Attachment F: Other Consultation Letters

DUKE ENERGY®

410 S. Wilmington St
NCRH 15
Raleigh, NC 27601
o: 919.546.5285

Nov. 4, 2020

Fran Marshall
Environmental Affairs Administrator
SCDHEC
2600 Bull Street
Columbia, SC 29201

Subject: Duke Energy – Oconee Nuclear Station Units 1, 2 & 3 Subsequent License Renewal - Thermophilic Organisms

Dear Ms. Marshall:

As per your recommendation regarding DHEC's assessment of thermophilic organisms for Oconee Nuclear Station, Duke Energy had Charles W. Calmbacher, Certified Industrial Hygienist, review the information and offer his professional opinion on this matter. This summary, and Mr. Calmbacher's findings, are provided in the attached letter. We request that DHEC consider this information in support of your review and response to our request concerning any potential public health concerns associated with thermophilic organisms. The original request dated Nov 11, 2019 for Oconee Nuclear Station is also attached.

If you have any questions, please feel free to contact me at 919-793-4220 or arun.kapur@duke-energy.com.

Sincerely,

Arun Kapur

Arun Kapur
Lead Environmental Specialist

<u>Attachments</u>
Charles W. Calmbacher Letter
Thermophilic Letter to DHEC

cc: John Estridge, ONS

ENERCON
Excellence—Every project. Every day.

October 21, 2020

Mr. Arun Kapur
Duke Energy
410 S. Wilmington St, NCRH 15
Raleigh, NC 27601

RE: Impact of Thermophilic Organisms – Oconee Nuclear Station

Dear Mr. Kapur:
An updated review of the potential impact of Naegleria fowleri on Lake Keowee, and associate waterways, was performed relative to the renewal application for the Duke Energy Oconee Nuclear Power Station. A review performed during the previous licensing cycle for the plant by the State of South Carolina (Dr. Brown) concluded that the potential for increased exposure to N. fowleri from ONS's thermal discharge is not a concern. The conditions leading to that conclusion have not changed, and are not proposed to change, since that determination. The cooling water is discharged at temperatures below levels that promote growth of the organism. Cooling towers are not used to cool discharge water. This reduces the potential for growth of the ameba before discharge. The deep discharge of cooling water in the lake keeps water temperature below optimal levels for propagation of the ameba.

In performing this updated review, I utilized the following information and documents:

- Centers for Disease Control and Prevention (CDC), Naegleria fowleri, Primary Amebic Meningoencephalitis (PAM), updated September 29, 2020.
- Hains, Dr. John, Practical Limnology, "What Lurks in that Water?", The Sentinel, September-October 2016. Pp. 10-11.
- Application for Renewed Operating Licenses Oconee Nuclear Station, Units 1, 2, and 3 Volume IV, June 1998. pp. 4-37-38 & Attachment I.

Based on current permit requirements, NPDES permits, discharge location, and discharge parameters, it is reasonable to assume that the potential for increasing the exposure to N. fowleri to the general public is negligible. I see no impediment to the license renewal, or continued operation, of the Oconee Nuclear Power Station based on the potential of impacting public health and safety relative to increased potential for the growth of thermophilic microorganisms. Infection by N. fowleri is considered "rare" by the CDC. There is little reason to believe that the actions of the ONS should promote or increase the potential for elevated exposure or risk to the public at large.

Respectfully submitted,

C. B. Calmbacher

Charles W. Calmbacher, Ph.D., CIH
Certified Industrial Hygienist

cc: Ed Asbury
Tom Slavonic

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield @duke-energy.com

BY U.S. MAIL - RETURN RECEIPT REQUESTED

November 11, 2019

Fran Marshall
Environmental Affairs Administration
SCDHEC
2600 Bull Street
Columbia, SC 29201

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Ms. Marshall:

Duke Energy is seeking a response from DHEC concerning the potential existence and Perceived public health risks associated with thermophilic organisms that may be present in the portion of Lake Keowee that receives the cooling water discharge from our Oconee Nuclear Station (ONS). Information concerning the reason for this request and specific microorganisms of concern is presented below. Figures depicting the station site and the vicinity within a 6-mile radius of the station are attached.

Reason for this Request and Microorganisms of Concern
Duke Energy Carolinas, LLC (Duke Energy) is preparing an application for renewing the operating licenses for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy is contacting you for assistance in assessing the impacts from continued operation during this renewed license period.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

As part of the renewal process, the U.S. Nuclear Regulatory Commission (NRC) requires that the license renewal application include an environmental report (ER) that assesses the impacts from continued operation and any refurbishment undertaken to enable the continued operation of the units. One area of potential environmental impact concerns potential public health risks associated with microorganisms.

Information to Support Consultation on Thermophilic Microorganisms

In Regulatory Guide 1437, Supplement 1, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), the NRC considered health impacts from thermophilic organisms posed to both the public and plant workers because ideal conditions for thermophilic bacteria can result from nuclear facility operations and discharges. The NRC designated public health impacts resulting from thermophilic organisms as a Category 2 issue requiring plant-specific analysis. Information to be considered in evaluating impacts includes thermal discharge temperature; thermal characteristics of the receiving water bodies; thermal conditions for the enhancement of Naegleria fowleri and other pathogens; and potential impacts to public health.

The GEIS discussion of microbiological hazards focuses on the thermophilic microorganisms Legionella spp. (which can be a hazard in cooling towers) and the pathogenic amoeba, N. fowleri (which can be a hazard resulting from cooling water discharges). ONS's cooling system does not use cooling towers but does have a thermal discharge to publicly accessible water.

Naegleria spp. is ubiquitous in nature and thrives in heated water bodies at temperatures ranging from 95-106°F or higher is rarely found in water cooler than 95°F, and infection rarely occurs in water temperatures of 95°F or less (NRC 2013, Section 3.9.3). SCDHEC, South Carolina's state public health agency, characterized the risk of infection from N. fowleri 's as very rare, but warns that "recreational water users should assume that N. fowleri is present in warm freshwater across the United States and be aware that there is always a low-level risk of infection." There have been only eight cases of Primary Amebic Meningoencephalitis, the infection caused by N. fowleri, in South Carolina from 1962 to 2018.

ONS utilizes an open-cycle cooling system in which cooling water is withdrawn from Lake Keowee from its intake channel on the south side of the ONS plant, heated in the condensers, and returned to Lake Keowee through the discharge point on the northeast side of the ONS plant. ONS discharges heated cooling water at a depth of approximately 20 feet. The lake waters near the discharge area are open to the public. Activities in the area include recreational boating, fishing, and scuba diving. Lake Keowee has residential housing and public swimming areas as well.

The current NPDES permit for ONS establishes both a maximum allowable discharge temperature, and a limit for increases of water temperature between the intake and discharge. The maximum discharge temperature is100°F as a daily average, unless critical hydrological, meteorological, and electrical demand conditions apply. In such situations, the discharge temperature shall not be allowed to exceed 103°F. The maximum temperature rise above the intake temperature is limited to 22°F when the intake temperature is greater than 68°F. In the 2013 permit renewal application, Duke Energy requested the daily maximum value of 100°F to a 7-day average not to exceed 100°F.

As part of a CWA Section 316(a) demonstration monitoring, Duke Energy monitors water temperatures at several Lake Keowee stations. The most recent report submitted to SCDHEC is from 2013 and covers the years 2006 – 2011. The closest station to the plant's discharge is Location 508, which is approximately 200 meters from the discharge. The annual maximum measured surface temperatures at Location 508 in the years 2006 to 2011 ranged from 92.5 °F in 2009 to 94.8°F in 2008. The annual maximum temperatures were similar to values reported in 1995 and 2007 reports. The report also noted that no exceedances of permit thermal limits occurred over the 2006 – 2011 period.

As noted above, N. fowleri is rarely found in water cooler than 95°F, and infection rarely occurs in water temperatures of 95°F or less. While the immediate discharge area could have temperatures in the summer above 95°F, the maximum temperatures recorded 200 meters from the discharge were below 95°F, indicating lower risk. In addition, the discharge point and this monitoring point is located in an area of deep water, approximately 23 meters. The N. fowleri infection risk is higher in shallow, warm water.

The Friends of Lake Keowee published an article in their newsletter in 2016 from Dr. J. Hains of Clemson University that addressed the risk posed by ONS's heated discharge for promoting N. fowleri. Dr. Hains wrote, "The temperature at which this organism grows best is reported to be far greater (approximately 112°F) than the ONS discharge. Moreover, that water originates in the coldest, deepest depths of Lake Keowee, not an optimal habitat. To my knowledge there have been no studies of the distribution of this organism in Lake Keowee (or in other nearby lakes in recent times)."

Additionally, for the first license renewal of ONS, Duke Energy consulted with SCDHEC to determine if the continued operation of Oconee will have public health impacts due to the enhancement of thermophilic organisms. By letter dated October 25, 1996, Dr. John F. Brown, State Toxicologist at SCDHEC, summarized the agency's position and opinion regarding the public health implications of continued operation of Oconee. Regarding the potential public health hazard from pathogenic microorganisms whose abundance might be promoted by ONS's artificial warming of recreational waters, Dr. Brown indicated that there seems to be no significant threat to off-site persons near such heated recreational waters.

As stated earlier, this letter seeks your input on any potential public health concerns associated with our proposed continued operation of ONS. We appreciate your notifying us of your comments and any information you believe Duke Energy should consider in the preparation of the ER. Duke Energy plans to include this letter and any response you provide in the ER.

Should you or your staff have any questions or comments, please contact Mike Ruhe at (980) 373-3231 / Mike.Ruhe@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

<u>Attachments:</u>

Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

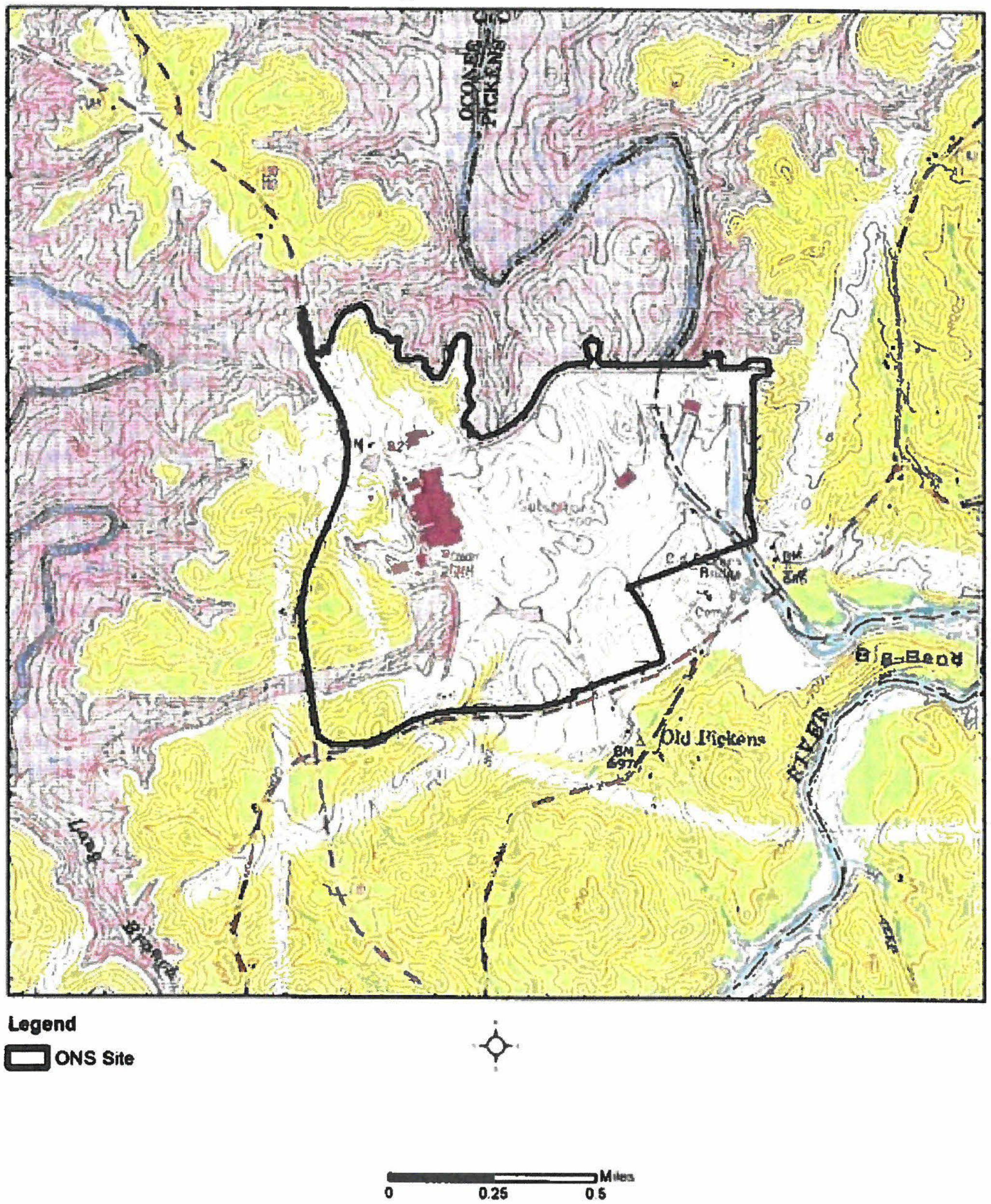
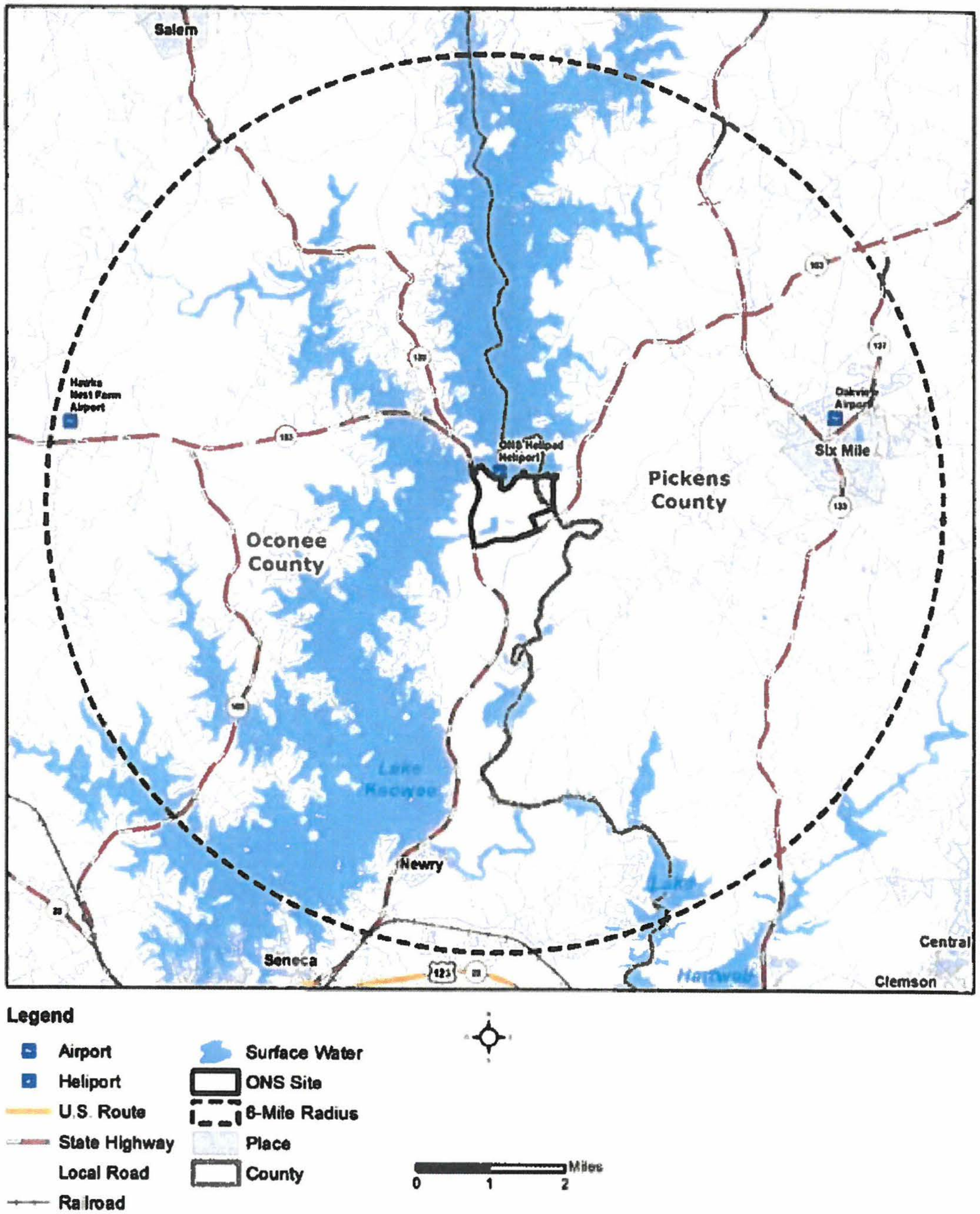


Figure 2. ONS 6-mile Vicinity



cc: Myra Reece
SCDHEC
2600 Bull Street
Columbia, SC 29201

January 8, 2021

Mr. Arun Kapur
Duke Energy
Lead Environmental Specialist
410 S. Wilmington Street NCRH15
Raleigh, North Carolina 27601

Re: Oconee Nuclear Station Units 1,2 and 3 Subsequent License Renewal – Consultation on Thermophilic Microorganisms

Dear Mr. Kapur:

After a review of the report on the impact of thermophilic organisms associated with discharge from the Oconee Nuclear Station (ONS) prepared by Enercon, DHEC technical staff does not take any exception with the findings, including that "there is little reason to believe that the actions of the ONS should promote or increase the potential for elevated exposure or risk to the public at large."

As stated in the Duke-Energy letter dated November 11, 2019, *Naegleria spp.* is ubiquitous in nature and thrives in water bodies at temperatures ranging from 95 to 106 degrees F. SCDHEC routinely warns recreational water users across the state that there is always a low risk of infection from naturally occurring organisms like *N. fowleri* in warm freshwater.

In our view, the Enercon consultant has met the NRC requirement of plant-specific analysis of this issue. The discharge temperature, location and demonstration monitoring all support the conclusion that an increased risk to the public to exposure to thermophilic microorganisms is not expected.

We agree with your consultant that the conditions that led the former DHEC State Toxicologist who evaluated this issue and memorialized his findings in the October 1996 letter have not substantially changed and are not expected to change. As Dr. Brown referenced, there is no ongoing monitoring program for these microorganisms in this location largely because no increased risk has been identified. That remains true today.

For all of the reasons listed in the Enercon letter/report, we concur that there is "no impediment to the license renewal or continued operation of the ONS based on the potential of impacting public health and safety relative to increased potential for growth of thermophilic microorganisms."

If you have questions or need any additional information, please don't hesitate to contact me by phone (803.422.1805) or e-mail (marshaf2@dhec.sc.gov) .

Thank you,

s/Fran W. Marshall

Fran W. Marshall, JD, MSPH
Director of the Office of Applied Science

cc: Mike Ruhe, Duke Energy, Director, Environmental Policy and Affairs
Rounette Nader, Duke Energy, Director Nuclear Engineering
Myra C. Reece, DHEC Director of Environmental Affairs
Bryan Rabon, DHEC Bureau of Water, Manager Aquatic Science Programs

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Ms. Kimberly Bose, Secretary
Federal Energy Regulatory Commission
888 First Street N.E.
Washington, DC 20426

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Ms. Bose:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships.

Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

Sincerely,

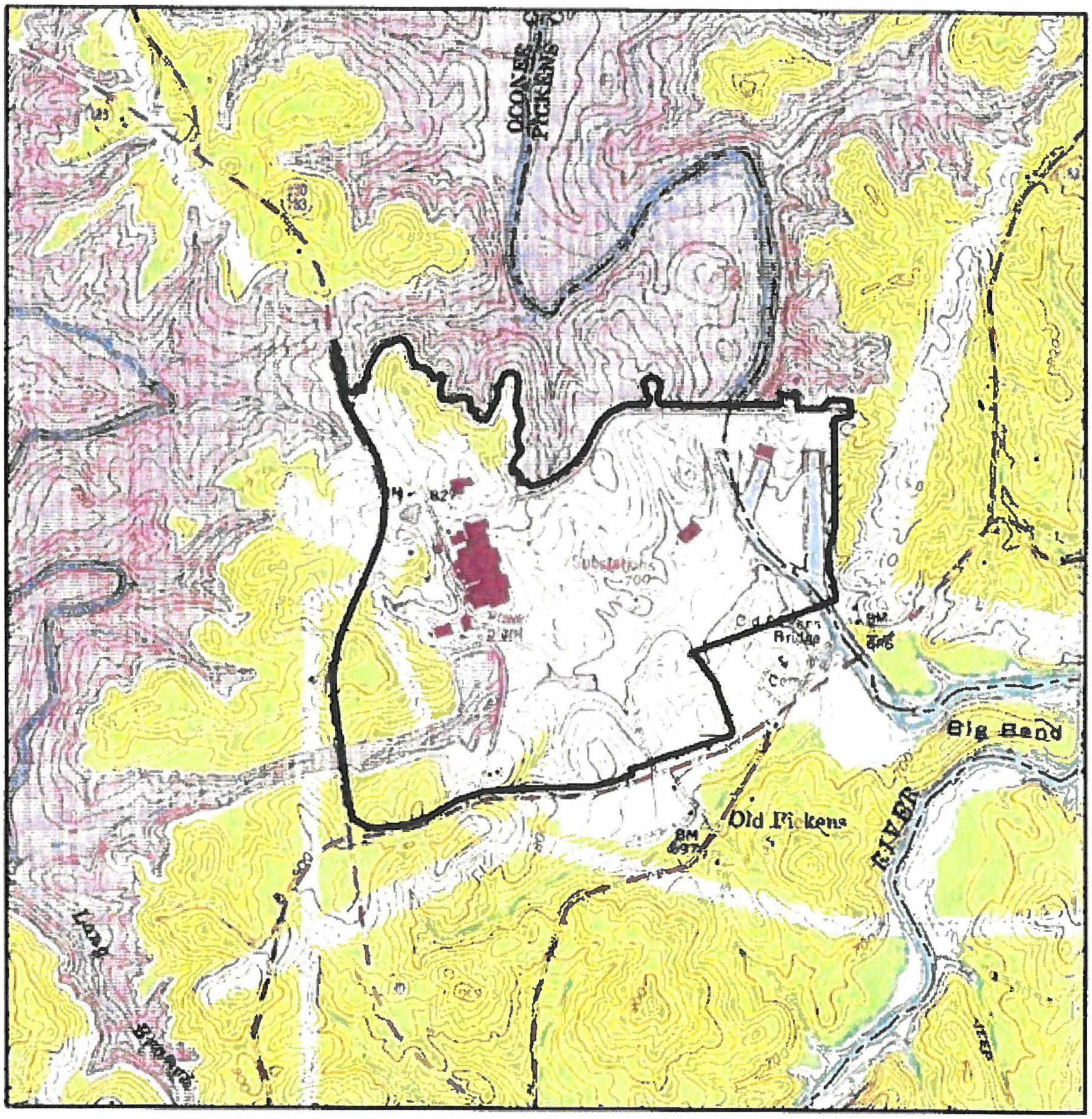
J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:

Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

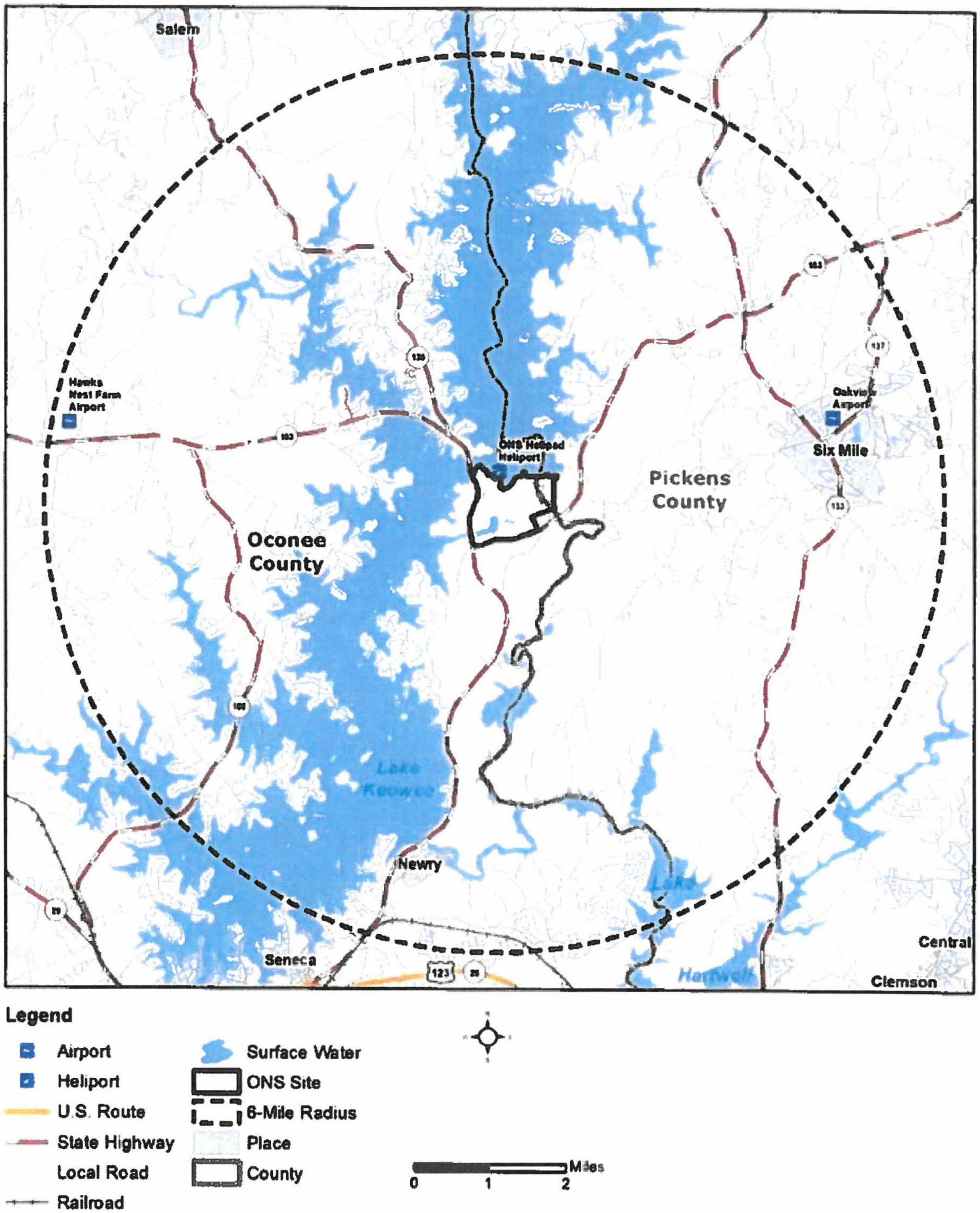


Legend

ONS Site

0 0.25 0.5 Miles

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Doug Hof, President
Friends of Lake Keowee Society
4065 Keowee School Rd
Seneca, SC 29672

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Hof:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com or Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

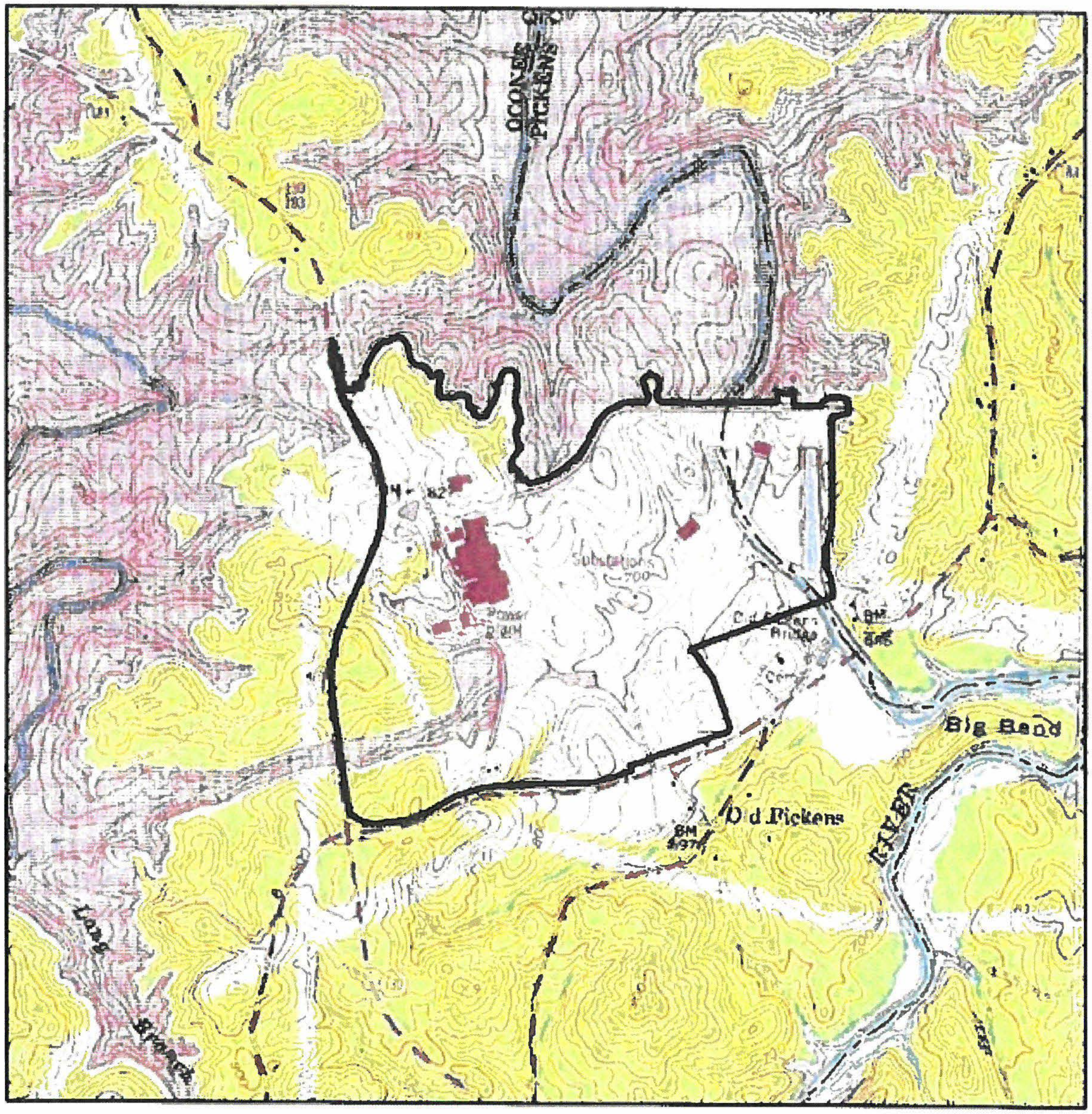
Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

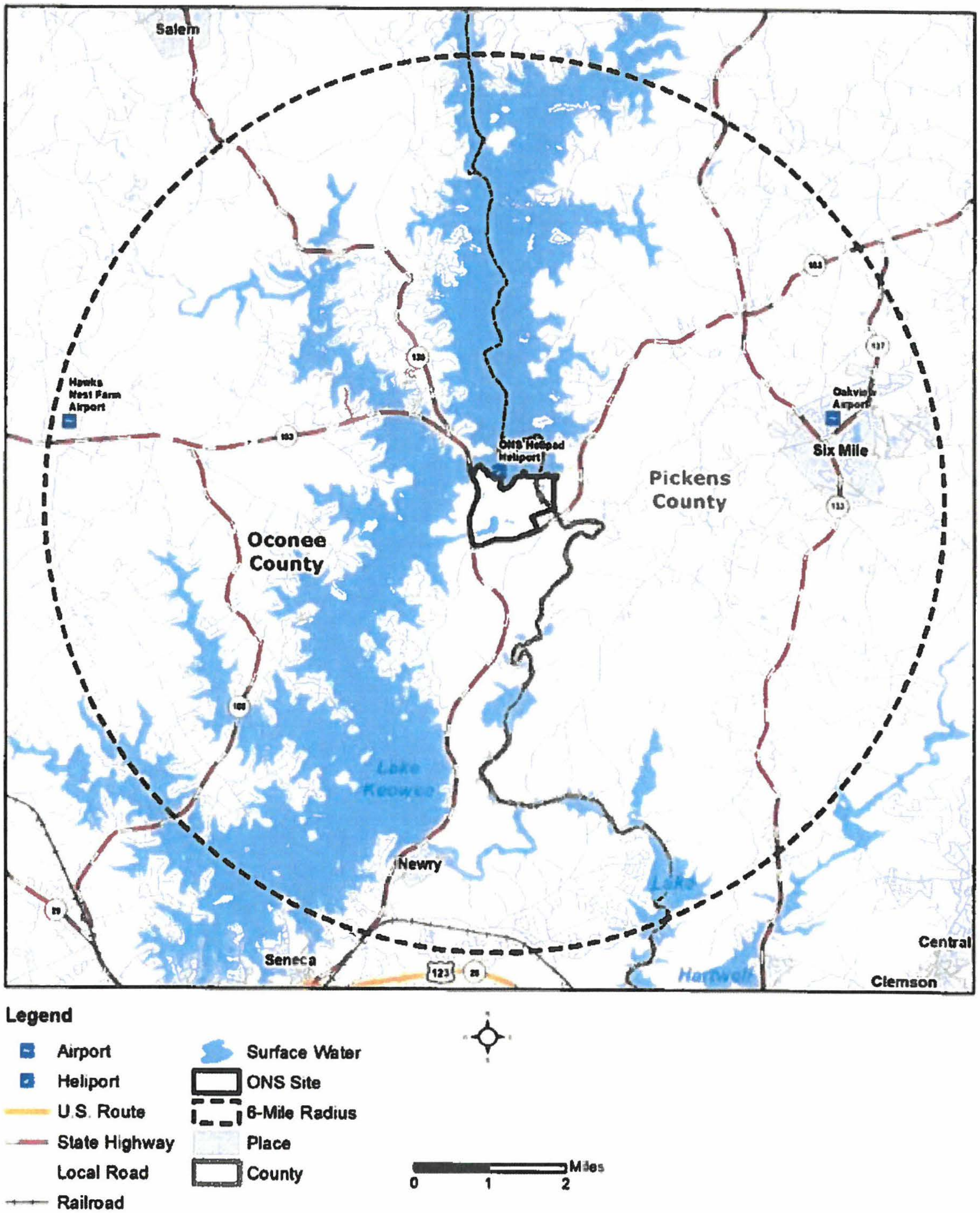


Legend

ONS Site

0 0.25 0.5 Miles

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Rob Auhlebach
Advocates for Quality Development
P.O. Box 802
Seneca, SC 29679

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Auhlebach:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com or Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

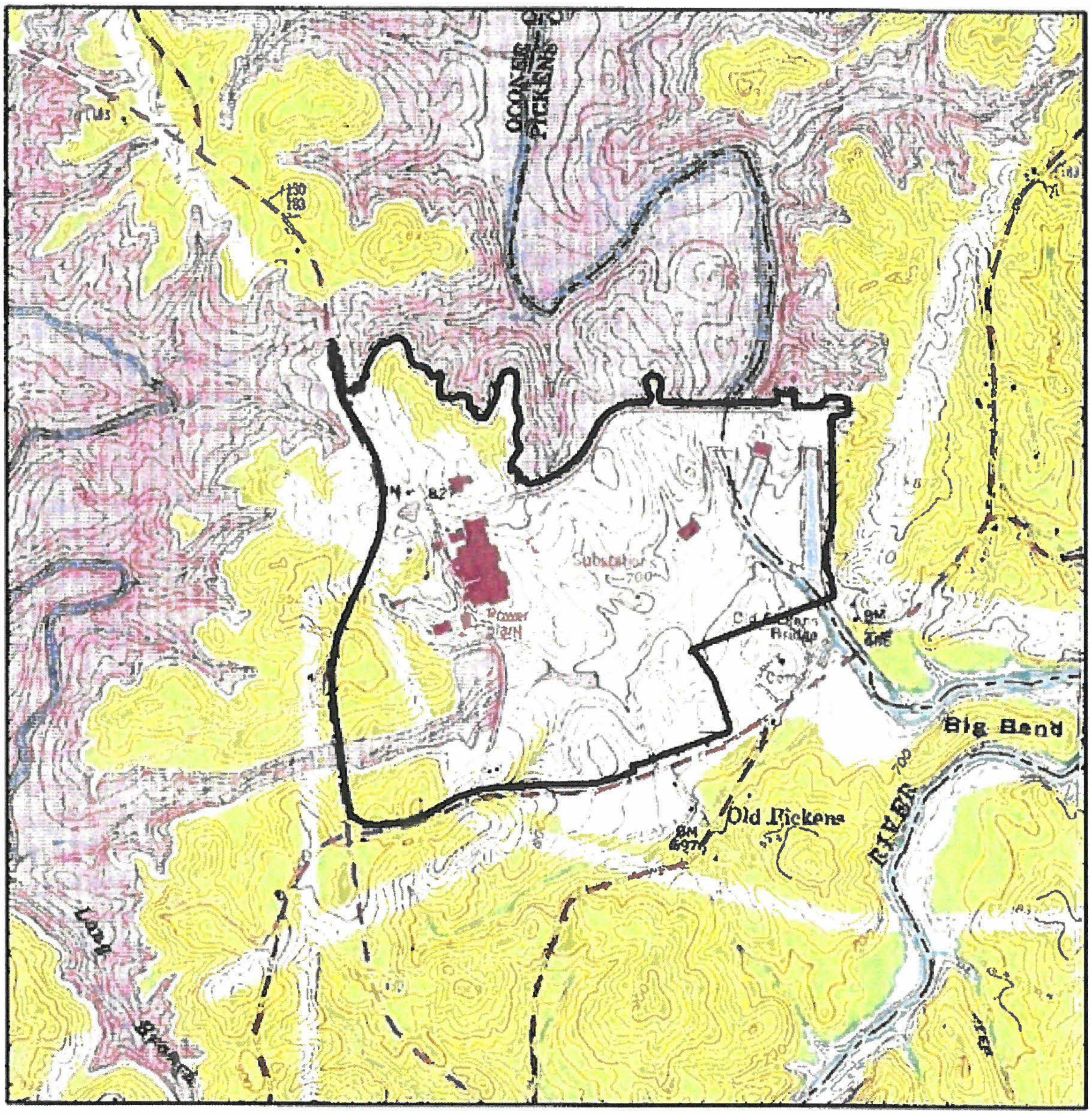
Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

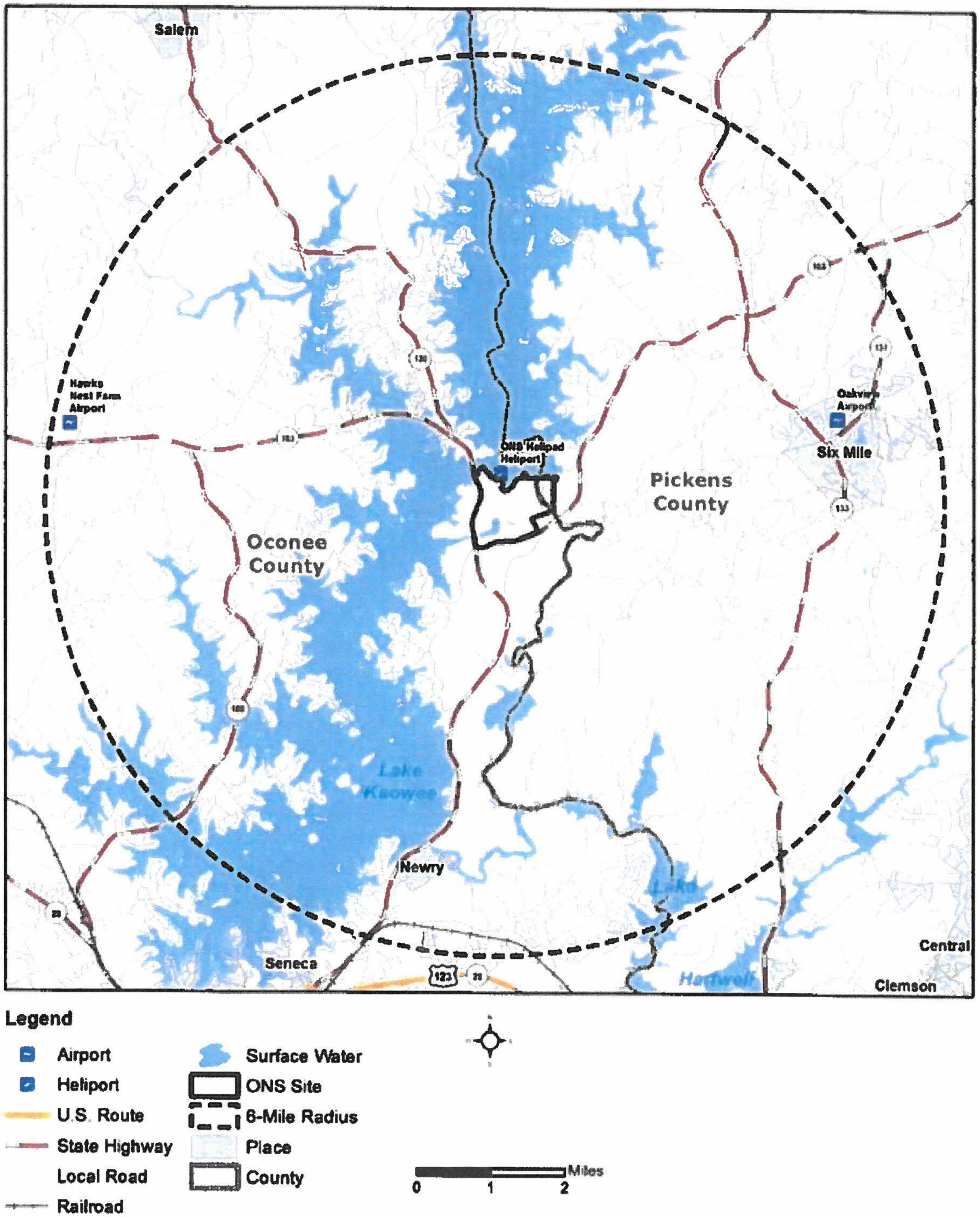


Legend

ONS Site

Miles
0 0.25 0.5

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Chris Starker, Land Conservation Manager
Upstate Forever
507 Pettigru St
Greenville, SC 29601

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Starker:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com or Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

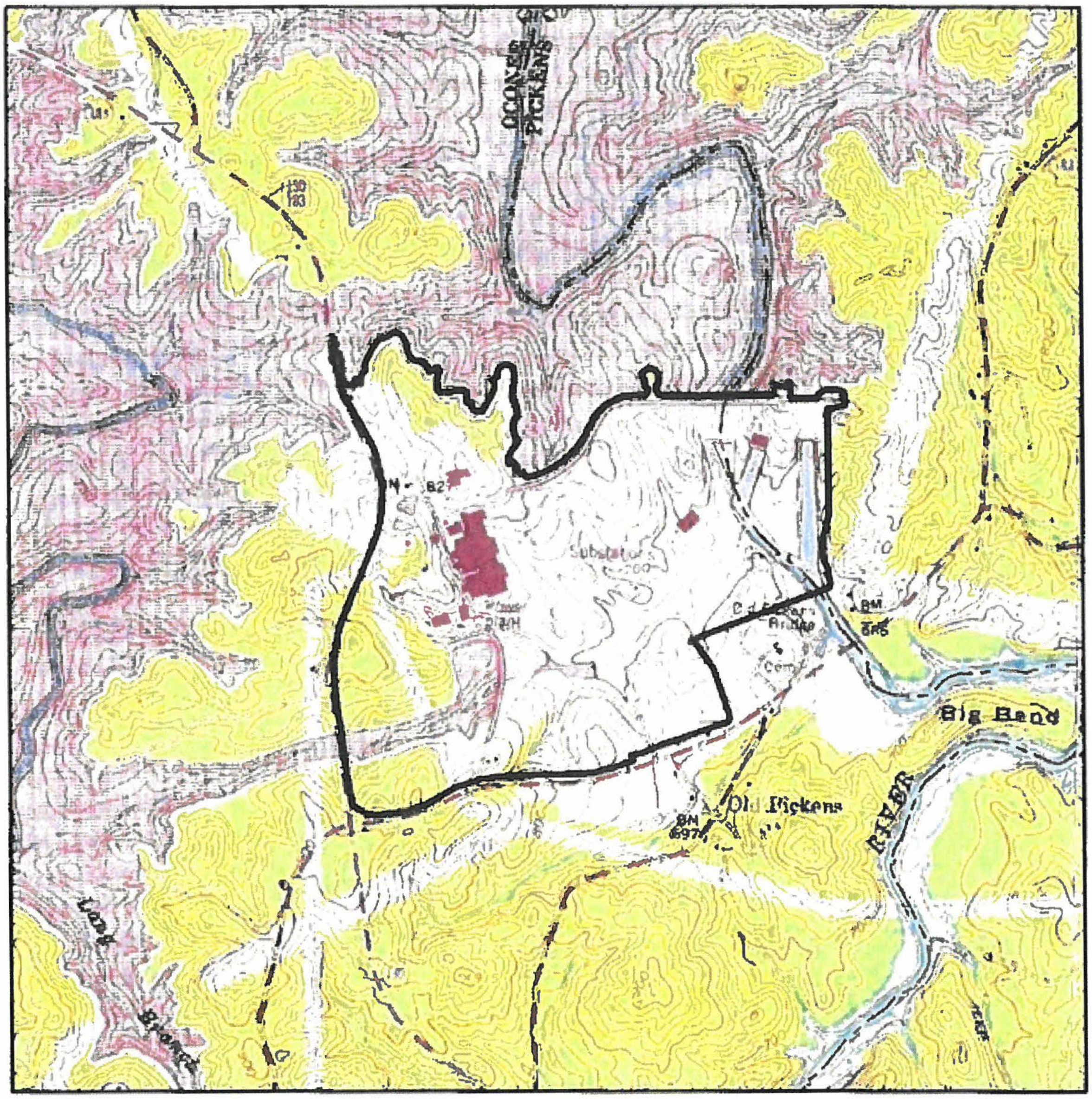
Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

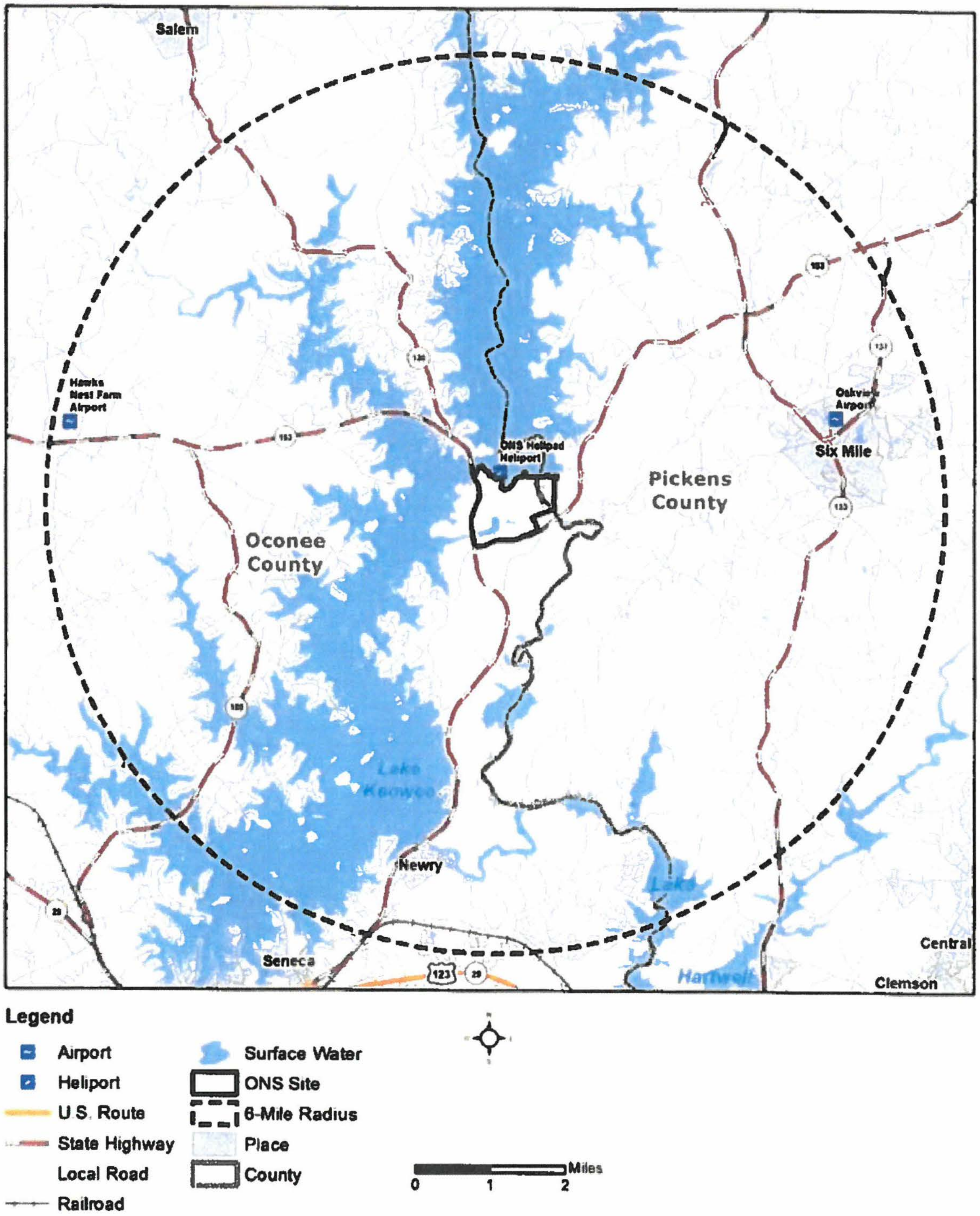


Legend

ONS Site

0 0.25 0.5 Miles

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Sara Green, Executive Director
SC Wildlife Federation
215 Pickens Street
Columbia, SC, 29205

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Ms. Green:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com or Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

Sincerely,

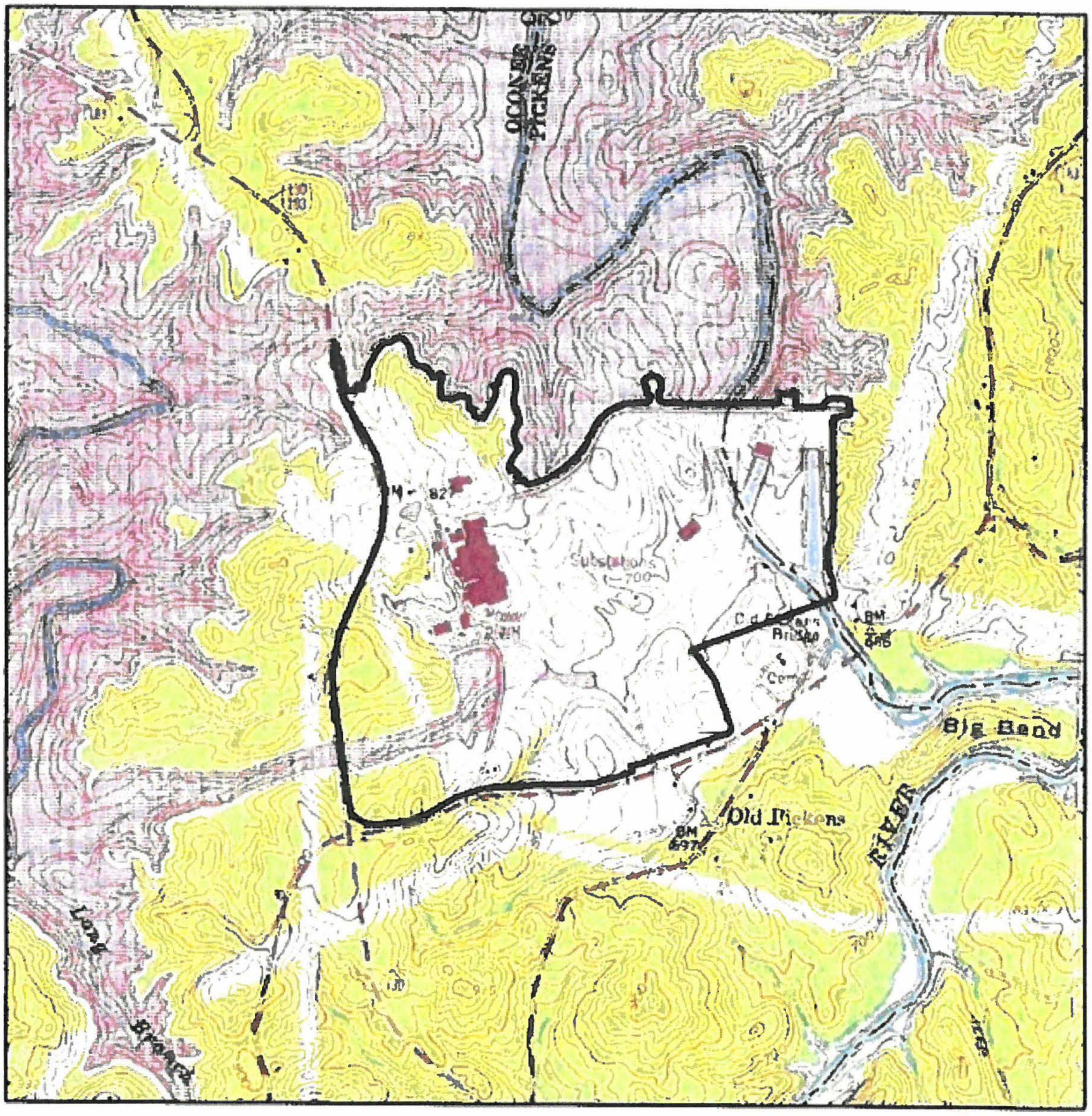
J Ed Burchfield Jr

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:

Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

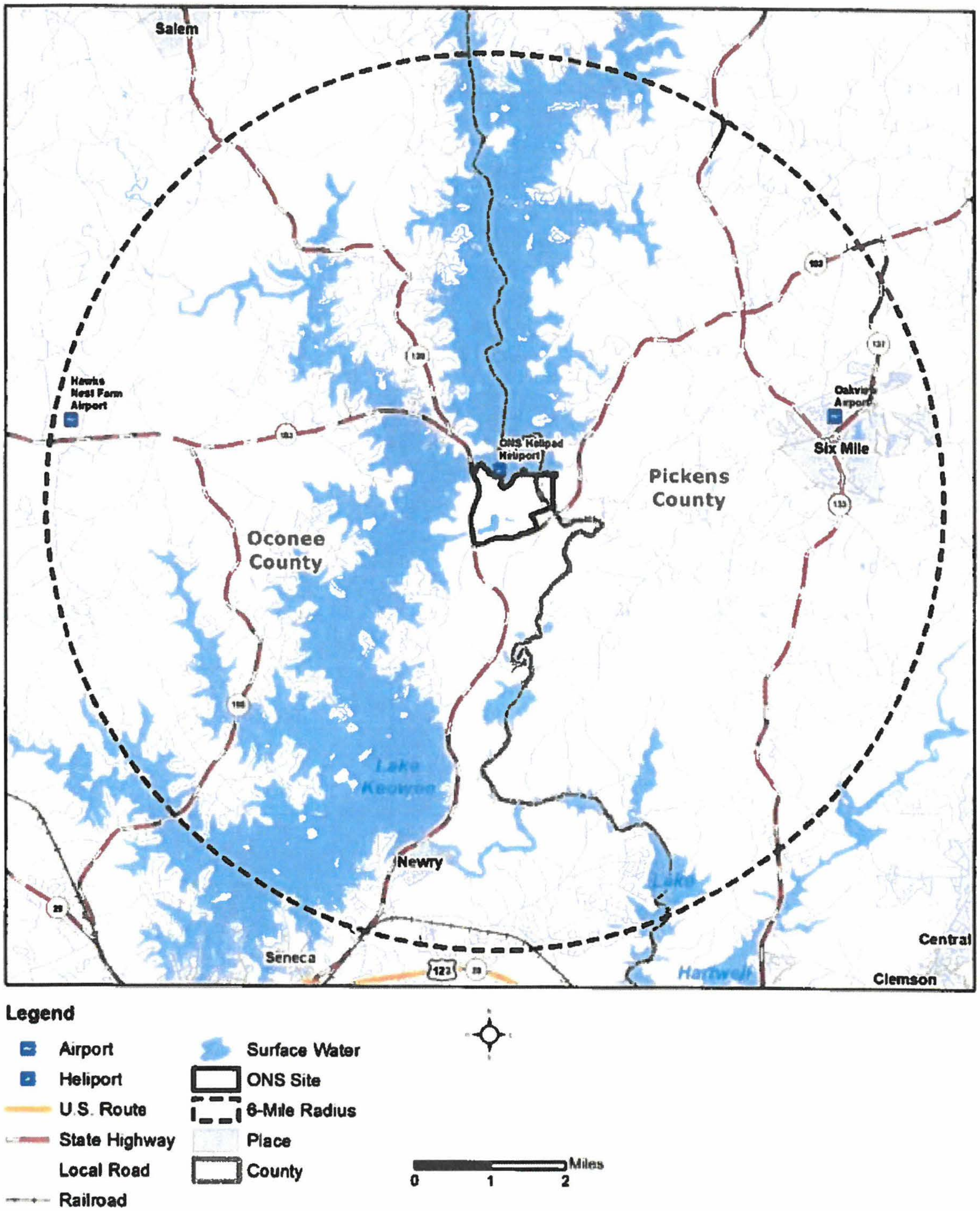


Legend

ONS Site

Miles
0 0.25 0.5

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Douglas Spencer
Southeastern Power Administration
1166 Athens Tech Rd.
Elberton, GA 30635

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Spencer:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Ed Bruce at (704) 382-5239 / Ed.Bruce@duke-energy.com.

