (OTSG) inservice inspections for TMI-1 Cycle 13 Refueling (13R) Outage examinations which would be applicable for one cycle of operation only, TMI Operating Cycle 13. The current Technical Specifications expire at the end of Operating Cycle 12, which is scheduled to occur on September 10, 1999. GPU Nuclear requests NRC review and approval of this change in a license amendment by April 15, 1999.

Using the standards in 10CFR50.92, GPU Nuclear has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10CFR50.91(a)(1). Also enclosed is the Certificate of Service for this request certifying service to the chief executives of the township and county in which the facility is located, as well as the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection.

If you have any questions regarding this information, please contact Mr. Bob Knight at (717) 948-8554.

Sincerely,

MJ Ross
for

James W. Langenbach
Vice President and Director, TMI

Enclosures:
1) TMI-1 TSCR No. 277 Safety Evaluation and No Significant Hazards Consideration Analysis
2) TMI-1 Technical Specification Revised Pages for TSCR No. 277
3) Certificate of Service for TMI-1 TSCR No. 277

A001

MRK

cc: Administrator NRC Region I
TMI Senior NRC Resident Inspector
TMI-1 Senior NRC Project Manager
File No. 98192

TMI-1 TSCR No. 277
Safety Evaluation and No Significant Hazards Consideration Analysis

I. Technical Specification Change Request (TSCR) No. 277

GPU Nuclear requests that the following changed replacement pages be inserted into the existing TMI-1 Technical Specifications (TS):

Revised Technical Specification Pages: 4-79 through 4-83.

These pages are included with Enclosure 2.

II. Reason For Change

Based on the results of the Cycle 12 Refueling Outage (12R) Once Through Steam Generator (OTSG) tube pull metallurgical analyses and the 12R eddy current test (ECT) growth analyses, this change proposes the continuation of repair criteria which address volumetric inter-granular attack (IGA) degradation identified on the inside diameter (ID) in the unexpanded portions of the TMI-1 OTSG tubes. The repair criteria were originally licensed for the plant under TMI-1 License Amendment No. 206 (Reference 1) for the 12R Outage and the present operating cycle (Operating Cycle 12). This TSCR proposes that these same repair criteria be used during the next scheduled refueling outage, the Cycle 13 Refueling Outage (13R) examinations and effective until the end of the next operating cycle, Operating Cycle 13. These proposed changes, which are identical to those previously licensed, continue to impose restrictions on the axial and circumferential length of volumetric ID IGA tube degradation that may remain in service. GPU Nuclear will perform growth analyses using comparisons of the ECT results from 13R with 12R and previous inspections. In-situ pressure testing will be performed, if the IGA indications are not bounded by previous test results, for a sample of tubes to demonstrate acceptable leakage integrity for tubes left in service with indications of volumetric ID IGA.

This change reflects changes to the voltage normalization procedures for bobbin coil probe eddy current inspections of the TMI-1 steam generator tubes. GPU Nuclear has been using a normalization procedure which sets the bobbin coil prime frequency peak-to-peak response from the four 20 percent through-wall holes of the B&W Owners Group (BWOG) ASME calibration standard to 10 volts. The new normalization value will be 4 volts for the same technique. This change in voltage normalization will align the plant's normalization with that specified in the current industry and BWOG standards and practices. This change will provide the benefits of direct ECT signal comparisons among the BWOG plants. Changes in the voltage normalization procedures will affect the voltage values that are presently specified in the steam generator tube volumetric ID IGA repair criteria. Included in the Bases section is a note to clarify that the voltages stated in TMI-1 TS Section 4.19 are based on 4 volt normalization.

This request also proposes a change to TS Section 4.19.5 (b) regarding "Reports" of steam generator inspection results to allow 12 months for reporting the complete results rather than 90 days and to clarify that the reporting period begins with closure of the plant's main electrical generator breaker.

The following lists the changes on each of the affected pages as proposed by TSCR No. 277:

Page 4-79
Section 4.19.2, Note (1): The threshold voltage increase at which a tube whose ID IGA indication is counted in the degraded tube population is revised from 0.6 volts to 0.24 volts. This revision is a result of the proposed change in voltage normalization for bobbin probe examinations.

Page 4-80
Section 4.19.4 (a), 3: The threshold voltage at which a tube whose ID IGA bobbin coil indication is considered degraded is revised from 0.5 volts to 0.2 volts. This revision is a result of the proposed change in voltage normalization for bobbin probe examinations. The definition of a degraded tube is also revised so that these criteria (pertinent to ID IGA indications) are extended from the "12 R outage" and "Cycle 12 operation" to the "13R Outage" and "Cycle 13 operation" respectively.

Page 4-81
Section 4.19.4 (a), 6: The definition of "repair limit" is revised so that these criteria (pertinent to ID IGA indications) are extended beyond Cycle 12 operation through Cycle 13 operation.

Page 4-82
Section 4.19.5 (b): The parenthetical phrase "(main generator breaker closure)" is added to define this event as the start date of the reporting period for submittal of the complete results of the 13R steam generator tube inspections; and the reporting period is changed from 90 days to 12 months consistent with Draft Regulatory Guide, DG-1074, dated March 1998 (Reference 2).

Page 4-83
Bases: Several references to "Outage 12R" are revised to indicate "Outage 13R;" and the references to "Cycle 12" operation are revised to indicate "Cycle 13" operation.

