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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

September 26, 2007

L-MT-07-071
10 CFR 50.55a(g)(5)(iii)

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

10 CFR 50.55a Request No. 15: Relief from Impractical Examination Coverage Requirements Pursuant to 10 CFR 50.55a(g)(5)(iii) for the Fourth Ten-Year Inservice Inspection Interval

Pursuant to 10 CFR 50.55a(g)(5)(iii), the Nuclear Management Company, LLC (NMC) requests relief from certain examination coverage requirements imposed by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for the Monticello Nuclear Generating Plant (MNGP). This 10 CFR 50.55a request is for weld examinations, performed during the 2007 refueling outage, where the required coverage of "essentially 100 percent" could not be obtained when examined to the extent practical. The basis for the 10 CFR 50.55a request is that compliance with the specified requirements is impractical due to plant design. The details of the 10 CFR 50.55a request are enclosed.

NMC is submitting this request for the Fourth Ten-Year Inservice Inspection Interval scheduled to end on May 31, 2012. If you have any questions or require additional information, please contact Lynne Gunderson at 715-377-3430.

This letter contains no new commitments and makes no revisions to existing commitments.

For T O'Connor

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
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ENCLOSURE 1
10 CFR 50.55a REQUEST NO. 15
IN ACCORDANCE WITH 10 CFR 50.55a(g)(5)(iii)
INSERVICE INSPECTION IMPRACTICALITY

1. ASME Code Component(s) Affected

Components affected are American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Class 1, Reactor Vessel Nozzle-to-Vessel welds specified below and in-detail in Table A:

Recirculation Inlet	Nozzle N-2B	Weld - N-2B NV
Recirculation Inlet	Nozzle N-2G	Weld - N-2G NV
Feedwater Inlet	Nozzle N-4A	Weld - N-4A NV
Reactor Head Spare	Nozzle N-6A	Weld - N-6A NV
Capped Control Rod Drive (CRD) Return	Nozzle N-9	Weld - N-9 NV

2. Applicable ASME Section XI Code Edition and Addenda

The applicable ASME Section XI Code for the Monticello Nuclear Generating Plant (MNGP), Fourth Ten-Year Inservice Inspection (ISI) Interval is the 1995 Edition with the 1996 Addenda.

3. Applicable Code Requirement

ASME Class 1 Nozzle-to-Vessel welds are subject to the examination requirements of Subsection IWB Table IWB-2500-1, as shown below, and 10 CFR 50.55a(b)(2)(xv)(G). The welds are required to be examined once within the Fourth Ten-Year Interval:

Code Class:	1
References:	IWB-2500, Table IWB-2500-1
Examination Category:	B-D
Item Number:	B3.90
Description:	Nozzle-to-Vessel Welds
Component Numbers:	See Section 1 and Table A
System:	Reactor Vessel
Examination Method:	Volumetric - Ultrasonic Testing (UT)
Examination Volume:	Figure IWB-2500-7(b)

In August 2005, the Nuclear Regulatory Commission (NRC) issued Regulatory Guide (RG) 1.147, Revision 14, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (Reference 1). In RG 1.147, the NRC identifies the ASME Code Cases that they have determined to be acceptable alternatives to applicable parts of Section XI, and that these Code Cases may be used by licensees without requesting authorization from the NRC provided that they are used with any identified limitations or modifications. RG 1.147, Table 1 lists the

following two Code Cases as acceptable to the NRC for use by a licensee with no identified limitations or modifications: 1) Code Case N-460 (Reference 2), and 2) Code Case N-613-1 (Reference 3).

Code Case N-460 states in part, "when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage on any Class 1 or Class 2 weld may be accepted provided the reduction in coverage for that weld is less than 10 percent."

NRC Information Notice (IN) 98-42 (Reference 4) termed a reduction in coverage of less than 10 percent to be "essentially 100 percent." IN 98-42 states in part, "The NRC has adopted and further refined the definition of 'essentially 100 percent' to mean 'greater than 90 percent'...has been applied to all examinations of welds or other areas required by ASME Section XI."

Code Case N-613-1 provides an alternative examination volume that includes the width of the weld plus one-half inch of adjacent base metal on each side of the widest part of the weld. In comparison, the examination volume required by the Figure IWB-2500-7(b) includes the width of the weld plus the adjacent base metal on each side of the widest part of the weld equal to one-half of the vessel shell wall thickness.

4. Impracticality of Compliance

Construction Permit CPPR-31 was obtained for the MNGP in 1967. The MNGP systems and components were designed and fabricated before the examination requirements of ASME Section XI were formalized and published. Therefore, MNGP was not specifically designed to meet the requirements of ASME Section XI and full compliance is not feasible or practical within the limits of the current plant design.

10 CFR 50.55a recognizes the limitations to inservice inspection of components in accordance with Section XI of the ASME Code that are imposed due to early plants' design and construction, as follows:

> 10 CFR 50.55a(g)(1): For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (5) of this section to the extent practical.

> 10 CFR 50.55a(g)(4): Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and pre-service examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code ... to the extent practical within the limitations of design, geometry and materials of construction of the components.
>
> 10 CFR 50.55a(g)(5)(iii): If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.

The inspection limitations on the subject components are due to inherent nozzle design geometric contours (see Table A).

