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10 CFR 50.73(e)
10 CFR 50.59(d)(2)

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

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

Submittal of Revision 34 to the Updated Safety Analysis Report

Pursuant to 10 CFR 50.71(e), Revision 34 to the Monticello Nuclear Generating Plant (MNGP) Updated Safety Analysis Report (USAR) is provided. This revision completes an update of the information in the USAR for the period from November 25, 2015, to November 15, 2016.

The changes in this revision reflect the incorporation of modifications, license amendments, and editorial corrections and clarifications. These changes are made in accordance with the guidance provided in Nuclear Energy Institute (NEI) Report NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports", Revision 1, dated June 1999 and Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in accordance with 10 CFR 50.71(e)", dated September 1999.

Enclosure 1, "Report of Changes, Tests and Experiments", indicates that one 10 CFR 50.59 evaluation was performed. This report is provided as required by 10 CFR 50.59(d)(2).

Enclosure 2, "Report of Changes to Licensee Docketed Commitments", indicates that in accordance with the guidance provided in NEI 99-04, "Guidelines for Managing NRC Commitment Changes", dated July 1999, for this period, there was one change to a commitment.

Enclosure 3, "Summary of Information Removed from the USAR", provides the information removed from the USAR for this revision cycle. This information is provided in accordance with Revision 1 of NEI 98-03 and Regulatory Guide 1.181.

Enclosure 4 contains Revision 34 of the MNGP USAR. The USAR is being submitted electronically, in its entirety, on CD-ROM according to the instructions in Regulatory Issue Summary (RIS) RIS 2001-005, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM".

A006
A053
NRR

Enclosure 5, "Report of Changes to the Monticello Fire Protection Program", provides a summary of changes to the program. Changes to the Fire Protection Program are provided in accordance with the guidance contained in Generic Letter 86-10, "Implementation of Fire Protection Requirements", dated April 24, 1986.

Summary of Commitments

This letter contains no new commitments.

In accordance with 10 CFR 50.71(e)(2), I certify that the information presented herein, accurately presents changes made since the previous submittal prepared pursuant to Commission requirement and identifies changes made under the provisions of 10 CFR 50.59 not previously submitted to the Commission.

Peter A. Gardner
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (5)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

The following includes a brief description and summary of the 10 CFR 50.59 evaluations performed between November 25, 2015 and November 15, 2016 for those changes, tests and experiments that were carried out without prior U.S. Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR 50.59(d)(2).

<u>Evaluation No. SCR-15-0378 Revision 0:</u>
<u>Replace SBGT (Standby Gas Treatment) Controllers to Improve Reliability</u>

<u>Activity Description</u>:
Replace both the original Bailey and Scientech/NUSI 2000 controllers for SBGT. The replacement controllers are digital. In order to address the potential for common cause failure, hardware, and software diversity will be provided: FIC-2943 will be replaced with a Siemens Model 353 and FIC-2942 will be replaced with a Yokogawa YS1700.

<u>Summary of Evaluation</u>:
The 10 CFR 50.59 evaluation's response to each of the eight 10 CFR 50.59(c)(2) criterion is "No." Therefore, the evaluation concluded that prior NRC approval is not required. The evaluation shows that the implementation of the digital aspects will not, increase the frequency of accidents, increase consequences of accidents, create new accidents, increase the occurrence of equipment malfunctions, increase consequences of equipment malfunctions, create malfunctions having different results, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in departure of a method of evaluation. The bases for this statement are as follows:

1. SBGT system is not an initiator of any accidents and no failure modes of the controllers or the SBGT system have the potential to create an accident.

2. The controller replacements are direct fit and function replacements for the existing controllers. There are no reductions to the controller response time. The ability of Operators to operate the equipment or the likelihood of misoperation is not changed.

   The replacement controllers were tested and qualified to the same or higher standards than the existing controllers. Testing included testing of the hardware specifications and software specifications; as well as seismic, and electromagnetic compatibility testing.

Although the introduction of digital programmable controllers into the SBGT system increases the internal complexity of the controller, it does not result in a more than minimal increase the likelihood of a malfunction. These controllers have been used at other facilities in the US Nuclear industry and are widely used in the nuclear industry in other countries. A FEMA analysis was performed.

