NMC
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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

L-PI-03-089
October 31, 2003 10 CFR 50.90

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60

LICENSE AMENDMENT REQUEST (LAR) DATED October 31, 2003
CHANGES TO TECHNICAL SPECIFICATIONS TO IMPLEMENT NEI 97-06, "STEAM GENERATOR PROGRAM GUIDELINES", AND INSPECTION REQUIREMENTS ASSOCIATED WITH THE UNIT 1 REPLACEMENT STEAM GENERATORS

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC), hereby requests the following amendment to Appendices A and B of the Operating Licenses for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The proposed amendment would implement the industry guidance contained in Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines". The proposed amendment also separates the Unit 1 and Unit 2 requirements for steam generator (SG) tube repair criteria, inspections, and repair methods. This separation is necessary due to the different steam generator design being used for the Unit 1 replacement steam generators. The proposed amendment involves:

1. Revising the surveillance requirements in Technical Specification (TS) 3.4.14 and associated Bases, from verifying steam generator tube integrity to requiring verification that primary to secondary LEAKAGE is within limits.
2. Adding a new Technical Specification TS 3.4.19 entitled "Steam Generator Tube Integrity," and associated Bases. The proposed Technical Specification requires that SG tube integrity be maintained and requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

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3. Revising TS 5.5.8, "Steam Generator (SG) Tube Surveillance Program," to require a Steam Generator Program be established and implemented to ensure that SG tube integrity is maintained, and to describe SG condition monitoring and performance criteria. The revision also describes the repair criteria, repair methods, and inspection intervals that need to be made unit specific as a result of the new tubing material in the Unit 1 replacement steam generators. The title of TS 5.5.8 is also changed from Steam Generator Tube Surveillance Program to Steam Generator Program.
4. Changing the reporting requirements in TS 5.6.7, "Steam Generator Tube Inspection Report". The reporting requirements are revised to require a report within 180 days of initial entry into MODE 4 following a SG inspection.
5. Deleting requirements of Additional License Conditions associated with the voltage based repair criteria. They are no longer applicable to Unit 1 and have been incorporated into specification TS 5.5.8 for Unit 2.

As discussed in Exhibit A, the proposed changes replace an outdated prescriptive technical specification with one that references Steam Generator Program requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them. The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than required by the current Technical Specifications. As a result, these proposed changes will result in added assurance that steam generator tube integrity is maintained and that the tubes will be capable of performing their intended safety functions consistent with the plant's licensing basis, including applicable regulatory requirements.

Based on the discussion in the attached Exhibit A, the NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

The NMC requests approval of the proposed amendment by August 1, 2004.

Exhibit A contains the licensee's evaluation of this proposed change. Exhibit B presents the proposed Technical Specifications, associated Bases, and Additional Condition pages marked-up. Exhibit C presents the revised Technical Specification and Additional Condition pages incorporating the proposed changes. Exhibit D provides the commitments made in this License Amendment Request (LAR).

In accordance with 10 CFR 50.91, the NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

This license amendment request is similar to one requested by the Catawba Nuclear Station in 2003.

Please address any comments or questions regarding this LAR to Mr. H Oley Nelson at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on October 31, 2003

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector - Prairie Island Nuclear Generating Plant
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Attachments:

Exhibit A, Licensee Evaluation
Exhibit B, Proposed Technical Specifications, Bases and Additional Condition Changes (marked-up)
Exhibit C, Revised Technical Specifications, Bases and Additional Condition Changes
Exhibit D, List of Commitments

Exhibit A
Letter L-PI-03-089

LICENSEE EVALUATION

Subject: Changes to Technical Specifications to Implement NEI 97-06, "Steam Generator Program Guidelines", and Inspection Requirements Associated with the Unit 1 Replacement Seam Generators

1.0 DESCRIPTION

This letter is a request to amend the Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Unit 1 and Unit 2 respectively.

The proposed change would revise Specification 3.4.14, "RCS Operational LEAKAGE," Specification 5.5.8, "Steam Generator (SG)Tube Surveillance Program, and Specification 5.6.7, "Steam Generator Tube Inspection Report," and adds a new specification for steam generator tube integrity. The proposed changes are necessary in order to implement the guidance for the industry initiative contained in Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

The proposed change also deletes the additional conditions on the operating licenses associated with Amendments 133 for Unit 1 and 125 for Unit 2. These conditions dealt with requirements associated with the voltage based repair criteria. They are no longer applicable to Unit 1 and have been incorporated into Specification 5.5.8 for Unit 2.

2.0 PROPOSED CHANGE

Brief descriptions of the proposed changes are provided below. The specific wording changes to the Technical Specifications (TS), Bases, and Additional Conditions are provided in Exhibits B and C. The technical justifications for the changes are discussed in Section 4 below.

- Revise Technical Specification 3.4.14, "RCS Operational LEAKAGE"

  A new Note is added to SR 3.4.14.1 to indicate that this surveillance is not applicable to primary to secondary LEAKAGE.

  Surveillance Requirement (SR) SR 3.4.14.2 is changed from verifying steam generator (SG) tube integrity to requiring verification that primary to secondary LEAKAGE is within limit with a frequency of 72 hours. SG tube integrity is verified under a new Limiting Condition of Operation (LCO). A Note is added to SR 3.4.14.2 stating that the SR is not required to be performed until 12 hours after establishing steady state operations.

  TS Bases changes are made to reflect the changes proposed to the Technical Specifications.

- Add a Steam Generator Tube Integrity Specification

  The proposed change adds a new Technical Specification TS 3.4.19 entitled "Steam Generator (SG) Tube Integrity," and associated Bases. The proposed Specification requires that SG tube integrity be maintained and requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program. The proposed Specification also specifies the actions to be taken if one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.

- Revise Technical Specification 5.5.8, "Steam Generator (SG) Tube Surveillance Program"

  The proposed change revises TS 5.5.8, "Steam Generator Tube Surveillance Program," to require a Steam Generator Program to be established and implemented to ensure that SG tube integrity is maintained, and to describe SG condition monitoring and performance criteria. The revision also describes the repair criteria, repair methods, and inspection intervals that had to be made unit specific as a result of the different steam generator designs. The title of TS 5.5.8 is revised from Steam Generator Tube Surveillance Program to Steam Generator Program.

- Revise Technical Specification 5.6.7, "Steam Generator Tube Inspection Report"

  The proposed change to TS 5.6.7, "Steam Generator Tube Inspection Report," provides the requirements for and contents of the SG tube inspection report. The reporting requirements are revised to require a report within 180 days of initial entry into MODE 4 following a SG inspection.

- Revise the TS Bases for Specifications 3.4.4

  The TS Bases for Specification 3.4.4, "RCS Loops – MODES 1 and 2," are revised to eliminate the reference to the Steam Generator Tube Surveillance Program as the method of establishing Steam Generator OPERABILITY.

- Delete the requirements of Additional Conditions imposed by Amendment 133 for Unit 1 and 125 for Unit 2.

  These conditions dealt with requirements associated with the voltage based repair criteria. They are no longer applicable to Unit 1 and have been incorporated into specification 5.5.8 for Unit 2.

The following table provides more details of the proposed changes. It summarizes steam generator related conditions or requirements under the current licensing basis and under the proposed changes. It also identifies the subsection in Section 4 that contains the technical justification for the proposed change.

In summary, this License Amendment Request will make the necessary changes to implement the guidance for the industry initiative contained in document NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

This License Amendment Request is similar to that submitted by Duke Energy Corporation for the Catawba Nuclear Station, Units 1 and 2, in References 12, 14 & 15.