A description of the examination methodology used to provide the maximum obtainable coverage is provided in Section 6 of this request. This methodology is based on ASME Section XI, Appendix VIII qualification and was applied to the extent practical within the design constraints of the components. Enclosure 3 provides cross-sectional diagrams of the subject welds showing the geometric contour of the component design in relation to the welds and the coverage obtained within the examination volume requirements of Code Case N-613-1, Figure 2.

5. Burden Caused by Compliance

Compliance with the examination coverage requirements of ASME Section XI would require modification, redesign, or replacement of components where geometry is inherent to the component design.

6. Proposed Alternative and Basis for Use

Proposed Alternative

In accordance with 10 CFR 50.55a(g)(5)(iii), relief is requested for the components listed in Table A on the basis that the required examination coverage of "essentially 100 percent" is impractical due to physical obstructions and the limitations imposed by design, geometry and materials of construction.

Nuclear Management Company (NMC) performed qualified examinations that achieved the maximum, practical amount of coverage obtainable within the limitations imposed by the design of the components. Additionally, as Class 1 examination Category B-P components, a VT-2 examination is performed on the subject components of the Reactor Coolant Pressure Boundary (RCPB) during system pressure tests each refueling outage. This was completed during the 2007 refueling outage and no evidence of leakage was identified for these components.

Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), NMC requests relief from the requirements of ASME Section XI Table IWB-2500-1, Category B-D, Item B3.90, and proposes to utilize these completed exams as acceptable alternatives that provide reasonable assurance of continued structural integrity.

Basis for Use

The NMC Nondestructive Examination (NDE) procedures incorporate inspection techniques qualified under Appendix VIII of the ASME Section XI Code by the Performance Demonstration Initiative (PDI) for examination of the subject nozzle-to-vessel welds, and allow the examination volume to meet the provisions of alternative requirements (i.e., Code Case N-613-1).

The examinations were performed using a manual contact method from the nozzle outside blend radius and vessel surfaces. Coverage was obtained by following the scan parameters designated within NMC NDE procedures and as defined by MNGP specific Electric Power Research Institute (EPRI) computer modeling reports (References 5 and 6) for each nozzle configuration and angle. It should be noted that that the scans defined by the EPRI report are only applicable to the inner 15 percent of the weld volume when scanning in the parallel direction.

The refracted longitudinal wave mode of propagation was applied for all the radial scans of the exam volume, and to the outer 85 percent of the exam volume for parallel scans. The shear wave mode of propagation was applied for each of the transducer and wedge combinations required for the remaining inner 15 percent of the parallel scan exam volume.

The subject components received the required examination(s) to the extent practical within the limited access of the component design. One hundred (100) percent coverage was obtained for the inner 15 percent of the examination volume. The examination limitations for the subject components were encountered within the outer 85 percent of the examination volume. For the examinations conducted, satisfactory results were achieved, and no evidence of unacceptable flaws was detected with the inspection techniques.

Due to the design of these welds it was not feasible to effectively perform a volumetric examination of "essentially 100 percent" of the required volume. The nozzle-to-vessel welds are accessible from the vessel plate side of the weld and are examined to the extent practical, but there are no qualified examinations to obtain coverage of the excluded areas within the outer 85 percent of the examination volume due to the nozzle forging curvature.

Additional coverage for the limited areas was not achievable or practical, based on the latest qualified ultrasonic technology, nor by other considered examinations methods, such as radiography. NMC has concluded that if significant degradation existed in the subject welds, it would have been identified by the examinations performed.

Additionally, as Class 1 examination category B-P components, VT-2 examinations were performed on the subject components in association with the Reactor Coolant Pressure Boundary system pressure test performed during the 2007 refueling outage. No evidence of leakage was identified during this system test.

The materials for the subject components are A508 Cl II nozzle forgings welded to A533 Cl I vessel shell plate. A review of operating experience within the nuclear industry did not reveal any instances of cracking in this location and type of weldment.

The MNGP reactor vessel water chemistry is controlled in accordance with the 2004 revision to the BWR Water Chemistry Guidelines (Reference 7). Also a hydrogen water chemistry system is used to reduce the oxidizing environment in the reactor coolant. These additional measures provide added assurance against the initiation of cracking or corrosion from the inside surface of the reactor vessel. An inerted primary containment environment during operation provides assurance of corrosion protection on the outside surface of the reactor vessel.

The provisions described above as an alternative to the code requirement will continue to provide reasonable assurance of the structural integrity of the subject welds. The examinations were completed to the extent practical and evidenced no unacceptable flaws present. VT-2 examinations performed on the subject components during system pressure testing each refueling outage (in accordance with examination Category B-P) provide continued assurance that the structural integrity of the subject components is maintained. Additionally, the MNGP Water Chemistry Program and inerted primary containment environment provide added measures of protection for the component materials. Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), NMC requests relief from the ASME Section XI examination requirements for the subject nozzle-to-vessel welds.

7. Duration of Proposed Alternative

NMC requests the granting of this relief for the Fourth Ten-Year Inservice Inspection Interval of the Inservice Inspection Program for the MNGP that is scheduled to end on May 31, 2012.