III. Safety Evaluation Justifying the Change

Background

In November 1981, while performing Reactor Coolant System hydrostatic testing with the reactor shut down, primary-to-secondary system leakage was detected in both TMI-1 OTSGs. Subsequent Eddy Current Examinations revealed many defective tubes. Metallographic examination of portions of removed tubes confirmed that the tube degradation initiated from the primary side (ID) of the tubes, principally in the form of circumferential stress-assisted inter-granular cracks at the upper tube sheet (UTS). The active chemical impurity causing the corrosion was sulfur in reduced forms, which had been inadvertently introduced into the Reactor Coolant System. The vast majority (approximately 95 percent) of the defects occurred within the top 2 to 3 inches of the 24-inch thick UTS since the corrosive attack occurred most rapidly at the air/water interface during lay up. The air/water interface was located within the UTS during a significant portion of the 1981 post-hot-functional shutdown period. To repair the defective OTSG tubes within the UTS, GPU Nuclear applied a kinetic expansion (explosive) tube repair technique. GPUN's repair of the TMI-1 OTSG tubing was reviewed and approved by the NRC in the Safety Evaluation Report (SER) that accompanied License Amendment No. 103, dated December 21, 1984, entitled "NUREG-1019, Supplement No. 1, Safety Evaluation Report by the Office of NRR - Three Mile Island Nuclear Station, Unit l (TMI-1) - Steam Generator Tube Repair and Return to Operation," (enclosure to Reference 3).

The kinetic expansion repair technique applied in the early 1980s addressed the existence of defects located within the UTS. However, a limited population of the tubes contained degradation located below the UTS secondary face that could not be repaired by the kinetic expansion technique. During plant outages from 1985 through 1997, GPU Nuclear inspected the steam generator tubing for this type of degradation (along with other types of degradation pertinent to OTSGs). Based on these examinations, GPU Nuclear determined that the ID IGA degradation due to the chemical intrusion had become inactive. Because of the uncertainty in sizing the depth of small amplitude ID IGA degradation that was not previously repaired, the NRC and GPU Nuclear agreed in 1997 that the TS tube repair criteria at TMI-1 should be amended to address the tubes identified with this mode of degradation. As mentioned above, TMI-1 License Amendment No. 206 (Reference 1) provided a set of surveillance criteria under which the ID volumetric IGA indications could remain in service for the TMI-1 12R Refueling Outage and the present Cycle 12 operation which followed. This TSCR proposes the extension of those criteria for an additional outage and cycle of operation.

GPU Nuclear has used a 10 volt normalization (as described above) for the OTSG tube bobbin coil eddy current examinations since the tube damage that occurred in 1981. Other nuclear plants have also used this normalization; however, a greater number of plants have adopted a 4 volt normalization. With this TSCR GPU Nuclear proposes to convert to the 4 volt normalization, which is becoming standard in the industry. This change will allow a more direct comparison between TMI-1 eddy current data and that of the industry, and particularly that of the BWOG plants. This change will also affect TS sections in which an eddy current voltage is specified, such as the present Section 4.19.4. In effect, the measured voltages representative of the eddy current signals under the proposed 4 volt normalization will be 40 percent of the measured voltages under the a 10 volt normalization. This voltage normalization change is applicable for the bobbin probe examinations only, and does not affect signal quality or phase angle analysis of the bobbin probe eddy current signals, rotating probe eddy current, or other aspects of eddy current signal analysis.

In the past GPU Nuclear has used conservative event dates at the end of an outage to start the schedule for providing the required report of the complete steam generator inspection results to the NRC. TS Section 4.19.5.b presently states that the complete results "shall be reported to the NRC within 90 days following completion of the inspection and repairs." In order to provide a more consistent and better defined event date to begin the reporting period, GPU Nuclear requests that main electrical generator breaker closure be utilized as the event signifying the end of an outage's steam generator inspection and repair work. This clarification of the reporting period start date is typical of the interpretation used to determine the reporting period at other plants and represents an improvement over the current TS. Changing the reporting period from 90 days to 12 months is consistent with the Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," dated March 1998 (Reference 2).

Proposed Extension Of Tube Repair Criteria

The depth-based tube repair limit (40 percent through-wall) in the TMI-1 TS is intended to help ensure adequate tube integrity through the end of the following operating cycle. This repair limit has remained in the TS each of the outages and along with the changes proposed by this TSCR, GPU Nuclear proposes to retain this repair limit. The ability to accurately size the depth of service-induced IGA degradation with eddy-current inspection techniques is complicated by a number of variables. Accurately dispositioning tubes with identified ID IGA degradation in accordance with the 40 percent depth repair limit is difficult using inspection methods that are currently available. For example, ID IGA indications that are detectable but very small are difficult to analyze because of the small size of the eddy current signal. However, currently available eddy current examination technology does possess the capability to conservatively estimate the axial and circumferential extents of this degradation. TMI-1 License Amendment No. 206 (Reference 1) provided the axial and circumferential extents for repair limits to disposition ID IGA volumetric degradation in addition to the 40 percent through-wall repair limit. This TSCR proposes to continue to assess the structural integrity of tubes with ID IGA indications based on the length-based repair limits in addition to the 40 percent through-wall limit as provided by License Amendment No. 206 (Reference 1).