Condition or Requirement	Current Licensing Basis	Proposed Change	Technical Justification subsection
Reactor Coolant System (RCS) LEAKAGE determined by water inventory balance (SR 3.4.14.1)	Note states: Not required to be performed until 12 hours after establishment of steady state operation.	RCS Operational LEAKAGE TS new Note: Added new Note indicating SR not applicable to primary to secondary LEAKAGE.	4.1
SG Tube integrity verification (SR 3.4.14.2)	Verify in accordance with the Steam Generator Program.	RCS Operational LEAKAGE TS: Revised the SR to verify specified primary to secondary LEAKAGE every 72 hours. Added Note stating that the SR is not required to be performed until 12 hours after establishing steady state operations.	4.2
Frequency of verification of tube integrity (TS 5.5.8.c)	Up to 40 months depending on results of previous inspection or special conditions.	SG Tube Integrity TS 3.4.19 – Requires surveillance frequency in accordance with the TS 5.5.8, Steam Generator Program. Frequency is dependent on tubing material and the previous inspection results and the anticipated defect growth rate. TS 5.5.8 – Establishes maximum inspection intervals.	4.3

Condition or Requirement	Current Licensing Basis	Proposed Change	Technical Justification subsection
Tube sample selection (TS 5.5.8.b)	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum.	Steam Generator Program procedures - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – currently 20% of all tubes as a minimum.	4.4
Inspection techniques	Not specified.	SG Tube Integrity – SR 3.4.19.1 requires that tube integrity be verified in accordance with the Steam Generator Program.	4.5
		Steam Generator Program and implementing procedures: • Establishes requirements for qualifying Non-destructive Examination (NDE) techniques. • Requires use of qualified techniques in SG inspections. • Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.	

Condition or Requirement	Current Licensing Basis	Proposed Change	Technical Justification subsection
Inspection Scope (TS 5.5.8.d.1(h))	Hot leg point of entry completely around the U-bend to the top support of the cold leg.	Steam Generator Program procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of the entire length of tube from tube-sheet weld to tube-sheet weld.	4.6
Performance criteria (TS 3.4.14 and Additional Conditions for Operating License)	Operational leakage ≤ 150 gallons per day through any one SG. Accident induced leakage - During the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F). No criteria specified for structural integrity or accident induced leakage.	RCS Oper. LEAKAGE – No change SG Tube Integrity TS 3.4.19 – Requires that tube integrity be maintained. TS 5.5.8 – Defines structural integrity and enhances accident induced leakage performance criteria. Provides provisions for condition monitoring assessment to verify compliance.	4.7

Condition or Requirement	Current Licensing Basis	Proposed Change	Technical Justification subsection
Repair criteria (5.5.8.d)	Plug or repair tubes with imperfections extending ≥50% through wall (40% for general thinning) and specific degradation limits for F*, EF*, and voltage based criteria.	TS 5.5.8 – Criteria unchanged for Unit 2. For Unit 1, the only applicable criteria is flaws with a depth ≥ 40% of nominal tube wall thickness	4.8
Failure to meet performance or repair criteria (TS 3.4.14, and 5.5.8)	Performance Criteria not defined. Primary to secondary LEAKAGE limit and actions included in the TS. Plug or repair tubes exceeding repair criteria.	RCS Oper. LEAKAGE TS 3.4.14 and SG Tube Integrity TS 3.4.19 – Contains primary to secondary LEAKAGE limit, SG tube integrity requirements and ACTIONS required upon failure to meet performance criteria. SG Program TS 5.5.8 - Plug or repair tubes exceeding repair criteria.	4.9
Repair methods (TS 5.5.8.d.3)	Welded sleeving per Combustion Engineering Nuclear, CEN-629-P "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".	TS 5.5.8 –Methods (except plugging) require Nuclear Regulator Commission (NRC) approval. No change for Unit 2. For Unit 1, there will be no approved repair methods.	4.10

Condition or Requirement	Current Licensing Basis	Proposed Change	Technical Justification subsection
Reporting requirements (TS 5.6.7)	Plugging and repair report required 15 days after each inservice inspection, 90 day report documenting inspection results, and 30 day report when the inspection results fall into category C-3. Notifications required prior to operation of inspection results which fall into Category C-3, special conditions related to implementation of voltage based repair criteria.	Serious SG tube degradation (i.e., tubing fails to meet the structural integrity and accident induced leakage criteria) requires reporting in accordance with 10 CFR 50.72 and 50.73. TS 5.6.7 - 180 days after the initial entry into MODE 4 after performing a SG inspection. Retained notification required prior to operation for special conditions related to implementation of voltage based repair criteria for Unit 2.	4.11
Definitions SG Terminology (TS 1.1)	TS 1.1 does not address SG Program issues.	TS 5.5.8, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs.	4.12
Additional License Conditions (Amendment 133 for Unit 1, and 125 for Unit 2)	Imposed a voltage based repair criteria primary to secondary leakage limit of 1.42 gallons per minute.	Deleted for Unit 1. Incorporated into specification TS 5.5.8 for Unit 2.	4.13

3.0 BACKGROUND

The SG tubes in pressurized water reactors have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant's licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. When the degradation of the tube wall reaches a prescribed repair criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s and assumed uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criteria (e.g., 40 percent) that was incorporated into Prairie Island Nuclear Generating Plant's Technical Specifications and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 360° wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow the flexibility to manage different types of SG tube degradation. This through wall criterion is used for all forms of degradation unless approval is obtained for a more appropriate repair criterion that considers the structural integrity implications of the given mechanism, e.g., voltage based repair at tube support plates.

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- improved scope and methods for SG inspection,
- industry incentive to continue to improve inspection methods, and
- development of plugging and repair criteria based on appropriate NDE parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.

Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program. These guidelines include:

- EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guideline." (Reference 2),
- TR-107621, "Steam Generator Integrity Assessment Guideline" (Reference 3),
- TR-107620, "Steam Generator In-situ Pressure Test Guideline" (Reference 4),
- TR-104788, "PWR Primary-to-Secondary Leak Guideline" (Reference 5),
- TR-105714, "Primary Water Chemistry Guideline" (Reference 6), and
- TR-102134, "Secondary Water Chemistry Guideline" (Reference 7).

These EPRI Guidelines, along with NEI 97-06 (Reference 1), define the Steam Generator Program, while outlining a comprehensive, performance based approach to managing SG performance.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability.

This License Amendment Request implements the new regulatory framework and is consistent, to the extent practicable, with the Technical Specification Task Force (TSTF) Traveler 449.

In addition, the Nuclear Management Company, LLC will be replacing the Unit 1 original Westinghouse Model 51 steam generators that have been in service since commercial operation was achieved in 1973, with steam generators designed and fabricated by Framatome ANP. This replacement is scheduled to occur during an outage in the Fall of 2004. The replacement steam generators will utilize a different design than that in the Unit 2 steam generators, e.g., thermally treated Alloy 690 vs. mill annealed Alloy 600. Thus the repair criteria, repair methods, and inspection intervals for each unit will be different.

4.0 TECHNICAL ANALYSIS

The proposed changes do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The primary coolant activity limit and its assumptions are not affected by the proposed changes to the technical specifications. The proposed changes are an improvement to the existing SG inspection requirements and provide additional assurance that the plant's licensing basis will be maintained between SG inspections.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than those in TS 3.4.14 "RCS Operational LEAKAGE" plus the leakage rate associated with a double-ended rupture of a single tube.

For design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture).

The consequences of these design basis accidents are, in part, a function of the radioactivity levels in the primary coolant and the accident primary to secondary LEAKAGE rates. As a result, limits are included in the plant's technical specifications for operational LEAKAGE and for DOSE EQUIVALENT I-131 in primary coolant to ensure that the plant is operated within its analyzed condition.

TSTF-449 proposes technical specification changes that include a reduction in the primary to secondary leakage to 150 gallons per day. PINGP's current Technical Specification 3.4.14 already limits primary to secondary leakage to 150 gallon per day. Therefore the primary to secondary leakage limit does not need to be changed as part of this proposed license amendment request.

The technical specification changes proposed are in general a significant improvement over current requirements. They replace an outdated prescriptive technical specification with one that references Steam Generator Program requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than required by current technical specifications. As a result, these proposed changes will result in added assurance that the SG tubes will perform their functions and maintain their integrity.