The modification does not change, degrade, or prevent the SBGT system from accomplishing any of the accident mitigation functions attributed to the system in the USAR.

3. Since the initial conditions and SBGT system response remains unchanged, there is no potential to change any of the assumptions made when determining the radiological consequences of any accidents and no impact on the radiological consequences of the LOCA analysis described in the USAR.

4. The replacement controllers perform the same function in the same manner as the previous controllers and failures of the replacement controllers have the same consequences as failures of the existing controllers. This activity does not result in an increase in the consequences of a malfunction of an SSC important to safety as the consequences of malfunctions remain the same.

5. The replacement of the SBGT flow controllers do not introduce the possibility of a new accident because the SBGT system is not an initiator of any accident and no failure modes of the controllers or the SBGT system have the potential to create an accident.

6. A comparison of the new and existing failure modes was performed and indicates that, with the introduction of the diversity concept to address software common cause failures, the effects of any new failure mechanisms introduced by the replacement controllers are bounded by existing failure modes and mitigated by the separation of system trains.

   These controllers are utilized to perform surveillances and are not required to perform any functions to support the SBGT system functions during design basis events. Additionally the valves controlled by the controllers are normally open and fail safe (open) on loss of air. As a result, most controller failure modes and also the most common failure modes cause the controller to fail safe (low output, valve open). Thus a malfunction of the controller which could affect the function of the SBGT system is rare and the potential for having two malfunctions that impair the system function on both of the trains at the same time is mitigated by the use of diverse designs in the hardware and software between the trains.

   The replacement of the SBGT flow controllers does not create the possibility for a malfunction of an SSC with a different result because the modification does not introduce a failure mechanism that has different

consequences than any of the failure mechanisms of the existing analog controllers, and the diversity aspect of the design ensures that the failures due to software common cause failures are no more likely than other potential common cause failures.

7. The replacement of the SBGT flow controllers does not have the potential to affect any fission product barrier as neither the controllers or the SBGT system involves any fission product barriers.

8. The modification does not involve or constitute a method of evaluation as defined in the USAR.

Evaluation No. SCR-16-0024 Revision 0:
Disconnect Faulty 46-19 PIP Over-travel Input

Activity Description:
EC 26642 (Disconnect Faulty 46-19 PIP Over-travel Input) is a temporary modification to remove the over-travel alarm indication for control rod 46-19 to allow control rod 46-19 to be declared operable. Currently, the position indicator probe (PIP) associated with the control rod 46-19 has faulted wiring that causes the withdraw over-travel alarm.

With the fault isolated, the withdraw overtravel alarm for control rod 46-19 clears and will not come in for rod 46-19. The proper "00" is displayed at full-in. Removing the withdraw overtravel alarm for rod 46-19 allows the overtravel alarm to be available for all the other control rods.

There are three actual indications of withdraw overtravel, the red full-out backlight, the "48" position display, both of which go dark when the drive piston moves past full-out, and the third indication is the alarm. The alarm indication has to have a decoupled condition to alarm such that it is rarely operated. The backlight and "48" display indications are derived via independent PIP reed switches and independent probe buffer card circuitry. These switches are exercised every time the rod is fully withdrawn which displays switch operation evidenced by the change from amber backlit "46" to red backlit "48". In a case of a full-out rod with a withdraw attempt and no change in the red backlit "48" display, this serves as a rod coupling check.

Summary of Evaluation:
The 10 CFR 50.59 evaluation's response to each of the eight 10 CFR 50.59(c)(2) criterion is "No." Therefore, the evaluation concluded that prior NRC approval is not required. The evaluation shows that the implementation of the temporary modification will not, increase the frequency of accidents, increase consequences of accidents, create new accidents, increase the occurrence of equipment malfunctions, increase consequences of equipment malfunctions, create malfunctions having different results, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in departure of a method of evaluation. The bases for this statement are as follows:

1. Removal of the overtravel alarm function for control rod 46-19 does not introduce the possibility of a change in the frequency of a Control Rod Drop Accident of concern because the ability to verify to control rod coupling is maintained using the red backlight and "48" display indications.

