June 25, 2004

Mr. L. M. Stinson
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 — REQUEST FOR ADDITIONAL INFORMATION RE: SERVICE WATER INTAKE STRUCTURE EXEMPTION FROM FIRE PROTECTION REQUIREMENTS (TAC NOS. MC0627 AND MC0628)

Dear Mr. Stinson:

By letter dated August 28, 2003, Southern Nuclear Operating Company requested a revision to an existing exemption concerning the Service water Intake Structure for Farley Nuclear Plant (FNP), Units 1 and 2. Specifically, to eliminate the FNP reliance on Kaowool, the proposal would allow an alternate compliance strategy based on a combination of changes to the current FNP fire protection program and an application of the risk-informed, performance-based methods of NFPA 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The U.S. Nuclear Regulatory Commission staff has reviewed the application and has determined that additional information is required, as identified in the Enclosure.

We discussed these issues with your staff on May 5, 2004. Your staff indicated that you would attempt to provide your response within 90 days of the date you receive the questions.

Please contact me at (301) 415-1842, if you have any other questions on these issues.

Sincerely,

/RA/
Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: Request for Additional Information

cc w/encl: See next page

June 25, 2004

Mr. L. M. Stinson
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 — REQUEST FOR ADDITIONAL INFORMATION RE: SERVICE WATER INTAKE STRUCTURE EXEMPTION FROM FIRE PROTECTION REQUIREMENTS (TAC NOS. MC0627 AND MC0628)

Dear Mr. Stinson:

By letter dated August 28, 2003, Southern Nuclear Operating Company requested a revision to an existing exemption concerning the Service water Intake Structure for Farley Nuclear Plant (FNP), Units 1 and 2. Specifically, to eliminate the FNP reliance on Kaowool, the proposal would allow an alternate compliance strategy based on a combination of changes to the current FNP fire protection program and an application of the risk-informed, performance-based methods of NFPA 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants." The U.S. Nuclear Regulatory Commission staff has reviewed the application and has determined that additional information is required, as identified in the Enclosure.

We discussed these issues with your staff on May 5, 2004. Your staff indicated that you would attempt to provide your response within 90 days of the date you receive the questions.

Please contact me at (301) 415-1842, if you have any other questions on these issues.

Sincerely,

/RA/
Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: Request for Additional Information

cc w/encl: See next page

DISTRIBUTION:

PUBLIC	RidsAcrsAcnwMail Center
PDII-1 R/F	RidsRgn2MailCenter
AKlein	RidsNrrDlpmLpdii
DFrumkin	RidsNrrPMSPeters
JHannon	RidsNrrLACHawes
RGallucci	RidsOgcRp
NIqbal	BBonser, RI
PLain	

ADAMS Accession No. ML NRR-088

OFFICE	PDII-1/PM	PDII-1/SC(A)
NAME	SPeters	SCoffin
DATE	06/21/04	06/24/04

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

The U. S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittal dated August 28, 2003, regarding a proposed revision to an existing exemption concerning the Service Water Intake Structure (SWIS). The NRC staff has identified the following information that is needed to enable the continuation of its review.

1. On Page B-7 of your submittal you state:

> "In addition, SNC determined that certain cables for MOVs that align the swing Service Water pump discharge valves no longer require reliance on Kaowool. Service water alignment procedures remove power from these valves, thus precluding spurious valve operation."

Is power removed from the MOVs prior to fire initiation, i.e., whenever Service Water is realigned, the MOVs are powered for the realignment and then, once realigned, have the power removed? How often are the pumps realigned and what is the time period during the realignment when power is provided to the MOVs? What is the probability (time-integrated frequency) of fire covering these time periods?

2. On Page B-11 of your submittal you state:

> "Target damage was assumed to occur if the target surface temperature reached 700°F ... The threshold damage condition for a target was a surface temperature of 700°F."

On Pages B-12 and B-13 of your submittal you state:

> "A Unit 1 Service Water pump scenario was assumed to involve 22.5 gallons of Texaco Regal 68 lubricant, a Class IIIB combustible liquid ... The maximum number of pumps that could thus be damaged by a single fire is three (two by flame impingement and one by thermal radiation or one by flame impingement and two by thermal radiation). The temperature of any pump target or cable tray target located beyond this range would be less than 610°F."