Although eddy current inspection techniques can measure the extent of ID IGA degradation, these methods do not currently by themselves adequately assess tube leakage integrity margins. During the 12R Outage, GPU Nuclear performed in-situ pressure testing, supplemental testing, and analyses to bound the eddy current findings and demonstrate that those tubes left in service will not leak or burst under postulated accident conditions. If eddy current indications are found during the 13R Outage which exceed those bounded by the in-situ pressure testing performed in Outage 12R, additional in-situ tests will be performed on a bounding sample of the larger 13R indications. Based on previous inspection data (from the several outages since the 1985 plant restart following the kinetic expansion repairs), GPU Nuclear has concluded that the ID IGA degradation is dormant and not expected to degrade further over the next cycle of operation. This, in conjunction with the in-situ pressure testing results, will again be used to demonstrate adequate leakage integrity margins for tubes with ID IGA indications through the end of the next operating cycle. Details of GPUN's proposal for the continuance of the repair criteria are included in the following sections.

Outage 13R Inservice Inspection of TMI-1's Steam Generator Tubes

For the 13R Outage, GPU Nuclear has proposed to examine 100 percent of the OTSG tubes with a bobbin coil eddy current probe, similar to the 12R scope. Tube examinations using this probe should identify any indications of ID IGA degradation that could potentially degrade a tube's structural and leakage integrity margins. The bobbin probe, however, cannot completely assess the morphology or extent of any detected indications. Because the repair criteria apply only to ID IGA degradation, during the 13R Outage examinations, GPU Nuclear will again examine all ID flaw-like indications (detected using a bobbin coil probe) with an MRPC (motorized rotating pancake coil) probe, including those bobbin coil indications that were confirmed as ID IGA degradation during the 12R Outage examinations. This ensures a conservative examination scope since the bobbin coil can identify a change in tube condition. The use of both the bobbin and MRPC probes will provide four examinations over two consecutive outages for a typical ID IGA indication. Examinations using the rotating probe can determine whether the morphology of a bobbin indication is volumetric and thus indicative of IGA. In addition, the data acquired by the MRPC probe inspections will be used to assess the axial and circumferential extent of confirmed tube degradation and whether the degradation initiated from the ID or outside diameter (OD) of the tube.

IGA degradation is characterized as a mode of degradation that is volumetric rather than "crack-like" in nature. That is, this degradation affects a small volume of tube material and typically has dimensions that extend axially, circumferentially, and through-wall in the tube. ID IGA degradation exhibits a morphology that extends both along the tube axis and around the circumference. Crack-like indications, however, extend primarily along only two tube directions (i.e., through-wall/axial or through-wall/circumferential). Because rotating probes are sensitive to degradation extending in both the axial and circumferential directions, these probes are capable of providing data to allow determination of whether an indication is crack-like or volumetric. In addition, they possess the capability to size the length (extent) of steam generator tube degradation. GPU Nuclear will inspect all ID flaw-like indications detected by bobbin coil with an MRPC (Plus Point) probe to determine the flaw morphology. This will enable GPU Nuclear to confirm the mode of degradation, and determine the axial and circumferential lengths of the indication using the MRPC probe, if applicable.

Eddy current inspection techniques are capable of distinguishing the surface of origin (i.e., ID or OD) of flaws included in calibration standards and have been used successfully to determine the initiating surface of steam generator tube degradation. Examination of each ID flaw-like indication using both the bobbin coil and Plus Point probes will provide complementary data to the analysts that will continue to indicate whether an indication is ID or OD in nature. GPU Nuclear will consider those signals with phase angles less than 30 degrees for the bobbin coil probe data to be indicative of ID degradation. For the Plus Point probe, indications with signal phase angles less than that of the signal obtained from the 100 percent through-wall notch in the calibration standard will be considered ID. As in the inspection during Outage 12R, only those indications, which display a phase rotation in the ID, flaw plane will be considered ID indications. The TMI-1 inspection guidelines require all bobbin coil indications of apparent wall loss with phase angles outside the ID phase plane (e.g., ≥ 30°) to be excluded from the scope of the proposed repair criteria. In addition, the analysis of rotating probe data will require that analysts verify that the indications are ID in origin. As in Outage 12R, the use of data from two inspection probes will minimize the potential for inadvertently applying the proposed repair criteria to OD-initiated degradation. From our experience with ID-initiated degradation at TMI-1, GPU Nuclear has found that significant degradation of this type can be identified using the bobbin coil.

Structural Integrity Assessment

The proposed modifications to the TMI-1 TS include changes to the steam generator tube repair limits defined in TS 4.19.4.a.6, "Repair Limit." The existing requirements specify that tubes shall be repaired or removed from service when degradation exists within the tube that is equal to or greater than 40 percent of the nominal tube wall thickness, along with additional limits on the dimensional extent of volumetric ID IGA indications for the 12R Outage and Cycle 12R operations. Specifically, the circumferential and axial lengths of ID IGA tube degradation are limited to less than 0.52 and 0.25 inches, respectively. The depth (when assigned), axial extent, and circumferential extent limits are applied separately; if any one of the three limits is exceeded, the tube must be repaired. These length-based repair limits for the ID IGA degradation are based on structural analyses of flawed steam generator tubing assuming it contains through-wall defects of the limiting length. GPU Nuclear continues to conclude that tubes with defects at these repair limits would retain their structural integrity under normal operating and postulated accident loads. These length-based repair limits are unchanged by the proposed TS. The proposed changes to TS 4.19.4.a.6 include a continuation of the current TS limits for the 13R Outage and Operating Cycle 13. Therefore, reference to the 12R Outage and Cycle 12 operations only is deleted from the proposed section.