4.1 RCS Operational LEAKAGE Determined by Water Inventory Balance

The proposed change adds a second Note to SR 3.4.14.1 that makes the water inventory balance method not applicable to determining primary to secondary LEAKAGE. This change is proposed because primary to secondary LEAKAGE as low as 150 gallons per day through any one SG cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the surveillance requirement appropriate for the LCO.

4.2 SG Tube Integrity Verification

Current SR 3.4.14.2 requires verification of tube integrity in accordance with the SG Tube Surveillance Program. This surveillance is no longer appropriate since tube integrity is addressed through the addition of a new SG Tube Integrity Specification, TS 3.4.19. Specification 3.4.14 now applies specifically to primary to secondary LEAKAGE. Surveillance Requirement 3.4.14.2 has been changed to verify the LCO requirement on primary to secondary LEAKAGE only. Steam generator tube integrity is verified in accordance with a SR in the SG Tube Integrity Specification, TS 3.4.19.

The SG program and the EPRI PWR primary to secondary leak guidelines (Reference 5) provide guidance on leak rate monitoring.

The proposed Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

4.3 Frequency of Verification of SG Tube Integrity

The current technical specifications contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The tube condition is classified into one of three categories based on the number of tubes found degraded and defective. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months. Additionally, unscheduled inspections are required if specific conditions such as exceeding primary to secondary leakage rates, seismic events, loss-of-coolant accidents, main steam line breaks, or feedwater line breaks occur.

The Surveillance Frequency in the proposed Steam Generator Tube Integrity specification is governed by the requirements in the Steam Generator Program and specifically by References 2 and 3. The proposed Frequency is also prescriptive, but has a stronger engineering basis than the existing technical specification requirements. The interval is dependent on tubing material and whether any active degradation is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established in Specification 5.5.8.

Maximum interval for Unit 1.

The Unit 1 replacement steam generators contain thermally treated Alloy 690 (690TT) tubing. Thus the maximum inspection interval requirement in Specification 5.5.8 for Unit 1 states:

> For the Unit 1 SGs, inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months (EFPMs). The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

Even though the maximum interval is longer than allowed by current technical specifications, it is only applicable to SGs with advanced tubing materials, and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection.

The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 8 shows that the performance of 690TT is significantly better than the performance of mill annealed Alloy 600 (600MA) tubing (the material used in the Unit 2 SG tubing). Taken in total, the proposed inspection intervals provide a larger margin of safety than the current requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the Steam Generator Program requires a minimum sample size (currently 20 percent) at each inspection that is significantly larger than that required by current technical specifications (3 percent); thus providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

Maximum interval for Unit 2.

The Unit 2 steam generators contain 600 MA tubing. Thus the maximum inspection interval requirement in Specification 5.5.8 for Unit 2 states:

> For the unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected. Each time a SG is inspected, all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the inspection requirements.

This frequency for the Unit 2 steam generators is as restrictive as the current technical specification requirement.

The completion of the first in-service examination defines the periods that remain fixed throughout the SG lifetime. For example, the periods for the Unit 1 steam generators would end 144, 252, 324, 384, 444, etc effective full power months after the first inservice inspection. The phrase "by the refueling outage nearest" is intended to mean that it is acceptable for the inspections to occur before or within ± ½ fuel cycle of the midpoint (or end) of a period.

In summary, the proposed change is an improvement over the current technical specifications. The current technical specifications base inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed technical specification uses information from previous plant inspections as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 3. The proposed inspection intervals maximize the potential that improved techniques and knowledge will be used since better knowledge of SG conditions supports longer intervals.

4.4 SG Tube Sample Selection

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes times the number of SGs in the plant. The proposed change refers to the Steam Generator Program degradation assessment guidance for sampling requirements. The minimum sample size is 20% of the tubes.

The Steam Generator Program requires the preparation of a degradation assessment before every SG inspection. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated prior to the outage. The degradation assessment addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary). In a degradation assessment, tube sample selection is performance based and is dependent upon actual SG conditions, plant operational experience, and industry experience. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI Steam Generator Examination Guidelines (Reference 2) and EPRI Steam Generator Integrity Assessment Guidelines (Reference 3) provide guidance on degradation assessment.

In general, the sample selection considerations required by the current technical specifications and the requirements in the Steam Generator Program as proposed by this change, are consistent, but the Steam Generator Program provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore the sample selection method proposed by this change is more conservative than the current technical specification requirements. In addition, the minimum sample size in the proposed requirements is larger.

4.5 SG Inspection Techniques

The surveillance requirements proposed in the Steam Generator Tube Integrity specification requires that tube integrity be verified in accordance with the requirements of the Steam Generator Program. The Steam Generator Program uses the EPRI Steam Generator Examination Guidelines (Reference 2) to establish requirements for qualifying NDE techniques.

The Steam Generator Program requires the performance of a degradation assessment before every SG inspection and refers utilities to EPRI Steam Generator Examination Guidelines (Reference 2) and EPRI Steam Generator Integrity Assessment Guidelines (Reference 3) for guidance on its performance. The degradation assessment will identify current and potential new degradation locations and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the *performance criteria*.

The current technical specifications contain no requirements on NDE inspection techniques. Therefore the proposed change is an improvement over the current technical specifications.

4.6 SG Inspection Scope

The current technical specifications include a definition of inspection that specifies the end points of the eddy current examination of each tube. The required inspection is from the hot leg point of entry completely around the U-bend to the top support of the cold leg. This definition is overly prescriptive and simplistic.

The provisions for steam generator tube inspections in proposed Technical Specification 5.5.8 states:

> Provisions for SG tube inspection. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may

satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

The Steam Generator Program provides extensive guidance and a defined process (the degradation assessment) for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

The proposed change is an improvement over the current technical specifications because it focuses the inspection effort on the areas of concern, thereby minimizing unnecessary data that the NDE analyst must review to identify indication of tube degradation.

4.7 SG Performance Criteria

The proposed change adds the requirement for a performance-based Steam Generator Program to the Technical Specifications. A performance-based approach has the following attributes:

- measurable parameters,
- objective criteria to assess performance based on risk-insights,
- deterministic analysis and/or performance history, and
- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. The structural integrity and accident induced leakage criteria were developed deterministically and are consistent with the plant's licensing basis. The operational LEAKAGE criterion was based on providing added assurance against tube rupture at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The performance criteria are specified in Specification 5.5.8. The requirements and methodologies established to meet the performance criteria are documented in the Steam Generator Program. The current technical specifications contain only the operational LEAKAGE criterion; therefore the proposed change is more restrictive than the current requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The structural integrity performance criterion in proposed Technical Specification 5.5.8 is:

> All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst. In the assessment of tube integrity, those loads that do significantly affect burst shall be determined and assessed in combination with the loads due to primary to secondary pressure differential using safety factors that are consistent with the licensing basis design criteria.

The structural integrity performance criterion is based on providing reasonable assurance that a SG tube will not burst during normal operation or postulated accident conditions.

Adjustments to include contributing loads are addressed in the applicable EPRI Guidelines to ensure that the evaluated or tested conditions are at least as severe as those expected during normal operating conditions and Level D accident events.

Normal steady state full power operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Future changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} will be assessed and included if significant.

The definition of normal steady state full power operation is important as it relates to application of the safety factor of three in the structural integrity performance criterion. The criterion requires "...retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential...". The application of the safety factor of three to normal steady state full power operation is based on past NRC positions, accepted industry practice, and the intent of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of three against burst. Additionally, the 3 times the primary to secondary pressure differential criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3 times the primary to secondary pressure differential limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI Steam Generator Integrity Assessment Guidelines (Reference 3).

The accident induced leakage performance criterion in proposed Technical Specification 5.5.8 is:

> The primary to secondary accident induced leakage rate for any design basis accidents, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42

gallons per minute (based on a reactor coolant system temperature of 578°F).

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences from primary to secondary LEAKAGE during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Part 100 guidelines, or the radiological limits to control room personnel imposed by General Design Criterion (GDC) 19.

