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Revision 0

U.S. EPR Fuel Assembly Mechanical Design Topical Report

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AREVA NP Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
AOO	Anticipated Operational Occurrence
BOL	Beginning of Life
CHF	Critical Heat Flux
CFM	Centerline Fuel Melt
CRGA	Control Rod Guide Assembly
CS	Cold Shutdown
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EG	End Grid
EOL	End of Life
FIV	Flow-Induced Vibration
FP	Full Power
GWd/mtU	Gigawatt-Day Per Metric Ton Uranium
HFP	Hot Full Power
HS	Hot Shutdown
HTP	High Thermal Performance
HMP	High Mechanical Performance
HZP	Hot Zero Power
ID	Inside Diameter
IFM	Intermediate Flow Mixer
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feed Water
LOOP	Loss of Offsite Power
LTL	Lower Tolerance Limit
MFW	Main Feed Water
MSIV	Main Steam Isolation Valve
MSMG	Mid-Span Mixing Grid
GWd/mtU	Gigawatt-Days Per Metric Ton Uranium
OBE	Operational Base Earthquake
OD	Outside Diameter
OL3	Olkiluoto 3, Finland
PICS	Process Instrumentation And Control System
PIE	Post-Irradiation Examination
ppm	Parts Per Million
psi	Pounds Per Square Inch
PWR	Pressurized Water Reactor
QD	Quick Disconnect
RCCA	Rod Cluster Control Assembly
RCPB	Reactor Coolant Pressure Boundary

Acronym	Definition
RCS	Reactor Coolant System
RMS	Root Mean Square
RT	Reactor Trip
RXA	Recrystallized Annealed
SG	Spacer Grid
SRSS	Square Root of the Squares Sum
SSE	Safe Shutdown Earthquake
SSPB	Secondary Side Pressure Boundary
SO	Stretch-Out
TD	Theoretical Density
UO2	Uranium Dioxide
UTL	Upper Tolerance Limit

1.0 INTRODUCTION

AREVA NP has developed a 17x17 fuel assembly design for use in U.S. EPR nuclear power plants. This deign is based on the collective experience from the Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report (Reference 3) and high thermal performance (HTP) fuel assembly designs detailed in the Generic Mechanical Design Criteria for PWR Design (Reference 6) and the Generic Mechanical Design Report High Thermal Performance Spacer reports (References 20), The assembly is referred to as the U.S. EPR fuel assembly in this document. This report provides an evaluation of the U.S. EPR fuel assembly performance in relation to the general criteria defined in Section 4.2 of NUREG-0800 (Reference 1). A set of specific criteria consistent with Section 4.2 of the SRP has been previously established in AREVA NP topical reports reviewed and approved by the NRC.

This report is divided into three major sections (Sections 3.0, 4.0, and 5.0), each addressing a significant aspect of the U.S. EPR fuel assembly, focusing on new features and similarities to existent designs. Section 3.0 describes the U.S. EPR design, highlighting the distinguishing features. Section 4.0 describes relevant operating experience applicable to the U.S. EPR design. Section 5.0 describes the fuel assembly and fuel rod example evaluations that address the performance of the U.S. EPR fuel assembly based on the criteria in Reference 1.

Section 6.0 describes a design change process that will be used to determine if follow-on designs beyond the initial cycle 1 fuel should be submitted to the NRC for review and approval in addition to referencing this topical report. Section 6.0 will also be used to determine if small follow-on design changes to the initial cycle 1 U.S. EPR fuel assembly design can be made without specific NRC review and approval. Examples of such changes are outlined in Section 6.0.

2.0 SUMMARY

The U.S. EPR fuel assembly contains design features similar to those of the proven Advanced Mark-BW and HTP fuel designs that serve as bases for comparison. Reference 3 was reviewed and approved by the NRC for generic use. Many of the design criteria and methodologies approved by the NRC for the Advanced Mark-BW and HTP designs are also applicable to the U.S. EPR fuel design. The applicability of Reference 3 design evaluation criteria and methodologies to the U.S. EPR is contained in the NRC-approved Codes and Methods Applicability Topical Report (Reference 2).

Additionally, the U.S. EPR fuel assembly design is based on the EPR base fuel design for the Olkiluto 3 (OL3) reactor with the following adjustments:

- Axial blanket fuel rods to reduce axial neutron leakage and improve fuel economy.
- Design power histories considering 18 and 24 month fuel cycles.
- Gadolinia fuel rods with 2, 4, 6, and 8 wt% Gd_2O_3.
- Varying radial and axial fuel enrichments to control power peaking.
- 0.3g U.S. safe shutdown earthquake (SSE) spectra.

Other features that have been previously approved for AREVA designed fuel assemblies and incorporated into the U.S. EPR fuel assembly include:

- M5™ clad fuel rods to increase resistance to corrosion and to reduce hydrogen uptake and reduce fuel rod irradiation growth.
- M5™ HTP spacer grids to improve corrosion resistance and reduce irradiation growth.

- Inconel 718 high mechanical performance (HMP) spacer grids to reduce cell relaxation during irradiation and increase strength.

- Low pressure drop quick disconnect (QD) top nozzle that uses a leaf spring holddown system and a low pressure drop nozzle structure.

- Robust FUELGUARD™ bottom nozzle that provides highly effective debris resistance with good flow characteristics and an acceptable pressure drop.

- M5™ MONOBLOC™ guide tubes with two inside diameters (ID) (for the upper region and the dashpot) and a single outside diameter (OD). This feature of the MONOBLOC™ guide tubes provides a robust cross-section to minimize fuel assembly distortion, while also providing rapid insertion of the control rod cluster and a dashpot region that provides rod cluster deceleration and acceptable impact loads on the top nozzle. QD sleeves are attached to the upper end of the guide tubes for connection to the top nozzle.

- Leaf holddown springs.

- Welded cage design for improved fuel assembly bow performance.

Features new to the U.S. EPR fuel assembly design include:

- Active fuel height of 13.8 ft.

- Interface with heavy reflector and incore detection system.

Extensive operating experience and data of AREVA pressurized water reactor (PWR) fuel throughout the world, using the M5™ alloy, provides the design bases for consistent irradiation performance and models used for the U.S. EPR fuel assembly design. The M5™ alloy and associated models and methods are addressed in the Evaluation of Advanced Cladding and Structural Material in PWR Reactor Fuel Topical Report (Reference 7) and the Incorporation of M5™ Properties in Framatome ANP Approved

Methods Report (Reference 8). The Safety Evaluation Report for the Extended Burnup Evaluation Report (Reference 9) limits the Mark-BW fuel rod burnup to 60 GWd/mtU. The justification for increasing the burnup limit to 62 GWd/mtU is provided in the Extended Burnup Evaluation Report, Revision 2 (Reference 10). The U.S. EPR fuel assembly is designed to achieve a peak fuel rod burnup of 62 GWd/mtU, which is consistent with the burnup limits in Reference 10.

Based on the results of comprehensive empirical testing and design evaluation analyses, the U.S. EPR fuel assembly design is demonstrated acceptable for batch and full core implementation in the U.S. EPR.

3.0 U.S. EPR FUEL DESIGN DESCRIPTION

The U.S. EPR fuel assembly uses a 17x17 rod array specifically developed for the U.S. EPR. Figure 3-1 highlights the primary design features of the U.S. EPR fuel assembly. Table 3-1 provides comparisons of basic fuel assembly parameters of the U.S. EPR fuel assembly to the Advanced Mark-BW and HTP 17x17 fuel assembly designs.

3.1 Fuel Assembly

The U.S. EPR fuel assembly is a 17x17 array of fuel pins that have been designed specifically for use with the core configuration of the U.S. EPR reactor. This configuration does not require a central instrument tube location in the assembly because of the insertion of instrumentation lances into a small subset of guide tube locations (see Figure 3-2). The U.S. EPR fuel assembly contains 265 fuel rods with a fuel rod in the center of the array.

The U.S. EPR fuel assembly uses 10 spacer grids that (along with the 24 guide tubes and associated connections, and the top and bottom nozzles) provide the structural cage or skeleton for supporting the 265 fuel rod assemblies. The top and bottom spacer grids are the HMP design, constructed of Inconel 718 strip material. The eight intermediate HTP spacer grids are constructed from M5™ strip material. The M5™ clad fuel rods are positioned **[ ]** above the bottom nozzle and are axially and laterally supported by the top and bottom HMP end spacer grids and the eight HTP intermediate spacer grids. The HMP and HTP spacer grids provide restraint along several lines of contact to support the fuel rod. Line contact provides a much larger bearing area than other types of spacer grids and, therefore, reduces the potential for fretting wear of the fuel rods.

The 24 guide tubes are the MONOBLOC™ design using M5™ alloy. The MONOBLOC™ design uses a constant OD with the dashpot features integral to the IDs. The intermediate M5™ HTP spacer grids are welded to each guide tube.

The Inconel HMP end grids are not welded directly to the guide tubes because of the difference in materials. Instead, they are axially constrained by M5™ alloy sleeves welded directly to each guide tubes above and below the corresponding grid positions, at all 24 guide tube locations.

The U.S. EPR top nozzle is constructed of 304 L stainless steel and accommodates the QD feature for the guide tube-to-top nozzle connections, which enable rapid removal and installation during fuel assembly reconstitution. The top nozzle also houses the holddown spring system, which consists of four sets of five-leaf springs made of Inconel 718 that are mounted to the top nozzle with Inconel 718 screws. The holddown springs maintain positive fuel assembly contact with the core support structure during normal operating conditions and provide positive holddown margins for precluding liftoff due to hydraulic flow forces while accommodating differential thermal expansion and irradiation growth of the fuel assembly.

The FUELGUARD™ bottom nozzle is a 304 L stainless steel brazement, which incorporates a series of parallel-curved blades that provide debris resistance by virtue of curved flow passages, allowing no direct line of sight through the nozzle, restricting the passage of debris but allowing coolant to pass through freely. The connection of the bottom nozzle to the 24 guide tubes is accomplished using 316 L stainless steel bolts that incorporate a mechanical locking feature.

Key fuel assembly dimensions establish compatibility with core and component interfaces.

3.2 Fuel Rod

The U.S. EPR fuel rod design consists of uranium dioxide (UO_2) pellets contained in a seamless M5™ alloy tube, with end plugs made from M5™ alloy barstock welded at each end. The design uses a fuel stack height of 165.354 inch. The fuel pellets have a diameter of 0.3225 inch. The fuel rod cladding has a 0.374 inch OD with a **[ ]** inch wall thickness. This configuration leaves a small radial clearance of approximately

[ ] between the ID of the cladding and the OD of the fuel pellets. The fuel rod uses a stainless steel spring in the upper plenum to prevent the formation of fuel stack axial gaps during shipping and handling, and which also allows fuel stack expansion during operation. The fuel column rests on a stainless steel support tube that sits on the lower end plug. The inner space within the support tube for the fuel column provides additional plenum volume in the fuel rod for reduced pressure due to fission gas retention. The upper end cap has a grippable shape to remove the fuel rods from the fuel assembly, if necessary.

The fuel rod cladding is M5^{TM} alloy. M5^{TM} cladding significantly increases resistance to corrosion and hydrogen uptake associated with longer cycles, high temperatures, and high burnup in comparison to early zircaloy constructions. The U.S. EPR fuel rod length and void volume provide adequate margin against failure due to pin internal pressure buildup. Table 3-2 shows a comparison of the fuel rod parameters for the U.S. EPR fuel assembly and the Advanced Mark-BW fuel assembly, which serves as the reference M5^{TM} rod design. Figure 3-3 shows an axial cross-section of the U.S. EPR fuel rod. Table 3-6 through Table 3-12 provide specific fuel rod design parameters.

The fuel pellets are sintered, high density ceramic. The fuel pellets are cylindrically shaped with a dish at each end. The corners of the pellets have outward land tapers (i.e., chamfers) that ease the loading of the pellets into the cladding and also reduce the chances of pellet chipping. The dish and taper geometry also prevents the pellets from forming an hourglass geometry during operation. The design density of the pellets is **[ ]** percent theoretical density (TD). Pellet density may increase as fabrication improvements occur allowing the potential for higher assembly loading. Pellet enrichments used for the U.S. EPR design may be as high as 5.0 weight percent U^{235}.

The U.S. EPR fuel rod design uses axial blankets and gadolinia fuel configurations similar to the Advanced Mark-BW fuel assembly, which can have up to five axial enrichment zones (Reference 3). Axially blanketed fuel rods for the U.S. EPR contain up to seven axial pellet enrichment zones depending on fuel cycle design specifics

([ ]) A fuel rod with the maximum number of axial zones would consist of the following:

- A central zone of enriched sintered UO_2 pellets or UO_2+Gd_2O_3 pellets.
- Two additional enriched zones; one above and one below the central zone.
- Two cutback zones of enriched sintered UO_2 pellets above and below the enriched zones.
- Two axial blanket zones, one at each end of the stack.

The axial blanket region consists of sintered UO_2 pellets with a low weight percent U^{235} enrichment. The fuel pellet may also use gadolinium, which serves as an integral burnable poison to control power peaking or core reactivity. Pellets containing gadolinium are typically not loaded into the cutback or blankets zones. The fuel rod dimensions are shown in Tables 3-2 and 3-6 through 3-12.

3.3 Spacer Grids

The U.S. EPR fuel assembly uses eight M5™ alloy HTP flow mixing spacer grids at the intermediate locations and HMP Inconel 718 alloy spacer grids at the top and bottom end locations of the assembly. AREVA NP received NRC approval of HTP spacer grids constructed of Zircaloy-4 in the Generic Mechanical Design Thermal Performance Spacer and Intermediate Flow Mixer Report (Reference 20). The U.S. EPR HTP intermediate spacer grids are M5™ alloy to improve corrosion resistance and reduce irradiation growth. The use of the M5™ alloy for spacer grids was previously approved by the NRC in see References 7 and 8. Table 3-3 provides a comparison of spacer grid parameters for the U.S. EPR fuel assembly, Advanced Mark-BW fuel assembly, and 12 foot 17x17 HTP fuel assembly designs.

The M5™ HTP spacer grid is constructed of pairs of die formed strips that interlock when they are assembled to form the overall HTP grid structure. Each cross strip is formed by resistance spot welding two stamped M5™ halves to form a subcomponent called a doublet. The assembled doublets form channels, slanted at the outlets, which induce a swirling pattern in the coolant flow as it passes through the HTP grid (see Figure 3-7). The channels are arranged so that there is no net torque on the fuel assembly. These channels also provide the integral springs and contact surfaces that hold the fuel rods in place. The channel strips are formed in the axial direction so that they provide a spring contact with the fuel rods in the mid-region of the spacer. At the inlet and outlet of the spacer, the channels (referred to as castellations) provide more rigid lateral constraint at a slight nominal clearance from the fuel rod. Sideplates are welded to the ends of the doublets to form the outer envelope of the grid. The sideplates are provided with top and bottom lead-in tabs to avoid assembly hang-up during fuel movement. Figure 3-7 shows the HTP grid assembly.

In addition to the HTP intermediate grids, HMP grids are used at the top and bottom end grid locations to provide additional support of the fuel rods. The HMP grids are made of low cobalt precipitation-hardened Inconel 718 that provides additional strength and reduced cell relaxation due to irradiation. The HMP spacer maintains line contact on the fuel rod similar to the HTP spacer. The lower relaxation provides fuel rod lateral support during operation for the design burnup range. The HMP spacer grid design is similar to the HTP spacer grid except the flow channels created by the doublets are straight. This minimizes the hydraulic resistance of the grid in the locations outside of the active fuel region where flow mixing is not needed. The HMP grid assembly is shown in Figure 3-8.

To establish axial alignment of spacer grids with adjacent fuel assemblies, the HTP grids are spot welded to the guide tubes. This limits grid axial movement after irradiation relaxation of the M5™ grids. Sleeves of M5™ are spot welded to the guide tubes above and below the HMP grids for axial location and restraint.

3.4 Low Pressure Drop Top Nozzle

The U.S. EPR fuel assembly design incorporates a low pressure drop top nozzle made of stainless steel. The low pressure drop feature is achieved by an optimization of flow path geometry with the nozzle structural integrity that accommodates each required normal and faulted load. The top nozzle design also incorporates a Quick Disconnect (QD) feature to attach to the 24 fuel assembly guide tubes, as shown in Figure 3-10. The primary features of the top nozzle include:

- Five leaf spring holddown system.
- Low pressure drop nozzle structure.
- QD guide tube attachment.

A representative QD guide tube assembly is shown in Figure 3-9. The design consists of a double-spline sleeve made of M5^{TM} alloy attached to the guide tube via multiple spot welds. The features in the top nozzle machining provide either clearance for removal, or restraint for securing the nozzle based on the orientation of QD features on the guide tube assemblies. The reconstitution tooling rotates the guide tube QD ring 90^{o} to lock or unlock the sleeve splines, and provides a positive lock when the ring rotation is complete.

The top nozzle assembly incorporates four sets of formed leaf springs made of Inconel 718 alloy fastened to the nozzle with Inconel 718 clamp screws captured in the nozzle body. During operation, the springs prevent fuel assembly lift due to hydraulic forces, while accommodating irradiation growth and thermal expansion. The upper leaf contains an extension that engages a cutout in the top plate of the nozzle. This arrangement provides spring leaf retention in the unlikely event of a spring leaf or clamp screw failure.

The top nozzle structure consists of a stainless steel frame that provides interfaces with the reactor upper internals, the core components, and fuel assembly handling tooling and equipment while providing coolant flow. The top nozzle flow-hole pattern enables an increased flow area, yielding a reduced pressure drop while satisfying the strength requirements for the top nozzle plate.

3.5 Debris Filter (FUELGUARD™) Bottom Nozzle

The FUELGUARD™ bottom nozzle, shown in Figure 3-4 and 3-5, provides a highly effective barrier to debris.

The bottom nozzle is stainless steel with a frame of deep ribs connecting the guide tube attachment bushings and conventional legs that interface with the lower reactor internals. The frame distributes the primary loads on the fuel assembly through the bottom nozzle. A set of parallel, curved blades as shown in Figure 3-5, are assembled by brazing into a frame structure. The blade spacing enables good flow characteristics while providing enhanced debris filtering. The lower end of the guide tubes contain threaded features that provide rigid connection of the guide tubes to the bottom nozzle with special stainless steel bolts that incorporate a mechanical locking feature.

The FUELGUARD™ lower tie plate (or grillage) is an effective barrier to debris with acceptable pressure drop. The pressure drop performance is equivalent to conventional debris filter designs.

3.6 M5™ Alloy Guide Tube

The MONOBLOC™ guide tubes are fabricated from M5™ alloy. The use of the M5™ alloy for guide tubes was previously approved (Reference 7 and 8). This material exhibits low corrosion and hydrogen uptake throughout the fuel design burnup ranges in addition to lower irradiation free growth. Table 3-4 shows a comparison of MONOBLOC™ and Advanced Mark-BW guide tubes parameters.

The MONOBLOC™ guide tube, as shown in Figure 3-9, has two inside diameters (ID) and a single OD. The larger ID at the top provides a relatively large annular clearance that permits rapid insertion of the rod cluster control assembly (RCCA) during a reactor trip and accommodates coolant flow during normal operation. The reduced ID section (i.e., the dashpot located at the lower end of the tube) provides a relatively close fit with the control rods to decelerate toward the end of the control rod travel. This deceleration limits the magnitude of the RCCA impact loads on the fuel assembly top nozzle. The guide tube wall thickness at the bottom is much greater in the dashpot region than at the upper end of the tube to maintain the same OD with the smaller ID. This design provides a more rigid tube and thus a more robust structure that helps to reduce fuel assembly distortion and bow.

Four small holes in the guide tube located just above the dashpot allow both outflow of water during RCCA insertion, and coolant flow to control components and instrumentation lances during operation. There is also a small flow hole in the guide tube bolt that enables flow through the reduced diameter section and flow venting during RCCA deceleration.

The QD sleeve is attached to the upper end of the guide tube for connection to the top nozzle (see Figure 3-12). At the lower end of an M5™ lower end plug is welded onto the end of the guide tube dashpot section. The lower end plug is internally threaded for engagement with the guide tube bolt that connects the guide tube to the bottom nozzle (see Figure 3-13).