8. Precedents

The NRC has granted relief for other nozzle-to-vessel shell welds at the MNGP, most recently for the current Fourth Ten-Year Inservice Inspection Interval (Reference 8). Also, the NRC has granted relief for the Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 9), the Dresden Nuclear Power Station, Units 2 and 3 (Reference 10), and the Prairie Island Nuclear Generating Plant, Unit 2 (Reference 11).

REFERENCES

1. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 14, August 2005.

2. ASME Section XI Code Case N-460, "Alternative Examination Coverage for Class 1 and Class 2 Welds."

3. ASME Section XI Code Case N-613-1, "Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Item No's. B3.10 and B3.90, Reactor Nozzle-To-Vessel Welds, Figures IWB-2500-7(a), (b), and (c)."

4. NRC Information Notice 98-42, "Implementation of 10 CFR 50.55a(g) In-service Inspection Requirements."

5. EPRI Internal Report IR-2004-63, "Monticello Nozzle Inner Radius and Nozzle-to-Shell Weld Examinations," dated December 2004.

6. EPRI Internal Report IR-2006-100, "Monticello Nozzle Inner Corner Regions and Nozzle-to-Shell Weld Examinations," dated January 2006.

7. BWRVIP-130, "BWR Water Chemistry Guidelines - 2004 Revision" (EPRI Topical Report TR-1008192).

8. NRC letter to NMC, "Monticello Nuclear Generating Plant (MNGP) - Fourth 10-Year Interval Inservice Inspection (ISI) Program Plan Relief Request No. 13 (TAC No. MC8882)," dated July 18, 2006.

9. Letter from NRC to Exelon Generation Company, LLC, "Quad Cities, Units 1 and 2 - Relief Request CR-39 for Third 10-Year Inservice Inspection Interval (TAC Nos. MC2427 and MC2428)," dated May 10, 2005.

10. Letter from NRC to Exelon Generation Company, LLC, "Dresden Nuclear Power Station, Units 2 and 3 - Relief Request CR-26 For Third 10-Year Inservice Inspection Interval (TAC Nos. MC3269 and MC3270)," dated October 1, 2004.

11. NRC letter to NMC, "Prairie Island Nuclear Generating Plant, Unit 2 – Evaluation of Relief Request No. 16 for the Unit 2 3rd 10-year Interval Inservice Inspection Program (TAC No. MC1775)," dated October 18, 2004.

ENCLOSURE 2
10 CFR 50.55a REQUEST NO. 15
IN ACCORDANCE WITH 10 CFR 50.55a(g)(5)(iii)
INSERVICE INSPECTION IMPRACTICALITY

TABLE A - Category B-D, "Full Penetration Welds of Nozzles in Vessels," Item No. B3.90
Percent Coverage and Limitations for Nozzles N-2B, N-2G, N-4A, N-6A, and N-9

Code Category and Item No.	System and Component Description	Component ID	Code Component and Examination Volume Required	Percent* Coverage Obtained	Limitations	Exam Report Number
B-D B3.90	Reactor Vessel, Recirculation Inlet Nozzle N-2B	N-2B NV	Nozzle-to-Vessel Weld, Code Case N-613-1 Figure 2	78%	Limited due to nozzle configuration.	2007UT058
B-D B3.90	Reactor Vessel, Recirculation Inlet Nozzle N-2G	N-2G NV	Nozzle-to-Vessel Weld, Code Case N-613-1 Figure 2	78%	Limited due to nozzle configuration.	2007UT061
B-D B3.90	Reactor Vessel, Feedwater Inlet Nozzle N-4A	N-4A NV	Nozzle-to-Vessel Weld, Code Case N-613-1 Figure 2	79%	Limited due to nozzle configuration.	2007UT103
B-D B3.90	Reactor Vessel, Top Head Spare Nozzle N-6A	N-6A NV	Nozzle-to-Vessel Weld, Code Case N-613-1 Figure 2	86%	Limited due to nozzle configuration.	2007UT104
B-D B3.90	Reactor Vessel, CRD Return Nozzle (capped) N-9	N-9 NV	Nozzle-to-Vessel Weld, Code Case N-613-1 Figure 2	85%	Limited due to nozzle configuration.	2007UT102

* Due to the nozzle design it was not feasible to effectively examine essentially 100 percent of the required examination volume as defined in Figure 2 of Code Case N-613-1. Percentages are conservatively rounded down to the nearest whole number. It should be noted that 100 percent of the inner 15 percent was examined for all components listed above.

ENCLOSURE 3

10 CFR 50.55a REQUEST NO. 15
IN ACCORDANCE WITH 10 CFR 50.55a(g)(5)(iii)
INSERVICE INSPECTION IMPRACTICALITY

EXAM LIMITATIONS IMPOSED BY COMPONENT DESIGN AND CONSTRUCTION

This enclosure contains a series of excerpts from the ISI Ultrasonic Testing (UT) reports applicable to the subject components.

These excerpts contain sketches depicting the component configuration with physical limitations imposed by the design, e.g., geometrical contour, weld position, interferences, and a cross sectional view depicting the UT coverage and limitations in relation to the required examination volume.

Also included is a sketch of a typical reactor vessel nozzle contour and the resulting effect that causes the UT transducer to lift and lose effective coupling when it reaches the nozzle blend radius.