Appendix F - Fire Protection Significance Determination Process (SDP), (Draft, February 2004), states the following in Section 6.2.3.3, "Task 2.3.3: Identify Nearest Ignition and Damage Targets:"

> *It is worth noting that in the IPEEE, a commonly applied screening failure threshold for IEEE-383 qualified cables applied by licensees was 370 °C (700 °F) ... The 700 °F value is recommended in the EPRI FIVE method (EPRI TR-100370), and appears again in the EPRI Fire PRA Implementation Guide (EPRI TR-105928). The original source cited for this value is the EPRI cable damage tests reported in a series of Factory Mutual*

Research Corporation (FMRC) studies from the early 1980s (see in particular EPRI NP-1767, March 1981). The method used to estimate the cable "critical" threshold values cited in the original FMRC work, and repeated in FIVE, has since been discredited, and has been disavowed by FMRC (see letter, A. Tewarson of FMRC, to R. Kasawara of EPRI, 5/10/95). There appears little basis for the continued reliance on 700 °F as a screening threshold for thermoset/qualified cables given the direct evidence of failures at substantially lower temperatures for a broad and common class of thermoset/qualified cable products. Recommended SDP Practice: A failure threshold of 330 °F (625 °F) is recommended for the generic class of thermoset cables.

What would be the impact when reassessing the potential damage to the pumps beyond the three-pump range mentioned above in light of the recommended reduced damage threshold of 625°F? In other words, what is the nature of the 610°F estimate? Is it an upper bound or mean value? If an upper bound, what is the confidence level and how robust is it? Sensitivity analyses may be appropriate to demonstrate robustness.

3. On Page B-14 of your submittal you state:

> "The fire risk analysis focused only on elements of the SWIS that had been or were proposed to be changed from the current licensing basis. These elements were associated with pump/motor lubricant fires (one for each pump or ten cases in all). The risk analysis determined that a conservative estimate of the CDF associated with the ten cases would be approximately 6.5E-07/yr per unit."

Presumably, each scenario assumes one of the following per unit:

(1) Fire in pump A or B, damaging the other pump and nearby pump C. Pumps D and E fail from non-fire causes (2 scenarios per unit);
(2) Fire in pump C, damaging pumps B and D. Pumps A and E fail from non-fire causes (1 scenario per unit);
(3) Fire in pump D or E, damaging the other pump and nearby pump C. Pumps A and B fail from non-fire causes (2 scenarios per unit).

The three pumps failed by the fire are assigned the common-cause failure frequency of that of the fire. The remaining two pumps are then assumed to fail randomly from non-fire causes. For these two pumps, was the possibility of both being out-of-service for maintenance, etc., considered, assuming this is not precluded by Technical Specifications? Was common-cause failure of these two pumps addressed, assuming other than fire-induced mechanisms (e.g., silting)? If so, how was common-cause failure modeled (e.g., MGL, alpha-factor, etc.)? Was the CDF for each scenario the same (i.e., 6.5E-07/yr * 1/5 = 1.3E-07/yr), or were they different? If different, what accounted for the asymmetry?

4. On Page B-14 of your submittal you state:

> "The scope of the analyses that were performed for this analysis included a re-analysis of the Service Water system performance. This re-analysis concluded that a single Service Water pump per unit was sufficient to satisfy the system performance requirements. The re-analysis results were integrated into the PRA Model by altering the number of Service

Water pumps per train that was required for system success from two to one."

On Page B-19 of your submittal you state:

> "An integrated risk assessment shows that safe shutdown can be achieved even if no credit is taken for the Kaowool raceway enclosures. This finding is also based, in part, on a re-determination of the safe shutdown success criterion, using traditional thermal-hydraulic techniques. The exemption should be revised not only to eliminate reliance on Kaowool but also to recognize the new success criterion."

Did this re-analysis of Service Water system performance address the Service Water requirements during a core damage scenario, or did it consider only reaching and maintaining hot shutdown for a limited period? Were there any differences between units? If so, what accounted for the asymmetry? What constituted the "traditional thermal-hydraulic techniques" (e.g., MAAP) that were used to verify the less limiting performance requirement during a core damage scenario? What was the sensitivity of the PRA results to the modeling assumption of one vs. two required Service Water pumps?