TS Section 4.19.4.a.3(a) currently defines a degraded tube as one with imperfections that exceed 20 percent in depth as well as those with indications of ID IGA with lengths greater than or equal to 0.26 inches circumferential extent or 0.13 inches axial extent. In addition, a tube is considered degraded if the bobbin coil voltage amplitude for an ID IGA indication is equal to or greater than 0.5 volts. The proposed TSCR retains the 20 percent depth, 0.26 inch circumferential extent, and 0.13 inch axial extent criteria without change. The voltage criterion, however, is adjusted from 0.5 to 0.2 volts because GPU Nuclear is revising its normalization of bobbin coil voltages to agree with those presently considered "industry-standard." Note that this revised voltage criterion is in effect identical to that previously licensed, because GPU Nuclear will adjust all of the previous voltage measurements obtained under the 10 volt normalization to a voltage measurement that would have been obtained under a 4 volt normalization (i.e., 40 percent of the 12R Outage voltage). For example, consider a hypothetical tube, which contained an ID IGA indication that measured 1.0 volt during Outage 12R - at a 10 volt normalization. This tube was considered degraded during Outage 12R because it exceeded the 0.5 volt limit provided in TS Section 4.19.4.a.3(a). This indication would have measured 0.4 volts at a 4 volt normalization. GPU Nuclear would continue to consider this tube degraded since the indication's 12R voltage, when adjusted for the new 13R normalization, would exceed the new 0.2 volt limit.

If 13R eddy current indications are found to exceed those bounded by the 12R in-situ pressure testing, GPU Nuclear will complete additional in-situ pressure testing of a sample of steam generator tubes containing ID IGA indications as necessary to supplement the in-situ tests performed during Outage 12R. Tubes selected for testing will include the most significantly degraded tubes (e.g., the lowest expected burst pressure) as determined by an assessment of eddy current signal characteristics. GPU Nuclear will again consider indication voltage, dimensional lengths, and phase angle depth measurements to choose potential candidates for in-situ testing. If the tested tubes retain their structural and leakage integrity throughout the test, there is additional assurance that tubes with less significant degradation will have adequate margins for tube integrity.

The in-situ pressure test equipment that GPU Nuclear intends to use during Outage 13R can effectively simulate both peak hoop stresses and peak axial stresses (corrected for thermal conditions) which might be applied to tubes containing volumetric flaws during a postulated Main Steam Line Break (MSLB). This is an improvement over the 12R Outage in which the available equipment was unable to subject the ID IGA indications to the maximum calculated axial tensile stresses arising from a postulated MSLB. To supplement the 12R in-situ pressure testing, GPU Nuclear performed additional analyses of the volumetric flaws during the 12R Outage to support NRC review prior to the issuance of License Amendment No. 206 (Reference 1), and subsequent laboratory tests on volumetric flaws in pulled tubing (Reference 5), to further demonstrate that the flaws had adequate margins for tube integrity. With the in-situ test equipment improvements, GPU Nuclear does not anticipate that the supplemental work undertaken for Outage 12R will be required for Outage 13R.

Demonstration of Leakage Integrity Margins

The existing depth-based repair criteria are established to ensure steam generator tubes have adequate structural and leakage integrity with appropriate margins of safety under normal operating and postulated accident conditions. GPU Nuclear will continue to apply the depth-based repair limits to ID IGA degradation (when depth is assigned) in addition to the length-based limits for the 13R examinations. Under high differential pressures, this degradation could become a leakage path for the reactor coolant to the steam generator secondary side. As a continuing part of the repair criteria to address the ID IGA degradation, GPU Nuclear will complete in-situ pressure testing of a sample of steam generator tubes with ID IGA indications as required to demonstrate a bounded low leakage potential for tubes containing the ID IGA mode of degradation.

In-situ pressure testing subjects degraded tubes to conditions that are conservative with respect to internal pressure loads postulated to occur under accident conditions. Internal pressure within the tube during the test induces axial and circumferential stresses within the tube wall. The purpose of the testing is to assess whether the degraded tubes exposed to these elevated stresses are capable of withstanding the test conditions while retaining leakage and structural integrity. The test pressure is adjusted to account for the temperature dependence of material strength and other factors that may not be simulated in the test. If a tube leaks during testing, the leak rate can be quantified provided it is not in excess of the capabilities of the test equipment.

Using the eddy current inspection results, GPU Nuclear will select tubes for testing that have indications of degradation which appear more limiting with respect to leakage integrity than those which were qualified by tests during 12R (as discussed above). Although the inspection may not accurately assess the depth of the degradation, other available data such as defect length, eddy current amplitude response, or phase angle depth measurements may be used to identify a subset of degraded tubes with indications that appear to bound the remaining tubes in the population. In-situ testing is a tool whereby GPU Nuclear can continue to assess the potential for leakage from all the degraded tubes. Assuming the limiting tubes retain full structural and leakage integrity throughout the test, it is reasonable to conclude that the other, less degraded tubes in the population would also have sufficient integrity to withstand accident-induced loads without failure.