2. Removing the overtravel alarm function from control rod 46-19 does not introduce the possibility of a change in likelihood of a malfunction because the overtravel alarm is not the initiator of any malfunctions that result in control rod decoupling and no new failure modes are introduced.

3. Removal of the overtravel alarm does not change the consequences of a CRDA because the alarm function has no effect on the CRDA bounding conditions.

4. Removal of the overtravel alarm results in minimal risk of not detecting a uncoupled control rod because two independent position indications remain available to detect control rod movement past position 48. In summary, there is no more than minimal increase in consequences of a malfunction.

5. As two other methods exist for verifying that the rod remains coupled via independent reed switches and circuitry, and each is more reliable than an alarm, there is no accident of a different type which could occur due to this activity.

6. The alarm function has no impact on the physical control rod drive to control blade coupling. Therefore the lack of an alarm cannot create the possibility of a malfunction of the control rod drive/control blade with a different result than a decoupled rod from known sources. The RPIS function of the alarm is to provide the operator a method of determining if a control rod is decoupled. As two other redundant methods exist for making such a determination and each is more reliable than an alarm which can't be routinely tested, there is no possibility for a malfunction of the RPIS with a different result.

7. The proposed change will not increase the likelihood of occurrence or the severity or consequences of a postulated rod drop accident because of the diversity of decoupled rod indications and reliability of unaffected methods of rod position indication and does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

8. This alarm function is not a method of evaluation described in the USAR.

Evaluation No. SCR-16-0043 Revision 0:
ECCS-LOCA Analyses Input Changes
Revisions to Calculations 13-055, Revision 1, and 11-180, Revision 1 to procedure 0255-03IA-2A
Changes to USAR Sections USAR-06.02, USAR-06.08, USAR-14.07,

USAR-14.11 and USAR-14.FIG

Activity Description:
The primary focus of the proposed activity is the new revisions to calculations 13-055 (Rev. 1) and 11-180 (Rev. 1). The changes to calculations 13-055 and 11-180 involve determination of new core spray and LPCI reactor vessel injection flow curves for use as replacements for the existing core spray and LPCI injection flow curve inputs in the Monticello ECCS-LOCA analyses. These new curves are slightly less conservative than those used in the ECCS-LOCA analyses but take into consideration new conservative criteria which the existing curves did not. These new criteria include: emergency diesel generator (EDG) Technical Specification frequency deviation of 2%, core spray and RHR pump flow reduction effects due to pump operation at reduced net positive suction head and wear ring degradation per NRC SECY-11-0014, and lower than expected shutoff heads for the core spray and RHR pumps determined from system flow models developed for these systems. The methods used to perform Monticello's ECCS analyses were approved by the NRC (Reference USAR Section 14.7.2). However, changes to ECCS analyses inputs are not considered a change to a method of evaluation. Consequently, addressing the impacts of the proposed activities on methods of evaluation is not applicable to this 50.59 evaluation.

The revisions to USAR-06.02, USAR-06.08, USAR-14.07, and USAR-14.11 involve updating discussion to reflect the flow data points from the new flow curves and changing the referenced source for the new flow curve data points to calculation 13-055.

The editorial changes described above for calculation 11-180 also impact USAR-14.07 and USAR-14.11 resulting in inconsequential changes as defined in Attachment 2 of fleet procedure FP-E-SAR-01, Rev. 8 (USAR Changes). These include changes in reference document locations, clarifications, excess detail and redundancy removal, and correction of inconsistencies all related to the discussion of the ECCS-LOCA analyses. Such inconsequential USAR changes are also not required to be screened or evaluated. Therefore, these USAR changes are not included in the 50.59 evaluation.

Summary of Evaluation:
The 50.59 evaluation's response to each of the eight 10 CFR 50.59(c)(2) criterion is "No". Therefore, the evaluation concluded that NRC approval for the revisions to calculations 13-055 and 11-180 and USAR is not required. The evaluation shows that use of the replacement injection flow curve inputs in lieu of those used in the ECCS-LOCA analyses will not increase the frequency of accidents, increase consequences of accidents, create new accidents, increase the occurrence of equipment malfunctions, increase consequences of equipment malfunctions, create malfunctions having different results, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in departure of a method of evaluation.