The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as an initial condition. Recent experience with degradation mechanisms involving tube cracking has revealed that leakage under accident conditions can exceed the level of operating leakage by orders of magnitude. Therefore, a separate performance criterion was established for accident induced leakage. The limit for accident induced leakage is 1 gpm, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. Note that utilizing the voltage based repair criteria is not currently approved for the thermally treated 690 tubing material in the Unit 1 replacement steam generators.

The operational LEAKAGE performance criterion is specified in TS 3.4.14 and is not changing.

Proposed Administrative Specification 5.5.8 contains the performance criteria and is more restrictive than the current technical specifications. The current technical specifications do not address the structural integrity and accident induced leakage criteria.

4.8 SG Repair Criteria

Repair criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through-wall depth-based criterion (e.g., 40% through-wall) or through-wall depth based criteria for repair techniques, or other Alternate Repair Criteria for the voltage-based repair limit.

The surveillance requirements of the proposed Steam Generator Integrity Specification require that tubes that exceed approved tube repair criteria will be plugged or repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity should be assessed. It cannot be guaranteed that every flaw will be detected with a given eddy current technique and, therefore, it is possible that some flaws will not be detected during an inspection. The Steam Generator Program was not violated if a flaw is discovered and it is determined that this flaw would have met the repair criteria at the time of the last inspection of the affected tube.

There are no technical changes being made to the currently approved repair criteria for the Unit 2 steam generators. The criteria had to be reworded to fit the new format for Specification 5.5.8.

The F*, EF*, and voltage based repair criteria that are in the current technical specifications are not approved for the thermally treated Alloy 690 tubing material being used in the Unit 1 replacement steam generators. Hence the only applicable repair criterion for the Unit 1 SGs is a flaw with a depth equal to or exceeding 40% of the nominal tube wall thickness. The repair criteria of 40% nominal tube wall thickness is selected because it is a historically accepted value and satisfies the minimum wall thickness required by section NB-3324.1 of Reference 11.

4.9 Exceeding Performance or Repair Criteria

The RCS Operational LEAKAGE and Steam Generator Tube Integrity specifications require the licensee to monitor SG performance against performance criteria in accordance with the Steam Generator Program.

During plant operation, monitoring is performed using the operational LEAKAGE criterion. Exceeding that criterion will lead to a plant shutdown in accordance with Technical Specification 3.4.14. Once shutdown, the Steam Generator Program will ensure that the cause of the operational LEAKAGE is determined and corrective actions to prevent recurrence are taken. Operation may resume when the requirements of the Steam Generator Program have been met. This requirement is unchanged from the current technical specifications.

Also during plant operation an error or omission may be discovered that indicates a failure to implement a required plugging or repair during a previous SG inspection. Under these circumstances, the actions required by Condition A in the proposed Steam Generator Tube Integrity specification will be taken. If a performance criterion has been exceeded, then a principal safety barrier has been challenged and 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requires NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The Steam Generator Program additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle.

During MODES 5 and 6, the operational LEAKAGE criterion is not applicable, and the SGs will be inspected as required by the surveillance requirements in the Steam Generator Tube Integrity specification. A condition monitoring assessment of the "as found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident leakage performance criteria. If the performance criteria are not met, the Steam Generator Program requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met.

The current technical specifications do not address actions required while operating if it is discovered that the structural integrity or accident induced leakage performance criteria or a repair criterion are exceeded, so the proposed change is more conservative than the current technical specifications.

If performance or repair criteria are exceeded while shutdown, the affected tubes must be repaired or plugged. A report will be submitted to the NRC in accordance with Technical Specification 5.6.7. The changes in the required reports are discussed in Section 4.11 below.

4.10 SG Repair Methods

Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair.

The purpose of a repair is typically to reestablish or replace the RCPB. The proposed Steam Generator Tube Integrity surveillance requirements require that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with the Steam Generator Program. Repair methods established in accordance with the Steam Generator Program are listed in Technical Specification 5.5.8.

Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity is assessed. This requirement is unchanged by the proposed Technical Specifications.

The proposed TS 5.5.8.f does not change the approved sleeving repair method for Unit 2. This repair method has not been approved for the thermally treated Alloy 690 tubing material being used in the Unit 1 replacement steam generators. Hence there are no additional acceptable SG tube repair methods for the Unit 1 SGs.

Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

4.11 Reporting Requirements

The current technical specification TS 5.6.7 requires the following reports:

- A report listing the number of tubes plugged or repaired in each SG to be submitted within 15 days of the end of the inspection.
- Within 30 days of inspection which falls into Category C-3, a report shall be submitted describing the investigation into the cause and corrective measures to prevent the degradation.
- A SG inspection results report to be submitted within 90 days after the inspection.

The current technical specifications also require the following notification prior to operations:

- Results of inspection which fall into Category C-3.

The proposed change to Technical Specification 5.6.7 replaces the 15, 30, and 90 day inspection reports with one report required within 180 days. The proposed report also contains more information than the current SG inspection report. This provision expands the report to provide more substantive information to the NRC.

The guidance in NUREG-1022, Rev. 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," identifies serious SG tube degradation as an example of an event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. Steam generator tube degradation is considered serious if the tubing fails to meet structural integrity and accident induced leakage performance criteria. Serious SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed reporting requirements are an improvement as compared to those required by the current technical specifications. The proposed reporting requirements are more useful in identifying the degradation mechanisms and determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 (Reference 1) requires submitting additional information on the root cause of the condition and an Operational Assessment establishing the basis for operating during the next cycle.

The changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports being generated and sent to the NRC, while ensuring that critical information related to problems and significant tube degradation is reported more completely.

4.12 SG Terminology

The proposed Steam Generator Tube Integrity Specification Bases (TS B 3.4.19) explain a number of terms that are important to the function of a Steam Generator Program. The Technical Specification Bases are controlled by the Technical Specification Bases Control Program, which appears in the Administrative Technical Specifications.

The terms are described below.

1. Accident induced leakage rate means the primary to secondary LEAKAGE rate occurring during postulated accidents other than a steam generator tube rupture. This includes the primary to secondary LEAKAGE rate existing immediately prior to the accident plus additional primary to secondary LEAKAGE induced during the accident.

   Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR Part 100 guidelines for offsite doses, or the GDC-19 requirements for control room personnel, or other NRC approved licensing bases.

2. The LCO section of the Steam Generator Tube Integrity Bases define the term "burst" as:

   The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.

Since a burst definition is required for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is necessary. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation.

The definition developed for tube burst is consistent with the testimony of James Knight (Reference 9), and historical guidance of draft Regulatory Guide 1.121 (Reference 10). The definition of burst per these documents is in relation to gross failure of the pressure boundary; e.g., "the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME Code." Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The above definition of burst was chosen for a number of reasons:

- The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished via in situ testing. Since these tests do not have the capability to provide an unlimited water supply, or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

- The definition does not characterize local instability or "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament, (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for deep wear indications.

3. The LCO section of Steam Generator Tube Integrity Bases define a SG tube as, "...the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube nor is the region of tube below the F* and EF* distance."

   This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to Steam Generator Program requirements. The definition is also intended to exclude tube ends that cannot be NDE inspected by eddy current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques, if possible, or by using other methods, if necessary.

   For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

4.13 Additional License Conditions

Amendment 133 for Unit 1, and 125 for Unit 2, imposed the following license condition.

> NMC will assure that during the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).

The voltage based repair criteria is not approved for the thermally treated Alloy 690 tubing material being used in the Unit 1 replacement steam generators. Hence it is not included as a repair method for Unit 1 in Specification 5.5.8, and the above license condition is no longer applicable to Unit 1.

The voltage-based repair criteria primary to secondary leakage limit has been incorporated into the leakage performance criterion for Unit 2 in Specification 5.5.8. Hence the above license condition is no longer necessary for Unit 2.