Steam generator tube flaws may progress in length or depth during operation and degrade the margins for tube integrity below acceptable limits. If it can be demonstrated that the expected flaw growth rate for the ID IGA degradation is negligible, then the bounding in-situ pressure test data will provide assurance that the affected tubes will have sufficient margins for structural and leakage integrity for operation. A discussion on the expected growth rate for ID IGA indications is included in the following section.

Analysis of Growth Rate for ID IGA Degradation

The ID IGA degradation present in the TMI-1 steam generators is unique with respect to other known damage modes affecting pressurized-water reactor steam generator tubes. The degradation at TMI-1 was introduced as a result of a chemistry excursion that occurred during plant shutdown years ago. Sodium thiosulfate in the reactor coolant caused significant cracking in many of the steam generator tubes. All of the tubes returned to service were repaired. However, some tubes currently in service contain indications of ID IGA that are the result of the past chemistry event.

Reference 4 provided the results of a growth rate analysis that was performed for the ID IGA flaws during the 12R Outage. Assessments of volumetric ID IGA growth rates based on changes in bobbin coil voltage, MRPC-indicated axial length, and MRPC-indicated circumferential length were performed. Each of these assessments indicated that there was no statistically significant growth in the ID IGA; the changes were less than the statistical uncertainty of the measurement techniques. For the forthcoming Outage 13R, GPU Nuclear will perform the same types of growth assessments (i.e., of changes in voltage, axial extent, and circumferential extent).

GPU Nuclear has reviewed eddy current inspection data of the tubes with known ID IGA indications to assess degradation growth rates. The signal characteristics (e.g., voltage amplitude, phase angle) have remained approximately constant for the known population of degraded tubes for several cycles of operation indicating no change in the degradation. In addition, considering that the root cause of the degradation was the inadvertent introduction of sodium thiosulfate into the plant during shutdown, and that these chemicals have been removed from the plant design and coolant, the driving mechanism for this mode of degradation is no longer present. In addition, the metallurgical and chemical analyses that were performed on the 12R Outage pulled tubes suggest strongly that there was no progression in ID volumetric degradation (Reference 6). Therefore, GPU Nuclear concluded that the ID IGA indications were not growing in either size or depth.

Classification of Inspection Results

The current TMI-1 TS require the classification of inspection results based on the number of degraded or defective steam generator tubes identified during inservice inspections per TS 4.19.2. The results are classified as C-1, C-2, or C-3, and depending on the classification, GPU Nuclear may be required to complete additional inspections during an outage. TS 4.19.2 exempts previously degraded tubes from the inspection results classification except those tubes with indications that have exhibited significant growth from the previous examinations. The term "significant" is defined within the present TS requirement as any indication that increases in depth by greater than 10 percent through-wall. GPUN's current TS is also based on criteria that are other than depth-based; TS 4.19.2 was modified prior to Outage 12R to also define "significant" as an increase in bobbin coil voltage for ID IGA indications of greater than 0.6 volts. [This value was determined based on a statistical review of the outage-to-outage variation in the voltage for ID IGA indications. Historically, a significant fraction of the ID IGA indications exhibited voltage changes less than this value. Therefore, any ID IGA indications that increase in voltage greater than this threshold are more likely to have grown from the previous inspection.] Under the proposed TS change, the 0.6 volts increase threshold is effectively unchanged except that it is adjusted to 0.24 volts to reflect the proposed new 4 volt normalization. (Applying a 4:10 volt ratio to the 0.6 volt increase threshold yields a 0.24 volt result.) Again, the indication voltages from prior examinations will be adjusted for proper comparison to data acquired under the new 4 volt normalization procedure.

Outage 12R Findings

GPU Nuclear used the subject repair criteria during the 1997 Outage 12R examinations with a 10 volt bobbin coil normalization. 100 percent of the inservice steam generator tubes were inspected with bobbin coil eddy current probes. Reference 4 provides information on the number of tubes that were removed from service as a result of ID IGA indications. While a number of tubes were plugged as a result of ID IGA indications which exceeded the 40 percent through-wall criterion, no ID IGA indications were found which exceeded the circumferential extent criterion, and only one tube was found with an indication that required repair as a result of exceeding the 0.25" axial extent criterion.

During Outage 12R, six TMI-1 volumetric ID IGA indications were in-situ pressure tested. This work is described in Reference 4, Tables III-20 and III-21. None of the indications leaked or burst at pressures simulating up to 3 times the normal operating delta P. A comparison of eddy current inspection data collected before and after testing indicated the degradation did not appear to have grown as a result of the in-situ pressure tests. This indicates that the tubes that were in-situ pressure tested have structural and leakage integrity margins beyond the load conditions encountered in the tests.