The bases for this statement are that the calculation revisions and associated USAR and procedure changes only involve inputs for the ECCS-LOCA analyses.

These analyses are used to evaluate the reactor's response to limiting pipe breaks due to a loss of coolant accident that has already occurred to demonstrate conformance with the ECCS acceptance criteria of 1 OCFR50.46. The changes have no impact on accident frequency, are not an initiator of any malfunction, and create no new failure modes. The ECCS-LOCA analyses acceptance criteria are still satisfied and no design basis limit fission product barrier is exceeded or altered as a result of the analyses flow curve input changes. The use of the replacement flow curves as inputs in lieu of the previous inputs is also not considered a departure from the method of evaluation used for the ECCS-LOCA analyses so criterion 8 is not applicable.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

Commitments are identified and reported to the Commission in accordance with guidance provided in NEI Technical Report 99-04 Revision 0, "Guidelines for Managing NRC Commitment Changes."

This enclosure provides a brief description and a summary of changes to commitments established with the NRC by the Monticello Nuclear Generating Plant (MNGP) per NEI 99-04 guidelines.

For Revision 34:

The following change was made to an existing Monticello Nuclear Generating Plant commitment and required the station to provide notification of the change to the NRC. Notification of the commitment change was made via Reference 1 below.

References: 1) Letter from Peter A. Gardner (NSPM), to Document Control Desk (NRC), "Revised Commitment to Reconcile Analysis of Bypass Voiding For Transition to AREVA Analysis Methodology (TAC No. MF5002)," L-MT-16-017, dated April 29, 2016 (ADAMS Accession No. ML16120A299)

2) Letter from Timothy J. O'Connor (NSPM), to Document Control Desk (NRC), "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Review Branch Request for Additional Information (RAI) dated January 16, 2009 (TAC No. MD9990)," L-MT-09-017, dated March 19, 2009 (ADAMS Accession No. ML090790388)

The commitment initially made in Reference 2 was revised so that it could be reconciled with the AREVA calculational methodology that was approved by MNGP License Amendment 188. The commitment change is justified because it perpetuates the bypass void fraction analysis, but invokes the approved AREVA methodology in place of the GeneralElectric – Hitachi (GEH) methodology.

Revised Regulatory Commitment	Due Date / Event
The steady state bypass void fraction for the EPU core will be calculated using the method described by NSPM letter to NRC L-MT-16-017, dated April 29, 2016.	Effective for reactor core reload analyses that will be implemented at startup for operating cycle 29 in 2017.

For additional information regarding this revised commitment please refer to Reference 1.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT
SUMMARY OF INFORMATION REMOVED FROM THE USAR

Consistent with the guidance in Nuclear Energy Institute (NEI) Report NEI 98-03, "Guidelines for Updated Final Safety Analysis Reports," Revision 1 and Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e)," Information removed from the Monticello Nuclear Generating Plant Updated Safety Analysis Report (USAR) is summarized below.

- USAR Change 01480460
  Affected section:
  Section USAR-09.02 – Plant Radioactive Waste Control Systems
  9.2.2.2.3, Detergent Wastes

  Deletion was due to the implementation of EC 21091, Laundry Drain Piping Rerouting. The reference to how detergent wastes are processed was deleted, as they are now processed with other chemical wastes as discussed in section 9.2.2.2.2.

- USAR Change 01519920
  Affected section:
  Section USAR-10.02 – Plant Auxiliary Systems-Reactor Auxiliary Systems
  Table 10.2-3, Reactor Auxiliary Systems Reactor Core Isolation Cooling System (RCIC) Principal Components Parameters

  Deletion of *"and/or close..."* removed from RCIC Pump Discharge Valves MO-2106 and MO-2107 requirement.