3.7 Materials

Table 3-5 summarizes the materials used on the U.S. EPR fuel assembly design, identifying the alloys and the corresponding components. The specific use of M5™ for fuel rod cladding, guide tubes, and spacer grids has been approved by the NRC per References 7 and 8. Low cobalt material requirements are imposed where applicable to reduce radiation exposure levels.

Table 3-1 Comparison of U.S. EPR to Advanced Mark-BW and 12-Foot HTP 17 Fuel Assembly Parameters

Fuel Assembly Parameter	Advanced Mark-BW	U.S. EPR	12 Foot 17x17 HTP
Fuel assembly overall length, in w/o HD springs	[ ]	[ ]	[ ]
FA matrix	17x17	17x17	17x17
Fuel rod overall length, in	152.16	179.134	151.50
Fuel assembly envelope, in	8.425	8.426	8.426
Fuel rod pitch, in	0.496	0.496	0.496
Fuel rods / assembly	264	265	264
Guide tubes / assembly	24	24	24
Instrument tubes / assembly	1	0	1
Fuel rod cladding material	M5™	M5™	M5™
Guide tube material	M5™	M5™	M5™
Guide tube design	Standard dashpot GT	MONOBLOC™	Standard dashpot GT
Top nozzle	Low pressure drop multi-leaf spring	Low pressure drop multi-leaf spring	Low pressure drop multi-leaf spring
Top nozzle attachment	QD	QD	QD
Bottom nozzle	TRAPPER™ coarse mesh or fine mesh	FUELGUARD™	FUELGUARD™
End Grids	2 monometallic Inconel 718	2 Inconel 718 HMP	2 Bi-Met Zr4/Inconel 718
Intermediate grids	6 monometallic M5™	8 M5™ HTP	6 Zr4 HTP
Intermediate grid types	5 Mixing, 1 non-mixing	8 HTP	6 Zr4 HTP
Intermediate grid / guide tube attachment	Swaged, deflection limiting ferrules with initial gap, 8 guide tube locations	Spot welded to guide tubes	Spot welded to guide tubes
MSMGs	3 monometallic M5™	Not used for U.S. EPR design	3 Intermediate Flow Mixers

Table 3-2 Comparison of U.S. EPR and Advanced Mark-BW Fuel Rod Parameters

Fuel Rod Parameters	Advanced Mark-BW	U.S. EPR
Clad material	M5™ Alloy	M5™ Alloy
Fuel rod length, in	152.16	179.134
Cladding OD, in	0.374	0.374
Cladding thickness, in	0.0225	0.0225
Cladding ID, in	0.329	0.329
Clad-to-pellet gap, in	0.0033	0.0033
Fuel pellet OD, in	0.3225	0.3225
Plenum spring	Top	Top

Table 3-3 Comparison of U.S. EPR, Advanced Mark-BW, and 17x17 HTP Grid Parameters

Table 3-4 Comparison of U.S. EPR and Advanced Mark-BW Guide Tube and Instrument Sheath Parameters

Table 3-5 Summary of Component Materials

Alloy	Component
M5™	Fuel rod clad, guide tube
	Guide tube sleeves
	Fuel rod end caps, guide tube plugs
	HTP spacer grids
	QD sleeve
304 L stainless steel	Top and Bottom nozzle structures
316 L stainless steel	QD locking ring
	Guide tube screw
302 stainless steel	Fuel rod spring
Inconel 718	HMP spacer grids
	Holddown spring clamp screws
	QD ring
	Holddown spring leaves
Stainless steel	Fuel stack support tube
UO_2 and UO_2 + Gd_2O_3	Fuel pellets

Table 3-6 Fuel Rod Parameters

[Parameter Description	Parameter Information	Tolerance
Fuel Column Length, in.	165.354	[ ]
Overall Rod Length, in.	179.134	[ ]
Rod Internal Plenum Volume, in^3	[ ]	[ ]
Fill Gas Type	Helium [ ]	-
Fill Gas Pressure, psig	[ ]	[ ]
Fissile Enrichment	≤ 4.95 wt% U-235	[ ]
Burnup Limit, GWd/mtU	62	-

Table 3-7 Fuel Rod Cladding Parameters

Parameter Description	Parameter Information	Tolerance
Type and Metallurgical State of Cladding	M5™ - Recrystallized	-
Cladding OD, in.	0.3740	[ ]
Cladding Inside Diameter, in.	0.3291	[ ]
Cladding Inside Roughness, μin	45	-

Table 3-8 UO_2 Pellet Parameters

Parameter Description	Parameter Information	Tolerance
Pellet OD, in.	0.3225	[ ]
Roughness, μin	[ ]	-
Density, %Theoretical Density	[ ]	[ ]
Resinter Densification Limits (24-hour test)	[ ]	-
Length, in.	0.531	[ ]
Dish Volume (for one pellet), mm^3	[ ]	]
Dish Diameter (minimum), in.	[ ]	-
Dish Depth (minimum), in.	[ ]	-
Pellet Grain Size	[ ]	-
Open Porosity Fraction	[ ]	[ ]
Sorbed Gas	[ ]	[ ]
	[ ]	[ ]
Pellet Void Volume, cm^3	[ ]	-

Table 3-9 Gadolinium Pellet Parameters

Parameter Description	Parameter Information	Tolerance
Pellet OD	0.3225	[ ]
Roughness, µin	[ ]	-
Density, %Theoretical Density	[ ]	]
Resinter Densification Limits (24-hour test)	[ ]	-
Length, in.	0.531	[ ]
Dish Volume (for one pellet), mm^3	[ ]	[ ]
Dish Diameter (minimum), in.	[ ]	-
Dish Depth (minimum), in.	[ ]	-
Pellet Grain Size, µm	[ ]	-
Open Porosity Fraction	[ ]	[ ]
Sorbed Gas	[ ]	[ ]
	[ ]	[ ]
Pellet Void Volume, cm^3	[ ]	-
U235 Enrichment Reduction	Enrichment Reduction	Wt % Gad
	[ ]	[ ]
	[ ]	[ ]
	[ ]	[ ]
	[ ]	[ ]

Table 3-10 Fuel Rod Spring Dimensions

Table 3-11 Fuel Rod Support Tube Dimensions

Table 3-12 Fuel Rod End Plug Dimensions

Figure 3-1 U.S. EPR Fuel Assembly

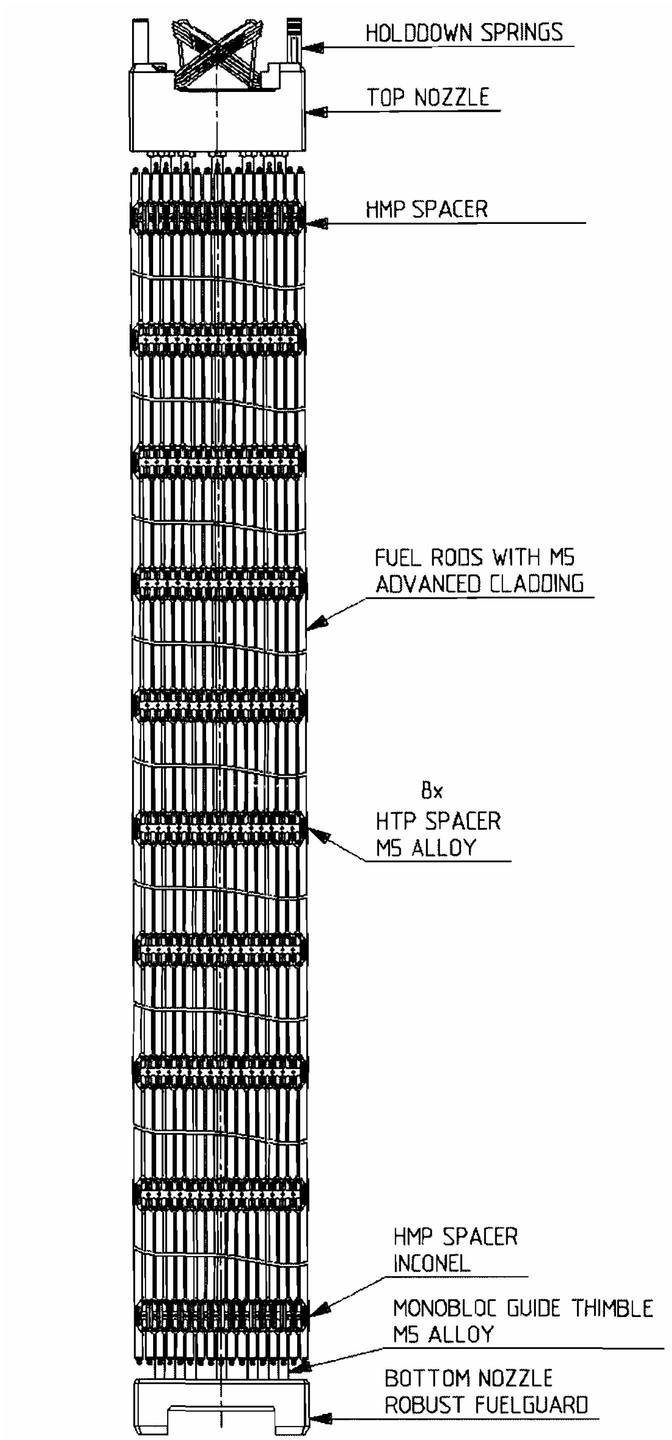


Figure 3-2 Instrument Lance Position

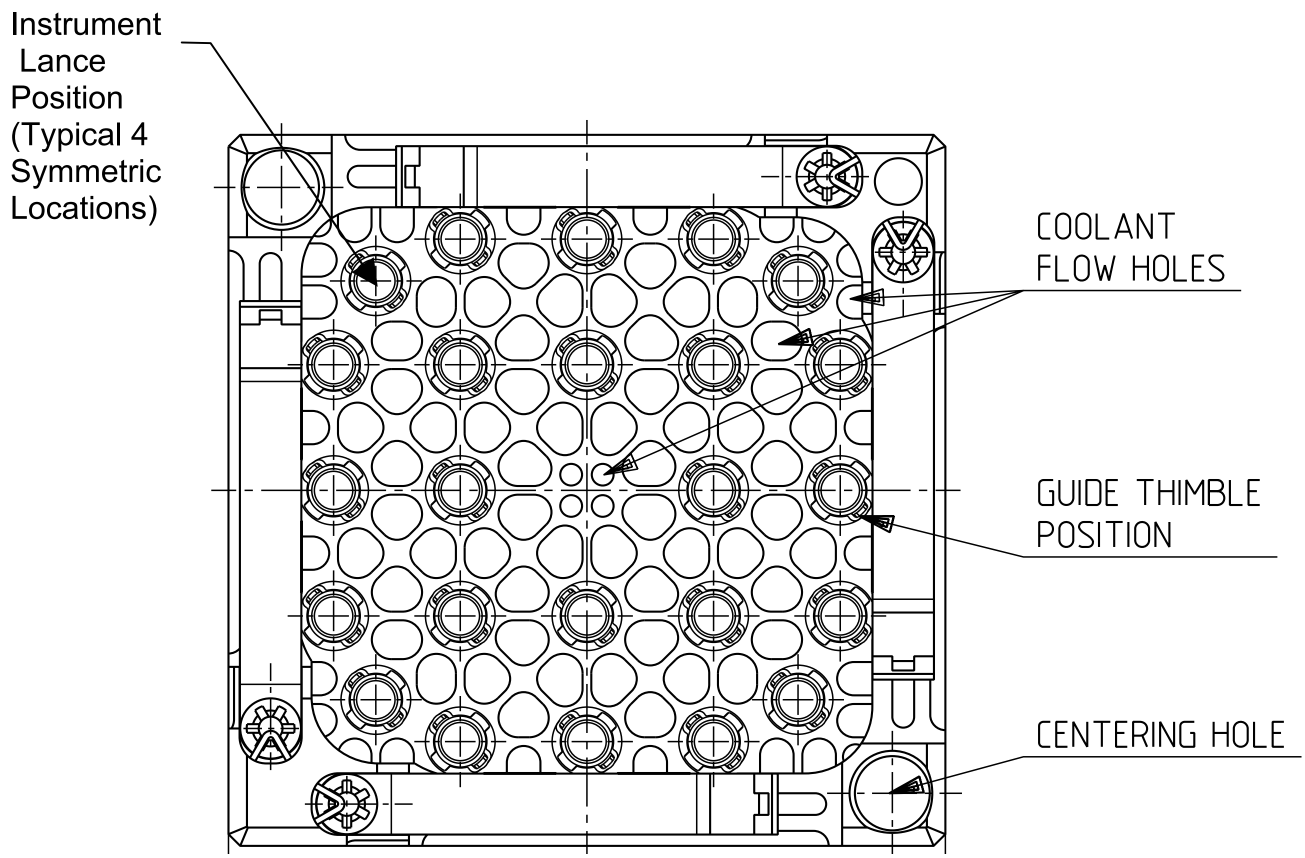


Figure 3-3 Fuel Rod Assembly

[ ]

Figure 3-4 FUELGUARD™ Bottom Nozzle Arrangement (Dimensions in millimeters)

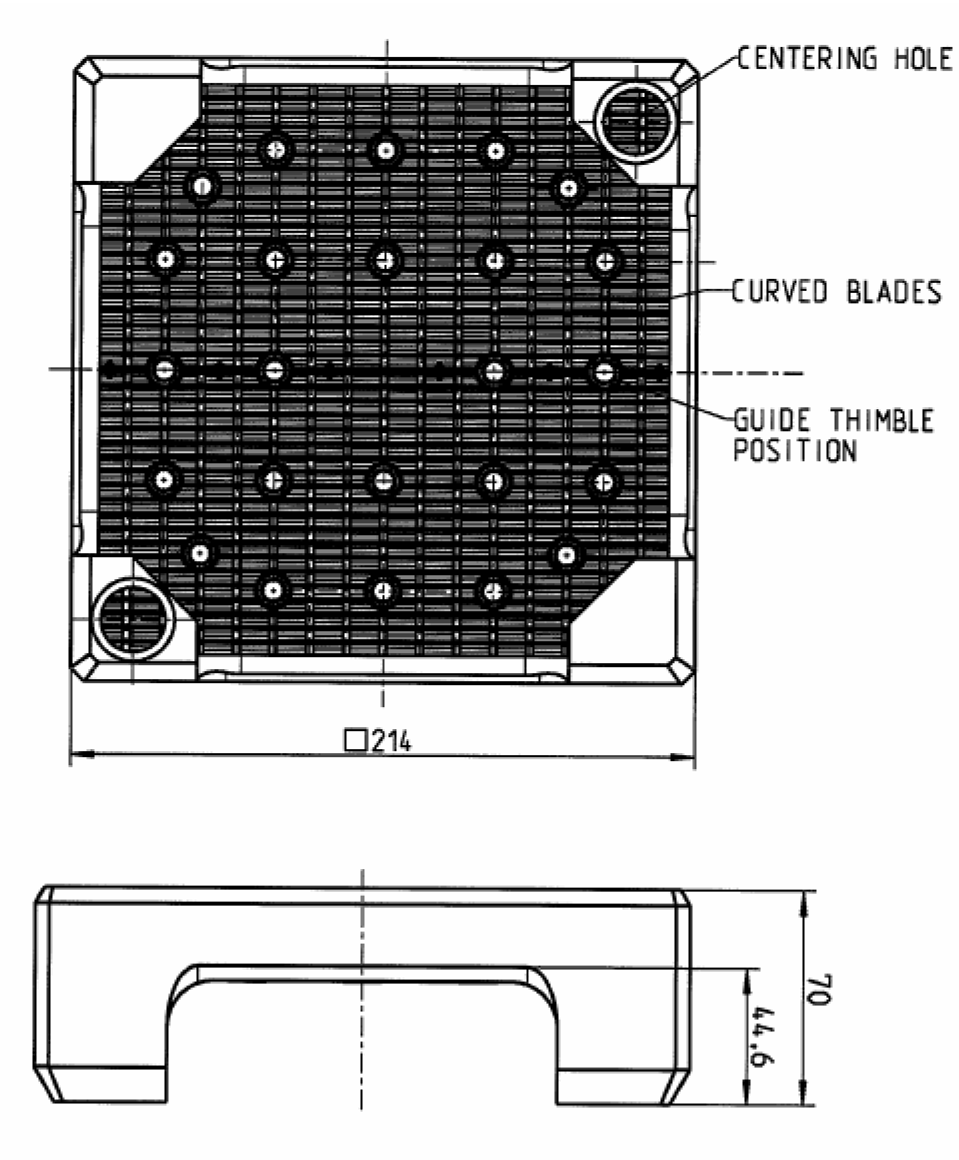
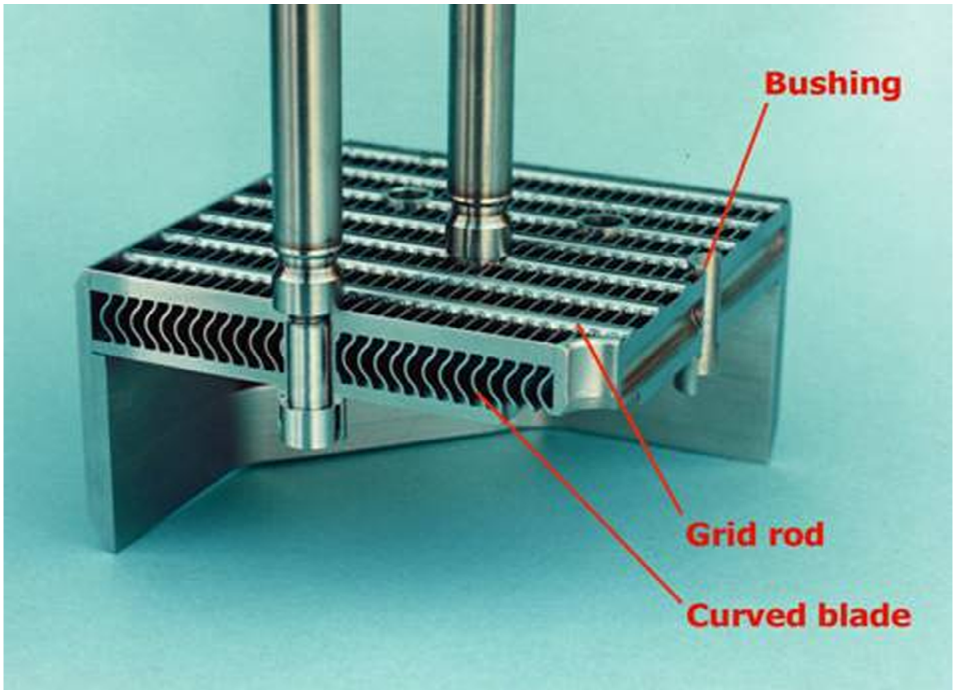