Sincerely,

J Ed Burchfield Jr

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

<u>Attachments:</u>
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

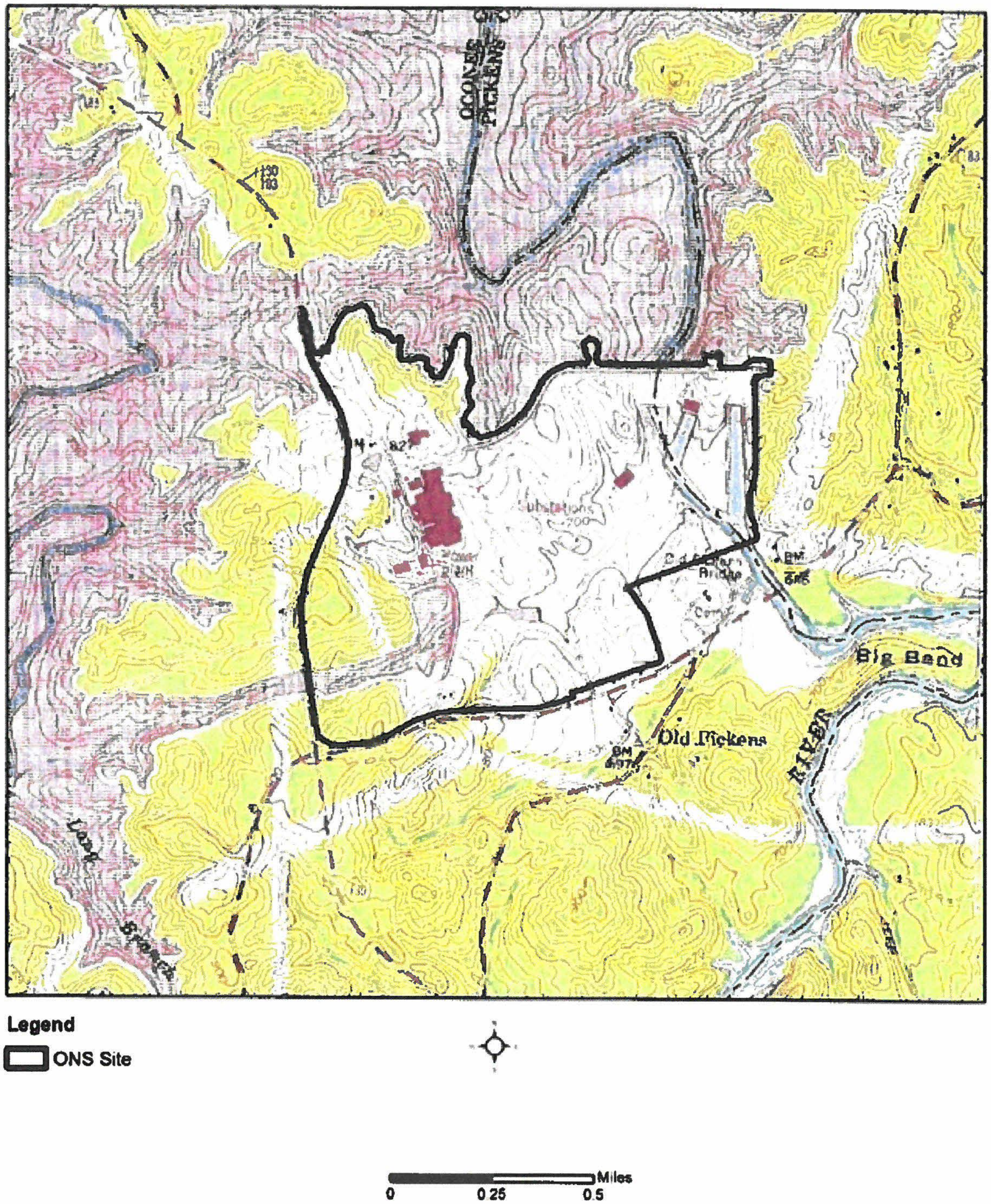
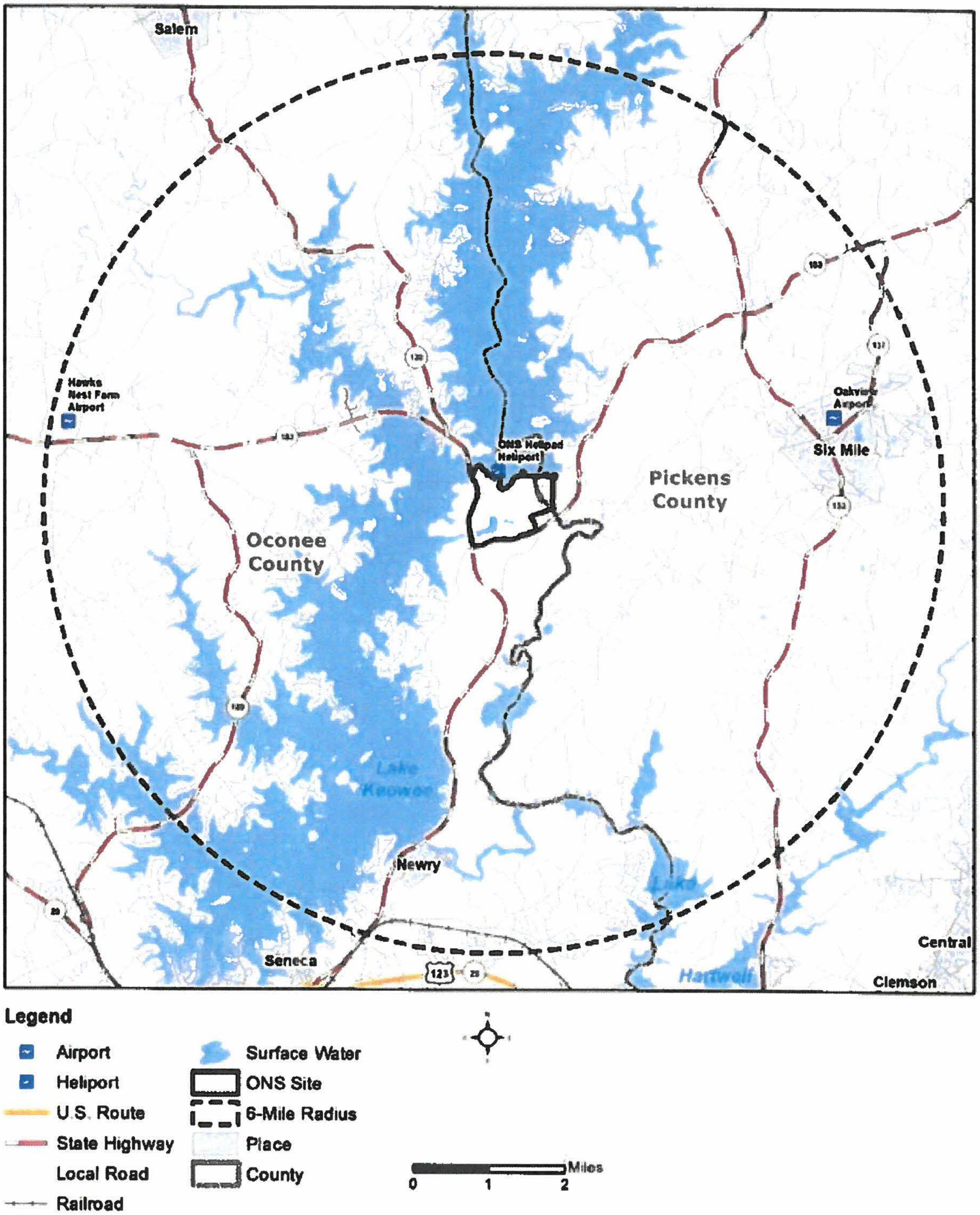


Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Amanda Brock, County Administrator
Oconee County
415 S Pine St
Walhalla, SC 29691

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Ms. Brock:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

cc: Phil Shirley, Director
Oconee County Parks, Recreation and Tourism
1031 South Dover Rd,
Seneca, SC 29672

Figure 1. ONS Site

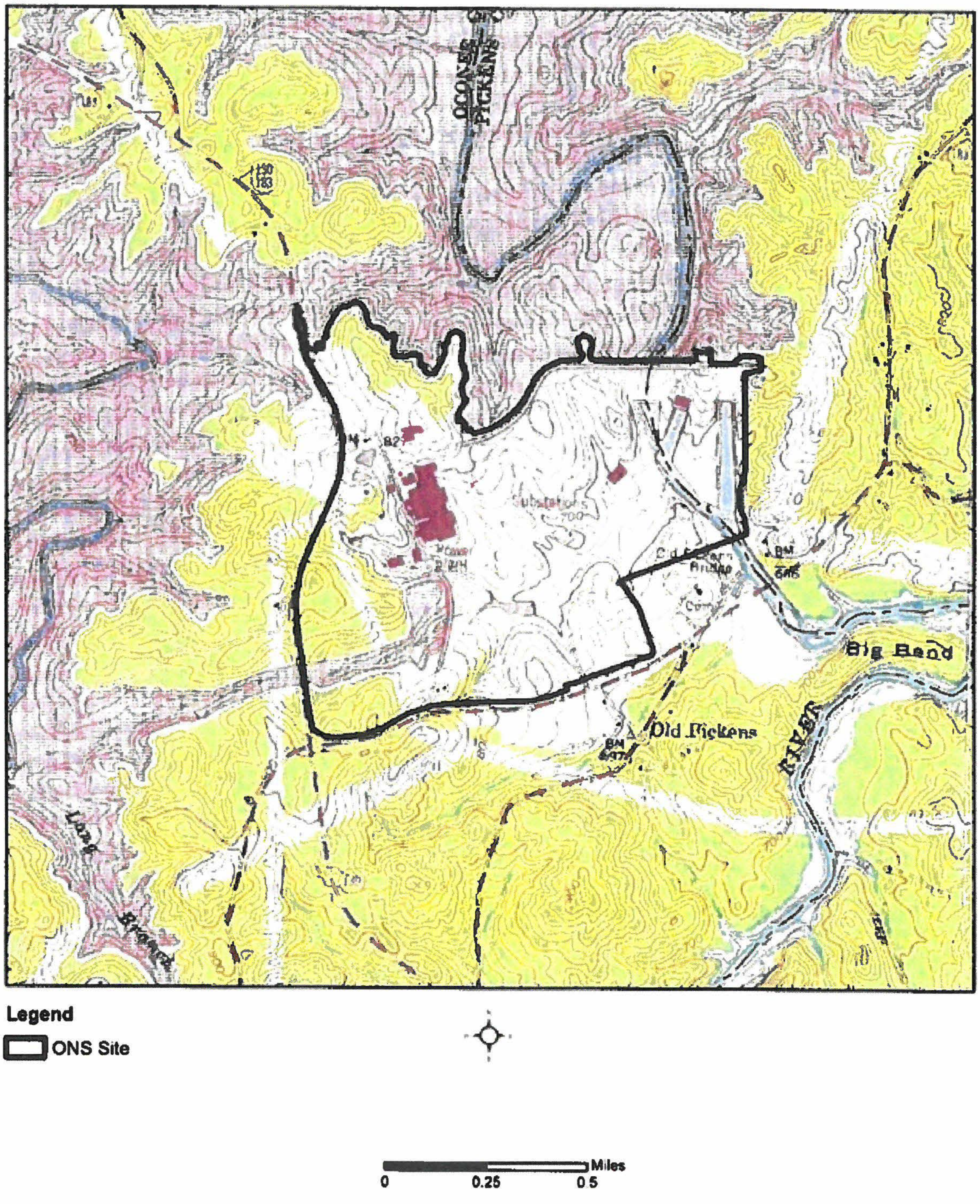
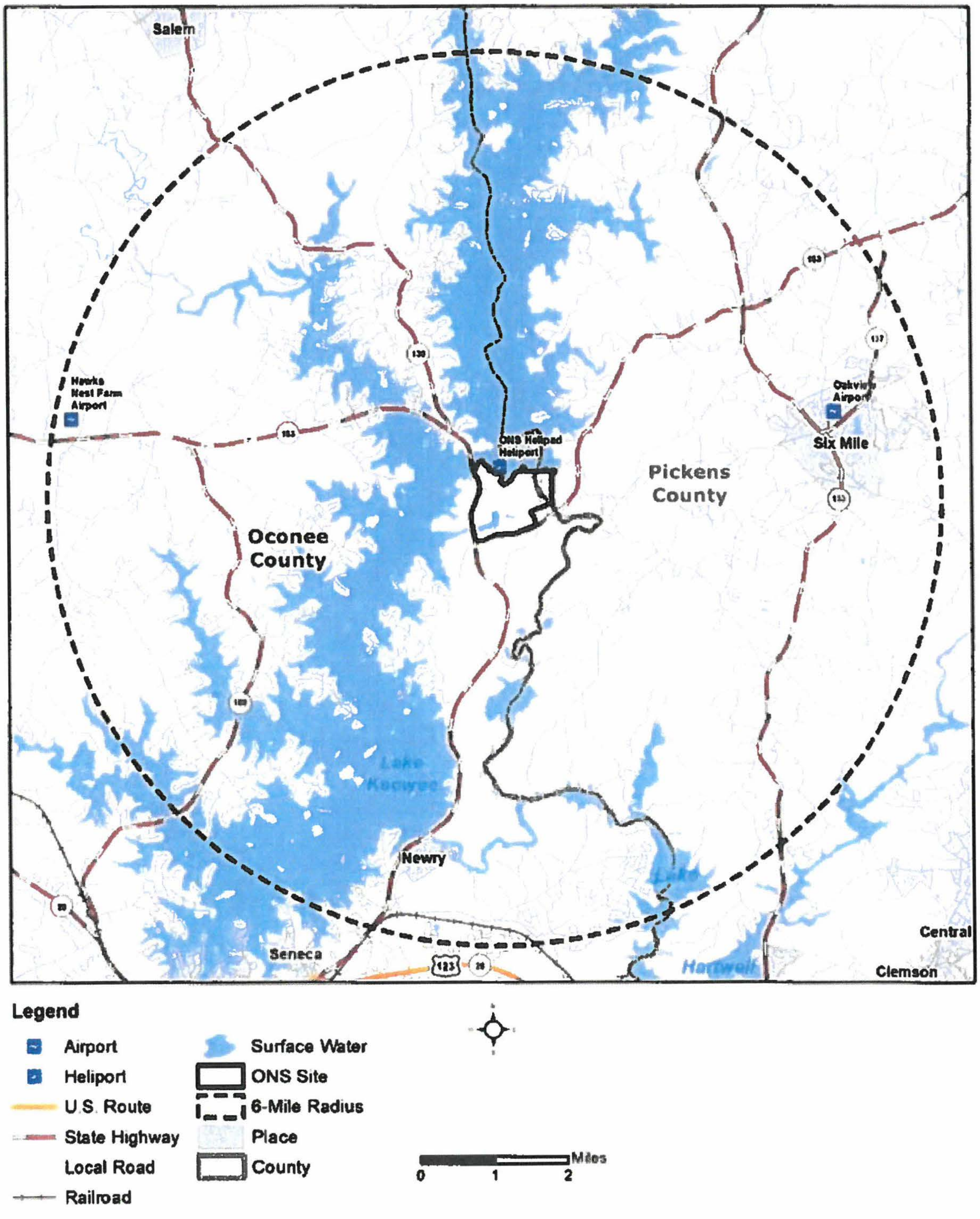


Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Ken Roper, County Administrator
Pickens County
222 McDaniel Ave Ste B2,
Pickens, SC 29671

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Roper:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:

Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

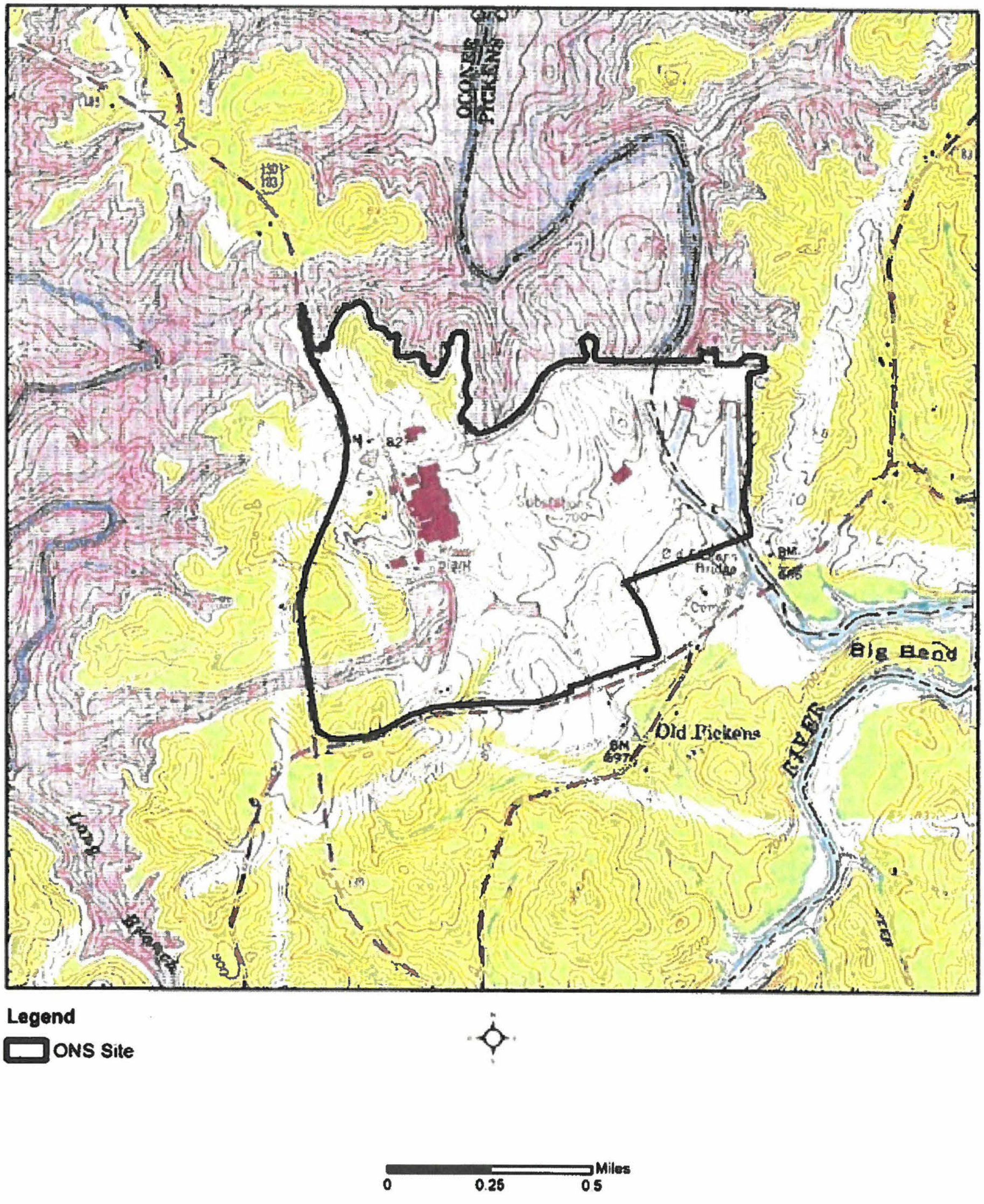
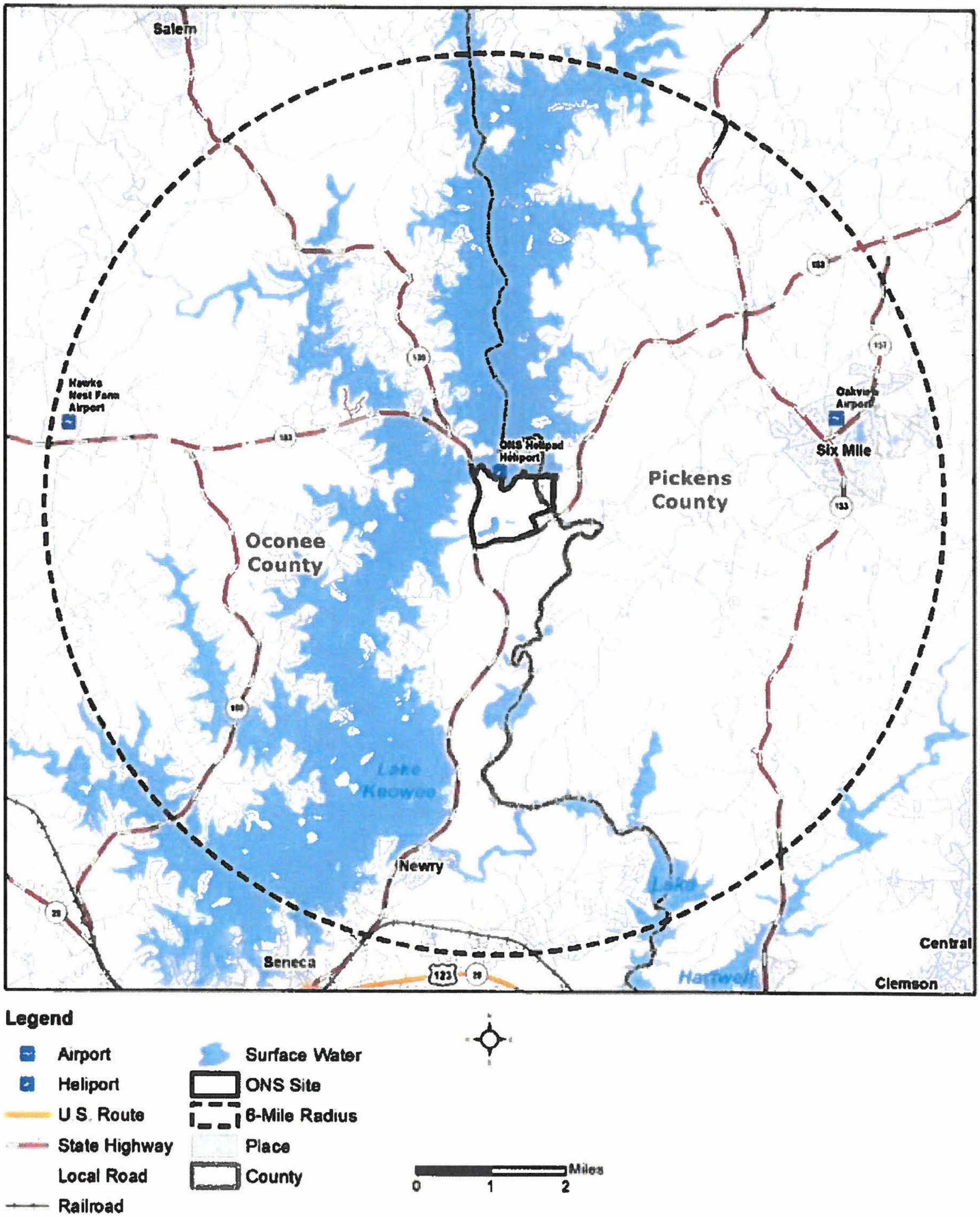


Figure 2. ONS 6-mile Vicinity



DUKE ENERGY.

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Paul McCormack, Director
SC Department of Parks, Recreation and Tourism
1205 Pendleton St
Columbia, SC 29201

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. McCormack:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Mikayla Kreuzberger at (864) 873-4204 / Mikayla.Kreuzberger@duke-energy.com or Scott Fletcher at (980) 875-6014 / Scott.Fletcher@duke-energy.com.

Sincerely,

J Ed Burchfield Jr

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

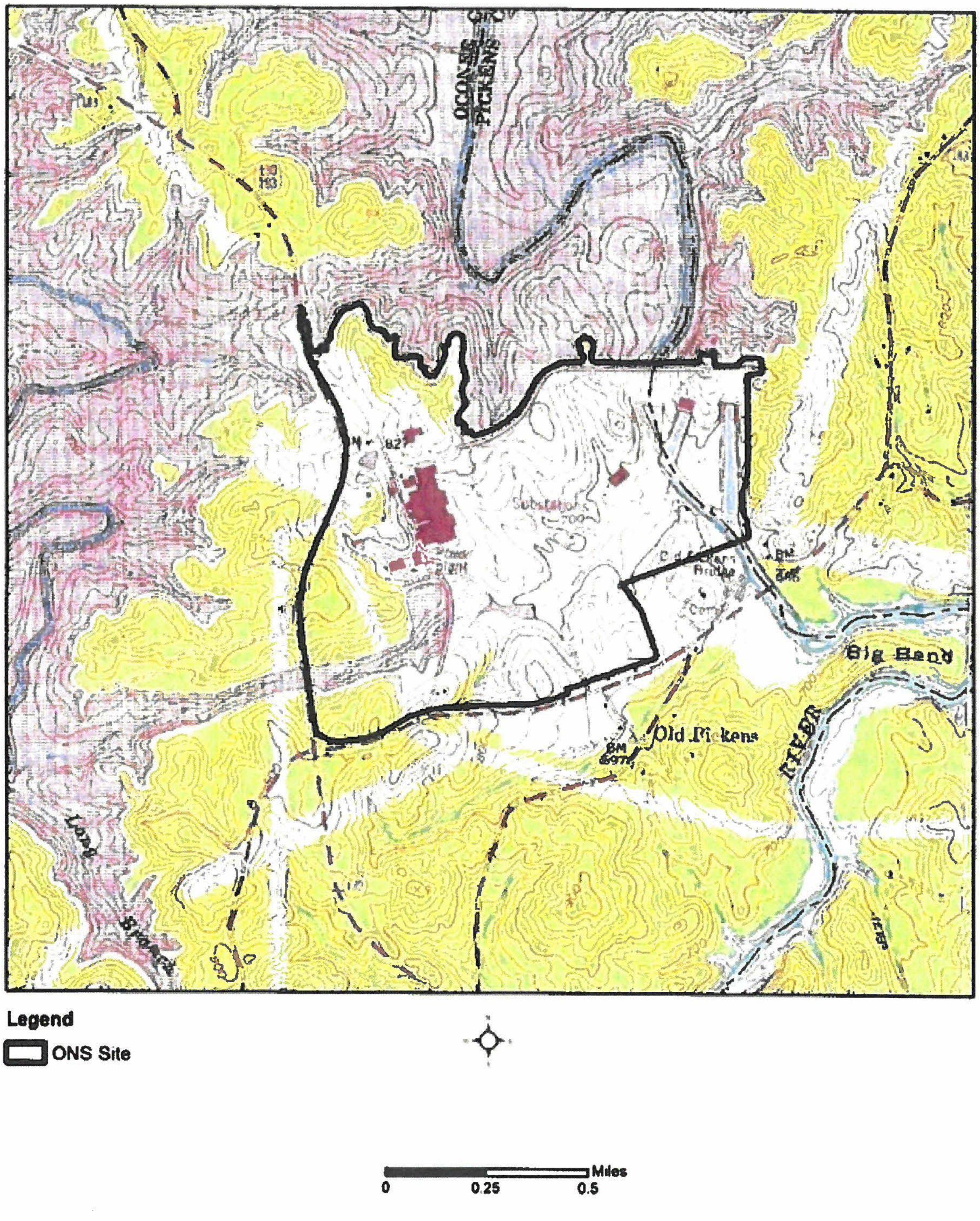
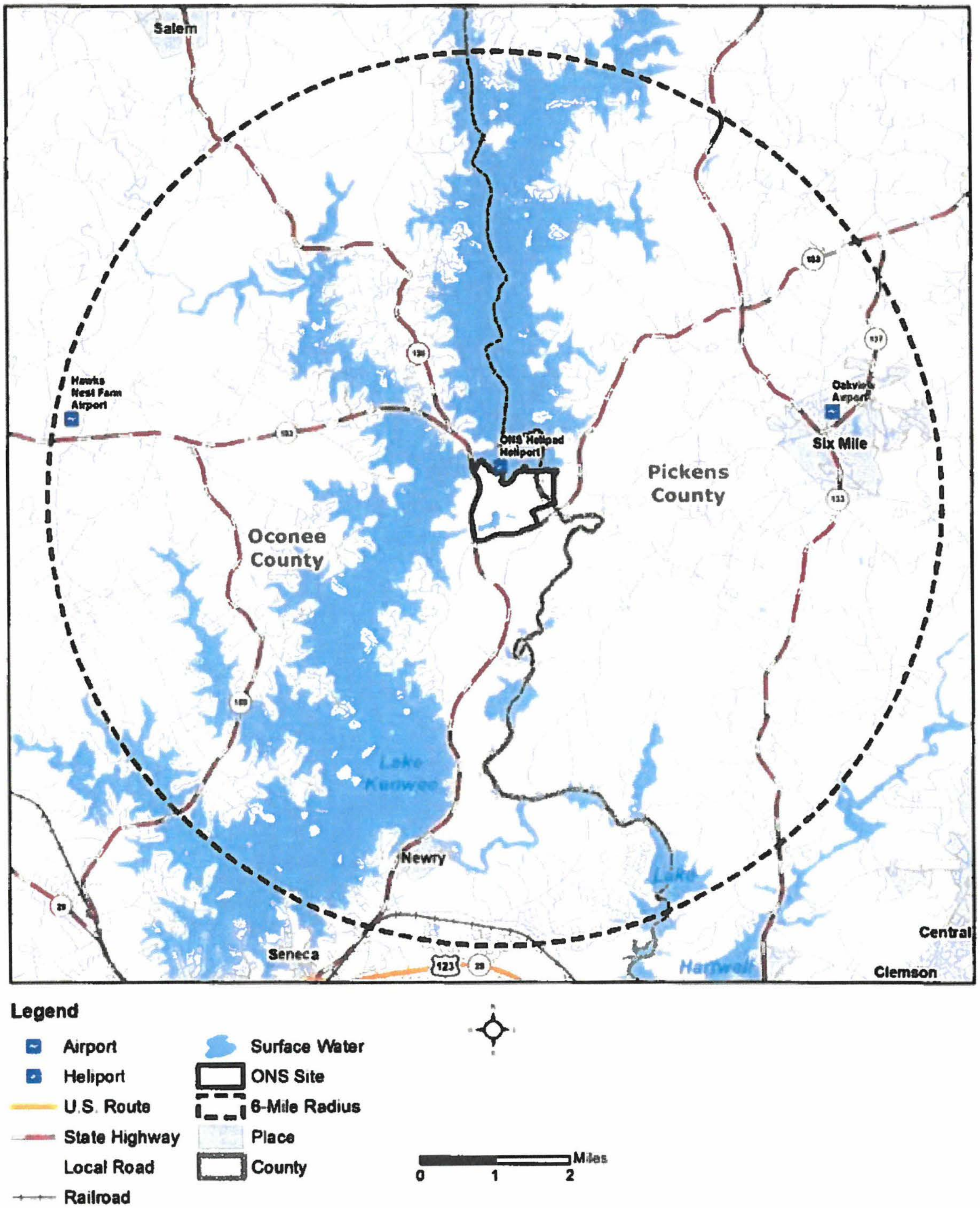


Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Jeff Phillips, Director of Water Resources
Greenville Water System
407 West Broad Street
Greenville, SC 29601

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Phillips:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Ed Bruce at (704) 382-5239 / Ed.Bruce@duke-energy.com.

Sincerely,

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

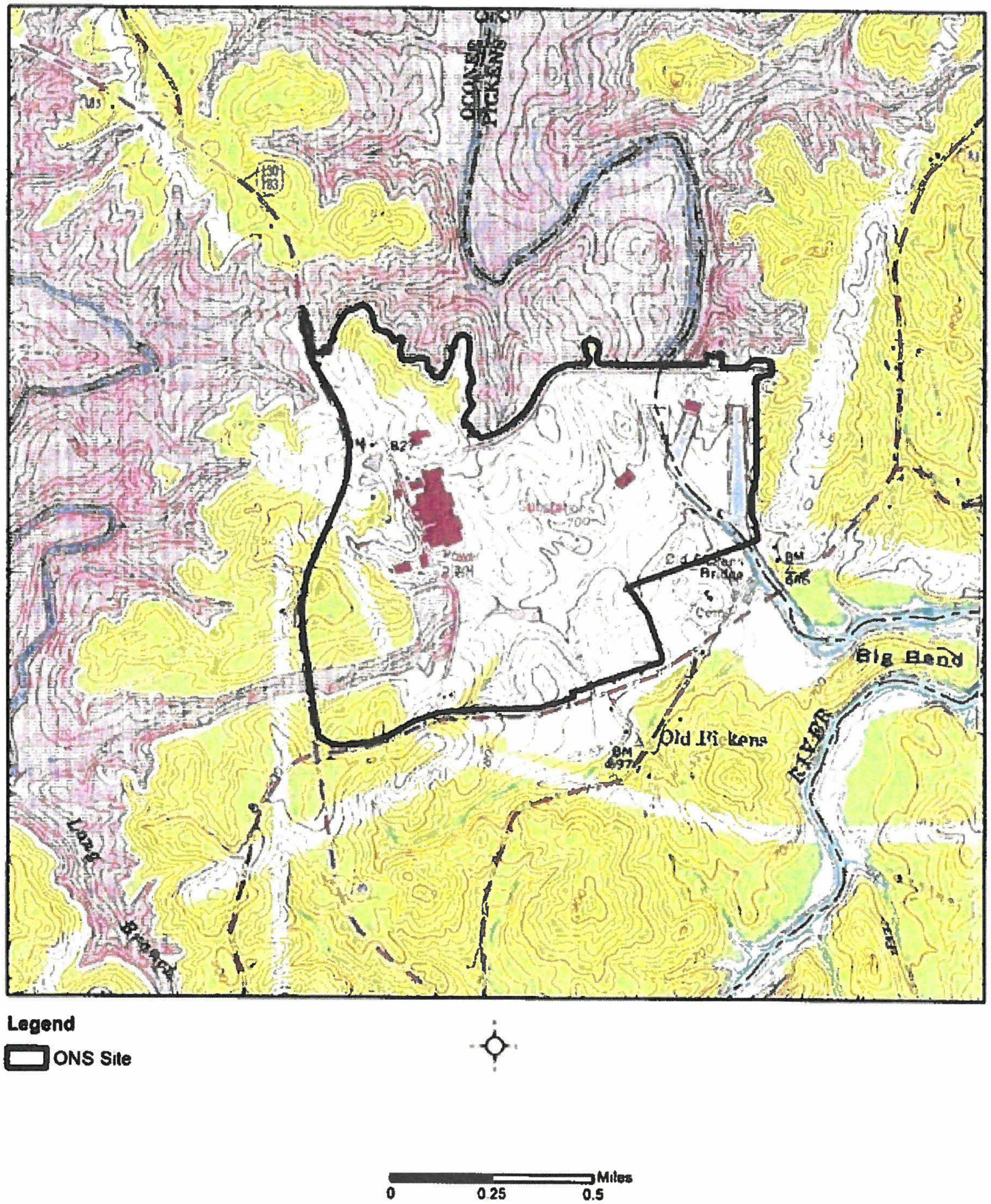
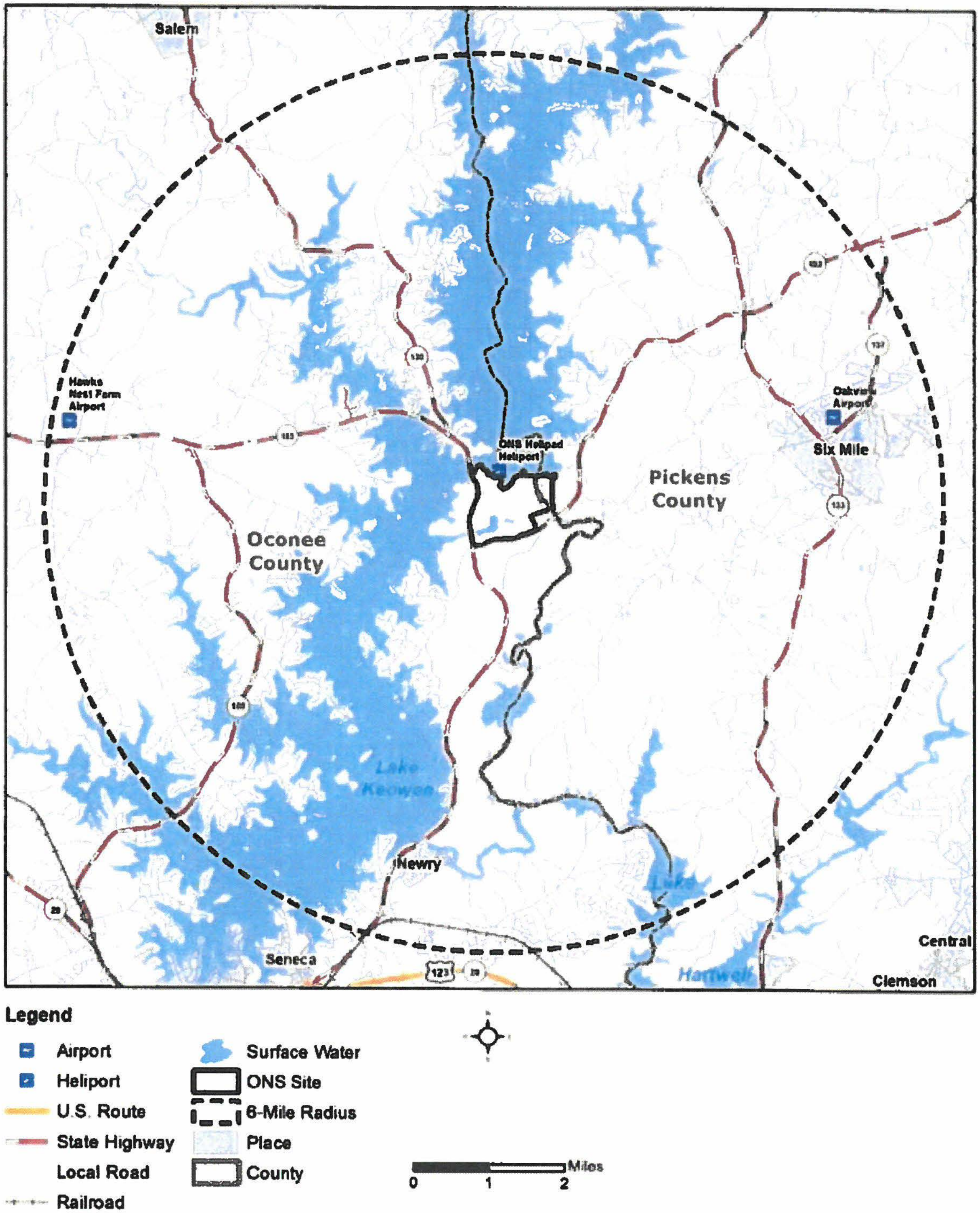


Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Bob Faires, Director
Seneca Light and Water
PO Box 4773
Seneca, SC 29679

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Faires:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Tony Garland at (864) 873-4216 / Tony.Garland@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

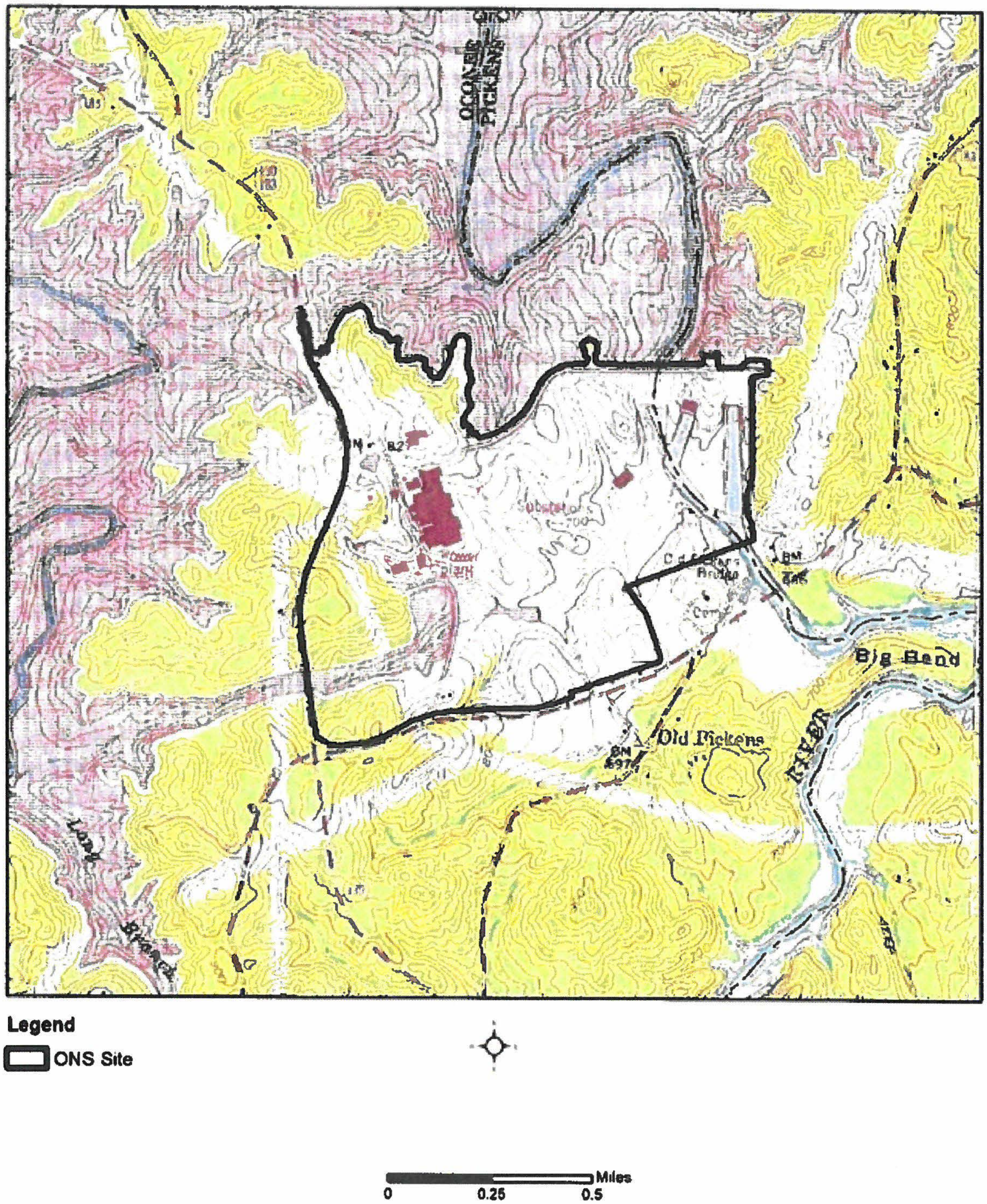
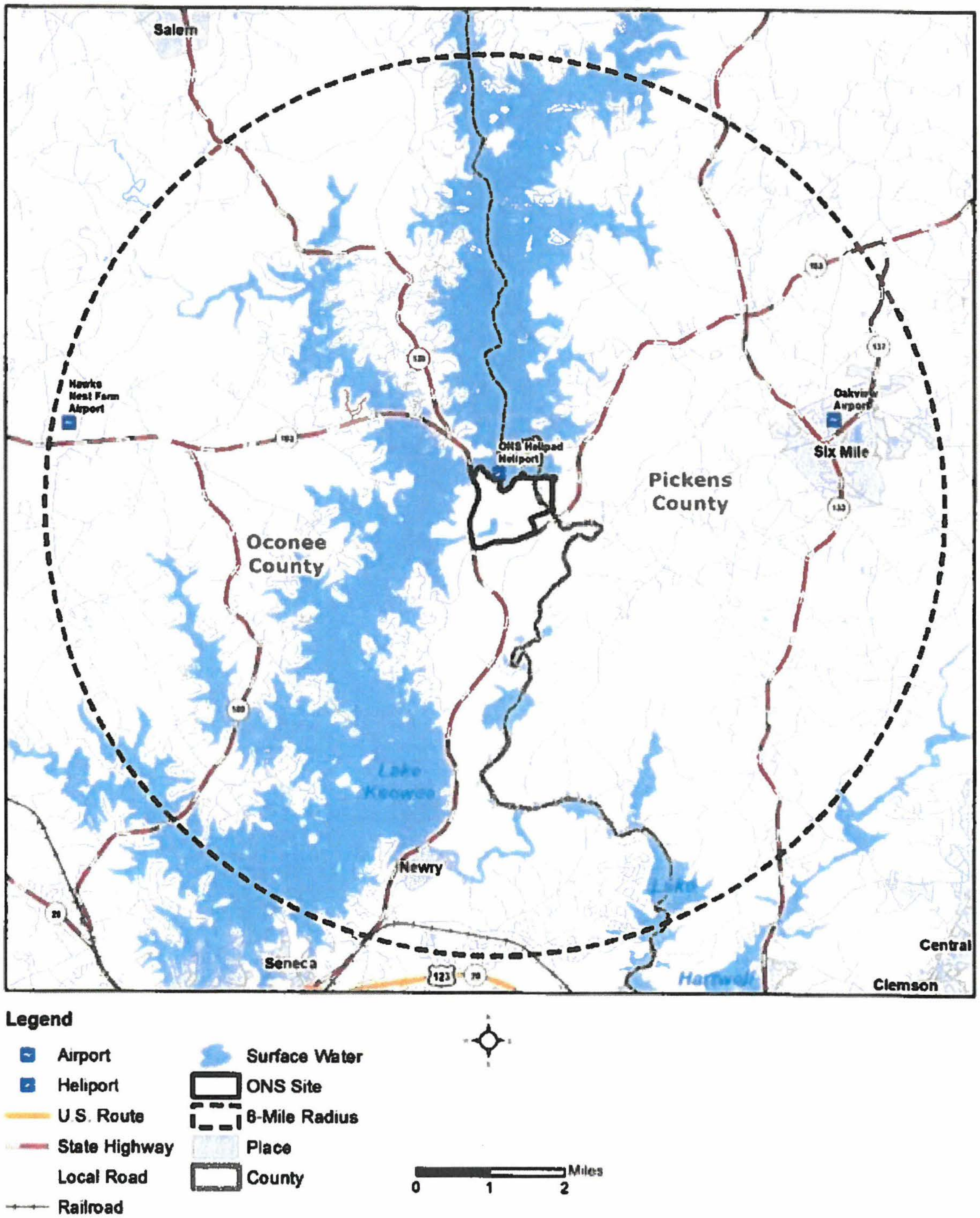


Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Scott Parris
City of Walhalla
PO Box 1099
Walhalla, SC 29691

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Parris:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Ed Bruce at (704) 382-5239 / Ed.Bruce@duke-energy.com.

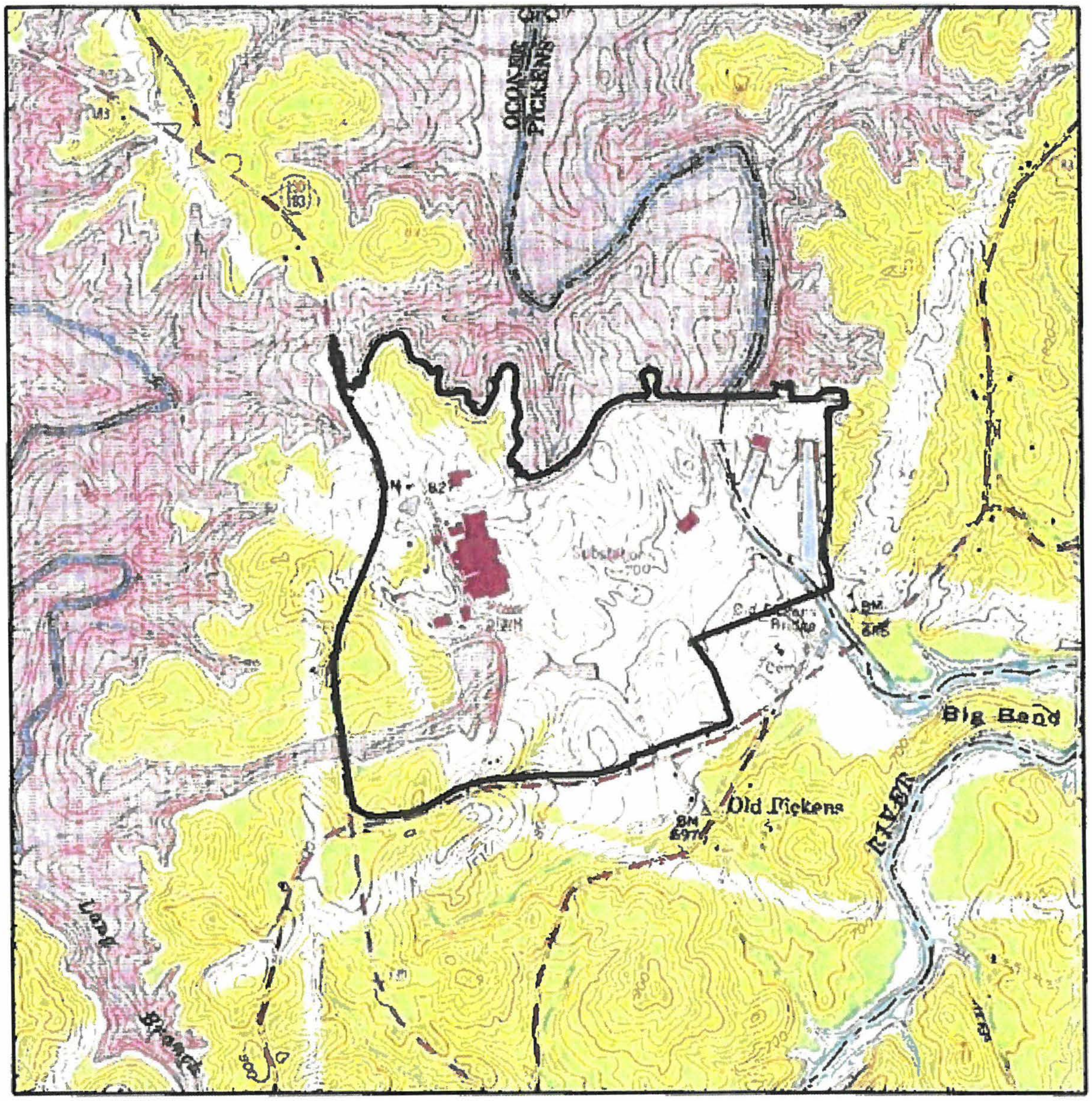
Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

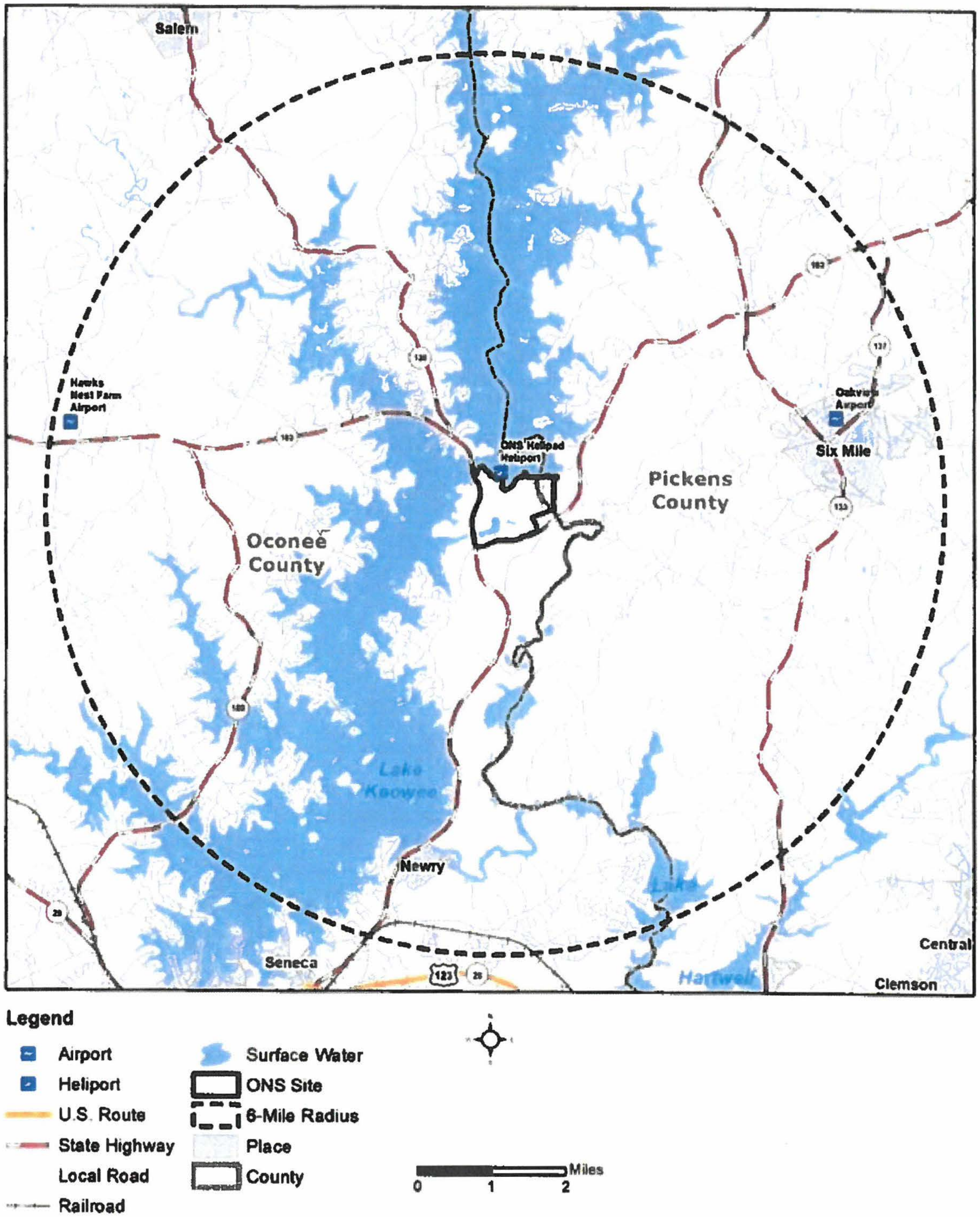


Legend

ONS Site

0 0.25 0.5 Miles

Figure 2. ONS 6-mile Vicinity



J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o 864.873.3478
f 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 7, 2019

Mr. Chris Eleazer, Director
Oconee Joint Regional Sewer Authority
623 Return Church Road
Seneca, SC 29678

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Eleazer:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Tony Garland at (864) 873-4216 / Tony.Garland@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

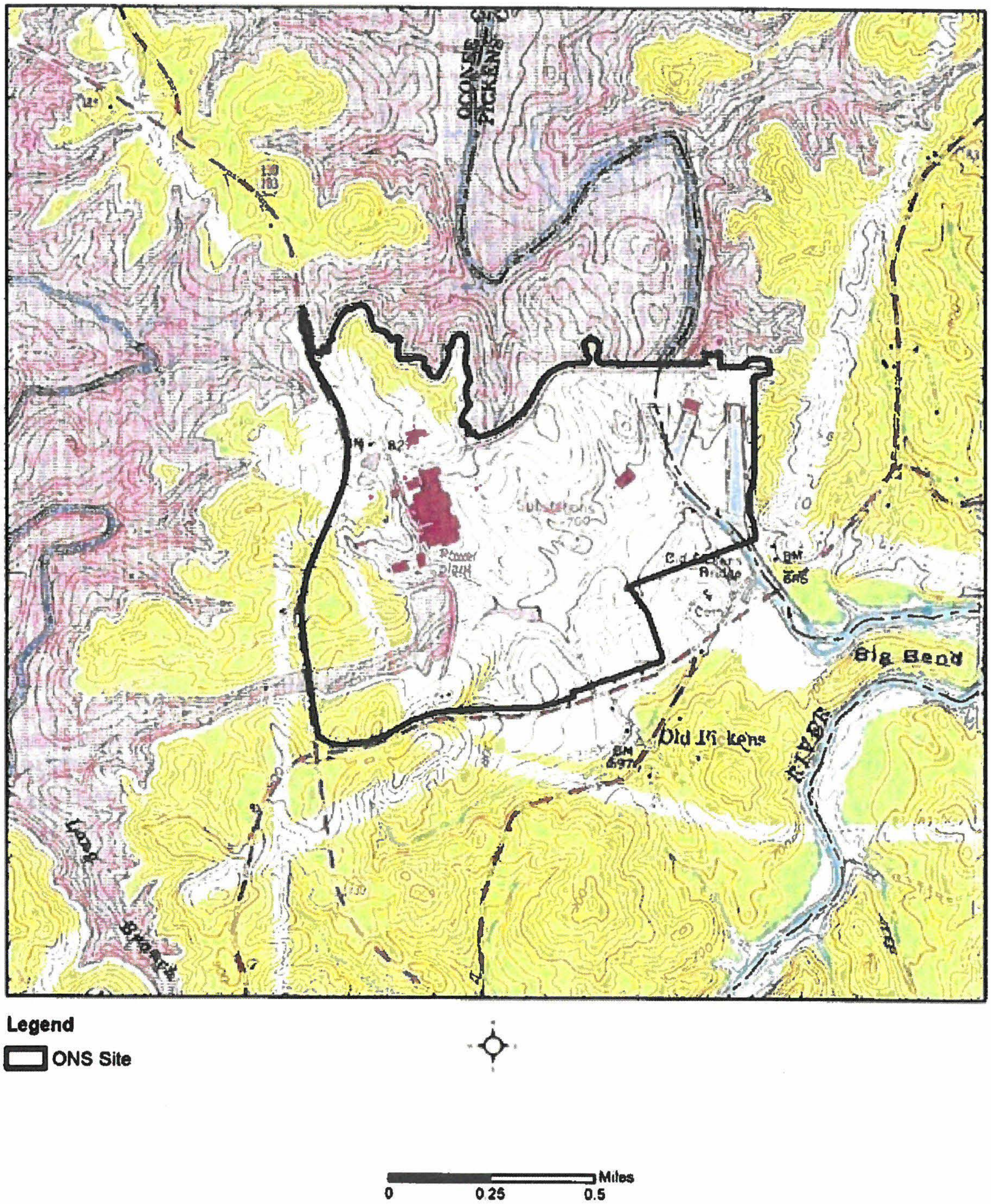
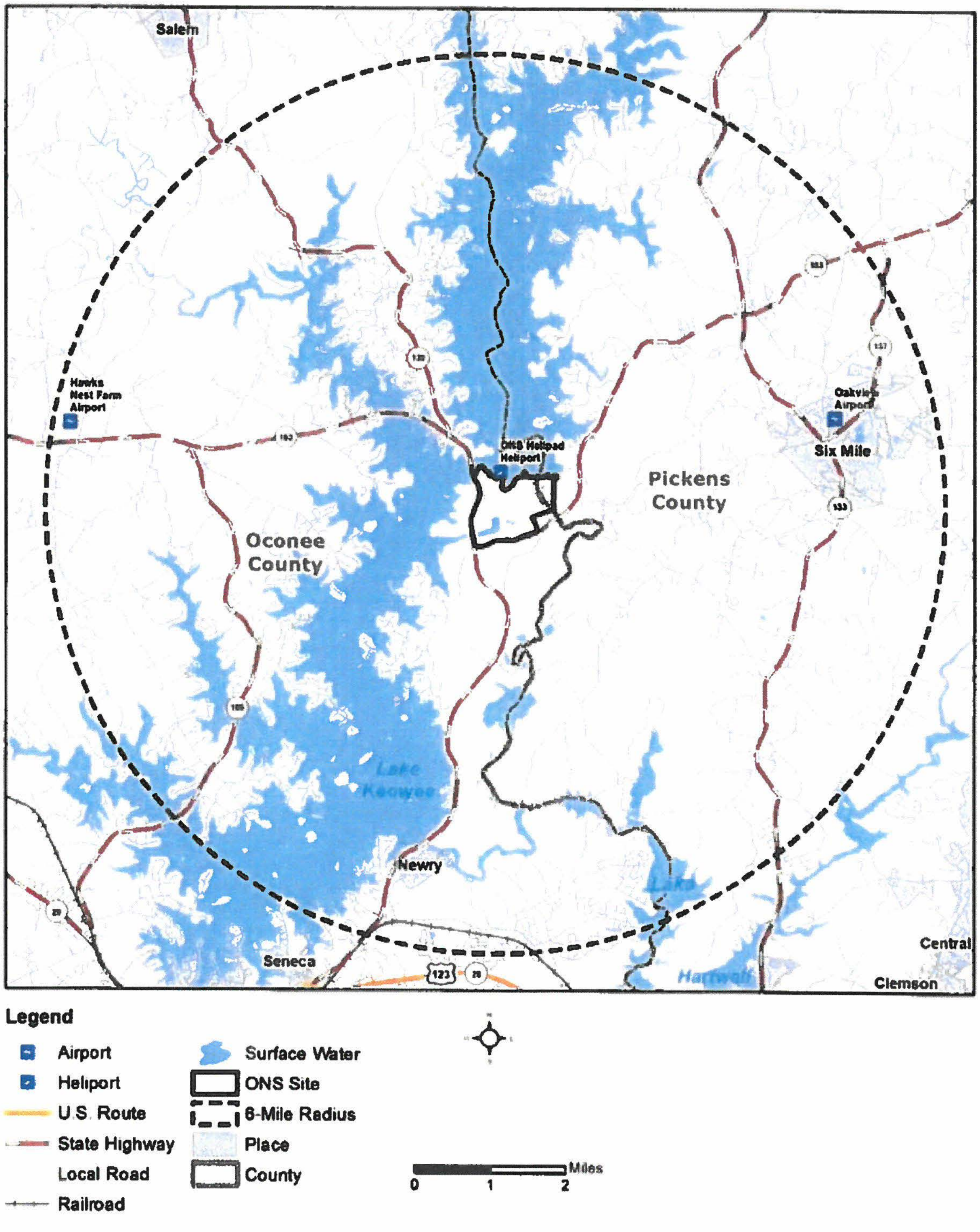


Figure 2. ONS 6-mile Vicinity



DUKE ENERGY.

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672
o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

BY U.S. MAIL
RETURN RECEIPT REQUESTED

November 11, 2019

Stan Simpson
United States Army Corps of Engineers
Savannah District
100 W. Oglethorpe Ave
Savannah, GA 31401

RE: Duke Energy – Oconee Nuclear Station Units 1, 2, and 3 Subsequent License Renewal

Dear Mr. Simpson:

Duke Energy Carolinas, LLC (Duke Energy) is seeking to renew the operating license for Oconee Nuclear Station Units 1, 2, and 3 (ONS) for an additional 20 years (see Table 1). Duke Energy and ONS have safely and reliably provided electricity to our Carolinas customers for decades. ONS has generated clean and cost-effective power, provided thousands of well-paying jobs, and produced substantial economic benefits for the Carolinas. Renewing the licenses of ONS is important for our customers, communities and environment.

Table 1. ONS Licensing Dates

ONS Unit	License Expiration Date	Extended License Expiration Date
Unit 1	February 6, 2033	February 6, 2053
Unit 2	October 6, 2033	October 6, 2053
Unit 3	July 19, 2034	July 19, 2054

Duke Energy's nuclear fleet plays an important role in the company's efforts to lower carbon emissions. In 2018, the Duke Energy nuclear fleet generated more than 72 billion kilowatt-hours of electricity and avoided the release of about 54 million tons of carbon dioxide – equivalent to keeping more than 10 million passenger cars off the road. The company has set aggressive carbon reduction goals of at least 50% by 2030 and net-zero by 2050 and keeping its nuclear fleet operating is key to achieving these goals.

Renewing the nuclear licenses will provide significant value to Duke Energy customers, as well as continue to support Carolinas communities through jobs, tax revenues and partnerships. Duke Energy employs about 5,000 workers in its nuclear group, with additional contract workers supporting refueling outages and project work. In 2018, the Duke Energy also paid more than $300 million in property and payroll taxes associated with the nuclear stations, benefiting local governments and school districts. In addition, nuclear employees support the communities where they live and work by donating time and funds through sponsorships and volunteer activities.

The ONS site is situated on 510 acres in eastern Oconee County, South Carolina (SC), approximately eight miles northeast of Seneca, SC, on the southern shore of Lake Keowee. During the license renewal term, Duke Energy proposes to continue operating the units as currently operated. There are currently no ground-disturbing activities or refurbishment anticipated at the ONS site during the subsequent license renewal period. Figures depicting the station site and the vicinity within a 6-mile radius of the station are enclosed.