COMPONENT	REPORT	PAGE(S)
N-2B NV	2007UT058	Pages 1-2
N-2G NV	2007UT061	Pages 3-4
N-4A NV	2007UT103	Page 5
N-6A NV	2007UT104	Page 6
N-9 NV	2007UT102	Page 7
Typical Reactor Vessel Nozzle Contour Affecting Transducer Contact at blend radius		Page 8

8 Pages Follow

Coverage drawings excerpted from applicable reports

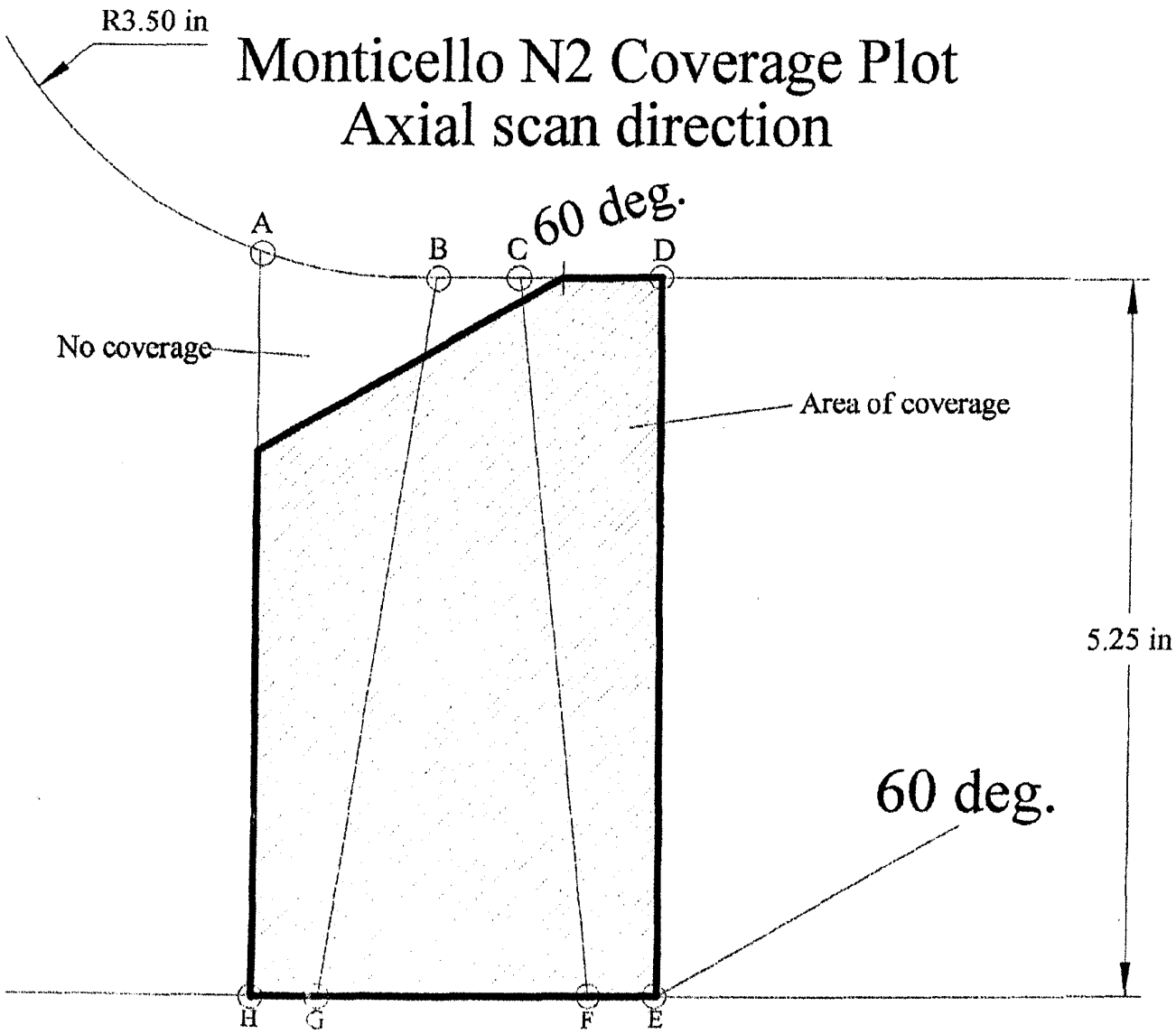
Component N-2B NV Report # 2007UT058

NMC

Supplemental Report

Report No.: 2007UT058

Summary No.: 102658

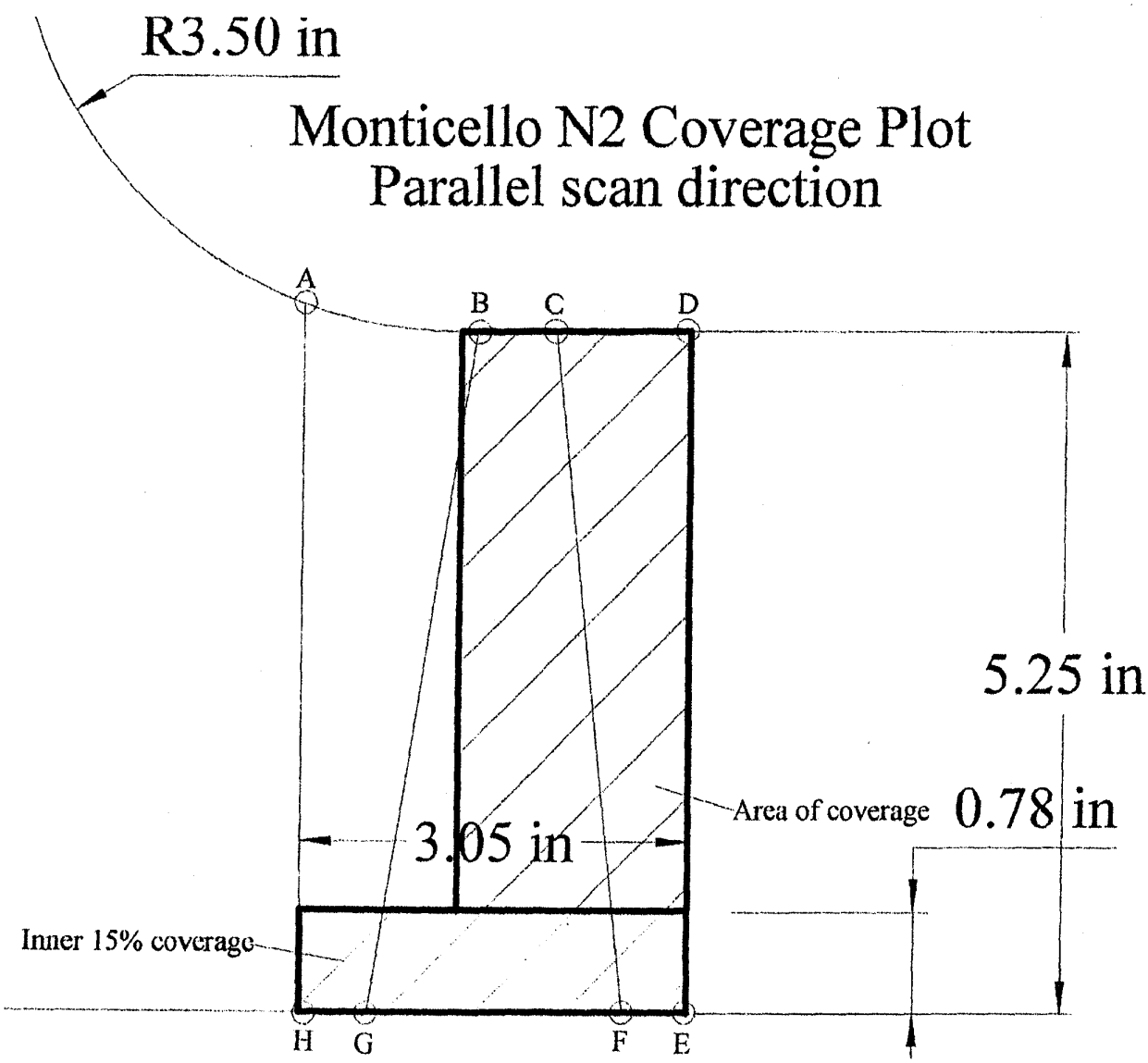


NMC

Supplemental Report

Report No.: 2007UT058

Summary No.: 102658

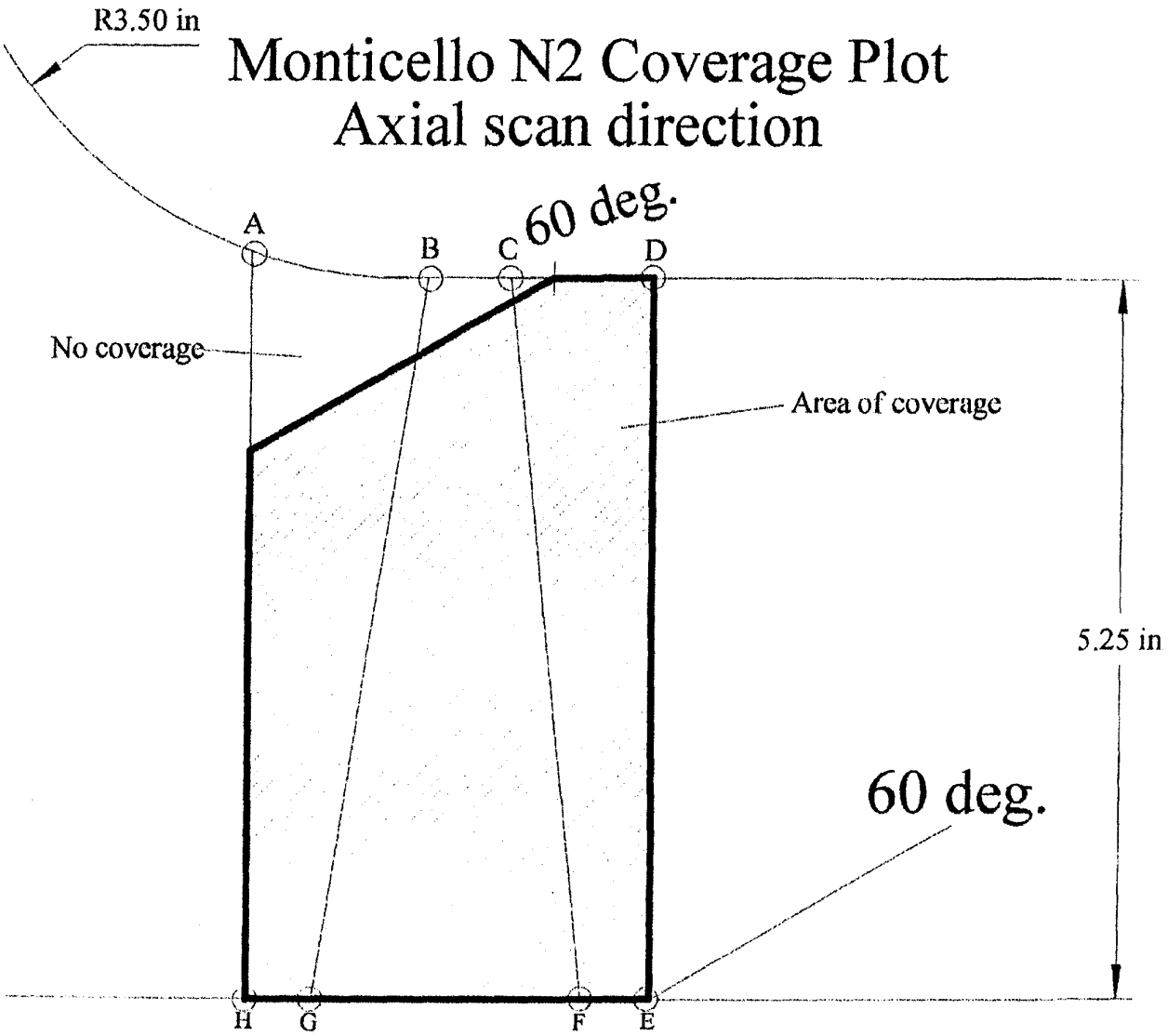


NMC