Figure 3-5 FUELGUARD^{TM} Bottom Nozzle



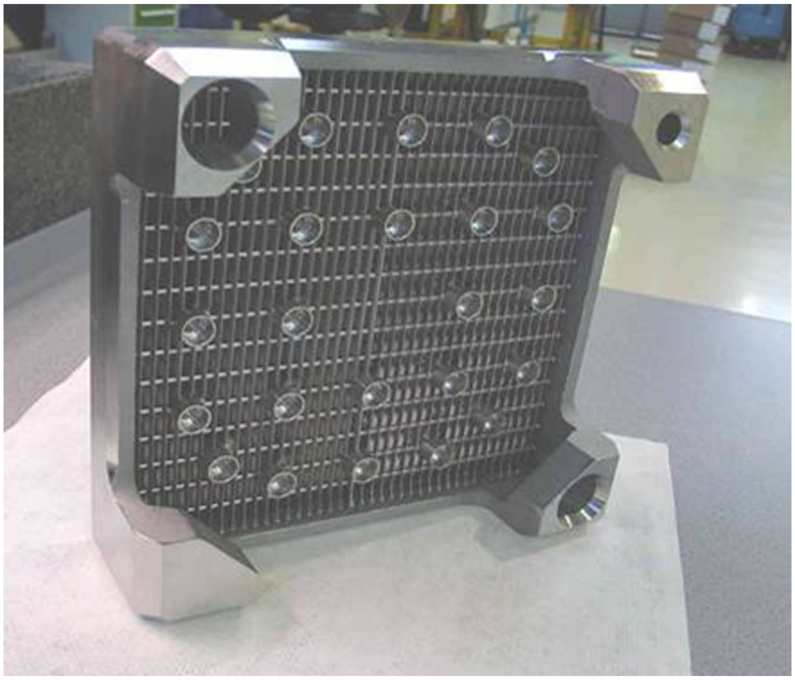
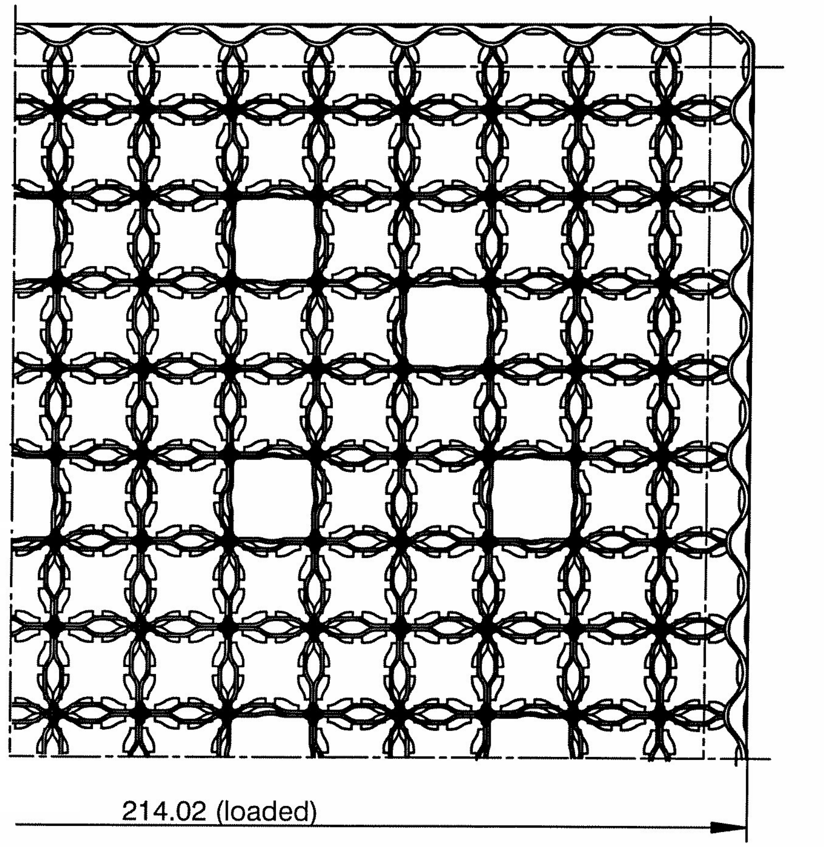


Figure 3-6 Intermediate-HTP Spacer Grid Assembly (Dimensions in millimeters)



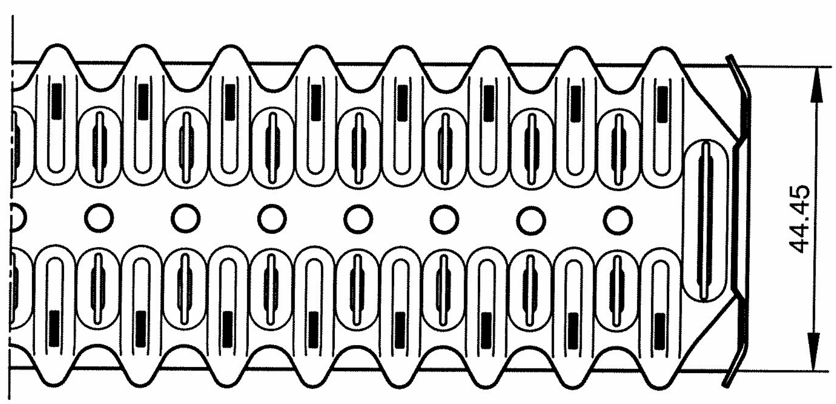


Figure 3-7 HTP Spacer Grid Characteristics

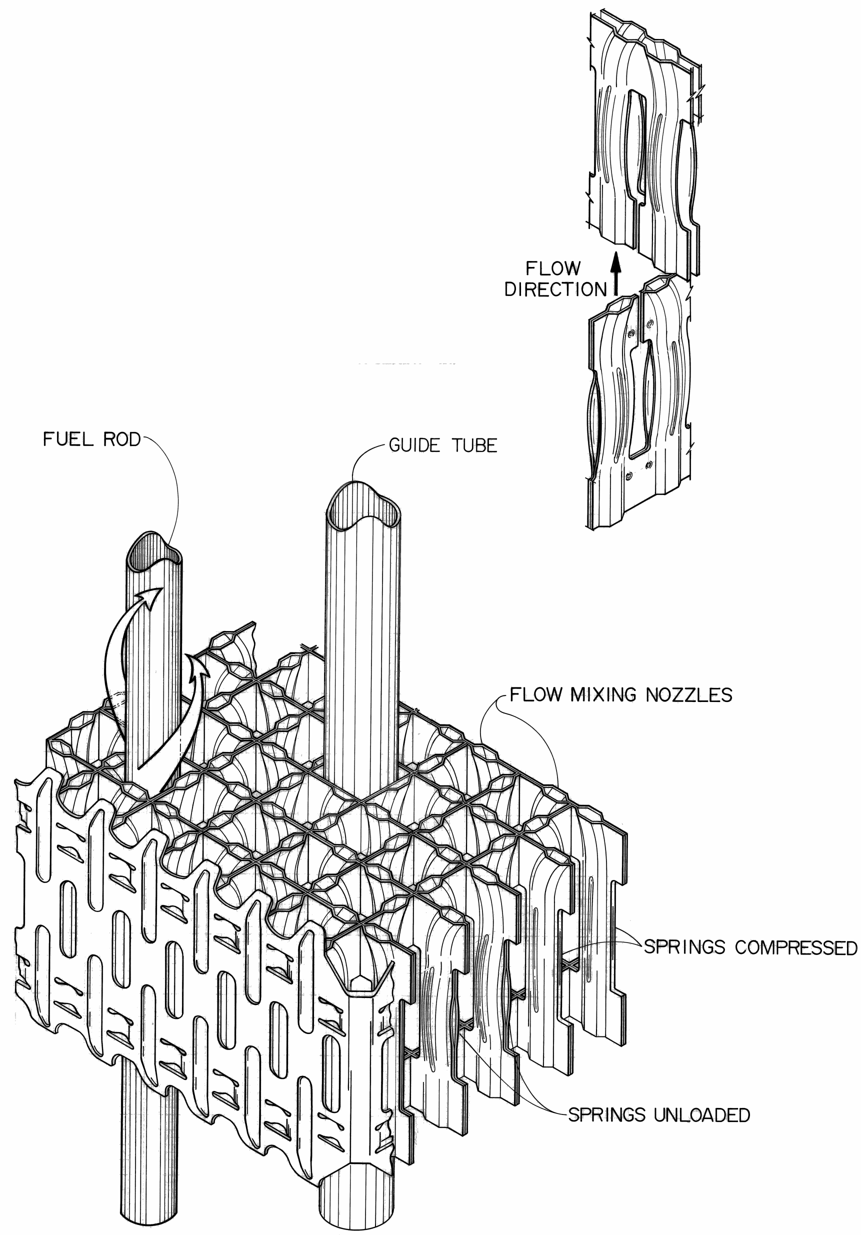


Figure 3-8 HMP End Grid Assembly
(Dimensions in millimeters)

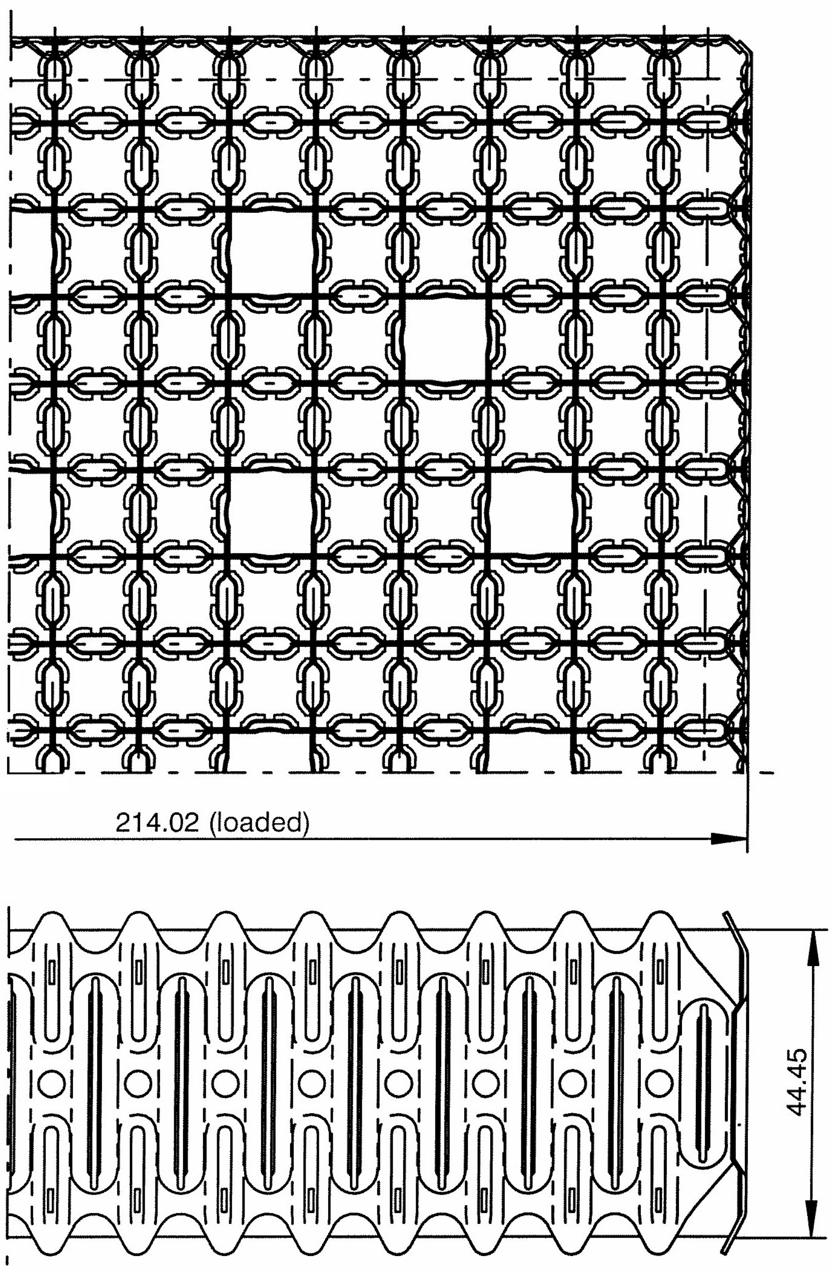


Figure 3-9 Guide Tube Assembly
(Dimensions in millimeters)

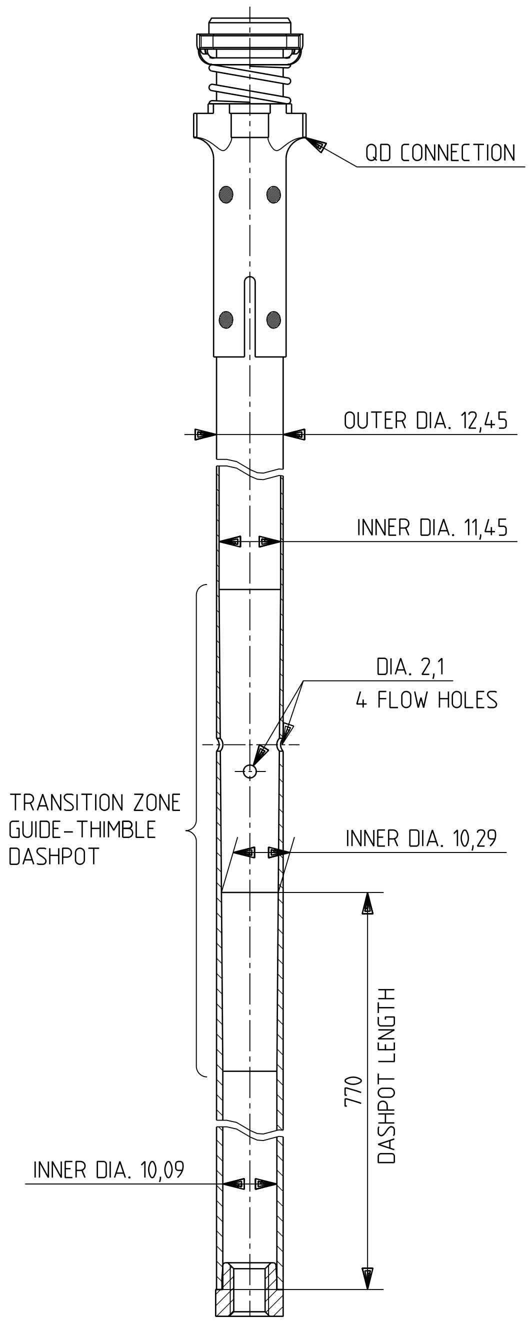


Figure 3-10 QD Top Nozzle Assembly

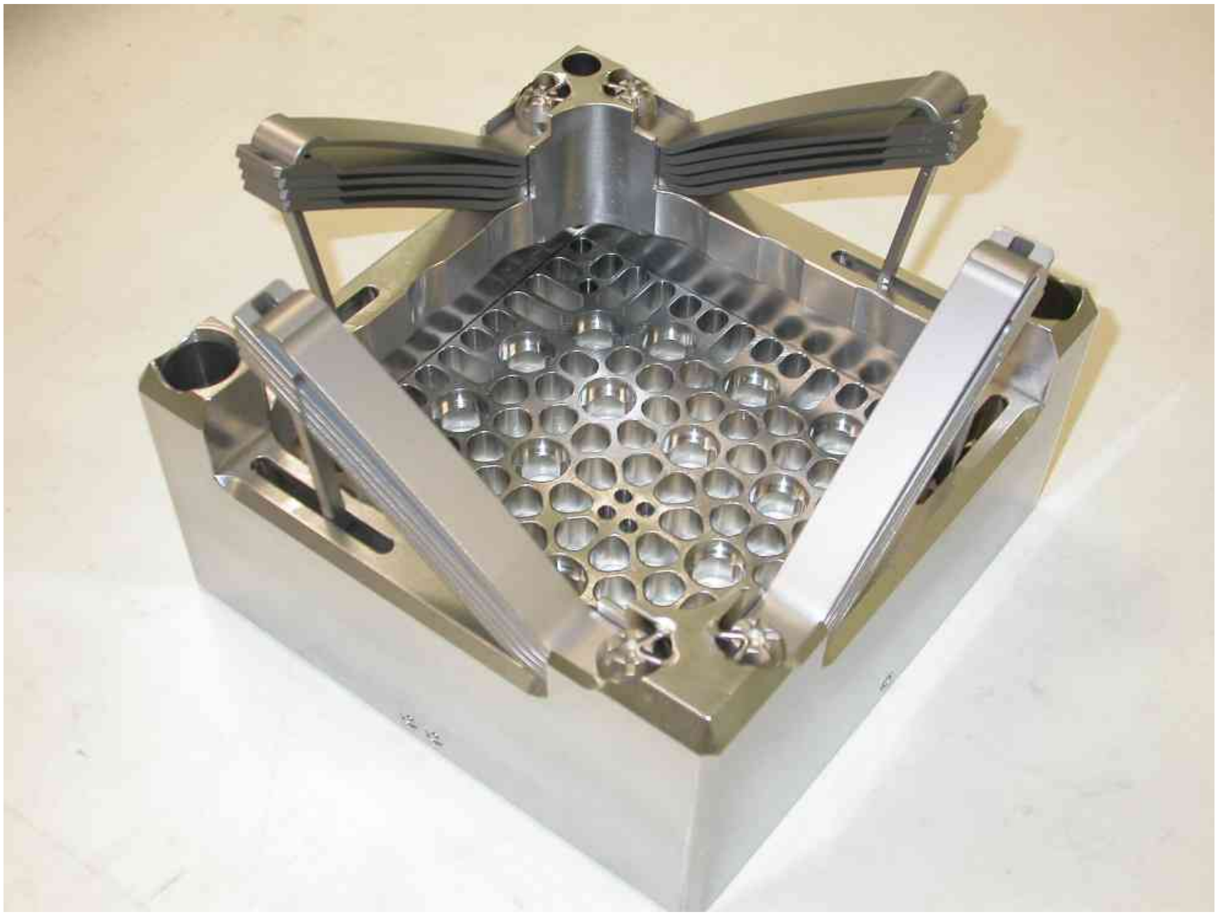


Figure 3-11 QD Top Nozzle Assembly
(Dimensions in millimeters)