5. On Page B-15 of your submittal you state:

> "The plant PRA model was modified to take advantage of recent vendor data related to RCP seal performance. The specific data is related to seal performance given loss of motor bearing cooling."

Did this modification utilize the WOG2000 RCP seal leakage model? If so, what version was used? Whatever model was used, how did the results compare with other RCP seal leakage models, such as Rhodes? What was the sensitivity of the PRA results, to the model change? How much of this sensitivity was reflected in the PRA performed for this exemption request?

6. On Page B-15 of your submittal you state:

> "The LERF associated with the proposed change is negligible given the acceptance criteria of Regulatory Guide 1.174."[a]

The Large Early Release Frequency (LERF) would be bounded by the Core Damage Frequency (CDF), indicating a maximum possible LERF of 6.5E-07/yr per unit. Assuming a typical probability of no more than 0.1 for a large early release given core damage, this would suggest a LERF of ~6.5E-08/yr per unit. This would fall in Region III of the LERF acceptance guidelines in Regulatory Guide 1.174. This would be in the same Region as the CDF calculation, therefore some additional detail that builds on the CDF result would be expected in the exemption request regarding the LERF. Was the

---

[a] As with CDF, it is assumed that "LERF" refers to the more conservative surrogate for delta-LERF.

LERF truly "negligible" (e.g., <1E-8/yr per unit), or was it also a Region III result like the CDF?

7. On Page B-18 of your submittal you state:

> "Nine PRA elements were judged by the peer review to have findings that resulted in their being considered 'Contingency Grade 3.' A 'Contingency Grade 3' reverts to a 'Grade 3' when items noted in the evaluation of the element are resolved. Such pending items are classified as one of four degrees of significance. None of the pending items noted in the Plant Farley PRA evaluation were judged to be of a level of significance to require prompt resolution to ensure the technical adequacy of the PRA for this specific application."

Were there any Contingent Grade 3 sub-elements or elements that could impact the PRA for this exemption request still pending resolution at the time of this analysis? If so, were any at an "A" or "B" level of significance? If so, please elaborate.

8. On Page B-15 of your submittal you state:

> "The results of the risk assessment show a conservatively estimated risk increase... The Farley Nuclear Plant does not have an updated fire risk assessment. The available analysis is the Fire IPEEE... Although a comprehensive update and upgrade of the plant PRA has not been performed, these estimates are sufficient to conclude that the proposed change is within the Region III limits."

Typically, an internal events PRA does not include fire-related failures of components already modeled for the internal events analysis or fire-related failures of components that would be credited in an internal fire PRA. The effect of the first exclusion can be non-conservative, in that additional failure modes are excluded, while the exclusion of the latter can be conservative because additional credit for means of avoiding core damage, identified only through fire safe-shutdown analyses, are not modeled. Ideally, the overall effect is that fire CDF estimates developed from an internal events PRA will be conservative relative to the fire CDF that would be estimated from a combined internal events and fire PRA. Please describe, at a high-level, the technical approach by which the Farley fire CDF analysis ensured the results were conservative. For example, a fire in an area was assumed to fail/spuriously actuate all susceptible components (e.g., either via presence of the components themselves or their power/control cables) as modeled in the internal events PRA, with the appropriate frequency of a non-suppressed fire in that area superimposed on the corresponding initiating events (e.g., loss of offsite power) that would result from the fire.

Fire Protection Related RAIs

9. Fire protection for nuclear power plants uses the concept of defense in depth to achieve the required degree of reactor safety by using echelons of administrative controls, fire protection systems and features, and safe shutdown capability. These defense-in-depth principles are aimed at achieving the following objectives.

- To prevent fires from starting,
- To detect rapidly, control, and extinguish promptly those fires that do occur, and
- To provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

On Page B-3 of your submittal you provide a general overview of the Service Water Intake Structure (SWIS) and the fire suppression and detection systems provided.