Should you or your team have any questions or comments about ONS or the license renewal process, please contact Alan Stuart at (980) 373-2079 / Alan.Stuart@duke-energy.com.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr
Site Vice President
Oconee Nuclear Station

Attachments:
Figure 1. ONS Site
Figure 2. ONS 6-mile Vicinity

Figure 1. ONS Site

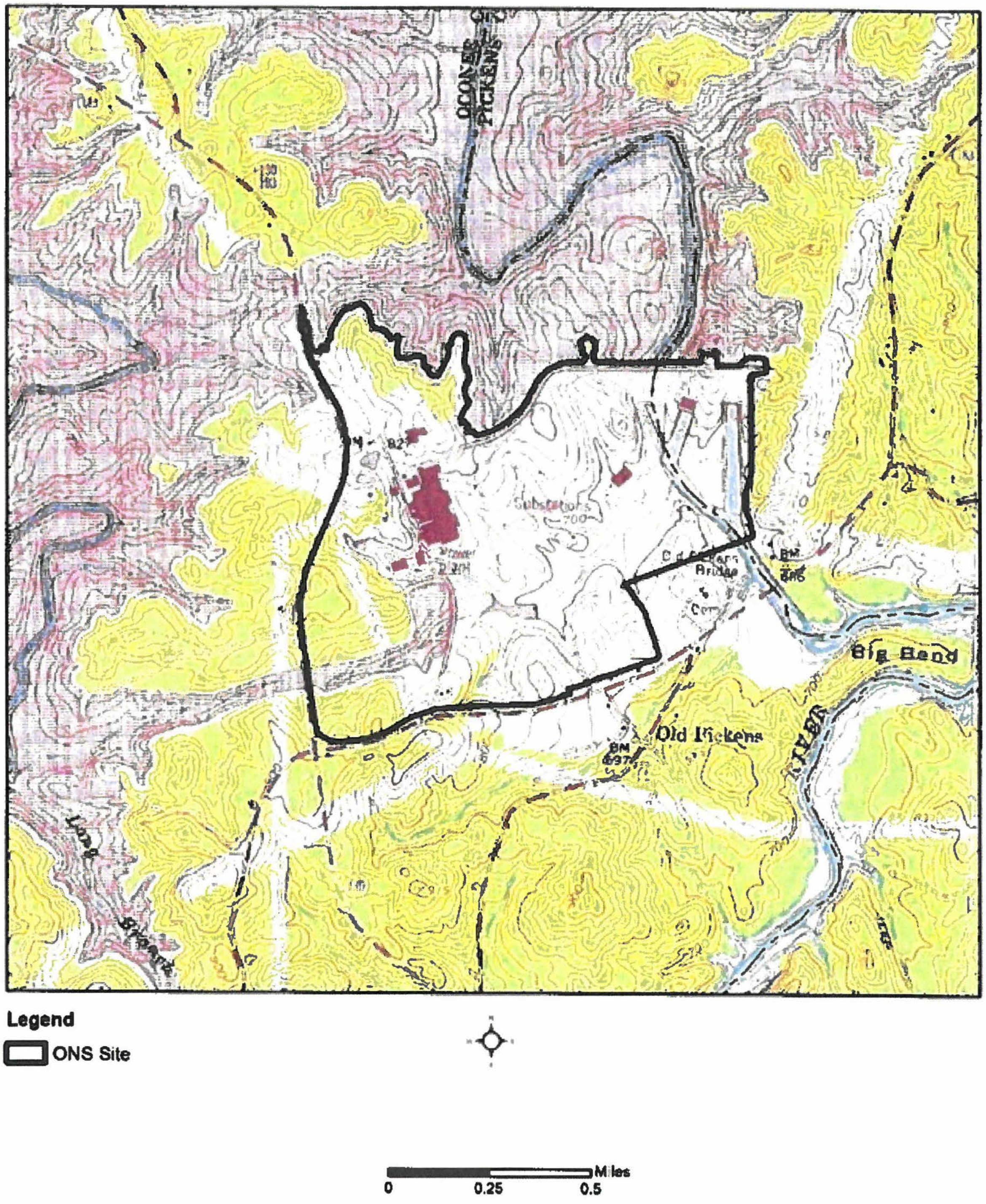
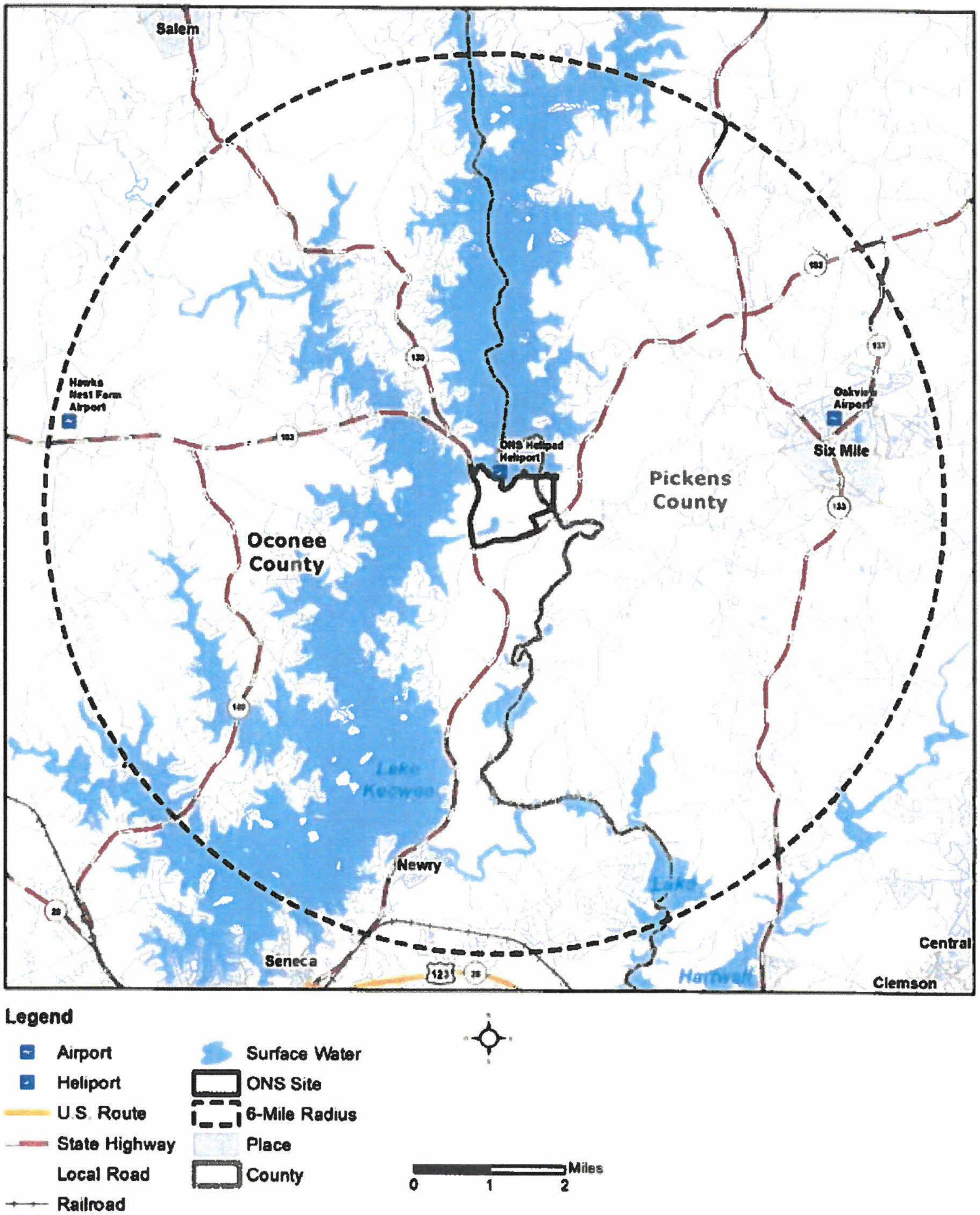


Figure 2. ONS 6-mile Vicinity



ENCLOSURE 4

OCONEE NUCLEAR STATION
NON-PROPRIETARY REFERENCE DOCUMENTS AND A REDACTED VERSION OF A PROPRIETARY REFERENCE DOCUMENT (PUBLIC VERSION)

ENCLOSURE 4
ATTACHMENT 1

OCONEE NUCLEAR STATION
Framatome Topical Report ANP 3898NP, Revision 0,
"Framatome Reactor Vessel TLAA and Aging Management Review
Input to the ONS SLRA," May 2021

Framatome Reactor Vessel and RCP TLAA and Aging Management Review Input to the ONS SLRA

ANP-3898NP
Revision 0

May 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
BTP	Branch Technical Position
CASS	Cast Austenitic Stainless Steel
CLB	Current Licensing Basis
CMTR	Certified Material Test Report
CVN	Charpy V-Notch
CVNp	Preirradiation Charpy V-Notch
CvUSE	Charpy V-Notch Upper Shelf Energy
DORT	Discrete Ordinate Transport
DPA	Displacement per Atom
EAF	Environmentally-Assisted Fatigue
EFPY	Effective Full Power Years
EMA	Equivalent Margins Analysis
EOL	End of Life
FME	Fracture Mechanics Evaluation
IE	Irradiation Embrittlement
INF	Inlet Nozzle Forging
IS	Intermediate Shell
J_d	J deformation
LEFM	Linear Elastic Fracture Mechanics
LNB	Lower Nozzle Belt
LS	Lower Shell
LTOP	Low Temperature Overpressure Protection
MCNP	Monte Carlo N-Particle
MUR	Measurement Uncertainty Recapture
ONF	Outlet Nozzle Forging
ONS	Oconee Nuclear Station
ONS Unit 1	Oconee Nuclear Station Unit 1
ONS Unit 2	Oconee Nuclear Station Unit 2
ONS Unit 3	Oconee Nuclear Station Unit 3
PTS	Pressurized Thermal Shock
P-T	Pressure-Temperature
RCP	Reactor Coolant Pump
RG	Regulatory Guide
RPV	Reactor Pressure Vessel

RVID2	Reactor Vessel Integrity Database, Version 2.0.1
RVWG	Reactor Vessel Working Group
TLAA	Time Limited Aging Analyses
TR	Topical Report
UCC	Underclad Cracking
US	Upper Shell
USE	Upper Shelf Energy

ABSTRACT

The purpose of this document is to provide supplemental information relative to the information reported in the subsequent license renewal application for Oconee Nuclear Station Units 1, 2, and 3 reactor pressure vessel (RPV) and reactor coolant pump (RCP) time limited aging analyses (TLAA) and aging management review topics that are reported in NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants. NUREG-2192, Section 4.2.2.1, Table 4.7-1, and Section 4.3.3.1.2 include the following TLAA topics that apply to the Oconee RPVs.

- Section 4.2.2.1.1—Neutron Fluence
- Section 4.2.2.1.2—Upper Shelf Energy, (10 CFR 50.60, 10 CFR 50 Appendix G)
- Section 4.2.2.1.3—Pressurized Thermal Shock (PWRs), (10 CFR 50.61)
- Section 4.2.2.1.4—Pressure-Temperature Limits (10 CFR 50.60, 10 CFR 50 Appendix G)
- Table 4.7-1—Reactor Pressure Vessel Underclad Cracking
- Section 4.3.3.1.2—Components Evaluated for CUF_{en}

Aging Management Review Topics—RPV Supports and RCPs

The NRC has developed guidance for aging management review (further evaluation) of RPV supports for irradiation embrittlement through draft interim staff guidance contained in ADAMS ML20049H359. In addition, the aging management program for evaluation of thermal embrittlement of cast austenitic steel, NUREG-2191, Volume 2, XI.M12, contains acceptable methods to manage thermal embrittlement of reactor coolant pump casings.

The supplemental information provided in this ANP report is intended to assist the NRC with review of the Oconee RPV and RCP TLAA aging management review topics listed above relative to the applicable “Review Procedures” reported in NUREG-2192. The topics addressed in this ANP report are as follows.

- Reactor Pressure Vessel Neutron Fluence Projections, Section 2.0
- Upper Shelf Energy, Section 3.0
- Equivalent Margins Analysis of Oconee Unit 3 RPV Outlet Nozzle and Transition Forgings, Section 4.0
- Pressurized Thermal Shock, Section 5.0
- P-T Limits, Section 6.0
- Fracture Mechanics Evaluation of Underclad Cracks, Section 7.0
- Reactor Vessel Environmentally-Assisted Fatigue, Section 8.0
- Irradiation Embrittlement of RPV Supports, Section 9.0
- Reactor Coolant Pumps Thermal Embrittlement (CASS), Section 10.0
- ONS Unit 2 and ONS Unit 3 RCP Flaw Tolerance Evaluation, Section 11.0

1.0 INTRODUCTION

The purpose of this document is to provide supplemental technical information relative to the information reported in the subsequent license renewal application for Oconee Nuclear Station Units 1, 2, and 3 reactor pressure vessel (RPV) time limited aging analyses (TLAA) and aging management review topics that are reported in NUREG-2192 (Reference 1-1), Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants. NUREG-2192, Section 4.2.2.1, Table 4.7-1, and Section 4.3.3.1.2 include the following TLAA topics that apply to the Oconee RPVs.

- Section 4.2.2.1.1—Neutron Fluence
- Section 4.2.2.1.2—Upper Shelf Energy, (10 CFR 50.60, 10 CFR 50 Appendix G)
- Section 4.2.2.1.3—Pressurized Thermal Shock (PWRs), (10 CFR 50.61)
- Section 4.2.2.1.4—Pressure-Temperature Limits (10 CFR 50.60, 10 CFR 50 Appendix G)
- Table 4.7-1—Reactor Pressure Vessel Underclad Cracking
- Section 4.3.3.1.2—Components Evaluated for CUF_{en}

Aging Management Review Topics—RPV Supports and RCPs

The NRC has developed guidance for aging management review (further evaluation) of RPV supports for irradiation embrittlement through draft interim staff guidance contained in ADAMS ML20049H359. In addition, the aging management program for evaluation of thermal embrittlement of cast austenitic steel, NUREG-2191, Volume 2, XI.M12, contains acceptable methods to manage thermal embrittlement of reactor coolant pump casings.

Oconee Current Licensing Basis (CLB)

The above topics were all identified as TLAA for the Oconee Unit 1, Unit 2, and Unit 3 RPVs for 60 years and were evaluated in the following Oconee CLB documents.

- Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U. S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998 (Reference 1-2).
- NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3 (ADAMS Accession No. ML003695154, Reference 1-3)
- The ONS LRA references BAW-2251A (Reference 1-4) relative to evaluation of various RPV TLAA for 60-years. All TLAA evaluations were reported at 48 EFPY and include upper shelf energy (10 CFR 50 Appendix G), pressurized thermal shock (10 CFR 50.61), underclad cracking, metal fatigue, and environmentally-assisted fatigue of the RPV.
- OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, Issuance of Amendments Regarding Revised Pressure-Temperature Limits (TAC NOS. MF0763, MF0764, AND MF0765) ADAMS Accession Number ML14041A093 (Reference 1-5).
  - NRC approval of ONS P-T limits to 54 EFPY and reconciliation of USE from 48 EFPY to 54 EFPY. Upper shelf energy was reconciled from 48 EFPY, as reported in BAW-2251A, to 54 EFPY by the NRC.
- ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate (MUR), February 19, 2020, ADAMS Accession Number ML20050D379 (Reference 1-6).
  - Reduces the Applicability for the RCS Heatup and Cooldown limit curves from 54 EFPY to 44.6 Effective Full Power Years (EFPY) for Unit 1, to 45.3 EFPY for Unit 2, and to 43.8 EFPY for Unit 3 based on updated reactor vessel (RV) material evaluations discussed in Section IV.1 of the MUR submittal.

- Maintains RT_{PTS} and Underclad Cracking at 48 EFPY and USE at 54 EFPY for all 3 units.

In accordance with NUREG-2192, Standard Review Plan for Subsequent License Renewal, aging management review recommendations for reactor pressure vessel supports are addressed in Section 3.5.2.2.2.6, and Table 3.5-1, Item 097. Following review of the first subsequent license renewal applications for Turkey Point 3 and 4, Surry 1 and 2, and Peach Bottom, the NRC determined that additional guidance was needed to clarify aging management review expectations by the NRC for Reactor Pressure Vessel (RPV) supports for subsequent license renewal. As such, the NRC has issued draft interim staff guidance (Reference 1-7) to revise the SLR Standard Review Plan NUREG-2192 to add a new Section 3.5.2.2.2.7 that includes additional guidance for evaluation of RPV supports relative to irradiation embrittlement. This topic is addressed in Section 9.0 of this report.

EFPY

Based on accrued EFPY through Cycles 31, 29, and 30 for Oconee Units 1 through 3 and assuming breaker-to-breaker operation and no outages per cycle (Capacity Factor = 1) to 80 years of operation, the bounding projected EFPY for 80 years for each Oconee Unit is less than 72 EFPY. Therefore, the Oconee Nuclear Station RPV TLAA evaluations are completed to 72 EFPY for SLR.

Measurement Uncertainty Recapture

All TLAA for subsequent license renewal reported in this ANP document consider the revised operating conditions (e.g., 1.64% increase in power) associated with MUR as reported in Reference 1-6. MUR is assumed at the beginning of Cycle 30 for ONS Unit 1, Cycle 29 for ONS Unit 2, and Cycle 29 for ONS Unit 3. At present, MUR has not been initiated and each unit is currently operating in Cycles 32, 30, and 31, respectively.

The supplemental information provided in this ANP report is intended to assist the NRC with review of the Oconee RPV and RCP topics above relative to the applicable "Review Procedures" reported in NUREG-2192. The topics addressed in this ANP report are as follows.

- Reactor Pressure Vessel Neutron Fluence Projections, Section 2.0
- Upper Shelf Energy, Section 3.0
- Equivalent Margins Analysis of Oconee Unit 3 RPV Outlet Nozzle and Transition Forgings, Section 4.0
- Pressurized Thermal Shock, Section 5.0
- P-T Limits, Section 6.0
- Fracture Mechanics Evaluation of Underclad Cracks, Section 7.0
- Reactor Vessel Environmentally-Assisted Fatigue, Section 8.0
- Irradiation Embrittlement of RPV Supports, Section 9.0
- Reactor Coolant Pumps Thermal Embrittlement (CASS), Section 10.0
- ONS Unit 2 and ONS Unit 3 RCP Flaw Tolerance Evaluation, Section 11.0

1.1 *References for Section 1.0*

1-1. NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

1-2. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U. S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998 (ADAMS Accession No. ML15254A151 and ML15112A661)

1-3. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3 (ADAMS Accession No. ML003695154)

1-4. BAW-2251A, Demonstration of Management of Aging Effects for the Reactor Vessel, ADAMS Accession Numbers ML20212G894 and ML20212G911

1-5. OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, Issuance of Amendments Regarding Revised Pressure-Temperature Limits (TAC NOS. MF0763, MF0764, and MF0765), ADAMS Accession Number ML14041A093

1-6. ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate, February 19, 2020, ADAMS Accession Number ML20050D379 and NRC SER, ADAMS Accession Number ML20335A001

1-7. RV Supports ISG SLR Document Changes: Add FE Section 3.5.2.2.2.7 and AMR Items for Irradiation Embrittlement of Reactor Vessel (RV) Steel Supports and Other Steel Structural Support Components near RV, ADAMS Accession Number ML20049H359

2.0 RPV NEUTRON FLUENCE PROJECTIONS

2.1 *Introduction*

Neutron fluence is defined as the time integral of the neutron flux density, expressed as number of particles (neutrons) per cm^2. Neutron fluence is used as an input to quantify the change in material properties of the reactor vessel (traditional beltline and extended beltline) regions, as required by 10 CFR 50, Appendix G, 10 CFR 50.61, over the life of the plant. Because these neutron embrittlement analyses are evaluated for the plant's service lifetime (10 CFR 50, Appendix G, IV., 1.) or end-of-life (10 CFR 50.61(a)(6)), they are identified as time-limited aging analyses.

2.2 *Regulatory Guidance for Subsequent License Renewal*

The regulatory guidance for NRC review of neutron fluence calculations performed in accordance with 10 CFR 54.21(c)(1)(ii) is reported in NUREG-2192, Section 4.2.3.1.1.2 (Reference 2-1).

“The reviewer confirms that the applicant adequately reevaluated its RPV neutron fluence analysis for the subsequent period of extended operation. As part of its review, the review confirms that the applicant identifies (a) the neutron fluence for each beltline material at the end of the subsequent period of extended operation, (b) the NRC staff-approved methodology used to determine the neutron fluence or submits the methodology for NRC staff review, and (c) whether the methodology is consistent with the guidance in NRC RG 1.190.”

Guidance for NRC review of acceptable methods and assumptions for determining reactor vessel neutron fluence for subsequent license renewal is provided in NUREG-2191 (Reference 2-2), X.M2, Neutron Fluence Monitoring as supplemented by the final NRC interim staff guidance for X.M2 (Reference 2-10).

“Guidance on acceptable methods and assumptions for determining reactor vessel neutron fluence is described in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” The methods developed and approved using the guidance contained in RG 1.190 are specifically intended for determining neutron fluence in the region of the RPV close to the active fuel region of the core and are not intended to apply to vessel regions significantly above and below the active fuel region of the core, nor to RVI components. Therefore, the use of RG 1.190-adherent methods to estimate neutron fluence for the RPV regions significantly above and below the active fuel region of the core and RVI components may require additional justification, even if those methods were approved by the NRC for RPV neutron fluence calculations. This program monitors in-vessel or ex-vessel dosimetry capsules and evaluates the dosimetry data, as needed. Additional in-vessel or ex-vessel dosimetry capsules may be needed when the reactor surveillance program has exhausted the available capsules for in-vessel exposure.”

2.3 *Methodology*

The calculation based fluence analysis methodology contained in BAW-2241P-A, Revision 2 (Reference 2-3) is used to predict the 72 EFPY fluence at reactor vessel shell locations (both traditional beltline and extended beltline) for Oconee Units 1, 2, and 3. BAW-2241P-A, Revision 2 is fully compliant with Regulatory Guide 1.190 (Reference 2-4) for RPV traditional beltline locations. Traditional beltline locations for the Oconee RPVs are defined as those items reported in the NRC approved topical report BAW-2251A (Reference 2-5), Tables 4-5 through 4-7, and Tables A-2 through A-4, which is referenced in the Oconee 60-year license renewal application. The RPV beltline items identified in BAW-2251A are consistent with the beltline items reported in the NRC's reactor vessel integrity database (RVID2, Reference 2-6). For SLR, extended beltline items are defined as those RPV locations, in addition to traditional beltline locations, that will receive projected neutron fluence values greater than 1.0E+17 n/cm^2 (E > 1 MeV) at the end of the subsequent period of extended operation (i.e., 72 EFPY for Oconee). A threshold of >1.0E+17 n/cm^2 (E > 1 MeV) is established through RIS 2014-11 (Reference 2-7).

For the Oconee Nuclear Station, neutron transport calculations, using the methodology from BAW-2241P-A, Revision 2, were completed for Cycles 27-29 for ONS Unit 1, Cycles 25-28 for ONS Unit 2, and Cycles 26-28 for ONS Unit 3, and used to project fast neutron fluence at the reactor vessel shell (i.e., traditional beltline and extended beltline locations) to 72 effective full power years (EFPYs). A measurement uncertainty recapture (MUR) power uprate is conservatively factored in at 2% and is assumed at the beginning of Cycle 30 for ONS Unit 1, Cycle 29 for ONS Unit 2, and Cycle 29 for ONS Unit 3.

Neutron fluence projections at 72 EFPY are provided for ONS Unit 1, ONS Unit 2, and ONS Unit 3 in Table 2-1, Table 2-2, and Table 2-3, respectively. Inside wetted surface fluence at 72 EFPY is reported for all RPV shell locations that exceed 1.0E+17 n/cm^2 (E > 1.0 MeV). Fluence is attenuated through the thickness of the shell by taking the ratio of dpa at the depth in question to the dpa at the inner surface, as permitted by Regulatory Guide 1.99, Revision 2, Section 1.1. Based on these tables, traditional beltline and extended beltline items for SLR are as follows.

ONS Unit 1 Traditional Beltline and Extended Beltline RPV Items (Figure 2-1)

- Forgings are ASTM A 508 Class 2
  - RPV outlet nozzles (2) and RPV inlet nozzles (4)**
  - Lower nozzle belt forging (NBF) (AHR 54: ZV 2861)
  - Transition forging (122S347VA1)** The transition forging is also known by Framatome as the dutchman forging. This part is identified as transition forging in this ANP report.
- Plates are ASTM A 302B, Modified
  - Intermediate shell (C2197-2)
  - Upper shell (C3265-1 and C3278-1)
  - Lower shell (C2800-1 and C2800-2)
- Linde 80 Welds
  - RPV inlet and outlet nozzle to NBF welds**
  - Lower NBF to intermediate shell circumferential weld (SA-1135)
  - Intermediate shell axial welds (SA-1073)
  - Intermediate shell to upper shell circumferential weld (SA-1229, 61% ID; WF-25, 39% OD)
  - Upper shell axial welds (SA-1493)

- Upper shell to lower shell circumferential weld SA-1585
- Lower shell axial welds (SA-1430 and SA-1426)
- Lower shell forging to transition forging circumferential weld (WF-9)

** Extended beltline items

ONS Unit 2 Traditional Beltline and Extended Beltline RPV Items (Figure 2-2)

- Forgings are all ASTM A 508 Class 2
  - RPV outlet nozzles (2) and RPV inlet nozzles (4)**
  - Lower nozzle belt forging (AMX 77: 123T382)
  - Upper shell forging (AAW 163:3P2359)
  - Lower shell forging (AWG 164: 4P1885)
  - Transition forging (122T293VA1)**
- Linde 80 Welds
  - RPV inlet and outlet nozzle to NBF welds**
  - Lower nozzle belt forging to upper shell forging circumferential weld (WF-154)
  - Upper shell forging to lower shell forging circumferential weld (WF-25)
  - Lower shell forging to transition forging circumferential weld (WF-112)

** Extended beltline items

ONS Unit 3 Traditional Beltline and Extended Beltline RPV Items (Figure 2-3)

- Forgings are all ASTM A 508 Class 2
  - RPV outlet nozzles (2) and RPV inlet nozzles (4)**
  - Lower nozzle belt forging (4680)
  - Upper shell forging (AWS 192: 522314)
  - Lower shell forging (ANK 191: 522194)

  - Transition forging (417543-1)**
- Linde 80 Welds
  - RPV inlet and outlet nozzle to USF welds**
  - Lower nozzle belt forging to upper shell forging circumferential weld (WF-200)
  - Upper shell forging to lower shell forging circumferential weld (WF-67, 75% ID; WF-70, 25% OD)
  - Lower shell forging to transition forging circumferential weld (WF 169-1)

** Extended beltline items

As the lifetime of the reactor vessel for each Oconee unit is extended to 80-years, reactor vessel regions that may be susceptible to reduction of fracture toughness may extend beyond traditional beltline locations. As such, estimation of the fluence in reactor vessel regions adjacent to those that surround the effective height of the active core (traditional beltline) is required and the accepted fluence threshold for irradiation damage of reactor vessel materials, relative to the monitoring of limiting materials, is RPV inside wetted surface fluence greater than 1.0E+17 n/cm^2 (E > 1.0 MeV) (Reference 2-7). For locations of interest beyond the reactor vessel regions that surround the effective height of the active core, the model developed using the methodology from BAW-2241P-A, Revision 2 is extended to provide estimates of the fluence rates, and calculated fluence at 72 EFPY, at the RPV nozzle and transition forging.

An adjustment of the method is employed to obtain best estimate cumulative (E > 1.0 MeV) fluence values for extended beltline locations. Specifically, a displacement per atom (dpa) adjustment is used wherein a ratio of the discrete ordinate transport (DORT) calculated dpa at the extended beltline location of interest (e.g., RPV nozzle) to the discrete ordinate transport (DORT) calculated dpa at the circumferential weld that connects the upper shell to the lower shell (SA-1585 for Unit 1, WF-25 for Unit 2, and WF-67 for Unit 3), is multiplied by the fluence at the circumferential weld to obtain the fluence at the extended beltline location of interest (e.g., RPV nozzle). The equation for dpa adjustment is as follows.

$$f_{location\ of\ interest} = f_{beltline\ \ surface} \times \frac{dpa_{location\ of\ interest}}{dpa_{beltline\ surface}}$$

Where *f* is fluence n/cm^2 (E > 1.0 MeV)

This formulation is also used to obtain the fluence at the ¼T and ¾T locations reported in Table 2-1 through Table 2-3 versus the Regulatory Guide 1.99, Revision 2 (Reference 2-8) attenuation formulation.

$$f = f_{surf}\ (e^{-0.24\,x})$$

Figure 2-4 and Figure 2-5 illustrate the fluence rate (flux) attenuation with and without using the technique above. Fast flux (E > 1 MeV) is obtained from the DORT calculated fluence as a function of thickness through the RPV (traditional beltline and extended beltline regions). **[**

]

This dpa adjustment factor method is shown to be conservative (i.e., provides a positive bias) for extended beltline nozzle belt locations relative to recent 3-D Monte Carlo N-Particle (MCNP) calculations reported in ANP-10348P, Revision 0, Figure 1-1, reproduced as Figure 2-6 below (Reference 2-9). That is, Framatome utilized a complex system of computer codes to analyze reactor regions beyond the traditional beltline that includes SOLIDWORKS3, VICTORIA 4 (a plugin for ANSYS 5), ADVANTG 6 and MCNP7; this new system is known as SVAM.

2.4 *Summary and Conclusions*

The 72 EFPY fluence analyses have been projected to the end of the subsequent period of extended operation, and these projections are used as inputs to the RPV neutron embrittlement TLAA evaluations. The methodology used to generate Oconee 72 EFPY RPV fluence is in accordance with Regulatory Guide 1.190 compliant methodology BAW-2241P-A, Revision 2. These Oconee 72 EFPY calculations meet the RG 1.190 uncertainty requirements of 20% (1 σ) or less for traditional beltline locations. For extended beltline locations, i.e., RPV outlet nozzle forgings, RPV inlet nozzle forgings, and transition forging, 72 EFPY fluence was estimated using the RG 1.190 compliant BAW-2241P-A, Revision 2 methodology that was modified by extending the model to include extended beltline locations and to implement dpa adjustment as described above. Therefore, the 72 EFPY fluence reported herein for extended beltline locations is justified for use in compliance with the Reactor Pressure Vessel Neutron Embrittlement Analyses TLAA guidance reported in NUREG-2192 for subsequent license renewal.

2.5 *References for Section 2.0*

2-1. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

2-2. NUREG-2191, Volume 2, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report

2-3. Framatome Inc. Topical Report BAW-2241P-A, Revision 2, “Fluence and Uncertainty Methodologies,” ADAMS Accession No. ML073310655 (Proprietary), ML073310660 (Non-Proprietary)

2-4. U.S. Nuclear Regulatory Commission, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” Regulatory Guide 1.190 (available @ https://www.nrc.gov/docs/ML0108/ML010890301.pdf)

2-5. Framatome Inc. Topical Report BAW-2251A, Demonstration of the Management of Aging Effects for the Reactor Vessel ADAMS, Accession Numbers ML20212G894 and ML20212G911

2-6. Reactor Vessel Integrity Database Version 2.0.1, https://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html

2-7. NRC Regulatory Issue Summary 2014-11, Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components

2-8. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials

2-9. Framatome Topical Report ANP-10348P and NP, Revision 0, “Fluence Methodologies for SLR,” ADAMS Accession No. ML20223A019

2-10. SLR-ISG-2021-02-MECHANICAL, Updated Aging Management Criteria for Mechanical Portions of Subsequent License Renewal Guidance, ADAMS Accession No. ML20181A434

Figure 2-1
ONS Unit 1 Reactor Vessel Shell

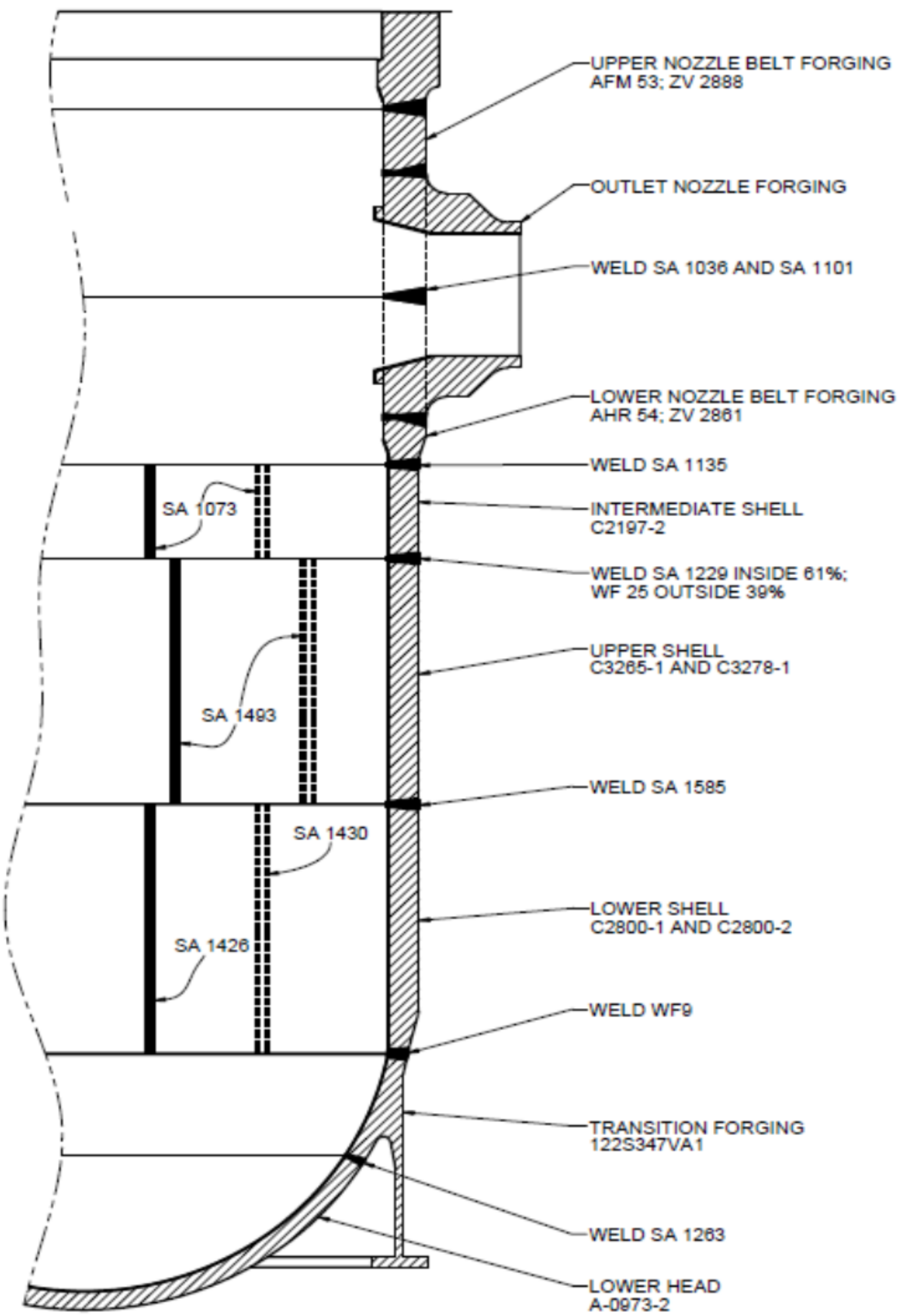


Figure 2-2
ONS Unit 2 Reactor Vessel Shell

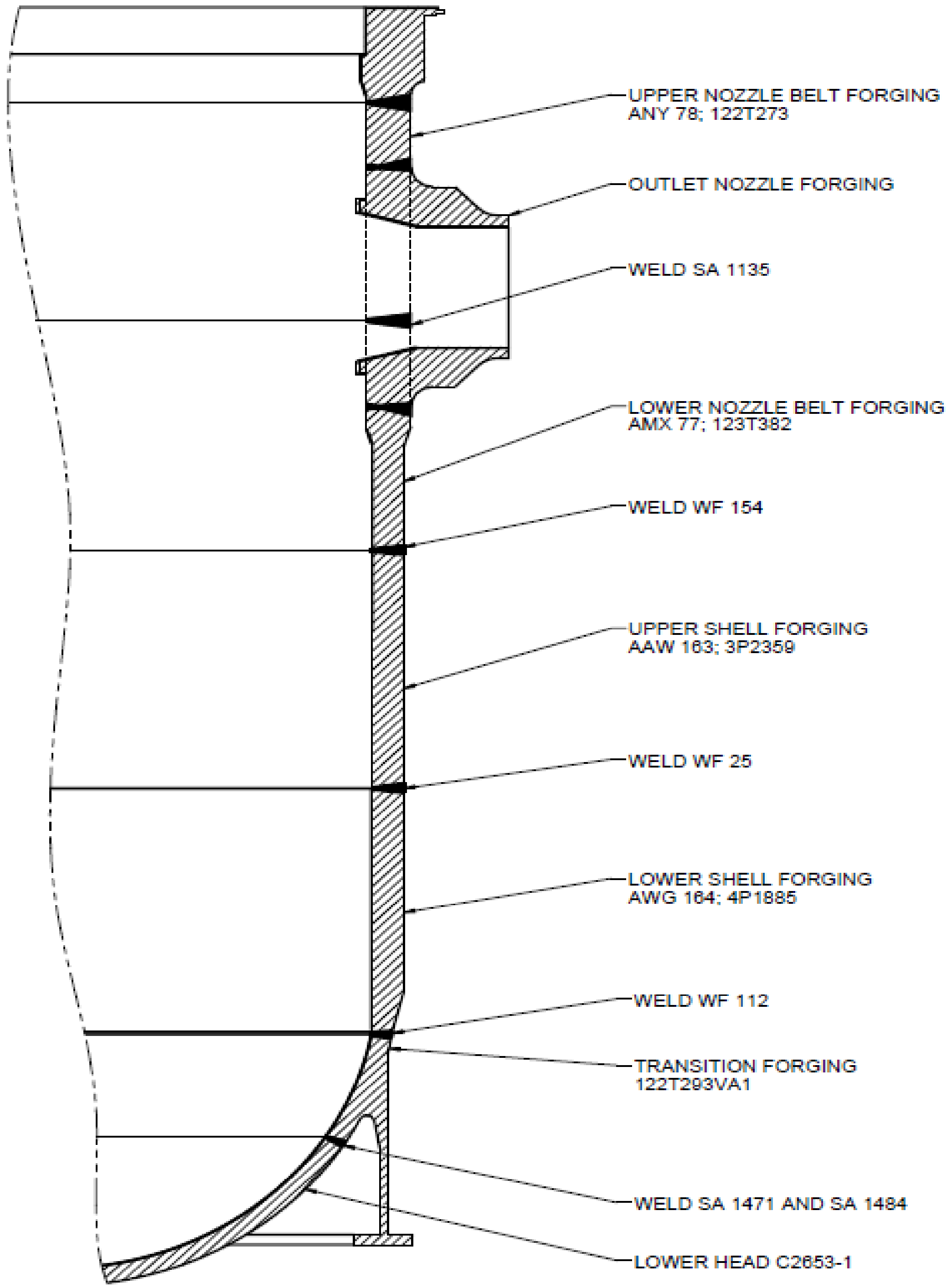


Figure 2-3
ONS Unit 3 Reactor Vessel Shell

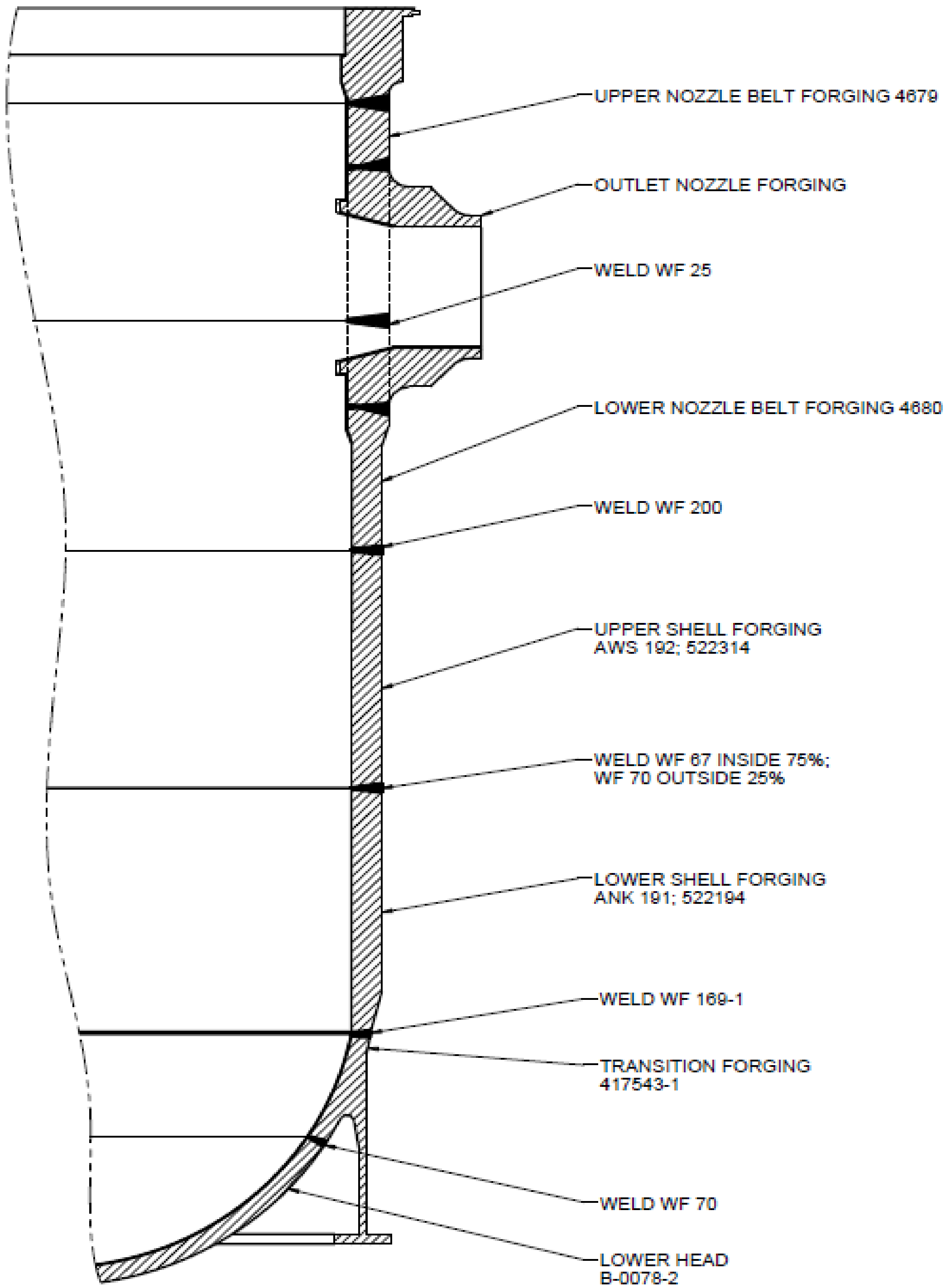


Figure 2-4
Attenuated Neutron Fluence Comparison – Core Center

[

]

Figure 2-5
Attenuated Neutron Fluence Comparison – Nozzle Region

Figure 2-6
ANP-10348P, Figure 1-1, Calculations and Measurements

[ ]

Table 2-1
72 EFPY ONS Unit 1 RPV Shell Locations with Fluence > 1.0E17 n/cm^2 (E > 1.0 MeV)

Reactor Vessel Material	Material ID and/or Heat Number	Inside Wetted Surface Fluence (n/cm^2)	Base Metal/Clad Interface Fluence (n/cm^2)	¼ T (n/cm^2)	¾ T (n/cm^2)
Upper Shell (US) Plates	C3265-1, C3278-1	2.10×10^{19}	2.02×10^{19}	1.32×10^{19}	4.35×10^{18}
Lower Shell (LS) Plates	C2800-1, C2800-2	2.10×10^{19}	2.02×10^{19}	1.31×10^{19}	4.32×10^{18}
Intermediate Shell (IS) Plates	C2197-2	1.85×10^{19}	1.78×10^{19}	1.15×10^{19}	3.72×10^{18}
Lower Nozzle Belt (LNB) Forging	AHR 54; ZV 2861	2.68×10^{18}	2.60×10^{18}	1.81×10^{18}	7.91×10^{17}
Bottom of 12” Thickness of LNB Forging	AHR 54; ZV 2861	1.48×10^{18}	1.43×10^{18}	8.82×10^{17}	4.02×10^{17}
Top of 8.438” Thickness of LNB Forging	AHR 54; ZV 2861	2.19×10^{18}	2.06×10^{18}	1.41×10^{18}	7.91×10^{17}
Bottom of 8.438” Thickness of LS	C2800-1, C2800-2	9.93×10^{17}	9.59×10^{17}	7.14×10^{17}	3.80×10^{17}
Transition Forging*	122S347VA1	2.70×10^{17}	2.52×10^{17}	2.17×10^{17}	1.96×10^{17}
Inlet Nozzle Forging (INF) Postulated Flaw	NA	1.11×10^{17}	NA	NA	NA
Outlet Nozzle Forging (ONF) Postulated Flaw	NA	2.20×10^{17}	NA	NA	NA
LNB to Bottom of ONF Welds	8T1762; 299L44; 8T1554B	3.49×10^{17}	3.38×10^{17}	2.29×10^{17}	2.52×10^{17}
LNB to Bottom of INF Welds	8T1762; 299L44; 8T1554B	1.62×10^{17}	1.57×10^{17}	1.13×10^{17}	1.73×10^{17}
LNB to IS Circ. Weld	SA-1135	2.91×10^{18}	2.82×10^{18}	1.97×10^{18}	8.80×10^{17}
IS Long. Welds (Both)	SA-1073	1.38×10^{19}	1.33×10^{19}	8.75×10^{18}	2.97×10^{18}
IS to US Circ. Weld	SA-1229; WF 25	1.86×10^{19}	1.79×10^{19}	1.16×10^{19}	3.79×10^{18}
LS to US Circ. Weld	SA-1585	2.05×10^{19}	1.97×10^{19}	1.29×10^{19}	4.30×10^{18}
US Long. Welds (Both)	SA-1493	1.36×10^{19}	1.31×10^{19}	8.58×10^{18}	2.94×10^{18}
LS Long. Weld (Both)	SA-1426, SA-1430	1.68×10^{19}	1.62×10^{19}	1.07×10^{19}	3.59×10^{18}
LS to Transition Circ. Weld	WF-9	2.70×10^{17}	2.52×10^{17}	2.17×10^{17}	1.96×10^{17}

Notes

* Values are the same as the LS to Transition Forging Circumferential Weld

Table 2-2
72 EFPY ONS Unit 2 RPV Shell Locations with Fluence > 1.0E17 n/cm^2 (E > 1.0 MeV)

Reactor Vessel Material	Material ID and/or Heat Number	Inside Wetted Surface Fluence (n/cm^2)	Base Metal/Clad Interface Fluence (n/cm^2)	¼ T (n/cm^2)	¾ T (n/cm^2)
Upper Shell (US) Forging	AAW 163; 3P2359	1.98×10^{19}	1.90×10^{19}	1.24×10^{19}	4.14×10^{18}
Lower Shell (LS) Forging	AWG 164; 4P1885	1.97×1^{019}	1.89×10^{19}	1.24×10^{19}	4.10×10^{18}
Lower Nozzle Belt (LNB) Forging	AMX 77; 123T382	1.74×10^{19}	1.67×10^{19}	1.09×10^{19}	3.52×10^{18}
Bottom of 12” Thickness of LNB Forging	AMX 77; 123T382	1.70×10^{18}	1.66×10^{18}	1.04×10^{18}	3.70×10^{17}
Top of 8.438” Thickness of LNB Forging	AMX 77; 123T382	2.42×10^{18}	2.32×10^{18}	1.62×10^{18}	7.45×10^{17}
Bottom of 8.438” Thickness of LS	AWG 164; 4P1885	9.19×10^{17}	8.87×10^{17}	6.63×10^{17}	3.58×10^{17}
Transition Forging*	122T293VA1	2.50×10^{17}	2.34×10^{17}	2.02×10^{17}	1.84×10^{17}
Inlet Nozzle Forging (INF) Postulated Flaw	NA	1.03×10^{17}	NA	NA	NA
Outlet Nozzle Forging (ONF) Postulated Flaw	NA	2.04×10^{17}	NA	NA	NA
LNB to Bottom of ONF Welds	[ ]	3.24×10^{17}	3.14×10^{17}	2.13×10^{17}	2.41×10^{17}
LNB to Bottom of INF Welds	[ ]	1.50×10^{17}	1.46×10^{17}	1.06×10^{17}	1.67×10^{17}
LNB to US Circ. Weld	WF 154	1.75×10^{19}	1.68×10^{19}	1.09×10^{19}	3.59×10^{18}
LS to US Circ. Weld	WF 25	1.92×10^{19}	1.85×10^{19}	1.22×10^{19}	4.08×10^{18}
LS to Transition Circ. Weld	WF 112	2.50×10^{17}	2.34×10^{17}	2.02×10^{17}	1.84×10^{17}

Notes
* Values are the same as the LS to Transition Forging Circumferential Weld

Table 2-3
72 EFPY ONS Unit 3 RPV Shell Locations with Fluence > 1.0E17 n/cm^2 (E > 1.0 MeV)

Reactor Vessel Material	Material ID and/or Heat Number	Inside Wetted Surface Fluence (n/cm^2)	Base Metal/Clad Interface Fluence (n/cm^2)	¼ T (n/cm^2)	¾ T (n/cm^2)
Upper Shell (US) Forging	AWS 192; 522314	2.06×10^{19}	1.98×10^{19}	1.30×10^{19}	4.30×10^{18}
Lower Shell (LS) Forging	ANK 191; 522194	2.05×10^{19}	1.97×10^{19}	1.29×10^{19}	4.26×10^{18}
Lower Nozzle Belt (LNB) Forging	4680	1.81×10^{19}	1.74×10^{19}	1.13×10^{19}	3.66×10^{18}
Bottom of 12" Thickness of LNB Forging	4680	1.81×10^{18}	1.77×10^{18}	1.11×10^{18}	3.87×10^{17}
Top of 8.438" Thickness of LNB Forging	4680	2.58×10^{18}	2.48×10^{18}	1.72×10^{18}	7.86×10^{17}
Bottom of 8.438" Thickness of LS	ANK 191; 522194	9.80×10^{17}	9.45×10^{17}	7.06×10^{17}	3.75×10^{17}
Transition Forging*	417543-1	2.68×10^{17}	2.50×10^{17}	2.15×10^{17}	1.95×10^{17}
Inlet Nozzle Forging (INF) Postulated Flaw	NA	1.14×10^{17}	NA	NA	NA
Outlet Nozzle Forging (ONF) Postulated Flaw	NA	2.21×10^{17}	NA	NA	NA
LNB to Bottom of ONF Welds	[ ]	3.50×10^{17}	3.39×10^{17}	2.30×10^{17}	2.51×10^{17}
LNB to Bottom of INF Welds	[ ]	1.62×10^{17}	1.58×10^{17}	1.14×10^{17}	1.73×10^{17}
LNB to US Circ. Weld	WF 200	1.82×10^{19}	1.75×10^{19}	1.14×10^{19}	3.74×10^{18}
LS to US Circ. Weld	WF 67; WF 70	2.01×10^{19}	1.93×10^{19}	1.27×10^{19}	4.24×10^{18}
LS to Transition Circ. Weld	WF 169-1	2.68×10^{17}	2.50×10^{17}	2.15×10^{17}	1.95×10^{17}

Notes
* Values are the same as the LS to Transition Forging Circumferential Weld

3.0 UPPER SHELF ENERGY

3.1 *Introduction*

Appendix G of 10 CFR 50 (Reference 3-1) requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb (102J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J) unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code." Since USE reduction is a function of the end of license fluence, which is associated with the 60-year licensed operating period, the USE calculations meet the criteria of 10 CFR 54.3(a) and have been identified as Time Limited Aging Analyses (TLAA) requiring evaluation for 80 years.

Current Licensing Basis

Compliance with the requirements of 10 CFR 50 Appendix G by Duke Energy for 60-years of operation is reported in Section 1.0 of this ANP report.

3.2 *Regulatory Guidance for Subsequent License Renewal*

The regulatory guidance for NRC review of upper-shelf energy evaluations performed in accordance with 10 CFR 54.21(c)(1)(ii) is reported in NUREG-2192, Section 4.2.3.1.2.2 (Reference 3-2) and is repeated below.

"The documented results of the revised USE analysis (or revised EMA analysis, as applicable) based on the projected neutron fluence at the end of the subsequent period of extended operation are reviewed for compliance with 10 CFR Part 50, Appendix G. The applicant may use NRC RG 1.99 Rev. 2 as the basis for using the ¼T neutron fluence values for the reactor vessel beltline components (as projected to the end of the SLR period) to project the USE values for the reactor vessel beltline components at the end of the subsequent period of extended operation. The applicant also may use ASME Code Section XI Appendix K for the purpose of performing an equivalent margins analysis to demonstrate that adequate protection for ductile failure is maintained to the end of the subsequent period of extended operation.

The NRC staff reviews the applicant's methodology for this evaluation. Branch Technical Position (BTP) 5-3, "Fracture Toughness Requirements," in Standard Review Plan, Section 5.3.2, "Pressure Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," provides additional NRC positions on estimations of USE values for RPV beltline materials.

The NRC staff confirms that the applicant has provided sufficient information for all USE and/or equivalent margins analysis calculations for the subsequent period of extended operation as follows:

> The applicant identifies the neutron fluence at the ¼T location for each beltline material at the expiration of the subsequent period of extended operation.

To confirm that the USE analysis meets the requirements of Appendix G of 10 CFR Part 50 at the end of the subsequent period of extended operation, the NRC staff determines whether:

1. For each beltline material, the applicant provides the unirradiated USE and the projected USE at the end of the subsequent period of extended operation, and whether the drop in USE was determined using the limit lines in Figure 2 of NRC RG 1.99, Rev 2, based on the material copper content, or from surveillance data.

2. If an equivalent margins analysis is used to demonstrate compliance with the USE requirements in Appendix G of 10 CFR Part 50, the applicant provides the analysis or identifies an NRC-approved topical report that contains the analysis which is applicable to the subsequent period of extended operation. Information the NRC staff considers to assess the equivalent margins analysis includes the unirradiated USE (if available) for the material, its copper content, the neutron fluence (¼T and at 1-inch depth), the projected SLR USE, the operating temperature in the downcomer at full power, the vessel radius, the vessel wall thickness, the J-applied analysis for Service Level C and D, the vessel accumulation pressure, and the vessel bounding heat-up/cool-down rate during normal operation."

3.3 *Methodology*

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb (102J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J) unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code." Since USE reduction is a function of the end-of-life fluence, which is associated with the 60-year licensed operating period, the USE calculations meet the criteria of 10 CFR 54.3(a) and have been identified as Time Limited Aging Analyses (TLAA) requiring evaluation for 80 years.

The 72 EFPY Charpy V-notch Upper Shelf Energy (CvUSE) projections are calculated at the ¼T thickness (¼T) for Oconee Nuclear Station Units 1, 2, and 3 base and weld metals that have peak inside wetted surface fluence >1.0E+17 n/cm^2 (E > 1 MeV) at 72 Effective Full Power Years (EFPY). Fluence at the ¼T location for traditional and extended beltline items is reported in Table 2-1, Table 2-2, and Table 2-3 for ONS Units 1, 2, and 3, respectively.

3.3.1 CvUSE at 72 EFPY Based on Regulatory Guide 1.99 Revision 2

Regulatory Guide 1.99, Revision 2 (Reference 3-3) provides two methods for determining CvUSE:

- Position 1.2 for material that does not have surveillance data available, and
- Position 2.2 for material that does have surveillance data.

For Position 1.2, the percent drop in CvUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CvUSE to obtain the adjusted CvUSE. For Position 2.2, the percent drop in CvUSE is determined by plotting the available data on Figure 2 and fitting the data with a line drawn parallel to the existing lines that bound all the plotted points. These calculations are reported in Section 3.4 below.

Material property input required for calculating 72 EFPY upper shelf energy using Positions 1.2 and 2.2, Figure 2, includes copper content (weight percent) of base metal and weld metal, and initial unirradiated upper shelf energy. Fluence is reported in Section 2.0 and material properties are addressed in Section 3.4.1 below.

3.3.2 Equivalent Margins Analyses

The ASME Code, Section XI, Appendix K acceptance criteria for Levels A through D Service Loadings for Oconee Units 1, 2, and 3 reactor vessel traditional beltline and extended beltline Linde 80 welds are satisfied and are reported in Framatome reports BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A and B Service Loadings (Reference 3-4), and BAW-2178, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C and D Service Loadings (Reference 3-5). These reports contain equivalent margins analyses for all Oconee Unit 1, 2, and 3 Linde 80 welds identified as traditional beltline and extended beltline material for 80 years. Confirmation that the Oconee Nuclear Station reactor pressure vessels are bounded by these topical reports is provided in Section 3.5 below.

The ASME Code, Section XI, Appendix K acceptance criteria for Levels A through D Service Loadings for Oconee Unit 3 reactor vessel outlet nozzle forgings and the transition forging are satisfied and are reported in Section 4.0 of this ANP report.

3.4 *Traditional Beltline and Extended Beltline 72 EFPY Upper Shelf Energy*

The 72 EFPY Regulatory Guide 1.99, Revision 2 (Position 1.2) USE values of the vessel materials are predicted using the corresponding ¼T fluence projection, copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2. The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using surveillance data along with the corresponding ¼T fluence projection. The ¼T fluence projections are ¼T dpa adjusted values reported in Table 2-1 for ONS Unit 1, Table 2-2 for ONS Unit 2, and Table 2-3 for ONS Unit 3 unless the attenuation of fluence from the inside wetted surface using RG 1.99, Revision 2, Equation (3), exceed the ¼T dpa adjusted values. The projected USE values were calculated to determine if the ONS Unit 1, Unit 2, and Unit 3 traditional beltline and extended beltline materials remain above the 50 ft-lb limit at 72 EFPY and are reported for Oconee Units 1, 2, and 3 in Table 3-6, Table 3-7, and Table 3-8, respectively.

All ONS Unit 1 reactor vessel traditional beltline and extended beltline plate and forging materials maintain a USE value greater than 50 ft-lbs at 72 EFPY. All ONS Unit 1 reactor vessel traditional beltline and extended beltline Linde 80 welds have USE values less than 50 ft-lbs at 72 EFPY and require an equivalent margins analysis to demonstrate that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

All ONS Unit 2 reactor vessel traditional beltline and extended beltline forging materials maintain a USE value greater than 50 ft-lbs at 72 EFPY. All ONS Unit 2 reactor vessel traditional beltline and extended beltline Linde 80 welds have USE values less than 50 ft-lbs at 72 EFPY and require an equivalent margins analysis to demonstrate that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

All ONS Unit 3 reactor vessel traditional beltline and extended beltline forging materials maintain a USE value greater than 50 ft-lbs at 72 EFPY with the exception of the RPV outlet nozzle forgings and the transition forging. All reactor vessel traditional beltline and extended beltline Linde 80 welds have USE values less than 50 ft-lbs at 72 EFPY. The RPV outlet nozzle forgings, transition forging, and traditional beltline and extended beltline Linde 80 welds require an equivalent margins analysis to demonstrate that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

3.4.1 Material Properties Including Copper and Initial Upper-Shelf Energy

Material properties required for the calculation of 72 EFPY USE are reported in Table 3-6, Table 3-7, and Table 3-8 and include copper content (weight %) and initial unirradiated upper-shelf energy (ft-lbs). For traditional beltline locations, the copper content and initial upper shelf energy are consistent with the current licensing basis (Section 1.0, Reference 1-2 and Reference 1-5) with some exceptions as reported in Section 3.4.2. For extended beltline locations (i.e., RPV inlet and outlet nozzles and associated welds and the transition forging), material properties have not been reported to the NRC and initial copper content and initial upper shelf energy for SLR are obtained as follows.

<u>Initial Upper Shelf Energies</u>

A generic upper shelf energy for Linde 80 welds is reported herein at **[ ]** and is obtained as follows. The upper-shelf energy values for Linde 80 material in the Reactor Vessel Working Group (RVWG) data base **[ ]** were compiled and are reported in Table 3-1. The mean was found to be **[ ]**, the standard deviation is **[ ]** and the one-sided tolerance factor (k), 95% confidence for 95% coverage, is **[ ]**. The generic value is calculated to be **[ ]** (mean less k*sigma).

Table 3-1
Linde 80 Initial CvUSE Values

[ ]

Generic initial upper shelf energy is reported for the ONS Unit 1 and ONS Unit 2 RPV inlet and outlet nozzle forgings and transition forgings and the ONS Unit 3 RPV inlet nozzle forgings procured from Bethlehem Steel at [ ] and is obtained from the data reported in Table 3-2 below. The upper-shelf energy values in the weak direction for all such material in the RVWG data base [ ] are reported in Table 3-2 . The mean was found to be [ ], the standard deviation is [ ] and the one-sided tolerance factor (k), 95% confidence for 95% coverage, is [ ]. The generic value is calculated to be [ ] (mean less k*sigma). The ONS Unit 3 RPV outlet nozzle forgings and transition forging were procured from Klockner-Werke and there are no initial upper shelf energy data available for these forgings from the Certified Material Test Reports (CMTRs).

Table 3-2
Upper-Shelf Energy Data for Extended Beltline Forging Materials Used to Determine Generic Upper-Shelf Energy

[ ]

<u>Copper Weight Percent—Extended Beltline Locations</u>

The ONS Unit 1 and ONS Unit 2 RPV inlet and outlet nozzle forgings and transition forgings and the ONS Unit 3 RPV inlet nozzle forgings were procured from Bethlehem Steel. The copper wt% for all such material in the RVWG data base [ ] are reported in Table 3-5. The mean was found to be [ ], the standard deviation is [ ] and the one-sided tolerance factor (k), 95% confidence for 95% coverage, is [ ]. The generic value is calculated to be [ ] (mean plus k*sigma). Data used to develop the generic mean are reported in (Table 3-5 below).

The copper weight percent of the RPV inlet and outlet nozzle forging to nozzle belt forging Linde 80 welds for all 3 Oconee Units are reported in Table 3-6 through Table 3-8 and in each case are the mean of measured values of the actual weld wire heats listed. Use of the mean of measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld is consistent with Regulatory Guide 1.99, Revision 2, Section 1.1 (Reference 3-3).

3.4.2 Comparison to Current Licensing Basis and NRC RVID2-Traditional Beltline

For traditional beltline materials, differences between the current licensing basis (see Section 1.0) and the Cu wt% content and initial upper-shelf energy values utilized for subsequent license renewal are identified in Table 3-3 and Table 3-4, respectively. For items not listed, the CLB and SLR Cu wt% and CvUSE values are identical. In addition, Cu wt% and CvUSE values reported in the NRC Reactor Vessel Integrity Database (RVID2-Reference 2-6) database are included for information. The data and information provided by licensees in their responses to the staff's Generic Letter (GL) 92-01, Revision 1 close-out letters, and in response to GL 92-01, Revision 1, Supplement 1, are included in RVID2. This database includes updates from June 1999 - July 2000. Duke Energy’s responses to GL92-01, GL 92-01, Revision 1, and GL 92-01, Revision 1, Supplement 1 are based on the following generic topical reports in chronological order.

- BAW-2166, Response to GL 92-01, June 1992, available through ADAMS Accession Number ML20101J787
- BAW-2222, Response to Closure Letters to GL 92-01, June 1994, ADAMS Forwards rept BAW-2222, "Reactor Vessel Working Group Response to Closure Ltrs to NRC GL 92-01, Rev 1,” ADAMS Legacy Accession 9407060101 (microfiche).
- BAW-2222, Revision 1, Response to Closure Letters to GL 92-01, June 1994, ADAMS Accession Number ML20070C083

- BAW-2257, July 1995, Response to Part (1) of GL 92-01, Revision 1, Supplement 1, ADAMS Accession Number ML17264A118
- BAW-2257, Revision 1, October 1995, B&WOG Reactor Vessel Working Group Response to GL 92-01, Rev 1, Suppl 1, ADAMS Accession Number ML20094D978
- BAW-2325, May 1998, Reactor Vessel Working Group Response to RAI Regarding Reactor Pressure Vessel Integrity, ADAMS Accession No. ML20248F744
- BAW-2325, Revision 1, January 1999, Response to RAI Regarding Reactor Pressure Vessel Integrity, ADAMS Accession No. ML14162A108

Based on a comparison of the NRC RVID2 data and updated SLR data for traditional beltline locations, additional explanatory information is provided for the following items due to the potential to increase margins to embrittlement limits.

- ONS Unit 2 LNB Forging AMX-77. Cu wt% decrease from 0.13% to [ ]

  A generic Cu wt % of [ ] is selected for forging AMX-77, which is derived in Section 3.4.1 above for Bethlehem forgings. Based on a review of the ONS Unit 2 RPV Quality Assurance Data Package, ONS Unit 2 LNB forging AMX-77 was forged by the Ladish Co. As reported in BAW-2222, Section 4.1, for Ladish forgings for which copper content was not determined, a mean was found to be 0.05 wt %, the standard deviation is 0.03 and the one-sided factor (k) is 2.453. The generic value is calculated to be 0.13 wt % (mean plus kσ).

  [

  ]

- ONS Unit 3 LNB 4680. Initial USE increase from 109 ft-lbs to 113 ft-lbs.

  With recent (2014) measured values for heat 4680 (100% shear) from a nozzle dropout at 111, 108, 114, and 119 ft-lbs, the mean is 113 ft-lbs. The CLB initial USE for heat 4680, originally at 109 ft-lbs, was based on BAW-2222. Recent testing of heat 4680 permitted the increase from 109 ft-lbs to 113 ft-lbs.