During Outage 12R, a tube (A 52-34) was pulled from the "A" TMI-1 steam generator that contained several volumetric ID IGA flaws. (This tube would have been repaired under the subject criteria as it contained an ID IGA flaw that was estimated as 50 percent through-wall by bobbin coil examination. Essentially the "full length" of the tube was pulled; the upper and lower tube sheet expansions of the tube were not pulled.) The purpose of the tube pull was to reconfirm the ability of bobbin eddy current to identify the important flaws, to confirm the ability of rotating probe eddy current to determine the morphology and conservatively estimate the extents of the flaws in the field, and to perform chemical and metallurgical tests on these flaws. Several of the volumetric ID IGA flaws from this tube were subjected to laboratory leak and burst tests. These tests were undertaken to supplement the in-situ pressure testing of ID IGA flaws that was performed in the generators during Outage 12R. The results of the laboratory leak and burst testing were provided to the NRC in Reference 5. None of the ID IGA flaws leaked at pressures simulating up to 3 times the normal operating delta P, with applied axial loads simulating those calculated for a hypothetical Main Steam Line Break (MSLB). No internal flaw bladders or supports were used for any of these leak or burst tests. The burst pressures of the tube sections containing ID IGA indications that were tested were above 10,000 psig, indicating that substantial structural margin exists for these flaws. These burst pressures are nearly equal to those of non-defective virgin tubing; recent burst tests of three non-defective virgin OTSG tube lengths had an average burst pressure of 11,216 psig.

The Outage 12R pulled tube laboratory results demonstrated that the MRPC probes are conservative in their evaluation of TMI-1's volumetric ID IGA axial and circumferential extents. For example, Table 1 compares the axial extents of the flaws in the TMI-1 pulled tube, as called by field MRPC eddy current, with the subsequent metallographic findings from the laboratory for those same flaws.

Table 1:
Comparison of Field Eddy Current Estimated and Laboratory Determined ID IGA Axial Extents of TMI-1 Pulled Tube Flaws

TMI-1 Tube A 52-34 Flaw Type (as called by MRPC)	Location (Field)	Field Axial Length (by MRPC)	Maximum Laboratory Axial Length (How Determined.)	Ratio of Field Axial Length to Lab Axial Length
ID Volumetric IGA	7 + 36.8"	0.11"	0.024" (Radial Grinding / Photo Exam)	4.6
ID Volumetric IGA	13 + 23.1"	0.16"	0.033" (Radial Grinding / Photo Exam)	4.8
ID Volumetric IGA	14 + 12.8"	0.16"	0.054" (Radial Grinding / Photo Exam)	3.0
ID Volumetric IGA	14 + 31.9"	0.16"	0.042" (Radial Grinding / Photo Exam)	3.8
ID Volumetric IGA	15 + 14.7"	0.10"	0.029" (Radial Grinding / Photo Exam)	3.4
ID Volumetric IGA	15 + 24.9"	0.10"	0.030" (Radial Grinding / Photo Exam)	3.3
ID Volumetric IGA	UTS + 0.06"	0.20"	0.040" (Radial Grinding / Photo Exam)	5.0
ID Volumetric IGA	13 + 2.9"	0.10"	≅ 0.066" (By longitudinal grind at 1 of 2 fracture surfaces* / Photo exam)	≅ 1.5
ID Volumetric IGA	15 + 38.2"	0.10"	≅ 0.020" (By longitudinal grind at 1 of 2 fracture surfaces* / Photo exam)	≅ 5

* These two flaws were located at the fracture surfaces resulting from the laboratory burst testing. The flaws were torn in half during the testing and one of the two fracture surfaces was ground to determine the flaw morphology.

A similar degree of conservatism was noted for the circumferential extents as was seen for the axial extents. The following table compares the circumferential extent of the flaws in the TMI-1 pulled tube as called by field MRPC eddy current with the subsequent metallographic findings from the laboratory for the same flaws:

Table 2:
Comparison of Field Eddy Current Estimated and Laboratory Determined ID IGA Circumferential Extents of TMI-1 Pulled Tube Flaws

TMI-1 Tube A 52-34 Flaw Type (as called by MRPC)	Location (Field)	Field Circumferential Width (by MRPC)	Maximum Laboratory Circumferential Width (How Determined)	Ratio of Field Circumferential Width to Lab Circumferential Width
ID Volumetric IGA	7 + 36.8"	0.11"	0.022" (Radial Grinding/ Photo Exam)	5.0
ID Volumetric IGA	13 + 23.1"	0.11"	0.020" (Radial Grinding/ Photo Exam)	5.5
ID Volumetric IGA	14 + 12.8"	0.19"	0.025" (Radial Grinding/ Photo Exam)	7.6
ID Volumetric IGA	14 + 31.9"	0.11"	0.019" (Radial Grinding/ Photo Exam)	5.8
ID Volumetric IGA	15 + 14.7"	0.06"	0.018" (Radial Grinding/ Photo Exam)	3.3
ID Volumetric IGA	15 + 24.9"	0.11"	0.018" (Radial Grinding/ Photo Exam)	6.1
ID Volumetric IGA	UTS + 0.06"	0.14"	0.025" (Radial Grinding/ Photo Exam)	5.6
ID Volumetric IGA	13 + 2.9"	0.11"	≅ 0.032" (By longitudinal grind at 1 of 2 fracture surfaces* /Photo exam)	≅ 3.4
ID Volumetric IGA	15 + 38.2"	0.06"	≅ 0.016" (By longitudinal grind at 1 of 2 fracture surfaces* /Photo exam)	≅ 3.8

* As mentioned above, these two flaws were located at the fracture surfaces resulting from the laboratory burst testing. The flaws were torn in half during the testing and one of the two fracture surfaces was ground to determine the flaw morphology. The circumferential extent

listed in Table 2 for these two flaws were estimated by doubling the circumferential extent that was determined for one of the two fracture surfaces.