Supplemental Report

Report No.: **2007UT061**

Summary No.: **102668**

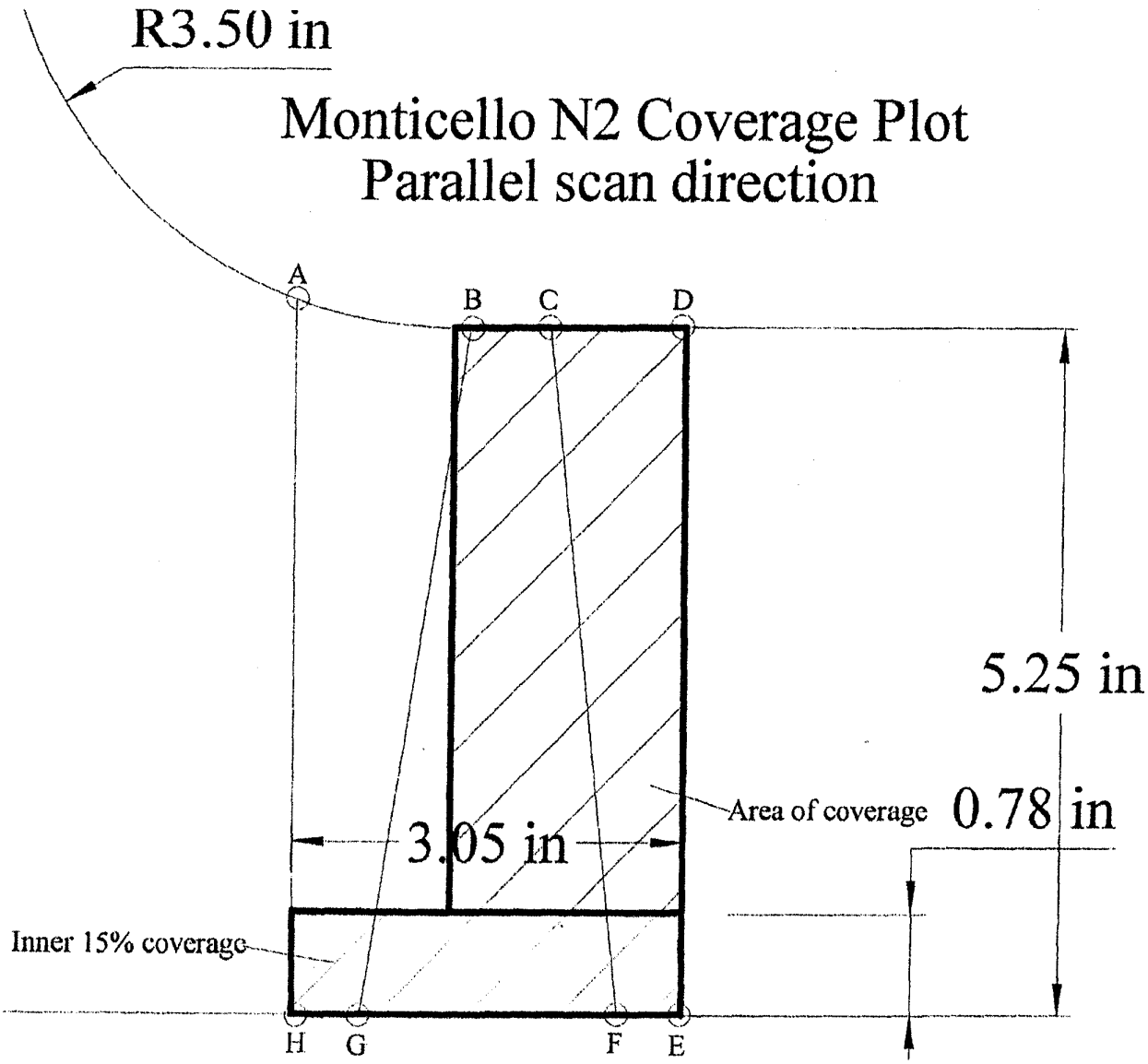


NMC

Supplemental Report

Report No.: **2007UT061**

Summary No.: **102668**



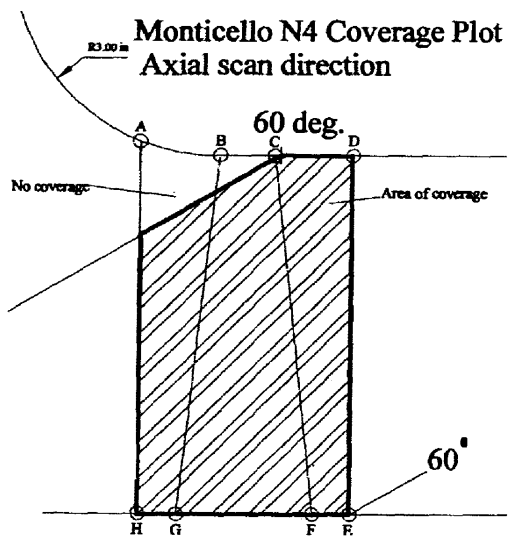
NMC

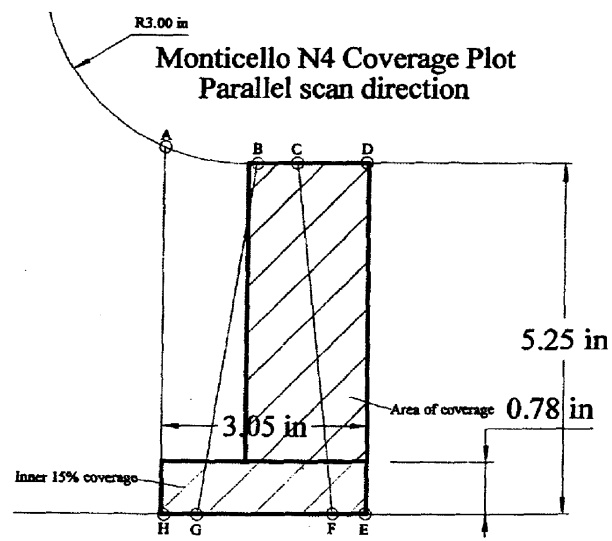
Supplemental Report

Report No.: 2007UT103