Table 3-3
Traditional beltline Cu Content Differences between CLB and SLR

[

]

Table 3-4
Traditional Beltline CvUSE Differences between CLB and SLR

Table 3-5
Copper Content Data for Pre-1971 ASTM A508 Class 2 Reactor Vessel Forgings: Bethlehem Steel Ingots

Table 3-6
Evaluation of 72 EFPY Charpy V-Notch Upper-Shelf Energy – Oconee Unit 1

Material Description								
RV Material	Mat. ID	Heat #	Type	Cu, wt%	Initial CvUSE (ft-lb)	72 EFPY, Fluence ¼T Location (n/cm^2)	ΔCVN Decrease (%)	72 EFPY Predicted CvUSE (ft-lb)
Regulatory Guide 1.99, Revision 2, Position 1.2								
Lower Nozzle Belt (LNB) Forging	AHR 54	ZV-2861	A-508, Cl. 2	[ ]	[ ]	1.81E+18	[ ]	90.3
Intermediate Shell (IS) Plate	C2197-2	C2197-2	SA-302 Gr. B Mod.	0.15	80	1.15E+19	24.8	60.2
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302 Gr. B Mod.	[ ]	[ ]	1.32E+19	[ ]	86.1
US Plate	C3278-1	C3278-1	SA-302 Gr. B Mod.	0.12	80	1.32E+19	22.4	62.1
Lower Shell (LS) Plate	C2800-1	C2800-1	SA-302 Gr. B Mod.	0.11	80	1.31E+19	21.3	62.9
LS Plate	C2800-2	C2800-2	SA-302 Gr. B Mod.	0.11	119	1.31E+19	21.3	93.6
Outlet Nozzle Forging (ONF) 1	NA	122S316VA2	A-508, Cl. 2	[ ]	[ ]	2.29E+17	[ ]	96.0
ONF 2	NA	122S316VA1	A-508, Cl. 2	[ ]	[ ]	2.29E+17	[ ]	96.0
Inlet Nozzle Forging (INF) 1	NA	123S346VA1	A-508, Cl. 2	[ ]	[ ]	1.13E+17	[ ]	96.0
INF 2	NA	123S346VA2	A-508, Cl. 2	[ ]	[ ]	1.13E+17	[ ]	96.0
INF 3	NA	124S502VA1	A-508, Cl. 2	[ ]	[ ]	1.13E+17	[ ]	96.0
INF 4	NA	124S502VA2	A-508, Cl. 2	[ ]	[ ]	1.13E+17	[ ]	96.0
Transition Forging	NA	122S347VA1	A-508, Cl. 2	[ ]	[ ]	2.17E+17	[ ]	96.0

Material Description				Cu, wt%	Initial CvUSE (ft-lb)	72 EFPY, Fluence ¼T Location (n/cm^2)	ΔCVN Decrease (%)	72 EFPY Predicted CvUSE (ft-lb)
RV Material	Mat. ID	Heat #	Type					
LNB to ONF Welds	NA	8T1762	Linde 80	[ ]	[ ]	2.29E+17	[ ]	45.0
	NA	299L44	Linde 80	[ ]	[ ]	2.29E+17	[ ]	40.1
	NA	8T1554B	Linde 80	[ ]	[ ]	2.29E+17	[ ]	45.9
LNB to INF Welds	NA	8T1762	Linde 80	[ ]	[ ]	1.13E+17	[ ]	45.0
	NA	299L44	Linde 80	[ ]	[ ]	1.13E+17	[ ]	40.1
	NA	8T1554B	Linde 80	[ ]	[ ]	1.13E+17	[ ]	45.9
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	[ ]	[ ]	1.97E+18	[ ]	41.6
IS Long. Welds (Both 100%)	SA-1073	1P0962	Linde 80	[ ]	[ ]	8.75E+18	[ ]	36.7
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	[ ]	[ ]	1.16E+19	[ ]	34.3
IS to US Circ. Weld (OD 39%)	WF-25	299L44	Linde 80	[ ]	[ ]	3.95E+18	[ ]	35.1
US Long. Welds (Both 100%)	SA-1493	8T1762	Linde 80	[ ]	[ ]	8.58E+18	[ ]	37.9
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	[ ]	[ ]	1.29E+19	[ ]	34.3
LS Long. Weld (100%)	SA-1426	8T1762	Linde 80	[ ]	[ ]	1.07E+19	[ ]	37.0
LS Long. Weld (100%)	SA-1430	8T1762	Linde 80	[ ]	[ ]	1.07E+19	[ ]	37.0
LS to Transition Forging Circ. Weld (100%)	WF-9	72445	Linde 80	[ ]	[ ]	2.17E+17	[ ]	44.0
Regulatory Guide 1.99, Revision 2, Position 2.2								
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302 Gr. B Mod.	[ ]	[ ]	1.32E+19	[ ]	86.1
LNB to ONF Weld	NA	299L44	Linde 80	[ ]	[ ]	2.29E+17	[ ]	39.4
LNB to INF Welds	NA	299L44	Linde 80	[ ]	[ ]	1.13E+17	[ ]	39.4
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	[ ]	[ ]	1.97E+18	[ ]	41.9
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	[ ]	[ ]	1.16E+19	[ ]	35.8

Material Description								
RV Material	**Mat. ID**	**Heat #**	**Type**	**Cu, wt%**	**Initial CvUSE (ft-lb)**	**72 EFPY, Fluence ¼T Location (n/cm^2)**	**ΔCVN Decrease (%)**	**72 EFPY Predicted CvUSE (ft-lb)**
IS to US Circ. Weld (OD 39%)	WF-25	299L44	Linde 80	[ ]	[ ]	3.95E+18	[ ]	33.1
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	[ ]	[ ]	1.29E+19	[ ]	33.0
LS to Transition Forging Circ. Weld (100%)	WF-9	72445	Linde 80	[ ]	[ ]	2.17E+17	[ ]	43.3

Table 3-7
Evaluation of 72 EFPY Charpy V-Notch Upper-Shelf Energy – Oconee Unit 2

Material Description								
RV Material	**Mat. ID**	**Heat #**	**Type**	**Cu, wt%**	**Initial CvUSE (ft-lb)**	**Fluence ¼T Location (n/cm^2)**	**ΔCVN Decrease (%)**	**72 EFPY Predicted CvUSE (ft-lb)**
Regulatory Guide 1.99, Revision 2, Position 1.2								
LNB Forging	AMX 77	123T382	A-508, Cl. 2	[ ]	[ ]	1.09E+19	[ ]	86.8
US Forging	AAW 163	3P2359	A-508, Cl. 2	0.04	133	1.24E+19	20.0	106.4
LS Forging	AWG 164	4P1885	A-508, Cl. 2	0.02	138	1.24E+19	20.0	110.4
ONF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.13E+17	[ ]	96.0
ONF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.13E+17	[ ]	96.0
INF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.06E+17	[ ]	96.0
INF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.06E+17	[ ]	96.0
INF 3	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.06E+17	[ ]	96.0
INF 4	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.06E+17	[ ]	96.0
Transition Forging	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.02E+17	[ ]	96.0
LNB to ONF Welds	NA	8T1762	Linde 80	[ ]	[ ]	2.13E+17	[ ]	45.0
	NA	72445	Linde 80	[ ]	[ ]	2.13E+17	[ ]	44.0
LNB to INF Welds	NA	8T1762	Linde 80	[ ]	[ ]	1.06E+17	[ ]	45.0
	NA	72445	Linde 80	[ ]	[ ]	1.06E+17	[ ]	44.0
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	[ ]	[ ]	1.09E+19	[ ]	32.3
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	[ ]	[ ]	1.22E+19	[ ]	31.3
LS to Transition Forging Circ. Weld (100%)	WF-112	406L44	Linde 80	[ ]	[ ]	2.02E+17	[ ]	42.4
Regulatory Guide 1.99, Revision 2, Position 2.2								
US Forging	AAW 163	3P2359	A-508, Cl. 2	0.04	133	1.24E+19	23.8	101.4
LNB to ONF Weld	NA	72445	Linde 80	[ ]	[ ]	2.13E+17	[ ]	43.3

Material Description				Cu, wt%	Initial CvUSE (ft-lb)	Fluence ¼T Location (n/cm^2)	ΔCVN Decrease (%)	72 EFPY Predicted CvUSE (ft-lb)
RV Material	**Mat. ID**	**Heat #**	**Type**					
LNB to INF Weld	NA	72445	Linde 80	[ ]	[ ]	1.06E+17	[ ]	43.3
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	[ ]	[ ]	1.09E+19	[ ]	32.6
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	[ ]	[ ]	1.22E+19	[ ]	26.3
LS to Transition Forging Circ. Weld (100%)	WF-112	406L44	Linde 80	[ ]	[ ]	2.02E+17	[ ]	42.5

Table 3-8
Evaluation of 72 EFPY Charpy V-Notch Upper-Shelf Energy – Oconee Unit 3

Material Description								
RV Material	**Mat. ID**	**Heat #**	**Type**	**Cu, wt%**	**Initial CvUSE (ft-lb)**	**Fluence ¼T Location (n/cm^2)**	**ΔCVN Decrease (%)**	**72 EFPY Predicted CvUSE (ft-lb)**
Regulatory Guide 1.99, Revision 2, Position 1.2								
LNB Forging	4680	4680	A-508, Cl. 2	0.15	113	1.13E+19	24.7	85.1
US Forging	AWS 192	522314	A-508, Cl. 2	0.01	112	1.30E+19	20.2	89.4
LS Forging	ANK 191	522194	A-508, Cl. 2	0.02	144	1.29E+19	20.2	114.9
ONF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.30E+17	[ ]	---
ONF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.30E+17	[ ]	---
INF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.14E+17	[ ]	96.0
INF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.14E+17	[ ]	96.0
INF 3	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.14E+17	[ ]	96.0
INF 4	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	1.14E+17	[ ]	96.0
Transition Forging	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	2.15E+17	[ ]	48.8
LNB to ONF Welds	NA	72105	Linde 80	[ ]	[ ]	2.30E+17	[ ]	40.8
	NA	406L44	Linde 80	[ ]	[ ]	2.30E+17	[ ]	42.4
LNB to INF Welds	NA	72105	Linde 80	[ ]	[ ]	1.14E+17	[ ]	40.8
	NA	72102	Linde 80	[ ]	[ ]	1.14E+17	[ ]	44.3
	NA	82102	Linde 80	[ ]	[ ]	1.14E+17	[ ]	39.8
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	[ ]	[ ]	1.14E+19	[ ]	33.8
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	[ ]	[ ]	1.27E+19	[ ]	32.1
US to LS Circ. Weld (OD 25%)	WF-70	72105	Linde 80	[ ]	[ ]	4.27E+18	[ ]	34.9
LS to Transition Forging Circ. Weld (100%)	WF-169-1	8T1554	Linde 80	[ ]	[ ]	2.15E+17	[ ]	45.9

Material Description								
RV Material	**Mat. ID**	**Heat #**	**Type**	**Cu, wt%**	**Initial CvUSE (ft-lb)**	**Fluence ¼T Location (n/cm^2)**	**ΔCVN Decrease (%)**	**72 EFPY Predicted CvUSE (ft-lb)**
Regulatory Guide 1.99, Revision 2, Position 2.2								
US Forging	AWS 192	522314	A-508, Cl. 2	0.01	112	1.30E+19	20.7	88.8
LNB to ONF Welds	NA	[ ]	Linde 80	[ ]	[ ]	2.30E+17	[ ]	42.1
	NA	[ ]	Linde 80	[ ]	[ ]	2.30E+17	[ ]	42.5
LNB to INF Weld	NA	[ ]	Linde 80	[ ]	[ ]	1.14E+17	[ ]	42.1
LS Forging	ANK 191	522194	A-508, Cl. 2	0.02	144	1.29E+19	26.8	105.5
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	[ ]	[ ]	1.14E+19	[ ]	33.4
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	[ ]	[ ]	1.27E+19	[ ]	37.1
US to LS Circ. Weld (OD 25%)	WF-70	72105	Linde 80	[ ]	[ ]	4.27E+18	[ ]	36.6

3.5 *Confirmation that Oconee is Bounded by NRC Approved Generic Technical Reports*

The ASME Code, Section XI, acceptance criteria for Levels A through D Service Loadings for Oconee Units 1, 2, and 3 reactor vessel traditional beltline and extended beltline Linde 80 welds are satisfied and are reported in Framatome reports BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A and B Service Loadings (Reference 3-4), and BAW-2178, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C and D Service Loadings (Reference 3-5). These reports contain equivalent margins analyses for all Oconee Unit 1, 2, and 3 Linde 80 welds identified as traditional beltline and extended beltline material for 80 years.

As illustrated in Table 3-9, the 80-year inside wetted surface fluence values reported in Table 3-1 of BAW-2178, Revision 0, Supplement 1P-A, Revision 0 and Table 3-1 of BAW-2192, Revision 0, Supplement 1P-A, Revision 0 bound the 72 EFPY fluence values reported in Table 2-1, Table 2-2, and Table 2-3 of this ANP report with the exception of ONS Unit 1 weld SA-1135, circumferential weld that connects the lower nozzle belt forging to the intermediate shell. Note that the EMAs reported in the topical reports (TRs) conservatively utilized 80-year fluence values of at least an order of magnitude higher than the 72 EFPY nozzle fluence reported herein.

In addition, the weld chemistry (Copper wt %) data reported in Table 3-1 of BAW-2178, Revision 0, Supplement 1P-A, Revision 0 and Table 3-1 of BAW-2192, Revision 0, Supplement 1P-A, Revision 0 is consistent with weld chemistry copper content reported in Table 3-6, Table 3-7, and Table 3-8. The Levels C and D limiting design transients reported in Section 4.3.1 of BAW-2178, Revision 0, Supplement 1P-A, Revision 0 are applicable to Oconee Units and are based on a review of the Oconee Reactor Vessel Design Specification transients and the UFSAR Chapter 15 events relative to transients that would result in the highest thermal stresses coupled with pressure stresses relative to the EMA analysis; this satisfies Regulatory Guide 1.161 with respect to Levels C and D transient selection. The materials of construction, RPV geometry, and range of explanatory variables for the J-R model (Section A.5 of BAW-2192, Revision 0, Supplement 1P-A, Revision 0) reported in the topical reports are confirmed to be applicable to Linde 80 traditional beltline and extended beltline welds at Oconee Units 1 through 3.

As such, Framatome has confirmed that Oconee Units 1, 2, and 3 are bounded by the topical report submittals BAW-2178, Revision 0, Supplement 1P-A, Revision 0, and BAW-2192, Revision 0, Supplement 1P-A, Revision 0 relative to fluence (with the exception of ONS Unit 1 weld SA-1135), weld chemistry, geometry, materials of construction, design transients, and the J-R model applicability. The results of the Oconee Units 1, 2, and 3 EMA are reported in BAW-2178, Revision 0, Supplement 1P-A, Revision 0 and BAW-2192, Revision 0, Supplement 1P-A, Revision 0 and are not repeated herein.

The equivalent margins analyses for the ONS Unit 3 RPV outlet nozzle forgings and transition forging are not included in the above NRC approved topical reports but are reported in Section 4.0 of this ANP report.

3.6 *ONS Unit 1 Weld SA-1135 EMA Reconciliation*

Oconee Unit 1 weld SA-1135 was evaluated in BAW-2192, Revision 0, Supplement 1P-A, Revision 0, at a fluence of 1.90E+18 n/cm^2 yet the fluence for weld SA-1135 is reported in Table 2-1 as 2.91E+18 n/cm^2. Therefore, an EMA reconciliation was performed for this weld at a fluence of 2.91E+18 n/cm^2. The ratio $J_{0.1}/J_1$ for weld SA-1135 decreased from [ ] to [ ], is greater than the required value of 1.0, and flaw extensions remain ductile and stable. The limiting weld remains Oconee Unit 1 axial weld SA-1073 as reported in BAW-2192, Revision 0, Supplement 1P-A, Revision 0.

3.7 *References for Section 3.0*

3-1. 10 CFR Part 50.60, Appendix G, Code of Federal Regulations, Title 10, Part 50 – Domestic Licensing of Production and Utilization Facilities, Appendix G – Fracture Toughness Requirements, Federal Register Vol. 84, Page 65644, November 29, 2019.

3-2. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

3-3. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials

3-4. BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads. ADAMS Accession Number ML19101A355.

3-5. BAW-2178, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads. ADAMS Accession Number ML19101A355.

Table 3-9
BAW-2178 and BAW-2192 Reactor Vessel Weld Locations--Copper Content and 80-Year Fluence Projections

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or n/cm^2 E> 1.0 MeV)	Cu, wt%	(IS) Inside Wetted Surface Fluence or n/cm^2 E> 1.0 MeV) Tables 2-1 to 2-3
Oconee Unit 1, 80-YearFluence (E > 1.0 MeV)						
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	1.50E+18	0.19	3.49×10^{17}
	Wire Ht. 299L44	0.34	0.68	1.50E+18	0.34	3.49×10^{17}
	Wire Ht. 8T1554B	0.16	0.57	1.50E+18	0.16	3.49×10^{17}
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	1.50E+18	0.19	1.62×10^{17}
	Wire Ht. 299L44	0.34	0.68	1.50E+18	0.34	1.62×10^{17}
	Wire Ht. 8T1554B	0.16	0.57	1.50E+18	0.16	1.62×10^{17}
LNB to Intermediate Shell (IS) Circ. Weld	SA-1135 (Wire Ht. 61782)	0.23	0.52	1.90E+18	0.23	2.91×10^{18}
IS Long. Welds (Both)	SA-1073 (Wire Ht. 1P0962)	0.21	0.64	1.58E+19	0.21	1.38×10^{19}
IS to Upper Shell (US) Circ. Weld (ID 61%)	SA-1229 (Wire Ht. 71249)	0.23	0.59	2.02E+19	0.23	1.86×10^{19}
US Long. Welds (Both)	SA-1493 (Wire Ht. 8T1762)	0.19	0.57	2.05E+19	0.19	1.36×10^{19}
US to Lower Shell (LS) Circ. Weld	SA-1585 (Wire Ht. 72445)	0.22	0.54	2.14E+19	0.22	2.05×10^{19}
LS Long. Weld (1)	SA-1426 (Wire Ht. 8T1762)	0.19	0.57	1.82E+19	0.19	1.68×10^{19}
LS Long. Weld (2)	SA-1430 (Wire Ht. 8T1762)	0.19	0.57	1.82E+19	0.19	1.68×10^{19}
LS to Transition Forging Circ. Weld	WF-9 (Wire Ht. 72445)	0.22	0.54	4.88E+17	0.22	2.70×10^{17}

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or n/cm^2 E> 1.0 MeV)	Cu, wt%	(IS) Inside Wetted Surface Fluence or n/cm^2 E> 1.0 MeV) Tables 2-1 to 2-3
Oconee Unit 2, 80-Year Fluence (E > 1.0 MeV)						
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	1.50E+18	0.19	3.24×10^{17}
	Wire Ht. 72445	0.22	0.54	1.50E+18	0.22	3.24×10^{17}
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	1.50E+18	0.19	1.50×10^{17}
	Wire Ht. 72445	0.22	0.54	1.50E+18	0.22	1.50×10^{17}
LNB to Upper Shell (US) Circ. Weld	WF-154 (Wire Ht. 406L44)	0.27	0.59	1.99E+19	0.27	1.75×10^{19}
US to Lower Shell (LS) Circ. Weld	WF-25 (Wire Ht. 299L44)	0.34	0.68	2.18E+19	0.34	1.92×10^{19}
LS to Transition Forging Circ. Weld	WF-112 (Wire Ht. 406L44)	0.27	0.59	5.20E+17	0.27	2.50×10^{17}
Oconee Unit 3, 80-Year Fluence (E > 1.0 MeV)						
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 72105	0.32	0.58	1.50E+18	0.32	3.50×10^{17}
	Wire Ht. 406L44	0.27	0.59	1.50E+18	0.27	3.50×10^{17}
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 72105	0.32	0.58	1.50E+18	0.32	1.62×10^{17}
	Wire Ht. 72102	0.21	0.58	1.50E+18	0.21	1.62×10^{17}
	Wire Ht. 82102	0.35	1.00	1.50E+18	0.35	1.62×10^{17}
LNB to Upper Shell (US) Circ. Weld	WF-200 (Wire Ht. 821T44)	0.24	0.63	1.92E+19	0.24	1.82×10^{19}
US to Lower Shell (LS) Circ. Weld (ID 75%)	WF-67 (Wire Ht. 72442)	0.26	0.60	2.04E+19	0.26	2.01×10^{19}
LS to Transition Forging Circ. Weld	WF-169-1 (Wire Ht. 8T1554)	0.16	0.57	4.78E+17	0.16	2.68×10^{17}

4.0 EMA OF ONS UNIT 3 RPV OUTLET NOZZLES AND TRANSITION FORGING

4.1 *Introduction*

Appendix G to 10 CFR Part 50, which addresses low upper-shelf toughness of the reactor pressure vessel (RPV) forgings, requires that an Equivalent Margins Analysis (EMA) be completed when the upper-shelf Charpy energy level falls below 50 ft-lb in order to demonstrate adequate material toughness. The 72 EFPY Charpy V-notch Upper Shelf Energy (CvUSE) calculations at the quarter thickness wall locations for Oconee (ONS) Units 1, 2, and 3, documented in Table 3-6, Table 3-7, and Table 3-8 concludes that the predicted CvUSE is above 50 ft-lbs at all three Oconee Units, with the following exceptions:

1. ONS Unit 3 Transition Forging: The ONS Unit 3 Transition forging (Heat 417543-1) predicted 72 EFPY CvUSE value is 48.8 ft-lb, which is below the 50 ft-lb threshold.
2. ONS Unit 3 Outlet Nozzle Forgings: The ONS Unit 3 outlet nozzle forgings have no recorded initial CvUSE. Consequently, predicted CvUSE values are not explicitly calculated.
3. All traditional and extended beltline Linde 80 welds, which are evaluated at 72 EFPY in Sections 3.5 and 3.6, and References 4-1 and 4-2.

The ONS Unit 3 RPV outlet nozzle forgings (2) and the transition forging (1) were procured from Klockner-Werke, are extended beltline materials for SLR, and Charpy V-Notch (CVN) testing on the upper shelf was not required by the construction code for these forgings. Therefore, 72 EFPY EMAs are required for the ONS Unit 3 RPV outlet nozzle forgings and transition forging to demonstrate compliance with 10 CFR Part 50, Appendix G. Consistent with BAW-2192 and BAW-2178, Revision 0, Supplement 1P-A, Revision 0, the Levels A and B Service Loadings for the RPV outlet nozzle forgings and transition forging are computed first, and only the limiting of these locations is evaluated for Levels C and D Service Loadings.

In addition, the EMAs for the ONS RPV outlet nozzle forgings and transition forging reported herein conservatively utilize a 72 EFPY peak inside surface fluence of 1.0×10^{18} n/cm^2 (E > 1.0 MeV) compared to the projected fluence of 2.68×10^{17} n/cm^2 for the transition forging and 2.21×10^{17} n/cm^2 for the RPV outlet nozzles (Table 2-3).

4.2 *Regulatory Requirements*

4.2.1 10 CFR 50 Appendix G

In accordance with 10 CFR 50, Appendix G (Reference 4-3), IV, A, 1, Reactor Vessel Upper Shelf Energy Requirements are as follows:

- Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into 10 CFR 50.55a(b)(2) at the time the analysis is submitted.
- Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.
- The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in 10 CFR 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate.

In accordance with NRC Regulatory Guide 1.161, the NRC has determined that the analytical methods described in the ASME Code, Section XI, Appendix K, provide acceptable guidance for evaluating reactor pressure vessels when the Charpy upper-shelf energy falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. However, the staff noted that Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. Selection of design transients and the methodology for calculation of applied J-integrals for the RPV outlet nozzles and transition forging are consistent with the methodology used to calculate applied J-integrals for Linde 80 welds (References 4-1 and 4-2).

4.2.2 COMPLIANCE WITH 10 CFR 50 APPENDIX G AND ACCEPTANCE CRITERIA

The current edition of the ASME Code, Section XI, listed in 10 CFR 50.55a is the 2017 Edition. Therefore, the analyses reported herein are performed in accordance with the 2017 Edition (Reference 4-4) of Section XI of the ASME Code, Appendix K. The material properties used in this analysis are based on the ASME Code, Section II, Part D, 2017 Edition. The ASME Code, Section XI, Subarticle K-2200, provides acceptance criteria for Levels A and B Service Loadings, and Subarticle K-2300 for Level C Service Loadings, and K-2400 for Level D Service Loadings. Both the RPV outlet forgings and transition forging are assessed against Subarticle K-2200. The limiting of these locations is evaluated against the ASME Code, Section XI, Subarticle K-2400, which provides acceptance criteria for Level D Service Loadings. However, for the Subarticle K-2400 evaluation reported herein the J-integral resistance versus flaw extension curve is a conservative representation for the vessel material under analytical evaluation and not a best estimate curve.

4.3 *Transition Forging Equivalent Margins Analysis (Levels A and B Service Loadings)*

The analytical procedure used for the 72 EFPY equivalent margins analysis of the ONS Unit 3 transition forging is in accordance with the ASME Code, Section XI, Appendix K, 2017 Edition (Reference 4-4), with selection of design transients based on the guidance in Regulatory Guide 1.161 (Reference 4-5).

4.3.1 Material Properties and Levels A and B Service Loadings

4.3.1.1 J-Integral Resistance Model

The J-integral resistance model is from NUREG/CR-5729 (Reference 4-6), i.e., RPV J_d base metal models reported in Table 11, Charpy Model and CVN_P Model. The J-integral resistance model (Charpy Model or CVN_P) that provides the most conservative J-integral resistance at a crack extension of 0.10 inches is selected for each evaluation. In order to utilize the NUREG/CR-5729 (Reference 4-6) base metal J_d Charpy and CVN_P models, initial unirradiated upper shelf energy is established for these extended beltline forgings procured from Klockner-Werke.

The ONS Unit 3 transition forging unirradiated CvUSE was established through NRC review of the ONS 54 EFPY P-T limits (NRC SER-ADAMS Accession Number ML14041A093, Page 8). The SER references an e-mail (ADAMS Accession Number ML13350A098) wherein the unirradiated CvUSE of 59.2 ft-lbs in the weak direction is reported for the ONS Unit 3 transition forging. This value is not a measured upper shelf energy but was obtained as follows.

Data taken from the actual RPV certified material test report where limited CVN test data were performed at a single test temperature (+10 °F) to confirm that at least 30 ft-lbs was obtained for the material. CVN test data does not represent the impact energy upper-shelf (i.e., >95% shear), but is a mean value taken at a reduced percent shear fracture and is a conservative estimate of the CvUSE in that the impact energy will increase as the percent shear fracture increases. This test report data was conservatively considered to be in the strong direction and the mean value of the reported CVN data at +10 °F was multiplied by 65% to obtain 59.2 ft-lbs in the weak direction.

4.3.1.2 Mechanical Properties of Transition Forgings

The reactor vessel lower shell and transition forgings are fabricated from A-508 Class 2 that is identified in the current ASME Code as SA-508 Grade 2 Class 1 material (Table 4-1). For the stainless steel cladding, 18Cr-8Ni Type 304 material properties are used.

The predicted Charpy impact energy value at 72 EFPY of 48.8 ft-lbs (Table 3-8) is used for the NUREG/CR-5729 Charpy Model and the associated initial Charpy impact energy value of 59.2 ft-lbs is used for the CVN_p model. The weak direction upper shelf energies, 72 EFPY and unirradiated, are conservatively used for both axial and circumferential flaws.

Table 4-1
Material Properties for SA-508 Class 2

Temp. (°F)	α (10^{-6} 1/°F)	E (10^{6} psi)	v (typ.)	C (BTU/lb-°F)	k (Btu/hr-in-°F)	ρ (lb/in^3)	σ_y (ksi)
70	6.4	27.8	0.3	0.105	1.975	0.2841	50.00
100	6.5	27.6	0.3	0.107	1.967	0.2839	50.00
200	6.7	27.1	0.3	0.113	1.958	0.2831	47.15
300	6.9	26.7	0.3	0.120	1.950	0.2823	45.25
400	7.1	26.2	0.3	0.125	1.925	0.2817	44.50
500	7.3	25.7	0.3	0.131	1.892	0.2809	43.20
600	7.4	25.1	0.3	0.136	1.850	0.2802	42.00
650	7.5	24.9	0.3	0.139	1.825	0.2797	41.40
700	7.6	24.6	0.3	0.142	1.800	0.2794	40.60

4.3.1.3 Levels A and B Service Loadings

The Levels A and B Service Loadings required by Appendix K are an accumulation pressure (internal pressure load) and a cooldown transient (thermal load). In accordance with Article K-1300 of Appendix K (Reference 4-4), the accumulation pressure used for flaw evaluations should not exceed 1.1 times the design pressure. With the design pressure of 2500 psig, the accumulation pressure is then 2750 psig (= 1.1×2500 psig). A conservative cooldown rate of 100°F/hour (from 560°F to 70°F in 4.9 hours) is used to represent the Levels A and B Service Loadings. This is the maximum attainable cooldown rate at ONS Unit 3 and the maximum cooldown rate required by Appendix K. This is consistent with BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Page 2-2.

4.3.1.4 Reactor Vessel Design Data

Pertinent design data for the thickness of the transition regions of the ONS Unit 3 reactor vessel are listed below.

4.3.2 Fracture Mechanics Analysis

4.3.2.1 Methodology and Acceptance Criteria

In accordance with the ASME Code, Section XI, Appendix K (Reference 4-4), Subarticle K-1200, the following analytical procedure was used for Levels A and B Service Loadings.

a. The flaws in the transition forging were postulated in accordance with Subarticle K-2200. When analytically evaluating adequacy of the upper shelf toughness for the base material, both interior axial and circumferential flaws with depths one-quarter of the wall thickness and lengths six times the depth shall be postulated, and toughness properties for the corresponding orientation shall be used. Smaller flaw sizes may be used when justified. The path lines of interest wherein flaws are postulated are illustrated in Figure 4-1.

b. Loading conditions at the locations of the postulated flaws were determined for Levels A and B Service Loadings. For Levels A and B Service Loadings the equations to calculate the stress intensity factor (SIF) due to pressure and thermal gradients for a given pressure and cooldown rate are given in Article K-4210. Consistent with BAW-2192, Revision 0, Supplement 1P-A, Revision 0 (Reference 4-1), the accumulation pressure is taken as ten percent above the design pressure and the maximum cooldown rate is 100°F/hr.

c. Material properties, including E, α, σ_y, and the J-integral resistance curve (J-R curve), were determined at the locations of the postulated flaws. Young's modulus, mean coefficient of thermal expansion and yield strength are addressed in Section 4.3.1.2. The J-R curve is discussed in Section 4.3.1.1.

d. The postulated flaws were evaluated in accordance with the acceptance criteria of Subarticle K-2200. Requirements for evaluating the applied J-integral are provided in Subarticle K-3200, and for determining flaw stability in Subarticle K-3400. Subarticle K-3500(a) invokes the procedure provided in Subarticle K-4200 for evaluating the applied J-integral for a specified amount of ductile flaw extension. Three permissible evaluation methods to address flaw stability are described in Subarticle K-3500(b). The evaluation method selected herein is the J-R curve crack driving force diagram procedure described in Subarticle K-4310.

4.3.2.2 Procedure for Evaluating Levels A and B Service Loadings

The EMA of the transition forging uses the fracture mechanics analytical procedures of the ASME Code, Section XI, Appendix K (Reference 4-4), augmented as necessary to incorporate location specific stresses determined by a 2 dimensional (2-D) finite element analysis and stress intensity factors calculated from an influence coefficient based solution for axial or circumferential surface flaws.

Stress intensity factors (SIFs) for Levels A and B Service Loadings are calculated based on the Raju and Newman solution (Reference 4-7) for internal semi-elliptical axial surface flaws in a cylinder, and the [ ] solution (Reference 4-8) for internal semi-elliptical circumferential surface flaws in cylindrical shells. An effective flaw depth is used to account for small scale yielding at the crack tip in computing the applied J-integral per Appendix K, Article K-4210. The adequacy of the upper-shelf toughness margin and demonstration of flaw stability is then evaluated according to Articles K-4220 and K-4310, respectively. The overall evaluation procedure for both axial and circumferential flaws is outlined below.

a. **Axial Flaw**

[ ]

[ ]

b. **Circumferential Flaw**

[ ]

<u>Temperature Range for Upper-Shelf Fracture Toughness Evaluations</u>: Upper-shelf fracture toughness is determined through use of Charpy V-notch impact energy versus temperature plots by noting the temperature above which the Charpy energy remains on a plateau, maintaining a relatively high constant energy level. Similarly, fracture toughness can be addressed in three different regions on the temperature scale, i.e., a lower-shelf toughness region, a transition region, and an upper-shelf toughness region. Fracture toughness of reactor vessel steel and associated weld metals are conservatively predicted by the ASME Code initiation toughness curve, K_{Ic}, in lower-shelf and transition regions. In the upper-shelf region, the upper-shelf toughness curve, K_{Jc}, is derived from the upper-shelf J-integral resistance model described. When upper-shelf toughness is plotted versus temperature, a plateau-like curve develops that decreases slightly with increasing temperature. Since the present analysis addresses the low upper-shelf fracture toughness issue, only the upper-shelf temperature range, which begins at the intersection of the K_{Ic} and upper-shelf toughness curves, is considered.

4.3.2.3 Evaluation for Levels A and B Service Loadings

The ASME Code, Section XI, Appendix K acceptance criteria have been satisfied for Levels A and B Service Loadings for ONS Unit 3 transition forging. For circumferential flaws, the minimum ratio $J_{0.1}/J_1$ is **[ ]** by the Charpy model, and **[ ]** by the CVN_p model. For axial flaws, the minimum ratio is **[ ]** by the Charpy model and **[ ]** by the CVN_p model. The overall minimum $J_{0.1}/J_1$ ratio is **[ ]**, which is greater than the minimum acceptable value of 1.0 required by Appendix K of the Code. In addition, using structural factors of 1.25 on pressure and 1.0 on thermal loading, flaw extension is demonstrated to be ductile and stable for the transition forging since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect.

The J-R curve data and applied J-integrals are plotted in Figure 4-2 through Figure 4-5 for circumferential flaws, and in Figure 4-6 through Figure 4-9 for axial flaws at each of the path lines shown in Figure 4-1. The limiting path line location is Base_1.

Figure 4-1
Stress Path Lines (Transition Forging)

[ ]

Figure 4-2
J-Integral versus Circumferential Flaw Extension – BASE_1
(Transition Forging)

Figure 4-3
J-Integral versus Circumferential Flaw Extension – BASE_2
(Transition Forging)

Figure 4-4
J-Integral versus Circumferential Flaw Extension – BASE_3
(Transition Forging)

Figure 4-5
J-Integral versus Circumferential Flaw Extension – BASE_4
(Transition Forging)

Figure 4-6
J-Integral versus Axial Flaw Extension – BASE_1 (Transition Forging)

Figure 4-7
J-Integral versus Axial Flaw Extension – BASE_2 (Transition Forging)

Figure 4-8
J-Integral versus Axial Flaw Extension – BASE_3 (Transition Forging)

Figure 4-9
J-Integral versus Axial Flaw Extension – BASE_4 (Transition Forging)

4.4 *RPV Outlet Nozzle Forgings Equivalent Margins (Levels A and B Service Loadings)*

The analytical procedure used for the 72 EFPY equivalent margins analysis, Levels A and B Service Loadings, of the ONS Unit 3 RPV outlet nozzle forging is in accordance with the ASME Code, Section XI, Appendix K, 2017 Edition (Reference 4-4), with selection of design transients based on the guidance in Regulatory Guide 1.161 (Reference 4-5).

4.4.1 Material Properties and Levels A and B Service Loadings

4.4.1.1 J-Integral Resistance Model

The J-integral resistance model is from NUREG/CR-5729 (Reference 4-6), i.e., RPV J_d base metal models reported in Table 11, i.e., Charpy Model and CV_{NP} Model. The J-integral resistance model (Charpy Model or CV_{NP}) that provides the most conservative J-integral resistance at a crack extension of 0.10 inches is selected for each evaluation. In order to utilize the NUREG/CR-5729 (Reference 4-6) base metal J_d Charpy and CV_{NP} models, initial unirradiated upper shelf energy is established for these extended beltline forgings procured from Klockner-Werke.

Unirradiated upper shelf energy for the ONS RPV outlet nozzle forgings procured from Klockner-Werke is not available from the certified material test report and was conservatively estimated at 78 ft-lbs in the strong direction and 52 ft-lbs in the weak direction based on the data reported in PWROG-17090-NP-A, Revision 0, Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination (Reference 4-9), Table 6. The mean minus 2 standard deviations CvUSE, in the strong and weak directions, for Fried-Krupp Huttenwerke AG Forgings reported in Table 6 were conservatively applied to the Klockner-Werke forgings. This assumption is justified based on review of PWROG-17090-NP-A, Revision 0, Table 7, wherein use of actual Klockner-Werke measured data and statistical mean in the strong direction minus 2 standard deviations would result in an unirradiated CvUSE greater than 78 ft-lbs in the strong direction.

4.4.1.2 Mechanical Properties

The reactor vessel outlet nozzle forgings are fabricated from A-508 Class 2 that is identified in the current ASME Code as A508-64 Grade 2 Class 1 material (Table 4-2). For the stainless steel cladding, 18Cr-8Ni Type 304 material properties are used.

Table 4-2
Material SA 508 Class 2 (3/4Ni-1/2Mo-1/3Cr-V)

Temperature (°F)	Modulus of Elasticity (psi)	Poisson's Ratio (-)	Yield Strength (ksi)
70	2.78E+07	0.30	50.00
100	2.76E+07	0.30	50.00
200	2.71E+07	0.30	47.00
270(1)	2.68E+07	0.30	45.95
300	2.67E+07	0.30	45.50
317(1)	2.66E+07	0.30	45.28
400	2.62E+07	0.30	44.20
500	2.57E+07	0.30	43.20
600	2.51E+07	0.30	42.10
650(1)	2.49E+07	0.30	41.40
700	2.46E+07	0.30	40.70

Note(s):
1. Interpolated value

The NUREG/CR-5729 RPV J_d base metal Charpy Model requires upper shelf energy at 72 EFPY as input. As such, a conservative fluence value of 1.0E+18 n/cm^2 at the inside surface was assumed at the RPV nozzles, with no attenuation, and the 72 EFPY CvUSE values at the RPV nozzles were calculated using RG 1.99, Position 1.2. The NUREG/CR-5729 RPV J_d CVN_p model requires unirradiated upper shelf energy, discussed above, and fluence (assumed at 1.0E+18 n/cm^2) as inputs. The estimate of 72 EFPY fluence of 1.0E+18 n/cm^2 is conservative relative to that reported in Table 2-3 for the ONS Unit 3 RPV outlet nozzles.

4.4.1.3 Levels A and B Service Loadings

The Levels A and B Service Loadings required by Appendix K are an accumulation pressure (internal pressure load) and a cooldown transient (thermal load). Per article K-1300 of Appendix K (Reference 4-4), the accumulation pressure used for flaw evaluation should not exceed 1.1 times the design pressure. With the design pressure of 2500 psig, the accumulation pressure is then 2750 psig. The ONS unit's design cooldown rate is 100 °F/hr, which is the maximum required by Appendix K of the ASME Code, Section XI and is consistent with BAW-2192, Revision 0, Supplement 1P-A, Revision 0. Pipe loads for Levels A and B Service Loadings are reported in Table 4-3.

Table 4-3
Service Levels A and B Outlet Nozzle Loads

[ ]

4.4.1.4 Reactor Vessel Design Data

Pertinent design data including the design pressure and geometry of the RPV shell and outlet nozzle forgings of the ONS Unit 3 reactor vessel are listed below.

4.4.2 Fracture Mechanics Analysis

4.4.2.1 Methodology and Acceptance Criteria

The analytical procedure was used for the evaluation of Levels A and B Service Loadings is in accordance with the ASME Code, Section XI, Appendix K (Reference 4-4), Subarticle K-1200.

a. Flaws in the RPV outlet nozzle forgings were postulated in accordance with the Subarticle K-2200. When analytically evaluating adequacy of the upper shelf toughness for the base material, both interior axial and circumferential flaws with depths one-quarter of the wall thickness and lengths six times the depth shall be postulated, and toughness properties for the corresponding orientation shall be used. Smaller flaw sizes may be used when justified. Path lines of interest in the RPV outlet nozzle are illustrated in Figure 4-10. Circumferential and axial flaws with depths one-quarter of the wall thickness and lengths six times the depth are postulated in the taper region (Path ntt). The defect in the corner region (Path nct) is an axial flaw in accordance with the ASME Code, Section XI, 2017 Edition, G-2223 and Figure G-2223-1, as modified below.

   - For the postulated nozzle corner defect, Appendix K requires that a ¼T flaw size be postulated when evaluating for Levels A and B Service Loadings. However, Appendix K, Subarticle K-2200 permits the analysis to be performed considering a smaller flaw size, if it can be justified. Herein, a 3-inch flaw is postulated at the nozzle corner, which is consistent with the maximum allowable postulated defect for the shell section that is 12-inches thick or greater. This postulated nozzle corner flaw is substantially greater than the allowable planar flaw at this location per the inservice inspection standards reported in Table IWB-3512-1 for the inside corner region.

b. Loading conditions at the locations of the postulated flaws were determined for Levels A and B Service Loadings. For Levels A and B Service Loadings the equations to calculate the stress intensity factor (SIF) due to pressure and thermal gradients for a given pressure and cooldown rate are given in Article K-4210. Consistent with BAW-2192P, Revision 0, Supplement 1P-A, Revision 0 (Reference 4-1), the accumulation pressure is taken as ten percent above the design pressure and the maximum cooldown rate is 100°F/hr.

c. Material properties, including E, σ_y, and the J-integral resistance curve (J-R curve), were determined at the locations of the postulated flaws. Young's modulus and yield strength are addressed in Section 4.4.1.2. The J-R curve is discussed in Section 4.4.1.1.

d. The postulated flaws were evaluated in accordance with the acceptance criteria of Subarticle K-2200. Requirements for evaluating the applied J-integral are provided in Subarticle K-3200, and for determining flaw stability in Subarticle K-3400. Subarticle K-3500(a) invokes the procedure provided in Subarticle K-4200 (K-4220) for evaluating the applied J-integral for a specified amount of ductile flaw extension. Three permissible evaluation methods to address flaw stability are described in Subarticle K-3500(b). The evaluation method selected herein is the J-R curve crack driving force diagram procedure described in Subarticle K-4310.

Figure 4-10
Path Lines of Interest (Outlet Nozzle Levels A and B)

4.4.2.2 Procedure for Evaluation

3. **<u>Upper Shelf Toughness at an extension of 0.1 inch</u>**: Evaluation of upper-shelf toughness at a flaw extension of 0.1 inch is performed for the flaw depths,

   a = 3 in. + 0.10 in. (for path line nct)

   a = 0.25t +0.10 in. (for path line ntt)

   Using:

   SF = 1.15

   $p = P_a$

Where P_a is the accumulation pressure for Levels A and B Service Loadings, such that:

$$J_1 < J_{0.1}$$

Where:

J_1 = the applied J-integral for a structural factor of 1.15 on pressure, and a structural factor of 1.0 on thermal loading

$J_{0.1}$ = the J-integral resistance at a ductile flaw extension of 0.10 in.

4. **Evaluation of Flaw Stability**: Evaluation of flaw stability is performed through use of a crack driving force diagram procedure by comparing the slopes of the applied J-integral curve and the material J-integral resistance curve, or J-R curve. The applied J-integral is calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw extension. The applied pressure is the accumulation pressure for Levels A and B Service Loadings, P_a, and the structural factor (SF) on pressure is 1.25. Flaw stability at a given applied load is verified when the slope of the applied J-integral curve is less than the slope of the J-R curve at the point on the J-R curve where the two curves intersect.

5. **Postulated Flaw**: For nozzle corner path line (nct) crack depth is selected as 3 inches (that corresponds to ¼ of the nozzle belt shell wall thickness) and for taper-transition region path line (ntt) crack depth is selected as ¼ of the wall thickness at the taper section.

4.4.2.3 Evaluation for Levels A and B Service Loadings

The ONS Unit 3 outlet nozzle EMA, in accordance with Appendix K, K-4200 applied stress intensity factor formulations at 72 EFPY demonstrate that the ASME Code, Section XI, Appendix K (Reference 4-4) acceptance criteria have been satisfied for Levels A and B Service Loadings considering both the CVN_p and the Charpy methods. The results of the analysis are as follows:

1. Figure 4-11 for the nozzle corner axial flaw, Figure 4-12 for the tapered transition axial flaw, and Figure 4-13 for the tapered transition circumferential flaw, show that with a factor of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral ($J_{applied}$) is less than the lower bound J-integral of the material resistance curve at a ductile flaw extension of 0.10 inch ($J_{0.1}$). The ratio of $J_{0.1}/J_{applied}$ for the CVN_p method is [ ] for the nozzle corner axial flaw, tapered transition axial flaw, and tapered transition circumferential flaw, respectively. Also, the ratio of $J_{0.1}/J_{applied}$ for the Charpy method is [ ] for the nozzle corner axial flaw, tapered transition axial flaw, and tapered transition circumferential flaw, respectively, which all are greater than the required value of 1.0.
2. Figure 4-11 for the nozzle corner axial flaw, Figure 4-12 for the tapered transition axial flaw, and Figure 4-13 for the tapered transition circumferential flaw, show that with a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable because the slope of the applied J-integral curve is less than the slope of the lower J-R curve at the point where the two curves intersect.

Figure 4-11
J-Integral versus Axial Flaw Extension – Nozzle Corner (Outlet Nozzle Levels A and B)

Figure 4-12
J-Integral versus Axial Flaw Extension – Taper Transition (Outlet Nozzle Levels A and B)

[ ]

Figure 4-13
J-Integral versus Circumferential Flaw Extension – Taper Transition (Outlet Nozzle Levels A and B)

4.5 *RPV Outlet Nozzle Forgings Equivalent Margins Analysis (Levels C and D Service Loadings)*

The analytical procedure used for the 72 EFPY equivalent margins analysis, Levels C and D Service Loadings, of the ONS Unit 3 RPV outlet nozzle forging is in accordance with the ASME Code, Section XI, Appendix K, 2017 Edition (Reference 4-4), with selection of design transients based on the guidance in Regulatory Guide 1.161 (Reference 4-5) and consistent with Levels C and D design transients reported in BAW-2178, Revision 0, Supplement 1P-A, Revision 0 (Reference 4-2). Consistent with BAW-2178, Revision 0, Supplement 1P-A, Revision 0, only the most limiting item (i.e., RPV outlet nozzle) is evaluated for Levels C and D Service Loadings based upon the Levels A and B EMAs reported in Sections 4.3 and 4.4 above.

4.5.1 Material Properties and Levels C and D Service Loadings

4.5.1.1 J-Integral Resistance Model

The J-integral resistance model is from NUREG/CR-5729 (Reference 4-6), i.e., RPV J_d base metal models reported in Table 11, i.e., Charpy Model and CVN_P Model as described in Section 4.4.1.1.

4.5.1.2 Mechanical Properties

The mechanical properties for the Levels C and D EMA are consistent with Section 4.4.1.2. The calculation of the plastic zone correction uses the yield stress limit,

S_y = **[ ]**

The NUREG/CR-5729 RPV J_d base metal Charpy Model requires upper shelf energy at 72 EFPY as input. As such, a conservative fluence value of 1.0E+18 n/cm^2 at the inside surface was assumed at the RPV nozzles with no attenuation, and the 72 EFPY CvUSE values at the RPV nozzles were calculated using RG 1.99, Position 1.2. The NUREG/CR-5729 RPV J_d CVNp model requires unirradiated upper shelf energy, discussed above, and fluence (assumed at 1.0E+18 n/cm^2) as inputs. The estimate of 72 EFPY fluence of 1.0E+18 n/cm^2 is conservative relative to that reported in Table 2-3 for the ONS RPV outlet nozzles.

4.5.1.3 Levels C and D Service Loadings

Thermal Transients

Levels C and D transients for the outlet nozzle forging are consistent with Levels C and D transients reported in BAW-2178, Revision 0, Supplement 1P-A, Revision 0 (Reference 4-2). There is one Level C transient and the other four (4) are Level D transients.

[ ]

Pipe Loads

In order to account for the influence of the nozzle loads, loads are developed for faulted service conditions and are shown in Table 4-4. These loads are also used for emergency (Level C) service level transient.

Table 4-4
Service Levels C and D Outlet Nozzle Loads

[ ]

4.5.1.4 Reactor Vessel Design Data

Pertinent design data including the design pressure and geometry of the RPV shell and outlet nozzle forgings of the ONS Unit 3 reactor vessel are reported in Section 4.4.1.4.

4.5.2 Fracture Mechanics Evaluation

4.5.2.1 Methodology and Acceptance Criteria

In accordance with the ASME Code, Section XI, Appendix K, Subarticle K-1200, the following analytical procedure was used for Levels C and D Service Loadings.

a. Flaws in the RPV outlet nozzle forging were postulated in accordance with the Subarticles K-2300 and K-2400. Three locations of postulated defects are illustrated in Figure 4-17: nozzle corner, nozzle boss, and taper region.

b. Loading conditions at the locations of the postulated flaws were determined for Levels C and D Service Loadings.

c. Material properties, including E, σ_y, and the J-integral resistance curve (J-R curve), were determined at the locations of the postulated flaws. Young's modulus and yield strength are addressed in Section 4.5.1.2. The J-R curve is discussed in Section 4.5.1.1.

d. The postulated flaws were evaluated in accordance with the acceptance criteria of Subarticle K-2300 (both Levels C and D Service Loadings) by calculating the applied J-integral according to the procedure provided by Subarticle K-5210. The applied J-integral was then evaluated to satisfy the criteria for flaw extension in Subarticle K-5220 and flaw stability in Subarticle K-5300.

4.5.2.2 Procedure for Evaluation

[

]

1. **Procedure for Evaluating Levels C and D Service Loadings**: The methodology used herein to evaluate Levels C and D Service Loadings is outlined as follows:

[ ]

2. **Temperature Range for Upper-Shelf Fracture Toughness Evaluations**: Upper-shelf fracture toughness is determined through use of Charpy V-notch impact energy versus temperature plots by noting the temperature above which the Charpy energy remains on a plateau, maintaining a relatively high constant energy level. Similarly, fracture toughness can be addressed in three different regions on the temperature scale, i.e., a lower-shelf toughness region, a transition region, and an upper-shelf toughness region. Fracture toughness of the outlet nozzle forgings material is conservatively predicted by the ASME Code initiation toughness curve, K_{Ic}, in lower-shelf and transition regions. In the upper-shelf region, the upper-shelf toughness curve, K_{Jc}, is derived from the upper-shelf J-integral resistance model. When upper-shelf toughness is plotted versus temperature, a plateau-like curve develops that decreases slightly with increasing temperature. Since the present analysis addresses the low upper-shelf fracture toughness issue, only the upper-shelf temperature range, which begins at the intersection of K_{Ic} and the upper-shelf toughness curves, is considered.

3. **Effect of Cladding Material on Stress Intensity Factor**: The Finite Element model utilized to develop inputs for this analysis features cladding; therefore, the effect of cladding is included.

   Significant stress discontinuities can be observed at plots of the through wall thermal stress at the cladding-to-base metal transition region. These stress discontinuities generate (in most cases) compressive stresses on the wetted surface of the shell and consequently relieve the overall tensile stresses which would open the postulated flaw. For this reason, the compressive stresses are neglected in the calculation of the final safety margin.

4.5.2.3 Evaluation for Levels C and D Service Loadings

The results of the low upper shelf toughness calculations for Levels C and D Service Loadings with projected fluence at 72 EFPY (80 calendar years) for Oconee Unit 3 Outlet Nozzle forging demonstrate that the acceptance criteria of the ASME Code, Section XI, Appendix K for Levels C and D Service Loadings have been met.

As shown in Table 4-5, the minimum equivalent margin for the outlet nozzle forging is [ ] (minimum required is ≥ 1.0). The limiting location on the outlet nozzle forging is the nozzle corner and the limiting transient is the [ ] transient.

The J-R curve data and applied J-integrals are plotted in Figure 4-14, Figure 4-15 and Figure 4-16 for the path lines shown in Figure 4-17, and demonstrate that:

1. With a structural factor of 1.0 on all loading, the applied J-integral (J_1) is less than the J-integral of the material at a ductile flaw extension of 0.10 inch ($J_{0.1}$).
2. With a structural factor of 1.0 on all loading, the flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point of intersection for both (CVNp and Charpy) material models.
3. The remaining ligament (limiting location at the safe end) is sufficient to preclude tensile instability by sufficient margin.

Note that the margins presented in Table 4-5 are calculated using lower bound J-R curve for both the Levels C and D Service Loadings, and that applied J-integral (J_1) neglects the effect of cladding and attached piping which in these specific configurations decrease the value of applied J-integral.

Table 4-5
ONS Unit 3 Outlet Nozzle EMA Results (Levels C and D)

Location	Temperature	$J_{applied}$	$J_{resistance}$	Margin
	T (°F)	(lb/inch)		
Nozzle Corner	590	[ ]	[ ]	[ ]
Nozzle Boss	467	[ ]	[ ]	[ ]
Nozzle Taper	492	[ ]	[ ]	[ ]

Figure 4-14
J-integral versus Flaw Extension for Nozzle Corner (Outlet Nozzle Levels C and D)

Figure 4-15
J-integral versus Flaw Extension for Nozzle Boss (Outlet Nozzle
Levels C and D)

Figure 4-16
J-integral versus Flaw Extension for Nozzle Taper (Outlet Nozzle Levels C and D)

[ ]

Figure 4-17
Locations of Interest (Outlet Nozzle Levels C and D)

[ ]

4.6 *References for Section 4.0*

4-1. BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads. ADAMS Accession Number ML19101A355

4-2. BAW-2178, Revision 0, Supplement 1P-A, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads. ADAMS Accession Number ML19101A355

4-3. 10 CFR 50 Appendix G, Code of Federal Regulations, Title 10, Part 50 – Domestic Licensing of Production and Utilization Facilities, Appendix G – Fracture Toughness Requirements, Federal Register Vol. 60. No. 243, December 19, 1995

4-4. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 2017 Edition

4-5. Regulatory Guide 1.161, Evaluation of Reactor Pressure Vessels With Charpy Upper Shelf Energy Less Than 50 ft-lb

4-6. NUREG/CR-5729, “Multivariable Modeling of Pressure Vessel and Piping J-R Data

4-7. I.S. Raju and J.C. Newman, Jr., “Stress-Intensity Factors for Internal and External Surface Cracks in Cylindrical Vessels,” Journal of Pressure Vessel Technology, Vol. 104, pp. 293-298, November 1982

4-8. [

]

4-9. PWROG-17090-NP-A, Revision 0, Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination, ADAMS Accession Number ML20024E238

5.0 PRESSURIZED THERMAL SHOCK

5.1 *Introduction*

Title 10 of the Code of Federal Regulation (CFR) 50.61 provides requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of pressurized thermal shock reference temperature (RT_{PTS}) whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility. Since RT_{PTS} is a function of the end of license fluence, which is associated with the 60-year licensed operating period, the RT_{PTS} calculations meet the criteria of 10 CFR 54.3(a) and have been identified as Time Limited Aging Analyses (TLAA) requiring evaluation for 80 years.

5.2 *Regulatory Guidance for Subsequent License Renewal*

The regulatory guidance for NRC review of pressurized thermal shock evaluations performed in accordance with 10 CFR 54.21(c)(1)(ii) is reported in NUREG-2192, Section 4.2.3.1.3.2 (Reference 5-1) and is repeated below.

“The documented results of the revised PTS analysis based on the projected neutron fluence at the end of the subsequent period of extended operation are reviewed for compliance with 10 CFR 50.61 or 10 CFR 50.61a.

The NRC staff confirms that the applicant has provided sufficient information for PTS for the subsequent period of extended operation as follows:

The applicant identified the neutron fluence at the clad-to-base metal interface for each beltline material at the expiration of the subsequent period of extended operation.

There are two methodologies from 10 CFR 50.61 that can be used in the PTS analysis, based on the projected neutron fluence at the end of the subsequent period of extended operation. RT_{NDT} is the reference temperature (NDT means nil-ductility temperature) used as an indexing parameter to determine the fracture toughness and the amount of embrittlement of a material. RT_{PTS} is the reference temperature used in the PTS analysis and is related to RT_{NDT} at the end of the facility's operating license.

The first methodology does not rely on plant-specific surveillance data to calculate delta RT_{NDT} (ΔRT_{NDT}, the mean value of the adjustment or shift in reference temperature caused by irradiation). The ΔRT_{NDT} is determined by multiplying a chemistry factor from the tables in 10 CFR 50.61 by a neutron fluence factor calculated from the neutron flux using Equation 3 in 10 CFR 50.61.

For the surveillance data to be defined as credible, the difference in the predicted values and the measured values for ΔRT_{NDT} must be less than 15.6 °C (28 °F) for weld metal components or less than 9.4 °C (17 °F) for base metal components. When a credible surveillance data set exists, the chemistry factor can be determined from these data in lieu of a value from the table in 10 CFR 50.61. The standard deviation for the ΔRT_{NDT} used in the margin term assessment (e.g., $\sigma\Delta$) of the RT_{PTS} calculations may be reduced from 15.6 °C (28 °F) to 7.8 °C (14 °F) for welds or from 9.4 °C (17 °F) to 4.7 °C (8.5 °F) for base metal materials. However, $\sigma\Delta$ need not exceed one-half of the RT_{NDT} value used in the RT_{PTS} calculations.

To confirm that the PTS analysis results in RT_{PTS} values below the screening criteria in 10 CFR 50.61 at the end of the subsequent period of extended operation, the applicant provides the following:

1. For each beltline material, provide the unirradiated RT_{NDT}, the method of calculating the unirradiated RT_{NDT} (either generic or plant-specific), the margin, chemistry factor, the method of calculating the chemistry factor, the mean value for the shift in transition temperature, and the RT_{PTS} value.

2. If there are two or more data for a surveillance material that is from the same heat of material as the beltline material, provide analyses to determine whether the data are credible in accordance with NRC RG 1.99, Revision 2, and whether the margin value used in the analysis is appropriate.

3. If a surveillance program does not include the vessel beltline controlling material but two or more data sets are available from other beltline materials, then provide an analysis of the data in accordance with NRC RG 1.99, Revision 2, Regulatory Position C.2.1, to show that the results either bound or are comparable to the values that would be calculated for the same materials using Regulatory Position C.1.1.

If the PTS screening criteria in 10 CFR 50.61 are projected to be exceeded during the subsequent period of extended operation, an analysis based on NRC RG 1.154 or 10 CFR 50.61a may be submitted for review. For applicants with PTS analysis based upon an NRC-approved submittal using 10 CFR 50.61a, the analysis is revised to reflect the subsequent period of extended operation."

5.3 *Methodology*

10 CFR 50.61(c) provides two methods for determining RT_{PTS}. These methods are also described in Section 1.1 (i.e., Position 1.1) and Section 2.1 (i.e., Position 2.1) of Regulatory Guide 1.99, Revision 2 (Reference 5-2). Position 1.1 applies for material without credible surveillance data available and Position 2.1 is used for material with two or more credible surveillance data sets available. The RT_{PTS} values are calculated for both Positions 1.1 (all 3 Units) and 2.1 (ONS Unit 3 only) by following the guidance in Regulatory Guide 1.99, Revision 2, using the copper and nickel content of the Units 1, 2, and 3 RPV materials, and subsequent period of extended operation 72 EFPY fluence projections at the clad/base metal interface.

5.4 *Traditional Beltline and Extended Beltline 72 EFPY Pressurized Thermal Shock*

The 10 CFR 50.61(c) methods were used with the 72 EFPY base metal/clad interface fluence values reported in Table 2-1, Table 2-2, and Table 2-3 to calculate RT_{PTS} values for Oconee Units 1, 2, and 3 at 72 EFPY. The subsequent period of extended operation 72 EFPY RT_{PTS} calculations for Oconee Units 1, 2, and 3 are summarized in Table 5-10, Table 5-11, and Table 5-12.

10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS} as 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds. In accordance with SECY 82-465, Enclosure A (Reference 5-3), which provides the bases for the PTS screening criteria, the NRC staff reported that further justification was needed from the B&W owners for longer term use of the PTS screening criteria for B&W plants. The B&WOG provided justification for long term use of the PTS screening criteria for B&W plants through BAW-1791 (Reference 5-4), and Duke Energy provided a separate justification for applicability of the PTS screening criteria to B&W-designed plants through Reference 5-5.

As reported in BAW-1791, the methods of probabilistic risk assessment were applied to a comprehensive set of initiating events to determine the frequencies of possible PTS sequences for a generic configuration of the B&W-designed 177-fuel assembly plant. Best-estimate thermal-hydraulics analyses were used to bound the temperature and pressure histories of PTS sequences in nine categories. The thermal responses and frequencies were input to a probabilistic fracture mechanics correlation to relate the probabilities of through-wall vessel cracking to mean surface transition temperatures.

The aggregate results were compared to the NRC evaluation set forth in SECY 82-465, Enclosure A. It was concluded that the aggregate frequencies for crack initiation without arrest in B&W designed plants are more than one order of magnitude less than those established by the NRC for other pressurized water reactor designs. BAW-1791 supplements and extends earlier pressurized thermal shock evaluations for B&W-designed plants, and the information presented confirms the applicability, without modification, of the proposed PTS screening criteria of SECY 82-465 to all B&W-designed plants. Therefore, the PTS screening criteria of 270°F for plates, forgings, and longitudinal welds and 300°F for circumferential welds remain applicable to the Oconee units for the subsequent period of extended operation.

All of the ONS Units 1, 2, and 3 RPV materials are below the RT_{PTS} screening criteria values of 270°F for base metal and longitudinal welds, and 300°F for circumferentially oriented welds through the subsequent period of extended operation. For Unit 1, the limiting materials are axial welds SA-1426 and SA-1430 with a projected value of RT_{PTS} at 72 EFPY of 207.2°F (screening criterion of 270°F). For Unit 2, the limiting material is circumferential weld WF-25 with a projected value of RT_{PTS} at 72 EFPY of 245.1°F (screening criterion of 300°F). For Unit 3, the limiting material is circumferential weld WF-67 with a projected value of RT_{PTS} at 72 EFPY of 240.2°F (screening criterion of 300°F).

5.4.1 Material Properties Including Copper, Nickel, and Initial RT_{NDT},

Material properties required for the calculation of 72 EFPY RT_{PTS} are reported in Table 5-10, Table 5-11, and Table 5-12, and include copper content (weight %), nickel content (weight %), and initial RT_{NDT}. Copper content (weight %) and a comparison to the current licensing basis is addressed in Sections 3.4.1 and 3.4.2, above. For traditional beltline locations, the nickel content, and initial RT_{NDT} are consistent with the current licensing basis (Section 1.0, Reference 1-2) with some exceptions as reported in Section 5.4.2. For extended beltline locations (i.e., RPV inlet and outlet nozzles and associated welds and the transition forging), material properties have not been reported to the NRC and nickel content, initial RT_{NDT}, and σ_I for SLR for these items are reported below.

<u>Nickel Content</u>

Nickel content for ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV outlet nozzle forgings and inlet nozzle forgings are plant-specific and are directly from the product analysis reported in the certified material test reports for the specific heat of material. Nickel content of the transition forgings are from the product analysis of specific heats of material measurements as reported in BAW-1820 (Reference 5-6). The nickel content of extended beltline Linde 80 welds are mean values from weld wire heat specific measurements, as permitted by Regulatory Guide 1.99, Revision 2. The nickel content in weight % for extended beltline locations (base metal and welds) are reported in Table 5-10, Table 5-11, and Table 5-12 for ONS Unit 1, ONS Unit 2, and ONS Unit 3, respectively.

Initial RT_{NDT} and σ_I

The initial RT_{NDT} for the ONS Unit 1 and ONS Unit 2 RPV inlet and outlet nozzle forgings, ONS Unit 1 and ONS Unit 2 transition forgings, and the ONS Unit 3 RPV inlet nozzle forgings is a generic mean from the data presented in Table 5-1 below. For ASME SA-508 (ASTM A 508) Class 2 forgings where no initial (unirradiated) reference temperatures are available, a statistical evaluation was performed on the available data to obtain a generic mean value. The mean RT_{NDT} was found to be [ ], and the standard deviation is [ ].

The data in Table 5-1 was obtained from forgings manufactured by US suppliers. A review of forging data from non-US suppliers indicates that the data for the non-US forging materials may not be representative of US-supplied materials. Therefore, the data illustrated in Table 5-1 is used for forgings manufactured by US suppliers, and not for initial RT_{NDT} values for forgings from a non-US supplier, such as Klockner Werke.

ONS Unit 3 Klockner-Werke RPV Outlet Nozzle Forgings and Transition Forging

The initial RT_{NDT} and σ_I of the ONS Unit 3 Klockner Werker forgings are estimated from the Rotterdam Dockyard data reported in Table 5-2 below. The generic mean is [ ] and standard deviation of [ ]. The actual suppliers of the forgings to Rotterdam Dockyards for the plants listed in Table 5-2 are not known but likely included Klockner-Werke, Fried-Krupp Huttenwerke AG, and Rheinstahl Huttenwerke AG based on review of PWROG-17090, Table 2 (Reference 5-7). As such, the above generic mean of [ ] and standard deviation of [ ] are acceptable due to consideration of the class of European forgings that are representative of the ONS Unit 3 RPV outlet nozzle forgings and transition forging.

Extended Beltline Linde 80 Welds

For weld metals fabricated with Linde 80 flux where no initial (unirradiated) reference temperatures are available (e.g., RPV nozzle welds), a statistical evaluation was performed on the available data to obtain a generic mean value. The generic mean was found to be [ ], and the standard deviation is [ ], based on test data reported in Table 5-3 below.