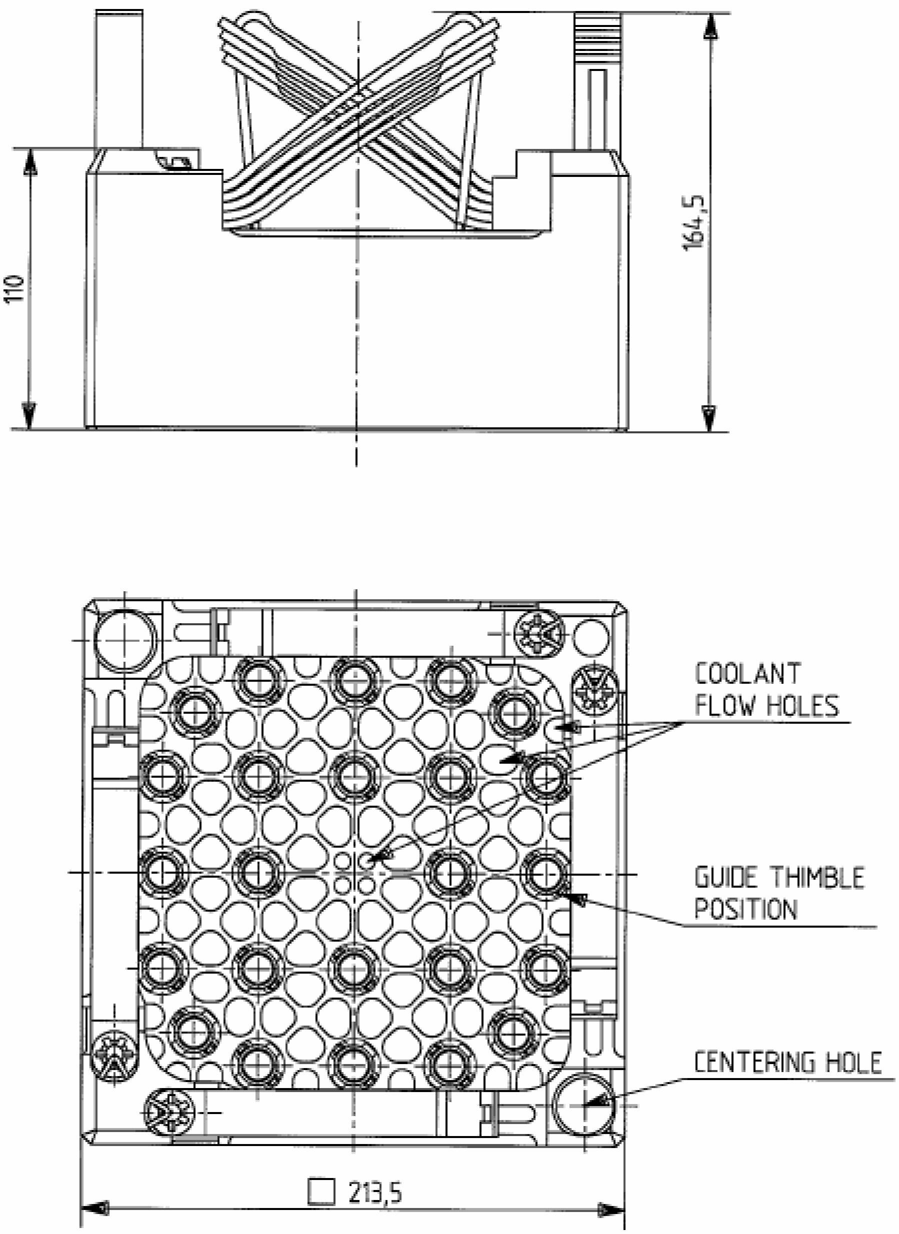


Figure 3-12 QD Connection at Top Nozzle

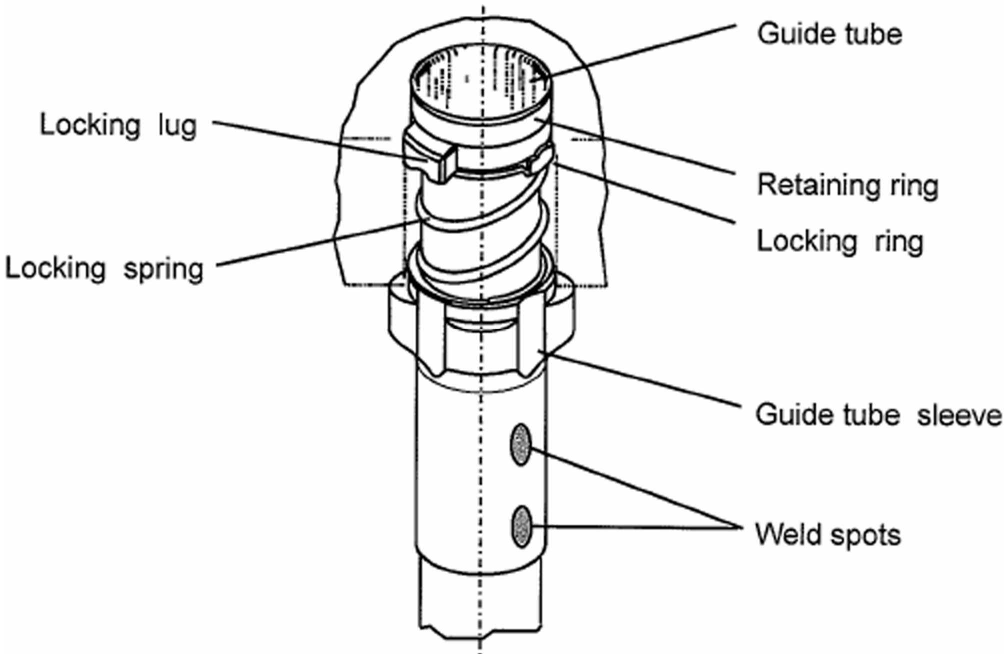
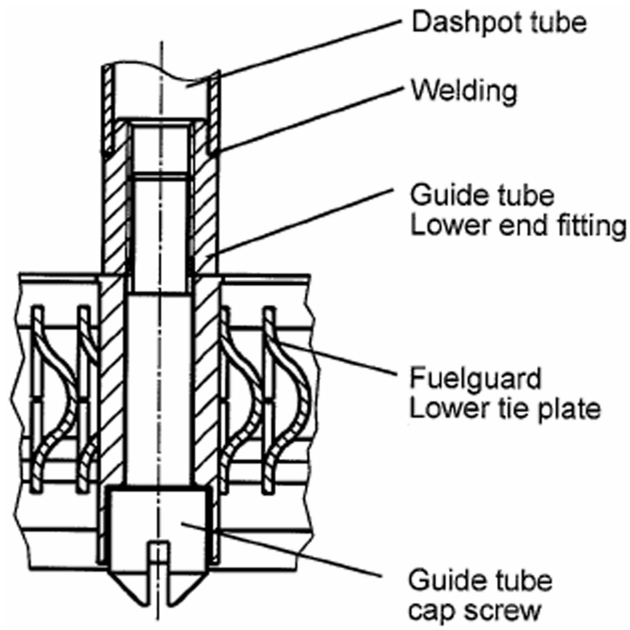


Figure 3-13 Guide Tube Connection at Bottom Nozzle



3.8 Fuel Assembly Testing

A comprehensive test program was conducted to characterize the performance of the U.S. EPR fuel assembly design. Testing was conducted on full-sized prototype fuel assemblies and on various assembly components. The full-size 14 ft prototype fuel assemblies were used for structural and mechanical as well as thermal-hydraulic testing. The prototype fuel assembly mechanical tests included:

- Static axial tension
- Compression tests to determine fuel assembly axial stiffness
- Static lateral bending to determine lateral stiffness
- Fuel assembly shaker tests to determine natural frequencies and mode shapes
- Lateral pluck tests with and without spacer grid impact to determine fuel assembly damping
- Vertical drop tests from various heights

The prototype fuel assembly thermal-hydraulic test scope included assembly pressure drop, life and wear testing consisting of a 1000 hour endurance test in the HERMES-P loop, and flow induced vibration testing. In addition, RCCA SCRAM tests (i.e., trip times) and stroking wear tests were performed in the KOPRA test loop.

These tests are described in detail in Sections 3.8.1 through 3.8.7. These tests were conducted in accordance with approved test plans, at QA approved (Approved Supplier List (ASL) compliant to AREVA NP Inc.) AREVA test facilities. These test results were used in benchmarking analytical models for U.S. EPR fuel assembly design evaluation in Section 5.0.

3.8.1 Shaker Tests

The prototype fuel assembly was supported by mock core plates with guide pins to simulate the end conditions in the reactor. The frequency and damping values and the mode shapes for the first five modes of vibration of the fuel assembly were measured. An electrodynamic shaker was used to excite the fuel assembly near the midplane at various frequencies and amplitudes as the responses of the fuel assembly nozzles and spacer grids were measured and recorded. These displacement time histories were analyzed to determine the dynamic characteristics of the fuel assembly.

3.8.2 Lateral Pluck Tests

A fuel assembly pluck test was performed on the prototype fuel assembly in air and at room temperature to obtain fundamental natural frequency and damping values at various amplitudes. The pluck test was conducted by measuring and recording the displacement of selected spacer grids as the fuel assembly was deflected laterally at the midplane and quickly released. The results were consistent with the shaker test results for the first mode. The test results were used to benchmark the lateral fuel assembly model to determine the faulted lateral loads from SSE and SSE + LOCA.

3.8.3 Lateral Impact Tests

Fuel assembly lateral impact tests were performed on the prototype fuel assembly in air and at room temperature to obtain impact force measurements. The lateral impact test was conducted by measuring and recording the impact velocity and force as the fuel assembly impacted stationary blocks placed at grids 5 and 6 after the fuel assembly was deflected laterally at the midplane and quickly released. The test results were used to benchmark the lateral fuel assembly model to determine the faulted lateral loads from SSE and SSE + LOCA. The model benchmark predicted the measured velocities and impacts within 3%.

3.8.4 Static Stiffness Tests

Force-versus-deflection tests were conducted to determine the axial and lateral stiffness of the fuel assembly. In the axial stiffness test the fuel assembly was compressed along its longitudinal axis by an application of forces at the nozzles. The lateral stiffness test consisted of loading the fuel assembly laterally at the two center spacer grids. The results of these stiffness tests were used to benchmark the analytical models usage for the faulted component analyses addressed in Section 5.3.4.3.

3.8.5 Fuel Assembly Drop Tests

Fuel assembly drop tests were performed on a prototype fuel assembly to obtain impact loads against which the vertical analytical model was benchmarked. The fuel assembly was dropped from various heights against an unyielding surface and the impact loads were measured and recorded. The effect of multiple drops in succession was accounted for in the model benchmark and a suitable correlation to the test results were obtained.

3.8.6 Fuel Hydraulic Flow Tests

Full scale flow testing on a full scale prototype fuel assembly was performed using the AREVA HERMES-P flow loop test facility. These flow loop tests were used to establish flow loss coefficients and other related flow characterization parameters for inputs to the AREVA LYNXT thermal hydraulic flow analysis computer code. The LYNXT code was approved by the NRC for use in evaluating fuel assembly designs in the LYNXT: Core Transient Thermal-Hydraulic Program (Reference 18). LYNXT was used to determine the flow lift forces on the fuel assembly as a part of the evaluation of fuel assembly liftoff resistance described in Section 5.1.9.

The prototype fuel assembly tests in the HERMES-P loop also were used to evaluate the fretting and wear performance at the grid to rod interfaces. Fretting resistance was also demonstrated using combined out of core testing performed in the PETER flow loop and Autoclave test facilities. The PETER loop tests provided short term measurement of flow-induced vibration (FIV) parameters on a full scale fuel assembly at temperature and pressure. From the rod and grid vibration parameters, such as amplitude and frequency obtained in the PETER flow loop, the worst case rod and axial region of fuel assembly flow were determined. These bounding vibration conditions were then replicated in the Autoclave test apparatus involving an individual rod and limited axial span but with extended time spans of up to 1000 hours to enable testing on a bounding and worst case rod to grid interface. Testing and evaluation results for verification of U.S. EPR fretting and wear resistance are addressed in Section 5.1.4.

3.8.7 CRDM Driveline/RCCA Drop Testing

The RCCA and fuel assembly have been performance tested in a full-scale test loop, which includes the CRDM, drive rod, RCCA and prototype fuel assembly cage mockup. The test program measures various parameters relating to RCCA drop kinetics, i.e., various pressure drop conditions and driveline drop time. The KOPRA testing program covers multiple phases and various components, including, but not limited to the RCCA and fuel assembly cage structure and guide thimble design.

Of particular interest to the fuel assembly guide thimble and RCCA design is the measured drop time data used to benchmark the CIGAL code used to calculate/predict drop kinetics of the driveline under various conditions, including maximum and minimum rod drop conditions. Maximum rod drop time was determined to be less than the tech spec limit of 3.5 seconds.

Validation of CIGAL was performed. The final results of the KOPRA testing confirm that drop times are not adversely affected following the endurance phase of the test program, which includes approximately **[ ]** steps and **[ ]** scrams.

3.9 Fuel Assembly Component Testing

In addition to full-scale prototype testing, various components were also characterized by testing. Static compression tests were performed on the holddown springs and clamp screw. The spacer grid design was subjected to static buckling and dynamic crush tests. The properties of the component connections and bottom nozzle were also tested. The results of the component prototype characterization testing described in Section 3.9 were incorporated into various analytical models used to verify the U.S. EPR design.

3.9.1 Holddown Spring Characteristic Tests

Static force deflection tests were performed on prototype sets of the U.S. EPR fuel assembly five-leaf holddown springs at room temperature to obtain the force and deflection characteristics of the spring. The force/deflection characteristics were used in the normal operating analysis to determine fuel assembly holddown forces. The holddown spring rate was determined from the test data and used in the analytical model of the fuel assembly. A typical holddown spring force versus deflection curve is shown in Figure 3-14.

Figure 3-14 U.S. EPR Holddown Spring Single Set Force/Deflection Curve

3.9.2 Strength Test of Upper Guide Tube Connection

Hot compression tests of the upper guide tube QD connection were conducted to determine the static strength of this connection. The tests determined that the guide tubes fail plastically at loads larger than **[ ]** without failure of the spot welds used to attach the QD fittings to the end of each guide tube. Therefore the test shows that the mechanical strength of the QD connector is limited by the performance of the guide tube and not by the welded connection between the sleeve and guide tube.

3.9.3 Spacer Grid Testing

The mechanical design bases of the U.S. EPR spacer grids were confirmed through a series of tests on prototype 17x17 M5™ HTP grids. These tests are described in Section 5.1.1.5.

3.9.4 Strength Test of Bottom Nozzle

Strength testing of the bottom nozzle was performed to establish the axial load limit criteria for evaluation. A prototype bottom nozzle was tested at room temperature in static axial compression by 24 springs on the guide tube positions. The spring stiffness was determined to be equal to the guide tubes stiffness in order to simulate the real load distribution of the guide tubes. The test piece and the bottom nozzle for the U.S. EPR design are identical except that the test piece accommodates a central instrument tube position that is not present in the U.S. EPR design. The influence of this difference on the yield behavior can be neglected. A maximum room temperature test load of **[ ]** was demonstrated without collapse of the structure. This tested maximum load was used to demonstrate the structural adequacy in the design evaluation by comparison with the normal operating and faulted loads as discussed in section 5.1.1.3.

4.0 RELEVANT OPERATING EXPERIENCE

Operational experience is used to demonstrate the reliability and the performance of a fuel assembly design. The U.S. EPR design operational performance is supported by those fuel designs upon which it is based. The AREVA HTP fuel product establishes the operating experience basis for the overall U.S. EPR fuel assembly construction including HTP and HMP spacer grids, welded skeleton, and FUELGUARD™ bottom nozzle. The AREVA M5™ fuel products provide the global and domestic operating experience basis for the M5™ fuel rod cladding, guide tube, and spacer grid implementation, including the 17x17 Advanced Mark-BW fuel design from which the low pressure drop top nozzle is demonstrated.

A brief overview of the experience gained from other HTP fuel assembly designs is described in Section 4.1. The operational experience with more advanced HTP design using higher performance requirements, introduced as a result of increased target burnups, is addressed in Section 4.2. M5™ fuel rod cladding, guide tube, and spacer grid operational experience is described in Section 4.3. Operational experience for 14 ft fuel designs is described in Section 4.4. Reliability and operational behavior are described in Section 4.5.

4.1 Operational Experience with HTP Fuel Assemblies

HTP is the designation of a specific AREVA-developed spacer grid and concurrently denotes a fuel assembly design in which this type of spacer is the major component, such as the U.S. EPR fuel design. Fuel assemblies equipped with traditional spacers employ springs and dimples to support each fuel rod in its spacer cell and use mixing vanes along the top edges of the spacer strips to enhance thermal-hydraulic performance. The HTP spacer represents a different but proven concept in spacer design for PWR fuel. The HTP spacer features strip doublets are shaped to serve as spring elements to firmly hold the fuel rods in radial alignment, and to produce curved internal flow channels to achieve the desired thermal-hydraulic performance. HTP fuel assemblies were first loaded in a U.S. plant in 1988. There is now over 18 years of operational experience with HTP fuel assemblies, including domestic and global applications. This experience indicates the HTP spacer design application to the U.S. EPR will provide FIV resistance and high impact strength for maintaining coolable geometry during severe postulated accidents, which is demonstrated further in sections 5.1.4 and 5.3.4.1 respectively.

As of December 2006, the operational experience with HTP fuel assemblies comprises a total of 7894 fuel assemblies irradiated in 41 reactors. Of these, the experience includes 4835 assemblies in 26 European plants located in Belgium, France, Germany, Spain, Sweden, Switzerland, UK, and The Netherlands; 2995 assemblies in 12 U.S. plants; 60 assemblies in 2 Japanese plants; and 4 assemblies in a Brazilian plant. The entire range of fuel rod arrays from 14x14 to 18x18 is represented, as well as reactors supplied by various vendors, comprising Combustion Engineering (CE), Framatome, Westinghouse, Siemens and Babcock & Wilcox. More than half of the HTP assemblies (i.e., 3816) used the 17x17 array for operation in the 12 ft Framatome and Westinghouse designed plants. HTP assemblies deployed in CE and Westinghouse designed plants comprise 7.5 to 9.5 percent of the total number of fuel assemblies for the 14x14 or 15x15 arrays, respectively. A total of 1120 HTP fuel assemblies have been inserted into 14 Siemens designed plants, comprising 15x15, 16x16, and 18x18 arrays.

The operational experience for HTP fuel covers a variety of core formations. Domestically, Shearon Harris, Robinson, Palisades, St. Lucie, Millstone, and Ft. Calhoun plants have operated for several years using full core reloads of HTP fuel assemblies. Currently, several plants operate using transition cores including HTP fuel and a mixture of different assembly designs (see Table 4-1).

Operating experience for HTP fuel assemblies covers a range of cycle design strategies from 6 to 24 months. For compensation of the initial excess reactivity as well as for flattening the power density distribution in case of low leakage loading patterns, the HTP fuel experience includes the use of fuel rods that employ gadolinium. As of December 2006, more than 3500 HTP fuel assemblies equipped with Gd rods have been loaded into 23 reactors worldwide. The number of Gd fuel rods within an assembly varied between 4 and 28 with Gd_2O_3 concentrations from 2 to 8 weight percent. HTP 15x15 and 17x17 fuel assemblies with configurations ranging from 4 Gd-rods of 2 weight percent to 28 Gd-rods of 8 weight percent have been placed in operation for Westinghouse type plants. A maximum fuel assembly burnup of 67 GWd/mtU has been achieved in a Westinghouse plant. Gadolinia fuel rods have also been used extensively in other AREVA fuel designs.

Table 4-1 Operational Experience with HTP Fuel Assembly (Status as of December 2006)

Most of the HTP fuel assemblies irradiated to date are equipped with bimetallic spacers (i.e., Zircaloy-4 strips with Inconel springs) at the top and bottom spacer locations, Zircaloy-4 HTP spacers at the intermediate positions, Zircaloy-4 cladding and structural material, and FUELGUARD™ debris filter bottom nozzles. Since 1991, the HTP 15x15 fuel assembly for a Westinghouse plant has been equipped with a Zircaloy-4 HTP spacer at the uppermost position.

With 4101 fuel assemblies, more than half of all inserted HTP fuel assemblies have achieved a burnup higher than 40 GWd/mtU. The maximum assembly burnup is 70 GWd/mtU. The burnup distribution of the HTP fuel assemblies as of December 2006 is shown in Figure 4-1.

Figure 4-1 Burnup Distribution of the HTP Fuel Assembly (Status as of December 2006)

4.2 Advanced HTP Fuel Assembly Designs

Advanced HTP fuel assembly designs have recently been loaded with enhanced robustness and elevated performance requirements as a result of increased target burnups. A more detailed technical description and the operational experience gained from these designs are described in this section.

Two features that increase the robustness of the design include an Inconel HTP spacer at the lowermost position and M5™ alloy as the strip material of the spacers on the other positions.

The use of an Inconel spacer at the lowermost position provides the benefits of enhanced strength and FIV resistance. The higher strength enables lower strip thicknesses for lower pressure drop. Inconel end grid construction also provides lower levels of spring relaxation, which provides enhanced fuel rod support under extended burnup conditions especially at the lower most span where greater levels of flow turbulence and cross-flow occur. The use of Inconel spacers at the lower most positions in conjunction with HTP spacers at the intermediate grid positions and M5™ has resulted in the near elimination of fretting failures in PWR environments.

Lateral growth of Zircaloy-4 spacer grids occurs at high burnups which is attributed to irradiation and corrosion. M5™ alloy has been introduced to stabilize the spacer grid given the material low corrosion, hydrogen uptake, and free growth properties and to reduce the growth at high burnups.

4.2.1 HTP Fuel Assemblies Equipped with an HMP Spacer at Lowermost Position

Fuel assemblies with HTP spacers made of Inconel at the lowermost position were first inserted into two German plants in 1992. These Inconel spacers had the same curved flow channels as the Zircaloy HTP spacers in the active region of the fuel. The initial insertion of the current version of Inconel spacer with straight flow channels, designated HMP, was implemented in 1988. Today, a large operational experience base with HTP fuel featuring the HMP spacer is available. Altogether, 2597 of these HTP fuel assemblies have been loaded into 26 plants worldwide. Figure 4-2 shows the status as of December 2006 for burnup distribution of the HTP fuel assemblies featuring the HMP at the lowermost position. A maximum assembly burnup of 70 GWd/mtU has been achieved.

Figure 4-2 Burnup Distribution of HTP Fuel Assembly featuring an HMP at Lowermost Position (Status as of December 2006)

[ ]

4.2.2 HTP Spacers Made of M5™ Alloy

The selection of M5™ strip material for construction of U.S. EPR HTP intermediate spacers is an extension of the favorable operating experience of M5™ alloy application for fuel rod cladding and guide tubes. The benefits of reduced corrosion (i.e., lower oxide layer formation rate) and hydrogen pickup results in greater strength at higher burnups from a lower loss of structural strip thickness and additional margins against loss of ductility.