- USAR Change 01506230
  Affected section:
  Section USAR-J.04, Appendix USAR-J.04 Safe Shutdown Analysis
  Table J.4.6-1, Monticello Appendix R Compliance Evaluation Fire Area I Division 1

  Deletion of cables for certain fire zones for components:
  MO-1749, MO-1751, MO-1753, MO-1754, P-111A, CV-1994, CV-1996, MO-2006, MO-2008, MO-2010, MO-2012, MO-2014, MO-2020, MO-2022, P-202A, P-109A, P-109C, LT-2-3-85A, PT-6-53A, D10, D11, D111, SV-2-71D, V-SF-10.

  Cables were incorrectly identified in these fire zones.

ENCLOSURE 5

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO THE MONTICELLO FIRE PROTECTION PROGRAM

This enclosure contains a report of changes to the Monticello Fire Protection Program (FPP) in accordance with the provisions of 10 CFR 50.71(e), 10 CFR 50.59, and Generic Letter (GL) 86-10.

In conformance with GL 86-10, the Updated Fire Hazards Analysis (UFHA) and the Safe Shutdown Analysis (SSDA) are incorporated directly into the Updated Safety Analysis Report (USAR) as Appendix J.05 and J.04 respectively. The following summarizes fire protection program document changes since the previous submittal.

1. USAR J.04, SAFE SHUTDOWN ANALYSIS

Revision 33*

- USAR 01502449 – The following clarifying note was added: "Note: Due to this fire area approach, fire area boundaries required for Appendix R compliance do not necessarily separate Safe Shutdown Divisions."

Revision 34

- USAR 01506230
    - Table J.4.5-1 – Editorial change the Fire Zone for AN2-TRAIN-A to 19A.
    - Table J.4.5-2 – Editorial changes were made to cable IDs for SV-2-71D to C32-D11/1 and C03-C32/4.
    - Table J.4.5-1, Table J.4.5-2, Table J.4.5-4, Table J.4.6-1 – revised to include SV-4234 and SV-4235 and the components associated information (system, fire zones, cable IDs, cable functions, suprious operations, etc.). These components were added as part of the credited safe shutdown path for Safety Relief Valves (SRVs) for Fire Area XII.
    - Table J.4.6-1 – Removed cables for certain fire zones for components MO-1749, MO-1751, MO-1753, MO-1754, P-111A, CV-1994, CV-1996, MO-2006, MO-2008, MO-2010, MO-2012, MO-2014, MO-2020, MO-2022, P-202A, P-109A, P-109C, LT-2-3-85A, PT-6-53A, D10, D11, D111, SV-2-71D, V-SF-10. This was due to the cables being incorrectly identified in these fire zones.
    - Table J.4.6-1 – Revised compliance strategy for SV-2-71D.

- Table J.4.6-1 – Editorial change to 152-511 equipment name from 2AR to 1AR.

2. USAR J.05, FIRE HAZARDS ANALYSIS

   Revision 34

   - USAR 01488669
     - Page J.05-13 – Revised to include Engineering Change (EC) 25741 which replaced the fire pump battery charger #1.

3. 4 AWI-08.01.00, FIRE PROTECTION PROGRAM PLAN

   - Revision 18 (PCR 01505381) – Added requirements for maintinaing the Independent Spent Fuel Storage Installation (ISFSI) Fire Hazards Analysis (FHA) to the Non-10CFR50.48 Program Elements section.
   - Revision 19 (PCR 01521810)
     - Revised reference to procedure FP-E-MR-09 (Maintenance Rule (A)(4) Fire Risk Assessment Mitigation Actions (RMAs)) to FP-OP-RSK-01 (Risk Monitoring and Risk Management). FP-OP-RSK-01 supercedes FP-E-MR-09.
     - Revised reference from 4 AWI-04.08.01 (Event Notifications) to FP-OP-REP-01 (Event Reporting and Notification Process). FP-OP-REP-01 supersedes 4 AWI-04.08.01.
     - Removed Appendix R Program Owner requirement to determine the availability of alternate safe shutdown paths for Fire Assessment for Maintenance Rule (a)(4) in accordance with FP-E-MR-09. FP-OP-RSK-01 which supercedes FP-E-MR-09 no longer has this as an action to be taken.
     - Added the use of the 3832 (ISFSI Fire Protection Change Review) form when reviewing changes made the ISFSI Fire Protection Requirements.