For weld wire heat number 299L44, the initial RT_{NDT} is [ ] and its standard deviation is [ ], respectively, based on the "group" data for this weld wire based on test data reported in Table 5-4 below. This data may be applied outside the traditional RPV beltline.

Table 5-1
Reference Temperature Data for Forging Materials Used to Determine Generic Initial Reference Temperature

[ ]

Table 5-2
Generic Initial RT_{NDT} for ASTM A508 Class 2 Forgings Supplied to Rotterdam Dockyard

Table 5-3
Reference Temperature Data for Linde 80 Weld Metals Used to Determine Generic Initial Reference Temperature

Table 5-4
Reference Temperature Data for Weld Wire Heat No. 299L44 Used to Determine Generic Initial Reference Temperature

[ ]

5.4.2 Comparison to Current Licensing Basis

For traditional beltline materials, differences between the current licensing basis (see Section 1.0) and the Ni wt% content and initial RT_{NDT} values utilized for subsequent license renewal are identified in Table 5-5 and Table 5-6, respectively. For items not listed, the CLB and SLR Ni wt% and initial RT_{NDT} values are consistent. In addition, Ni wt% and Initial RT_{NDT} values reported in the NRC Reactor Vessel Integrity Database (RVID2, Reference 5-8) database are included for information. The data and information provided by licensees in their responses to the staff's Generic Letter (GL) 92-01, Revision 1 close-out letters, and in response to GL 92-01, Revision 1, Supplement 1, are included in RVID2. This database includes updates from June 1999 - July 2000. Duke Energy’s responses to GL 92-01, GL 92-01, Revision 1, and GL 92-01, Revision 1, Supplement 1 are based on generic topical reports. The generic Topical Reports that provided Oconee’s response to GL 92-01 are reported in Section 3.4.2.

The Ni wt% values and initial RT_{NDT} reported herein for SLR were calculated with additional information and subsequent to those found in the CLB; therefore, they are an improved representative of the materials.

<u>Nickel differences-CLB versus SLR-Traditional Beltline Materials</u>

Differences for Ni content values are identified in Table 5-5. The value of 0.57% for heat 8T1762 is the mean value of 6 weld qualification samples and is identical to the nickel wt% reported in the NRC RVID2.

Initial RT_{NDT}-CLB versus SLR-Traditional Beltline Materials

Differences for the mean and standard deviation for initial RT_{NDT} traditional beltline items are identified in Table 5-6. The NRC RVID2 database has not been updated to reflect the changes to Linde 80 weld initial RT_{NDT} based on BAW-2308, Revisions 1-A and 2-A (Reference 5-9). The mean and standard deviation for initial RT_{NDT} values reported in the CLB for forgings and plates were calculated based a dataset comprised of Bellefonte Unit 1 and Unit 2 (BLN-1 and BLN-2) reactor vessel forgings and reactor vessel & steam generator plates, respectively. BLN-1 and BLN-2 are later generation B&W units and, therefore, may have more favorable initial RT_{NDT} values than would be representative of the earlier vintage B&W units.

Therefore, material data that were more representative of the earlier vintage B&W units are utilized and initial RT_{NDT} and associated σ_I of traditional beltline plates, forgings, and welds were recalculated for subsequent license renewal with removal of the BLN-1 and BLN-2 data. These more representative results for the mean and standard deviation for initial RT_{NDT} values are reported herein and apply to forgings, plates, and welds fabricated with Linde 80 flux and are listed in Table 5-6. Details of the revised initial RT_{NDT} values relative to NRC RVID2 are provided below if the revision for SLR provides greater margins to embrittlement limits (e.g., if either initial RT_{NDT} is reduced or σ_I is reduced). Traditional beltline forging and plate material items for which material properties have been revised such that margins may be reduced to embrittlement limits for SLR are as follows. Traditional beltline Linde 80 welds have all been revised based on BAW-2308.

- Oconee Unit 1—LNB forging AHR 54, IS plates C2197-2, US plates C3265-1 and C3278-1, LS plate C2800-1 and C2800-2
- Oconee Unit 2—LNB forging AMX 77
- Oconee Unit 3—LNB forging 4680

For those instances where test data is not available for the determination of unirradiated initial reference temperature (RT_{NDT}), for base metal plate, forgings, and weld metals fabricated with Linde 80 flux, generic mean values have been established as follows.

Plates

For ASME SA-533 (ASTM A 533) Grade B Class 1 plates and ASME SA-302 (ASTM A 302) Grade B Modified plates where no initial (unirradiated) reference temperatures are available, a statistical evaluation was performed on the available data to obtain a generic mean value. The mean was found to be [ ], and the standard deviation is [ ]. The data is reported in Table 5-7.

Forgings

For ASME SA-508 (ASTM A 508) Class 2 forgings where no initial (unirradiated) reference temperatures are available, a statistical evaluation was performed on the available data to obtain a generic mean value. The mean was found to be [ ], and the standard deviation is [ ]. The data is reported in Table 5-8. This data was obtained from forging manufactured from US suppliers and is not applicable to the following ONS Unit 3 forgings supplied by Klockner Werke and Rotterdam Dockyard: ONS Unit 3 RPV outlet nozzle and transition forgings (see Section 5.4.1) by Klockner Werke, and ONS Unit 3 lower nozzle belt forging by Rotterdam Dockyard.

LNB Forging (4680) for ONS Unit 3 was procured from the Rotterdam Dockyard, for which measured values were not available from the time of manufacturing; therefore, the values reported in the CLB and NRC RVID2 are generic initial RT_{NDT} values determined by B&W from U.S. forging companies, e.g., Ladish Co, Bethlehem Steel Corporation, or Midvale-Hepppenstall. Framatome located archive ONS Unit 3 LNB forging material and testing was performed in accordance with ASTM E208-06 to determine the nil-ductility transition temperature. Testing resulted in an initial RT_{NDT} of +10 °F with σ_I=0.0 °F, which is reported in Table 5-6.

For direct test measurements, ASTM E208, Section 17.1, addresses precision and bias and contains the following: no information is presented about either the precision or bias of Test Method E208 since the test result is non-quantitative. As such, Framatome's position is that when actual drop weight test data is available for a specific plate, forging, or weld to establish T_{NDT} in accordance with ASTM E208, determination of uncertainty in the test method is not applicable since the test result is non-quantitative and σ_I=0.0. This position is consistent with that reported in BAW-2222 (Reference 5-10), Reactor Vessel Working Group Response to Closure Letters to NRC Generic Letter 92-01, Revision 1 for Oconee Units 1, 2, and 3, for all plant-specific entries. When establishing a generic mean, the margin (σ_I) is calculated from the statistical evaluation of the test data using the standard deviation.

As such, σ_I values have been reduced to 0.0 °F for ONS Unit 1 plates C3265-1 and C2800-2 based on test data reported in Table 5-7. In each case, initial RT_{NDT} values have increased and are based on the measured values reported in Table 5-7.

Weld Metal Fabricated with Linde 80 Flux

For weld metals fabricated with Linde 80 flux where no initial (unirradiated) reference temperatures are available, a statistical evaluation was performed on the available data to obtain a generic mean value. The mean value initial RT_{NDT} is **[ ]**, and the standard deviation is **[ ]** as reported in Section 5.4.1 above.

For weld wire heat number 299L44, the mean value initial RT_{NDT} is **[ ]**, and standard deviation is **[ ]**, based on the "group" data for this weld wire as reported in Section 5.4.1 above.

BAW-2308 (Reference 5-9) was prepared for the B&W Owners Group (B&WOG) Reactor Vessel Working Group (RVWG) to justify alternative initial reference temperatures (IRT_{NDT}) for the Linde 80 beltline welds in B&W fabricated reactor vessels. The alternative IRT_{NDT} was determined based on brittle-to-ductile transition range fracture toughness test data of these weld metals obtained in accordance with ASTM Standard E1921 and using ASME Boiler and Pressure Vessel Code Case N-629. BAW-2308 was submitted to the NRC for review and acceptance as a B&WOG topical for application to the pressurized thermal shock (PTS) rule (10 CFR 50.61) and 10 CFR Part 50, Appendix G, pressure temperature limits. The heat-specific and generic RT_{To} and initial margin terms are tabulated in Table 5-9. The data in Table 5-9 is an excerpt from Table 9 of BAW-2308 Revision 2-A.

Duke Energy requested an exemption (Reference 5-11) from the requirements of 10 CFR 50.61 and 10 CFR Part 50, Appendix G to revise plant-specific initial RT_{NDT} values using BAW-2308, Rev 1-A, and its supplement BAW-2308, Rev 2A. This exemption request was approved by the NRC staff on April 26, 2012 (Reference 5-12). There is no specific time dependency associated with this exemption request and it may appropriately be applied to traditional RPV beltline locations for subsequent license renewal. The exemption may not be applied to extended beltline locations, as neither the Duke Energy exemption request nor the NRC SER addressed applicability of the exemption request to extended beltline locations.

Table 5-5
Traditional Beltline Ni Content Differences between CLB and SLR

[ ]

Table 5-6
Traditional Beltline Initial RT_{NDT} Differences between CLB and SLR

Table 5-7
Reference Temperature Data for Plate Materials Used to Determine Generic Initial Reference Temperature

Table 5-8
Reference Temperature Data for Forging Materials Used to Determine Generic Initial Reference Temperature

[ ]

Table 5-9
Heat Specific and Generic Initial RT_{To} with Associated Initial Margin

Linde 80 Heat	With Proposed ASTM E1921 Loading Rate Adjustment	
	IRT_{T0} (°F)	Initial Margin a_I (°F)
406L44	-98.0	11.6
71249	-53.5	12.8
72105	-31.1	13.7
821T44	-84.2	9.6
299L44	-74.3a	12.8a
72442	-33.2	12.2
72445	-72.5	12.0
61782b	-58.5	15.4
Other heats	-48.6a	18.0a

Notes:

a. This value includes new data from that reported within BAW-2308-1A.

b. This heat was not reported within BAW-2308-1A.

5.5 *References for Section 5.0*

5-1. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

5-2. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials

5-3. SECY 82-465, Pressurized Thermal Shock, ADAMS Accession Number ML16232A574

5-4. BAW-1791, B&W Owners Group Probabilistic Evaluation of Pressurized Thermal Shock, Phase 1 Report, ADAMS Accession Number ML20024A860, 8307010051 (legacy)

5-5. Duke Power Company Comments on Pressurized Thermal Shock Screening Criteria, ADAMS Accession Number ML20028E598, 8301280040 (legacy)

5-6. BAW-1820, Babcock & Wilcox Owner's Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information, ADAMS Accession Number ML20108D545

5-7. PWROG-17090-NP-A, Revision 0, Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination, ADAMS Accession Number ML20024E238

5-8. Reactor Vessel Integrity Database Version 2.0.1, https://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html

5-9. BAW-2308, Revision 1-A and 2-A, Initial RTNDT of Linde 80 Weld Metals ADAMS, Accession Number ML081270388 for 2-A. Final Safety Evaluation for Revision 1 is ML052070408

5-10. BAW-2222, Reactor Vessel Working Group Response to Closure Letters to NRC Generic Letter 92-01, Revision 1, ADAMS Legacy No. 9407060101

5-11. Duke Energy Letter Request for Exemption from Certain Requirements Contained in 10 CFR 50.61 and 10 CFR 50, Appendix G, ADAMS Accession No. ML11223A010

5-12. Oconee Nuclear Station, Units 1, 2, and 3, Exemption From the Requirements of 10 CFR PART 50.61 and 10 CFR Part 50, Appendix G (TAC NOS. ME7000, ME7001, ME7002, ME7003, ME7004, AND ME7005), ADAMS Accession No. ML120580196

Table 5-10
Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 72 EFPY – Oconee Unit 1

Reactor Vessel Beltline/Beltline Extended Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT} (°F)	Chemistry Factor	72 EFPY Fluence (n/cm^2)	ΔRT_{PTS} at 72 EFPY (°F)	Margin at 72 EFPY (°F)	RT_{PTS} at 72 EFPY (°F)	Applicable Screening Criteria (°F)	Pass Criteria
Lower Nozzle Belt (LNB) Forging	AHR 54	ZV-2861	A-508, Cl. 2	[ ]	[ ]	[ ]	115.3	2.60E+18	73.1	[ ]	149.9	270	Y
Intermediate Shell (IS) Plates	C2197-2	C2197-2	SA-302 Gr. B Mod.	0.15	0.50	6.7	104.5	1.78E+19	121.0	55.6	183.4	270	Y
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302 Gr. B Mod.	0.10	0.50	21	65.0	2.02E+19	77.5	34.0	132.5	270	Y
US Plate	C3278-1	C3278-1	SA-302 Gr. B Mod.	0.12	0.60	6.7	83.0	2.02E+19	98.9	55.6	161.2	270	Y
Lower Shell (LS) Plate	C2800-1	C2800-1	SA-302 Gr. B Mod.	0.11	0.63	6.7	74.5	2.02E+19	88.8	55.6	151.1	270	Y
LS Plate	C2800-2	C2800-2	SA-302 Gr. B Mod.	0.11	0.63	20	74.5	2.02E+19	88.8	34.0	142.8	270	Y
Outlet Nozzle Forging (ONF) 1	NA	122S316VA2	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.38E+17	[ ]	[ ]	86.1	270	Y
ONF 2	NA	122S316VA1	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.38E+17	[ ]	[ ]	85.9	270	Y
Inlet Nozzle Forging (INF) 1	NA	123S346VA1	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	78.7	270	Y
INF 2	NA	123S346VA2	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	78.8	270	Y
INF 3	NA	124S502VA1	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	78.9	270	Y
INF 4	NA	124S502VA2	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	78.9	270	Y
Transition Forging	NA	122S347VA1	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	2.52E+17	[ ]	[ ]	82.8	270	Y
LNB to ONF Welds	NA	8T1762	Linde 80	[ ]	[ ]	[ ]	[ ]	3.38E+17	[ ]	[ ]	84.2	270	Y
	NA	299L44	Linde 80	[ ]	[ ]	[ ]	[ ]	3.38E+17	[ ]	[ ]	111.2	270	Y
	NA	8T1554B	Linde 80	[ ]	[ ]	[ ]	[ ]	3.38E+17	[ ]	[ ]	80.9	270	Y
LNB to INF Welds	NA	8T1762	Linde 80	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	63.0	270	Y
	NA	299L44	Linde 80	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	78.1	270	Y
	NA	8T1554B	Linde 80	[ ]	[ ]	[ ]	[ ]	1.57E+17	[ ]	[ ]	61.1	270	Y
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	0.23	0.52	-58.5	167.0	2.82E+18	109.3	63.9	114.7	300	Y
IS Long. Welds (Both 100%)	SA-1073	1P0962	Linde 80	0.21	0.64	-48.6	170.6	1.33E+19	184.1	66.6	202.1	270	Y
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	0.23	0.59	-53.5	167.6	1.79E+19	194.4	61.6	202.5	300	Y
US Long. Welds (Both 100%)	SA-1493	8T1762	Linde 80	0.19	0.57	-48.6	167.0	1.31E+19	179.5	66.6	197.5	270	Y
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	0.22	0.54	-72.5	167.0	1.97E+19	197.9	60.9	186.3	300	Y
LS Long. Weld (100%)	SA-1426	8T1762	Linde 80	0.19	0.57	-48.6	167.0	1.62E+19	189.2	66.6	207.2	270	Y
LS Long. Weld (100%)	SA-1430	8T1762	Linde 80	0.19	0.57	-48.6	167.0	1.62E+19	189.2	66.6	207.2	270	Y
LS to Transition Forging Circ. Weld (100%)	WF-9	72445	Linde 80	[ ]	[ ]	[ ]	158.0	2.52E+17	31.3	[ ]	76.7	300	Y

Table 5-11
Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 72 EFPY – Oconee Unit 2

Reactor Vessel Beltline/Beltline Extended Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT} (°F)	Chemistry Factor	72 EFPY Fluence (n/cm^2)	ΔRT_{PTS} at 72 EFPY (°F)	Margin at 72 EFPY (°F)	RT_{PTS} at 72 EFPY (°F)	Applicable Screening Criteria (°F)	Pass Criteria
LNB Forging	AMX 77	123T382	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.67E+19	[ ]	[ ]	155.1	270	Y
US Forging	AAW 163	3P2359	A-508, Cl. 2	0.04	0.75	20	26.0	1.90E+19	30.6	30.6	81.1	270	Y
LS Forging	AWG 164	4P1885	A-508, Cl. 2	0.02	0.80	20	20.0	1.89E+19	23.5	23.5	67.0	270	Y
ONF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.14E+17	[ ]	[ ]	85.2	270	Y
ONF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.14E+17	[ ]	[ ]	85.4	270	Y
INF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	78.2	270	Y
INF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	78.4	270	Y
INF 3	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	78.4	270	Y
INF 4	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	78.3	270	Y
Transition Forging	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	2.34E+17	[ ]	[ ]	82.2	270	Y
LNB to ONF Welds	NA	8T1762	Linde 80	[ ]	[ ]	[ ]	[ ]	3.14E+17	[ ]	[ ]	81.8	270	Y
	NA	72445	Linde 80	[ ]	[ ]	[ ]	[ ]	3.14E+17	[ ]	[ ]	83.9	270	Y
LNB to INF Welds	NA	8T1762	Linde 80	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	61.5	270	Y
	NA	72445	Linde 80	[ ]	[ ]	[ ]	[ ]	1.46E+17	[ ]	[ ]	62.6	270	Y
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	0.27	0.59	-98.0	182.6	1.68E+19	208.7	60.6	171.3	300	Y
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	0.34	0.68	-74.3	220.6	1.85E+19	257.8	61.6	245.1	300	Y
LS to Transition Forging Circ. Weld (100%)	WF-112	406L44	Linde 80	[ ]	[ ]	[ ]	182.6	2.34E+17	[ ]	[ ]	82.1	300	Y

Table 5-12
Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 72 EFPY – Oconee Unit 3

Reactor Vessel Beltline/Beltline Extended Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT} (°F)	Chemistry Factor	72 EFPY Fluence (n/cm^2)	ΔRT_{PTS} at 72 EFPY (°F)	Margin at 72 EFPY (°F)	RT_{PTS} at 72 EFPY (°F)	Applicable Screening Criteria (°F)	Pass Criteria
10 CFR 50.61 (CF Tables)													
LNB Forging	4680	4680	A-508, Cl. 2	0.15	0.91	10	116.1	1.74E+19	133.8	34.0	177.8	270	Y
US Forging	AWS 192	522314	A-508, Cl. 2	0.01	0.73	40	20.0	1.98E+19	23.7	23.7	87.5	270	Y
LS Forging	ANK 191	522194	A-508, Cl. 2	0.02	0.76	40	20.0	1.97E+19	23.7	23.7	87.4	270	Y
ONF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.39E+17	[ ]	[ ]	128.6	270	Y
ONF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	3.39E+17	[ ]	[ ]	128.6	270	Y
INF 1	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	79.0	270	Y
INF 2	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	79.0	270	Y
INF 3	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	78.8	270	Y
INF 4	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	78.8	270	Y
Transition Forging	NA	[ ]	A-508, Cl. 2	[ ]	[ ]	[ ]	[ ]	2.50E+17	[ ]	[ ]	121.9	270	Y
LNB to ONF Welds	NA	72105	Linde 80	[ ]	[ ]	[ ]	[ ]	3.39E+17	[ ]	[ ]	103.4	270	Y
	NA	406L44	Linde 80	[ ]	[ ]	[ ]	[ ]	3.39E+17	[ ]	[ ]	96.5	270	Y
LNB to INF Welds	NA	72105	Linde 80	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	73.9	270	Y
	NA	72102	Linde 80	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	65.0	270	Y
	NA	82102	Linde 80	[ ]	[ ]	[ ]	[ ]	1.58E+17	[ ]	[ ]	91.8	270	Y
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	0.24	0.63	-84.2	178.0	1.75E+19	205.4	59.2	180.4	300	Y
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	0.26	0.60	-33.2	180.0	1.93E+19	212.4	61.1	240.2	300	Y
LS to Transition Forging Circ. Weld (100%)	WF-169-1	8T1554	Linde 80	[ ]	[ ]	[ ]	143.9	2.50E+17	28.4	[ ]	72.0	300	Y
10 CFR 50.61 (Surveillance Data)- Position 2.1													
US Forging	AWS 192	522314	A-508, Cl. 2	0.01	0.73	40	36.0	1.98E+19	42.7	34.0	116.7	270	Y
LS Forging	ANK 191	522194	A-508, Cl. 2	0.02	0.76	40	17.4	1.97E+19	20.6	20.6	81.2	270	Y

6.0 P-T LIMITS

6.1 *Introduction*

10 CFR Part 50, Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the RPV is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must account for the anticipated RPV fluence. The current ONS 54 EFPY P-T limits are based upon fluence projections for 60 years of plant operation (Reference 6-1). Because they were based upon a fluence assumption of 60 years of operation, the P-T limits analyses meet the definition of 10 CFR 54.3(a) and have been identified as time-limited aging analyses (TLAA).

6.2 *Regulatory Guidance for Subsequent License Renewal*

Since the P-T limits will be updated through the 10 CFR 50.90 process at a later, more appropriate date, the effects of aging on the intended function(s) of the RPVs will be adequately managed for the subsequent period of extended operation. The ONS Reactor Vessel Material Surveillance Program (XI.M31), ONS Neutron Fluence Monitoring program (X.M2), and plant Technical Specifications will ensure that updated P-T limits based upon updated ART values will be submitted to the NRC for approval prior to exceeding the period of applicability for Units 1, 2, and 3. The regulatory guidance for NRC review of P-T limit evaluations performed in accordance with 10 CFR 54.21(c)(1)(iii) is reported in NUREG-2192, Section 4.2.3.1.4.3 (Reference 6-2), and is reported below.

"Updated P-T limits for the subsequent period of extended operation must be established and implemented prior to entry into the subsequent period of extended operation. The 10 CFR 50.90 process for P-T limits located in the TS LCOs or the TS Administrative Controls Process for P-T limits that are administratively amended through a PTLR process can be considered adequate AMPs within the scope of 10 CFR 54.21(c)(1)(iii), such that P-T limits will be maintained through the subsequent period of extended operation.

For plants whose P-T limits are controlled by an applicable Administrative Controls TS Section and an NRC-approved PTLR process, the methodologies referenced in the applicable TS Section are reviewed to verify that they will comply with the requirements in 10 CFR Part 50, Appendix G and conform to the recommended position for minimum methodology contents in GL 96-03. Otherwise, the methodology bases for generating updates of the P-T limits during the subsequent period of extended operation are reviewed to determine whether a 10 CFR 54.22-implemented license amendment and TS change of the methodology requirements is necessary for the SLRA."

At present, the Oconee reactor coolant system P-T limits reside within plant Technical Specifications (Reference 6-3), Section 3.4.3, and are applicable to 54 EFPY (60 years). The ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate (MUR), February 19, 2020, ADAMS Accession Number ML20050D379 (Reference 6-4) reduces the applicability of these HU/CD curves as follows.

- Reduces the Applicability for the RCS Heatup and Cooldown limit curves from 54 EFPY to 44.6 Effective Full Power Years (EFPY) for Unit 1, to 45.3 EFPY for Unit 2, and to 43.8 EFPY for Unit 3 based on updated reactor vessel (RV) material evaluations discussed in Section IV.1 of the MUR submittal.

Therefore, the 10 CFR 50.90 process will be used by Duke Energy to update the P-T limits prior to entering the subsequent period of extended operation.

6.3 *P-T Limits at 72 EFPY*

Consistent with the Oconee 60-year license renewal application, NUREG-1723 (Reference 6-5), Section 4.2.4.3.1, Duke Energy prepared 72 EFPY (equivalent to 80 years of operation) P-T limits to demonstrate that the predicted operating window is sufficient to conduct heatups and cooldowns. The 72 EFPY P-T limits were developed with consideration of traditional and extended beltline materials (i.e., locations where RPV neutron fluence > 1.0E+17 n/cm^2, E > 1 MeV) using the analytical methods and flaw acceptance criteria of topical report BAW-10046A, Revision 2 (Reference 6-6) and ASME Code, Section XI, Appendix G (2013 Edition, which permits use of K_{Ic}). Geometric discontinuities at the lower shell to lower transition forging and the nozzle belt forging to intermediate/upper shell (i.e., taper transition regions) were not explicitly modeled for the development of the ONS 72 EFPY P-T limits. However, to support 60-year P-T limits for a B&W-designed plant, detailed 2-D ANSYS finite element models were prepared to evaluate the impact of taper transition on P-T limits. The taper transition region P-T limits were compared to the existing uncorrected 60-year P-T limits and the existing limits were found to be more restrictive. This conclusion is not expected to change for operation to 72 EFPY.

Limiting materials relative to ART at 54 EFPY (MUR submittal maintained the 54 EFPY P-T limit curves and reduced their EFPY applicability) and 72 EFPY are reported in Table 6-1 for Oconee Unit 1, Table 6-2 for Oconee Unit 2, and Table 6-3 for Oconee Unit 3. The 72 EFPY ARTs are higher than the 54 EFPY ARTs, with the exception of the lower shell to transition forging, at all locations for Oconee Unit 1. The 72 EFPY ARTs are higher than the 54 EFPY ARTs at all locations for Oconee Unit 2 with the exception of the lower shell to transition forging weld and nozzle belt forging AMX-77; the reduction in ART for AMX-77 is attributed to a reduction in the chemistry factor owing to an updated and revised copper content of forging AMX-77. The 72 EFPY ARTs are higher than the 54 EFPY ARTs at all locations for Oconee Unit 3 with the exception of the lower shell to transition forging weld and nozzle belt forging 4680. Reduction in ART at 72 EFPY for nozzle belt forging 4680 is attributed to testing that was completed on a lower nozzle belt forging nozzle cutout thus significantly reducing the initial RT_{NDT} and margin term relative to what was reported for 54 EFPY. The 54 EFPY ART reported for the lower shell to transition forging weld of <115°F conservatively includes consideration of the lower shell to transition forging weld, transition forging, and lower head.

Two sets of uncorrected heatup and cooldown 72 EFPY P-T limits are developed to assess the 72 EFPY operating window: one set (Figure 6-1 and Figure 6-2) for ONS Unit 1, and a second set (Figure 6-3 and Figure 6-4) for combined ONS Unit 2 and ONS Unit 3, where the limiting adjusted reference temperatures considering ONS Unit 2 and ONS Unit 3 were utilized. Note that typical P-T limit development includes various RCP start and decay heat removal system (DHRS) initiation combinations. The heatup and cooldown curves reported herein are for illustrative purposes only to demonstrate that plant heatups and cooldowns may be conducted at 72 EFPY.

6.4 *Adjusted Reference Temperatures*

Limiting ARTs for each RPV region are reported in Table 6-1, Table 6-2, and Table 6-3. Adjusted reference temperatures for all traditional beltline and extended beltline locations used to develop the 72 EFPY P-T limits are reported in Table 6-4 for ONS Unit 1, Table 6-5 for ONS Unit 2, and Table 6-6 for ONS Unit 3. Material properties of extended beltline locations and changes to material properties tor traditional beltline locations for SLR are reported in Sections 3.4 and 5.4. The limiting ARTs for ONS Unit 2 and ONS Unit 3 were used to develop combined P-T curves for ONS Unit 2 and ONS Unit 3.

Table 6-1
Limiting Adjusted Reference Temperature (ART) Values for ONS Unit 1 Reactor Vessel

Vessel Item	Wall Location	Limiting RT_{NDT} (°F) at 54 EFPY	Limiting RT_{NDT} (°F) at 72 EFPY
Beltline Base Metal/Axial Weld	¼T	171.0 (SA-1493)	188.1 (SA-1426/1430)
	¾T	132.9 (C2197-2)	139.8 (C2197-2)
Beltline Circ. Weld	¼T	164.2 (SA-1229)	182.6 (SA-1229)
	¾T	132.1 (WF-25)	151.1 (WF-25)
Nozzle Belt (at 12-inch Wall Thickness)	¼T	111.9 (AHR-54)	122.1 (AHR-54)
	¾T	83.5 (AHR-54)	104.9 (AHR-54)
Outlet Nozzle (Postulated Corner Flaw)	(1)	(2)	81.6
Lower Shell to Transition Forging Weld	¼T	< 115 (3)	72.4 (WF-9)
	¾T	< 115 (3)	69.6 (WF-9)

1. Inside wetted surface
2. No shift assumed; initial RT_{NDT} at 60°F per BAW-10046A, Revision 2. Fluence at 54 EFPY < 1.0E+17 n/cm^2 (E > 1.0 MeV)
3. ANP-3127P, Q1-- Response to Second Bullet of RAI-1. Includes consideration of lower shell to transition forging weld, transition forging, and lower head.

Table 6-2
Limiting Adjusted Reference Temperature (ART) Values for ONS Unit 2 Reactor Vessel

Vessel Item	Wall Location	Limiting RT_{NDT} (°F) at 54 EFPY	Limiting RT_{NDT} (°F) at 72 EFPY
Beltline Base Metal/Axial Weld	¼T	161.8 (AMX-77)	147.1 (AMX-77)
	¾T	135.7 (AMX-77)	126.6 (AMX-77)
Beltline Circ. Weld	¼T	193.1 (WF-25)	220.1 (WF-25)
	¾T	132.5 (WF-25)	153.0 (WF-25)
Nozzle Belt (at 12-inch Wall Thickness)	¼T	102.4 (AMX-77)	112.5 (AMX-77)
	¾T	79.4 (AMX-77)	97.5 (AMX-77)
Outlet Nozzle (Postulated Corner Flaw)	(1)	(2)	81.0
Lower Shell to Transition Forging Weld	¼T	< 115 (3)	77.2 (WF-112)
	¾T	< 115 (3)	74.3 (WF-112)

1. Inside wetted surface
2. No shift assumed; initial RT_{NDT} at 60°F per BAW-10046A, Revision 2, Table 3-1. Fluence at 54 EFPY < 1.0E+17 n/cm^2 (E > 1.0 MeV)
3. ANP-3127P, Q1-- Response to Second Bullet of RAI-1. Includes consideration of lower shell to transition forging weld, transition forging, and lower head.

Table 6-3
Limiting Adjusted Reference Temperature (ART) Values for ONS Unit 3 Reactor Vessel

Vessel Item	Wall Location	Limiting RT_{NDT} (°F) at 54 EFPY	Limiting RT_{NDT} (°F) at 72 EFPY
Beltline Base Metal/Axial Weld	¼T	190.8 (4680)	164.1 (4680)
	¾T	160.0 (4680)	129.4 (4680)
Beltline Circ. Weld	¼T	195.6 (WF-67)	219.9 (WF-67)
	¾T	162.1 (WF-70)	183.4 (WF-70)
Nozzle Belt (at 12-inch Wall Thickness)	¼T	106.3 (4680)	106.0 (4680)
	¾T	88.8 (4680)	87.0 (4680)
Outlet Nozzle (Postulated Corner Flaw)	(1)	(2)	118.4
Lower Shell to Transition Forging Weld	¼T	< 115 (3)	68.1 (WF-169-1)
	¾T	< 115 (3)	65.8 (WF-169-1)

1. Inside wetted surface
2. No shift assumed; initial RT_{NDT} at 60°F per BAW-10046A, Revision 2, Table 3-1. Fluence at 54 EFPY < 1.0E+17 n/cm^2 (E > 1.0 MeV)
3. ANP-3127P, Q1-- Response to Second Bullet of RAI-1. Includes consideration of lower shell to transition forging weld, transition forging, and lower head

Table 6-4
Adjusted Reference Temperature (ART) Evaluation for the ONS Unit 1 Reactor Vessel Traditional Beltline and Extended Beltline Materials at 72 EFPY

Table 6-5
Adjusted Reference Temperature (ART) Evaluation for the ONS Unit 2 Reactor Vessel Traditional Beltline and Extended Beltline Materials at 72 EFPY

Table 6-6
Adjusted Reference Temperature (ART) Evaluation for the ONS Unit 3 Reactor Vessel Traditional Beltline and Extended Beltline Materials at 72 EFPY

6.5 *Assessment of 72 EFPY Operating Window*

[

]

For subsequent license renewal, the temperature range where ONS Unit 3 RPV outlet nozzles are predicted to restrict the operation window (i.e., 60 F to 170 F) will have a reduced margin to the NPSH limit. This margin may be regained by removing conservatisms in the 72 EFPY analysis (e.g., revision of heat transfer coefficients to be consistent with RCS flow during 1 and 2 pump operation at low RCS temperatures, reducing the HU/CD rates at low RCS temperatures, and utilizing isothermal conditions to establish LTOP P-T limits) or utilizing provisions permitted by Generic Letter (GL) 88-11.

Figure 6-1
ONS Unit 1 72 EFPY Uncorrected P-T Limits for Normal Heatup with RCP Start at 170°F

Figure 6-2
ONS Unit 1 72 EFPY Uncorrected P-T Limits for Normal Cooldown with DHRS Initiation at 190°F and RCP Stop at 155°F (CD2)

Figure 6-3
ONS Unit 2 and ONS Unit 3 72 EFPY Uncorrected P-T Limits for Normal Heatup with RCP Start at 170°F

[ ]

Figure 6-4
ONS Unit 2 and ONS Unit 3 72 EFPY Uncorrected P-T Limits for Normal Cooldown with DHRS Initiation at 190°F and RCP Stop at 155°F (CD2)

Figure 6-5
ONS Unit 3 Low Range 54 EFPY Cooldown Curves

Figure 6-6
ONS Unit 3 Wide Range 54 EFPY Cooldown Curves

6.6 *References for Section 6.0*

6-1. Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Revised Pressure-Temperature Limits (TAC NOS. MF0763, MF0764, and MF0765), ADAMS Accession Number ML14041A093

6-2. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

6-3. Oconee Nuclear Station Units 1, 2, and 3, Plant Technical Specifications, November 11, 2020

6-4. ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate, February 19, 2020, ADAMS Accession Number ML20050D379 and NRC SER, ADAMS Accession Number ML20335A001

6-5. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3 (ADAMS Accession No. ML003695154)

6-6. BAW-10046A, Revision 2, Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G, ADAMS Accession Number ML20207G642

7.0 FRACTURE MECHANICS EVALUATION OF UNDERCLAD CRACKS

7.1 *Introduction*

Intergranular separations in the heat-affected zone (HAZ) of RPV low-alloy steel under austenitic stainless steel (SS) cladding (i.e., underclad cracking) is a potential TLAA, as indicated in Table 4.7-1 of the subsequent license renewal standard review plan, NUREG-2192 (Reference 7-1). Duke Energy identified and evaluated this TLAA for 60-years in the ONS License Renewal Application, Section 5.4.2.3, wherein reference is made to BAW-2274, which is contained in BAW-2251A (Reference 7-2) as Appendix C. NRC approval of the application of BAW-2251A, Appendix C, to the ONS 60-year LRA is reported in NUREG-1723 (Reference 7-3), Section 4.2.4.4. Underclad cracking was recently evaluated for MUR conditions as reported in the ONS License Amendment Request (Reference 7-4) Section IV.1.C.vii. The MUR evaluation concluded that the generic 48 EFPY evaluation reported in BAW-2251A, Appendix C remained bounding for ONS when considering MUR conditions at 54 EFPY.

7.2 *Regulatory Guidance for Subsequent License Renewal*

The regulatory guidance for NRC review of plant-specific TLAA evaluations performed in accordance with 10 CFR 54.21(c)(1)(ii) is reported in NUREG-2192, Section 4.7.3.1.2 (Reference 7-1), and is reported below.

“The documented results of the revised analyses are reviewed to verify that their period of evaluation is extended such that they are valid for the subsequent period of extended operation. The applicable analysis technique can be the one that is in effect in the plant’s CLB at the time that the subsequent license renewal application (SLRA) is filed.

The applicant may recalculate the TLAA using an 80-year period to show that the acceptance criteria continue to be satisfied for the subsequent period of extended operation. The applicant also may revise the TLAA by recognizing and reevaluating any overly conservative conditions and assumptions. Examples include relaxing overly conservative assumptions in the original analysis, using new or refined analytical techniques, and performing the analysis using an 80-year period. The applicant should provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the subsequent period of extended operation.

As applicable, the plant's code of record is used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

In some cases, the applicant may identify activities to be performed to verify the assumption basis for the calculation (e.g., cycle counting). An evaluation of that activity is provided by the applicant. The reviewer assures that the applicant's verification activities are sufficient to confirm the validity of the calculation assumptions for the subsequent period of extended operation."

7.3 *Description of RV Shell Region with Postulated Underclad Cracks*

The reactor vessel shell is an approximately 14-foot inner diameter, 37-foot high vertical cylindrical shell with a concave lower head (Reference 7-2), Figure 7-1. The closure heads at Oconee Units 1, 2, and 3 have been replaced and are not susceptible to the postulated underclad cracks owing to compliance with Regulatory Guide 1.43. The reactor vessel shell consists of three sub-assemblies:

1. Upper shell assembly (upper shell flange and upper shell forgings, also called the nozzle belt region)
2. Shell assembly (intermediate and lower shell areas)
3. Lower vessel head assembly

For the evaluation of underclad cracks, there are five locations of interest (Figure 7-2).

- Flange
- Nozzle belt
- Shell taper
- Shell
- Transition forging

7.3.1 Geometry

There are three locations without discontinuities: bottom sphere dome, shell, and nozzle belt shell. Thicknesses and radii of the RPV shell at these locations are listed in Table 7-1.

Table 7-1
Reactor Vessel Shell Dimensions

[ ]

7.3.2 Adjusted Reference Temperatures

The adjusted reference temperatures for nil-ductility transition, RT_{NDT}, at locations applicable to axially and circumferentially oriented flaws are listed in Section 5.0 (RT_{PTS} Values at 72 EFPY) for the three Oconee units. The values are listed for thickness t = 0.0 inch, that is for cladding to base metal interface, and projected fluence corresponding to 72 EFPY (80 calendar years). The overall summary of RT_{PTS} temperatures according to the unit and applicable flaw orientation is reported in Table 7-2. The table lists only the bounding values for forgings, plates, and welds.

Table 7-2
Summary of RT_{PTS} Temperatures for 72 EFPY

[ ]

The highest value for axially oriented flaw occurs at Oconee Unit 1: RT_{PTS} = 207.2°F; the highest value for circumferentially oriented flaw occurs at Oconee Unit 2: RT_{PTS} = 245.1°F. [ ]

7.3.3 Material Properties

There are two materials of construction in the reactor vessel shell: (1) low-alloy base metal, and (2) stainless steel cladding. Table 7-3 and Table 7-4 list physical properties for both of these materials. (Note that density 'ρ', thermal conductivity 'TC', and specific heat list only one value. The thermal run is made using one constant value for the entire temperature range. Thermal stresses are calculated using temperature-dependent modulus of elasticity and temperature-dependent coefficient of thermal expansion).

Table 7-3
Properties for Base Metal (¾Ni ½Mo ⅓Cr-V)

Table 7-4
Properties for Cladding (18Cr-8Ni)

7.3.4 Service Loadings

7.3.4.1 Design Basis Transients

The inside surface of the reactor vessel is subjected to transient loads in the form of primary coolant cold leg temperatures and pressures as defined in the RPV ASME Code, Section III, Design Specification. The normal, upset, emergency and faulted condition transient events that are considered in the underclad cracking analysis are listed in Table 7-5 along with number of occurrences over complete projected life span of the plant. Note that Transients [ ] are listed for completeness, but do not contribute to crack growth of the postulated underclad cracks since these transients are describing events occurring either in the pressurizer or in the steam generator, and their representation in the reactor vessel downcomer region is as a steady state condition. These transients do not contribute to the growth of postulated underclad cracks in the reactor vessel and are therefore not considered.

Table 7-5
List of Applicable Design Transients

[ ]

7.3.4.2 Piping Loads

Stresses at the five locations of interest are included due to deadweight, operating basis earthquake (OBE), thermal expansion at steady state conditions, safe shutdown earthquake (SSE), and loss of coolant accident (LOCA) from attached piping to the reactor vessel, such as inlet and outlet nozzles. These stresses include localized effects from the RPV inlet/outlet nozzles as well.

7.3.4.3 Effects of Discontinuities

Discontinuity (interaction) stresses are stresses due to taper-transition regions as they occur in ‘flange top,’ ‘shell taper,’ and ‘transition forging’ location. These stresses were added to the cylindrical shell stresses at locations near structural discontinuities to account for the effects of closure head bolting or impact of taper-transition between two shells of different thicknesses.

7.4 *Fracture Mechanics Analysis*

7.4.1 Methodology

The methodology used to evaluate intergranular separations for the ONS reactor vessels at 80-years is consistent with the methodology reported in the update of BAW-10013 included as Appendix C of BAW-2251A (Reference 7-2). The Oconee-specific analysis was performed for 80-years (72 EFPY and MUR conditions) using current fracture toughness information, applied stress intensity factor solutions, fatigue crack growth correlations for SA-508 Class 2 materials, and is evaluated in accordance with the criteria prescribed in the ASME Code, Section XI, 2013 Edition, IWB-3612 (Reference 7-5).

The underclad cracks are conservatively postulated as surface flaws with an initial flaw depth of 0.353 inches (Figure 7-3, maximum known flaw depth plus the thickness of clad) length of the flaw at 2.12 inches, and aspect ratio at 6:1. Furthermore, the flaws are conservatively postulated at the highly stressed regions of the reactor vessel which includes structural discontinuities. The effect of additional stresses and the applied stress intensity factor due to the higher thermal expansion coefficient of the clad is also considered in the analysis. All the significant normal and upset condition transients with the associated number of cycles given in Table 7-5 are considered in the fatigue crack growth analysis. The final flaw size at the end of 80-years is thereby predicted. The stress intensity factor due to this flaw size is then computed for both the Levels A and B (normal/upset) service level conditions as well as the Levels C and D (emergency/faulted) service level conditions. The applied stress intensity factors are then compared against the ASME Code, Section XI, IWB-3612 standards to ensure that the required safety margins for each of the loading conditions are met.

7.4.1.1 Procedure

Transient through-wall temperatures are calculated using a constant heat transfer (film) coefficient. The value of a constant heat transfer coefficient used is the maximum value occurring during the applicable thermal transients for any given location.

[ ]

7.4.2 Acceptance Criteria

In accordance with the ASME Code, Section XI, IWB-3612 criteria, a flaw is acceptable if the applied stress intensity factor at the final flaw size is less than the available fracture toughness (at the crack tip temperature) with appropriate safety factors, satisfies the following criteria:

1. For Levels A and Level B (Normal and Upset) service conditions:

$$K_I(a_f) < K_{Ic} / \sqrt{10}$$

2. For Levels C and Level D (Emergency and Faulted) service conditions:

$$K_I(a_f) < K_{Ic} / \sqrt{2}$$

Where $K_I(a_f)$ is the maximum applied stress intensity factor at the final flaw size a_f, and K_{Ic} is the fracture toughness (based on fracture initiation) of the material at the corresponding crack tip temperature and irradiation level obtained from Figure A-4200-1 (Reference 7-5).

7.4.3 Fracture Toughness Curve

From Article A-4200 (Section XI, Reference 7-5), the lower bound K_{Ic} fracture toughness for critical crack initiation is expressed as:

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 \cdot (T - RT_{NDT})].$$

where T is the crack tip temperature, RT_{NDT} is the nil-ductility reference temperature of the material, K_{Ic} is in units of ksi√in, and T and RT_{NDT} are in units of °F.

In this present flaw evaluation, K_{Ic} is limited to a maximum value of 200 ksi√in (upper-shelf fracture toughness).

7.4.4 Fatigue Crack Growth Model

Flaw growth due to cyclic loading is calculated using the fatigue crack growth rate model from Reference 7-5, Appendix A, Section A-4300:

$$\frac{da}{dN} = C_o(\Delta K_I)^n ,$$

where ΔK_I is the stress intensity factor range in ksi√in and da/dN is in inches/cycle. For a surface flaw in a water environment,

$\Delta K_I = K_{Imax} - K_{Imin}$

$R_{pwr} = K_{Imin} / K_{Imax}$ ($R_{pwr} = 0$ for $K_{Imin} \leq 0$)

$0 \leq R_{pwr} \leq 0.25$: $\Delta K_I < 17.74$: $n = 5.95$

$C_o = 1.02 \times 10^{-12} \times S$

$S = 1.0$

$\Delta K_I \geq 17.74$: $n = 1.95$

$C_o = 1.01 \times 10^{-7} \times S$

$S = 1.0$

$0.25 < R_{pwr} < 0.65$: $\Delta K_I < 17.74\,[(3.75R_{pwr} + 0.06) / (26.9R_{pwr} - 5.725)]^{0.25}$:

$n = 5.95$

$C_o = 1.02 \times 10^{-12} \times S$

$S = 26.9R_{pwr} - 5.725$

$\Delta K_I \geq 17.74\ [(3.75R_{pwr} + 0.06) / (26.9R_{pwr} - 5.725)]^{0.25}$:

$n = 1.95$

$C_o = 1.01 \times 10^{-7} \times S$

$S = 3.75R_{pwr} + 0.06$

$0.65 \leq R_{pwr} \leq 1.0$: $\Delta K_I < 12.04$: $n = 5.95$

$C_o = 1.02 \times 10^{-12} \times S$

$S = 11.76$

$\Delta K_I \geq 12.04$: $n = 1.95$

$C_o = 1.01 \times 10^{-7} \times S$

$S = 2.5$

The same section also specifies that the following in-air rates must be used if it is greater than the in-water rates specified above.

$R_{air} = K_{Imin} / K_{Imax}$

$n = 3.07$

$C_o = 1.99 \times 10^{-10} \times S$

$0 \leq R_{air} \leq 1$: $S = 25.72 \times (2.88 - R_{air})^{-3.07}$

$\Delta K_I = K_{Imax} - K_{Imin}$

$-2 \leq R_{air} < 0$ and $K_{Imax} - K_{Imin} \leq (0.8) \times 1.12\sigma_f\sqrt{\pi a}$: $S = 1$

$\Delta K_I = K_{Imax}$

$R_{air} < -2$ and $K_{Imax} - K_{Imin} \leq (0.8)\times 1.12\sigma_f\sqrt{\pi a}$: $S = 1$

$\Delta K_I = (1 - R_{air}) \times K_{Imax} / 3$

$R_{air} < 0$ and $K_{Imax} - K_{Imin} > (0.8)\times 1.12\sigma_f\sqrt{\pi a}$: $S = 1$

$\Delta K_I = K_{Imax} - K_{Imin}$

Where σ_f is the flow stress defined by $\sigma_f = ½(\sigma_{ys} + \sigma_{ult})$, where σ_{ys} is the yield strength and σ_{ult} is the ultimate tensile strength calculated at the maximum crack tip temperature between the transient time points where the maximum and minimum stress intensity factor occur. The (*0.8)* reduction factor is established by NRC in 10 CFR 50.55a, Item (xxviii), Section XI condition: Analysis of Flaws, (Reference 7-7).

The fatigue crack growth calculations contain an explicit check to ensure that the maximum crack growth rate is used on the present flaw evaluations.

7.4.5 Calculation of K_I by Polynomial Stress Representation

The Polynomial Stress Representation method is used to calculate K_I due to pressure, pipe loads, and (shell) interaction stresses.

Reference 7-8, the ASME Code, Appendix A, paragraph A-3211 solution for an internal semi-elliptical axial and circumferential surface flaw in a cylindrical vessel is used as a general form of stress intensity factor equation for calculating crack tip stress intensity factor for arbitrary through-wall stress.

$$\sigma = A_0 + A_1\left(\frac{x}{a}\right) + A_2\left(\frac{x}{a}\right)^2 + A_3\left(\frac{x}{a}\right)^3 + A_4\left(\frac{x}{a}\right)^4$$

Where a is the total crack depth of a surface flaw, x is the distance through the wall, where the origin is at the center of the flaw, A_0, A_1, A_2, A_3, and A_4 are fitting constants in units of stress derived from fitting the actual stress distribution at the flaw plane. Coefficients for inside circumferential semi-elliptical flaws at the deepest point ("point 1") are found in Reference 7-8, Tables A-3630-1, 3, 5, and 7, for R_i/t = 1, 5, 10, and 20 respectively. The coefficients for inside axial semi-elliptical flaws at the deepest point ("point 1") are given in Reference 7-8, Tables A-3650-1, 3, 5, and 7, respectively for R_i/t = 1, 5, 10, and 20.

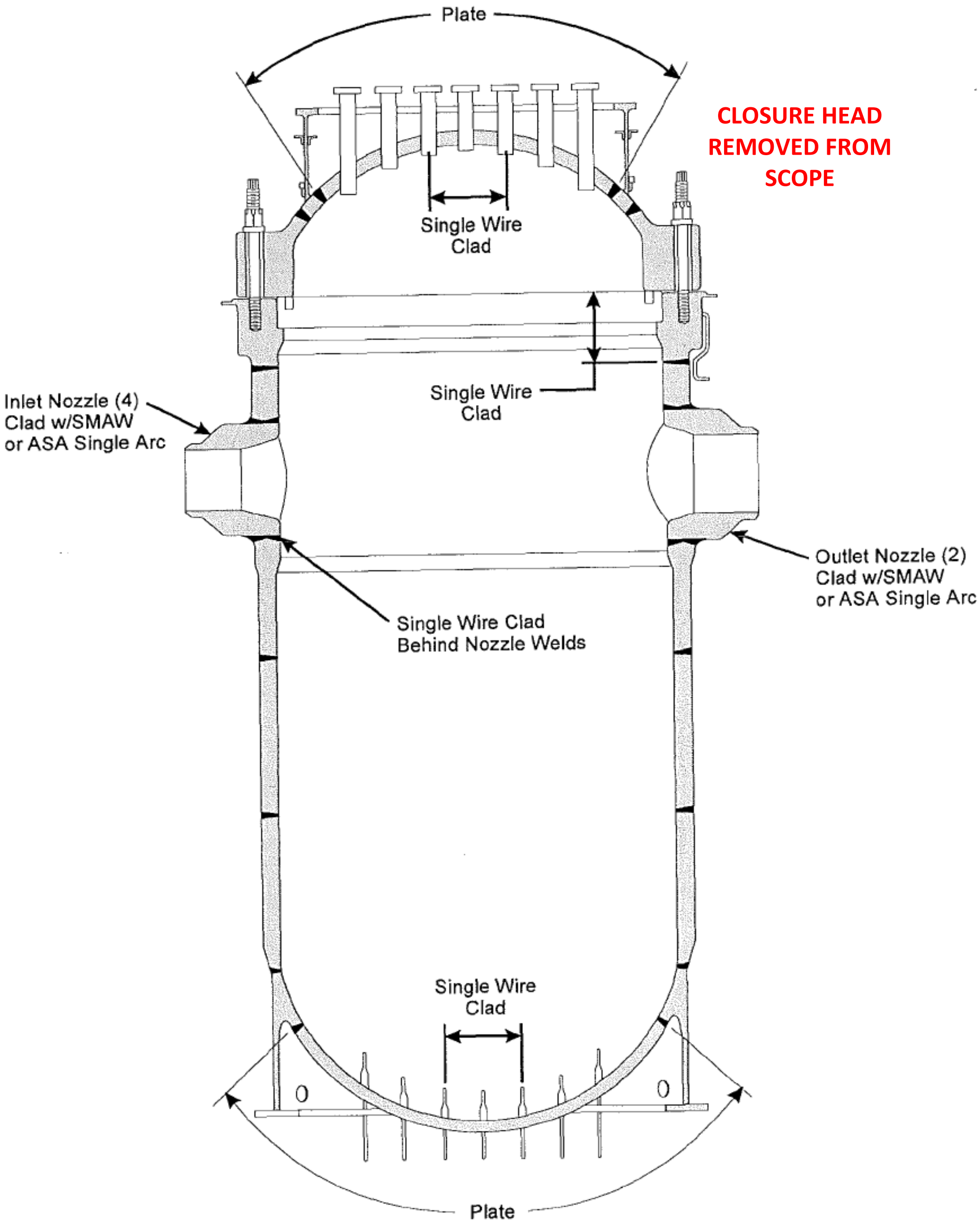
7.4.6 Calculation of K_I by Weight Function

A weight function solution is used to calculate K_I from transient thermal stresses with cladding effect. This solution allows direct calculation of K_I for both the axial and circumferential orientations of the flaw. Weight function method solution is selected over 'cladding stress extrapolation' solutions due to its superior ability to represent non-linear stress profiles.

7.4.7 Irwin Plastic Zone Correction

The final crack size endpoint of the fatigue crack growth is adjusted by the plastic zone correction calculated as: $a_e = a + \frac{1}{6\pi} \cdot \left(\frac{K_I(a)}{\sigma_y}\right)^2$, Reference 7-9, where σ_y is material yield strength. Conservatively use the value of σ_y @ 700°F rounded down (see Table 7-3), i.e., σ_y = 40 ksi, and the final stress intensity factor is calculated as: $K_I(a_e) = K_I(a)\sqrt{\frac{a_e}{a}}$.

Figure 7-1
Regions with Clad Forging Material



Note: All inside surfaces six-wire clad except as noted
All material ASTM A508, Class 2 except plate,
as noted.

Figure 7-2
Reactor Vessel Shell High Stress Locations

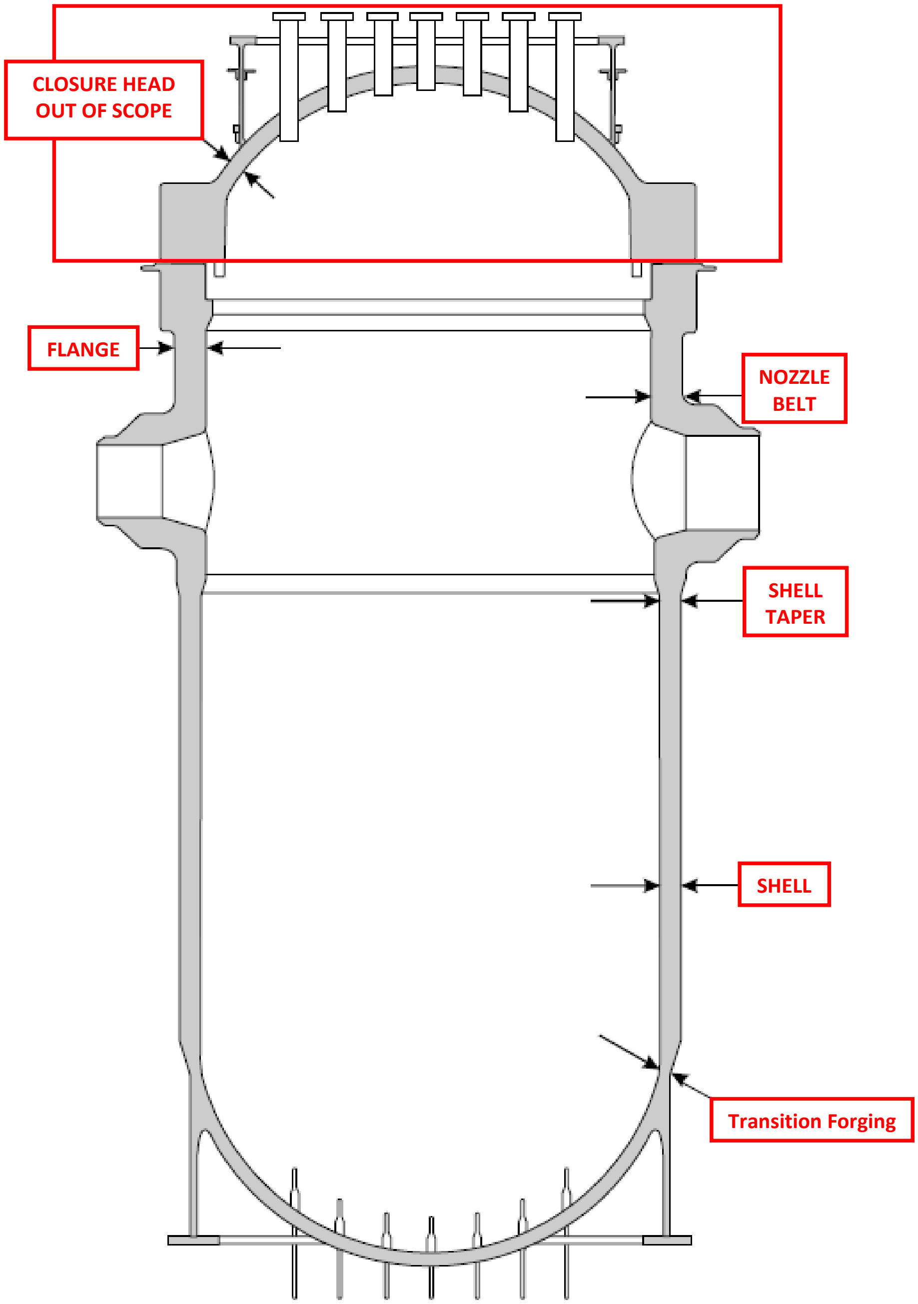
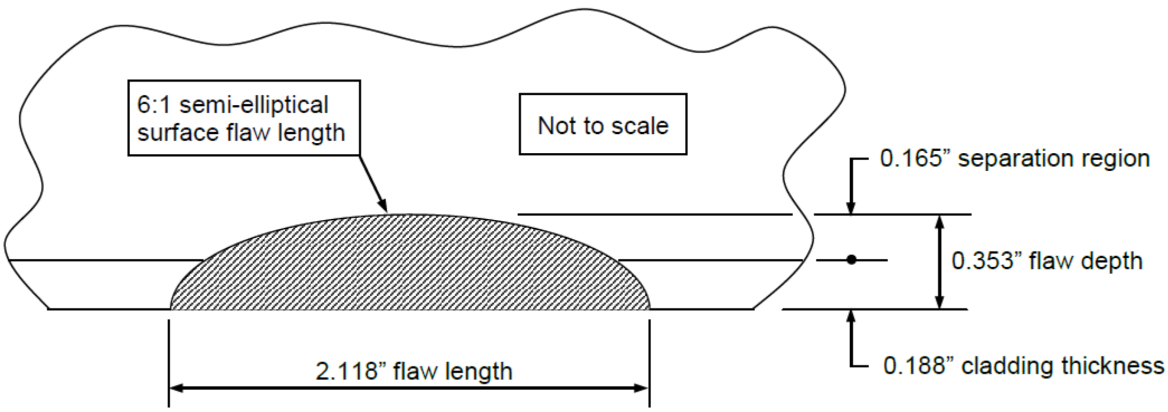


Figure 7-3
Postulated Surface Flaw Dimensions



7.4.8 Evaluation Results

Table 7-6 reports the final crack sizes at each of the five locations along with the calculated safety margin that accounts for plastic zone correction. It was demonstrated that at all the investigated locations, postulated flaws in both the axial and circumferential orientations meet the acceptance criteria of Article IWB-3612 (Reference 7-5) for Levels A and Level B (normal/upset), and Levels C and Level D (emergency/faulted) loading conditions after accounting for plastic zone correction.

The lowest fracture toughness margin for Levels A and B (normal/upset) loading conditions is 3.94 and occurs at the flange top axially oriented flaw, which is higher than the minimum required margin of 3.16 ($\sqrt{10}$ = 3.16). The lowest fracture toughness margin for Levels C and D (emergency/faulted) loading condition is 1.62 and occurs at the shell taper due to a circumferentially oriented flaw, which is greater than the minimum required margin of 1.41 ($\sqrt{2}$ = 1.41). Therefore, all the postulated flaws for the underclad cracking in the reactor vessel for all the three Oconee Units remain acceptable at the end of the 80-year service evaluation period.

Table 7-6
Results Summary

7.5 *References for Section 7.0*

7-1. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

7-2. BAW-2251A, Demonstration of the Management of Aging Effects for the Reactor Vessel, ADAMS Accession Numbers ML20212G894 and ML20212G911

7-3. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3 (ADAMS Accession No. ML003695154)

7-4. ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate, February 19, 2020, ADAMS Accession Number ML20050D379 and NRC SER, ADAMS Accession Number ML20335A001

7-5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2013 Edition

7-6. **[**

]

7-7. Code of Federal Regulations, Title 10, Part 50.55a, Domestic Licensing of Production and Utilization Facilities, Codes and Standards, 85 FR 26576, Jun. 3, 2020; 85 FR 34088, Jun. 3, 2020

7-8. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 2019 Edition

7-9. T.L. Anderson, Fracture Mechanics – Fundamentals and Applications, Taylor and Francis, 3rd Edition

8.0 REACTOR VESSEL ENVIRONMENTALLY-ASSISTED FATIGUE

8.1 *Introduction*

As outlined in Section X.M1 of NUREG-2191 (Reference 8-1) and Section 4.3 of NUREG-2192 (Reference 8-2), the effects of the reactor water environment on fatigue cumulative usage factors (CUFs) must be examined for a set of sample critical components for the plant. This sample set includes the locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Reference 8-3) and additional plant-specific locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260, Table 6-1. Any additional limiting locations are identified through an environmentally-assisted fatigue (EAF) screening evaluation. The EAF screening process evaluates existing CLB fatigue usage values for the ASME Code, Section III components, including the NUREG/CR-6260 locations, to determine the lead indicator (also referred to as sentinel) locations for EAF.

The current ONS analysis of record for reactor vessel EAF NUREG/CR-6260 locations is reported in BAW-2251A (Reference 8-4) and NUREG-1723 (Reference 8-5, Page 4-16). Review of non-NUREG/CR-6260 locations was not required for 60-years by the NRC for the ONS in accordance with NUREG-1723.

The purpose of this section is to summarize the environmentally-assisted fatigue usage factors for the Oconee RPVs for 80 years. The EAF evaluation is based on the F_{en} methodology provided in NUREG/CR-6909, Revision 1 (Reference 8-6), for the following Reactor Pressure Vessel (RPV) NUREG/CR-6260 locations at Oconee Nuclear Station (ONS) Units 1, 2, and 3:

1. Reactor vessel shell and lower head
2. Reactor vessel inlet and outlet nozzles
3. ICI nozzle to lower head weld
4. Reactor vessel core flood nozzle

8.2 *Regulatory Guidance for Subsequent License Renewal*

The regulatory guidance for NRC review of components evaluated for CUF_{en} TLAA evaluations performed in accordance with 10 CFR 54.21(c)(1)(ii) is reported in NUREG-2192, Section 4.3.3.1.2.2 (Reference 8-2), and is reported below.

"The operating cyclic load experience and a list of the assumed transients used in the existing fatigue parameter calculations is reviewed for the current operating term to ensure that the number of assumed occurrences for each transient are projected to the end of the subsequent period of extended operation. The reviewer verifies that a comparison of the operating cyclic load severity to the severity assumed in the existing fatigue parameter calculations for each transient has been made to demonstrate that the cyclic load severities used in the fatigue parameter calculations remain bounding. In addition, the reviewer verifies that a comparison of the water chemistry conditions to those assumed in the F_{en} calculations has been made to demonstrate that the water chemistry conditions used in the F_{en} calculations are appropriate. For consistency purposes, the review also includes an assessment of the TLAA information against relevant design basis information and CLB information. The review includes verification that the applicant has updated the CUF_{en} calculations for the applicable NUREG/CR–6260 or more limiting component locations using the methods of analysis in either RG 1.207, Revision 1, NUREG/CR–6909, Revision 0 (with "average temperature" used consistent with the clarification that was added to NUREG/CR–6909, Revision 1); or other subsequent NRC-endorsed alternatives.

The Code of Record should be used for the reevaluation, or the applicant may update to a later Code edition pursuant to 10 CFR 50.55a using an appropriate Code reconciliation. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met."