Conclusion

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current technical specifications. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

Adopting the proposed changes will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company, LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

The change that is being evaluated below is the revision of Technical Specifications that will:

- Change the surveillance frequency of determining the primary to secondary leakage from every 24 hours to 72 hours.
- Add a new specification for steam generator tube integrity.
- Replace the Steam Generator Tube Surveillance Program requirements with requirements for a Steam Generator Program. The new program implements the guidance provided in the Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines".
- Modify the various steam generator tube inspection reporting requirements to provide more information 180 days after initial entry into MODE 4.
- Delete Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No

The proposed change would:

- Change the frequency of determining the primary to secondary leakage from every 24 hours to 72 hours.
- Add a new requirement for steam generator tube integrity.
- Replace the Steam Generator Tube Surveillance Program requirements with requirements for a Steam Generator Program. The new program implements the guidance provided in the Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines".
- Modify the various steam generator tube inspection reporting requirements to provide more information 180 days after initial entry into MODE 4.
- Delete Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits.

Determining the primary to secondary leakage is performed to assess one of the steam generator performance criteria discussed in the Steam Generator Program below. The frequency of performing that assessment is not an accident initiator, nor can it affect the consequences of an accident.

The new requirement for steam generator tube integrity is that the tube integrity shall be maintained and all tubes that satisfy the tube repair criteria are either repaired or plugged. Adding this requirement will not change the operation of or modify any equipment in the plant other than repairing or plugging steam generator tubes that met the defined repair criteria. Repairing or plugging steam generator tubes will ensure that the tubes remaining in service will meet the integrity requirements assumed in the previously evaluated accidents. Thus this requirement will not increase the probability or consequences of previously evaluated accidents.

The new Steam Generator Program includes performance criteria that will provide reasonable assurance that the steam generator tubing will retain its required integrity. The performance criteria are based on tube structural integrity and leakage. The tube structural integrity performance criteria will ensure that a tube does not burst due to the primary to secondary pressure difference. This performance criteria provides reasonable assurance that the steam generator tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. Thus the probability of a previously evaluated accident is not increased.

The leakage performance criteria will ensure that the primary to secondary leakage will not exceed the values assumed in the safety analysis. This will ensure that the consequences of a previously evaluated accident are not increased.

The Steam Generator Program includes repair criteria that provide a reasonable assurance that the performance criteria will be met until the next required inspection. The Steam Generator Program also relies on a stronger engineering basis for determining the inspection frequency and selecting the tubes to be inspected than the current steam generator tube surveillance program. The repair criteria, inspection frequency, and tube sample selection are all chosen so that the probability of an accident is not increased. The repair criteria, inspection frequency, and tube sample selection do not impact the evolution of an accident and therefore do not impact consequences of accidents.

Thus replacing the steam generator tube surveillance program with the Steam Generator Program will not increase the probability or consequences of an accident previously evaluated.

Steam generator tube inspection reports are an administrative requirement, and do not affect the probability or consequences of an accident. Thus modifying the requirements can not increase the probability or consequences of an accident previously evaluated.

Deletion of the Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits is an administrative change. Thus it can not increase the probability or consequences of an accident previously evaluated.

Therefore changing the frequency of determining the primary to secondary leakage, adding a steam generator tube integrity requirement, replacing the steam generator tube surveillance program with a Steam Generator Program, modifying the steam generator tube inspection reporting requirements, and deleting the Additional License conditions related to voltage-based repair criteria does not increase the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change would:

- Change the frequency of determining the primary to secondary leakage from every 24 hours to 72 hours.
- Add a new requirement for steam generator tube integrity.
- Replace the Steam Generator Tube Surveillance Program requirements with requirements for a Steam Generator Program. The new program implements the guidance provided in the Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines".
- Modify the various steam generator tube inspection reporting requirements to provide more information 180 days after initial entry into MODE 4.
- Delete Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits.

The proposed changes do not involve a physical change to the plant nor do they change the operation of the plant. Thus they cannot introduce a new failure mode.

Therefore changing the frequency of determining the primary to secondary leakage, adding a steam generator tube integrity requirement, replacing the steam generator tube surveillance program with a Steam generator Program, modifying the steam generator tube inspection reporting requirements, and deleting the Additional License conditions related to voltage-based repair criteria does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No

The proposed change would:

- Change the frequency of determining the primary to secondary leakage from every 24 hours to 72 hours.
- Add a new requirement for steam generator tube integrity.
- Replace the Steam Generator Tube Surveillance Program requirements with requirements for a Steam generator Program. The new program implements the guidance provided in the Nuclear Energy Institute document NEI 97-06, "Steam Generator Program Guidelines".
- Modify the various steam generator tube inspection reporting requirements to provide more information 180 days after initial entry into MODE 4.
- Delete Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits.

Determining the primary to secondary leakage is performed to assess one of the steam generator performance criteria discussed in the Steam Generator Program below. The frequency of performing the assessment does not alter actual integrity of the tubing, thus changing it does not impact the margin of safety.

The new requirement for steam generator tube integrity is that the tube integrity shall be maintained and all tubes that satisfy the tube repair criteria are either repaired or plugged. Adding this requirement will not change the operation of or modify any equipment in the plant other than repairing or plugging steam generator tubes that met the defined repair criteria. Repairing or plugging steam generator tubes will ensure that the tubes remaining in service will meet the integrity requirements assumed in the previously evaluated accidents. Maintaining the integrity requirements the steam generator tubes will maintain the margin of safety. Thus adding this requirement will not significantly reduce the margin of safety.

The new Steam Generator Program includes performance criteria that will provide reasonable assurance that the steam generator tubing will retain its required integrity. The performance criteria are based on tube structural integrity and leakage. The tube structural integrity performance criteria will ensure that a tube does not burst due to the primary to secondary pressure difference. The leakage performance criteria will ensure that the primary to secondary leakage will not exceed the values assumed in the safety analysis. The safety factors used in developing these criteria are not being changed from those used to establish the current tube repair and leakage limits. Thus the margin of safety is not reduced.

The Steam Generator Program includes repair criteria that provide a reasonable assurance that the performance criteria will be met until the next required inspection. The safety factors used in developing these repair criteria are not being changed from those used to establish the current tube repair limits.

The Steam Generator Program also relies on a stronger engineering basis for determining the inspection frequency and selecting the tubes to be inspected than the current steam generator tube surveillance program. The inspection frequency and tube sample selection are all chosen so that the margin of safety is not reduced.

Thus replacing the Steam Generator Tube surveillance program with the Steam Generator Program will not involve a significant reduction in the margin of safety.

Steam generator tube inspection reports are an administrative requirement, and do not affect the margin of safety. Thus modifying the requirements cannot reduce the margin of safety.

Deletion of the Additional License Conditions related to voltage-based repair criteria primary to secondary leakage limits is an administrative change. Thus it cannot reduce the margin of safety.

Therefore changing the frequency of determining the primary to secondary leakage, adding a steam generator tube integrity requirement, replacing the steam generator tube surveillance program with a Steam Generator Program, modifying the steam generator tube inspection reporting requirements, and deleting the Additional License conditions related to voltage-based repair criteria does not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory requirements applicable to SG tube integrity are the following:

10 CFR 50.55a: Codes and Standards

Section (b), American Society of Mechanical Engineers code.

Section (c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

The proposed changes and the Steam Generator Program requirements which underlie them are in full compliance with the American Society of Mechanical Engineers code. The proposed technical specifications are more effective at ensuring tube integrity and, therefore, compliance with the American Society of Mechanical Engineers code, than the current technical specifications.

10 CFR 50.65 Maintenance Rule – Each holder of a license to operate a nuclear power plant under 10 CFR 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken.

Under the Maintenance Rule, steam generators are classified as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria included in the proposed technical specifications are used to demonstrate that the condition of the steam generators "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule, 10 CFR 50.65 paragraph (a)(2)). If the performance criteria are not met, a root cause determination of appropriate depth is done and the results evaluated to determine if goals should be established per the Maintenance Rule, 10 CFR 50.65 paragraph (a)(1).

Nuclear Energy Institute document NEI 97-06, Steam Generator Program Guidelines, and its referenced Electric Power Research Institute guidelines define a steam generator program that provides the appropriate preventive maintenance that meets the intent of the Maintenance Rule. Nuclear Management and Resources Council, (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 13) offers guidance for implementing the Maintenance Rule should a licensee elect to incorporate additional monitoring goals beyond the scope of those documented in NEI 97-06.