Summary No.: 102684

Comments: Coverage Plots





Component N-6A NV Report # 2007UT104

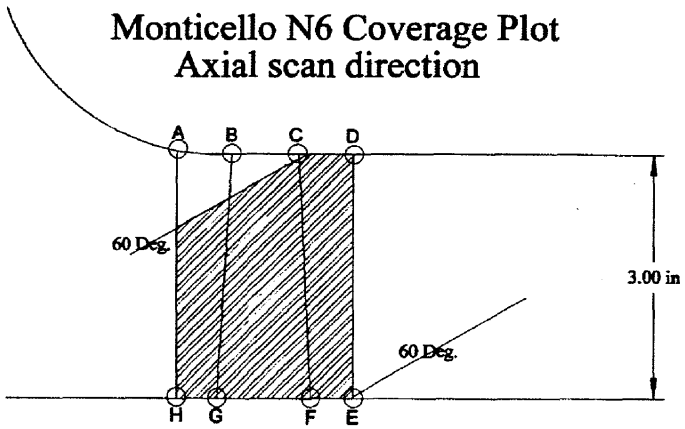
NMC

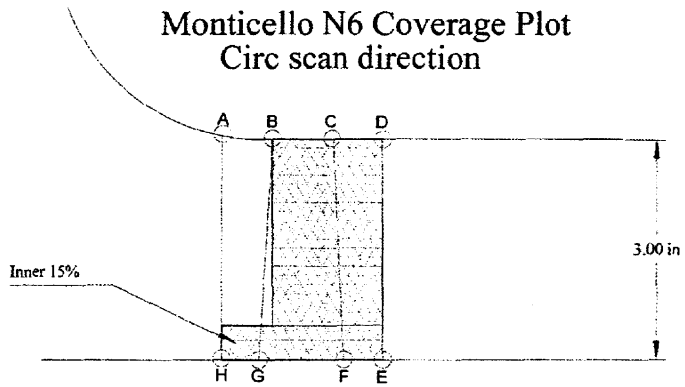
Supplemental Report

Report No.: 2007UT104

Summary No.: 102375

Comments: N-6A NV Coverage Plots





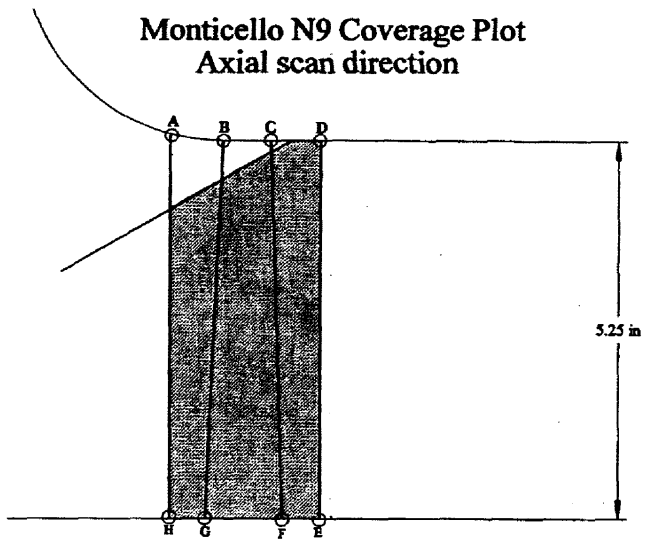