8.3 *Methodology*

The following methodology was used for the determination of the EAF usage factors (CUF_{en}).

1. Retrieve the bounding in-air fatigue usage factor ($CUF_{in\text{-}air}$) for each location of interest from the current license basis (CLB) ASME Code, Section III design reports for Oconee Units 1, 2, and 3.

2. Calculate the maximum environmentally-assisted fatigue correction factor (F_{en}) for each location of interest using NUREG/CR-6909, Rev. 1 (Reference 8-6).

3. For locations constructed of low alloy steel and featuring relatively low $CUF_{in\text{-}air}$ magnitudes, CUF_{en} (= $CUF_{in\text{-}air}$ x F_{en}).

4. For locations featuring high fatigue usage factor due to high level of conservatism, recalculate the in-air fatigue usage factor ($CUF_{in\text{-}air}$) using appropriate stress ranges and number of operating transient cycles using NUREG/CR-6909, Rev. 1, fatigue curves. Locations featuring high fatigue usage factor due to high level of conservatism are: inlet nozzle, inlet nozzle safe end (inlet nozzles are buttered with carbon steel at their terminal ends), outlet nozzle, outlet nozzle safe end (outlet nozzles are buttered with carbon steel at their terminal ends), ICI J-weld, core flood nozzle, core flood nozzle safe end, and venturi.

5. Calculate EAF usage factor based on recalculated $CUF_{in\text{-}air}$ per step 4, CUF_{en} (= $CUF_{in\text{-}air}$ x F_{en}).

8.4 *Assumptions*

1. In NUREG/CR-6909 Rev. 1, Reference 8-6, when calculating T* in Appendix A, the upper bound limit of equation for T is 325°C (617°F). For the purposes of calculating a F_{en}, this temperature is considered bounding for locations exposed to LWR coolant.

2. Since the strain rate is unknown for stress intensity ranges, the slowest strain rate used in Reference 8-6 is considered for conservatism.

3. Sulfur content of 0.015 wt.% is assumed. This value is taken from Reference 8-6 as bounding value.

4. Low alloy steel material with cladding is conservatively assumed to be exposed to the LWR coolant environment.

5. The in-air cumulative fatigue usage factors extracted from the ASME Code, Section III design report summary documents and used in the calculation may be of locations not exposed to the RCS fluid as detailed location of fatigue usages reported are not readily available. However, since it is the maximum usage factor for that component, for conservatism, it is used herein.

6. The number of NSSS design transients defined in the RPV ASME Code, Section III, Design Specifications, which are applicable to 60-years, are not revised for 80-years.

8.5 *Summary of Results*

Table 8-1 includes a summary of the results of the environmentally-assisted fatigue usage factors for the RPV NUREG/CR-6260 items. The final CUF_{en} is based on the original design number of operating transient cycles. Note that the operating transient cycles are not adjusted or prorated for the 80 years plant operation from the current license basis.

The maximum CUF_{en} is 10.01 for the core flood venturi; however, the core flood venturi is not a pressure retaining item and was not evaluated in NUREG/CR-6260. The CUF_{en} values for the remaining locations are less than 1.0 and therefore acceptable.

Table 8-1
CUF_{en} Results Summary

Component	Material	Bounding F_{en}	$CUF_{in\text{-}air}$	CUF_{en}
Lower Head	SA533-Gr2, Code Case 1339	[ ]	[ ]	0.756
Nozzle Belt	A-508-64 Cl.2	[ ]	[ ]	0.107
Shell	A-508-64 Cl.2	[ ]	[ ]	0.624
Inlet Nozzle	A-508-64 Cl.2	[ ]	[ ]	0.832
Inlet Nozzle Safe End	SA-106 Gr.C	[ ]	[ ]	0.832
Outlet Nozzle	A-508-64 Cl.2	[ ]	[ ]	0.832
Outlet Nozzle Safe End	SA-106 Gr.C	[ ]	[ ]	0.832
ICI-J-Weld	SB-166	[ ]	[ ]	0.744
Core Flood Nozzle ID	A-508-64 Cl.2	[ ]	[ ]	0.882
Core Flood Safe End[1]	A-336-F8m	[ ]	[ ]	0.525
Core Flood Venturi	SA-376 TP304	[ ]	[ ]	10.010

Notes:

1. NUREG/CR-6260, Page 5-41, reported that the CUF for the safe end was essentially zero and was not evaluated in NUREG/CR-6260.

8.6 *References for Section 8.0*

8-1. NUREG-2191, Volume 2, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report

8-2. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

8-3. NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curve to Selected Nuclear Power Plant Components,” March 1995

8-4. BAW-2251A, Demonstration of the Management of Aging Effects for the Reactor Vessel, ADAMS Accession Numbers ML20212G894 and ML20212G911

8-5. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, ADAMS Accession Number ML003695154

8-6. NUREG/CR-6909, “Effect of LWR Water Environments on the Fatigue Life of Reactor Materials,” Rev. 1, May 2018 Final Report

9.0 IRRADIATION EMBRITTLEMENT OF ONS RPV SUPPORTS FOR SLR

9.1 *Introduction*

The purpose of this section is to evaluate the ONS RPV supports relative to the draft NRC interim staff “further evaluation” draft guidance reported in Reference 9-1 with respect to determination of the applicability of reduction of fracture toughness for subsequent license renewal. The evaluation reported herein is consistent with the guidance reported in NUREG-1509 (Reference 9-2).

9.2 *Regulatory Guidance for Subsequent License Renewal*

In accordance with NUREG-2192, Standard Review Plan for Subsequent License Renewal (Reference 9-3), aging management review recommendations for reactor vessel (RV) supports are addressed in Section 3.5.2.2.2.6, and Table 3.5-1, Item 097. Following review of the first subsequent license renewal applications for Turkey Point 3 and 4, Surry 1 and 2, and Peach Bottom, the NRC determined that additional guidance was needed to clarify aging management review expectations by the NRC for Reactor Pressure Vessel (RPV) supports for subsequent license renewal. As such, the NRC has issued draft interim staff guidance (Reference 9-1) to revise the SLR Standard Review Plan NUREG-2192 to add a new Section 3.5.3.2.2.7 that includes the following review guidance.

“Further evaluation is recommended of a plant-specific program (or plant-specific enhancements to selected GALL-SLR AMPs) to manage reduction of fracture toughness due to irradiation embrittlement from accumulated neutron fluence and gamma dose, which could occur in BWR and PWR steel structural support components (including associated weldments and bolted connections) located in the vicinity of the Reactor Vessel (RV) during the subsequent period of extended operation. These components include the RV steel supports, neutron shield tank, steel structural support components of reactor shield wall and sacrificial shield wall, or other steel structural support components located in the vicinity of the RV. Further, loss of function due to radiation exposure (neutron and/or gamma) for non-steel, non-concrete materials (e.g., Lubrite® or other lubricant/coating in support sliding feet) that are used in these structural support components should also be evaluated and dispositioned, with supporting technical information, on a plant-specific basis for the subsequent period of extended operation. If a plant-specific program or program enhancements are determined to be necessary, the reviewer confirms that the acceptance criteria for AMP program elements described in BTP RLSB-1 {Appendix A.1 of NUREG-2192 (SRP-SLR)}. Otherwise, the reviewer confirms the adequacy of the justification provided for the aging effects not requiring management such that intended function(s) are maintained consistent with the CLB for the subsequent period of extended operation.

The applicant may address irradiation aging effects by analysis, testing, aging management inspections (e.g., one-time, periodic), or a combination of these methods; the reviewer confirms the evaluation is supported by an adequate technical basis that also accounts for uncertainties and limitations of available technical data and knowledge. NUREG-1509, “Radiation Effects on Reactor Pressure Vessel Supports,” (Ref. 21[1]) provides general guidance and an acceptable approach (with exception of structural consequence analysis approach) to evaluate loss of fracture toughness via a screening evaluation, a fracture mechanics analysis or a transition temperature analysis (relative to the lowest operating temperature). A plant-specific AMP or plant-specific enhancements to selected GALL-SLR AMPs may not be necessary for the steel structural support components if (i) the screening criteria or criteria for reevaluation in Chapter 4, “RPV Support Reevaluation Criteria,” of NUREG-1509 (with exception of the structural consequence analysis approach in Section 4.5) are satisfied on a plant-specific basis using the technical evaluation procedures described therein; (note that adequacy of technical justification provided for other applicant-proposed demonstrated technical evaluation approach used will be reviewed on a case-by-case basis) and (ii) there is no plant-specific operating experience of irradiation embrittlement observed to date. The methodology in Nonmandatory Appendix A of the ASME Code, Section XI may be used for analytical evaluation of postulated flaws in a fracture mechanics analysis. For related non-steel, non-concrete structural support components, the applicant should provide supporting technical information and data to justify its determination regarding the need for a plant-specific program to manage irradiation aging effects; the reviewer evaluates this on a case-by-case basis.

[1] Reference 21 is per NUREG-2192

It should be noted that the conclusions in NUREG-1509 were based on analyses and limited to neutron fluences relevant to 40 years of operation. Since the neutron fluences for the subsequent period of extended operation to 80 years often exceed the neutron fluence values used in the analyses of NUREG-1509, the analytical methodologies (except the structural consequence analysis approach in Section 4.5) in NUREG-1509 remain applicable but the conclusions are no longer supported.

The reviewer confirms that the applicant's technical evaluation estimated the neutron and gamma fluence incident on the RV steel supports and other applicable steel structural support components for the subsequent period of extended operation and evaluated the susceptibility of components to loss of fracture toughness. The evaluation should demonstrate that the RV steel supports and other steel structural support components noted above will remain capable of performing their intended function consistent with the current licensing basis through the subsequent period of extended operation. The reviewer confirms that the evaluation included sufficient description that demonstrate that appropriate methodologies and conservative assumptions were implemented to estimate the levels of neutron fluence and gamma dose for the subsequent period of extended operation. The damage parameter "displacements per atom (dpa)" should include neutrons of all energies (high and low) rather than only those with E > 1 MeV (i.e., embrittlement predictions should include damage from the entire neutron energy spectrum based on E > 0.1 MeV). Alloying elements, such as copper, can increase the rate of radiation embrittlement. Therefore, the evaluation should take into consideration the material properties and chemical composition of the steel (e.g., initial nil ductility temperature (NDT), type of steel, copper content, weld material) and the lowest service (operating) temperature to which the components are exposed.

A structural integrity evaluation of the RV steel supports and other steel structural support components should include all design basis loading combinations referenced in the current licensing basis (CLB). It is essential that the evaluation account for: (1) plant specific operating experience of RV steel supports and other structural components degradation due irradiation embrittlement and other susceptible aging effects; (2) the current as-found physical condition of the supports and structural components; and (3) account for the effects of observed signs of degradation (including but not limited to corrosion, cracks, or permanent deformation) and potential future degradations reasonably projected to the end of the subsequent period of extended operation (i.e., 80 years). This should typically be based on a detailed physical examination (to the extent possible) of the RV supports that is documented in an inspection report that serves as the basis for the evaluation and decisions regarding further actions. Where a detailed physical examination is not feasible, adequately justified assumptions of potential degradations through the end of the subsequent period of extended operation (i.e., 80 years) should be made, or appropriate monitoring actions proposed to manage the susceptible aging effects consistent with the assumptions made in the evaluation."

9.3 *Description of RPV Support and Evaluation Methodology*

The RPV support assembly and embedment detail are illustrated in Figure 9-1 and Figure 9-2, and consist of a support skirt, a support flange, anchor bolts and associated washers and hex nuts, sole plate, vertical bearing plate and associated nelson studs, grout, and reinforced concrete pedestal that contains the embedded anchor bolts. For the discussion that follows the RPV support assembly is defined as the RPV support skirt and the RPV support flange, which were attached to the reactor vessel by Babcock & Wilcox during fabrication of the reactor vessel. The RPV embedment includes the anchor bolts and associated washers and hex nuts, sole plate, vertical bearing plate and associated nelson studs, grout, and reinforced concrete pedestal that contain the embedded anchor bolts, and these items were supplied by the Architect Engineer.

The support skirt is fabricated from two semi-circular carbon steel (SA-516 Grade 70) rolled plates welded together longitudinally to form a 2-inch thick, 60-inch high, 175.5-inch ID cylinder. The top of the support skirt cylinder is welded to the bottom of the reactor vessel transition forging. Twelve 9.25-inch diameter holes and twelve 2-inch diameter holes are included in the support skirt for ventilation of the reactor vessel cavity, emergency flooding of the cavity, and steam venting in the event of flooding. The support skirt cylinder is welded at the bottom to the support flange, which is constructed from four 90-degree carbon steel (SA-515 Grade 70) segments welded together to form a 3.5-inch high cylinder (ID of 161.5 inches and OD of 193.5 inches).

The carbon steel support flange contains forty-eight 3-inch diameter holes, equally spaced at 7 degrees 30 minutes, on both the inside (on a diameter of 167.5 inches) and outside (on a diameter of 187.5 inches) of the support flange to accept 96 high strength alloy steel (A490) anchor bolts. Each anchor bolt is 2-inches in diameter and 6 ft 10-inches in length with the top of each anchor bolt threaded and extending approximately 7-inches above the sole plate. Each anchor bolt is secured to the embedment by one heavy hex nut, one jamb nut, one standard size hardened washer, and a 2 1/8-inch ID by 4 1/8-inch OD by 1-inch thick plate washer. In addition, the support flange contains forty-eight 1.5 inch diameter holes (on a diameter of 187.5 inches) for shear pins (A490). The shear pin holes are equally spaced and located azimuthally midway between adjacent anchor bolts; forty-two 7-inch long shear pins are inserted into the 1.5-inch diameter holes in the support flange and extend into the underlying sole plate (since the sole plate is formed by six sub-sections, only 42 shear pins are installed). The anchor bolts are specified to be prestressed (approximately 80 ksi each) such that lift off of the support flange will not occur during a postulated operating basis earthquake.

Evaluation Methodology

The evaluation process for the RPV support structural steel is documented in NUREG-1509 (Reference 9-2), and is illustrated through flow charts (i.e., Figures 4-2 and 4-4 of NUREG-1509), reproduced as Figure 9-3 and Figure 9-4 below. The initial step for the evaluation involves assessment of the existing condition of the RPV support at the time of re-evaluation, comparison with the initial construction condition, and the degree of degradation predicted by the end of plant life. In addition, a review of the original design and safety margin is performed. This review includes the original design methodology, load combinations for which the supports were designed, allowable stresses and their margins with respect to the actual stresses in the members, and codes governing the original design. To confirm that there is adequate RPV support fracture resistance, the assessment is based on a transition temperature analysis (Figure 9-4) wherein it is sufficient to demonstrate that there is an adequate margin between the minimum operating temperature (i.e., lowest service temperature) and the nil-ductility temperature (NDT) at 72 EFPY.

Based on the configuration of the RPV support assembly and embedment detail described above, the following items directly support the RPV support intended function, i.e., to provide structural support for the reactor vessel, and are evaluated for susceptibility to reduction of fracture toughness by irradiation embrittlement.

- RPV support skirt (SA-516 Grade 70)
- RPV support flange (SA-515 Grade 70)
- Anchor bolts (A490)
- Anchor bolt jamb nuts, hex nuts, and washers (assumed to be equivalent to A490)

The embedment vertical bearing plate and associated nelson studs do not support the RPV support assembly intended function and are not evaluated for susceptibility to reduction of fracture toughness by irradiation embrittlement herein. NUREG-2192, Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation, includes the following guidance relative to grout and concrete. "Data related to the effects and significance of neutron and gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. However, based on literature review of existing research, radiation fluence limits of 1×10^{19} neutrons/cm^2 neutron radiation (fluence cutoff energy E > 0.1 MeV) and 1×10^{8} Gy (1×10^{10} rad) gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly." 72 EFPY fluence at the ONS RPV support embedment is estimated at 1.63E+18 n/cm^2 (E > 0.1 MeV) and gamma dose at 1.75E+09 rad. As such, the embedment concrete and grout are not susceptible to irradiation embrittlement.

Figure 9-1
Reactor Pressure Vessel Support Assembly

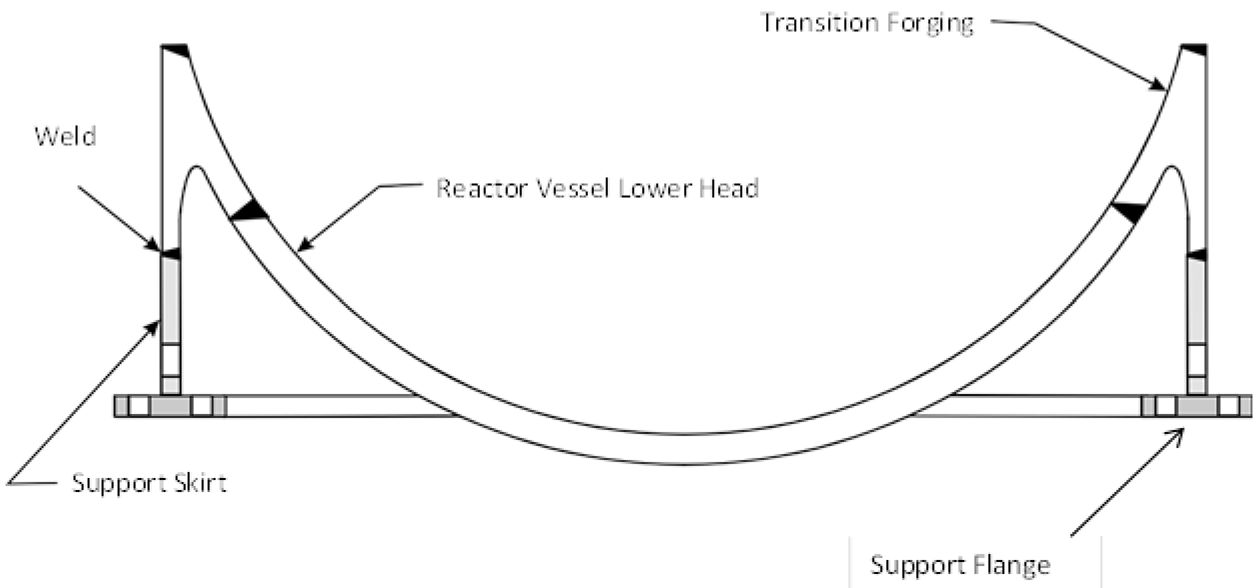


Figure 9-2
Reactor Pressure Vessel Support Embedment Detail

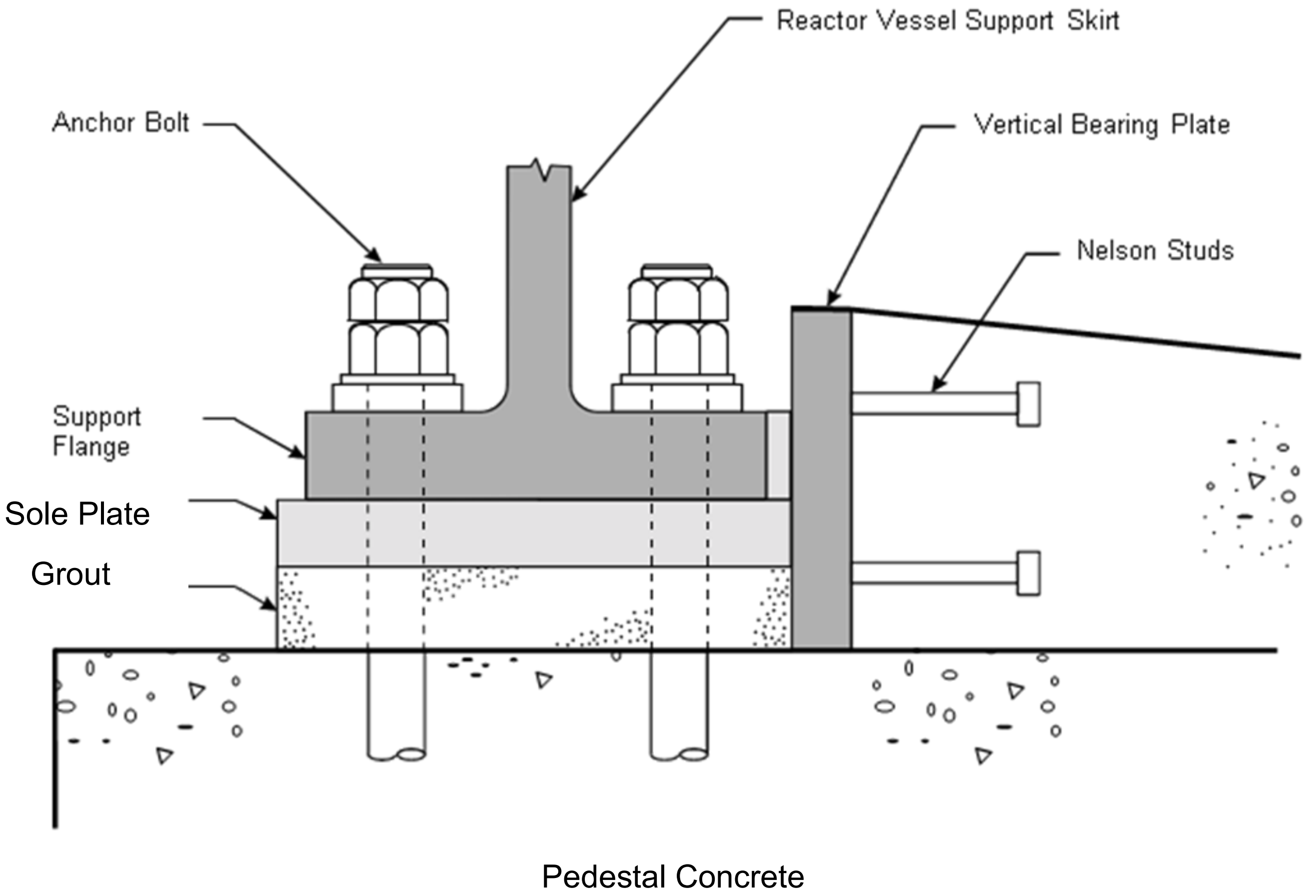


Figure 9-3
NUREG-1509 Figure 4-2, Preliminary Evaluation

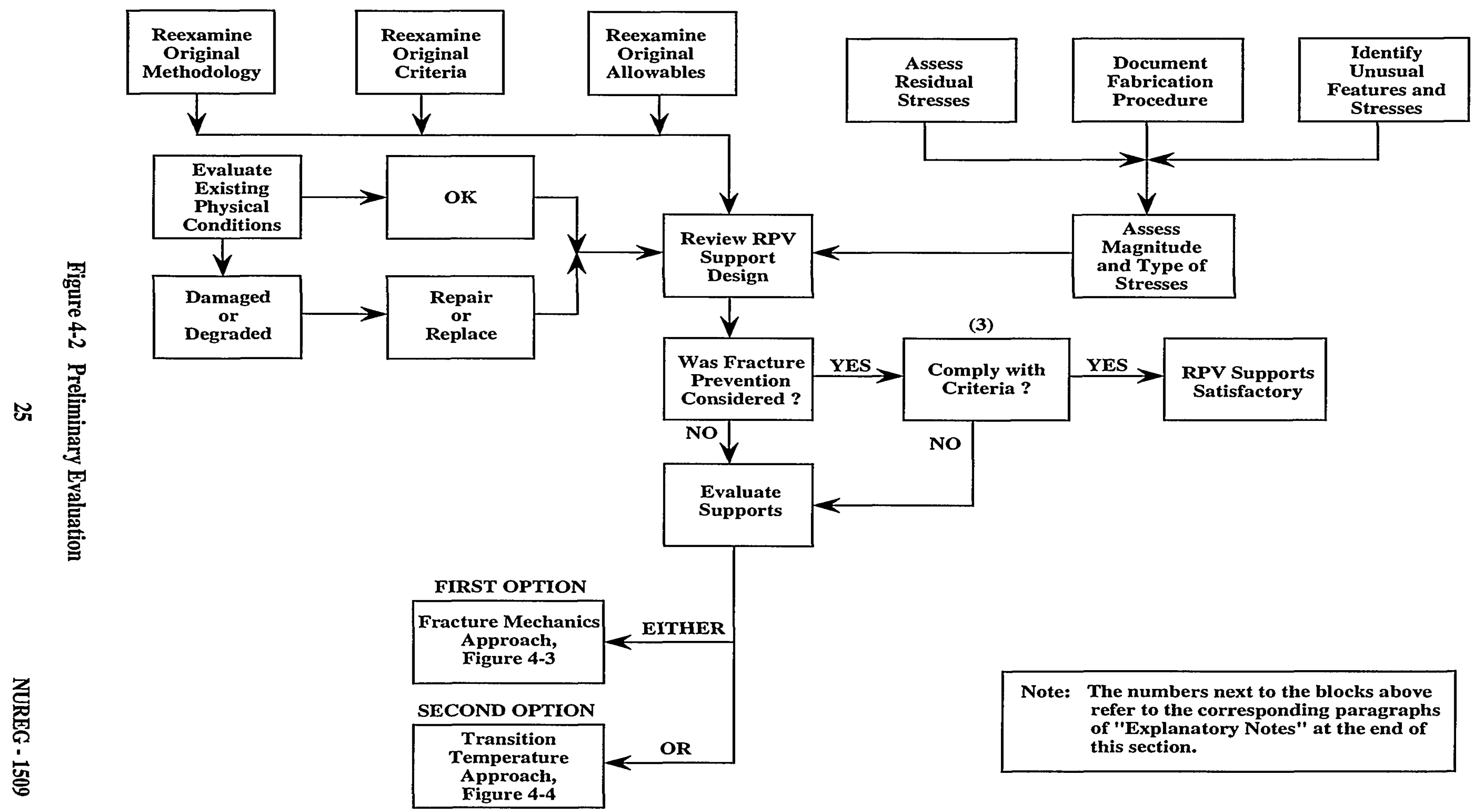


Figure 9-4
NUREG-1509 Figure 4-4, Transition Temperature Approach

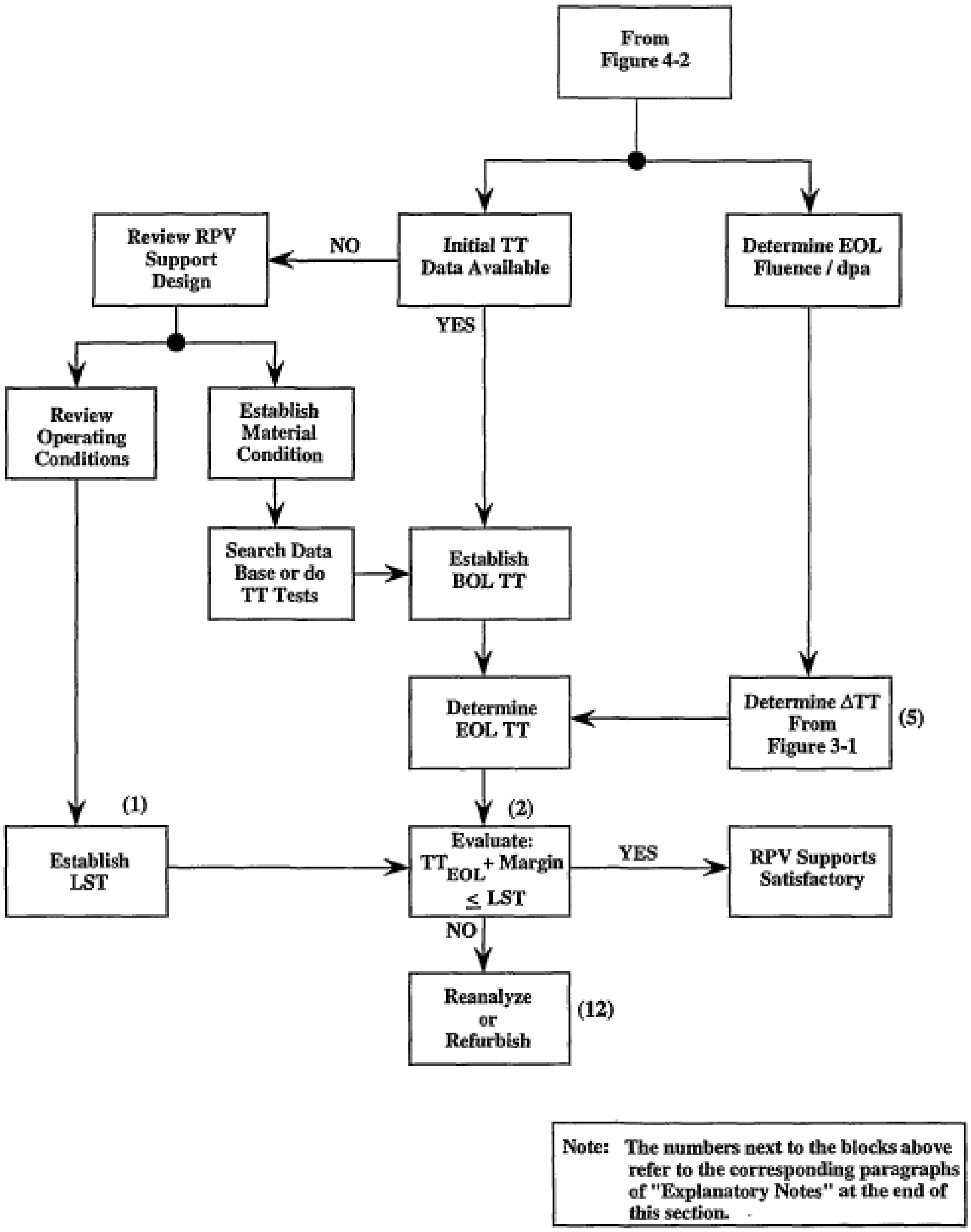


Figure 4-4 Transition Temperature Approach

9.4 *Irradiation Embrittlement Further Evaluation*

9.4.1 Assessment of Current Condition

Assessment of the existing condition of the ONS Units 1, 2, and 3 RPV supports includes a mandatory, visual physical condition inspection of the vital parts of the supports. Rust, cracks, or permanent deformation of any part of the RPV supports should be noted as evidence that some distress has been sustained.

Visual inspections of the ONS RPV supports were performed in 2012 (ONS Unit 1), 2013 (ONS Unit 2), and 2014 (ONS Unit 3). For each ONS unit, a VT-3 visual examination was performed on accessible surfaces of the RPV support with a calculated coverage of approximately 66.5% of the support surface areas within the examination boundary. The VT-3 visual examinations were performed using personnel, equipment, and procedures qualified in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 1998 Edition with the 2000 Addenda, and there were no unacceptable conditions or indications detected during these examinations. The ONS Unit 1 examination record indicates that the inspection was rejected; this was due to limited coverage that is discussed below.

These examinations were considered limited because the ASME Code, Section XI inspection required a VT-3 visual examination of 100% of the RV support. Based on this limited inspection, Duke Energy submitted Relief Request 15-ON-004 to the NRC requesting relief from the ASME Code, Section XI examination requirements for the RPV supports at ONS. After review of Relief Request 15-ON-004, the NRC staff, through a Safety Evaluation, granted Duke Energy relief for the limited examination of the ONS RPV supports, and further concluded that the examinations performed to the extent practical provide reasonable assurance of structural integrity of the RPV support.

9.4.2 Design Stress Summary

The design stresses for the ONS RPV supports (ONS Unit 1, ONS Unit 2, and ONS Unit 3) were developed in accordance with the ASME Code, 1965 Edition with Addenda through Summer 1967. Stress intensities were calculated for the following load cases and locations.

Normal/Upset Condition Loads

1. Primary Stresses – Dead Weight (DW) + Operational Basis Earthquake (OBE) + Pressure:
    - Support Skirt: Maximum Calculated Stress Intensity = [ ] Location – towards bottom of skirt at 9-1/4″ hole section
    - Support Flange Bending Stress = [ ]
    - Support Flange Shear Stress = [ ]
    - Shear Pins Stress = [ ] (includes thermal)
2. Primary + Secondary Stresses – DW + OBE + Pressure + Thermal transients:
    - Support Skirt: Stress Intensity Range = [ ]
    - Location – bottom of skirt just above flange

Faulted Condition Loads:

1. Primary Stresses – DW + SSE + LOCA + Pressure:
    - Support Skirt: Maximum Calculated Stress Intensity = [ ]
    - Location – towards bottom of skirt at 9.25″ hole section
    - Bolts: Maximum Calculated Shear Stress [ ]

- Shear Pins: Maximum Calculated Shear Stress [

]

Notes:

(1) The faulted stress analyses looked at the bolts as the weak link such that if the bolts were acceptable, the flange was acceptable.

(2) The faulted stresses above consider Large Break LOCAs. Considering Leak Before Break (LBB) along with OTSG replacement, the loads (thus stresses) are less but the component stresses with regards to ranking are expected to be the same.

(3) Definitions:

- DW: Deadweight
- OBE: Operating Basis Earthquake
- SSE Safe Shutdown Earthquake
- LOCA: Loss of Coolant Accident

[

]

9.4.2.1 Load Reduction Considering LBB

When the steam generators were replaced at the Oconee Units, the RCS loop model was reanalyzed considering LBB of the primary piping (hot and cold leg piping). With regard to the RPV support, load comparisons were performed. The load ratios for the faulted condition loading of DW + SSE + LOCA are reported in Table 9-1.

Table 9-1
RPV Support Load Comparison: LBB Loads / Original Design Loads

[ ]

[

]

9.4.3 Lowest Service Temperature

The lowest service temperature (LST), as defined in the NUREG-1509, Page 40, is estimated for the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support assembly and embedment. The lowest service temperature (LST) is defined as the minimum temperature of the most vulnerable part of the fracture-critical member when design-basis accident loads occur. RPV support temperatures can be established either from measurements or theoretical calculations.

For the ONS, design basis accidents (DBAs) are addressed in Chapter 15 of the ONS updated final safety analysis report (UFSAR). Based on a review of the ONS Chapter 15 DBAs, lowest service temperature may be defined by calculating the temperature distribution in the RPV support skirt at 100% steady state conditions. The B&W-designed reactors operate at a constant average temperature between approximately 15% power and 100% power; thus 100% power represents a condition with the lowest cold leg and RV downcomer temperatures prior to the postulated DBA.

A 2-D model of the lower RPV, RPV skirt and support flange, and anchoring and supporting structure was prepared. This model is based on an existing 3-D solids model of the Oconee RPV and internals. The resulting 2-D model is supplied to the ANSYS Mechanical application. A solution mesh is generated, thermal boundary conditions (loads) are applied, and a steady-state thermal solution is calculated. Two different temperature contours of the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support skirt assembly were developed using models based on two different locations (i.e., 2-D cross sections); the anchor bolt location, Figure 9-5, and the shear pin location, Figure 9-6. The RPV support assembly temperatures, which are estimated using the anchor bolt location model, are used for conservatism. Based on the anchor bolt location model temperature contours, the minimum LST for the RPV support assembly and embedment is 139.05°F (59.5°C) at the location of the anchor bolts, and is conservatively applied for all the RPV support assembly materials in this assessment.

Figure 9-5
RPV Skirt Temperatures from Anchor Bolts Location Model

Figure 9-6
RPV Skirt Temperatures from Shear Pin Location Model

9.4.4 Establish EOL NDT Temperature

9.4.4.1 DPA of RPV Support Skirt, Flange, Anchor Bolts, and Shear Pins

The projected 72 EFPY dpa of the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support skirt weld and embedment anchor bolts is conservatively estimated at **5.53E-04 dpa**. This projected 72 EFPY dpa is obtained by calculating dpa at the bottom of a RG 1.190 compliant RPV DORT model (BAW-2241P-A, Revision 2, Reference 9-5), which is approximately 17.49 inches above the transition forging to RPV skirt weld, and then by generating RPV skirt weld flux and dpa rate profiles on the outer RPV surface in the air cavity region and extrapolating dpa from the bottom of the DORT model to the RPV skirt to transition forging weld. The dpa of 5.53E-04 is conservatively assumed to be applicable to all the RPV support assembly and embedment materials (i.e., skirt, flange, anchor bolts, nuts, washers, and shear pins).

Neutron fluence and gamma dose at 80-years (72 EFPY) are calculated using source terms that bound all Oconee units to provide bounding estimates for RPV support and the biological shield wall. [

]

9.4.4.2 RPV Support Assembly Materials

The ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support skirt base metal materials (SA-516 Grade 70) are classified as carbon-manganese steels and the RPV support flange materials (SA-515 Grade 70) are classified as plain carbon steels in accordance with NUREG-1509, Table 4-2, Classification of Wrought Grades into Groups. The anchor bolts were furnished in the field as ASTM A490 steel; the shear pins are also manufactured per ASTM A490. Materials of construction for the jamb nut and hex nut are assumed to be the same as the anchor bolts with respect to material properties. The ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support assembly welds are produced using either the manual metal arc process or semi-automatic gas shielded metal arc process. Table 9-2 lists the material descriptions for the RPV skirt/flange base metals and Table 9-3 lists the RPV skirt/flange associated welds for ONS Unit 1, ONS Unit 2, and ONS Unit 3.

Table 9-2
ONS RPV Support Skirt Assembly Base Metal Materials

[ ]

Table 9-3
ONS RPV Support Skirt Assembly Weld Materials

9.4.4.3 Radiation Induced NDT Shift

In accordance with NUREG-1509, Section 4.3.4.2, when using the transition temperature approach to evaluate the RPV support integrity, the NDT temperature at end-of-life should include the irradiation-induced shift (ΔNDTT) and account for uncertainties related to the NDT determination. Therefore, the NDT temperature at end-of-life (ART) is expressed by the following equation:

$$ART = Initial\ NDT + \Delta NDTT + Margin$$

To determine the dpa at which the RPV support skirt assembly materials will equal the estimated LST, the above equation can be rearranged as follows:

$$\Delta NDTT = ART - (Initial\ NDT + Margin)$$

where ART = LST (see Section 9.4.3 above).

The ΔNDTT values for the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support assembly and embedment materials are reported in Table 9-4 and Table 9-5.

Based on the calculated ΔNDTT for the RPV support assembly materials, a dpa is estimated from Figure 3-1, upper bound curve, of NUREG-1509 for each ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support assembly material. These estimated dpa values along with the projected 72 EFPY dpa are illustrated in Table 9-6 and Table 9-7 for comparison. Based on this comparison, the projected 72 EFPY dpa for the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support flange and the ONS Unit 1 and ONS Unit 2 RPV support flange welds are greater than the estimated dpa for the minimum LST of the RPV support skirt assembly, indicating the expected ART values for these materials will be greater than the LST at 72 EFPY of operation. These items require further evaluation to demonstrate that the intended function(s) of the RPV support assembly will be maintained consistent with the CLB during the Subsequent Period of Extended Operation (SPEO) when considering irradiation embrittlement.

Table 9-4
Calculated ΔNDTT Values for the ONS RPV Support Skirt Assembly Base Metal Materials

[ ]

Table 9-5
Calculated ΔNDTT Values for the ONS RPV Support Skirt Assembly Weld Materials

[ ]

Table 9-6
Comparison of the ΔNDTT dpa at ONS RPV Support Skirt Assembly Base Metal LST to Projected 72 EFPY dpa

[ ]

Table 9-7
Comparison of the ΔNDTT dpa at ONS RPV Support Skirt Assembly Weld Metal LST to Projected 72 EFPY dpa

[ ]

9.4.5 Comparison of EOL NDT to LST for Critical RPV Support Locations

In accordance with Table 9-6 and Table 9-7, the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support flange and the ONS Unit 1 and ONS Unit 2 RPV support flange welds connecting the 90 degree segments together to form a circular support plate are potentially susceptible to reduction of fracture toughness by irradiation embrittlement at 72 EFPY of operation. All remaining RPV support assembly and embedment items that support the RPV support intended function are not susceptible to irradiation embrittlement.

The design stresses for the ONS RPV support flange and its associated welds indicate the normal/upset loads show low bending and shear stresses in the support flange compared to their respective allowables, and that the RPV support skirt is more highly stressed. The faulted stresses show that the RPV support skirt is the highly stressed item with little margin. The RPV support flange was not explicitly addressed for faulted loads as the bolts were determined to be the weak link and the bolts were shown acceptable with margin. The faulted loads considered were large break primary piping breaks. With consideration of LBB of the primary piping, the loads for the faulted condition have been reduced considerably in the RPV support skirt, flange, and bolts.

The support flange is bolted to the concrete with 48 bolts spaced equally around the outside of the flange, and 48 bolts spaced equally around the inside part of the flange;

[

]

Based on consideration of the potential for irradiation embrittlement of the RPV support assembly and embedment items, only the ONS Unit 1, ONS Unit 2, and ONS Unit 3 RPV support flange and the ONS Unit 1 and ONS Unit 2 RPV support flange welds are potentially susceptible to irradiation embrittlement. [

]

As discussed in Section 9.4.2.1, load reduction considering LBB is approximately one-quarter of the original design loads with the exception of Fy. Consideration of LBB would significantly reduce the faulted stress intensities in the RPV support skirt and support flange and would likely significantly increase the margin to the allowable stress intensities.

Since the RPV support skirt is the critical location of the RPV support assembly and is not susceptible to irradiation embrittlement based on the NDT evaluation reported above, the RPV support intended function will be maintained consistent with the CLB during the SPEO when considering damage due to irradiation.

9.5 *References for Section 9.0*

9-1. RV Supports ISG SLR Document Changes: Add FE Section 3.5.2.2.2.7 and AMR Items for Irradiation Embrittlement of Reactor Vessel (RV) Steel Supports and Other Steel Structural Support Components near RV, ADAMS Accession Number ML20049H359

9-2. U.S. Nuclear Regulatory Commission, NUREG-1509, Radiation Effects on Reactor Pressure Vessel Supports, May 1996

9-3. NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

9-4. Framatome Inc. Topical Report BAW-1621, Effects of Asymmetric LOCA Loads, Phase II Analysis, ADAMS Accession Number ML19320B058

9-5. Framatome Inc. Topical Report BAW-2241P-A, Revision 2, “Fluence and Uncertainty Methodologies,” ADAMS Accession No. ML073310655 (Proprietary), ML073310660 (Non-Proprietary)

10.0 REACTOR COOLANT PUMPS THERMAL EMBRITTLEMENT (CASS)

10.1 *Introduction*

The Oconee Nuclear Station (ONS) reactor coolant pumps (RCPs) were fabricated by Sulzer Bingham, 28 by 28 by 41 Type RQV (ONS Units 2 and 3) and Westinghouse, Type 93A (ONS Unit 1). Pressure retaining RCP items that are subject to aging management review, fabricated from cast austenitic stainless steel (CASS), and potentially subject to reduction of fracture of fracture toughness by thermal embrittlement include the following: Unit 1 – main flange and casing, Units 2 and 3 – cover/stuffing box and casing (Figure 10-1 and Figure 10-2). RCP pressure boundary items that are fabricated from cast austenitic steel may be susceptible to thermal embrittlement if the ferrite content of the castings exceeds the screening criteria established by the NRC in NUREG-2191, Section XI.M12, Evaluation and Technical Basis (Reference 10-1) as supplemented by the final interim staff guidance for mechanical, SLR-ISG-2021-02-MECHANICAL (Reference 10-2).

Figure 10-1
Sulzer Bingham Reactor Coolant Pump

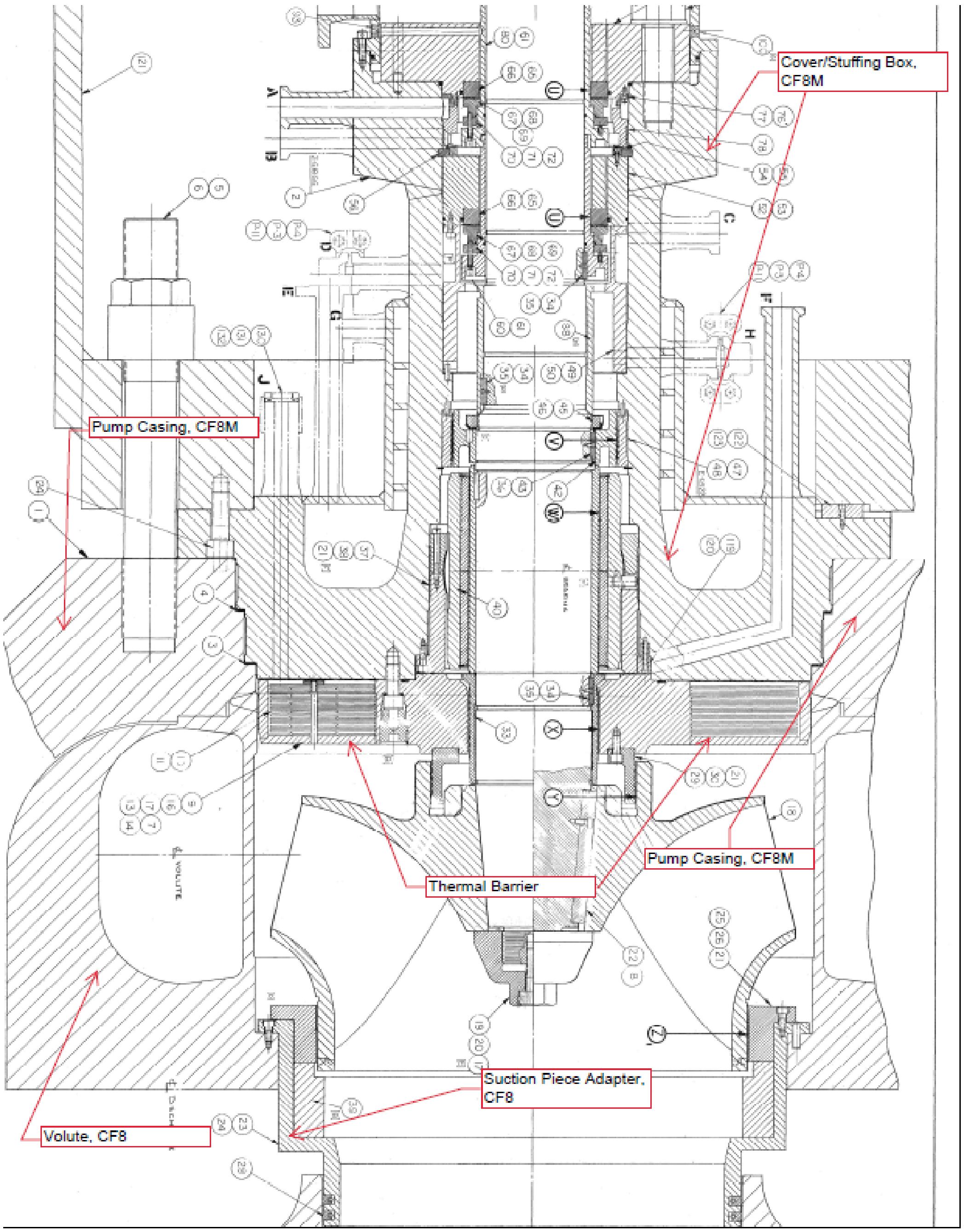
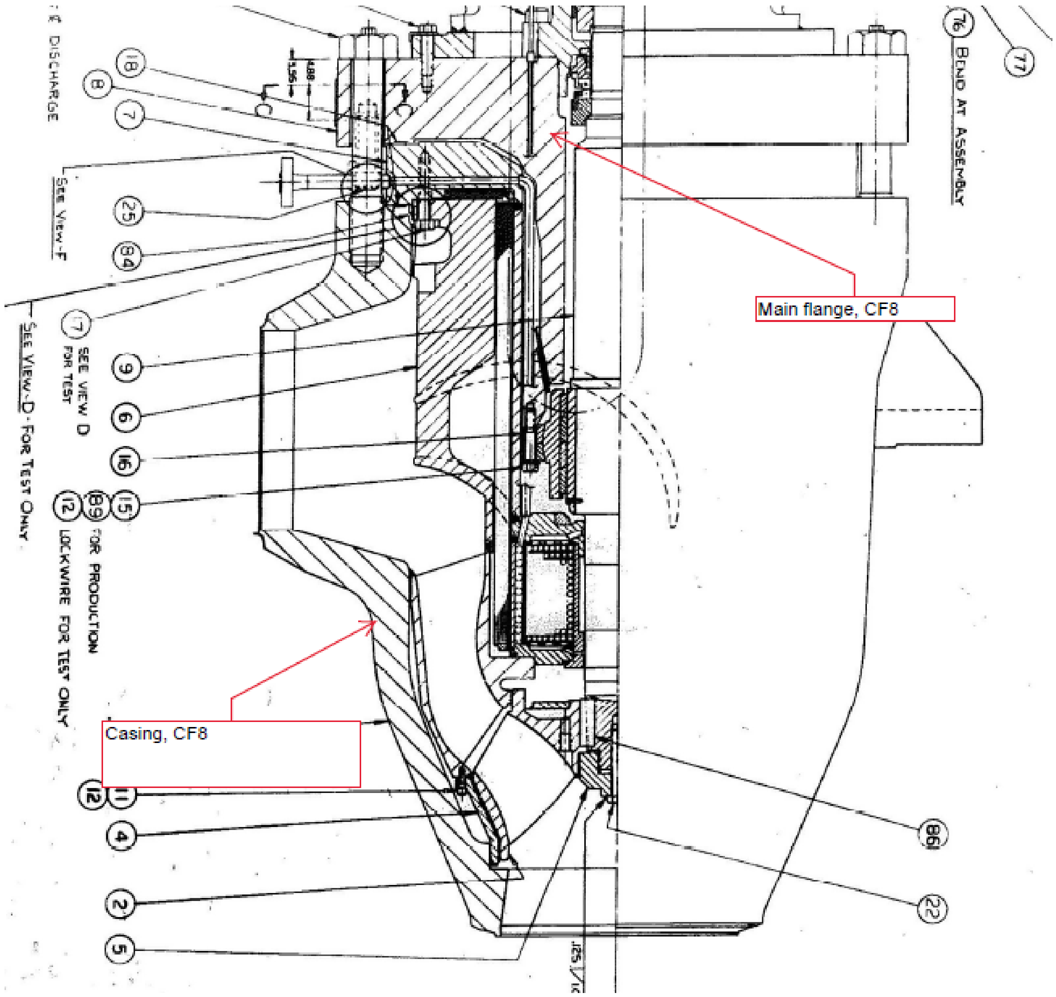


Figure 10-2
Westinghouse Type 93 A Reactor Coolant Pump



10.2 *Regulatory Guidance for SLR*

Regulatory guidance for NRC review of thermal embrittlement of reactor coolant system components fabricated from cast austenitic stainless steel is contained in NUREG-2191, Volume 2 (Reference 10-1), aging management program XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) as supplemented by the final interim staff guidance for mechanical, SLR-ISG-2021-02-MECHANICAL (Reference 10-2). The XI.M12 program description is as follows.

"The reactor coolant system components are inspected in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping components except for valve bodies. This aging management program (AMP) includes determination of the potential significance of thermal aging embrittlement of CASS components based on casting method, molybdenum content, and percent ferrite. For components for which thermal aging embrittlement is "potentially significant" as defined below, aging management is accomplished through either (a) qualified visual inspections, such as enhanced visual examination (EVT-1); (b) a qualified ultrasonic testing (UT) methodology; or (c) a component-specific flaw tolerance evaluation in accordance with the ASME Code, Section XI. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components for which thermal aging embrittlement is not significant. The scope of the program includes ASME Code Class 1 piping components constructed from CASS with service conditions above 250 °C (482 °F).

For pump casings, as an alternative to the screening and other actions described above, no further actions are needed if applicants demonstrate that the original flaw tolerance evaluation performed as part of Code Case N-481 implementation remains bounding and applicable for the SLR period or the evaluation is revised to be applicable for 80 years. For valve bodies, based on the results of the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, U.S. Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (May 19, 2000 NRC letter), screening for significance of thermal aging embrittlement is not required. The existing ASME Code, Section XI inspection requirements are adequate for valve bodies."

RCP items that are fabricated from cast austenitic steel may be susceptible to thermal embrittlement if the ferrite content of the castings exceeds the screening criteria established by the NRC in NUREG-2191, Section XI.M12, Evaluation and Technical Basis, as repeated below.

Based on the criteria set forth in the May 19, 2000, NRC letter, the potential significance of thermal aging embrittlement of CASS materials is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A or other steels with ≤ 0.5 weight percent (wt.%) Mo), only static-cast steels with >20 percent ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20 percent ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content steels (SA-351 Grades CF3M, CF3MA, and CF8M or other steels with 2.0 to 3.0 wt.% Mo), static-cast steels with >14 percent ferrite and centrifugal-cast steels with >20 percent ferrite thermal embrittlement can be potentially significant (i.e., screens in). For static-cast high-molybdenum steels with ≤14 percent ferrite and centrifugal-cast high-molybdenum steels with ≤20 percent ferrite, thermal aging embrittlement is not significant (i.e., screens out). The thermal embrittlement screening criteria of CASS with different molybdenum and ferrite contents are summarized in NUREG-2191, Table XI.M12-1, reproduced as Table 10-1 below.

In the significance screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Revision 1) or a staff-approved method for calculating delta ferrite in CASS materials. A fracture toughness value of 255 kilo-joules per square meter (kJ/m^2) (1,450 inch-pounds per square inch) at a crack extension of 2.5 millimeters (0.1 inch) is used to differentiate between CASS materials for which thermal aging embrittlement is not significant and those for which thermal aging embrittlement is potentially significant. Extensive research data indicate that for CASS materials without the potential for significant thermal aging embrittlement, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Revision 1).

The final interim staff guidance (Reference 10-2) provides clarification regarding acceptance criteria for a flaw tolerance approach. NUREG-2191, XI.M12, specifies that flaws detected in CASS components are evaluated in accordance with the applicable procedures of the ASME Code, Section XI. The most recent version of the ASME Code, Section XI incorporated by reference in 10 CFR 50.55a (2013 Edition), does not contain evaluation procedures applicable to CASS with ferrite content ≥ 20 percent. The final ISG permits use of the 2019 Edition of ASME Code, Section XI, Appendix C, wherein the flaw evaluation procedures in the 2019 Edition of the Code were developed by considering the ferrite content, fracture toughness, tensile data of CASS materials, and the relevant elastic-plastic correction factors (Z-factors) as a function of ferrite content.

Table 10-1
Thermal Embrittlement Susceptibility from NUREG-2191, Table XI.M12-1

Table XI.M12-1. Thermal Embrittlement Susceptibility				
Molybdenum (Mo) Content	**Fe Content**	**Casting Method**	**Potentially Susceptible (Screens In)**	**Not Susceptible (Screens Out)**
Low or ≤ 0.5 wt.%	>20% ferrite	Static	X	—
Low or ≤ 0.5 wt.%	≤20% ferrite	Static	—	X
Low or ≤ 0.5 wt.%	Any	Centrifugal	—	X
High or 2.0-3.0 wt.%	>14% ferrite	Static	X	—
High or 2.0-3.0 wt.%	>20% ferrite	Centrifugal	X	—
High or 2.0-3.0 wt.%	≤14% ferrite	Static	—	X
High or 2.0-3.0 wt.%	≤20% ferrite	Centrifugal	—	X

10.3 *Identification of RCP Items Subject to Aging Management Review*

RCP items subject to aging management review are reported in the ONS 60-year license renewal application (Reference 10-3), Section 3.4.8, and include the following: Casing, Cover, and Pressure-Retaining Bolting. Reference 10-3, Table 3.4-1, identifies that the casing and cover are fabricated from CASS with an applicable aging effect of reduction of fracture toughness, and aging management in accordance with the ASME Code, Section XI, In-service Inspection, as supplemented by the CASS Flaw Evaluation Procedure reported in Section 4.18.2.1.

Oconee reactor coolant pump quality assurance data packages, which contain material certification reports for pressure retaining items, were used to confirm the materials of construction for RCP CASS items to obtain material chemistry data required to calculate ferrite content using Hull's Equivalent Factors. In some instances, non-pressure retaining items (e.g., volute and suction piece adapter for ONS Unit 2 and ONS Unit 3) were included for conservatism.

The following Oconee RCP pressure retaining items are subject to aging management review and made from CASS.

- ONS Unit 1—main flange and pump casing (CF8), Figure 10-2.
- ONS Units 2 and 3—cover/stuffing box and pump casing (CF8M), Figure 10-1. Non-pressure retaining parts volute and suction piece adapter are made from CF8 and are shown for more information only.

10.4 *Calculation of Ferrite Numbers for CASS Parts*

Ferrite numbers are calculated for the RCP pressure boundary items. Ferrite % is calculated using Hull's Equivalent factors and heat-specific chemistry as follows (Reference 10-4, Equations (3), (4), and (5)). There is no indication that niobium was used in any of the casting heats.

$$Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) – 4.99$$

$$Ni_{eq} = (Ni) + 0.11(Mn) – 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77,$$

where the concentrations of the various alloying and interstitial elements are given in wt.%. The concentration of N is often not available in a CMTR; if not known, it is assumed to be 0.04 wt % (Reference 10-4). The maximum molybdenum content for CF8 material is assumed to be 0.5% in accordance with ASTM-A351. The ferrite content δc is given by

$$\delta c = 100.3(Cr_{eq}/ Ni_{eq})^2 – 170.72(Cr_{eq}/ Ni_{eq}) + 74.22$$

Results of RCP items (CF8, CF8M) that exceed the ferrite screening criteria listed in Section 10.2 are summarized in Table 10-2.

Table 10-2
ONS RCP Items Susceptible to Thermal Embrittlement Based on Ferrite %

ONS Unit 1		
Not susceptible to thermal embrittlement-all casings and main flanges are made from CF8 and ferrite % of each item is below 20%.		
ONS Unit 2		
Items potentially susceptible to thermal embrittlement include selected CF8M casings (upper and lower) and selected stuffing boxes (CF8M) based on ferrite percentages. Volute and suction piece adapters are made from CF8, ferrite % are all below 20%, and are not susceptible to thermal embrittlement. In all instances, static castings are assumed.		
RCP Item	**Heat Number**	**Ferrite % >14%**
RCP Casings (lower and upper)-CF8M		
2RC-P1A1-Serial Number 1	Lower Half-Heat 17368-1	14.46
2RC-P1A2-Serial Number 5	Upper Half-Heat 19427-1	17.55
2RC-P1A2-Serial Number 5	Lower Half-Heat 20683	18.59
2RC-P1B2-Serial Number 3	Upper Half-Heat 20723	16.46
Stuffing Boxes-CF8M		
	Heat 20011-1	14.72
	Heat-19880-2	16.98
	Heat 16858-3	16.45
	Heat 19969-1	19.57
	Heat 19880-3	17.33
ONS Unit 3		
Items potentially susceptible to thermal embrittlement include CF8M casings (upper and lower) and selected stuffing boxes based on ferrite percentages. Volute and suction piece adapters are made from CF8, ferrite % are all below 20%, and are not susceptible to thermal embrittlement. In all instances, static castings are assumed.		
RCP Item	**Heat Number**	**Ferrite %>14%**
RCP Casings (lower and upper)-CF8M		
3RC-P3A1-Serial Number 8	Lower Half-Heat 21795	14.20
3RC-P3A1-Serial Number 8	Upper Half-Heat 24245-1	15.96
3RC-P3A2-Serial Number 10	Upper Half-Heat 13011	19.94
3RC-P3B1-Serial Number 6	Upper Half-Heat 22270-1	15.65
3RC-P3B1-Serial Number 6	Lower Half-Heat 20910-1	15.38
3RC-P3B2-Serial Number 7	Lower Half-Heat 24685	20.06
Stuffing Boxes-CF8M		
	Heat 15384-2	17.25
	Heat 16496-4	14.86
	Heat 17434	16.46
	Heat 19969-3	20.32
	Heat 17278-2	16.93

ONS Unit 1 CF8 castings are not susceptible to thermal embrittlement since all casings and main flanges are made from CF8 and ferrite % of each item is below 20%. For ONS Unit 2 and ONS Unit 3, items potentially susceptible to thermal embrittlement include CF8M casings (upper and lower) and selected stuffing boxes based on ferrite percentages. Volute and suction piece adapters are made from CF8, ferrite % are all below 20%, and are not susceptible to thermal embrittlement. In all instances, static castings are assumed.

10.5 *Temperatures of ONS Unit 2 and ONS Unit 3 Stuffing Boxes During Normal Operation*

In accordance with Reference 10-1, Volume 2, Program XI.M12, the program includes screening criteria to determine which CASS components have the potential for significant reduction of fracture toughness due to thermal aging embrittlement and require augmented inspection. The screening criteria are applicable to all primary pressure boundary components constructed from CASS with service conditions above 250 °C (482 °F). CASS components that see normal service at 482 °F and below are not susceptible to reduction of fracture toughness by thermal embrittlement. The stuffing boxes for ONS Unit 2 and ONS Unit 3 are located above the thermal barrier and the temperature of the CASS stuffing box is below 482 °F during normal full power operation, assuming a conservative cold leg temperature of 580 °F. As such, the stuffing boxes are below the temperature threshold for thermal embrittlement and are not susceptible to reduction of fracture toughness.

10.6 *Calculation of Thermally Aged J_d*

The methodology for calculating thermally aged J_d for each affected material heat (ONS Unit 2 and ONS Unit 3) is in accordance with NUREG/CR-4513, Revision 2 (Reference 10-4), Section 3.1, Estimation of Thermal Embrittlement of CASS Materials of Known Composition and Service Condition – Service Time Values. Equations are as follows.

- Material heat chemistry is identified and ferrite numbers are calculated based on Reference 10-4, Equation (5)

- For each pump casing half, calculate Room Temperature CV sat using Reference 10-4, CF8M equations (21) through (23) where Ni < 10%, and equations (24) through (26) where Ni > 10%
- Calculate decrease in room temperature CV using Reference 10-4 equations (11), (13) to (17) and assume Cv init is 200 J/cm^2 per page 25 since that is not available from the QADP. Aging parameter P based on 72 EFPY at 555 °F.
- Develop J-R curve J_d=C(delta a)^n by using Reference 10-4 Equations (31) and (32) for C and at 290 °C to 320 °C, and assume static castings. For n, use Reference 10-4, Equation (44) for 290°C to 320 °C.

This results in a fracture toughness estimate for each pump casing half for the temperature range 290 °C (554 °F) – 320 °C (608 °F). Pump casings with thermally aged J_d > 1450 in-lb/in^2 (see Section 10.2) at a crack extension of 0.1 inches are not susceptible to reduction of fracture toughness by thermal embrittlement.

There were only four heats of material whose J_d (0.1) inches are less than 1450 in-lb/in^2, and are reported in Table 10-3. The pump heat with minimum J_d (0.1) is 2RC-P1A1-Serial Number 1, Lower Half-Heat 17368-1.