NOTE: The Prairie Island Nuclear Generating Plant was designed and constructed to comply with Northern States Power Company's understanding of the intent of the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Hence the following proposed 1967 Atomic Energy Commission General Design Criteria are discussed.

<u>Proposed 1967 Atomic Energy Commission General Design Criteria 9</u> – *Reactor Coolant Pressure Boundary.* The reactor coolant pressure boundary shall be designed, and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

The evaluation below concludes that the proposed changes will continue to comply with this regulatory requirement.

<u>Proposed 1967 Atomic Energy Commission General Design Criteria 33</u> – *Reactor Coolant Pressure Boundary Capability.* The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The evaluation below concludes that the proposed changes will continue to comply with this regulatory requirement.

<u>Proposed 1967 Atomic Energy Commission General Design Criteria 36</u> – *Reactor Coolant Pressure Boundary Surveillance.* Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during service lifetime. For reactor vessel, a

material surveillance program conforming with ASTM-E-185-66 shall be provided.

The evaluation below concludes that the proposed changes will continue to comply with this regulatory requirement.

Evaluation of Proposed 1967 Atomic Energy Commission General Design Criteria 9, 33, & 36, define requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the reactor coolant pressure boundary surface area. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The Steam Generator Program required by the proposed technical specification establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity will be met in the interval between steam generator inspections. Thus the proposed changes will continue to comply with the proposed General Design Criteria to which Prairie Nuclear Plant was designed and constructed.

5.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10

CFR 51.22 (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines."
3. EPRI TR-107621, "Steam Generator Integrity Assessment Guideline."
4. EPRI TR-107620, "Steam Generator In-situ Pressure Test Guideline."
5. EPRI TR-104788, "PWR Primary-to-Secondary Leak Guideline."
6. EPRI TR-105714, "Primary Water Chemistry Guideline."
7. EPRI TR-102134, "Secondary Water Chemistry Guideline."
8. EPRI Report R-5515-00-2, "Experience of US and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves," June 5, 2002.
9. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.
10. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded Steam Generator Tubes," August 1976.
11. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components.
12. Letter from Duke Energy Corporation to NRC, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, "Proposed Technical Specifications (TS) Amendments Revision to Steam Generator TS" February 25, 2003.
13. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

14. Letter from Duke Energy Corporation to NRC, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, "Proposed Technical Specifications (TS) Amendments Revision to Steam Generator TS" June 9, 2003.

15. Letter from Duke Energy Corporation to NRC, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, "Proposed Technical Specifications (TS) Amendments Revision to Steam Generator TS" July 30, 2003.

Exhibit B
Letter L-PI-03-089

Proposed Technical Specifications, Bases, and Additional Condition Changes
(marked-up)
(Additions shaded, deletions strikethrough)

Subject: Changes to Technical Specifications to Implement NEI 97-06, "Steam Generator Program Guidelines", and Inspection Requirements Associated with the Unit 1 Replacement Steam Generators

<u>Appendix A Technical Specification pages</u>
(page numbers are as shown in exhibit B)

3.4.14-3
3.4.19-1 & 3.4.19-2
5.0-13 through 5.0-27
5.0-35 & 5.0-36
5.0-38 through 5.0-40

<u>Appendix B, Additional Conditions</u>
(page numbers are as shown in exhibit B)

Facility Operating License No. DPR-42 (Unit 1) B-1
Facility Operating License No. DPR-60 (Unit 2) B-1

<u>Technical Specification Bases pages</u>
(page numbers are as shown in exhibit B)

B 3.4.4-2
B 3.4.14-2
B 3.4.14-4
B 3.4.14-7 through B 3.4.14-9
B 3.4.19-1 through B 3.4.19-9

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	---NOTES--- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational leakage within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	---NOTE--- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify ~~steam generator tube integrity is in accordance with the Steam Generator Program~~primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator.	~~In accordance with the Steam Generator Program~~ 72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG inspection

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst. In the assessment of tube integrity, those loads that do significantly affect burst shall be determined and assessed in combination with the

loads due to primary to secondary pressure differential using safety factors that are consistent with the licensing basis design criteria.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F).

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational Leakage".

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

2. Unit 2 steam generator tubes that meet the following criteria shall be plugged or repaired.

~~Steam generator tubes in each unit shall be determined OPERABLE by the following:~~

~~a. Steam Generator Sample Selection and Inspection~~

~~Each steam generator shall be determined OPERABLE in accordance with the in-service inspection schedule in Specification 5.5.8.c. The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.~~

~~b. Steam Generator Tube Sample Selection and Inspection~~

~~The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.5.8-1 and 5.5.8-2. The in-service inspection of steam generator tubes shall be performed at the Frequencies specified in Specification 5.5.8.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.d. The tubes selected for each in-service inspection shall include at least 3% of the total number of tubes in all steam generators and at least 20% of the total number of sleeves in service in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

~~1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.~~

~~2. The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(a) all tubes that previously had detectable wall penetrations (> 20%) that have not been plugged or sleeve repaired in the affected area.~~

~~(b) tubes in those areas where experience has indicated potential problems.~~

~~(c) a tube inspection (pursuant to Specification 5.5.8.d.1.(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~3. In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).~~

~~4. The tubes selected as the second and third samples (if required by Tables 5.5.8-1 or 5.5.8-2) during each in-service inspection may be subjected to a partial tube or sleeve inspection provided:~~

~~(a) the tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.~~

~~(b) the inspections include those portions of the tubes or sleeves where imperfections were previously found.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~The results of each sample inspection shall be classified into one of the following three categories:~~

~~Category~~	~~Inspection Results~~
~~C-1~~	~~Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.~~
~~C-2~~	~~One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.~~
~~C-3~~	~~More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.~~

~~Note: In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.~~

~~5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.~~

~~6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~c. Inspection Frequencies~~

~~The above required in-service inspections of steam generator tubes shall be performed at the following Frequencies:~~

~~1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.~~

~~2. If the results of the in-service inspection of a steam generator conducted in accordance with Table 5.5.8-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.c.1; the interval may then be extended to a maximum of once per 40 months.~~

~~3. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:~~

~~(a) primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.~~

~~(b) a seismic occurrence greater than the Operating Basis Earthquake.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(c) a loss-of-coolant accident requiring actuation of the engineered safeguards.~~

~~(d) a main steam line or feedwater line break.~~

~~d. Acceptance Criteria~~

~~1. As used in this Specification:~~

~~(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.~~

~~(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~(c) Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.~~

~~(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is~~

(a) **Tubes found by inservice inspection containing flaws with a depth** equal to **or exceeding** 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this ~~criteria will be~~**criterion** is reduced to 40% wall penetration. **This criterion does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. It also**~~This definition~~ does not apply to the portion of the tube in the tubesheet below the F* **or EF*** distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance~~for F* or EF* tubes~~. **The F* distance is defined as the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet. The EF* distance is defined as the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). The EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.**

(b) **Tubes found by inservice inspection containing flaws in** ~~The repair limit for~~ the pressure boundary region of any sleeve **with a depth equal to or exceeding** ~~is~~ 25% of the nominal sleeve wall thickness. ~~This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.5.8.d.4 for~~

~~the repair limit applicable to these intersections.~~

(c) **Tubes found by inservice inspection that are experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates.**

~~(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.~~

~~(h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.~~

~~(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.~~

~~(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.~~

~~(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.~~

~~(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.~~

~~2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8-1 and 5.5.8-2.~~

~~3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:~~

~~CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak-Tight Sleeves".~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:~~

~~(a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 Volts will be allowed to remain in service.~~

~~(b) Steam generator tubes,~~

i. ~~whose~~ **with indications of potential** degradation ~~is~~ attributed to **predominately axially oriented** outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts **unless no degradation is detected with a rotating pancake coil (or comparable examination technique) inspection.**

~~, will be repaired or plugged, except as noted in Specification 5.5.8.d.4(c) below.~~

~~(c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes,~~

ii. with indications of **predominately axially oriented** outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit. ~~will be plugged or repaired.~~

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program (continued)

iii. ~~(d)~~ **inspected during** ~~If~~ an unscheduled mid-cycle inspection~~ is performed~~, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.**c.2.(c).i**~~d.4(a), (b)~~ and **5.5.8.c.2.(c).ii** above~~(e)~~. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program (continued)

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2.(c).i ~~d.4(a), (b)~~ and 5.5.8.c.2.(c).ii above~~(c)~~.