4. B.08.05-05, FIRE PROTECTION - SYSTEM OPERATION, Tables A.2-1, A.2-2, A.2-3 and A.2-4

   - Revision 67 (PCR 01493654)
     - Table A.2-2, Specification B.1.f, diesel-driven fire pump engine inspection frequency was changed from 18 months to 2 years. Also added industry recommendations and operating experience as references for developing the inspection criteria.
     - Table A.2-4, editorial change was made to Power Supply list. C-397 was changed to C-379.

- Table A.2-4, Specification D.1, Changes were made to the Diesel Fire Pump Startup steps based on EC Modification 25741 which replaced the Diesel Fire Pump Battery Charger.
- Table A.2-4, Specification D.1, Bases section was revised to provide guidance regarding the Diesel Fire Pump Lube Oil pressures and what actions should be taken at specific levels.
- Table A.2-4, Specification G.6, Operators Actions, Steps were revised for resetting the Exciter Sprinkler System. This change splits steps into individual steps and adds clarification for operators performing the actions.

▪ Revison 68 (PCR 01525417)

- Table A.2-1, Specification E, added an alternate name Condenser Bay Sprinkler System for Lube Oil Piping System Sprinkler.
- Table A.2-1, Specification G.2 was reworded to clarify how to establish fire watch patrols.
- Table A.2-1 Specification I was added to include requirements for Appendix R Emergency Lighting.
- Table A.2-1, Bases section for fire detection was revised to incorporate the Alternate Compensatory measures described in Generic Letter 86-10 evalution FPEE-16-002. This requires all detectors in five (5) fire zones to be functional.
- Table A.2-2, Specification I was added to include requirements for Appendix R Emergency Lighting.
- Table A.2-1, Bases section for system flow testing was revised.
- Table A.2-3, Title was changed to include Appendix R Emergency Lighting Units.
- Table A.2-3, Fire Detector requirements were revised for fire zones 13C, 14A, 16, 19B, and 20 based on the Alternate Compensatory measures described in Gernic Letter 86-10 evalution FPEE-16-002.
- Table A.2-3, changed to include a list of Appendix R Emergency Lighting Units.

▪ Revision 69 (PCR 01539322) – Table A.2-1, Specification G.2, removed the use of electrical supervision on non-functional closers on fire doors instead of establishing a fire watch.

*Denotes changes that were incorporated during the Revision 33 of the USAR as submitted in letter L-MT-16-004 dated January 26, 2016; however, these changes are included in this submittal for completeness because the effective dates of the changes occurred between the period of November 25, 2015, and November 15, 2016.

ENCLOSURE 4 – DISC 1

MONTICELLO NUCLEAR GENERATING PLANT
USAR REVISION 34

ENCLOSED CD-ROM

File Name Rev. 34	File Size (KB)
001 – USAR LOEP	313
002 – USAR TOC	98
003 – USAR Section 1	1,041
004 – USAR Section 2	2,180
005 – USAR Section 3	2,032
006 – USAR Section 4	6,781
007 – USAR Section 5	3,550
008 – USAR Section 6	1,039
009 – USAR Section 7	2,689
010 – USAR Section 8	1,546
011 – USAR Section 9	390
012 – USAR Section 10	830
013 – USAR Section 11	584
014 – USAR Section 12	819
015 – USAR Section 13	360
016 – USAR Section 14	4,794
017 – USAR Section 15, Part 1 of 4	8,076
017 – USAR Section 15, Part 2 of 4	4.358
017 – USAR Section 15, Part 3 of 4	9,172
017 – USAR Section 15, Part 4 of 4	7,105
018 – USAR Appendix A	7,379
019 – USAR Appendix C	63
020 – USAR Appendix D	235
021 – USAR Appendix E	145
022 – USAR Appendix F	373
023 – USAR Appendix G	3,844
024 – USAR Appendix H	10,310
025 – USAR Appendix I	1,748
026 – USAR Appendix J	4,494
027 – USAR Appendix K	517