Table 10-3
Thermally Aged J_d (0.1-inch) for ONS Unit 2 and ONS Unit 3 RCP Casings

ONS Unit 2

RCP	Heat	Ferrite	J_d (0.1-inch) Thermally Aged (in-lb/in^2)	Comment
2RC-P1A1-Serial Number 1	Lower Half-Heat 17368-1	14.46	837.81	less than 1450 in-lb/in^2. Susceptible to TE and candidate for flaw tolerance
2RC-P1A2-Serial Number 5	Upper Half-Heat 19427-1	17.55	1182.42	""
2RC-P1A2-Serial Number 5	Lower Half-Heat 20683	18.59	1063.91	""
2RC-P1B2-Serial Number 3	Upper Half-Heat 20723	16.46	1835.41	Not susceptible since J_d > 1450

ONS Unit 3

RCP	Heat	Ferrite	J_d (0.1-inch) Thermally Aged (in-lb/in^2)	Comment
3RC-P3A1-Serial Number 8	Lower Half-Heat 21795	14.2	2105.39	Not susceptible since J_d > 1450
3RC-P3A1-Serial Number 8	Upper Half-Heat 24245-1	15.96	2084.61	Not susceptible since J_d > 1450
3RC-P3A2-Serial Number 10	Upper Half-Heat 13011	19.94	1214.81	less than 1450 in-lb/in^2. Susceptible to TE and candidate for flaw tolerance
3RC-P3B1-Serial Number 6	Upper Half-Heat 22270-1	15.65	2133.33	Not susceptible since J_d > 1450
3RC-P3B1-Serial Number 6	Lower Half-Heat 20910-1	15.38	2070.08	Not susceptible since J_d > 1450
3RC-P3B2-Serial Number 7	Lower Half-Heat 24685	20.06	2143.71	Not susceptible since J_d > 1450

Note:
Items potentially susceptible include CF8M casings (upper and lower). Stuffing boxes made from CF8M not susceptible due to temperature < 482 °F at full power.
Ferrite > 14% potentially susceptible to TE for CF8M

10.7 *J_{Ic} for Limiting ONS Unit 2 RC-P1A1, Lower Half-Heat 17368-1*

Should a linear elastic flaw tolerance evaluation be performed, a value of J_{Ic} must be obtained from the formulation of J_d reported in Section 10.6, so that a value of K_{Ic} may be calculated. A bounding J_{Ic} may be obtained from RCP Heat 17368-1. The procedure used to calculate J_{Ic} from the J_d power law formulation is consistent with ASTM E813-89, Section 9.2, Figure 3. Steps are as follows for Heat 17368-1.

- Develop a blunting line in accordance with the following equation, $J = 2*\sigma_y*\Delta a$
  - σ_y = yield strength from the QADP, psi; **[ ]**
  - Δa=crack extension, inches
- Develop and plot an offset line parallel to the blunting line at an offset value of 0.2 mm (0.008 in.)
- The estimate of J_{Ic} shall be at the intersection of the power law J-R curve and the offset line
- Calculate K_{Ic} from the following equation
  - $K_{Ic} = (E'J_{Ic})^{1/2}$, where the normalized elastic modulus is given by $E' = E/(1 - v^2)$, E is the elastic modulus, and v is the Poisson ratio. Modulus of elasticity and Poisson's ratio are from the ASME Code, Section II, 2017 Edition, at 555 °F.
    - ♦ E=25.9E06-(25.9E06-25.3E06)*55/100=25.6E06 psi from Table TM-1 (Group G Material)
    - ♦ Poisson's ratio = 0.30 from Table PRD

J_d = **[ ]**.

J_d = **[**

].

J_{Ic} = **[** **]** and K_{Ic}= **[** **]**.

10.8 *Summary and Conclusions*

In accordance with Table 10-2, four RCP casing heats of material for ONS Unit 2 are potentially susceptible to thermal embrittlement, and six RCP pump casing heats of material for ONS Unit 3 are potentially susceptible to thermal embrittlement based on ferrite screening. When considering thermally aged J_d (0.1-inch) for the affected heats and comparing those values to a screening criterion of 1,450 in-lb/in^2, RCP pump Heats 17368-1, 19427-1, 20683 for ONS Unit 2, and Heat 13011 for ONS Unit 3 require further evaluation for subsequent license renewal. As permitted by NUREG-2191, Volume 2, XI.M12, further evaluation may be performed for these RCP casing pump heats by completing a bounding flaw tolerance evaluation to demonstrate that the thermally aged material toughness adequately protect against a loss of structural integrity in CASS components. This evaluation is reported in Section 11.0 herein.

10.9 *References for Section 10.0*

10-1. NUREG-2191, Volume 2, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report

10-2. SLR-ISG-2021-02-MECHANICAL, Updated Aging Management Criteria for Mechanical Portions of Subsequent License Renewal Guidance Interim Staff Guidance (ADAMS Accession No. ML20181A434)

10-3. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U.S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998 (ADAMS Accession No. ML15254A151 and ML15112A661)

10-4. NUREG/CR-4513, Revision 2, Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems, ADAMS Accession Number ML16145A082

11.0 ONS UNIT 2 AND ONS UNIT 3 RCP BOUNDING FLAW TOLERANCE EVALUATION

11.1 *Introduction*

As permitted by NUREG-2192, Volume XI,M12 (see Section 10.2), a bounding flaw tolerance evaluation of the ONS Units 2 and 3 RCP casings that are potentially susceptible to reduction of fracture toughness by thermal embrittlement (Section 10.8) was completed by performing a fatigue crack growth analysis and a Linear Elastic Fracture Mechanics (LEFM) based fracture toughness margin evaluation using the existing 2-D linearized finite element stresses from the analyses of the suction side of the casing. These LEFM evaluations for subsequent license renewal are based, in part, on the Code Case N-481 (Reference 11-1) evaluations for ONS Unit 2, Pump Casing Heats 17368-1 and 16729-1, reported in Reference 11-2, and for ONS Unit 3, Pump Casing Heats 21795-1 and 24245-1, reported in Reference 11-3.

In accordance with NRC Regulatory Guide 1.147 (Reference 11-4), Code Case N-481 was annulled on 3/28/2004 and ASME Code, Section XI, Examination Category B-L-1, was removed from the ASME Code, Section XI, in the 2007 Edition, 2008 Addenda (Reference 11-5). At present, Duke is utilizing the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI and portions of the 2017 Edition at Oconee (Reference 11-6) for the fifth ISI interval at Oconee. As such, Code Case N-481 is no longer utilized at ONS Unit 2 and ONS Unit 3 for inservice inspection of RCP casings. However, as reported in NUREG-2191, Volume 2, XI.M12, for RC pump casings, as an alternative to the screening and other actions described above, no further actions are needed if applicants demonstrate that the original flaw tolerance evaluation performed as part of Code Case N-481 implementation remains bounding and applicable for the SLR period or the evaluation is revised to be applicable for 80 years.

As such, the ONS Unit 2 and ONS Unit 3 Code Case N-481 evaluations are revisited with modifications discussed below for SLR to demonstrate that no further actions are needed for the CASS for Unit 2 and Unit 3 RCP heats that are susceptible to thermal embrittlement. In accordance with Regulatory Guide 1.147, an annulled Code Case cannot be used in a subsequent ISI interval unless it is implemented as an approved alternative under 10 CFR 50.55a(z). However, NUREG-2191, XI.M12, places no restrictions on the Code Case N-481 evaluation applicability duration and NRC approval of the updated flaw tolerance analysis, based in part on the current Code Case N-481 evaluation, will be provided through NRC review of the Oconee subsequent license renewal application.

11.2 *Methodology*

The methodology for the RCP LEFM evaluation for SLR is as follows. The limiting ONS Unit 2 RCP 2RC-P1A1 is selected for the bounding evaluation.

[ ]

[ ]

11.4 *Summary of Results*

The circumferential and axial fatigue flaw growth for an initial 0.3-inch semielliptical flaw with a 6:1 aspect ratio for the most limiting locations are summarized in Table 11-1. Since the final a/t ratio is less than 0.25 (quarter thickness), the flaw tolerance objective is met. In addition, since the maximum stress intensity factor, K_{max} of [ ] is less than the limiting K_{Ic} value of [ ] (limiting K_{Ic} value occurs in 2RC-P1A1 Heat 17368-1), the RCP pump casings demonstrate adequate fracture toughness margin for the 80 year SLR period. Note that the limiting K_{IC} reported for Heat 17368-1 in the Code Case N-481 analysis in Reference 11-2 is 140.6 ksi-√in, which is a saturated lower bound value. The reduction at 80 years to [ ] is due to the use of NUREG/CR-4513, Revision 2, and consideration of thermally aged properties of Heat 17368-1 (i.e., J_d and calculation of J_{IC}) at 72 EFPY.

Aging management of the potentially susceptible RCP pump heats 17368-1, 19427-1, 20683 for ONS Unit 2, and heat 13011 for ONS Unit 3 is accomplished through a bounding flaw tolerance evaluation using guidance from the evaluation procedures and acceptance criteria contained in ASME Code, Code Case N-481. Since Code Case N-481 does not provide guidance on safety factors to be used in the evaluation; safety factors consistent with Appendix G of the ASME Code, Section XI are used. The flaw tolerance evaluation is done by performing a fatigue crack growth analysis (using ASME B&PV Code Case N-809) and Linear Elastic Fracture Mechanics (LEFM) based fracture toughness evaluation to demonstrate that the postulated flaw does not grow to the reference quarter thickness (¼T) flaw size and the maximum stress intensity factor (K_{max}) at the final flaw size does not exceed the critical stress intensity factor, K_{Ic} determined for the susceptible heats. The results from the bounding evaluation for the potentially susceptible four RCP casing heats at ONS Units 2 and 3 demonstrate that there is adequate fracture toughness margin for the 80 year Subsequent License Renewal (SLR) period and therefore the thermally aged material toughness adequately protects against a loss of structural integrity in CASS components.

Table 11-1
Summary of Results

Flaw Type	Location	Actual	Allowable	Margin (1-Actual/Allowable)*100
Fracture Toughness		$K_{max,AppG}$ ksi$\cdot\sqrt{in}$	K_{Ic}, ksi$\cdot\sqrt{in}$	
Circumferential	Path 4			
Axial	Path 2			
Flaw Size				
Circumferential	Path 4			
Axial	Path 1			

11.5 *References for Section 11.0*

11-1. ASME Boiler and Pressure Vessel Code, Code Case N-481, "Alternate Examination Requirements for Cast Austenitic Pump Casings, Section XI, Division 1," March 5, 1990

11-2. ISI Report, Unit 2 Oconee 1998 Refueling Outage 16, ADAMS Accession Number ML15239A572 (SIA Report SIR-98-039-Duke Report OSC-7225-provides CC-N481 report for ONS-2, RCP heat 17368-1, but not available on ADAMS)

11-3. ISI Report for Oconee Unit 3 Refueling Outage 17, ADAMS Accession Number ML15239A177 (SIR-98-077 included in ML15239A177)

11-4. Regulatory Guide 1.147, Revision 19, Inservice Inspection Code Case Acceptability, ASME Code, Section XI, Division 1

11-5. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition through 2008 Addenda

11-6. Brunswick 1 & 2; Catawba 1 & 2; Robinson 2; McGuire 1 & 2; Oconee 1, 2, & 3; Harris 1 - Request to Use a Provision of a Later Edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI (EPID L-2020-LLR-0126), ADAMS Accession Number ML21029A335

11-7. ASME Boiler and Pressure Vessel Code, Section XI, 2019 Edition.

11-8. ASME Boiler and Pressure Vessel Code, Code Case N-809, "Reference Fatigue Crack Growth Rate Curves for Austenitic Stainless Steels in Pressurized Water Environment, Section XI, Division 1," June 23, 2015.

11-9. EPRI Report NP-1406-SR, Nondestructive Examination Acceptance Standards, Technical Basis and Development of Boilers and Pressure Vessel Code, ASME Code, Section XI, Division 1, May 1980

ENCLOSURE 4
ATTACHMENT 2

OCONEE NUCLEAR STATION
Framatome Topical Report ANP 3899NP, Revision 0,
"Framatome Reactor Vessel Internals TLAA Input to the ONS SLRA,"
May 2021

Framatome Reactor Vessel Internals TLAA Input to the ONS SLRA

ANP-3899NP
Revision 0

May 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
10 CFR	Title 10, Code of Federal Regulations
AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
CLB	Current Licensing Basis
dpa	Displacements per atom
EAF	Environmentally-Assisted Fatigue
EFPY	Effective Full Power Year
FIV	Flow-Induced Vibration
LBB	Leak-Before-Break
LCB	Lower Core Barrel
LOCA	Loss of Coolant Accident
LRA	License Renewal Application
MRP	Materials Reliability Program
NRC	U.S. Nuclear Regulatory Commission
ONS	Oconee Nuclear Station
ONS-1	Oconee Nuclear Station Unit 1
ONS-2	Oconee Nuclear Station Unit 2
ONS-3	Oconee Nuclear Station Unit 3
RAI	Request for Additional Information
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
RVI	Reactor Vessel Internals
SER	Safety Evaluation Report
SSHT	Surveillance Specimen Holder Tube
Sy	Yield Strength
TLAA	Time Limited Aging Analysis
UCB	Upper Core Barrel

ABSTRACT

The purpose of this document is to provide supplemental information relative to the information reported in the subsequent license renewal application for Oconee Nuclear Station Units 1, 2, and 3 reactor vessel internals (RVI) time limited aging analyses (TLAA) topics that are reported in NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants. NUREG-2192, Table 4.1-2, Section 4.7, and Section 4.3.3.1.1 include the following TLAA topics that apply to the RVI.

- Table 4.1-2 - Ductility Reduction Evaluation for Reactor Internals (B&W designed PWRs only)
- Section 4.3.3.1.1 - Components Evaluated for Fatigue Parameters Other Than CUF_{en}
- Section 4.7.3 - Plant-Specific (Based on review of ONS 60-year license renewal application and associated SER).

The supplemental information provided in this ANP report is intended to assist the NRC with review of the Oconee RVI TLAA topics listed above relative to the applicable "Review Procedures" reported in NUREG-2192. The topics addressed in this ANP report are as follows.

- Reduction of Ductility, BAW-10008, Part 1, Revision 1, addressed in Section 2.0
- Flow-Induced Vibration, BAW-10051, Revision 1 & BAW-10051A, Supplement 1, addressed in Section 3.0
- Metal Fatigue of RVI Replacement Bolting, addressed in Section 4.0
- Reactor Internals Fluence/dpa, as input to the BAW-10008, Part 1, Revision 1 update for SLR and the XI.M16A gap analysis, addressed in Section 5.0

1.0 INTRODUCTION

The purpose of this document is to provide supplemental technical information relative to the information reported in the subsequent license renewal application for Oconee Nuclear Station Units 1, 2, and 3 reactor vessel internals (RVI) time limited aging analyses (TLAA) topics that are reported in NUREG-2192 (Reference 1-1), Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants. NUREG-2192, Table 4.1-2, Section 4.7, and Section 4.3.3.1.1 include the following TLAA topics that apply to the RVI.

- Table 4.1-2 - Ductility Reduction Evaluation for Reactor Internals (B&W designed PWRs only)
- Section 4.3.3.1.1 - Components Evaluated for Fatigue Parameters Other Than CUF_{en}
- Section 4.7.3 - Plant-Specific (Based on review of ONS 60-year license renewal application and associated SER).

Current Licensing Basis—RVI TLAA

The Oconee 60 year license renewal application (Reference 1-2) and associated NRC SER (Reference 1-3), Section 4.2.5.3, contain the following assessment of TLAA applicable to the Oconee RVI.

- Reduction in ductility (BAW-10008, Part 1, Revision 1, Reference 1-4). The staff determined that the ONS RVI aging management program will adequately manage the irradiation aging effect in accordance with 10 CFR 54.21(c)(1)(iii). This TLAA was addressed by Duke Energy in the ONS RVI inspection program and the NRC acceptance of this TLAA evaluation is contained in Reference 1-5.

- Transient cycle count assumptions for the replacement bolting. The NRC determined that this TLAA was managed by the ONS thermal fatigue monitoring program in accordance with 10 CFR 54.21(c)(1)(iii). The ONS thermal fatigue monitoring program must be updated for 80-years in accordance with the requirements of GALL-SLR NUREG-2191, X.M1.
- Flow-induced vibration endurance limit assumptions (BAW-10051, Revision 1 and BAW-10051A, Supplement 1, Reference 1-6, Reference 1-7). This TLAA was evaluated in BAW-2248A (Reference 1-8) and the NRC determined that component stress values were found to be less than the endurance limit, rendering the evaluation acceptable, according to the requirements of 10 CFR 54.21(c)(1)(ii). This TLAA must be updated for 80-years.
- Flaw growth acceptance in accordance with the ASME B&PV Code, Section XI ISI requirements. Duke Energy confirmed that there were no analytical evaluations of flaws applicable to the ONS reactor vessel internals for 60 years. This topic is addressed as part of Metal Fatigue, Section 4.3, evaluation in NUREG-2192 and is not addressed herein.

EFPY

Based on accrued EFPY through Cycles 31, 29, and 30 for Oconee Units 1 through 3 and assuming breaker-to-breaker operation and no outages per cycle (Capacity Factor = 1), the bounding projected EFPY for 80 years for each Oconee Unit is less than 72 EFPY. Therefore, for the Oconee Nuclear Station, TLAA evaluations for SLRA are completed to 72 EFPY. A measurement uncertainty recapture (MUR) power uprate is conservatively factored in at 2% for all TLAA evaluations that utilize neutron fluence as an input.

Measurement Uncertainty Recapture

- ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate (MUR) (Reference 1-9), Section IV.1.A.ii, Reactor core support structures and vessel internals, includes the following relevant information for the applicable TLAA.
  - The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the reactor vessel internals. No changes to the RCS pressure were made as part of the power uprate. The existing analyses are based on the design conditions in the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid.
  - The structural adequacy of RV internals and incore instrument nozzles of ONS Units 1, 2 and 3 was also reviewed with respect to flow-induced vibration (FIV) relative to the MUR power uprate. The components currently analyzed for FIV include the incore instrumentation nozzles, the flow distributor assembly, the thermal shield, and the inlet baffle. From the comparative analysis, the new operational condition of ONS Units 1, 2 and 3 after the MUR power uprate are bounded by the current analysis (BAW-10051, Revision 1 and BAW-10051A, Supplement 1). The RV internals and incore instrument nozzles are structurally adequate with regard to flow-induced vibration including the effects of the MUR power uprate.
  - The MUR (Reference 1-9) was silent regarding the BAW-10008, Part 1, Revision 1. As such, this topic is addressed herein for subsequent license renewal.

All RVI TLAA for subsequent license renewal reported in this ANP document consider the revised operating conditions (e.g., 1.64% increase in power) associated with MUR as reported in Reference 1-9. MUR is assumed at the beginning of Cycle 30 for ONS Unit 1, Cycle 29 for ONS Unit 2, and Cycle 29 for ONS Unit 2. At present, MUR has not been initiated and each unit is currently operating in Cycles 32, 30, and 31, respectively.

The supplemental information provided in this ANP report is intended to assist the NRC with review of the Oconee RVI topics above relative to the applicable “Review Procedures” reported in NUREG-2192. The topics addressed in this ANP report are as follows.

- Reduction of Ductility, BAW-10008, Part 1, Revision 1, is addressed in Section 2.0
- Flow-Induced Vibration, BAW-10051, Revision 1, & BAW-10051A, Supplement 1 is addressed in Section 3.0
- Metal Fatigue of RVI Replacement Bolting is addressed in Section 4.0
- Reactor Internals Fluence/dpa, as input to the BAW-10008, Part 1, Revision 1 update for SLR and the XI.M16A gap analysis is addressed in Section 5.0.
  - The fluence/dpa input to BAW-10008, Part 1, Revision 1 is a TLAA. Framatome does not consider fluence/dpa input to the XI.M16A gap analysis to be a TLAA since these parameters, while time dependent, are used for screening purposes only. Specifically, time dependent damage mechanism screening is used as input to the failure modes and effects criticality analysis (FMECA), which is a qualitative evaluation performed by a panel of subject matter experts.

- The FMECA results are used to perform categorization of RVI items into one of 3 categories (A, B, or C). This categorization may be modified for Category B and C items by further evaluation as reported in MRP-229, and MRP-231, and MRP-355. The FMECA followed by further evaluation for Category B and C items are used to place all RVI items into the following inspection classifications reported in MRP-227: primary, expansion, existing programs, and no additional measures. The final inspection classifications are a result of quantitative input (e.g., fluence and CUFs) and qualitative assessment (FMECA), and as such are not considered by Framatome to meet the definition of a TLAA since fluence and CUFs are not used exclusively to determine the final inspections classifications reported in MRP-227.

1.1 *References*

1-1. NUREG-2192, Standard Review Plan Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants

1-2. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U. S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998, ADAMS Accession Number. ML15254A151 and ML15112A661

1-3. NUREG-1723, “Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287”, ADAMS Accession Number ML003695154

1-4. BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake,” Duke Energy Letter Incorporating BAW-10008, Part 1, Revision 1 into the Oconee Current Licensing Basis (CLB), ADAMS Accession Number ML19312B713. AEC letter to Duke regarding approved SER for BAW-10008, Part 1, Revision 1, ML19319B162

1-5. Oconee Nuclear Station, Units 1, 2, and 3 - Approval of Time-Limited Aging Analysis for Reactor Vessel Internals, ADAMS Accession Number ML13045A489

1-6. BAW-10051, Revision 1, “Design of Reactor Internals and Incore Instrumentation Nozzles for Flow-Induced Vibration,” September 1972, revised in November 1972, Acceptability of BAW-10051, Revision 1, ADAMS Accession Number ML19316A566.

1-7. BAW-10051A, Supplement 1, “Structural Analysis of 177-Fuel Assembly Redesigned Surveillance Specimen Holder Tube," ADAMS Accession Number ML19248D133, 7908020516 (legacy)

1-8. BAW-2248A, Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, March 2000, ADAMS Accession Number ML003708443

1-9. ONS License Amendment Request for Measurement Uncertainty Recapture Power Uprate, February 19, 2020, ADAMS Accession Number ML20050D379 and Safety Evaluation, January 26, 2021, ADAMS Accession Number ML20335A001

2.0 REDUCTION OF FRACTURE TOUGHNESS DUE TO NEUTRON EMBRITTLEMENT (BAW-10008) (SLRA SECTION 4.7.1.1)

2.1 *Introduction*

Framatome Topical Report BAW-10008, Part 1, Revision 1, dated June 1970 (Reference 2-1) documents the acceptability of the reactor vessel internals under LOCA and a combination of LOCA and seismic loadings. The effect of irradiation on the material properties and deformation limits for the internals is presented in Appendix E where it is concluded that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits. Because the conclusion is based on a fluence determination of 40 years of operation, this meets the definition of 10 CFR 54.3(a) and was identified as time-limited aging analysis (TLAA).

Framatome Topical Report BAW-2248A (Reference 2-2) states that this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21(c)(1)(iii). Plant-specific analysis is required to demonstrate that, under LOCA and seismic loading and with irradiation accumulated at the expiration of the period of extended operation, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and will meet the deformation limits. Subsequently, in a letter to the NRC dated December 17, 1999 (Reference 2-3), Duke committed to perform the plant-specific analysis and develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewal license. In response, the NRC in its SER related to the license renewal of three Oconee Units, determined that this program will adequately manage the irradiation aging effect in accordance with 10 CFR 54.21(c)(1)(iii) (Reference 2-4) for the 60 year period.

In a letter to the NRC dated February 20, 2012 (Reference 2-5), Duke submitted the plant-specific analysis constituting the updated TLAA for reduction of fracture toughness of the reactor vessel internals for NRC staff review. The NRC reviewed the licensee's basis for the update of the reduction of ductility TLAA for the three Oconee Units (Reference 2-6) and determined that:

- The licensee has projected the neutron fluence for the RVI for the 60 year period using an acceptable methodology consistent with RG 1.190.
- The licensee compiled test data from materials irradiated in operating light-water reactors, plus the Halden test reactor. These materials were irradiated under conditions more similar to the conditions of the ONS, 1, 2, and 3 RVI and therefore should more accurately represent the behavior of the Type 304SA (solution annealed) material in the ONS 1, 2 and 3 RVI. The newer test data, when plotted on the original graph, confirms the conservatism of the original figure.
- The licensee's evaluation of the deformation limits of BAW-10008, Part 1, Rev. 1, considering the change in tensile properties of the Type 304SA material due to irradiation, is correct.
- The licensee appropriately revised Appendix E of BAW-10008, Part 1, Rev.1 to conclude the RVI would have adequate ductility at 60 years (54 EFPY) to withstand the postulated LOCA plus seismic event.
- The disposition of the TLAA for loss of fracture toughness was not changed by this analysis. Since the NRC staff-approved disposition of this TLAA is that aging will be adequately managed in accordance with 10 CFR 54.21(c)(1)(iii), the licensee must reevaluate this TLAA if new relevant data on loss of ductility of irradiated stainless steel is generated.

Based on the above, the NRC staff concluded that the licensee's evaluation of the TLAA for reduction of ductility of the RVI is acceptable, and the license renewal commitment documented in Section 4.2.5.3 of NUREG-1723 to perform a plant-specific analysis and develop data to demonstrate that the internals will meet the deformation limits at the expiration of the 60 year renewed license, is fulfilled. Therefore, this TLAA must be re-evaluated for subsequent license renewal.

2.2 *Methodology*

Section 3.2 of BAW-10008, Part 1, Revision 1 notes that there are two primary safety considerations that govern the deformation limits of the internals: deformation shall not prevent insertion of control rods, nor shall it prevent adequate post-accident cooling of the core. The resultant faulted condition stresses (Case IV) are provided in Table 1 of BAW-10008, Part 1, Revision 1. Each reactor vessel internals component item listed in Table 1 of BAW-10008, Part 1, Revision 1 is assessed in accordance with one of three process steps (Categories 1-3 – Phase I) for Faulted Conditions to determine if each reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY. This assessment is similar to that submitted and approved by the NRC for the Davis-Besse Nuclear Power Station, Unit 1 for the period of 60-year license renewal (References 2-7, 2-8). For completeness, more recent stress calculations (i.e., developed after the data reported in Table 1 of BAW-10008, Part 1, Revision 1) applicable to the ONS units were reviewed to ensure that the original design basis calculations reported in Table 1 of BAW-10008, Part 1, Revision 1 capture the limiting reactor vessel internals component items for reduction of ductility. This review included asymmetric loading stress calculations for Faulted Conditions reported in BAW-1621, related RAIs, and the ONS final safety evaluation report (SER) for BAW-1621 (References 2-9, 2-10, 2-11), and original and replacement high-strength bolting stress calculations. BAW-1621 is not identified as TLAA but is included for completeness since it contains faulted load stress intensities that post-date the stress intensities reported in BAW-10008, Part 1, Revision 1. The three process steps (Categories 1-3) to determine the impact of reduction of ductility at 72 EFPY on each component item (in order of evaluation) are as follows:

1. Determine if the faulted stress intensity for the reactor vessel internals component item is less than the unirradiated ASME B&PV Code yield strength at operating temperature (600°F) and therefore plasticity will not occur at 72 EFPY. Since the material remains elastic (and neutron embrittlement would increase the yield strength), reduction of ductility is acceptable.

2. Determine if the reactor vessel internals component item is already highly irradiated (i.e., only reactor vessel internals component items directly adjacent to the fuel assemblies - e.g., baffle plates), such that the faulted stress intensity remains below the irradiated yield strength (increases as fluence increases) and plasticity will not occur at 72 EFPY. Since the material remains elastic, with large margin to the irradiated yield strength, reduction of ductility is acceptable.

3. Determine if the expected fluence exposure is low enough such that neutron embrittlement is considered negligible, reduction of ductility is minimal or will not occur at 80 years/72 EFPY and unirradiated ductility properties are still applicable.

For each process step, further evaluation is required for those component items that do not satisfy the screening criterion defined within each process step. For example, component items that do not fall below the yield strength criterion in Category 1 are carried forward to Category 2. Category 2 items, developed from the first process step, that do meet the irradiated yield strength criterion are carried forward to Category 3. Component items that exceed the Category 3 screening criterion require further evaluation to demonstrate that those reactor vessel internals component items will have sufficient ductility at 72 EFPY to meet the deformation limits at the expiration of the subsequent period of extended operation. Further evaluation of those component items are addressed in Phase II.

2.3 *Evaluation*

2.3.1 Phase I

2.3.1.1 Categorization of Component Items from Table 1 of BAW-10008, Part 1, Revision 1

Each reactor vessel internals component item listed in Table 1 of BAW-10008, Part 1, Revision 1 is assessed in accordance with one of three categories to determine if each reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY. The three categories to determine the impact of reduction of ductility at 72 EFPY on each component item are as follows:

1. The Case IV faulted stress intensity listed in Table 1 of BAW-10008, Part 1, Revision 1 for the reactor vessel internals component item is less than the unirradiated yield strength at operating temperature and therefore plasticity will not occur at 72 EFPY. Since the material remains elastic (and neutron embrittlement would increase the yield strength), reduction of ductility is acceptable.
2. The reactor vessel internals component item is already highly irradiated (i.e., only reactor vessel internals component items directly adjacent to the fuel assembles - e.g., baffle plates), such that the faulted stress intensity remains below the irradiated yield strength (increases as fluence increases) and plasticity will not occur at 72 EFPY. Since the material remains elastic, with large margin to the irradiated yield strength, the reduction of ductility is acceptable.
3. The neutron embrittlement is negligible, reduction of ductility is minimal or will not occur at 72 EFPY and unirradiated ductility properties are still applicable.

Component items that are not shown acceptable for the three categories above are evaluated in Phase II.

2.3.1.1.1 Assessment per Criteria Defined by Category #1

For the Category 1 assessment, the faulted stress intensity for each the reactor vessel internals component item is compared to the unirradiated ASME B&PV Code yield strength for the applicable material type at operating temperature (i.e., about 550°F to 600°F). For those reactor vessel internals component items that have a reported stress intensity value less than the unirradiated ASME B&PV Code yield strength at 600°F, the material should remain elastic (and neutron embrittlement would increase the yield strength), reduction of ductility is acceptable; therefore, no further analysis is required. The unirradiated ASME B&PV Code yield strength (S_y) values at 600°F for the reactor vessel internals component items materials of interest are presented in Table 2-1. Reactor vessel internals component items reported in BAW-10008, Part 1, Revision 1, Table 1, that satisfy Category 1 screening criterion are reported in Table 2-2. Seven component items were carried forward to Process Step 2: lower grid plate, plenum cover, plenum cylinder reinforcing plate, core support shield top flange, core support shield lower flange, baffle plates, and upper core barrel bolts for ONS-3 only.

Table 2-1
ASME B&PV Code Yield Strength Values at 600°F

Temperature (ºF)	Yield Strength (psi) of Type 304 (e.g., SA-240)	Yield Strength (psi) of Type 304L (e.g., SA-240)	Yield Strength (psi) of Alloy A-286 (e.g., SA-453 Grade 660)	Yield Strength (psi) of Alloy X-750 (SB-637, Grade 688, Type 3)
600	17,300	14,400	81,000	92,500

Table 2-2
Faulted Stress Summary From BAW-10008, Part 1, Revision 1 vs ASME Code Strengths at 600°F

Component Item	Material	Stress Type	Case IV Faulted Stress Intensity (psi)	Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Lower grid plate **Outlet pipe rupture**	**Type 304**	**P_L+P_b**			**17,300**	**Further evaluation is required**
Inlet pipe rupture		**P_L+P_b**			**17,300**	**Further evaluation is required**
Plenum cover	**Type 304**	**P_L+P_b**			**17,300**	**Further evaluation is required**
Plenum cylinder reinforcing plate	**Type 304**	**P_L+P_b**			**17,300**	**Further evaluation is required**
Upper guide tubes	Type 304L	P_m			14,400	No further analysis is required
		P_L+P_b			14,400	No further analysis is required
Upper guide tube sectors	Type 304L	P_L+P_b			14,400	No further analysis is required
Core support shield, top flange **Subcooled portion of LOCA**	**Type 304**	**P_m**			**17,300**	**Further evaluation is required**
		P_L+P_b			**17,300**	**Further evaluation is required**
Saturated portion		P_m			17,300	No further analysis is required
		P_L+P_b			**17,300**	**Further evaluation is required**
Core support shield, lower flange **Subcooled portion of LOCA**	**Type 304**	P_m			17,300	No further analysis is required
		P_L+P_b			**17,300**	**Further evaluation is required**
Saturated portion of LOCA		P_m			17,300	No further analysis is required
		P_L+P_b			**17,300**	**Further evaluation is required**
Core barrel, top flange Subcooled portion of LOCA	Type 304	P_m			17,300	No further analysis is required
		P_L+P_b			17,300	No further analysis is required
Saturated portion of LOCA		P_m			17,300	No further analysis is required
		P_L+P_b			17,300	No further analysis is required
Baffle plates	**Type 304**	**P_L+P_b**			**17,300**	**Further evaluation is required**
Internals bolts **Core barrel-core support shield joint**	**Alloy A-286**	**P_m**			**81,000**	**Further evaluation is required (ONS-3 only)**
		P_L+P_b			**81,000**	**Further evaluation is required (ONS-3 only)**
Core barrel-lower grid cylinder joint	Alloy A-286	P_m			81,000	No further analysis is required
		P_L+P_b			81,000	No further analysis is required

(1) Stress analysis of the current upper core barrel (UCB) bolting configurations was performed in 2008 for missing or deficient UCB bolts. The maximum calculated faulted stresses for the bolting are: ONS-1 ([ ]); ONS-2 ([ ]); and ONS-3 ([ ]).

(2) Stress analysis of current lower core barrel (LCB) bolting configuration was performed in 2008 for deficient LCB bolts. The maximum calculated faulted stresses for the bolting are: ONS-1 ([ ]); ONS-2 ([ ]); and ONS-3 ([ ]).

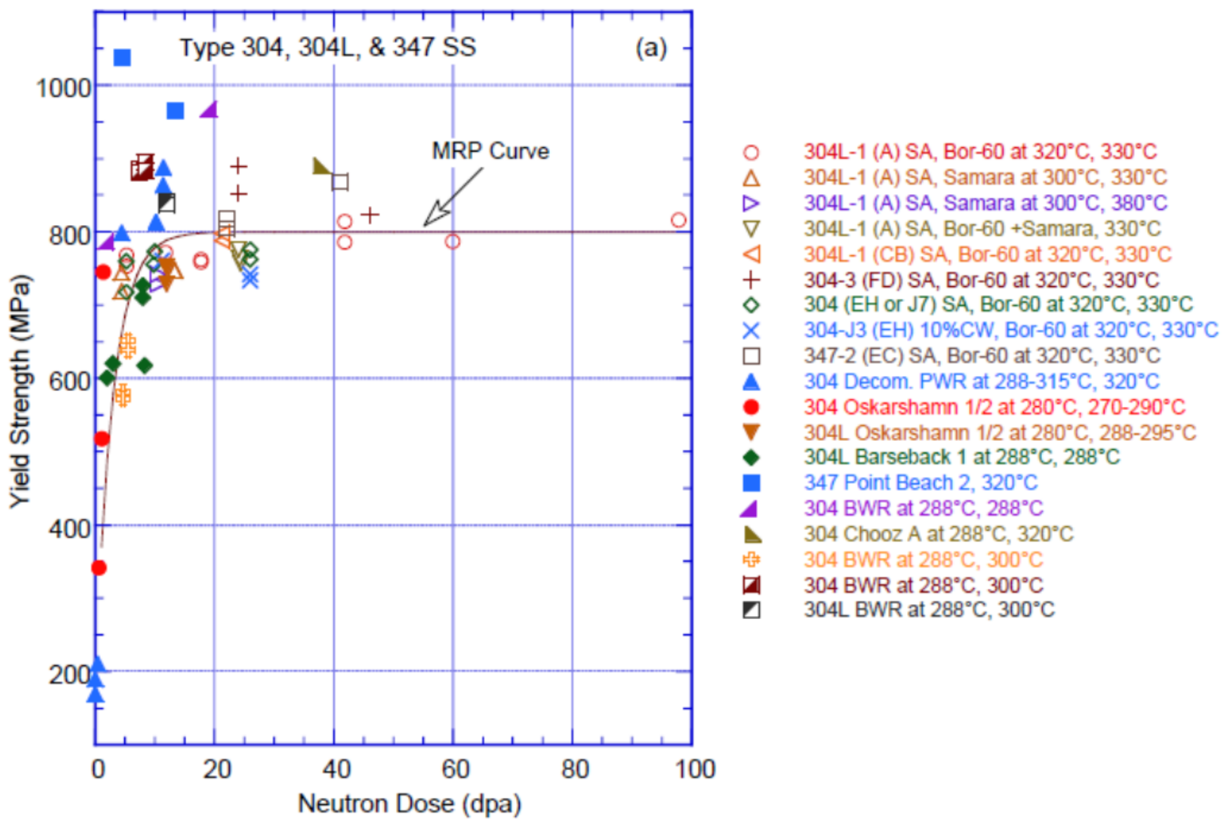
2.3.1.1.2 Assessment per Criteria Defined by Category #2

For the Category #2 assessment, the expected reactor vessel fluence (E > 1.0 MeV) exposure of the applicable ONS reactor vessel internals component items listed in Table 1 of BAW-10008, Part 1, Revision 1 are reviewed to determine if the component item is highly irradiated to the level where saturation of the material's yield strength has occurred as shown in Figure 2-1 (Figure 13(a) of NUREG/CR-7027). Only reactor vessel internals component items directly adjacent to the fuel assembles are expected to be highly irradiated. For the case where the component item is highly irradiated at 72 EFPY and the faulted stress intensity is below the irradiated yield strength (which increases as fluence increases), plasticity will not occur at 72 EFPY; therefore, the material remains elastic with large margin to the irradiated yield strength, and the reduction of ductility is acceptable.

Based on the 72 EFPY fluence estimates for the stress types of the seven component items identified in Section 2.3.1.1.1 where the faulted stress intensities are greater than the unirradiated yield strengths at 600°F, only the baffle plates are expected to be highly irradiated ([ ] - Table 5-1). Applying the baffle plate estimated fluence to the curve in Figure 2-1, the irradiated yield strength is expected to be about [ ]. Note that, per the correlation in Figure 2-1, above about 5 dpa the irradiated yield strength is greater than a factor of three higher than the baffle plate stress intensity in Table 2-2. This correlation is consistent with Section 3.5.3 of MRP-135, Rev 2 (Reference 2-12), which describes factors for temperature and irradiation dose that can be used to calculate yield strength for Type 304 solution-annealed stainless steel based on updated unirradiated and irradiated tensile test data. The baffle plates faulted stress intensity in Table 1 of BAW-10008, Part 1, Revision 1 ([ ]) is significantly less than this irradiated saturated yield strength, therefore the material remains elastic with large margin to the irradiated yield strength, and the reduction of ductility is acceptable and no further evaluation is required.

For the remaining six reactor vessel internals component items identified in Table 2-2 where the faulted stress intensities are greater than the unirradiated yield strengths at 600°F, further assessment in accordance with criteria defined by Category #3 is performed to determine if the reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY. The six component items carried forward to process step 3 include the lower grid plate, plenum cover, plenum cylinder reinforcing plate, core support shield top flange, core support shield lower flange, and upper core barrel bolting for ONS-3 only.

Figure 2-1
Change in Yield Strength as a Function of Neutron Dose for Solution-Annealed Type 304, 304L, and 347 Stainless Steels at Elevated Temperature, 270-380°C (Figure 13(a) of NUREG/CR-7027)



2.3.1.1.3 Assessment Per Criteria Defined by Category #3

For the remaining six ONS reactor vessel internals component items not screened out per the Category #1 and #2 assessments, a review is performed to determine if the expected fluence (E > 1.0 MeV) exposure of the reactor vessel internals component item is low enough such that neutron embrittlement is considered negligible, thus reduction of ductility is minimal or will not occur at 72 EFPY for the component item.

The projected 72 EFPY dose and fluence values for the remaining six reactor vessel internals component items are listed in Table 2-3. The methodology used to develop these dose and fluence values is described in Section 5.2. If the projected fluences listed in Table 2-3 are applied to Figure E-3 of BAW-10008, Part 1, Revision 1, the decrease in uniform elongation for the plenum cover, plenum cylinder reinforcing plate, and core support shield top flange (fabricated from Type 304 stainless steel) at both 572°F and 752°F (temperatures between which these component items would be expected to experience) is such that the 20 percent uniform elongation of irradiated material credited for 40 years in Appendix E of BAW-10008, Part 1, Revision 1 and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1 is met for these component items. Note that Figure 3-12 in Reference 2-13 provides irradiated Type 304 stainless steel solution annealed test data to compare to the curves in Figure E-3 of BAW-10008, Part 1, Revision 1. The test data validates the conservatism of the curves in Figure E-3 of BAW-10008, Part 1, Revision 1. This slight decrease in uniform elongation at the projected fluence levels for the plenum cover, plenum cylinder reinforcing plate, and core support shield top flange is confirmed in Figure 13(c) of NUREG/CR-7027 (Reference 2-14).

Section 3.5.3 of MRP-135, Rev 2 describes factors for temperature and irradiation dose that can be used to calculate uniform elongation for Type 304 solution-annealed stainless steel based on updated unirradiated and irradiated tensile test data. This data requires a dose value in dpa, so these equations can be used to calculate the uniform elongation for the plenum cover only, as the plenum cylinder reinforcing plate and core support shield top flange do not have calculated dose values listed in dpa in Table 2-3. Using the equations in Section 3.5.3 of MRP-135, Rev 2, the percent uniform elongation calculated for a dose of [ ] (Table 2-3) and a temperature of 600°F (315°C) for solution-annealed Type 304 stainless steel is about [ ]. It should be noted that the dose value for the plenum cover is actually listed in Table 2-3 as less than (<) [ ], so the percent uniform elongation expected is most likely greater than [ ].

In addition, as shown in Figure 5 of Volume 5, Issue 3 of the Journal of Engineering Materials and Technology (Reference 2-15), the uniform elongation of unirradiated Type 304SA stainless steel at 600°F only decreases slightly with increasing strain rate. Further observation of the data shows that even at the highest tested strain rates (10^1/sec and 10^2/sec) at 600°F, the uniform elongation is above the 20 percent uniform elongation of irradiated austenitic stainless steel material credited for 40 years in Appendix E and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1. It is also observed that yield strength increases with increasing strain rate at 600°F as shown in Figure 3 of Volume 5, Issue 3 of the Journal of Engineering Materials and Technology. In addition to having sufficient ductility at 72 EFPY relative to the allowable stresses of Table 1 of BAW-10008, Part 1, Revision 1, the plenum cover, plenum cylinder reinforcing plate, and core support shield top flange will have greater resistance to plastic deformation at increased strain rates.

For the three remaining reactor vessel internals component items (lower grid plate, core support shield lower flange, and upper core barrel bolts at ONS-3), further assessment will be performed in Phase II to determine if these reactor vessel internals component items should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY.

Table 2-3
Projected Fast Fluence (E > 1.0 MeV) of Select Reactor Vessel Internals Component Items from Table 1 of BAW-10008, Part 1, Revision 1

Component Item	Stress Type	Case IV Faulted Stress Intensity (psi)	72 EFPY Dose(1) (dpa)	72 EFPY Neutron Fluence(1), E > 1.0 MeV (n/cm^2)	Comment
Lower grid plate Outlet pipe rupture	**P_L+P_b**				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Inlet pipe rupture	**P_L+P_b**				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Plenum cover	P_L+P_b				Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
Plenum cylinder reinforcing plate	P_L+P_b				Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
Core support shield, top flange Subcooled portion of LOCA	P_m				Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
	P_L+P_b				Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
Saturated portion	P_L+P_b				Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
Core support shield, lower flange Subcooled portion of LOCA	**P_L+P_b**				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Saturated portion of LOCA	**P_L+P_b**				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Internals bolts Core barrel-core support shield joint (ONS-3 only)	**P_m**				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
	P_L+P_b				**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**

(1) See Section 5.0 (Table 5-1 and Table 5-2)

2.3.1.2 Categorization of Component Items from BAW-1621

More recent asymmetric LOCA loading stress calculations (i.e., developed after the data reported in Table 1 of BAW-10008, Part 1, Revision 1) are available and reported in Framatome Topical Report BAW-1621, which is supplemented by NRC RAI responses. The applicable ONS asymmetric loading stress intensity calculations (i.e., skirt-supported) are reviewed to ensure that the original design basis calculations reported in Table 1 of BAW-10008, Part 1, Revision 1 capture the limiting reactor vessel internals component items for reduction of ductility. Based on the stress limits criteria for faulted conditions (consistent with BAW-10008, Part 1, Revision 1), the assessment in the following sections is limited to only reactor vessel internals component items associated with the stress category "P_m" (primary membrane stress intensity) or "P_m+P_b" (primary membrane plus primary bending stress intensity).

Similar to the actions performed for the reactor vessel internals component items in Section 2.3.1.1, each reactor vessel internals component item listed in Tables 3.24-1, 3.24-2, 3.24-3, 3.24-4, and 3.24-5 of BAW-1621, Supplement 1 is assessed in accordance with one of three categories to determine if each component item should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY. The three categories are the same as those listed in Section 2.3.1.1. Component items that are not shown acceptable for the three categories are evaluated in Phase II.

2.3.1.2.1 Assessment per Criteria Defined by Category #1

For the Category #1 assessment, the faulted condition maximum calculated "P_m" or "P_m+P_b" stress intensity values listed in Tables 3.24-1, 3.24-2, 3.24-3, 3.24-4, and 3.24-5 of BAW-1621, Supplement 1 for each reactor vessel internals component item is compared to the unirradiated yield strength at operating temperature (i.e., 600°F). For those reactor vessel internals component items that have reported faulted condition maximum calculated "P_m" or "P_m+P_b" stress intensity values less than the unirradiated yield strength at operating temperature, the material should remain elastic (and neutron embrittlement would increase the yield strength), reduction of ductility is acceptable; therefore, no further analysis is required.

The unirradiated yield strength (S_y) values at 600°F for the reactor vessel internals component items materials of interest are presented in Table 2-1.

The comparison of the faulted condition maximum calculated "P_m" or "P_m+P_b" stress intensity values listed in Tables 3.24-1, 3.24-2, 3.24-3, 3.24-4, and 3.24-5 of BAW-1621 Supplement 1 to the applicable unirradiated yield strengths at 600°F are shown in Table 2-4, Table 2-5, Table 2-6, Table 2-7, and Table 2-8. Eight component items were carried forward to Process Step 2: ONS-3 UCB bolts, core support shield lower flange, lower grid rib section, lower grid rib section/lower grid shell forging bolt, support post/support forging weld, plenum cylinder, upper grid rib section, and upper grid pad joint bolt.

Table 2-4
Core Support Shield Stress Analysis Results for Faulted Conditions Reproduced from BAW-1621 Supplement 1 & ASME Code Yield Strength at 600ºF

Component Item	Material	Stress Type	Max Calculated Stress (psi)	Maximum Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Core Support Shield	Type 304	P_m	[	]	17,300	No further analysis is required
		Buckling			N/A	N/A
Core Support Shield/Core Barrel Bolted Joint	**Alloy A-286**	**P_m**			**81,000**	**Further evaluation is required (ONS-3 only)**
Core Barrel Upper Flange	Type 304	P_m			17,300	No further analysis is required
		Bearing			N/A	N/A
		Shear			N/A	N/A
Core Support Shield Lower Flange	**Type 304**	P_m			17,300	No further analysis is required
		P_m+P_b			**17,300**	**Further evaluation is required**
		Bearing			N/A	N/A
Core Support Shield Upper Flange	Type 304	Load Limit, kips		]	N/A	N/A

(1) Stress analysis of the current UCB bolting configurations was performed in 2008 for missing or deficient bolts. The maximum calculated faulted stress for the bolting are: ONS-1 ([ ]), ONS-2 ([ ]), and ONS-3 ([ ]).

Table 2-5
Lower Grid Assembly Stress Analysis Results for Faulted Conditions Reproduced from BAW-1621 Supplement 1 & ASME Code Yield Strength at 600ºF

Component Item	Material	Stress Type	Max Calculated Stress (psi)	Maximum Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Core Barrel/Lower Grid Assembly Bolts	Alloy A-286	P_m			81,000	No further analysis is required
		Stripping			N/A	N/A
		Bearing			N/A	N/A
Core Barrel Lower Flange	Type 304	P_m			17,300	No further analysis is required
Grid Pad	Type 304	Bearing			N/A	N/A
Grid Pad/Rib Section Joint – Bolt	Grade B8	P_m			17,300	No further analysis is required
Grid Pad/Rib Section Joint – Dowel	Alloy X-750	Shear			N/A	N/A
Rib Section	**Type 304**	**P_m**			**17,300**	**Further evaluation is required**
		P_m+P_b			**17,300**	**Further evaluation is required**
Support Post/Rib Section Joint – Bolt	Grade B8	P_m			17,300	No, further analysis not required
Rib Section/Lower Grid Shell Forging Joint Bolt	**Grade B8**	**P_m**			**17,300**	**Further evaluation is required**
Support Posts	Type 304	P_m			17,300	No further analysis is required
Flow Distributor Plate	Type 304	P_m			17,300	No further analysis is required
Flow Distributor Plate/Lower Grid Shell Forging Joint – Weld	Type 308/308L	P_m			14,400	No further analysis is required
Support Post/Flow Distributor Plate Welded Joint	Type 308L	P_m			14,400	No further analysis is required
Support Forging	Type 304	P_m			17,300	No further analysis is required
		P_m + P_b			17,300	No further analysis is required
Support Post/Support Forging Joint – Weld	**Type 308/308L**	**P_m**			**14,400**	**Further evaluation is required**
Lower Grid Shell Forging	Type 304	P_m			17,300	No further analysis is required

Table 2-6
Flow Distributor Assembly Stress Analysis Results for Faulted Conditions Reproduced from BAW-1621 Supplement 1 & ASME Code Yield Strength at 600ºF

Component Item	Material	Stress Type	Max Calculated Stress (psi)	Maximum Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Flow Distributor Head/Lower Grid Assembly Bolts	Alloy A-286	P_m			81,000	No further analysis is required
		Tear-out			N/A	N/A
		Bearing			N/A	N/A
Flow Distributor Shell	Type 304	P_m			17,300	No further analysis is required
		P_m+P_b			17,300	No further analysis is required
Flow Distributor Flange	Type 304	P_m			17,300	No further analysis is required
		P_m+P_b			17,300	No further analysis is required

Table 2-7
Core Barrel Assembly Stress Analysis Results for Faulted Conditions Reproduced from BAW-1621 Supplement 1 & ASME Code Yield Strength at 600ºF

Component Item	Material	Stress Type	Max Calculated Stress (psi)	Maximum Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Core Barrel	Type 304	P_m	[		17,300	No further analysis is required
		Buckling			N/A	N/A
Baffle-Former A-A Bolts	Grade B8	P_m			17,300	No further analysis is required
Barrel-Former A-A Bolts	Grade B8	P_m			17,300	No further analysis is required(1)
Thermal Shield Lower End	Type 304	P_m			17,300	No further analysis is required
Thermal Shield Upper End	Type 304	P_m			17,300	No further analysis is required
Thermal Shield/Lower Grid Shell Forging Bolts	Alloy X-750	P_m			92,500	No further analysis is required
Thermal Shield Upper Restraint	Type 304	Bearing			N/A	N/A
		Shear			N/A	N/A
Bolts	Alloy A-286	P_m			81,000	No further analysis is required
Dowels	Alloy A-286	Shear		]	N/A	N/A

(1) Based on B&W fabrication records, the lowest measured room temperature yield strength is [ ] psi. Per empirically based correlations between yield strength and temperature for unirradiated stainless steels, it is estimated the barrel-former bolts measured yield strength will be reduced to [ ] psi at 600°F. Since the calculated stress of [ ] psi is less than [ ] psi, reduction of ductility is acceptable.

Table 2-8
Plenum Assembly Stress Analysis Results for Faulted Conditions Reproduced from BAW-1621 Supplement 1 & ASME Code Yield Strength at 600ºF

Component Item	Material	Stress Type	Max Calculated Stress (psi)	Maximum Allowable Stress (psi), not used in Comparison	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Plenum Cover	Type 304	Bearing			N/A	N/A
Plenum Cylinder	**Type 304**	**P_m**			**17,300**	**Further evaluation is required**
		Buckling			N/A	N/A
CRGT Slotted Region	Type 304	P_m			17,300	No further analysis is required
		Buckling			N/A	N/A
CRGT Perforated Region	Type 304	P_m			17,300	No further analysis is required
		Buckling			N/A	N/A
CRGT Lower Joint – Bolt	Grade B8	P_m			17,300	No further analysis is required
CRGT Lower Joint – Dowel	Type 304	Shear			N/A	N/A
CRGT/Plenum Cover Joint	Type 308L	P_m			14,400	No further analysis is required
Upper Grid Rib Section	**Type 304**	P_m			17,300	No further analysis is required
		$P_m + P_b$			**17,300**	**Further evaluation is required**
Upper Grid Pad Joint – Dowel	Alloy X-750	Dowel			N/A	N/A
Upper Grid Pad Joint – Bolt	**Grade B8**	**Bolt**			**17,300**	**Further evaluation is required**

2.3.1.2.2 Assessment per Criteria Defined by Category #2

For the Category #2 assessment, the expected reactor vessel fluence (E > 1.0 MeV) exposure of the applicable ONS reactor vessel internals components items are reviewed to determine if the component item is highly irradiated to the level where saturation of the material's yield strength has occurred as shown in Figure 2-1 (Figure 13(a) of NUREG/CR-7027). Only reactor vessel internals component items directly adjacent to the fuel assembles are expected to be highly irradiated. For the case where the component is highly irradiated at 72 EFPY and the faulted condition maximum calculated "P_m" or "P_m+P_b" value is below the irradiated yield strength (which increases as fluence increases), plasticity will not occur at 72 EFPY; therefore, the material remains elastic with large margin to the irradiated yield strength, and the reduction of ductility is acceptable.

Based on the fluence estimates for the eight component item "P_m" or "P_m+P_b" stress category types identified in Table 2-4, Table 2-5, and Table 2-8 where the faulted condition maximum calculated "P_m" or "P_m+P_b" stress intensity values are greater than the unirradiated yield strengths at 600°F, none are expected to be highly irradiated.

For those eight reactor vessel internals component items identified in Table 2-4, Table 2-5 and Table 2-8 where the maximum calculated stress are greater than the unirradiated yield strengths at 600°F, further assessment in accordance with criteria defined by Category #3 is performed to determine if these reactor vessel internals component items should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY.

2.3.1.2.3 Assessment per Criteria Defined by Category #3

For the eight remaining ONS reactor vessel internals component items not screened out per the Category #1 and #2 assessments, a review is performed to determine if the expected fluence (E > 1.0 MeV) exposure of the reactor vessel internals component item is low enough such that neutron embrittlement is considered negligible, thus reduction of ductility is minimal or will not occur at 72 EFPY for the component item.

The projected 72 EFPY dose and fluence values for the remaining eight reactor vessel internals component items are listed in Table 2-9. The methodology used to develop these dose and fluence values is described in Section 5.2. If the projected fluences listed in Table 2-9 are applied to Figure E-3 of BAW-10008, Part 1, Revision 1, the decrease in uniform elongation for the plenum cylinder (fabricated from Type 304 stainless steel) at both 572°F and 752°F (temperatures between which these component items would be expected to experience) is such that the 20 percent uniform elongation of irradiated material credited for 40 years in Appendix E of BAW-10008, Part 1, Revision 1 and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1 is met for this component item. Note that Figure 3-12 in Reference 2-13 provides irradiated Type 304 stainless steel solution annealed (SA) test data to compare to the curves in Figure E-3 of BAW-10008, Part 1, Revision 1. The test data validates the conservatism of the curves in Figure E-3 of BAW-10008, Part 1, Revision 1. This slight decrease in uniform elongation at the projected fluence levels for the plenum cylinder is confirmed in Figure 13(c) of NUREG/CR-7027.

Section 3.5.3 of MRP-135, Rev 2 describes factors for temperature and irradiation dose that can be used to calculate uniform elongation for Type 304 solution-annealed stainless steel based on updated unirradiated and irradiated tensile test data. This data requires a dose value in dpa, so these equations cannot be used to calculate the uniform elongation for the plenum cylinder as the plenum cylinder does not have calculated dose values listed in dpa in Table 2-9.

In addition, as shown in Figure 5 of Volume 5, Issue 3 of the Journal of Engineering Materials and Technology, the uniform elongation of unirradiated Type 304SA stainless steel at 600°F only decreases slightly with increasing strain rate. Further observation of the data shows that even at the highest tested strain rates (10^1/sec and 10^2/sec) at 600°F, the uniform elongation is above the 20 percent uniform elongation of irradiated austenitic stainless steel material credited for 40 years in Appendix E and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1. It is also observed that yield strength increases with increasing strain rate at 600°F as shown in Figure 3 of Volume 5, Issue 3 of the Journal of Engineering Materials and Technology. In addition to having sufficient ductility at 72 EFPY relative to the allowable stresses of Table 1 of BAW-10008, Part 1, Revision 1, the plenum cylinder will have greater resistance to plastic deformation at increased strain rates.

For the seven remaining reactor vessel internals component items (ONS-3 UCB bolts, core support shield lower flange, lower grid rib section, lower grid rib section/lower grid shell forging bolt, support post/support forging weld, upper grid rib section, and upper grid pad joint bolt), further assessment will be performed in Phase II to determine if these reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY.

Table 2-9
Projected Fast Fluence (E > 1.0 MeV) of Select Reactor Vessel Internals Component Items from BAW-1621 Supplement 1

Component Item	Stress Type	Max Calculated Stress (psi)	72 EFPY Dose (dpa)	72 EFPY Neutron Fluence, E > 1.0 MeV (n/cm^2)	Comment
Core Support Shield/Core Barrel Bolted Joint (ONS-3 only)	P_m	[ ]	[ ] (Note 1)	[ ] (Note 1)	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Core Support Shield Lower Flange	P_m+P_b	[ ]	[ ] (Note 1)	[ ] (Note 1)	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
(Lower Grid) Rib Section	P_m	[ ]	[ ] (Note 1)	[ ] (Note 1)	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
	P_m+P_b	[ ]	[ ] (Note 1)	[ ] (Note 1)	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
(Lower Grid) Rib Section/Lower Grid Shell Forging Joint Bolt	P_m	[ ]	[ ]	[ ]	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Support Post/Support Forging Joint – Weld	P_m	[ ]	[ ]	[ ]	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Plenum Cylinder	P_m	[ ]	[ ] (Note 1)	[ ] (Note 1)	Fluence sufficiently low that the effect of irradiation embrittlement on the ductility of component item is minimal; further analysis not required.
Upper Grid Rib Section	P_m+P_b	[ ]	[ ] (Note 1)	[ ] (Note 1)	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**
Upper Grid Pad Joint – Bolt	**Bolt**	[ ]	[ ]	[ ]	**Fluence sufficient to cause neutron embrittlement of component item. Further evaluation is required.**

(1) See Section 5.0 (Table 5-1 and Table 5-2)

2.3.2 Phase II

Based on the results of the Phase I categorical assessments, the following reactor vessel internals component items were determined to be potentially susceptible to an unacceptable amount of reduction of ductility at 72 EFPY. Duplicate component item results between the two evaluations are <u>underlined</u>.

From Table 1 of BAW-10008, Part 1, Revision 1 (Section 2.3.1.1):

1. <u>Lower grid plate (i.e., lower grid rib section)</u>
2. <u>Core support shield lower flange</u>
3. <u>Upper core barrel bolts (ONS-3 only based on stress analysis of the current ONS-3 upper core barrel (UCB) bolting configuration)</u>

From RAI responses to BAW-1621, Supplement 1 (Section 2.3.1.2):

1. <u>Upper core barrel bolts (ONS-3 only based on stress analysis of the current ONS-3 upper core barrel (UCB) bolting configuration)</u>
2. <u>Core support shield lower flange</u>
3. <u>Lower grid rib section</u>
4. Lower grid rib section/lower grid shell forging joint bolt
5. Support post/support forging joint – weld
6. Upper grid rib section
7. Upper grid pad joint – bolt

To reconcile these component items/welds, a recalculated set of loads were developed to determine the faulted condition stresses for the six component items/welds excluding the ONS-3 upper core barrel bolts. These recalculated loads were based on smaller break loadings due to Leak-Before-Break (LBB) of the primary piping. The recalculated stress intensities are compared to the applicable unirradiated ASME Code yield strength values at 600°F in Table 2-10. The recalculated faulted condition stress intensity values are less than the unirradiated ASME Code yield strength at 600°F for the six components items/welds. Thus reduction of ductility is acceptable for these six component items and no further analysis is required.

For the ONS-3 UCB bolts the calculated P_m+P_b stress is [ ] psi. Per Table 2-1, the yield strength for Alloy A-286 at 600°F that was used for the Category 1 assessments is 81,000 psi. A limited records search for measured high temperature yield strength data for Alloy A-286 was performed. The data has been gathered and summarized in Table 2-11. This data shows that at these elevated temperatures, the measured yield strength values of these tested materials are greater than [ ] psi, with the lowest reported being 88.6 ksi at 800.6ºF. Therefore Category 1, as described in Sections 2.3.1.1.1 and 2.3.1.2.1 can be used for assessment and the reduction of ductility is expected to be acceptable for the ONS-3 UCB bolts.

Table 2-10
Select Component/Weld Recalculated Stress Analysis Results for Faulted Conditions and ASME Code Yield Strength at 600°F

Component Item	Material	Stress Type	Recalculated Faulted Stress Intensity (psi)	ASME Code Yield Strength (S_y) at 600ºF (psi)	Comparison of Case IV Faulted Stress Intensity vs ASME Yield Strength (S_y) at 600ºF
Lower grid plate (lower grid rib section)	Type 304	P_m		17,300	No further analysis is required
		P_m+P_b		17,300	No further analysis is required
Core support shield lower flange	Type 304	P_m+P_b		17,300	No further analysis is required
Upper core barrel bolts (ONS-3)	**Alloy A-286**	**N/A**		**81,000**	**Further evaluation is required**
Lower grid rib section/lower grid shell forging joint bolt	Grade B8	P_m		17,300	No further analysis is required
Support post/support forging joint – weld	Type 308/308L	P_m		14,400	No further analysis is required
Upper grid rib section	Type 304	P_m		17,300	No further analysis is required
		P_m+P_b		17,300	No further analysis is required
Upper grid pad joint – bolt	Grade B8	Bolt		17,300	No further analysis is required

Table 2-11
High Temperature Alloy A-286 Yield Strength Data

Material Heat	Heat Treatment Details	Product Form	Strain Rate	Temperature	Measured Yield Strength	Reference
K-58139-2	Annealed at 982ºC (1800ºF) for 30 minutes Precipitation heat-treated at 718ºC (1325ºF) for 16 hours	Strip	3×10^{-5} sec^{-1}	427ºC (800.6ºF)	88.6 ksi	(2-16)
			3×10^{-5} sec^{-1}		89.3 ksi	
			3×10^{-3} sec^{-1}		100.0 ksi	
			6×10^{-2} sec^{-1}		105.6 ksi	
			1.0 sec^{-1}		103.4 ksi	
8790		Rod	3×10^{-5} sec^{-1}		102.4 ksi	
			6×10^{-2} sec^{-1}		102.6 ksi	
			1.0 sec^{-1}		101.2 ksi	
90456	Solution heat treated at 1800ºF for 30 minutes, air cooled Age-hardened at 1325ºF for 16 hours, air cooled	Sheet	Unknown	1140ºR (680.33ºF)	94.5 ksi	(2-17)
82312		Sheet	Unknown		94.3 ksi	
K-58139-2	Solution annealed at 982ºC (1800ºF) for 30 minutes, water quenched Aged at 718ºC (1325ºF) for 16 hours	Strip	Unknown	427ºC (800.6ºF)	730 MPa (105.877 ksi)	(2-18)
K-58139-2	Solution annealed at 1800ºF for 30 minutes, water quenched Aged at 1325ºF for 16 hours, air cooled	Strip	3×10^{-4} sec^{-1}	600ºF	96.5 ksi	(2-19)
8790	Solution annealed at 1800ºF for 1 hour, water quenched Aged at 1325ºF for 16 hours, air cooled	Bar	3×10^{-5} sec^{-1}		113.5 ksi	
Unknown	Solution annealed at 1088ºF (982ºC) for 1 hour, oil quenched Aged at 1325ºF for 16 hours, air cooled	Bar	Unknown	600ºF	~92 ksi	(2-20)

2.4 *Summary and Conclusions*

The effect of irradiation on the material properties and deformation limits of the reactor vessel internals at all three units at Oconee is acceptable for an 72 EFPY lifetime such that the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits under faulted condition loadings.