Tables 1 and 2 indicate that the MRPC probes are able to conservatively assess the axial and circumferential extents of the TMI-1 ID IGA flaws. In general, eddy current measurements of volumetric IGA axial and circumferential flaw length are typically conservative because of a combination of field effect and the degradation morphology. An eddy current coil interrogates a volume of tube material that is larger than the physical dimensions of the coil. Because a coil's electromagnetic field extends beyond the coil dimensions, inspection probes have the ability to detect degradation before and after the coil passes over the actual tube degradation. This has the effect of extending the measured bounds of tube degradation beyond the actual bounds of the degradation. As illustrated in Tables 1 and 2, this was the case for the pulled tube during Outage 12R. The ability to detect degradation before passing a coil over the affected tube areas also depends, in part, on the geometry of the degradation. The three-dimensional morphology of TMI-1's ID IGA tube degradation facilitates its detection by eddy current methods.

The TMI-1 ID IGA flaws that remain in service were inspected with the same MRPC probes and techniques as the pulled tube. Thus, it is reasonable to assume that axial and circumferential extents of the ID IGA flaws remaining in service were also conservatively evaluated with the MRPC probes used in Outage 12R.

Three of the ID IGA flaws in the Outage 12R pulled tube were detected by the bobbin probe and assigned through-wall estimates based on phase angle analysis prior to their removal from the generator. The following table compares the through-wall extent of those flaws in the TMI-1 pulled tube as called by bobbin coil phase angle analysis versus the subsequent metallographic findings from the laboratory for the same flaws. The depths in the table that were determined by radial grinds are prefaced by a less than ("<") sign. These signs are due to the fact that the some of the flaws were incrementally ground from the inside surface of the tubing until the flaws were no longer present. The depth of the flaws was then assigned as less than the grind depth at which the flaw is no longer present in the tube wall.

**Table 3:
Comparison of Field Eddy Current Estimated and Laboratory Determined ID IGA Depths of TMI-1 Pulled Tube Flaws**

TMI-1 Tube A 52-34 Flaw Type	Location (Field)	Field Depth based on Bobbin Coil Phase Angle Analysis (in percent through-wall)	Maximum Laboratory Depth in percent through-wall based on 0.037" wall thickness and in inches (How Determined)
ID Volumetric IGA	13 + 23.1"	37%	< 49% < 0.018" (Radial Grinding / Photo Exam)
ID Volumetric IGA (fracture surface at burst location)	13 + 2.9"	50%	30% 0.011" (By longitudinal grind at 1 of 2 fracture surfaces / Photo exam)
ID Volumetric IGA (fracture surface at burst location)	15 + 38.2"	17%	19% 0.007" (By longitudinal grind at 1 of 2 fracture surfaces / Photo exam)

Nine (9) ID IGA flaws from the pulled tube were examined in a laboratory to determine morphology (length, width, and depth). Tables 1 and 2 provided the axial and circumferential extent data on the nine flaws. Table 3, above, provided the depth data for the three (3) flaws that were assigned a depth estimate by the bobbin probe prior to pulling the tube. Table 4 provides the laboratory depth data for the other six (6) flaws, which had insufficient bobbin coil eddy current signal to be given depth estimates using the bobbin probe, or were not detected by the bobbin coil probe. It is noteworthy that none of the flaws represented in Table 4 (i.e., flaws that were not provided with estimated depths in the field) were deeper than the deepest flaw represented in Table 3.

Table 4:
Laboratory Determined ID IGA Depths of TMI-1 Pulled Tube Flaws without Field Eddy Current Depth Estimates

TMI-1 Tube A 52-34 Flaw Type	Location (Field)	Maximum Laboratory Depth in percent through-wall based on 0.037" wall thickness and in inches (How Determined)
ID Volumetric IGA	7 + 36.8"	< 32% < 0.012" Radial Grinding / Photo Exam
ID Volumetric IGA	14 + 12.8"	< 38% < 0.014" Radial Grinding / Photo Exam
ID Volumetric IGA	14 + 31.9"	< 32% < 0.012" Radial Grinding / Photo Exam
ID Volumetric IGA	15 + 14.7"	< 32% < 0.012" Radial Grinding / Photo Exam
ID Volumetric IGA	15 + 24.9"	< 43% < 0.016" Radial Grinding / Photo Exam
ID Volumetric IGA	UTS + 0.06"	< 38% < 0.014" Radial Grinding / Photo Exam

Tables 3 and 4 illustrate that many of the TMI-1 volumetric ID IGA flaws are too small to be detected with the bobbin coil probe, or may be too small to be assigned a through-wall extent using bobbin coil phase angle analysis The very small eddy current signal, and the resulting low signal-to-noise ratio, of many of these small flaws precludes reliable detection and/or through-wall extent estimates using the bobbin coil probe. The bobbin coil is, however, able to detect the larger ID IGA flaws; thus, the bobbin probe provides a screen with which the most significant flaws are detected. GPU Nuclear performed laboratory leak and burst testing on pulled tube flaws detected by the bobbin probe, as well as flaws that were not detected by the bobbin probe. Reference 5 reported the laboratory leak and burst test results to the NRC. All of the ID IGA flaws that were tested met the structural requirements required by Draft Regulatory Guide 1.121 (Reference 7), and demonstrated no leakage at 3 times normal operating delta P.