The application of M5™ for intermediate spacer grids in the U.S. EPR is consistent with the intention of AREVA to maximize the use of M5™ within the entire product line of PWR fuels in the future. In 2004, the first HTP lead assemblies equipped with M5™ spacers and M5™ MONOBLOC™ guide tubes were loaded into two German plants. In

2005, two reloads with M5™ HTP 15x15 spacers were loaded into two U.S. plants. Two additional reloads were also loaded in 2006 and 2007. One reload of M5™ HTP 14x14 fuel assemblies was loaded in a U.S. plant in 2006. Four M5™ HTP 17x17 lead assemblies were installed in a Swedish plant in 2006.

Since 1991, HTP fuel assemblies with Zircaloy-4 spacers at the top end grid position have been inserted into U.S. plants. As of December 2006, a total of 3515 HTP fuel assemblies equipped with an upper HTP spacer made of zirconium alloy have been irradiated in 29 plants worldwide. The burnup distribution is shown in Figure 4-3.

Figure 4-3 Burnup Distribution of HTP Fuel Assemblies Featuring a Zirconium Alloy HTP Spacer at the Uppermost Position (Status as of December 2006)

[ ]

4.2.3 HTP Fuel Assemblies Equipped with M5™ Clad Rods

HTP fuel assemblies equipped with M5™ clad rods were first inserted into four plants in 2003, including four lead test assemblies each in a South American and a U.S. plant, a reload consisting of (36) 16x16 type assemblies in a German plant, and a reload of (85)

15x15 type assemblies in a U.S. plant. By December 2006, 1171 HTP fuel assemblies with M5™ clad have been irradiated in 20 plants in Germany, the Netherlands, Sweden, Switzerland, South America, and the U.S. The operational experience of the combination HTP fuel assembly and M5™ clad covers arrays from 14x14 up to 18x18. To date a maximum assembly average burnup of 51 GWd/mtU has been achieved. Figure 4-4 shows the burnup distribution of HTP fuel assemblies equipped with M5™ cladding material as of December 2006.

Figure 4-4 Burnup Distribution of HTP Fuel Assemblies with Fuel Rods Fabricated with M5™ Cladding Material (Status as of December 2006)

4.2.4 Summary of Advanced HTP Fuel Assembly Designs

The advanced HTP fuel designs also feature other advanced components in addition to the advanced spacer designs and cladding material already mentioned such as M5™ guide tube material, MONOBLOC™ guide tube design, and the robust variant of the FUELGUARD™ debris filter.

With respect to the more advanced components, the essential characteristics of the different HTP designs are listed in Tables 4-2 through 4-7 for each plant type and each array.

Table 4-2 Operating Experience in 17x17 HTP Fuel Assemblies for (12 ft) Framatome and W-Plants (Status as of December 2006)

Design Feature		No M5™ or HTP End Grids	No M5™ with HTP/HMP End Grids	All M5 with HMP/HMP End Grids
Spacer grid material	Top	(Bi-metallic) Zircaloy-4 /Inconel 718 spring	(HMP) Zircaloy-4	(HMP) Inconel 718
Spacer grid material	Intermediate flow mixer	---------------	---------------	M5
Spacer grid material	Intermediate (HTP)	Zircaloy-4	Zircaloy -4	M5
Spacer grid material	Bottom	(Bi-metallic) Zircaloy-4 /Inconel 718 spring	(HMP) Inconel 718	(HMP) Inconel 718
Cladding material		Zircaloy-4	Zircaloy-4	M5
Guide tube material		Zircaloy-4	Zircaloy-4	M5 (MONOBLOC™)
Debris filter		FUELGUARD™	FUELGUARD™	Robust FUELGUARD™
First insertion (year)		1993	1998	2006
Maximum assembly burnup (GWd/mtU)		[ ]	[ ]	[ ]
Number of plants		10	3	1
Total number of assemblies		[ ]	[ ]	[ ]

Table 4-3 Operating Experience in 15x15 HTP Fuel Assemblies for (12 ft) W-Plants (Status as of December 2006)

Design Feature		Characteristics	
		Without FUELGUARD™	**With FUELGUARD™**
Spacer grid material	Top	(HTP) Zircaloy-4	(HTP) Zircaloy-4
	IFM	IFM	IFM
	(HTP) Intermediate	Zircaloy-4	Zircaloy -4
	Bottom	(Bi-metallic) Zircaloy -4 /Inconel 718 Springs	(Bi-metallic) Zircaloy -4 /Inconel 718
Cladding material		Zircaloy-4	Zircaloy-4
Guide tube material		Zircaloy-4	Zircaloy-4
Debris filter		--------------------------	FUELGUARD™
First insertion		1991	1993
Maximum assembly burnup (GWd/mtU)		**[ ]**	**[ ]**
Number of plants		1	1
Total number of assemblies		**[ ]**	**[ ]**

Table 4-4 Operating Experience in 14x14 HTP Fuel Assemblies in W-Plants (Status as of December 2006)

<table>
<tr><th colspan="2">Design Feature</th><th>Characteristics</th></tr>
<tr><td rowspan="4">Spacer grid material</td><td>Top</td><td>(Bi-metallic) Zircaloy-4 /Inconel 718 Spring</td></tr>
<tr><td>IFM</td><td>---</td></tr>
<tr><td>Intermediate (HTP)</td><td>Zircaloy-4</td></tr>
<tr><td>Bottom</td><td>(Bi-metallic) Zircaloy-4 /Inconel 718 Spring</td></tr>
<tr><td colspan="2">Cladding material</td><td>Zircaloy-4</td></tr>
<tr><td colspan="2">Guide tube material</td><td>Zircaloy-4</td></tr>
<tr><td colspan="2">Debris filter</td><td>FUELGUARD™</td></tr>
<tr><td colspan="2">First insertion</td><td>1994</td></tr>
<tr><td colspan="2">Maximum assembly burnup (GWd/mtU)</td><td>[ ]</td></tr>
<tr><td colspan="2">Number of plants</td><td>3</td></tr>
<tr><td colspan="2">Total number of assemblies</td><td>[ ]</td></tr>
</table>

Table 4-5 Operating Experience in 15x15 HTP Fuel Assemblies for CE Plants (Status as of December 2006)

Design Feature		Characteristics		
		No M5™ No HTP End Grids	**No M5™ HTP/HMP End Grids**	**No M5™ All HTP**
Spacer grid material	Top	(Bi-metallic) Zircaloy-4 /Inconel 718 Spring	(HTP) Zircaloy-4	(HTP) Zircaloy-4
	IFM	---	---	---
	Intermediate (HTP)	Zircaloy-4	Zircaloy-4	Zircaloy-4
	Bottom	(Bi-metallic) Zircaloy-4 /Inconel 718 Spring	(HMP) Inconel 718	(HTP) Zircaloy-4
Cladding material		Zircaloy-4	Zircaloy-4	Zircaloy-4
Guide tube material		Zircaloy-4	Zircaloy-4	Zircaloy-4
Debris filter		---	FUELGUARD™	FUELGUARD™
First insertion		1988	1998	1998
Maximum assembly burnup (GWd/mtU)		[ ]	[ ]	[ ]
Number of plants		1	1	1
Total number of assemblies		[ ]	[ ]	[ ]

Table 4-6 Operating Experience in 14x14 HTP Fuel Assemblies for CE Plants (Status as of December 2006)

Design Feature		Characteristics				
		No M5™ No HTP End Grids	**No M5™ with HTP/HMP End Grids**	**No M5™ All HTP**	**All HTP with M5™ Cladding material**	**All M5™ with HTP/HMP End Grids**
Spacer grid material	Top	(Bi-metallic) Zircaloy-4 /Inconel 718 Spring	(HTP) Zircaloy-4	(HTP) Zircaloy-4	(HTP) Zircaloy-4	(HTP) M5
	IFM	---	---	---	---	---
	Intermediate (HTP)	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	M5
	Bottom	(Bi-metallic) Zircaloy-4 /Inconel 718 Spring	(HMP) Inconel 718	(HTP) Zircaloy-4	(HTP) Zircaloy-4	(HMP) Inconel 718
Cladding material		Zircaloy-4	Zircaloy-4	Zircaloy-4	M5	M5
Guide tube material		Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	M5
Debris filter		---	FUELGUARD™	FUELGUARD™ See Note 1	FUELGUARD™	FUELGUARD™
First insertion		1988	2001	2002	2003	2006
Maximum assembly burnup (GWd/mtU)		[ ]	[ ]	[ ]	[ ]	[ ]
Number of plants		1	2	2	1	1
Total number of assemblies		[ ]	[ ]	[ ]	[ ]	[ ]

Note 1: (72 ea) assemblies without FUELGUARD™

Table 4-7 Operating Experience in 15x15 HTP Fuel Assemblies for B&W Plants (Status as of December 2006)

<table>
<tr><th colspan="2" rowspan="2">Design Feature</th><th colspan="2">Characteristics</th></tr>
<tr><th>Mixed M5™
HTP/HMP End Grids</th><th>All M5 with
HTP/HMP End Grids</th></tr>
<tr><td rowspan="4">Spacer grid material</td><td>Top</td><td>(HTP)
Zircaloy-4</td><td>(HTP)
M5</td></tr>
<tr><td>IFM</td><td>---</td><td>---</td></tr>
<tr><td>Intermediate
(HTP)</td><td>Zircaloy-4</td><td>M5</td></tr>
<tr><td>Bottom</td><td>(HMP)
Inconel</td><td>(HMP)
Inconel</td></tr>
<tr><td colspan="2">Cladding material</td><td>M5</td><td>M5</td></tr>
<tr><td colspan="2">Guide tube material</td><td>M5</td><td>M5</td></tr>
<tr><td colspan="2">Debris filter</td><td>FUELGUARD™</td><td>FUELGUARD™</td></tr>
<tr><td colspan="2">First insertion</td><td>2003</td><td>2005</td></tr>
<tr><td colspan="2">Maximum assembly
burnup (GWd/mtU)</td><td>[ ]</td><td>[ ]</td></tr>
<tr><td colspan="2">Number of plants</td><td>1</td><td>3</td></tr>
<tr><td colspan="2">Total number of
assemblies</td><td>[ ]</td><td>[ ]</td></tr>
</table>

4.3 Operational Experience with M5™ Alloy

M5™ is an advanced zirconium alloy developed and implemented by AREVA to improve corrosion resistance, to reduce hydrogen uptake, and to reduce irradiation growth. In 2000, the NRC approved M5™ for domestic use (see initial version of Reference 7). The applicability of Reference 7 for the U.S. EPR was approved in Reference 2. To date, 33 reloads in 15 different U.S. reactors have used the M5™alloy in more than 2300 fuel assemblies. Globally, over 1.5 million M5™ fuel rods have operated in approximately 6500 fuel assemblies within 57 reactors and more than 3000 fuel assemblies with M5™ fuel rods and guide tubes operated in 37 reactors. Table 4-8 shows the global scale of M5™ fuel rod usage through January 2007 and the international distribution of these fuel assemblies is shown in Figure 4-5. Table 4-9 shows the global usage of M5™ guide tube material and each M5™ fuel assembly (i.e., cladding, guide tubes, and spacer grids).

Figure 4-5 International Distribution of Fuel Assemblies containing M5™ Fuel Rods (Status as of December 2006)

Table 4-8 Global M5™ Fuel Rod Operational Experience (as of January 2007)

Table 4-9 Global M5™ Structure Operational Experience (as of January 2007)

4.4 Operational Experience with 14 ft Fuel Assembly Designs

AREVA also has extensive experience in the design and supply of 14 ft 17x17 fuel assemblies for two reactor designs in Europe (i.e., N4 reactors (4) and 1300 MW reactors (20)). The first batch of 14 ft fuel was installed in 1983, and to date (i.e., mid-2007) AREVA has supplied over 25,000 of these fuel assemblies. Approximately 9600 fuel assemblies have been fabricated with the MONOBLOC™ guide tube. A total of 1454 fuel assemblies have been fabricated with M5™ clad fuel rods and 974 of those also use M5™ MONOBLOC™ guide tubes and M5™ spacer grids. The operational experience totals over 384,000 M5™ clad fuel rods in 14 ft fuel assemblies.

4.5 Operational Behavior and Reliability

Over a period of 18 years close to 8,000 HTP fuel assemblies have been in service, demonstrating excellent operational behavior. During this time only 20 rods in 16 HTP fuel assemblies in 6 reactors have failed, for a variety of reasons summarized in Table 4-10. This corresponds to a fuel rod failure rate of 5×10^{-6}. The failure rate is defined as the ratio of failed rods to the total number of fuel rods operated in the HTP fuel assemblies.

Building on this operational experience, the U.S. EPR fuel assembly design incorporates features to mitigate the cause of these failures. The use of an Inconel bottom HMP spacer grid eliminates the cause of 12 failures. The U.S. EPR fuel assembly design incorporates the FUELGUARD™ debris filtering bottom nozzle with excellent debris filtering performance. The FUELGUARD™ debris filtering bottom nozzle, coupled with continuously improving foreign material exclusion practices at fuel fabrication sites and at operating plants, has substantially reduced the risk of debris failures.

Four of the fretting failures observed at a CE 15x15 plant have been attributed to the plant design, specifically the control blades and associated wide fuel assembly gaps. This design feature is not used in the U.S. EPR plant.

The goal of AREVA NP is zero fuel failures. Failures, if they occur, are investigated with the support of the plant operator and AREVA NP global organization. The cause of fuel failure is eliminated through continuous fuel design enhancement and plant operation improvement.

Table 4-10 HTP Fuel Assembly Rod Failures and Causes
(Status as of December 2006)

[ ]

4.6 Post Irradiation Examination of U.S. EPR 14 foot Fuel Assembly

AREVA NP maintains a strong multifaceted post-irradiation examination (PIE) surveillance program for all of its licensed fuel assembly designs that are a key part of all of its fuel assembly development and deployments. AREVA NP's PIE database provides an extensive and comprehensive range of fuel assembly experience that covers many operating plants, several fuel assembly designs and an extensive range of industrial partnership spanning both the globe and U.S. nuclear industry. The AREVA NP PIE program is designed to maintain compliance to regulatory requirements, maintain awareness of emerging issues in the LWR industry and regulatory environment, and to support design optimization, product development and address operational issues. The program is committed to a culture of safety, quality, continuous improvement and Zero Tolerance for Failure (ZTF). PIE activities include both poolside and hot cell examination campaigns, either of which may include inspections and examinations for clad oxide/corrosion thickness, hydrogen content, crud deposition, irradiation growth of fuel rods and assemblies, shoulder gap measurements, fuel rod fretting, damage incurred from handling or debris, fuel assembly and fuel rod bow, and other general dimensional attributes. AREVA NP's PIE program, compiled into a four-year rolling schedule of PIE campaigns at various nuclear power plants and laboratories, is part of a comprehensive fuel reliability program. AREVA NP fuel reliability program works closely with INPO (Institute of Nuclear Power Operations) and EPRI (Electric Power Research Institute) in developing strategies for PIE exams toward reducing or eliminating fuel damage and fuel rod failures. PIEs also apply to control components (RCCAs) to monitor the presence and/or extent of component failure mechanisms. This program will be used by AREVA NP to monitor performance on selected fuel assemblies at U.S. EPR reactor sites.

5.0 DESIGN EVALUATION

This section presents the design evaluation of the U.S. EPR fuel assembly to establish that the design meets the applicable criteria to maintain safe plant operation. The mechanical analysis demonstrates that the fuel assembly satisfies the requirements outlined in Reference 1.

Methodologies and models specific to M5™ application are provided in References 2, 7 and 8. Methodologies for the fuel assembly faulted structural evaluations are described in the Mark-C Fuel Assembly LOCA-Seismic Analysis (Reference 13) and Addendums 1 and 2 to Mark-C Fuel Assembly LOCA-Seismic Analyses (Reference 14). The design bases follow those established for the Mark-BW fuel assembly, Advanced Mark-BW fuel assembly and generic HTP grids in References 4, 3 and 20 respectively. These topical reports have received NRC approval for referencing in licensing applications.

The results of the analyses are applicable to fuel assembly operation in 17x17 U.S. EPR plants. The analyses were performed for a peak fuel rod burnup of 62 GWd/mtU.

5.1 Fuel System Damage Criteria

5.1.1 Stress

Design Criterion

> Stress intensities for U.S. EPR fuel assembly components shall be less than the stress limits based on ASME Code, Section III criteria.

The structural design requirements for the U.S. EPR fuel assembly are mostly derived from AREVA NP experience, both in the design and incore operation with similar designs. The design bases and design limits for the U.S. EPR fuel assembly are essentially the same as those for previously licensed and approved fuel assembly designs such as those approved in Reference 3, Reference 4, and Reference 14. The requirements are consistent with the acceptance criteria in Reference 1. Stress intensities, and in some cases Von-Mises stresses, were shown to be less than the stress limits based on ASME Code, Reference 21. Code level A criteria are used for normal operating conditions and code level D criteria are used for the LOCA and seismic (i.e., faulted analyses).

The design evaluations performed to verify the adequacy of the U.S. EPR fuel assembly for normal and faulted operating conditions are presented in the following subsections. The fuel assembly components that were evaluated include:

- Guide tubes
- Spacer grids
- Top and bottom nozzles
- HMP end grid restraint sleeves
- QD mechanism
- Holddown spring
- Connections
- Fuel rod cladding

The fuel assembly loads at hot zero power (HZP) and hot full power (HFP) were considered for evaluation of components for normal operating conditions.

In accordance with the SRP criteria in Section 3.7.3 of NUREG-0800 (Reference 27), structurally significant fuel assembly components were evaluated for normal operating plus fatigue stress cycling of five operational base earthquake (OBE) events followed by one SSE event of 10 maximum stress cycles per event. The normal operating fatigue cycle counts for components are listed in Table 5-14, which includes a count of the reactor coolant system (RCS) design transients evaluated. The RCS life events were adjusted to eight effective full power (FP) years per fuel assembly for determining fuel component cycle counts. In the fatigue evaluations for earthquakes, an SSE stress cycle is used as an enveloping stress cycle for all earthquake events. A total of 60 SSE stress cycles were considered. In all cases a total fatigue usage factor of less than 1.0 was demonstrated considering the full life stress cycles for normal operation and up to 60 SSE stress cycles.

Both the fuel assembly and individual components were evaluated for structural adequacy for shipping and handling loads in the amount of 6g lateral and 4g in the axial direction. The evaluations resulted in positive design margins against the stress limits.

5.1.1.1 Material Properties

For stainless steel and nickel-based alloy components, the material properties used in stress and strength evaluations are from the ASME Code, Reference 21. The M5™ grid and guide tube material mechanical properties and the effects of design temperature on those properties are provided in Reference 7.

5.1.1.2 Guide Tubes

Design Criterion

> Buckling of the guide tubes shall not occur during AOOs or any other transient where control rod insertion is required. In addition, the primary and primary + secondary stresses shall be lower than the material allowable stresses as shown in Reference 4.

The U.S. EPR guide tubes were shown not to buckle and the guide tubes remain elastic, thereby permitting control rod insertion during normal operation. A guide tube buckling safety margin of **[ ]** was calculated for axial loading for the HZP condition. The HZP condition was determined as the limiting normal operating case for compressive loading relative to HFP operation. RCCA impact loads due to SCRAM operations were considered. The critical buckling load was determined by an ANSYS finite element model using large deflection criteria to identify the onset of buckling. An initial lateral deflection of [ ] was imposed on the fuel assembly model at mid-height to account for potential reduction in the critical load due to fuel assembly bow. The HZP power load distribution was incrementally scaled to the point that large deflections indicative of the onset of buckling were observed. A minimum normal operating margin against buckling of [ ] was calculated based on the minimum ratio of the axial load within a single span for the HZP condition to the derived critical buckling load for the span. Guide tube corrosion tolerances, and temperature effects were considered for:

- 100% FP mechanical design flowrate
- 120% FP mechanical design flowrate

In Reference 3, margins were provided on the load required to produce a midspan deflection of [ ] using the secant formula for a 100% FP mechanical design flowrate conditions and a 120% FP mechanical design flowrate pump overspeed condition. A midspan lateral deflection criterion of [ ] was used to demonstrate no adverse effect on the control rod insertion or trip performance. This criterion was also met for the U.S. EPR HZP case by examining the delta midspan nodal deflections from the initial to the onset of buckling. All the midspan nodal deflections between grids for HZP were less than the [ ] deflection criteria. Therefore the calculated [ ] margin against inelastic behavior also bounds the supplemental displacement criteria for demonstrating control rod insertion under HZP. The evaluation also showed that primary and primary + secondary stresses are lower than the material allowable stresses. The results of the evaluation of guide tube stresses are shown in Table 5-1.