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

d. Provisions for SG tube inspection. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following a SG replacement.

2. For the Unit 1 SGs, inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. For the Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected. Each time a SG is inspected, all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the inspection requirements.

4. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. There are no approved SG tube repair methods for the Unit 1 SGs.

2. An approved SG tube repair method for the Unit 2 SGs is the use of welded sleeves in accordance with the methods described in CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

~~Table 5.5.8-1~~

~~STEAM GENERATOR TUBE INSPECTION~~

~~1st SAMPLE INSPECTION~~			~~2nd SAMPLE INSPECTION~~		~~3rd SAMPLE INSPECTION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of S Tubes per S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Repair defective tubes and inspect additional 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Repair defective tubes and inspect additional 4S tubes in this S.G.~~	~~C-1~~	None
					~~C-2~~	~~Repair defective tubes~~
					~~C-3~~	~~Perform action for C-3 result of first sample~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~	~~N/A~~	~~N/A~~
	~~C-3~~	~~Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G.~~ ~~Prompt notification to NRC~~	~~All other S.G.s are C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~Some S.G.s C-2 but no additional S.G. are C-3~~	~~Perform action for C-2 result of second sample~~	N/A	N/A
			~~Additional S.G. is C-3~~	~~Inspect all tubes in each S.G. and repair defective tubes. Prompt notification to NRC.~~	~~N/A~~	~~N/A~~

~~S=3%; When two steam generators are inspected during that outage.~~
~~S=6%; When one steam generator is inspected during that outage.~~

~~Table 5.5.8-2~~

~~STEAM GENERATOR TUBE SLEEVE INSPECTION~~

~~1st Sample Inspection~~	~~2nd Sample Inspection~~

~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of 20% of Tube Sleeves (1)~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.~~	~~C-1~~	~~None~~
			~~C-2~~	~~Plug or repair defective sleeved tubes~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes~~	~~The other S.G. is C-1~~	~~None~~
			~~The other S.G. is C-2~~	~~Perform action for C-2 result of first sample~~
			~~The other S.G. is C-3~~	~~Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes~~

~~(1) Each type of sleeve is considered a separate population for determination of scope expansion~~

5.6 Reporting Requirements (continued)

5.6.7 Steam Generator Tube Inspection Report

~~1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.~~

~~2. The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.~~

~~3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:~~

~~a. Identification of F* and EF* tubes, and~~

~~b. Location and extent of degradation.~~

~~5.6 Reporting Requirements~~

~~5.6.7 Steam Generator Tube Inspection Report (continued)~~

5~~a~~. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1~~a~~. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

2~~b~~. If circumferential crack-like indications are detected at the tube support plate intersections.

3~~c~~. If indications are identified that extend beyond the confines of the tube support plate.

4~~d~~. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5~~e~~. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

b. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged or repaired to date,

7. The results of condition monitoring including the results of tube pulls and in-situ testing,

8. The effective plugging percentage for all plugging and tube repairs in each SG, and

9. Repair method utilized and the number of tubes repaired by each repair method.

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Nuclear Management Company, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
128	1. NSP* will provide a licensed operator in the control room on an interim basis for the dedicated purpose of identifying an earthquake which results in a decreasing safeguards cooling water bay level. This operator will be in addition to the normal NSP administrative control room staffing requirements and will be provided until License Condition 2 is satisfied.	Prior to Unit 2 entering Mode 2 Completed – See Amendment No. 140
128	2. NSP* will submit dynamic finite element analyses of the intake canal banks by July 1, 1997 for NRC review. By December 31, 1998, NSP will complete, as required, additional analyses or physical modifications which provide the basis for extending the time for operator post-seismic cooling water load management and eliminating the dedicated operator specified in License Condition 1.	July 1, 1997, and December 31, 1998, as stated in Condition 2. Completed – See Amendment No. 140
128	3. Based on the results of License Condition 2, NSP* will revise the Updated Safety Analysis Report to incorporate the changes into the plant design bases. These changes will be included in the next scheduled revision of the Updated Safety Analysis Report following completion of License Condition 2 activities.	At the next USAR update following completion of Condition 2, but no later than June 1, 1999.
130	4. Prairie Island will assure that heavy loads do not present a potential for damaging irradiated fuel through use of: 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.	This is effective immediately upon issuance of the amendment.
~~133~~Deleted	5. ~~NMC will assure that during the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).~~Deleted	~~This is effective immediately upon issuance of the amendment~~Deleted

*Reference to NSP is maintained for historical purposes.

Amendment No. ~~153~~

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Nuclear Management Company, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
120	1. NSP* will provide a licensed operator in the control room on an interim basis for the dedicated purpose of identifying an earthquake which results in a decreasing safeguards cooling water bay level. This operator will be in addition to the normal NSP administrative control room staffing requirements and will be provided until License Condition 2 is satisfied.	Prior to Unit 2 entering Mode 2 Completed – See Amendment No. 131
120	2. NSP* will submit dynamic finite element analyses of the intake canal banks by July 1, 1997 for NRC review. By December 31, 1998, NSP will complete, as required, additional analyses or physical modifications which provide the basis for extending the time for operator post-seismic cooling water load management and eliminating the dedicated operator specified in License Condition 1.	July 1, 1997, and December 31, 1998, as stated in Condition 2. Completed – See Amendment No. 131.
120	3. Based on the results of License Condition 2, NSP* will revise the Updated Safety Analysis Report to incorporate the changes into the plant design bases. These changes will be included in the next scheduled revision of the Updated Safety Analysis Report following completion of License Condition 2 activities.	At the USAR update following completion of Condition 2, but no later than June 1, 1999.
122	4. Prairie Island will assure that heavy loads do not present a potential for damaging irradiated fuel through use of: 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.	This is effective immediately upon issuance of the amendment.
~~125~~Deleted	5. ~~NMC will assure that during the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).~~Deleted	~~This is effective immediately upon issuance of the amendment~~Deleted

*Reference to NSP is maintained for historical purposes.

Amendment No. ~~144~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses include the effect of flow on the departure from nucleate boiling ratio (DNBR). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, misaligned rod, and rod withdrawal events (Ref. 1).

The plant is designed to operate with both RCS loops in operation to maintain DNBR within limits during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis~~e 1 gpm primary to secondary LEAKAGE as the initial condition~~.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm (at 70°F) primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One Steam Generator (SG)

~~The 150 gallons per day (gpd) limit on one SG is based on implementation of the Steam Generator Voltage Based Alternate Repair Criteria and is more restrictive than standard operating leakage limits to provide additional margin to accommodate a crack which might grow at greater than the expected rate or unexpectedly extend outside the thickness of the tube support plate.~~

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be met with the reactor at steady state operating condition (stable RCS pressure, temperature, power level, ~~equilibrium xenon,~~ pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. ~~Therefore, a~~ Note 1 states ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

The 24 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.14.2

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions~~This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.19, "Steam Generator (SG) Tube Integrity" should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 3. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary Leakage and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling.