Laboratory analyses were performed on the Outage 12R pulled tube to assess whether the volumetric ID IGA flaws were actively growing or were the result of a dormant degradation mechanism. These analyses concluded that the ID IGA volumetric pits appear to be stable and not increasing in depth or extent with time.

GPU Nuclear presently attributes no operational leakage to the volumetric ID IGA flaws that remain in service. (TMI-1's operating leakage history since 1985 demonstrates that previous cycle primary-to-secondary leakage was corrected by repairs of leaking tube plugs.) With the results of the in-situ pressure tests that were performed in Outage 12R, and the results from additional laboratory leak and burst tests, GPU Nuclear has concluded that no increase in accident-induced leakage would be due to the presence of these flaws.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this TSCR involves no significant hazards consideration as defined by NRC in 10 CFR 50.92:

A. The proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed flaw disposition strategy, based on measurable eddy current parameters of axial and circumferential extent for Inside Diameter (ID) Initiated Inter-Granular Attack (IGA), will continue to provide high confidence that unacceptable flaws that do not have the required structural integrity to withstand a postulated MSLB are removed from service. The axial and circumferential length limits for eddy current ID degradation indications meet the Draft Regulatory Guide 1.121 (Reference 7) acceptance criteria for margin to failure for MSLB-applied differential pressure and axial tube loads. The capability for detection of flaws is unaffected; and the identification of tubes that should be repaired or removed from service is maintained. The operation of the OTSGs or related structures, systems, or components is otherwise unaffected. Therefore, neither the probability nor consequences of a SGTR is significantly increased either during normal operation or due to the limiting loads of a MSLB accident.

   Neither the change in voltage normalization for the eddy current examinations, nor the administrative change in clarification of the reporting requirements, as described above, could significantly affect the probability of occurrence or consequences of any accident previously evaluated. These changes are administrative only.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no hardware changes involved nor changes to any operating practices. These changes involve only the OTSG tube inservice inspection surveillance requirements, which could only affect the potential for OTSG primary-to-secondary leakage. The proposed changes continue to impose flaw length limits for ID IGA to assure tube structural and leakage integrity, as confirmed by 12R (and post 12R) tube pull sample examinations and pressure testing.

In addition, neither the change in voltage normalization for the eddy current examinations nor the administrative change in the description of the reporting requirements, as described above, could possibly create the possibility of an accident of a new or different type from any previously evaluated. These changes are included only to modify the plant's eddy current normalization to the industry standard, and clarify the reporting period for submittal of the OTSG inspection results to the NRC. Therefore, these changes do not create the potential for any other kind of accident different from those that have been evaluated.

C. These proposed changes do not involve a significant reduction in a margin of safety because the margins of safety defined in Draft Regulatory Guide 1.121 (Reference 7) are retained. The probability of detecting degradation is unchanged since the bobbin coil eddy current methods will continue to be the primary means of initial detection and the probability of leakage from any indications left in service remains acceptably small. The strategy for dispositioning ID initiated IGA will continue to provide a high level of confidence that tubes exceeding the allowable limits for tube integrity are repaired or removed from service.

In addition, neither the change in voltage normalization for the eddy current examinations nor the administrative change in the description of the reporting requirements, as described above, could significantly affect a margin of safety. These changes are administrative in nature and are included only to align TMI-1's voltage normalization to the industry standard, and clarify the reporting period, respectively.

VI. Environmental Impact Evaluation

10 CFR51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

GPU Nuclear has reviewed this TSCR and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment for changes affecting the surveillance specification changes in TS 4.19, "OTSG Tube Inservice Inspections," described above.

VI. Implementation

GPU Nuclear, Inc. requests that the amendment authorizing this change be issued by April 15, 1999 effective immediately and through Operating Cycle 13.

VII. References

(1) NRC Letter dated October 16, 1997, from Bart C. Buckley to James W. Langenbach, "Three Mile Island – Issuance of Amendment Re: Changes to the Technical Specifications Surveillance Requirements for Once – Through Steam Generator Inservice Inspection for Cycle 12 Operation (TAC No. M99392)," Amendment No. 206 to Facility Operating License No. DPR-50.

(2) U.S. NRC, Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," March 1998.

(3) NRC Letter dated December 21, 1984, from John F. Stolz to Henry D. Hukill, "License Amendment No. 103, "Steam Generator Tube Repair and Return to Operation, Three Mile Island Nuclear Station, Unit 1 (TMI-1)."

(4) GPU Nuclear Letter dated January 12, 1998, from James W. Langenbach to U. S. Nuclear Regulatory Commission, "Cycle 12 Refueling (12R) Outage Once Through Steam Generator (OTSG) Tube Inspection Report with ASME NIS Data Reports for Inservice Inspections (ISI)."

(5) GPU Nuclear Letter dated May 19, 1998, from James W. Langenbach to U. S. Nuclear Regulatory Commission, "Results from Cycle 12 Refueling (12R) Outage Pulled Tube Examinations."

(6) ABB Combustion Engineering Nuclear Operations Letter, PENG-98-230, letter dated October 23, 1998, from Philip House to Richard Freeman, "Transmittal of Metallographic Data and Pit Activity for TMI-1."

(7) U.S. NRC, Regulatory Guide 1.121, "Bases for Plugging PWR Steam Generator Tubes," (For Comment), August 1976.