Table 5-1 Guide Tube Stress Margins for Normal Operation

[ ]

5.1.1.3 Top and Bottom Nozzles

Design Criterion

> The top and bottom nozzle design criterion is based on the ASME B&PV Code, Reference 21, limits and meets the requirements in Reference 1.

The evaluation of the bottom nozzle during normal operation was performed in accordance with ASME Code, Article NG-3228.4 (Reference 21) using a design limit of 44% of the maximum cold test load of **[ ]**. Bottom nozzle testing is described in Section 3.9.4. The limit based on the maximum test load is further discounted for normal operating temperature evaluations. Axial loading only is considered because the normal operating loads on the bottom nozzle are applied axially by the guide tubes. The maximum normal operating load used in the evaluation was conservatively taken as **[ ]**. This is the limiting holddown spring load **[ ]** plus the dry weight of the fuel assembly **[ ]** with no flow lift conservatively considered. The limiting holddown spring load is taken from the 4th pump startup case for UTL guide tube growth derived in the evaluation of the margin against flow liftoff in Section 5.1.9. A margin of safety of **[ ]** and a fatigue usage of **[ ]** was obtained.

The evaluation of the bottom nozzle for faulted operation was performed in accordance with ASME Code, Appendix F, paragraph F-1440(a) (Reference 21) using a design limit of 80% of the maximum cold test load of **[ ]**. The maximum test load is further discounted for operating temperature conditions. The maximum normal operating load used in the evaluation was conservatively taken as **[ ]** plus moment loads of **[ ]** for a total LOCA plus SSE axial loading of **[ ]**. The SRSS combination of the maximum LOCA and SSE loads is **[ ]**, which includes the weight of the assembly. The axial load equivalent of the moment couples of **[ ]** is created by the position of the guide tubes in relation to the center of the bottom nozzle. A margin of safety of **[ ]** was obtained.

The calculated margin of safety for the bottom nozzle during 4g shipping load is **[ ]** percent and **[ ]** for handling. The bottom nozzle is structurally adequate under shipping and handling loads.

The top nozzle structure was evaluated for normal operating and shipping and handling loads using an ANSYS FEA model. The normal operating evaluation used the combined maximum holddown spring load established by the holddown evaluation per Section 3.9.1, and the weight of the fuel assembly as the limiting case to be evaluated. The faulted evaluation considered the SRSS combination of the LOCA and SSE axial loads. The applicable margins of safety obtained for the membrane and membrane + bending stresses are summarized in Table 5-2 for normal operation, Table 5-3 for faulted conditions, and Table 5-4 for shipping and handling. A fatigue usage of **[ ]** was obtained.

Table 5-2 Top Nozzle Summary of Margins of Safety for Normal Operation

Table 5-3 Top Nozzle Summary of Margins of Safety for Faulted Conditions

Table 5-4 Top Nozzle Summary of Margins of Safety for Shipping and Handling

[ ]

5.1.1.4 Connections

Design Criterion

> The design criterion of the fuel assembly connections is the same as that given in Reference 3, which is based on the ASME B&PV Code, Reference 21, limits and meets the requirements of Reference 1.

The evaluation showed that sufficient margin exists for each connection during normal operation and handling. Table 5-7 provides a summary of normal operating stresses and margins of safety evaluated for each connection.

The upper guide tube QD connection to the top nozzle is shown in Figure 3-12. Hot compression tests of the upper QD connection (see Section 3.9.2) showed that the mechanical strength of the QD connector is limited by the performance of the guide tube and not by the welded connection between the sleeve and guide tube. The design load limit for the QD connector was established as 0.44 times the minimum compression test load of **[ ]** or **[ ]** per connector. The evaluated maximum load on the connectors is **[ ]** per connector a margin of safety of **[ ]** can be demonstrated.

The U.S. EPR lower guide tube connection between the guide tube end plug and the bottom nozzle is illustrated in Figure 3-13. A 304 stainless steel fastener threads into a threaded M5[TM] end plug that is welded to the end of each guide tube. The 304 stainless steel bottom nozzle is captured and compressed by the fastener to form the joint. The Von Mises stress based on the initial preloaded condition of both the fastener and the end plug were evaluated to be within design limits (see Table 5-5).

These stresses are then adjusted to reflect the addition of handling loads as shown in Table 5-6. And finally, the fourth pump startup (4 PSU) and HFP conditions of the end plug are determined from the initial preloaded condition. The coefficient of thermal expansion of the fastener and the bottom nozzle are greater than for the threaded plug.

Therefore, preload is lost during system warm-up because the axial growth of the fastener moves the plug further than the plug expands. In addition, the differences in thermal expansion coefficients introduce large hoop stresses, exceeding the yield point of the material within the end plug. Since these thermal stresses are strain limited they shake down to elastic action without strain rupture of the end plug or loss of function after the initial heating cycle. The end plug was evaluated on a plastic basis in which a factor of safety of **[ ]** was calculated based on the strain rupture limit of the end plug.

Table 5-5 Guide Tube Screw and End Plug Preload Stress, psi

Table 5-6 Guide Tube Screw and End Plug Preload Stress+Handling, psi

Table 5-7 Connections Normal Operating Stress Evaluation

Component	Parameter	Criterion	Value	Margin of Safety, %
Screw Connection Guide Tube to Bottom Nozzle	Total Strain Resulting from Reactor Startup	Strain ≤ 3.23%	[ ]	[ ]
Screw Connection Holddown Spring to Top Nozzle	Von Mises Stress	Axial Strs ≤135 ksi (Bolt Body)	[ ]	[ ]
		Thd Shr≤10,733 psi (Top Nozzle)	[ ]	[ ]
QD Guide Tube to Top Nozzle	Force	F ≤ 394 lb_f	[ ]	[ ]
Welded Connections	Axial Force	F ≤ 220 lb_f	[ ]	[ ]
Spacer Sleeves	Von Mises Stress	S_{vm} ≤ 9,430 psi	[ ]	[ ]

5.1.1.5 Spacer Grids

Design Criterion

> No grid crushing deformations occur during normal operation and OBE conditions. The grids shall also provide adequate support to maintain the fuel rods in a coolable configuration for each condition (References 1, 4 and 14).

The mechanical design bases of the U.S. EPR spacer grids were confirmed through a series of tests on U.S. EPR representative prototype 17x17 M5^{TM} HTP grids.

- Dynamic Impact (HTP) - The dynamic characteristics (i.e., impact force, impact duration, pre- and post-impact velocity, grid permanent deformation, dynamic stiffness, and damping) were used as input properties for the analytical models of the fuel assembly to establish allowable impact loads, and to characterize the plastic deformation of the grids. The grid strength was taken as the 95/95 one sided confidence of the mean elastic limit for grids at BOL conditions. The elastic limit was established as the impact force corresponding to a 0.2 % permanent deflection across the grid width using a quadratic regression of the impact force/deflection data obtained from testing. See Table 5-8 for test results. Structural margins for the HTP grid for faulted conditions are provided in Table 5-17.

The grid evaluation for dynamic impact was performed against the strength criterion defined in the recently revised Standard Review Plan (Reference 1). The standard review plan indicates that the grid strength can be evaluated at BOL conditions and that this is also appropriate for irradiated conditions. The methodology used to evaluate the grids in the horizontal faulted analysis in section 5.3.4.1, per reference14, is in compliance with the Standard Review Plan (Reference 1).

- Static Crush (HTP) - The static characteristics (i.e., static stiffness and elastic load limit) were used to establish allowable grid clamping loads during shipping.

- Slip Load (HTP, HMP) - The forces required to slip the grid relative to the fuel rods were measured at BOL conditions. These data, which represent the friction force between the grids and fuel rods, were used as input in analytical models of the fuel assembly. Typical BOL slip loads at room temperature of **[ ]** for HTP grids and **[ ]** for unrelaxed HMP grids were obtained.

- Corner Hang-up (HTP) - The U.S. EPR HTP grid corners have been designed, through the use of lead-in surfaces, to minimize the potential for grid hang-up. Previous testing on 15x15 M5™ HTP grids fuel assemblies showed that M5™ grids have a corner strength of no less than **[ ]** which is much greater than the maximum pull and push handling load criteria

[ ] for the U.S. EPR design. The test results were conservative because the grid corners are designed with lead-in to force grids to lead-off laterally and reduce the interference. The testing method allowed no lead-off.

In addition, these spacer grid tests determined that the failure mode of the corner cell (i.e., simulating grid hang-up) was through weld fracture with very little outer strip and corner deformation. Given the similarities in the design, these test results are considered applicable for the U.S. EPR.

Table 5-8 U.S. EPR HTP Spacer Grid Dynamic Impact Test Results

[ ]

5.1.1.6 Holddown Spring

Design Criterion

The design criterion of the fuel assembly holddown spring is based on the ASME B&PV Code (Reference 21) which meets the requirements of Reference 1.

Stress analysis of the U.S. EPR fuel assembly holddown spring examined stresses, strains, and fatigue usage to confirm that it does not break. The evaluation confirmed that the ASME Code criteria are satisfied.

The maximum normal operating loading occurs when the head is bolted on and the fourth pump is started (i.e., at cold conditions). Even the faulted loads do not exceed this normal condition. Since the reactor coolant pump lift forces do not raise the fuel assembly off the lower core plate, the holddown spring stresses are displacement controlled by the total separation between the lower and upper core plates.

Section 5.1.9 shows that worst case flow rates do not lift the fuel, so the stresses in the springs are displacement controlled. Because of this the spring stresses can be treated as secondary stresses. However, some fraction of the spring force is necessary to hold the fuel assembly down (i.e., to satisfy internal equilibrium). A portion of' the total stress was treated as primary.

The spring stresses are also displacement limited because they cannot be deflected past the solid position where testing has demonstrated that the leaves can be nearly flattened against the top nozzle, with no known failures. This means the maximum test load is not the ultimate load to be used in the ASME Code primary stress evaluations and that the ultimate load is very high in comparison to the compressive limit of the spring cavity in the top nozzle. The primary loads are only a small fraction of the ultimate load. The intent of the primary stress limits is to prevent a primary load from forcing the structure to ultimate failure. Therefore, the primary stress criteria are satisfied because failure of the structure due to primary stresses cannot occur because the geometry of the spring leaves.

Given that slow fuel assembly growth due to radiation fits Article NG-3213.17 of Reference 21 definition of creep, the secondary stresses limits are satisfied by performing a plastic analysis to Article NG-3228.1 of Reference 21. The holddown springs shake down to elastic action, as required.

The fatigue usage factor was analyzed using Rigide, a special-purpose finite element program. This program was written specifically for the design of leaf holddown springs and benchmarked to actual force deflection test data for the U.S. EPR spring design, which agrees with the full range of actual test data.

The normal and upset loads are the imposed displacements derived from the evaluation of fuel assembly liftoff (see Section 5.1.9), vertical SSE, and vertical LOCA (see Section 5.3.4.2). The number of cycles was developed by sorting and grouping the listing of RCS transients in Table 5-14, resulting in the grouped list of transients shown in Table 5-9 that were used in the evaluation.

Table 5-9 Grouped Transients

[ ]

Strain ranges were determined using the Rigide program, converted to alternating stress intensity ranges and compared to fatigue allowables.

The strains associated with 3 cycles of heat-up and cool-down are determined by imposing the displacements derived in the evaluation of fuel assembly liftoff (see Section 5.1.9). The displacements are shown in Figure 5-3 for the 3 cycles of heat-up and cool-down transients. The points in the figure for each cycle are as follows.

- Cycle 1 points: BOL cold – 1. BOL hot – 2. EOC1 hot – 3 with UTL growth. Intermediate point – 4. EOC1 cold with UTL growth – 5.

- Cycle 2 points: BOC2 cold – 5. BOC2 hot – 6. EOC2 hot – 7 with UTL growth. Intermediate point – 8. EOC2 cold with UTL growth – 9.

- Cycle 3 points: BOC3 cold – 9. BOC3 hot – 10. EOC3 hot – 11 with UTL growth. Intermediate point – 12. EOC3 cold with UTL growth – 13.

Figure 5-1 Transients Strains over 3 cycles of Heat-Up and Cool-Down

[ ]

Similarly, the local strain ranges were determined using the Rigide program, converted to alternating stress intensity ranges and compared to fatigue allowables.

The number of alternating stress cycles are then compared to Ni, the number of cycles allowed, as determined in NUREG/CR-6909 (Reference 26). The normal and upset transient fatigue usage calculation results are summarized in Table 5-10.

Table 5-10 Holddown Spring Derived Fatigue Usage Factors, Ui

[ ]

Multiplying the total usage factor by the ASME code environmental correction factor of 4.3 yields a final fatigue usage factor of **[ ]**. Note that this environmental correction factor is associated with a Class 1 component. Since the holddown spring is not a Class 1 component this fatigue usage factor is conservative.

The faulted loads for the SSE and LOCA events are from the vertical evaluations in Section 5.3.4.2 where the fuel assembly lifts about **[ ]** maximum during an earthquake and **[ ]** during a LOCA. These loads produce trivial strain ranges for the fatigue calculations.

Although the LOCA and SSE stresses in the springs become primary, these stresses or loads are still below 0.8 Lu (i.e., ultimate test load) for faulted conditions, regardless of when they occur on the hot tracks in Figure 5-1. They are also well below the ultimate stress or load.

5.1.1.7 Cladding Stress

Design Criterion

> Fuel rod cladding stress shall not exceed stress limits established in Reference 7 that are provided below:

- Pm < 1.5 Sm in compression and < Sm in tension
- Pm + Pb < 1.5 Sm
- Pm + Pb + PL < 1.5 Sm
- Pm + Pb + PL + Q < 3.0 Sm.

The classification of the types of stresses that were evaluated is summarized in Table 5-11 and described as follows:

- Pressure Stresses - These are membrane stresses from external and internal pressure on the fuel rod cladding.
- Flow Induced Vibration (FIV) - These are longitudinal bending stresses from vibration of the fuel rod. The vibration is caused by coolant flow around the fuel rod.
- Ovality - These are bending stresses from external and internal pressure on the fuel rod cladding that is oval. This does not include the stresses resulting from creep ovalization into an axial gap.
- Thermal Stresses - These are secondary stresses that arise from the temperature gradient across the fuel rod during reactor operation.
- Fuel Rod Growth Stresses - These secondary stresses are from the fuel rod slipping through the spacer grids. These may occur when the fuel assembly expands more than the fuel rod due to heat-up, or occur because of fuel rod growth from irradiation.
- Fuel Rod Spacer Grid Interaction - These are secondary stresses from contact between the fuel rod cladding and the spacer grid.
- Plenum Spring Force – This is a primary membrane stress from axial loading from the plenum spring.

Table 5-11 Classifications of Fuel Rod Clad Stresses

Loading Condition	Stress Category	
Pressure Stresses	Pm	Primary membrane
Ovality Stresses	Pb	Primary membrane bending
Spacer Grid Interaction	Q	Secondary
FIV	Pb	Primary membrane bending
Radial Thermal Expansion	Q	Secondary
Differential Rod Growth	Q	Secondary
Plenum Spring Force	Pm	Primary membrane

The fuel rod cladding was analyzed for the stresses induced during operation using the approved methodology of Reference 7 for which the applicability is demonstrated in Reference 2. The clad stress analysis follows the guidelines of Reference 21, which specifies the use of stress intensities. This method is known as the maximum shear stress criteria. The stress states modeled for the M5^{TM} cladding are maximum compression and maximum tension.

Conservative values are used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential thermal expansion, and unirradiated cladding yield strength. The fuel rod stress analysis calculates the worst-case cladding stress state based on the thinnest clad wall and largest cladding ovality. The likelihood that the two conditions occur at the same location on the cladding is remote. Therefore, the use of the two conditions together to calculate the cladding stress state is conservative. The analyses of the fuel rod clad stresses is summarized in Table 5-12 which demonstrates positive margins for each operating condition.

Table 5-12 U.S. EPR Fuel Rod Stress Result Summary

Category	Limits	Stress Allowable (psi)	% Margin	
			Compressive	Tensile
Primary Membrane (Compressive)	1.5 Sm	[ ]	[ ]	[ ]
Primary Membrane (Tensile)	Sm	[ ]	[ ]	[ ]
Primary Membrane + Bending	1.5 Sm	[ ]	[ ]	[ ]
Primary Membrane + Bending + Local	1.5 Sm	[ ]	[ ]	[ ]
Primary Membrane + Bending + Local + Secondary	3.0 Sm	[ ]	[ ]	[ ]

Note: The minimum unirradiated hoop yield strength of the cladding at 664°F used is **[ ]**.

The fuel rod cladding is analyzed to confirm that buckling does not occur. The two critical buckling pressures, Pcr and Pyp, are calculated. Pcr is the bifurcation buckling pressure of a perfectly circular shell and is calculated to check the elastic stability of the cladding. Pyp is the pressure at which the cladding extreme fiber is loaded beyond the yield point and it accounts for cladding initial ovality. The maximum differential pressure is less than the buckling pressure and the critical pressure, thereby proving that the cladding does not buckle.

5.1.2 Cladding Strain

Design Criterion

The U.S. EPR fuel rod transient strain limit is 1% for AOOs per Reference 7.

The U.S. EPR fuel rod was analyzed to determine the maximum transient load the fuel rod cladding could experience before exceeding the transient strain limit of 1 %. The transient strain limit analysis uses cladding circumferential changes before and after a linear heat rate (LHR) transient to determine the strain. The analysis was conducted using the NRC-approved COPERNIC Fuel Rod Design Computer Code (Reference 17). The formula for determining the transient strain is:

$$\varepsilon_{hoop} = \left[D_{Clad,et} - D_{Clad,bt}\right] / D_{Clad,bt}$$

Where:

$\varepsilon_{hoop} =$ cladding uniform hoop strain

$D_{Clad,bt} =$ local cladding mean fiber diameter before the transient

$D_{Clad,et} =$ local cladding mean fiber diameter at the end of the transient.

The calculated LHRs for transients that induce one percent cladding strain are not limiting to the operation of the plant and are much greater than the maximum transient the fuel rod is expected to experience.

5.1.3 Cladding Fatigue

Design Criterion

The cumulative fuel rod fatigue usage factor shall not exceed 1.0.

The fuel rod was analyzed for the total fatigue usage factor using the approved methodology from Reference 7 and the procedures outlined in the Article NG-3222.4 of Reference 21. Testing has been conducted by AREVA NP to determine the fatigue performance of M5™ cladding. These tests have shown similar fatigue endurance performance for recrystallized annealed (RXA) claddings as compared to Zircaloy-4, with the lower yield strength of the RXA claddings limiting the applied stresses. The values for S_{alt} versus N cycles obtained are well enveloped by the standard O'Donnell-Langer design fatigue curve for irradiated Zircaloy-4. A fuel rod life of eight years and a vessel life of 60 years are assumed. The fuel rod cladding, therefore, experiences 13 % of the number of transients the reactor pressure vessel experiences. The expected normal operating upset, and test transients were evaluated to determine the total fatigue usage factor experienced by the fuel rod cladding. In accordance with Reference 21, faulted conditions are not included in the fatigue evaluation.

Conservative inputs in terms of cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, and differential temperature across the cladding were assumed.

The results of the example fatigue analysis for the U.S. EPR fuel rod show that the fatigue usage factor is **[ ]**, which is well below the limit of 1.0.

5.1.4 Fretting

Design Criterion

> Span average cross-flow velocities shall be less than 2 ft/sec., per Reference 3.