2.5 *References*

2-1. BAW-10008, Part 1, Revision 1, “Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake,” June 1970. Duke Energy Letter Incorporating BAW-10008, Part 1, Revision 1 into the Oconee Current Licensing Basis (CLB), ADAMS Accession Number ML19312B713. AEC letter to Duke regarding approved SER for BAW-10008, Part 1, Revision 1, ML19319B162.

2-2. BAW-2248A, “Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,” March 2000, ADAMS Accession Number ML003708443

2-3. Letter from M. S. Tuckman (Duke Energy), to U. S. Nuclear Regulatory Commission Document Control Desk “License Renewal, Response to NRC Letter dated November 18, 1999, Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287”, ADAMS Accession Number ML993620451, December 17, 1999.

2-4. NUREG-1723, “Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287”, ADAMS Accession Number ML003695154, March 2000.

2-5. Letter from T. Preston Gillespie (Duke Energy) to U. S. Nuclear Regulatory Commission Document Control Desk, “Duke Energy Carolinas, LLC, Oconee Nuclear Station (ONS), Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, License Renewal Commitment to Submit a Time Limited Aging Analysis for the Reactor Vessel Internals to the NRC for Review,” ADAMS Accession Number ML12053A332, February 20, 2012.

2-6. Letter from John Boska (U. S. NRC) to Preston Gillespie (Duke Energy), “Oconee Nuclear Station, Units 1, 2, and 3 – Approval of Time-Limited Aging Analysis for Reactor Vessel Internals (TAC NOS. ME8436, ME8437, ME8438),” ADAMS Accession Number ML13045A489, February 19, 2013.

2-7. ANP-3542P Revision 1 (103-3542P-001), “Time-Limited Aging Analysis (TLAA) Regarding Reactor Vessel Internals Loss of Ductility for Davis-Besse Nuclear Power Station, Unit No. 1 at 60 Years,” March 2017. ADAMS Accession Number ML17026A007, Davis-Besse Nuclear Power Station Unit 1 Regarding MRP-227-A Applicant Action Item 8: Time-Limited Aging Analysis (TLAA) Regarding Reactor Vessel Internals Loss of Ductility at 60 Years

2-8. Letter from Blake Purnell (U. S. NRC) to Mark B. Bezilla (FirstEnergy), “Davis-Besse Nuclear Power Station, Unit No. 1 - Time-Limited Aging Analysis for the Reactor Vessel Internals Loss of Ductility at 60 Years (CAC NO. MF9126, EPID L-2017-LRO-0002),” ADAMS Accession Number ML17256B190, October 12, 2017.

2-9. BAW-1621, “B&W 177-FA Owners Group – Effects of Asymmetric LOCA Loadings – Phase II Analysis,” July 1980, ADAMS Accession Number ML19320B058.

2-10. BAW-1621, Supplement 1, “B&W 177-FA Owners Group – Effects of Asymmetric LOCA Loadings – Phase II Analysis – Supplement 1 Responses to NRC Questions,” June 1981, ADAMS Accession Number ML20009B628.

2-11. Letter from John F. Stolz, NRC to Mr. H. B. Tucker, Duke Power Company, “Safety Evaluation Report on Asymmetric LOCA Loads, Oconee 1, 2, and 3,” NRC Legacy Accession Number 8305310083, May 20, 1983, ADAMS Accession Numbers ML15238A802 and ML15238A804

2-12. Materials Reliability Program: Development of a Material Constitutive Model for Irradiated Austenitic Stainless Steels (MRP-135, Revision 2). EPRI, Palo Alto, CA: 2019. 300213216.

2-13. “Information in Support of the EPRI Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0) Review,” ADAMS Accession Number ML103090248, October 29, 2010.

2-14. NUREG/CR-7027 (ANL-10/11), “Degradation of LWR Core Internal Materials due to Neutron Irradiation,” ADAMS Accession Number ML102790482, December 2010.

2-15. “High Strain Rate Tensile Properties of AISI Type 304 Stainless Steel,” J. M. Steichen, Journal of Engineering Materials and Technology, Volume 95, Issue 3, pp 182-185, July 1973.

2-16. J. M. Steichen, Hanford Engineering Development Laboratory Report HEDL-TME 75-81, UC-70 h, p, “The Effect of Strain Rate, Thermal Aging, and Irradiation on the Tensile Properties of A-286,” March 1976.

2-17. Westinghouse Astronuclear Laboratory Report WANL-TME-2791, “Neutron Irradiation Effects on the Tensile Properties of Inconel 718, Waspaloy and A-286,” June 1971.

2-18. W. J. Mills and L. A. James, “Fracture Mechanics Behavior of Neutron Irradiated Alloy A-286,” 1983.

2-19. L. A. James, Hanford Engineering Development Laboratory Report HEDL-TME 75-82, UC-79 b, h, “The Effect of Temperature on the Fatigue-Crack Propagation Behavior of A-286 Steel,” January 1976.

2-20. R. F. Muraca and J. S. Whittick, Western Applied Research & Development, Prepared for National Aeronautics and Space Administration, “Materials Data Handbook Stainless Steel Alloy A-286,” June 1972.

3.0 REACTOR VESSEL INTERNALS FLOW-INDUCED VIBRATION ENDURANCE LIMITS, BAW-10051 (SLRA SECTION 4.7.1.2)

3.1 *Introduction*

BAW-10051, Revision 1 and BAW-10051A, Supplement 1 (References 3-1, 3-2) calculated stress values for the ONS reactor vessel internals for 40 years and compared them to endurance limit (stress) values. This methodology was extended from 40 years to 60 years in Section 4.5.1 of BAW-2248A (Reference 3-3) by conservatively increasing the number of cycles and determining the endurance limit using the latest ASME fatigue curves. In the SER related to the License Renewal of the Oconee stations for 60 years (NUREG-1723, pages 4-23 and 4-24) (Reference 3-4), the staff notes that the TLAAs applicable to the ONS reactor vessel internals are addressed in BAW-2248A which include:

- Flow-induced vibration endurance limit assumptions
- Transient cycle count assumptions for the replacement bolting
- Reduction in fracture toughness

The finding in NUREG-1723 relative to the flow-induced vibration endurance limit is as follows.

> "The flow-induced vibration fatigue limit assumptions were based on 10^{12} cycles for 40 years. The analysis was extended into the period of extended operation for license renewal by conservatively increasing the number of cycles to 10^{13}, then determining the endurance limit using the latest ASME fatigue curves. The component stress values were found to be less than the endurance limit, rendering the evaluation acceptable, according to the requirements of 10 CFR 54.21(c)(1)(ii)."

Based on the above, the NRC staff concluded that the licensee's evaluation of the TLAA for flow-induced vibration endurance limit assumptions is acceptable. Therefore, this 60-year TLAA must be re-evaluated for subsequent license renewal.

3.2 *Methodology*

The source references that were used to justify the B&W Reactor Pressure Vessel internals flow-induced vibration endurance limit assumptions for 60 years are reviewed and updated for applicability to 80 years. The rationale and methods that are applied to SLR for 80 years are identical to those previously considered for 60 years but augmented with additional justification to conservatively address environmentally-assisted fatigue (EAF) affects utilizing the EAF criteria developed in NUREG/CR-6909, Revision 1 (Reference 3-5). In addition, the more limiting fatigue curves published in the 2013 edition of the ASME Section III code (Reference 3-6) (compared to the 1986 edition used with the ONS LRA for 60 years) (Reference 3-7) are considered with the fatigue evaluation of the ONS RV internals.

3.3 *Evaluation (Update of FIV Endurance Limit TLAA for 80 years)*

3.3.1 Environmentally-Assisted Fatigue Affects

3.3.1.1 RV Internals (Austenitic Steels)

The fatigue curves published in the 2013 ASME Code for austenitic steels (Figure I-9.2) do not consider environmental effects. Appendix “A” of NUREG/CR-6909 provides an acceptable method to determine the environmental correction factor for austenitic steels. Per Section A.2 (Equation A.13) of NUREG/CR-6909, the environmental correction factor (F_{en}) is equal to 1.0 if the strain amplitude (ε) is less than 0.10% (or 28.3 ksi stress amplitude). The summary of stress resulting from FIV shows the maximum amplitude of stress is equal to [ ] ksi (Table 3-3). Therefore, [

]

3.3.1.2 RV Internal Bolts (High Strength Bolt Steel)

The RV internal bolts have an axial preload. Provided the structural members jointed by the bolts do not separate during the cyclic loads, the bolt preload (in conjunction with the compressive clamping force imparted to the structural members by the bolt) greatly reduces the cyclic loading imparted to the bolts that result from high cycle FIV loadings created by the response of the RV internals to random turbulence. This source of flow excitation is relatively weak and thus, separation of the mating surfaces is not possible. As such, the cyclic strain amplitude (ε) resulting from FIV loadings in the bolts is **[ ]** than 0.10% (or less than 28.3 ksi stress amplitude), as shown in Table 3-3. Per Section A.2 (Equation A.13) of NUREG/CR-6909, the environmental correction factor (F_{en}) is equal to 1.0 if the strain amplitude (ε) is less than 0.10% (or stress amplitude less than 28.3 ksi). Thus, the environmental correction factor (F_{en}) for the bolts is equal to **[ ]**. Therefore, **[ ]**

3.3.2 ASME Fatigue Curves for Design Analysis (80 Years)

3.3.2.1 RV Internals

The ASME fatigue curves for austenitic steel were extended from 10^6 cycles to 10^{11} cycles in the 1983 edition of the ASME Section III code (Reference 3-8). The "extended" fatigue curves "A", "B" and "C" are depicted in Figure I-9.2.2 of the aforementioned ASME code year. The 2013 edition of the ASME Section III code eliminated fatigue curves "A" and "B" of Figure I-9.2.2 (or Figure I-9.2 in 2013 edition). Therefore, only one fatigue curve (curve "C") is provided in the 2013 edition of the ASME code. The values from curve "C" are illustrated in Table 3-1 of this report in the column labeled "SLR for 80 years". Table 3-1 shows that the allowable alternating stress for austenitic steels is 13.6 ksi at 10^{11} cycles. This allowable cyclic stress is conservatively extrapolated to **[ ]** ksi at 10^{13} cycles based upon the decay rate of **[ ]**% decay per decade of cycles used in the previous decade (10^{10} to 10^{11} cycles).

3.3.2.2 RV Internal Bolts

As shown in Table 3-2, the number of cycles and allowable stress associated with the fatigue curves for the high strength steel bolting (Figure I-9.4) published in the ASME Code has not changed over the years of interest. The largest number of cycles published in the 2013 ASME code for this material is 10^6. The allowable cyclic stress (S_a) for cycles greater than 10^6 were extrapolated to 10^{13} cycles based upon the decay rate of **[ ]** consistent with the methodology applied in BAW-10051, Revision 1 and BAW-10051A, Supplement 1 and for the ONS LRA for 60 years.

3.3.3 Fatigue Usage Factor for RV Internals

In BAW-10051, Revision 1 and BAW-10051A, Supplement 1, 10^{12} cycles was postulated for the 40 year plant life. For an 80 year plant life, the number of cycles to be postulated is 2.0 x10^{12} cycles. However, 10^{13} cycles is conservatively evaluated herein. A multiplication factor of [ ] is considered for the thermal adjustment of the fatigue curve (e.g., the modulus of elasticity at temperature is approximately [ ] smaller than the one at room temperature). The environmental fatigue correction factor of [ ] is also considered (Section 3.3.1). The allowable cyclic stress for 80 years of operation is determined below.

For the RV Internals:

Allowable cyclic stress = [ ] ksi at 10^{13}
(per Table 3-1)

For the RV Internal Bolts:

Allowable cyclic stress = [ ] ksi at 10^{13}
(per Table 3-2)

Table 3-3 provides a comparison of the alternating stress value (S_a) for the ONS RV internals and bolting components. The component with the least margin reported in this table has a stress of [ ] psi for the upper support bolts of the thermal shield and is smaller than the allowable cyclic stress of [ ] psi. Therefore, at least [ ] margin (or a safety factor of [ ]) exists between the maximum FIV stress and the allowable cyclic stress from the fatigue curve. Since this stress is below the endurance limit ([ ] psi at 10^{13} cycles), the maximum fatigue usage factor associated with FIV for the RV internals is effectively equal to [ ] for an 80 year life.

Table 3-1
ASME Fatigue Curve for Austenitic Steels (RV Internals)

Number of Cycles	Allowable Cyclic Stress: $Sa_{(allowable)}$ Units: psi (0-peak)	
	SLR for 80 years	% Decay
10^1	870.0	N/A
10^2	287.0	67.0%
10^3	108.0	62.4%
10^4	53.4	50.6%
10^5	28.4	46.8%
10^6	18.3	35.6%
10^7	14.4	21.3%
10^8	14.1	2.1%
10^9	13.9	1.4%
10^{10}	13.7	1.4%
10^{11}	13.6	0.73%
10^{12}	[ ]	[ ]
10^{13}	[ ]	[ ]

Table 3-2
ASME Fatigue Curve for High Strength Steel Bolts (RV Internal Bolts)

Number of Cycles	**Allowable Cyclic Stress: $Sa_{(allowable)}$ Units: psi (0-peak)**	
	SLR for 80 Years	**% Decay**
10^{1}	1150.0	N/A
10^{2}	320.0	
10^{3}	100.0	
10^{4}	34.0	
10^{5}	19.0	
10^{6}	13.5	
10^{7}		
10^{8}		
10^{9}		
10^{10}		
10^{11}		
10^{12}		
10^{13}		

Table 3-3
Summary of FIV Stress Results for the SLR for 80 Years

RV Internal Part	Sa (BAW-10051)	$Sa_{(allowable)}$	$Sa_{(allowable)}$ / Sa
Incore Instrumentation Nozzle	[ ]	[ ]	[ ]
Incore Instrumentation Guide Tubes			
Cantilevered Portion below flow distributor	[ ]	[ ]	[ ]
Between flow distributor and support plate	[ ]	[ ]	[ ]
Between support plate and spider casting	[ ]	[ ]	[ ]
Gusset welds	[ ]	[ ]	[ ]
J-groove welds	[ ]	[ ]	[ ]
Flow Distributor			
Ligament stresses	[ ]	[ ]	[ ]
Support plate ledge area	[ ]	[ ]	[ ]
Flow distributor assembly to lower grid assembly	[ ]	[ ]	[ ]
Flow Distributor Assembly Support Plate			
Ligament stresses	[ ]	[ ]	[ ]
Thermal Shield			
Upper support bolts	[ ]	[ ]	[ ]
Upper support blocks	[ ]	[ ]	[ ]
Lower support bolts	[ ]	[ ]	[ ]
Surveillance Holder Tube	[ ]	[ ]	[ ]
Inlet Baffle	[ ]	[ ]	[ ]
Redesigned Surveillance Specimen Holder Tubes			
Bracket	[ ]	[ ]	[ ]
Tube	[ ]	[ ]	[ ]
Capsule	[ ]	[ ]	[ ]

Notes:

(1) All stresses are in units of psi (0-peak)

(2) Portions of the SSHT were removed from the RV internals of ONS-1, ONS-2, and ONS-3 in 1976 due to structural damage that resulted from FIV (Section 3.3.2 of ANP-3186P, Rev 2) (Reference 3-9). The upper portion of the SSHT assembly attached to the core support shield was left intact. The majority of the lower portion of the SSHT assembly attached to the thermal shield was removed, leaving only brackets, pins, etc. (e.g., the tube was removed). Therefore, the FIV results for the "tube" part of the SSHT are not determined for the SLR for 80 years.

3.4 *Summary and Conclusions*

Table 3-3 provides a summary of the FIV stress associated with the Oconee RV internals and bolting items. The item with the least margin reported in this table has a stress for the thermal shield upper support bolts that is smaller than the allowable cyclic stress. Therefore, a safety factor exists between the maximum FIV stress and the allowable cyclic stress from the fatigue curve for this limiting location. Since this stress is below the endurance limit, the maximum fatigue usage factor associated with FIV for the RV internals is below the endurance limit for an 80 year life.

Therefore, the FIV integrity of the Oconee RV internals and bolting is deemed acceptable for SLR for 80 years in accordance with 10 CFR 54.21(c)(1)(ii). The high cycle fatigue affects associated with FIV during the subsequent period of extended operation (60 to 80 years) and the concerns associated with the environmental fatigue affects have no impact upon the BAW-10051, Revision 1 and BAW-10051A, Supplement 1 analyses and thus have been demonstrated not to be detrimental to the FIV performance of the Oconee RV internals and bolting.

3.5 *References*

3-1. BAW-10051, Revision 1, “Design of Reactor Internals and Incore Instrumentation Nozzles for Flow-Induced Vibration,” September 1972, revised in November 1972, Acceptability of BAW-10051, Revision 1 ADAMS Accession Number 19316A566.

3-2. BAW-10051A, Supplement 1, “Structural Analysis of 177-FA Redesigned Surveillance Specimen Holder Tube”, ADAMS Accession Number ML19248D133, 7908020516 (legacy)

3-3. BAW-2248A, “Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,” March 2000, ADAMS Accession Number ML003708443

3-4. NUREG-1723, “Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287”, ADAMS Accession Number ML003695154, March 2000.

3-5. NUREG/CR-6909 (Revision 1), “Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials,” May 2018.

3-6. 2013 ASME Boiler and Pressure Vessel Code, Section III, Appendices.

3-7. 1986 ASME Boiler and Pressure Vessel Code, Section III, Appendices.

3-8. 1983 ASME Boiler and Pressure Vessel Code, Section III, Appendices.

3-9. ANP-3186P, Revision 2, “ONS Licensing Renewal Scope and MRP-189, Rev 1 Comparison,” August 2013, ADAMS Accession Number ML13275A297

4.0 REACTOR VESSEL INTERNALS REPLACEMENT BOLTING METAL FATIGUE (SLRA SECTION 4.3.2.2)

4.1 *Introduction*

As described in BAW-2248A (Reference 4-1), Sections 2.0 and 4.5.1, the reactor vessel internals were designed and constructed prior to the development of ASME Code requirements for core support structures. Because of the lack of specific ASME design rules for core support structures at the time of design and construction of the ONS reactor vessel internals, Section III of the ASME code was used as a guideline for the design criteria for the reactor vessel internals. Qualification of the internals was accomplished by both analytical and test methods. The only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue reported in BAW-10051, Revision 1 and BAW-10051A, Supplement 1 (References 4-2, 4-3), which is a TLAA and is addressed in Section 3.0.

Inservice inspection at the three Oconee nuclear plants during 1981 and 1982 revealed failed bolts in the joint fastening the lower end of the reactor vessel thermal shield to the lower grid assembly. These bolts were made of A-286 material. The failed bolts were replaced with X-750 bolting and fatigue analyses were performed for the replacement bolts as reported in BAW-1843PA (Reference 4-4). These topical reports summarize fatigue analyses performed to the ASME code (Section III, Subsection NG) including both high cycle fatigue (FIV) and low cycle fatigue (design transients). The fatigue evaluations for internal replacement bolting reported in BAW-1843PA are TLAAs that were evaluated in BAW-2248A for 60-years and must be addressed for subsequent license renewal.

4.2 *Methodology*

The total usage factor was recalculated for the replacement bolting items by considering both the existing design transient usage factors, which are acceptable for 80-years, and by the recalculation of fatigue usage factors due to flow-induced vibration. The current CUFs of replacement bolting items are acceptable since 40-year design cycles (CLB cycles) are postulated to bound 80 years of plant operation. The methods that are applied to SLR for 80 years are identical to those previously considered for 60 years considering the fatigue analysis presented in BAW-1843PA for the replacement bolting items.

4.3 *Evaluation (Update of Metal fatigue CUF for 80 years)*

4.3.1 Environmentally-Assisted Fatigue Effects

As discussed in Section 3.3.1.2, the environmental correction factor (F_{en}) is equal to 1.0 if the strain amplitude (ε) is less than 0.10% (or stress amplitude less than 28.3 ksi). The cyclic stress in the replacement bolts resulting from FIV is conservatively calculated to be 1.07 ksi (Section 4.3.3), which is less than 28.3 ksi. Thus, the environmental correction factor (F_{en}) for the replacement bolts is equal to **[ ]** Therefore, **[**

]

4.3.2 ASME Fatigue Curves for Design Analysis

The fatigue curve for the RV internal replacement bolts shown in Table 4-1 has not been modified since its original determination. Therefore, reconciliation of differences associated with the number of cycles and alternating stress is not necessary for this fatigue curve as has been performed for the fatigue curves of the original bolts in Section 3.3.2.2.

4.3.3 Fatigue Usage Factor for the Replacement Bolts

The fatigue analysis calculation below for 80 years adheres to the analytical methodology applied for 40 years and 60 years and follows Crandall's method (Reference 4-5) for a random stress process.

The usage factor for low cycle fatigue is [ ]. This value is the same for 40 years since the RCS transients are unchanged from a 40 year life.

The allowable FIV usage factor is therefore 1.0 - $UF_{low\ cycle}$ = [ ].

The number of FIV cycles at [ ] for 80 years is [ ].

Where [ ] is based on a modal analysis of the thermal shield.

The design fatigue curve for Inconel X-750 bolt material is reported in Table 4-1 and can be mathematically defined by the following expression:

$$(N_{cycles})(S_a)^b = c$$

where:

b = [ ] (Constant for X-750 fatigue curve for cycles between 10^9 and 10^{11})

c = [ ] (Constant for X-750 fatigue curve for cycles between 10^9 and 10^{11})

S_a = Allowable alternating / cyclic stress for a given number of cycles.

The usage factor for a random stress process when determined using Crandall's method can be expressed as:

$$UF = \frac{N_{cycles}}{c}\left(\sqrt{2} * S_a\right)^b \left(\Gamma\left((1+\frac{b}{2}\right)\right)$$

$$\log(UF) = \log\left(N_{cycles}\right) - \log(c) + b * \log\left(\sqrt{2}\right) + b * \log\left(S_a\right) + \log\left(\Gamma\left(1+\frac{b}{2}\right)\right)$$

Solving for the log(S_a) term yields;

$$\log\left(S_a\right) = \left(\frac{1}{b}\right) * \left[\log(UF) - \log\left(N_{cycles}\right) + \log(c) - b * \log\left(\sqrt{2}\right) - \log\left(\Gamma\left(1+\frac{b}{2}\right)\right)\right]$$

Substituting the variables into the above equation and considering Γ = [ ] yields:

S_a = [ ] rms for 80 years

The stress due to FIV (S_{FIV}) for the replacement bolts = [ ] rms.

The margin is thus S_a / S_{FIV} = [ ].

UF_{FIV} = [ ] for 80 years

$UF_{cumulative} = UF_{FIV} + UF_{low\_cycle}$ = [ ]

Since the FIV stress for the Oconee replacement lower thermal shield to lower grid assembly bolts is less than the 80 year allowable FIV stress, and the 80-year CUF that includes consideration of FIV is less than 1.0, these bolts will be acceptable for the period of SLR.

Table 4-1
Fatigue Curve for RV Internal Replacement Bolts (Inconel X-750 Material)

Number of Cycles	Allowable Cyclic Stress: $Sa_{(allowable)}$ Units: ksi (0-peak)
4×10^3	
10^4	
2×10^4	
4×10^4	
10^5	
2×10^5	
4×10^5	
10^6	
2×10^6	
4×10^6	
10^7	
10^8	
10^9	
10^{10}	
10^{11}	

4.4 *Summary and Conclusions*

The stress for the replacement bolts (lower thermal shield to lower grid assembly) is smaller than the allowable cyclic stress. As such, a safety factor exists between the maximum FIV stress and the allowable cyclic stress from the fatigue curve. Therefore, the FIV integrity of the Oconee RV replacement bolting is deemed acceptable for SLR for 80 years.

4.5 *References*

4-1. BAW-2248A, “Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,” March 2000, ADAMS Accession Number ML003708443.

4-2. BAW-10051, Revision 1, “Design of Reactor Internals and Incore Instrumentation Nozzles for Flow-Induced Vibration,” September 1972, revised in November 1972, Acceptability of BAW-10051, Revision 1, ADAMS Accession Number 19316A566

4-3. BAW-10051A, Supplement 1, “Structural Analysis of 177-Fuel Assembly Redesigned Surveillance Specimen Holder Tube," ADAMS Accession Number ML19248D133, 7908020516 (legacy)

4-4. BAW-1843PA, “Evaluation of Internals Bolting Concerns in 177 FA Plants,” January 1986, ADAMS Accession Number ML20197G640.

4-5. S. H. Crandall and W. D. Mark, “Random Vibration in Mechanical Systems,” 1963.

5.0 REACTOR VESSEL INTERNALS FLUENCE/DPA AT 72 EFPY (SLRA SECTION 4.7.1.3 AND XI.M16A)

5.1 *Introduction*

In accordance with Section 2.0 above, 72 EFPY Oconee-specific fluence/dpa estimates were used to assess susceptibility to irradiation embrittlement relative to the BAW-10008, Part 1, Revision 1, TLAA evaluation for the following reactor vessel internals component items.

- Baffle plates (Category 2 item)
- Plenum cover (Category 3 item)
- Plenum cylinder reinforcing plate (Category 3 item)
- Core support shield top flange (Category 3 item)

The 72 EFPY fluence/dpa values used to evaluate these items are reported in Table 5-1. The fluence/dpa values used for the Oconee gap analysis (GALL SLR XI.M16A), which is based on MRP-189 Revision 3 and was developed to bound the B&W fleet of reactors, is reported in Table 5-2. Oconee-specific DORT and MCNP 72 EFPY evaluations reported herein were developed to demonstrate that the fluence/dpa input to the TLAA evaluation reported in Section 2.0 above and the fluence/dpa input to MRP-189 Revision 3 used to develop the Oconee-specific gap analysis are acceptable for the subsequent period of extended operation.

5.2 *Methodology*

Framatome's Oconee-specific evaluation of the RVI included both deterministic discrete ordinate transport (DORT) and Monte Carlo N-Particle (MCNP) codes. The purpose of these calculations is to verify that the projected 72 EFPY fluence values developed for the update of BAW-10008, Part 1, Revision 1 (Section 2) and the fluence/dpa input to MRP-189 Revision 3 used to develop the Oconee-specific gap analysis are acceptable. The DORT evaluation followed the modeling procedures defined in Framatome's NRC approved methodology described in BAW-2241PA, Revision 2, "Fluence and Uncertainty Methodologies." There were no additional modeling considerations to address the inherent limitations of the DORT code. Instead, an additional analysis was performed using Framatome's hybrid 3D Deterministic-Monte Carlo methodology, called SVAM (Reference 5-5). The hybrid model include full three dimensional detail of the reactor internals components of interest. **[**

]

5.3 *Oconee Specific 72 EFPY Reactor Internals Fluence/dpa*

This section lists the key inputs and modeling approximations in the Oconee-specific DORT and MCNP RVI fluence calculations. Both analyses followed a best-estimate, rather than a bounding or conservative fluence methodology. Oconee-specific values were used for all of the key inputs, including the power level, axial and radial power distributions, materials, geometry definition, and operating history.

[

]

5.3.1 DORT 2-D Calculation of RVI Neutron Exposure to 72 EFPY

The Oconee-specific DORT calculation utilized Framatome's NRC approved fluence methodology. To determine the three-dimensional (3D) fluence rate (flux) and fluence, two separate 2-dimensional (2D) models are combined in accordance with the methodology presented in BAW-2241PA. In a manner consistent with previous fluence analyses, the Oconee reactor is modeled in 2D planar cylindrical coordinates (Rθ) and 2D axial cylindrical coordinates (RZ).

[

]

[

]

The synthesized fluence rates provide the basis for calculating the fluence and dpa results based on appropriate response functions. For fluence, the response function includes only fluence rates with energies > 1.0 MeV. However, the dpa response function extends to the thermal region. It is assumed that the fluence impact of a 1.64% MUR will be bounded by a 2% increase in fluence rate. This is accounted for by multiplying the resultant Cycle 26 flux and dpa rate values by 1.02.

A cross sectional view of the DORT model is provided in Figure 5-1. [

]

Figure 5-1
Sketch of the Oconee RVI DORT model

5.3.2 MCNP Calculation of RVI Neutron Exposure to 72 EFPY

The Oconee-specific MCNP model was developed to address known deficiencies in DORT RVI fluence calculations. For example,

1. The reactor internals are homogenized in the DORT model. A study performed by Oak Ridge National Lab found that RVI homogenization leads to a systemic under prediction of the fast flux, particularly at locations above and below the core.
2. The neutron flux spectrum used to collapse the iron isotopes in steel in the BUGLE-96 library is based on a fine-group (VITAMIN-B6) solution at ¼ of the RV thickness, effectively at the core midplane. The spectrum in the internals at locations above and below the core is likely to be different. Therefore, the BUGLE-96 cross section library may not be ideal for flux calculations outside of the beltline.
3. The BUGLE-96 cross section library only has 4 energy groups in the thermal range. This resolution may is not sufficient for dpa calculations. Particularly in locations far away from the core.
4. Symmetric quadratures may not be accurate in regions above and below the core. Higher order or biased quadrature sets may be more appropriate.

The Oconee MCNP model is three dimensional (3-D), with a heterogeneous representation of the reactor core and internals components. The inputs were selected to address the known deficiencies in the DORT model. [

]

Figure 5-2 through Figure 5-4 provide an example of the detail included in the MCNP model. [

]

Figure 5-2
View of Lower Grid Assembly and Core Barrel

[

]

Figure 5-3
View of Upper Vessel Base Metal

Figure 5-4
View of Incore Guide Support Plate

Although the MCNP model contains extremely precise representations of the reactor core and internals, model precision does not guarantee accurate results. Accuracy is a measure of how close the MCNP estimate of flux is to the “truth”. The difference between this true value and MCNP’s flux estimate is called the systematic error (or bias), which is seldom known. Error or uncertainty estimates for the results of Monte Carlo calculations refer only to the precision of the result and not to the accuracy. It is possible to calculate a highly precise result that is far from the physical truth because nature has not been modeled faithfully. For example, the exact coolant temperature is not known at all locations. Therefore, the MCNP model assumes that the temperature is constant over large regions of the RVI. Additionally, no reactor component is manufactured with perfect symmetry. However, the MCNP model assumes perfect symmetry of the reactor vessel and all internals component items. Typically, the error associated with modeling approximations is quantified by comparing the calculations to measurements that have a well-defined uncertainty. At the time this document was prepared, uncertainty and sensitivity calculations have not been performed for the MCNP (or the DORT) RVI fluence methodology.

5.3.3 Estimate of Uncertainty

The uncertainty for the RV internals is currently being evaluated through the PWROG (PA-MSC-1403R1). It is expected that consideration of uncertainty in the RV internals fluence will not impact the RV internals TLAA (BAW-10008, Part 1, Revision 1) or the gap analysis (XI.M16A). This is because there is a large margin between the fluence values used to support the ONS RV internals (TLAA BAW-10008, Part 1, Revision 1 and the XI.M16A gap analysis) and the ONS best estimate fluence values (DORT and MCNP). This margin is discussed in Sections 5.4 and 5.5.

5.4 *Oconee TLAA (BAW-10008, Part 1, Revision 1) Fluence Margin*

The fluence values used to support the TLAA in Section 2.0 were intended to be [ ] These values have significant margin relative to the Oconee-specific best estimate fluences calculated for 80 years (72 EFPY) by the DORT and MCNP methodologies, which are discussed above in Sections 5.3.1 and 5.3.2, respectively.

The margin between the Oconee-specific best estimate fluence values and the values used to support the TLAA in Section 2.0 are shown in Table 5-3. The ONS 72 EFPY values reported in Table 5-3 are the maximum considering the DORT and MCNP best estimate values. The margin (or the allowed percent difference) is very large far from the core and is lower close to the core. This is because the uncertainty in the best estimate fluence values is expected to increase with increasing distance from the core. Therefore, more conservative fluence values were used in the TLAA for component items farther from the core. For component items like the baffle plates and the bottom of the plenum cylinder that are closer to the core, the uncertainty in the Oconee-specific best estimate fluence values is expected to be low, so the margins allowed for these component items should remain adequate.

5.5 *Oconee Gap Analysis (XI.M16A) Fluence Margin*

The fluence values used to support the gap analysis (XI.M16A) were intended to be [ ] These values have significant margin relative to the Oconee-specific best estimate fluences calculated for 80 years (72 EFPY Table 5-2) by the DORT and MCNP methodologies, which are discussed above in Sections 5.3.1 and 5.3.2, respectively.

The margin between the Oconee-specific best estimate fluence values and the values used to support the gap analysis (GALL SLR XI.M16A) are shown in Table 5-4. Note that, since there are hundreds of component items in the RV internals, only the lowest margin component item for each assembly are included in the table. The margin (or the allowed percent difference) is very large far from the core and is lower close to the core. This is because the uncertainty in the best estimate fluence values is expected to increase with increasing distance from the core. Therefore, more conservative fluence values were used in the gap analysis for component items farther from the core. For component items close to the core, the uncertainty in the Oconee-specific best estimate fluence values is expected to be low, so the margins allowed for these component items should remain adequate. In addition, while exceeding the screening criteria may change the preliminary categorization, it may not actually impact the inspection recommendations in the gap analysis (GALL SLR XI.M16A).

5.6 *Summary and Conclusions*

The 72 EFPY fluence/dpa is reported in Table 5-1. The fluence/dpa used for the Oconee gap analysis (GALL SLR XI.M16A), which is based on MRP-189 Revision 3 and was developed to bound the B&W fleet of reactors, is reported in Table 5-2. Oconee-specific DORT and MCNP 72 EFPY evaluations reported herein demonstrate that the fluence/dpa input to the TLAA evaluation reported in Chapter 2 above and the fluence/dpa input to MRP-189 Revision 3 used to develop the Oconee-specific gap analysis are acceptable for the subsequent period of extended operation. Reactor vessel internals fluence shall be monitored for the Baffle plates (Category 2 item), Plenum cover (Category 3 item), Plenum cylinder reinforcing plate (Category 3 item), and Core support shield top flange (Category 3 item) in accordance NUREG-2191, X.M2, Neutron Fluence Monitoring to ensure that the TLAA reported in Section 2.0 remains valid.

Table 5-1
Projected Fast Fluence (E > 1.0 MeV) of Reactor Vessel Internals Component Items in Support of TLAA in Section 2.0

[ ]

Table 5-2
Projected 72 EFPY, 80 EFPY Fluence and dpa, MRP-189 Revision 3, used to Support the Gap Analysis

Table 5-3
Margin between Projected Fast Fluence (E > 1.0 MeV) of Reactor Vessel Internals Component Items in Support of TLAA in Section 2.0 and the ONS-Specific Best-Estimate 72 EFPY Fluence Values

[ ]

Table 5-4
Margin between Projected Fast Fluence (E > 1.0 MeV) of Reactor Vessel Internals Component Items in Support of the Gap Analysis and the ONS-Specific Best-Estimate 72 EFPY Fluence Values

5.7 *References*

5-1. BAW-10008, Part 1, Revision 1, “Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake,” Duke Energy Letter Incorporating BAW-10008, Part 1, Revision 1 into the Oconee Current Licensing Basis (CLB), ADAMS Accession Number ML19312B713. AEC letter to Duke regarding approved SER for BAW-10008, Part 1, Revision 1, ML19319B162.

5-2. BAW-2241PA, Revision 2, “Fluence and Uncertainty Methodologies”, ADAMS Accession Number ML073310655 (Proprietary) / ML073310660 (Non-Proprietary)

5-3. Materials Reliability Program: Screening, Categorization, and Ranking of Babcock & Wilcox–Designed Pressurized Water Reactor Internals Component Items and Welds (MRP-189, Revision 3). EPRI, Palo Alto, CA: 2019. 3002013218. ADAMS Accession Number ML20091K286, February 2020.

5-4. Materials Reliability Program: PWR (Pressurized Water Reactor) Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175, Revision 1). EPRI, Palo Alto, CA: 2017. 3002010268. ADAMS Accession Number ML17361A187, December 28, 2017.

5-5. ANP-10348P, “Fluence Methodologies in SLR,” 2020, ADAMS Accession Number ML20223A019.

ENCLOSURE 4
ATTACHMENT 3

OCONEE NUCLEAR STATION
SLR-ONS-TLAA-0306NP, Revision 0,
“Environmentally-Assisted Fatigue Oconee Subsequent License Renewal Application Supplemental Section 4.3.4”

OCONEE NUCLEAR STATION

UNITS 1, 2, and 3

SLR-ONS-TLAA-0306NP

Environmentally-Assisted Fatigue
Oconee Subsequent License Renewal Application
Supplemental Section 4.3.4

Revision 0

[NON-PROPRIETARY]

NOTE: Text that is within brackets is proprietary to Duke Energy or Framatome. The subscripts D or F respectively are used to identify the appropriate company.

LIST OF ABBREVIATIONS

ABBREVIATION	DEFINITION
ANSI	American National Standard Institute
AOR	Analysis of Record
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CUF	Cumulative Usage Factor
DO	Dissolved Oxygen
EAF	Environmentally-Assisted Fatigue
EPRI	Electric Power Research Institute
FAD	Functional Area Document
IPA	Integrated Plant Assessment
NUREG	U.S Nuclear Regulatory Commission technical report designation
NUREG/CR	Contractor-prepared NUREG
OSC	Oconee Calculation
PWR	Pressurized Water Reactor
PZR	Pressurizer
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
ROTSG	Replacement Once-Through Steam Generator
RV	Reactor Vessel
SLR	Subsequent License Renewal
TLAA	Time Limited Aging Analysis
U_{EN}	Usage considering Environmental Effects
UFSAR	Updated Final Safety Analysis Report
USAS	United States of America Standards

INTRODUCTION

The purpose of this document is to provide supplemental information to the non-proprietary information reported in the subsequent license renewal application for Oconee Nuclear Station Units 1, 2, and 3 for environmentally-assisted fatigue (EAF). Table 4.1-2 of NUREG-2192, Standard Review Plan of Subsequent License Renewal Applications for Nuclear Power Plants, provides the generic time-limited aging analyses (TLAA) and metal fatigue (Section 4.3) contains environmentally-assisted fatigue as a subsection. This report includes information that will assist the NRC review of environmentally-assisted fatigue, but could not be included in the Oconee subsequent license renewal application, which is non-proprietary.

NOTE: The following section supplements Section 4.3.4 of the Oconee Subsequent License Renewal Application.

ENVIRONMENTALLY-ASSISTED FATIGUE (EAF)

TLAA Description:

As outlined in Section X.M1 of NUREG-2191 (Reference 7) and Section 4.3 of NUREG-2192 (Reference 8) the effects of the reactor water environment on cumulative usage factor (CUF) must be examined for a set of sample critical components for the plant. This sample set includes the locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Reference 3) and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260. These additional limiting locations are identified through an environmental fatigue screening evaluation. The environmentally-assisted fatigue (EAF) screening process made use of existing fatigue usage values for the ASME Code, Section III components. The EAF screening evaluation reviewed the CLB fatigue evaluations for all ASME Code, Section III reactor coolant pressure boundary components and piping and ANSI B31.7 piping, including the NUREG/CR-6260 locations, to determine the lead EAF indicator (also referred to as Sentinel) locations for EAF. As a result of the EAF screening evaluation, there were other locations found that could potentially be more limiting than the NUREG/CR-6260 locations.

TLAA Evaluation:

To support subsequent license renewal, calculations were prepared to document the evaluations of environmentally-assisted fatigue for ASME Code, Section III pressure boundary components and piping and ANSI B31.7 piping that contact the reactor coolant and determine fatigue-sensitive locations for comparison and ranking. These evaluations are for subsequent license renewal purposes and do not amend the existing design reports. Discussion of the screening approach used for the ASME Code, Section III components and piping and ANSI B31.7 piping are provided below. A consolidated tabulation for pressure boundary components and piping is presented for the Sentinel locations in Table 4.3.4-1.

<u>EAF Screening Methodology</u>

As background, for initial License Renewal, EAF was evaluated only for the NUREG/CR-6260 locations applicable to ONS. To ensure that any additional plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than those considered in NUREG/CR-6260 are addressed, EAF screening was performed for all ASME Code, Section III components with existing fatigue usage values, including B31.7 piping which was reanalyzed using ASME Code Section III during the steam generator replacement project. These were reviewed and categorized into the following five groups of major components listed below.

6. Reactor Vessel (RV) and sub-components

7. Reactor Coolant Pumps (RCP's) and sub-components

8. Steam Generators (ROTSG's) – primary side

9. Pressurizer (PZR) and sub-components

10. RCS Loop and Branch Piping

Screening F_{en} factors were developed for each component so that EAF usage (U_{en}) can be calculated and compared. NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels" [Reference 9] was used for carbon and low-alloy steels. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," [Reference 10] was used for stainless steels and NUREG/CR-6909, Revision 0, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, Final Report" [Reference 11] was used for Ni-Cr-Fe alloys.

Conservative values are chosen for each of the F_{en} input parameters: sulfur content, service temperature, strain rate, and dissolved oxygen (DO). EPRI Technical Report, "Pressurized Water Reactor Primary Water Chemistry Guidelines" [Reference 12] is followed. A value of 0.005 ppm for dissolved oxygen (DO) content when the system temperatures are elevated is the action level 1 value for pressurized water reactors (PWR) environmental conditions under normal operation. Therefore, the F_{en} calculations use a value below the lowest threshold value for the dissolved oxygen (DO) content. For NUREG/CR-6583 and NUREG/CR-5704 this value is 0.05 ppm and for NUREG/CR-6909 this value is 0.04 ppm. Sample dissolved oxygen (DO) information from Oconee plant operation provides a maximum DO less than 10 ppb (0.010 ppm) during heatups and cooldowns.

For the periods during heatup/cooldown operations when DO may be elevated, pressurizer temperature is ≤ 250°F (121°C). During these times, the system ΔT between the pressurizer and RCS is low (≤ 80°F), and pressurizer insurge/outsurge transient events with this magnitude of system ΔT are not significant contributors to fatigue. For pressurizer temperatures above 250°F, oxygen scavenger addition is used to control dissolved oxygen. For cooldowns, H_2 concentration in the RCS is used to minimize dissolved oxygen in the RCS through the pressurizer steam bubble collapse. Furthermore, fluid temperatures during the elevated DO times are in the range where service temperature values are the lowest. Therefore, the use of 0.05 ppm and 0.04 ppm for DO content is acceptable for the F_{en} evaluations.

Any location that was not part of the Class 1 reactor coolant boundary was removed from consideration. Other locations were also excluded during this step. These locations included locations not in contact with primary coolant, locations excluded fatigue usage factor calculation based on fatigue waivers (ONS SLRA, Section 4.3.2.9) and locations where the CUF was so low that applying the maximum possible F_{en} factor results in a U_{en} less than 0.8.

For the components and subcomponents where the NUREG/CR-6260 locations have the highest screening U_{en} no additional locations were considered for EAF. For those components and subcomponents where the NUREG/CR-6260 locations do not have the highest screening

U_{en} or a NUREG/CR-6260 location does not exist, the locations within the components and subcomponents that have the highest screening U_{en} are the Sentinel locations. The final set of sentinel locations are meant to supplement those identified in NUREG/CR-6260, resulting in a comprehensive list of plant-specific sentinel locations for EAF consideration.

The screening process for the sentinel locations were identified as follows:

- NUREG/CR-6260 locations that were analyzed for EAF as part of license renewal were included as sentinel locations.
- Prescreening was performed where a location was eliminated if the location has a usage factor that is so low that, even if the maximum possible environmental fatigue correction factor (F_{en}) were applied, the resulting EAF usage (U_{en}) would be less than 0.8. The location is not exposed to reactor water, and therefore EAF is not applicable. The location is not part of the primary pressure boundary and is therefore excluded from the EPRI screening process.
- A bounding F_{en} is calculated for each material type using NUREG/CR-6583 (for carbon/low alloy steels), NUREG/CR-5704 (for stainless steels), and NUREG/CR-6909 (for Ni-Cr-Fe).
- Piping and/or vessel components that undergo essentially the same thermal and pressure transients during plant operations are grouped into thermal zones. Within each material type, the location with the highest U_{en}* (estimated U_{en}) is selected during initial screening; the second location with the second highest U_{en}* is also selected if the top two U_{en}* values are within a factor of two. However, if the third highest U_{en}* value is within 25% of the highest U_{en}* value within a thermal zone, then the top three locations in that thermal zone are selected.
- To reduce excess conservatism for stainless steel location due to the very large maximum F_{en}, an estimated F_{en} (F_{en}*) is calculated as the average of the value based on a qualitative estimate of strain rate, and the value based on the worst possible strain rate, using the same values of DO and estimated upper bound temperature for design transients in both cases.
- Additional screening was performed to determine that, within any system, a location can be considered bounded if all of the following criteria are true:
  - The F_{en} for the bounded location is smaller than that for the bounding location.
  - The U for the bounded location is smaller than that for the bounding location.
  - The U_{en} for the bounded location is less than half of that for the bounding location.
  - The bounded location is not a NUREG/CR-6260 location.

Initial Oconee screening was completed in 2014 using NUREG/CR-6909, Revision 0. This work was done as part of first steps in expanding the initial LR NUREG/CR-6260 work. In 2019 for SLR, the F_{en} values calculated as part of the initial screening were compared to the calculation methods proposed in Revision 1 of NUREG/CR-6909. It was concluded that revising the screening methodology would result in a similar ranking. Furthermore, conservatism added as part of the initial screening for F_{en} values for stainless steel were removed as part of the analysis for 80-years of operation.

Assessment of Sentinel Locations

For locations where the screening U_{en} exceeded 1.0, further evaluations were performed using NUREG/CR-6909, Revision 1, “Effect of LWR Water Environments on the Fatigue Life of

Materials, Final Report" [Reference 11] was used. This approach was applied to the following components:

- CRDM Weld
- Bottom Mounted Instrumentation (BMI) Weld
- Core Flood Nozzles
- Loop Drain 2B2/3B2 piping
- Pressurizer Surge Line Piping
- Pressurizer Surge Nozzle Weld Overlay
- Pressurizer Spray Piping, pt. 190
- HPI Piping, Stop Vlv-to-Check (bounds HPI Nozzles)
- Decay Heat Drop Lines
- Core Flood Piping

Sentinel locations are those locations chosen for more detailed analysis, monitoring, inspection, or replacement. These locations are chosen to have bounding U_{en} values compared with other locations. Except for the HPI Nozzles where the HPI piping bounded the nozzle, NUREG/CR-6260 locations were retained, regardless of the revised calculated U_{en}. The results of the EAF calculations for the sentinel locations are summarized in Table 4.3.4-1. In addition to the U_{en} values, the original analysis of record (AOR) CUF, the reduced CUF developed for SLR, and a brief summary of the conservatisms removed from the AOR for Safety Class 1 components are also provided in Table 4.3.4-1. Components with a U_{en} less than unity do not require fatigue management per GALL-SLR. Transients associated with components with a U_{en} less than unity will be monitored by the *Fatigue Monitoring* AMP (ONS SLRA, Appendix B3.1).

Appendix L Flaw Tolerance Evaluations

For two of the sentinel piping locations (Pressurizer Surge Line and HPI piping), the effects of fatigue will be managed by application of the Inservice Inspection Program (*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* program (ONS SLRA, Appendix B2.1.1)) during the subsequent period of extended operation based on results of flaw tolerance evaluations conducted per the guidance of ASME Code, Section XI [Reference 13], Nonmandatory, Appendix L. NUREG-2192 permits inspections as a management method for fatigue as long as a flaw tolerance evaluation is performed to determine the acceptable time between inspections. The ASME Code, Section XI, Appendix L crack growth evaluation is used in conjunction with calculated allowable flaw sizes to determine the required inspection interval for a postulated flaw in the piping at the bounding location. For a postulated initial flaw, crack growth is simulated until the flaw has reached the allowable flaw depth or the end of the subsequent period of extended operation, whichever comes first.

The purpose of the flaw tolerance evaluation is to establish an appropriate inspection frequency that is consistent with the typical 10-year inservice inspection program. The input used from each of the piping systems (geometry, transient cycles and definitions, material properties, piping loads, etc.) bounds Units 1, 2 and 3. The ASME Code, Section XI, Appendix L flaw tolerance evaluation consists of postulating a hypothetical inside surface axial and circumferential flaw. There are two Sentinel locations analyzed using ASME Code, Section XI, Appendix L presented in Table 4.3.4-2.

For the branch line piping, the transients used to simulate growth of the postulated flaws in the ASME Code, Section XI, Appendix L evaluations used ten years of projected cycles. For the pressurizer surge line elbow, as discussed in Section 4.3.1, a reduced set of allowable transient cycles were used for the ten years of projected cycles. Following re-inspection, the cycle counts used in the ASME Code, Section XI, Appendix L evaluation are set to zero when no flaw is disclosed for each new inspection interval. The ASME Code, Section XI, Appendix L evaluations used ten years of projected cycles and the branch line piping will be inspected on a ten-year inspection frequency. The selection of transients is conservative, thereby ensuring the inspection frequencies remain adequate. The *Fatigue Monitoring* AMP (ONS SLRA, Appendix B3.1) tracks significant transient cycles for the Safety Class 1 components. The Corrective Action Program tracks specific activities such as HPI injection and insurge and outsurge cycles pertaining to the surge line.

The maximum allowable end-of-evaluation period flaw size was determined based on the acceptance criteria and evaluation procedures in ASME Code, Section XI, Appendix C, 2007 Edition with 2008 Addenda which is the current code of record. Based on previous inspection records, there are no detected indications at these locations; therefore, the methodology of ASME Code, Section XI, Appendix L can be used. As per ASME Code, Section XI, Appendix L, a postulated flaw size larger than the ASME Code, Section XI acceptance standards in Table IWB-3514-2 was used in the fatigue crack growth (FCG) analysis. The piping systems evaluated here are constructed from stainless steel material, where the only significant crack growth mechanism of consideration is fatigue crack growth.

Based on the fatigue crack growth evaluation, the allowable operating period was determined as the length of time it takes for the postulated initial flaw size to grow to the maximum allowable end-of-evaluation period flaw size. The fatigue crack growth analysis was completed using the crack growth rates from ASME Code, Section XI, Code Case N-809, "Reference Fatigue Crack Growth Rate Curves for Austenitic Stainless Steels in Pressurized Water Reactor Environments" [Reference 14]. The results of the ASME Code, Section XI, Appendix L evaluations are provided in Table 4.3.4-2.

The ASME Code, Section XI, Appendix L inspections will be conducted by the Inservice Inspection (*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* program (ONS SLRA, Appendix B2.1.1)) program. The ASME Code, Section XI, Appendix L inspections are identified in Table 4.3.4-3. Each weld in the inspection population will be volumetrically inspected using the code required techniques once prior to establishing the Inspection Interval schedule for Units 1, 2, and 3 ASME Code, Section XI, Appendix L locations. Going forward after the first ultrasonic inspection, one weld in each of the piping lines will be ultrasonically inspected every 10 years.

<u>Loop Drain Piping</u>

As shown in Table 4.3.4-1, the loop drain 2B2/3B2 piping contains a U_{en} that is less than unity. The alternating stress associated with the MRP-146 loading are low stress high cycle fatigue and do not result in an EAF penalty factor provided the alternating is at or below the threshold of 28.3 ksi. This is discussed in MRP-146, Revision 2, Appendix F and is defined in NUREG/CR-6909, Revision 1. In Appendix F of MRP-146, a sample B&W 2.5" drain line was evaluated and shown to have an alternating stress of 22.37 ksi (< 28.3 ksi). Therefore, there is no EAF penalty expected for the drain lines.

Oconee currently has these drain lines scoped in for MRP-146 and will continue inspection under the MRP-146 management program for the subsequent period of operation. These inspections are currently part of the Augmented Inservice Inspection (AISI) program.

TLAA Disposition: 10 CFR 54.21(c)(1)(iii)

The effects of fatigue on the intended function(s) of ASME Code, Section III pressure boundary components and piping and ANSI B31.7 piping that contact reactor coolant will be adequately managed by the *Fatigue Monitoring* aging management program (ONS SLRA, Appendix B3.1) and the *ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD* aging management program (ONS SLRA, Appendix B2.1.1) through the subsequent period of extended operation.

Table 4.3.4-1 Sentinel Locations

System / Thermal Zone	Location	Material	AOR[4] CUF	SLR CUF	F_{en}	U_{en}	Analysis Method	Fatigue Management Method
RC / RC transients only	CRDM weld(1)	Ni-Cr-Fe Alloy	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	0.772	The CRDM weld is part of the RVCH replacement. The SLR CUF of **[ ]**$_{D:a,b,c}$ is based on reduced Power Loading/Unloading cycles. Power loading/unloading cycles is excluded from the Fatigue Monitoring program, which will require reconsideration if ONS implements flexible power operation.	None Required(5)
	Bottom Mounted Instrumentation (BMI)(1)	Ni-CR-Fe Alloy	**[ ]**$_{F}$	**[ ]**$_{F}$	**[ ]**$_{F}$	0.744	Recalculated CUF_{in-air} using appropriate stress ranges and number of operating transient cycles using NUREG/CR-6909 fatigue curves and utilizing maximum F_{en}.	None Required(5)
	Rx Vessel Shell Lower Head(2)	LAS	**[ ]**$_{F}$	**[ ]**$_{F}$	**[ ]**$_{F}$	0.756	Used the bounding CUF_{in-air} and applied the maximum F_{en}.	None Required(5)
	Rx Vessel Inlet and Outlet Nozzles(2)	LAS	**[ ]**$_{F}$	**[ ]**$_{F}$	**[ ]**$_{F}$	0.832	Recalculated CUF_{in-air} using appropriate stress ranges and number of operating transient cycles using NUREG/CR-6909 fatigue curves and utilizing maximum F_{en}	None Required(5)
	Core Flood Nozzles(2)	LAS	**[ ]**$_{F}$	**[ ]**$_{F}$	**[ ]**$_{F}$	0.882	Recalculated CUF_{in-air} using appropriate stress ranges and number of operating transient cycles using NUREG/CR-6909 fatigue curves and utilizing maximum F_{en}	None Required(5)

System / Thermal Zone	Location	Material	AOR[4] CUF	SLR CUF	F_{en}	U_{en}	Analysis Method	Fatigue Management Method
	Loop drain 2B2/3B2 piping(3)	SS	[ ]D:a,b,c	[ ]D:a,b,c	[ ]D:a,b,c	0.51	Usage is due to thermal stratification caused by turbulent penetration, which has an alternating stress below the threshold value of 28.3 ksi giving a F_{en} = [ ]D:a,b,c	*ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD* program (B2.1.1)
RC / PZR lower head and surge line	Pressurizer heater penetration weld	SS	[ ]F	[ ]D:a,b,c	[ ]D:a,b,c	0.723	Used service temperature of 550°F to calculate F_{en} with AOR alternating stress of 68.83 ksi to calculate usage from NUREG/CR-6909 fatigue curve.	None Required(5)
	Pressurizer surge line piping(2)	SS	[ ]D:a,b,c	[ ]D:a,b,c	[ ]D:a,b,c	1.28	Appendix L	*ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD* program (B2.1.1)
	Pressurizer Hot Leg surge nozzle weld overlay (path 17)	SS	[ ]D:a,b,c	[ ]D:a,b,c	[ ]D:a,b,c	0.702	Refined evaluation to develop updated alternating stresses and apply a F_{en} of 1 to load pairs where alternating stress is below the threshold value of 28.3 ksi. Max F_{en} = [ ]D:a,b,c	None Required(5)
	Pressurizer surge nozzle weld overlay (Path 3)	SS	[ ]D:a,b,c	[ ]D:a,b,c	[ ]D:a,b,c	0.979	Refined evaluation to develop updated alternating stresses and apply a F_{en} of 1 to load pairs where alternating stress is below the threshold value of 28.3 ksi. Max F_{en} = [ ]D:a,b,c	None Required(5)
RC/PZR spray	Pressurizer spray piping, pt. 190 (Aux. Spray Tee)	SS	[ ]D:a,b,c	[ ]D:a,b,c	[ ]D:a,b,c	0.816	Used service temperature of 557°F to calculate F_{en} with AOR alternating stresses to calculate usage from NUREG/CR-6909 fatigue curve.	None Required(5)

System / Thermal Zone	Location	Material	AOR4 CUF	SLR CUF	F_{en}	U_{en}	Analysis Method	Fatigue Management Method
High Pressure Injection(2)	HPI piping, stop valve-to-check (usage bounds HPI Nozzle)	SS	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	3.38	Appendix L	*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD* program (B2.1.1)
Decay Heat Removal System	Decay Heat Drop Lines(2)	SS	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	0.755	Refined evaluation using calculated strain rate and a service temperature of 500°F to calculate F_{en} with AOR alternating stresses to calculate usage from NUREG/CR-6909 fatigue curve.	None Required(5)
Core Flood	Core Flood Piping	SS	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	**[ ]**$_{D:a,b,c}$	0.752	Refined evaluation using a service temperature of 450°F to calculate F_{en} with AOR alternating stresses to calculate usage from NUREG/CR-6909 fatigue curve.	None Required(5)

Notes:

1. Installed equipment is not original. Since the CRDM Weld is not original equipment the next highest location was the RV Instrumentation (BMI) weld. These locations also bound steam generator Ni-Cr-Fe alloy locations.
2. NUREG/CR-6260 location.
3. The usage presented for drain line is based on MRP-146 turbulence penetration behavior which is a low stress amplitude/high cycle loading. Per NUREG/CR-6909 Rev. 1, no EAF penalty would apply to alternating stress < 28.3 ksi. The fatigue damage occurs under normal operation, therefore, the 60-year usage of **[ ]**$_{D:a,b,c}$ is adjusted by 80/60 x **[ ]**$_{D:a,b,c}$ = 0.51.
4. Analysis of Record.
5. U_{en} less than unity. No fatigue management required per GALL-SLR. Transients for this location are monitored in the Fatigue Monitoring program.

D. Duke Energy Proprietary
F. Framatome Proprietary

Table 4.3.4-2 ASME Section XI Appendix L Results

Location	Flaw Configuration	Interval (Years)	Acceptable Standards Flaw Size Table IWB-3514-2 (a/t)(2)	Final Flaw Size (a/t)(3)	Maximum Allowable End-of-Evaluation Flaw Size (a/t)(4)	Allowable Operating Period (Years)
Pressurizer Surge Line Piping (elbow cheek)	Axial	10	0.15	0.383	0.623	10(5)
	Circumferential(1)	10	0.15	0.187	0.750	10(5)
Pressurizer Surge Line Piping (weld)	Axial	10	0.15	0.157	0.623	10(5)
	Circumferential(1)	10	0.15	0.192	0.407	10(5)
HPI piping, stop valve-to-check valve (usage bounds HPI nozzle)	Axial	10	0.15	0.2322	0.75	69
	Circumferential(1)	10	0.15	0.1894	0.56	178

Notes:

1. Aspect ratio (AR) is limited to 1/8 for semi-elliptical circumferential flaw, so a full 360 degree circumferential flaw model is conservatively used when the initial aspect ratio is less than 1/8 limits = constant as the crack grows through the wall thickness.
2. Initial postulated flaw size which is based on ASME Code, Section XI Table IWB-3514-2 for an aspect ratio of 6. The methodology of the initial flaw size is based on ASME Code, Section XI Appendix L-3210.
3. The final flaw size based on fatigue crack growth per ASME Code Case N-809 with a constant aspect ratio. The aspect ratio for the FCG is determined per ASME Code, Section XI Appendix L.
4. The maximum allowable end-of-evaluation flaw size is determined per ASME Code, Section XI Appendix C. The final flaw size after fatigue crack growth should be less than the maximum allowable end-of-evaluation flaw size.
5. The Appendix L interval of 10 years was evaluated as the allowable operating period.

Table 4.3.4-3 Appendix L Inspections

Location	Line	Weld No.	Last ISI Inspection
Pressurizer Surge Line Piping - U1	Pressurizer Surge Line Piping	• Butt weld PSL-6 • Butt weld PSL-7 • Cheeks of base metal elbow "B" (bounded by welds PSL-6 and PSL-7)	O1R29 (F2016)(1)
HPI Piping, stop valve-to-check valve - U1	High Pressure Injection Piping • Loop 1B1 • Loop 1B2	Stop valve-to-check valve butt weld • Loop 1B1: Weld 165 • Loop 1B2: Weld 285	O1R30 (F2018)(2)
Pressurizer Surge Line Piping - U2	Pressurizer Surge Line Piping	• Butt weld PSL-6 • Butt weld PSL-7 • Cheeks of base metal elbow "B" (bounded by welds PSL-6 and PSL-7)	O2R28 (F2017)(1)
HPI Piping, stop valve-to-check valve - U2	High Pressure Injection Piping • Loop 1B1 • Loop 1B2	Stop valve-to-check valve butt weld • Loop 2B1: Weld 165 • Loop 2B2: Weld 285	O2R29 (F2019)(2)
Pressurizer Surge Line Piping - U3	Pressurizer Surge Line Piping	• Butt weld PSL-6 • Butt weld PSL-7 • Cheeks of base metal elbow "B" (bounded by welds PSL-6 and PSL-7)	O3R29 (S2018)(1)
HPI Piping, stop valve-to-check valve - U3	High Pressure Injection Piping • Loop 1B1 • Loop 1B2	Stop valve-to-check valve butt weld • Loop 3B1: Weld 155 • Loop 2B2: Weld 250	O3R30 (S2020)(2)

Note(s):

1. The Pressurizer Surge Line inspections were previously established to meet GL 88-11.
2. The HPI Piping stop valve-to-check valve weld inspection was previously established to meet GL 88-08 commitments and supplemented by RT.

REFERENCES:

1. 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants
2. OSC-2077-03-SLR-0500, Time-Limited Aging Analysis Identification for Subsequent License Renewal
3. NUREG/CR-6260 (INEL-95/0045), Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, March 1995. (ML031480219)
4. OSS-0254.00-00-4015, Rev. 3, (MECH) Design Basis Specification for Oconee Piping Classification
5. ONS UFSAR, Revision 28
6. SLR-ONS-TLAA-0300, Rev. 1, Oconee Nuclear Station TLAA of Mechanical System Thermal Fatigue for SLR
7. NUREG-2191, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report, Revision 0.
8. NUREG-2192, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants, Revision 0.
9. NUREG/CR-6583 (ANL-97/18), Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels, February 1998.
10. NUREG/CR-5704, Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels, April 1999.
11. NUREG/CR-6909, Rev. 0, Effect of LWR Water Environments on the Fatigue Life of Materials, Final Report.
12. EPRI Technical Report TR3002000505, “Pressurized Water Reactor Primary Water Chemistry Guidelines,” Volume 1, Revision 7, Palo Alto, CA, April 2014.
13. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Nonmandatory Appendix L, Operating Plant Fatigue Assessment.
14. ASME Code Case N-809, Reference Fatigue Crack Growth Rate Curves for Austenitic Stainless Steel in Pressurized Water Reactor Environments, Section XI, Division 1, ASME International, dated June 23, 2015
15. NUREG/CR-6909, Rev. 1, Effect of LWR Water Environments on the Fatigue Life of Materials, Final Report.
16. Framatome document ANP-3898P/NP, Framatome Reactor Vessel and RCP TLAA and Aging Management Review Input to the ONS SLRA