BASES

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2.
2. USAR, Section 14.5.
3. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
4. NEI 97-06, "Steam Generator Program Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria

BASES

BACKGROUND (continued)

are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than the operational LEAKAGE rate limits in LCO 3.4.14, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F). For accident analyses that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to or greater than the LCO 3.4.17, "RCS Specific Activity," limits. For accident analyses that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Ref. 3), or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. This portion of the LCO is meant to apply to all inservice tubes. It does not apply to tubes that have been removed from service via plugging. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube, nor is the region of tube below the F* and EF* distance.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

LCO
(continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed those discussed in the APPLICABLE SAFETY ANALYSES section above. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.14, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount

BASES

LCO (continued)

would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, and 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG tube inspections and assessments of degradation performed per Specification 5.5.8 include allowances for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met

BASES

ACTIONS (continued)

until the next SG inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tube(s). However, the affected tube(s) must be plugged or repaired prior to entering MODE 4 following the next refueling outage or SG inspection (which ever occurs first). This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected.) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.19.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The SG tube inspections and assessments of degradation performed per Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 and Reference 6 provide guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

BASES

REFERENCES (continued)

5. **Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.**

6. **EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines."**

Exhibit C
Letter L-PI-03-089

Revised Technical Specifications, Bases and Additional Condition Changes

Subject: Changes to Technical Specifications to Implement NEI 97-06, "Steam Generator Program Guidelines", and Inspection Requirements Associated with the Unit 1 Replacement Steam Generators

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	---------------------------NOTES------------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational leakage within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	---------------------------NOTE------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG inspection

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst. In the assessment of tube integrity, those loads that do significantly affect burst shall be determined and assessed in combination with the

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

loads due to primary to secondary pressure differential using safety factors that are consistent with the licensing basis design criteria.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F).

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational Leakage."

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

2. Unit 2 steam generator tubes that meet the following criteria shall be plugged or repaired.

(a) Tubes found by inservice inspection containing flaws with depth equal to or exceeding 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criterion is reduced to 40% wall penetration. This criterion does not apply to tube support plate intersections for which

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

the voltage-based repair criteria are being applied. It also does not apply to the portion of the tube in the tubesheet below the F* or EF* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance. The F* distance is defined as the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet. The EF* distance is defined as the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). The EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.

(b) Tubes found by inservice inspection containing flaws in the pressure boundary region of any sleeve with a depth equal to or exceeding 25% of the nominal sleeve wall thickness.

(c) Tubes found by inservice inspection that are experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates:

i. with indications of potential degradation attributed to predominantly axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage greater than 2.0 Volts unless no degradation is detected with a rotating pancake coil (or comparable examination technique) inspection.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

ii. with indications of predominantly axially oriented outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit.

iii. inspected during an unscheduled mid-cycle inspection, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.c.2.(c).i and 5.5.8.c.2.(c).ii above. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2.(c).i and 5.5.8.c.2.(c).ii above.

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

d. Provisions for SG tube inspection. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage following a SG replacement.

2. For the Unit 1 SGs, inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. For the Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected. Each time a SG is inspected, all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the inspection requirements.

4. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

   1. There are no approved SG tube repair methods for the Unit 1 SGs.

   2. An approved SG tube repair method for the Unit 2 SGs is the use of welded sleeves in accordance with the methods described in CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

5.5 Programs and Manuals (continued)

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5.5 Programs and Manuals (continued)

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5.6 Reporting Requirements (continued)

5.6.7 Steam Generator Tube Inspection Report

a. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

b. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged or repaired to date,

7. The results of condition monitoring including the results of tube pulls and in-situ testing,

8. The effective plugging percentage for all plugging and tube repairs in each SG, and

9. Repair method utilized and the number of tubes repaired by each repair method.

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Nuclear Management Company, LLC shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
120	1. NSP* will provide a licensed operator in the control room on an interim basis for the dedicated purpose of identifying an earthquake which results in a decreasing safeguards cooling water bay level. This operator will be in addition to the normal NSP administrative control room staffing requirements and will be provided until License Condition 2 is satisfied.	Prior to Unit 2 entering Mode 2 Completed – See Amendment No. 131
120	2. NSP* will submit dynamic finite element analyses of the intake canal banks by July 1, 1997 for NRC review. By December 31, 1998, NSP will complete, as required, additional analyses or physical modifications which provide the basis for extending the time for operator post-seismic cooling water load management and eliminating the dedicated operator specified in License Condition 1.	July 1, 1997, and December 31, 1998, as stated in Condition 2. Completed – See Amendment No. 131.
120	3. Based on the results of License Condition 2, NSP* will revise the Updated Safety Analysis Report to incorporate the changes into the plant design bases. These changes will be included in the next scheduled revision of the Updated Safety Analysis Report following completion of License Condition 2 activities.	At the USAR update following completion of Condition 2, but no later than June 1, 1999.
122	4. Prairie Island will assure that heavy loads do not present a potential for damaging irradiated fuel through use of: 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.	This is effective immediately upon issuance of the amendment.
Deleted	5. Deleted	Deleted

*Reference to NSP is maintained for historical purposes.

APPENDIX B

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Deleted	5. Deleted	Deleted

*Reference to NSP is maintained for historical purposes.

BASES

APPLICABLE SAFETY ANALYSES (continued)	forced flow rate, which is represented by the number of RCS loops in service. Both transient and steady state analyses include the effect of flow on the departure from nucleate boiling ratio (DNBR). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, misaligned rod, and rod withdrawal events (Ref. 1). The plant is designed to operate with both RCS loops in operation to maintain DNBR within limits during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops - MODES 1 and 2 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at power. An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.
APPLICABILITY	In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increase to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm (at 70°F) primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One Steam Generator (SG)

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE in excess of the LCO limits must be identified or reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1, B.2.1, and B.2.2

If unidentified LEAKAGE cannot be identified or cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals, gaskets, and pressurizer safety valves seats is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours. If the LEAKAGE source cannot be identified within 54 hours, then the reactor must be placed in MODE 5 within 84 hours. This action reduces the LEAKAGE and also reduces the

BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued)

factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

C.1, C.2.1, and C.2.2

If RCS identified LEAKAGE, other than pressure boundary leakage, is not within limits, then the reactor must be placed in MODE 3 within 6 hours. In this condition, 14 hours are allowed to reduce the identified leakage to within limits. If the identified LEAKAGE is not within limits within this time, the reactor must be placed in MODE 5 within 44 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

D.1 and D.2

If RCS pressure boundary LEAKAGE exists or if SG LEAKAGE (150 gpd limit) is not within limits, the reactor must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating condition (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

BASES

SURVEILALNCE REQUIREMENTS

SR 3.4.14.1 (continued)

The 24 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.14.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.19, "Steam Generator Tube (SG) Integrity" should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 3. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary Leakage and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2.

2. USAR, Section 14.5.

3. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines.

4. NEI 97-06, "Steam Generator Program Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

BASES

BACKGROUND (continued)	Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).
APPLICABLE SAFETY ANALYSES	The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than the operational LEAKAGE rate limits in LCO 3.4.14, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed

BASES

APPLICABLE SAFETY ANALYSES (continued)	1.42 gallons per minute (based on a reactor coolant system temperature of 578°F). For accident analyses, that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to or greater than the LCO 3.4.17, "RCS Specific Activity," limits. For accident analyses that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), or the NRC approved licensing basis (e.g., a small fraction of these limits). Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that SG tube integrity be maintained. This portion of the LCO is meant to apply to all inservice tubes. It does not apply to tubes that have been removed from service via plugging. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program. During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity. In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube, nor is the region of tube below the F* and EF* distance.

BASES

LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO. The structural integrity performance criterion provides a margin of safety against tube burst under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions.

BASES

LCO (continued)

The accident analysis assumes that accident induced leakage does not exceed those discussed in the APPLICABLE SAFETY ANALYSES section above. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.14, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, and 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG tube inspections and assessments of degradation performed per Specification 5.5.8 include allowances for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next SG inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

BASES

ACTIONS

A.1 and A.2 (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tube(s). However, the affected tube(s) must be plugged or repaired prior to entering MODE 4 following the next refueling outage or SG inspection (which ever occurs first). This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected.) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1 (continued)

The Steam Generator Program defines the Frequency of SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.19.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The SG tube inspections and assessments of degradation performed per Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 and Reference 6 provide guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.2 (continued)

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Exhibit D
Letter L-PI-03-089

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

LIST OF COMMITMENTS

The following table identifies those actions committed to by Nuclear Management Company, LLC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments.

REGULATORY COMMITMENT	DUE DATE
None	