A full core analysis of the U.S. EPR fuel demonstrated span average cross-flows less than **[ ]**. The 2 ft/sec criterion precludes unacceptable FIV of the fuel rods. The cross-flow velocities were determined using the NRC-approved LYNXT code per Reference 18, which established the flow and pressure drop characteristics of the U.S. EPR fuel assembly for full core implementation.

Design Criterion

> The fuel assembly design shall provide sufficient support to limit fuel rod vibration and clad fretting wear.

The favorable U.S. EPR fuel rod fretting and wear performance is based on the following tests and evaluations:

- Full scale 1000 hour endurance flow testing performed on a 12 ft fuel assembly design using Zircaloy-4 fuel rods and Zircaloy-4 HTP grids which was the basis for evaluation and approval by the NRC for HTP grids in Reference 20.

- Favorable U.S. operating experience with 12 ft fuel assemblies incorporating both Zircaloy-4 and M5™ fuel rods and HTP grids.

- Full scale 1000 hour endurance flow testing on a 14 ft prototype (with characteristics representative of the U.S. EPR fuel assembly) in the HERMES-P flow test facility. Extremely unfavorable rod support conditions and extremely large cross-flows were tested.

- Supplemental out-of-core life and wear and FIV testing using the PETER Loop Autoclave test methodology accompanied by performance benchmarks against fretting performance of other designs.

- Negative results for specifically targeted tests for self induced vibration modes performed with full scale fuel assembly prototypes in two independent test facilities, namely the PETER Loop and Hermes-P facilities.

The basis for extension of the fretting resistance of Zircaloy-4 HTP grids on Zircaloy-4 fuel rods demonstrated in Reference 20 to M5™ HTP grids and fuel rods is provided in Reference 7. The fretting resistance of 12 ft HTP fuel assemblies based operating experience can also be extended to the U.S. EPR 14 ft assembly that is in other ways similar to existing 12 ft designs. Operating experience demonstrates that the fretting of PWR fuel rods is typically localized in the lowermost regions of fuel within the flow recovery regions just beyond the bottom nozzle where the flows are most turbulent and cross-flow conditions are most likely to exist. The fretting behavior is independent of fuel assembly length since the key observed parameters governing fretting resistance are associated with rod support characteristics, cross-flow velocities, and lower span distance and materials.

The U.S. EPR design does not introduce additional features or characteristics other than overall length to the evaluation. Span lengths between spacer grids are no greater than those used on existing 12 ft designs.

The 1000 hour HERMES-P flow testing was conducted under extreme and very conservative tests conditions. These tests were not intended to provide the sole basis for licensing but are presented only to characterize the fretting performance. These fretting observations are based on the extreme conditions needed to produce significant discriminating levels of fretting characterizing the fretting behavior associated with grid features and test conditions. The tests used varying levels of grid contact ranging from very slight interference of 0.002 inch to open gap clearances as large as 0.004 inch between fuel rods and spacer grids. The tests were performed at operating temperature using a full scale fuel assembly prototype with characteristics representative of a U.S. EPR fuel assembly. The testing incorporated a large cross-flow injection port at the bottom span so that the total flow exiting the fuel assembly is greater than the flow entering the bottom nozzle. Rods associated with very light support conditions, not including the two rods directly in front of the cross-flow injection, showed fretting 2 mils deep or less over the full duration of the endurance testing.

Supplementary tests to establish FIV performance have been conducted with OL3 EPR fuel using the PETER Loop flow test method. This testing is applicable to the U.S. EPR. The PETER Loop tests measure fuel rod and fuel assembly behavior for a full scale fuel assembly prototype during parametric in-reactor flow conditions including twice the nominal cross-flow. The FIV testing served to characterize the FIV behavior of the fuel assembly. Vibration amplitudes for the fuel assemblies were low, which measured **[ ]** microns root mean square (RMS) **[ ]** with a 1 Hz high pass filter. No abnormal flowrate dependencies were observed for the fuel assembly vibration amplitudes.

Additional fretting wear tests were performed on a single fuel rod in water in the Autoclave test. The dynamic input from PETER Loop flow test with added conservatism is used for single rod fretting test. The rod is subjected to precisely replicate mechanical support conditions and mechanical excitations representing worst case rod support conditions including EOL plus open gap rod-to-grid interface conditions.

Fretting marks were not found on the EOL Autoclave tested fuel rod specimens with flow exposures of up to 1000 hours. The good fretting behavior is correlated with good line contact and higher inherent damping of the HTP grid design. AREVA has concluded that the U.S. EPR fuel assembly is fretting resistant against excitations that could be experienced in an EPR reactor.

The PETER Loop and Autoclave fretting methodology is benchmarked against actual in reactor experience on fuel designs for which fretting and wear parameters have been previously measured and characterized. The Autoclave tests showed good correlation to the fretting behavior experienced on non-HTP fuel assemblies that use the spring dimple rod support configuration.

The test results confirm that due to cross-flow within the flow recovery regions near the bottom of the assemblies, the bottom span experiences the most turbulent flow regimes. Very conservative and bounding rod motions were imposed during the wear tests for significant duration and no excessive wear was produced. The applicability of these results and conclusions to reactor conditions is supported by extensive operating experience.

Operating experience also supports the out-of-core test results. As of December 2006, more than 7894 fuel assemblies with HTP spacers have been in operation in 41 nuclear power plants worldwide. Of this population, more than 4000 assemblies have achieved a burnup greater than 40 GWd/mtU, with a maximum assembly burnup of 70 GWd/mtU achieved. In particular of the 7894 assemblies, only 4 fuel rods (two fuel assemblies) associated with designs using all Zircaloy HTP grids and bi-metallic end grids have experienced failure due to grid-to-rod fretting. When HTP grids were introduced at the end grid locations, the grid-to-rod fretting failures ended.

In summary, the U.S. EPR fuel rod fretting wear performance is determined acceptable based on relevant incore experience, in conjunction with extensive and conservative out-of-core testing.

5.1.5 Oxidation, Hydriding, and Crud Buildup

Design Criterion

> The fuel rod cladding best-estimate corrosion shall not exceed 100 microns per Reference 10. Hydrogen pickup is controlled by the corrosion limit.

Corrosion data of M5™ fuel rod cladding are provided in Reference 7. The data confirm that M5™ fuel rod cladding exhibits a strong resistance to corrosion. From previous irradiation experience with this cladding type, the corrosion has been found to be less than one half the corrosion of low-tin Zircaloy-4 cladding. For the present application, a corrosion prediction based on the database of M5™ corrosion measurements in the current version of COPERNIC for typically enveloping fuel cycles shows that the maximum cladding corrosion to be **[ ]** μm versus a limit of 100 μm. Recently updated PIE data show the actual average oxide thickness to be significantly lower than this predicted maximum. Crud buildup is included as part of the oxidation measurement. Therefore, crud is limited and is within the total acceptable range. The hydrogen pickup rate of the M5™ cladding has been found to be approximately **[ ]**.

At this corrosion level, the maximum hydrogen content of the M5™ cladding at 65 GWd/mtU is approximately **[ ]** ppm. The upper limit for hydrogen pickup is **[ ]** ppm. This level of corrosion and associated hydriding does not adversely affect the structural integrity of the fuel rod during its design lifetime.

5.1.6 Fuel Rod Bow

Fuel rod bowing is evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. Although there is no specific design mechanical criterion for fuel rod bow, the design follows the rod bow limits as established in Reference 10.

The current post-irradiation examination (PIE) database of fuel rod bow performance, measured as the standard deviation (i.e., mils) of the minimum of water channel (i.e., gap space between adjacent fuel rods) is provided in Figure 5-2. Assembly features such as M5™ cladding are segregated for convenience.

Figure 5-2 Worst Case Water Channel Closure versus Burnup

Because there is no U.S. fuel rod bow data specific for fuel assemblies with HTP grids, AREVA has performed a comparative evaluation of the U.S. EPR fuel assembly with respect to existing fuel designs in order to predict future performance relative to current bowing data. Key factors such as slip load, rod dimensions, span lengths, rod growth, and physics were evaluated.

The design slip load of the U.S. EPR HMP upper end grid is less than any other approved 17x17 designs. Because the upper end grid slip is 44% lower than the lower end grid the slip of the upper end grid governs the performance of the assembly with regard to rod bow. Intermediate M5™ grid slip loads were not considered to be a factor because they will relax early in the first cycle of operation. The high degree of fuel rod rotational fixity afforded by HTP and HMP grid line contacts significantly improves any adverse tendency for rod bow.

The influence of span length between spacer grids on rod performance is not an adverse factor because the average span lengths for the U.S. EPR design are shorter than that for 12 ft MK-BW designs for which PIE data is available. Likewise, the magnitude of flux applied to the U.S. EPR fuel rods is consistent with other core designs for which PIE data is available.

The lower growth characteristics of the M5™ advanced material that is used on the U.S. EPR can be expected to result in fuel rod bow behavior that is no more severe than the Zircaloy-4 clad fuel. The growth of M5™ for equivalent burnups across the entire range of the database has been demonstrated by irradiation experience to be consistently lower than Zircaloy-4 for each fuel design type, both in the U.S. and in Europe.

In consideration of the PIE data and the comparative design feature evaluations, AREVA has concluded that rod bow performance is similar to that of other AREVA NP designs and that the rod bow correlations from Reference 10 and Reference 15, previously approved by the NRC, are applicable to the U.S. EPR fuel assembly design including consideration of extended burnup conditions.

5.1.7 Axial Growth

Design Criterion

> The fuel assembly-to-reactor internals gap allowance shall be designed to provide positive clearance during the assembly lifetime (see Reference 4).

The axial gap between the top nozzle and reactor upper core plate was conservatively analyzed to show that sufficient margin exists to accommodate the fuel assembly growth for the design burnup. The peak rod burnup for the U.S. EPR design is 62 GWd/mtU. The analysis was conducted using the latest irradiation growth models for Alloy M5™ guide tubes and fuel rods based on PIE data for the AREVA NP fuel designs.

The core plate gap margin is evaluated using UTL fuel assembly length, fuel assembly thermal expansion and irradiation growth, and lower tolerance limit (LTL) core plate distances including thermal expansion. The BOL cold (i.e., room temperature) core plate gap is obtained by subtracting the length of the fuel assembly from the distance between the upper and lower core plates. BOL hot (i.e., HFP operation) is obtained by adding the LTL core cavity thermal expansion to the cold BOL core plate gap and subtracting the fuel assembly thermal expansion. The EOL cold core plate gap is determined by subtracting the UTL fuel assembly irradiation growth from the BOL cold core plate gap.

Subsequently, the EOL hot core plate gap is determined by subtracting the UTL fuel assembly irradiation growth from the BOL hot core plate gap.

Fuel assembly irradiation growth is evaluated using U.S. EPR specific growth limits (Section 5.1.7.1) whereas the distance between the upper and lower core plates is assumed to be unaffected by irradiation. Fuel assembly thermal expansion is evaluated at the average coolant temperature at HFP using fuel assembly expansion rates for the M5™ guide tubes and for the stainless steel end fittings. Thermal expansion of the stainless steel core cavity as a function of average coolant temperature at the operating points of interest was considered.

The minimum fuel assembly to reactor core plate gap at EOL for a 65 GWd/mtU maximum assembly burnup was determined to be **[ ]** inch at worst-case (i.e., cold) conditions. A conservative maximum fuel assembly growth prediction was used given the low fuel assembly growth.

Design Criterion

> The fuel assembly top nozzle-to-fuel rod gap allowance shall be designed to provide positive clearance during the assembly lifetime.

The axial gaps between the top nozzle adapter plate and fuel rods were conservatively analyzed to show that sufficient margin exists to accommodate the fuel assembly and fuel rod growth for the design burnup. The peak rod burnup for the U.S. EPR design is 62 GWd/mtU. U.S. EPR specific fuel assembly growth and fuel rod limits were used in the evaluation as addressed in Section 5.1.7.1.

The fuel rod shoulder gap calculation considers as-built fuel rod length, fuel rod thermal expansion, UTL fuel rod irradiation growth, LTL fuel assembly distance between end fittings, guide tube thermal expansion, and LTL guide tube irradiation growth. The BOL cold fuel rod shoulder gap is calculated by subtracting the length of the fuel rods from the distance between the top and bottom nozzles. Combined tolerances on as-fabricated dimensions are determined using the square root of the sum of the squares method. The minimum EOL cold fuel rod shoulder gap is calculated by adding the LTL fuel assembly growth to the BOL cold shoulder gap and subtracting the UTL fuel rod growth due to irradiation. The EOL hot shoulder gap is calculated by adding the fuel assembly thermal growth to the EOL cold shoulder gap and subtracting the fuel rod elongation due to thermal expansion at elevated temperature. Accumulated UTL fuel rod irradiation growth at EOL was obtained from the U.S. EPR growth limits (see Section 5.1.7.1).

The fuel rod thermal expansion was evaluated using the M5^{TM} coefficient of thermal expansion ([ ]), per Reference 7 at the maximum average fuel rod temperature [ ]. The maximum average fuel rod temperature was obtained using the enveloping case rod temperature determined by the COPERNIC fuel rod code (applicable for the U.S. EPR per Reference 2) runs that were used to evaluate the bounding linear heat generation rates. UTL fuel rod irradiation growth at 62 GWd/mtU was obtained from the M5^{TM} growth limits addressed in Section 5.1.7.1. The fuel assembly thermal elongation was derived with the coefficient of thermal expansion for M5^{TM} guide tubes provided in Reference 7 and average coolant temperatures for HFP operation.

The minimum fuel rod shoulder gap at EOL for the analyzed peak fuel rod burnup of 62 GWd/mtU was calculated to be [ ] inch at worst-case (i.e., hot) conditions using highly conservative methods. For the fuel rod growth evaluations, worst case was considered to be maximum fuel rod growth and minimum] guide tube growth. Differential thermal expansion was also considered at operating temperatures.

5.1.7.1 Fuel Assembly and Fuel Rod Growth Limits

U.S. EPR specific axial fuel assembly and fuel rod growth limits were developed for the following evaluations which consider fuel assembly axial growth due to irradiation:

- Fuel assembly-to-reactor internals gap allowance
- Fuel rod shoulder gap allowance
- Verification of margin against flow liftoff (holddown)

U.S. EPR Fuel Assembly Growth Limits

The U.S. EPR fuel assembly growth model was derived by the application of the empirical M5^{TM} guide tube irradiation growth data. These data, obtained to date in conjunction with calculations, consider the specific U.S. EPR design attributes and M5^{TM} free irradiation growth and creep as a function of temperature and stress. The calculation method was validated by benchmarking against existing M5^{TM} growth data for the AREVA 12 ft Mark-BW fuel assembly.

The consideration of guide tube stress as defined by the governing design variables and loadings in combination with correlations with the PIE growth is appropriate in establishing the design criteria. Operating experience regarding fuel assembly growth has been applied to the U.S. EPR fuel design.

The growth limits thereby established for the U.S. EPR design are shown in Figure 5-3.

Figure 5-3 U.S. EPR Fuel Assembly Growth Limits

U.S. EPR Fuel Rod Growth Limits

The U.S. EPR fuel rod growth model was derived by the application of the empirical M5^{TM} fuel rod irradiation growth data obtained to date by PIEs. The growth limits consider the statistical upper and lower limits of all the M5™ data collected considering all fuel designs and operating conditions as represented in the data.

The fuel rod growth limits established for the U.S. EPR design are shown in Figure 5.4.

Figure 5-4 U.S. EPR Fuel Rod Growth Limits

5.1.8 Fuel Rod Internal Pressure

Design Criterion

Fuel rod internal pressure limits are established in the Fuel Rod Gas Pressure Criterion Report (Reference 16). The design basis is that the fuel system will not be damaged due to excessive internal pressure. Fuel rod internal pressure is limited to that which would cause the diametral gap to increase because of outward creep during steady-state operation, and extensive departure from nucleate boiling (DNB) propagation to occur.

The U.S. EPR fuel rod internal gas pressure was determined using the COPERNIC computer code per Reference 17 and the methodology defined in Reference 16.

The results indicated the fuel rod can attain the design maximum burnup of 62 GWd/mtU. Inputs to the analysis included a power history that was assumed to envelop the operation of any individual fuel rod and worst-case manufacturing variations allowed by the fuel rod specifications. On a cycle-specific basis, if peak pin powers violate the envelope resulting in predicted pressure greater than the licensed limit, acceptable pin pressure results can be demonstrated by using fuel rod-specific power histories and fuel assembly as-built manufacturing data.

5.1.9 Assembly Liftoff

Design Criterion

> The U.S. EPR fuel holddown springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating AOOs, except for the pump overspeed transient. The fuel assembly shall not compress the holddown spring to solid height for AOOs. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all AOOs and design basis accidents (DBA) per Reference 3.

The U.S. EPR holddown springs were analyzed to show that the holddown springs can accommodate irradiation growth of the fuel assembly and the differential thermal expansion between the fuel assembly and the core internals. The fuel assembly lift evaluation was performed by comparing the holddown force from the leaf springs and fuel assembly weight with that of the hydraulic forces at both normal operating conditions and pump overspeed conditions. The spring characteristics were determined by load deflection testing. Hydraulic forces were determined using the NRC-approved LYNXT code per Reference 18, which was used to determine the worst case flow lift forces. Spring plasticity from deflections exceeding the elastic limit and spring relaxation from irradiation were considered.

In addition to irradiation growth of the fuel assembly, the lift forces change with flow rate, pressure and temperature according to specific operating state points, as given in Table 5-13. The following normal operating state points were considered:

- 4PSU - 4th Pump Startup 140°F and 370 psi
- HFP - Hot full power
- HZP - Hot Zero Power
- 120%OS - 120% Pump Over-speed
- HSD - Hot Shutdown
- HSB - Hot Standby
- 4PSUP - Proposed Higher Temp. for 4th Pump Startup 450°F ; 2250 psi

Table 5-13 System State Point Parameters for Assembly Liftoff Evaluation

[ ]

Flow conditions ranging from the fourth pump startup at 140°F to pump overspeed at 120% flow at full power (FP) conditions were considered. Although a pump overspeed event is not considered a credible event for the U.S. EPR, the evaluation considers this case as enveloping with respect to the magnitude of the flow lift forces that are possible for any other anticipated operational occurrence (AOO); therefore no other AOOs need to be evaluated. Beginning-of-life (BOL) and end-of-life (EOL) conditions were also evaluated to consider the change in load paths and loads due to grid material relaxation resulting in lower grid slip loads and fuel rod contact with the bottom nozzle.

A very conservative deterministic methodology was used to determine spring deflections and loads. This methodology uses the algebraic sum of extreme variations in core plate separation and fuel assembly growth and length when determining spring deflections and loads. This conservative method results in smaller margins than those that may be calculated using a statistical methodology. In the future, additional U.S. EPR holddown evaluations may use the statistical holddown methodology, Reference 22, approved by NRC in Reference 2.

Using the deterministic evaluation approach, liftoff during normal operating conditions does not occur, except for 120% pump overspeed conditions. A minimum **[ ]** margin-to-fuel assembly liftoff occurs at EOL, at 600^{o}F under HFP conditions assuming LTL fuel assembly growth, UTL core plate separations, and LTL fuel assembly length tolerances.

For the 120% pump overspeed condition (not applicable but bounding) the fuel assembly experiences some liftoff. The liftoff is minimal, and the holddown spring deflection is less than the worst-case normal operating cold-shutdown condition. The holddown spring does not go solid for any operating condition. These margins are calculated assuming a full core of U.S. EPR fuel assemblies. In addition, the fuel assembly top and bottom nozzles are shown to maintain engagement with reactor internals for each operating condition.

5.2 Fuel Rod Failure Criteria

5.2.1 Internal Hydriding

Design Criterion

> Internal hydriding shall be precluded by appropriate manufacturing controls per References 8 and 10.

The absorption of hydrogen by the cladding can result in cladding failure because of reduced ductility and the formation of hydride platelets. This failure mechanism is precluded in AREVA NP fuel rods by tight controls in the moisture hydrogen impurities in the rod during fabrication. Cleaning and drying of the cladding, and careful moisture control of the fuel pellets are used to minimize the total hydrogen within the fuel rod assemblies. The AREVA NP fabrication limit for total hydrogen in the fuel pellets is **[ ]** ppm.