NMC

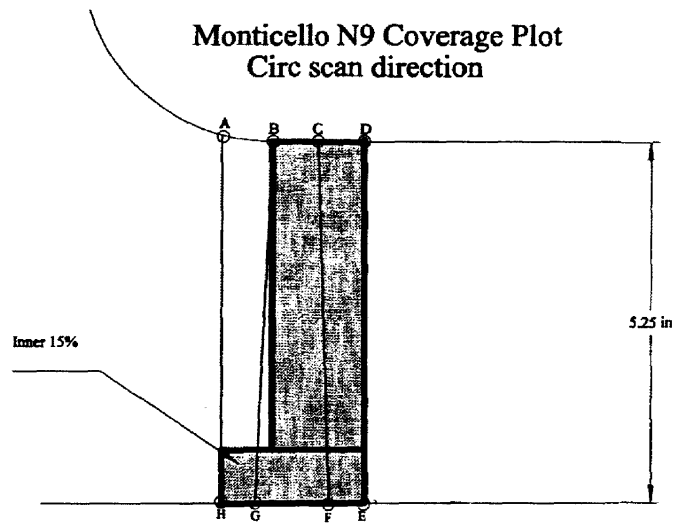
Supplemental Report

Report No.: 2007UT102

Summary No.: 102700

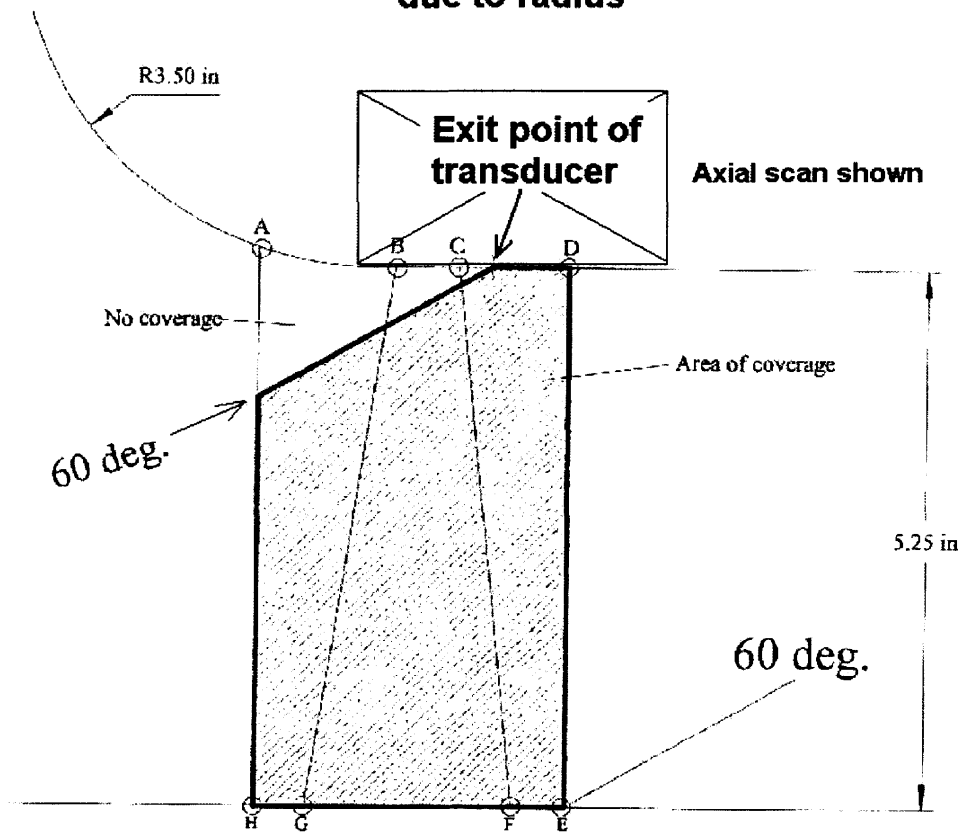
Comments: **Coverage Plots**





Typical Representation of Nozzle Limitations

Coverage affected by liftoff due to radius



N2 Nozzle shown as example