5.2.2 Cladding Collapse

Design Criterion

> The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected incore life.

A creep ovalization analysis program, CROV, developed and certified for licensing AREVA NP fuel rods, is used to evaluate the resistance of the U.S. EPR fuel rod cladding to creep collapse. Use of the CROV code to perform the creep collapse analysis for M5^{TM} cladding, as addressed in Reference 7, has been approved by the NRC. Inputs to the analysis include differential pressure, temperature gradients, and fast flux. The enveloping power histories from the COPERNIC thermal-hydraulic analysis are used to initialize the creep collapse code per Reference 17, thus providing inputs to CROV. As addressed in Reference 7, the creep rate of M5^{TM} is approximately

50% slower than Zircaloy-4; therefore, a multiplier of 0.5 or higher (for conservatism) can be applied to the CROV input for the M5™ analysis. For this analysis, a multiplier of 0.9 was used for conservatism.

The following conservatisms were used in determining creep collapse life of the fuel rod:

- Minimum fuel rod pre-pressure
- No fission gas release
- A worst-case or enveloping power history
- Worst-case cladding dimensions
- Bounding value for cladding thickness
- Bounding value for cladding ovality

Fuel rod creep collapse is determined when either of the following happens:

- The rate of creep ovalization exceeds 0.1 mil/hr.
- The maximum fiber stress exceeds the unirradiated yield strength of the cladding.

Using the methodology and conservatisms described above, the fuel rod creep collapse lifetime was shown to be greater than the design burnup of 62 GWd/mtU.

5.2.3 Overheating of Cladding

Design Criterion

For a 95% probability at a 95% confidence level, DNB will not occur on a fuel rod during normal operation and AOOs.

The requirements related to overheating and cladding are addressed in plant-specific transient analyses. NRC-approved methods are used to perform the transient analyses.

DNB Correlation

Two critical heat flux (CHF) correlations are used for the DNB analysis of the U.S. EPR fuel assembly: BWU-N CHF Reference 11 is applicable from the beginning of the heated length to the leading edge of the first HTP mixing grid. From the leading edge of the first HTP mixing grid to the leading edge of the upper non-mixing HMP grid, the ACH-2 CHF correlation (Reference 12) should be used. The BWU-N and ACH-2 CHF correlations are used as the licensing basis for the U.S. EPR fuel assembly.

BWU-N CHF Correlation

The applicable CHF correlation for DNB analysis of the U.S. EPR fuel assembly in the non-mixing region of the fuel assembly is the BWU-N CHF correlation documented in the BWU Critical Heat Flux Correlations Report (Reference 11). The bottom non-mixing region of the fuel assembly extends from the beginning of the heated length to the leading edge of the first HTP mixing grid.

ACH-2 CHF Correlation

The applicable CHF correlation for analysis of the U.S. EPR fuel assembly in the mixing region is ACH-2 as addressed in the ACH-2 CHF Correlation for the U.S. EPR Topical Report (Reference 12). The mixing region of the fuel assembly extends from the leading edge of the first HTP mixing grid to the leading edge of the upper non-mixing HMP grid.

5.2.4 Overheating of Fuel Pellets

Design Criterion

> For a 95% probability at a 95% confidence level, fuel pellet centerline melting shall not occur for normal operation and AOOs.

The design basis for the U.S. EPR fuel rod centerline melt follows that given in Reference 3. Fuel melting is not permitted during normal operating conditions or AOOs. The COPERNIC computer code documented in Reference 17 was used to determine the local LHR throughout the fuel rod lifetime that results in centerline temperature predictions exceeding T_L, which is a limit value chosen so that a 95% probability exists at the 95% confidence level that centerline melting does not occur. A bounding centerline fuel melt limit for the U.S. EPR fuel is **[ ]** kW/ft for UO_2 rod and **[ ]** kW/ft for UO_2-Gd_2O_3 rod (8 weight % Gd).

5.2.5 Pellet/Cladding Interaction

Per Section 4.2 of the SRP (Reference 1), there are no generally applicable criteria for pellet-clad interaction failure. Clad strain and fuel melt criteria are used to establish that the fuel rod design is acceptable per Reference 7.

5.2.6 Cladding Rupture

The requirements of cladding rupture are addressed in the plant-specific LOCA analyses. NRC-approved methods are used to perform the LOCA analyses.

5.3 Fuel Coolability

5.3.1 Cladding Embrittlement

The requirements on cladding embrittlement are addressed in the plant-specific LOCA analyses. NRC-approved methods are used to perform the LOCA analyses.

5.3.2 Violent Expulsion of Fuel

The requirements on violent expulsion of fuel during a reactivity accident are addressed in the plant-specific safety analyses. NRC-approved methods are used to evaluate the event (i.e., control rod ejection).

5.3.3 Fuel Rod Ballooning

The requirements on fuel rod ballooning are addressed in the plant-specific LOCA analyses. NRC-approved methods are used to perform the LOCA analyses.

5.3.4 Fuel Assembly Structural Damage from External Forces

Design Criteria

- OBE - Allow continued safe operation of the fuel assembly following an OBE event by establishing that the fuel assembly components do not violate their dimensional requirements.
- SSE - Establish safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertibility, and a coolable geometry within the deformation limits consistent with the emergency core cooling system (ECCS) and safety analysis.

- LOCA or LOCA + SSE - Establish safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.

The U.S. EPR faulted evaluation addresses both the horizontal (i.e., LOCA and seismic) and vertical (i.e., LOCA) effects. The horizontal faulted analysis models and methodology are consistent with that approved by the NRC in Reference 14. The axial faulted analysis methodology is consistent with that approved by the NRC in Reference 13.

The design bases used to establish the acceptance criteria for the U.S. EPR fuel assembly are provided in Reference 3 and are consistent with Reference 1 and follow the guidelines established by Reference 21.

5.3.4.1 Horizontal Analysis

The U.S. EPR fuel assembly component stresses were shown to be less than the allowable limits based on the criteria or testing in Reference 21.

The horizontal component of the faulted analysis determines the structural integrity of the U.S. EPR fuel assembly in the horizontal direction. The following loading conditions were evaluated:

- SSE
- LOCA
- Combined seismic and LOCA events

U.S. EPR fuel assembly models were benchmarked using properties established through testing. Fuel assembly models were combined to represent row configurations of fuel assemblies in the core. Row models with 7 to 17 assemblies were created.

The U.S. EPR fuel assemblies were analyzed under SSE and LOCA conditions. The CASAC program, used for design evaluation in Reference 14, was used to analyze the core response due to lateral loading. A single beam model was benchmarked against the results of mechanical tests performed with a mockup featuring a similar design. Bounding seismic SSE displacement time histories at the lower core plate, upper core plate, and upper end of the heavy reflector were evaluated. The LOCA lateral displacements evaluated corresponded to a worst-case attached pipe break based on leak-before-break. The SSE and LOCA time histories were applied to the reactor core model. The fuel assembly response was determined based on methodology described in Reference 14. The maximum grid impact forces and grid deformations were obtained for SSE and SSE + LOCA conditions for a full core configuration of U.S. EPR fuel assemblies.

The maximum impact load on the spacer grids under seismic SSE + LOCA conditions was determined to be **[ ]** for BOL hot conditions. An intermediate grid location on a peripheral fuel assembly within an intermediate length row configuration produced the maximum impact force for both SSE and LOCA. The loads for seismic and LOCA were combined, resulting in no grid deformation based on the BOL hot grid allowable load of **[ ]** per Table 5-8.

The spacer grid impact loads for the SSE conditions were within the allowable elastic load limits, resulting in no plastic deformation of the spacer grid geometry. Therefore, the core coolable geometry is maintained for the faulted loads.

Because the spacer grid loads are within the elastic limit for the SSE, these results also satisfy the OBE requirements that the assembly or components do not exceed their applicable yield limit (the magnitude of the OBE is half the magnitude of the SSE); therefore, a separate OBE analysis is not required.

5.3.4.2 Vertical Analysis

The U.S. EPR fuel assembly was evaluated for component loads in the axial direction resulting from LOCA and SSE conditions using the approved methodology in Reference 13. Finite element lumped mass models similar in approach to the model provided in Reference 13 but with applicable modifications to the elements to represent the U.S. EPR design was analyzed using the ANSYS general finite element code for LOCA events and the CASAC program for SSE. Fuel assembly axial stiffness properties and axial drop impact loads were obtained from testing and were used to benchmark the fuel assembly axial ANSYS model at room temperature. Additional modifications were made to the model to simulate conditions at operating conditions in compressed water for modeling the LOCA events. The relevant component properties that were developed in the ANSYS model for LOCA were then transferred to the CASAC model for SSE. The ANSYS model used an adjusted gravitational constant whereas the CASA model used fluid masses with appropriate coupling terms to account for the fuel assembly behavior in water.

The ANSYS LOCA model was subjected to example vertical core force time histories that correspond to bounding small pipe breaks based on leak-before-break methodology applied to the large breaks. The total LOCA flow surge force per fuel assembly was applied to the model elements representing grids, nozzles, guide tubes, and fuel rod element portions of the fuel assembly model in proportion to the pressure drop of each element as a percentage of total assembly pressure drop. Total and individual element pressure drops were derived from LYNXT distributions based on measured flow loss coefficients.

In determining the loads stemming from LOCA events BOL and EOL conditions were evaluated considering the changes in grid slip load and holddown due to irradiation effects including fuel assembly growth and holddown spring relaxation.

The peak internal forces for the fuel assembly due to the vertical component of a bounding SSE event were evaluated. The maximum loads were determined by a numerical simulation of the seismic event using a representative structural model of the U.S. EPR fuel assembly at BOL. The analysis was assessed using the soil case that generated the highest lateral loads (i.e., impact and reaction forces) on the fuel. A single fuel assembly was analyzed in the most penalizing configuration which is defined by the minimum holddown margin at BOL. The CASAC program was used to analyze the core response. A mass-spring model benchmarked against vertical stiffness and drop tests was used as the reference for the CASAC model.

With a vertical acceleration of the core plates featuring the same order of magnitude (0.3 g) as the horizontal acceleration at the ground level, liftoff is unavoidable. The internal loads reached during set-down features are acceptable levels that are comparable to the shipping and handling conditions **[ ]**.

All fuel assembly components including guide tubes, nozzles, and fuel rods were evaluated against the stress and deflection criteria of the ASME Code for the LOCA and LOCA + SSE loads derived from the analytical models.

5.3.4.3 Combined Horizontal and Vertical Faulted Analysis

The U.S. EPR fuel components were shown to satisfy all design criteria for faulted loading conditions. SSE and SSE + LOCA loading were used for the component analyses. The U.S. EPR fuel assembly component stress analyses for faulted conditions were performed using combined axial and lateral loads generated separately by seismic and LOCA loading analyses. The loads for the worst-case LOCA break were conservatively combined with those of the SSE to determine maximum fuel assembly loads. The component stress intensity limits for the components were based on the Level D service limit of Reference 21.

The design margins indicate that the major components of the U.S. EPR fuel assembly meet the design criteria for the SSE and SSE + LOCA loading events. The results of the faulted component analysis for the U.S. EPR are shown in Table 5-17.

5.4 U.S. EPR Fuel Assembly Thermal Hydraulic Evaluation

This section describes the thermal-hydraulic characteristics and analysis methods used to evaluate the U.S. EPR 17x17 HTP fuel assembly.

The basis for the thermal-hydraulic design of the fuel assembly is to enable the reactor to operate at rated power with sufficient margin to withstand operational and moderate frequency transients without sustaining damage to the core. The following design criteria have been established:

- For AOO events and normal operation, there is at least a 95 percent probability (at a 95 percent confidence level) that no fuel rod experiences departure from nucleate boiling.
- For AOO events and normal operation, there is at least a 95 percent probability (at a 95 percent confidence level) that no fuel melting occurs.
- The fuel assembly does not liftoff the lower core plate during normal operating conditions.

The following section describes the analyses and evaluations which are performed to assure that these criteria are satisfied by the U.S. EPR fuel assembly.

5.4.1 Core Pressure Drop

As described in Section 3.8.6, the pressure drop characteristics of the U.S. EPR 17x17 HTP fuel assembly were determined through a series of flow tests at the HERMES-P loop. The results of these tests were used as the basis for the calculation of form loss coefficients for the end fittings and spacer grids.

The pressure drop across a span is determined by the Darcy equation:

$$\Delta P = \left(K + f\,\frac{L}{D_h} \right) \frac{\rho V^2}{2}$$

Where:

ΔP	=	pressure drop (Pa)
$\rho V^2/2$	=	dynamic head (Pa)
ρ	=	fluid density (kg/m3)
V	=	fluid velocity (m/s)
L	=	length (m)
D_h	=	tube region hydraulic diameter (m)
K	=	form loss coefficient (dimensionless)
f	=	friction factor (dimensionless)

This equation can be rearranged to give the following expression:

$$K = \frac{\Delta P_L}{\frac{\rho V^2}{2}} - \left(f\,\frac{L}{D_e} \right)$$

The friction factor loss is determined using the LYNXT friction factor (Reference 18, pg. B-14) as the basis.

[ ]

Using the results from the HERMES-P tests, the form loss coefficients were then determined over a range of Reynolds numbers and adjusted to reflect the cold assembly tube region flow area. These results were then fit to the following relationship to determine the pressure loss coefficient, K:

$$K = A(\mathrm{Re})^b$$

Where A and b are constants determined by the HERMES-P tests.

5.4.2 Fuel Assembly Hydraulic Lift

The hydraulic lift force on an assembly is attributed to the unrecoverable pressure drop across the length of the assembly. Using the calculated form loss coefficients and the LYNXT code, along with appropriate assumptions on the core flow conditions and core configuration, several analyses were performed to evaluate the hydraulic lift forces on the U.S. EPR 17x17 HTP fuel assembly including:

- 4th Pump Startup
- Hot Full Power at Upper Temperature Band Limit
- Hot Full Power at Lower Temperature Band Limit
- Hot Zero Power at Upper Temperature Band Limit
- Hot Zero Power at Lower Temperature Band Limit
- 20% Pump Overspeed and Upper Temperature Limit and Full Power

The total lift resistance is calculated accounting for any additional ΔP in the lower portion of the core due to core bypass.

$$\Delta P_{lift} = \Delta P_{friction} + \Delta P_{form\_loss} + \Delta P_{bypass\_correction}$$

The spacer grid form losses represent approximately 70% of the overall lift resistance. Nozzle and rod friction losses account for the remaining resistance.

5.4.3 Core DNB Analyses

The purpose of the core DNB analyses is to ensure that there is at least a 95% probability, with 95% confidence that no fuel rod experiences a departure from nucleate boiling (DNB) event during normal operation or anticipated operational occurrences. The thermal-hydraulic analysis methodology employs the LYNXT sub-channel analysis code (Reference 18) to calculate departure from nucleate boiling ratios (DNBRs). The DNB criterion is met if the calculated DNBR is greater than the design limit DNBR, which is established independently for the critical heat flux (CHF) correlation being used. For the U.S. EPR 17x17 HTP fuel assembly, LYNXT is used in conjunction with NRC approved methodologies for evaluating DNB margin. The ACH-2 CHF (Reference 12) and BWU-N (Reference 11) correlations are used in the DNBR evaluation.

5.4.4 Fuel Rod Performance

The fuel performance is determined as a function of burnup using the COPERNIC computer code (Reference 17). The code includes analytical models for fuel densification and swelling, fuel restructuring, gas release, cladding creep, and gap closure. Using these models, the COPERNIC code conservatively calculates fuel pin temperature and pressure.

Figures 5-5 through 5-8 provide typical results from the thermal analysis of the U.S. EPR 17x17 HTP fuel rod. Figure 5-5 shows the power history assumed for the COPERNIC analysis, while Figures 5-6 and 5-7 provide fuel temperature and internal pressure results, respectively. The linear heat rate to prevent fuel centerline melt and or exceed 1 % clad strain as a function of rod average burnup is shown in Figure 5-8.

Figure 5-5 Average Heating Rate - UO_2, 18-month Equilibrium

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Figure 5-6 Fuel Average Temperature - UO_2, 18-month Equilibrium

Figure 5-7 Rod Internal Pressures (RIP) - UO_2, 18-month Equilibrium

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Figure 5-8 U.S. EPR Limiting Clad Strain (CS) & Centerline Fuel Melt (CFM) Curves

5.5 Design Evaluation Summary

The U.S. EPR fuel assembly meets all fuel assembly design criteria for safe and reliable operation. The U.S. EPR assembly incorporates reactor-proven design parameters that provide a basis for successful future performance. Design verification testing and analyses have demonstrated the acceptability of the design features and establish that the U.S. EPR fuel assembly will operate safely and reliably.

U.S. EPR fuel assembly and fuel rod mechanical and thermal-hydraulic performance capability is acceptable for fuel rod burnups up to 62 GWd/mtU.

Table 5-14 Summary of RCS Design Transients (Sheet 1 of 2)

Table 5-14 Summary of RCS Design Transients (Sheet 2 of 2)

Table 5-15 Summary of Required and Minimum Available Holddown Forces for Lower Tolerance Limit (LTL) Guide Tube Growth

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Table 5-16 Summary of Required and Minimum Available Holddown Forces for Best Estimate (BE) Guide Tube Growth

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Table 5-17 Summary of Faulted Analysis Safety Margins for Fuel Assembly Components

6.0 DESIGN CHANGE PROCESS

The generic design change process described in Reference 6, in conjunction with the specific design criteria in this topical report, will be used to justify fuel design changes for the U.S. EPR without requiring NRC review and approval when this topical report is referenced. The design change process may not be used to alter the design basis or configuration for the initial cycle of operation for the U.S. EPR fuel assembly. Reference 6 provides generic design criteria and a process to demonstrate their application. In summary, compliance to these criteria can be demonstrated by:

- Documenting the fuel system and fuel assembly design drawings
- Performing analyses with NRC-approved models and methods
- Confirming the adequacy of significant new design features using prototype tests or lead test assemblies prior to full reload implementation
- Continuing irradiation surveillance programs, including past irradiation examinations, to confirm fuel assembly performance
- Using the Quality Assurance procedures, Quality Control inspection program, and design control requirements set forth in the NRC approved AREVA NP quality assurance program, Reference 28.

6.1 Criteria to be Used When Making Design Changes

Small design changes in a fuel assembly are any changes that meet all of the following criteria:

- The change does not apply to the initial cycle design configuration of the fuel
- The change does not result in an un-reviewed safety question

- Changes in plant technical specifications are not required
- The applicability of NRC-approved methodologies is demonstrated to be valid
- Burnup limits are within those approved by the NRC.

Changes shall be developed within the conditions of the NRC-approved methods. If a change in methodology is made that meets any of the following criteria, the modified methodology will be submitted to the NRC for review and approval.

- An existing approved design code or method is replaced
- A new core power distribution monitoring method is implemented
- A method is applied beyond its approved limits

6.2 Types of Design Changes

Examples of design changes of which Reference 6 can be used were presented in a letter of clarification to the NRC (Reference 23), and the NRC concurrence is documented in Reference 24. The examples provided in Reference 23 are summarized as follows.

- A change in the attachment of the spacer to the guide tubes
- A change in the strip thickness of the spacer
- A change in cladding thickness
- The first use of an assembly design feature previously irradiated in conjunction with one lattice (i.e., 14x14) in a different lattice (i.e., 17x17)
- A change in enrichment
- A change in gadolinia-bearing rod locations

Additional examples are provided in Reference 23 which, while not requiring a submittal to the NRC for the design change, would require an NRC submittal to gain approval for a new or revised model, such as:

- New cladding material
- A spacer with a new functional mixing behavior or new rod support mechanism
- A change that would alter the fuel behavior relative to NRC-approved models (e.g., rod growth, assembly growth, or clad corrosion).

6.3 Summary

In summary, the generic design change process described in Reference 6, in conjunction with the specific design criteria specified in this topical report, will be used to justify fuel design changes after the initial cycle for the U.S. EPR fuel assembly without requiring NRC review and approval when this topical report is referenced.

7.0 REFERENCES

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