Based on the description in your letter dated August 28, 2003 and our previous Safety Evaluation (Enclosure 2 to letter dated December 29, 1986 from NRC to Alabama Power Company), it is our understanding that the SWIS Fire Area 72 has the following fire protection systems: (Please confirm our understanding of the fire protection systems provided in Fire Area 72 and provide any necessary clarifications.)

1. Smoke detection throughout Fire Area 72.

Clarify and confirm that all portions of Fire Area 72 in the SWIS are protected by smoke detectors.

2. Preaction ‘spray’ system protecting the service water pumps.

Is this a separate preaction system from the area wide preaction system? If so, please provide a summary description.

3. Area wide preaction sprinkler system protecting the area in the strainer pit beneath the pump deck.

Describe the extent of coverage for this system. Does this system extend underneath the pump deck’s approximate 12' overhang above the strainer pit?

4. Preaction sprinkler system protecting the safe-shutdown cabling in the upper northeast corner of the service water pump room.

Is this a separate preaction system from the area wide preaction system? If so, please provide a summary description.

5. Portable fire extinguishers located throughout Fire Area 72.

Please confirm that portable fire extinguishers are provided in Fire Area 72.

6. Two fire hose/hydrant houses located directly outside of the SWIS within the security fence.

Please confirm that the hose houses contain adequate lengths of hose such that all areas of Fire Area 72 can be reached with an effective hose stream.

10. Fire protection for nuclear power plants uses the concept of defense in depth to achieve the required degree of reactor safety by using echelons of administrative controls, fire protection systems and features, and safe shutdown capability. The first objective in the concept of defense-in-depth for fire protection is:

"To prevent fires from starting"

Several areas of your submittal discuss transient combustibles; and some of the fire modeling accounts for specific quantities and types of in-situ combustibles; and other areas of the submittal discuss the need for administrative controls to limit the amount of lubrication oil during maintenance activities in the SWIS. Given the very specific transient and in-situ combustibles discussed, please describe the combustible and ignition controls that will be put in place to ensure that the first objective of defense-in-depth is maintained; and to ensure that the described margins between a Maximum Expected Fire Scenario (MEFS) and a Limiting Fire Scenario (LFS) are maintained (see pages B-12 through B-14); and to ensure that any combustibles due to maintenance activities (e.g., pump oil change-out that may cause double the amount of assumed oil to be in the area) are adequately controlled.

Describe any designated storage areas in the SWIS and the amounts and types of combustibles in those storage areas. If there are any designated storage areas in the SWIS containing combustibles, describe how the fire modeling considered these combustibles (as an intervening combustible and/or as a primary combustible). The staff is concerned with 'non-target' and secondary combustibles that may propagate fire from the initiating source to the target cables.

11. Page B-9, Paragraph under, "Initial Assessment," states, "Three fire scenarios required a more detailed evaluation in order to make an adequate preliminary assessment of fire risk in the SWIS. Other fire scenarios were determined to not contribute to the change in risk being assessed."

Provide a summary description of the 'other fire scenarios' and the bases for the determination that these other fire scenarios do not contribute to the change in risk.

12. Page B-9, Scenario 1 states that "A transient fire was considered to be the bounding fire in this area because of a lack of *in-situ* ignition sources and combustible material." Provide a summary basis for this statement considering that the cable jacketing/insulation itself is combustible.

13. Page B-1 states that the SWIS is located about half a mile from the nuclear power block and its support buildings. Describe operator and fire brigade response to a fire alarm or to a fire suppression system actuation alarm in the SWIS. Specifically, describe operator response upon receipt of a fire alarm signal; how manual fire fighting response is planned considering the distance from the main power block; how manual fire fighting protective gear (bunker gear, SCBA, extra air supply, etc.) is ensured at the scene and how fire brigade members are transported to the SWIS.

14. Two out of the three fire scenarios discussed in the submittal involve oil fires. Curbs are described in the submittal to contain oil spills (therefore, curbed volumes are limited). Describe how the fire brigade plans to fight a lube oil fire in the SWIS. Considering there is an automatic water based suppression system in the SWIS and the fire brigade has hose houses available, describe how the potential of oil spreading over the curbs due to sprinkler system or fire fighting water has been considered. Does the fire brigade have a readily available supply of foam and corresponding fire fighting equipment?

Fire Modeling Related RAIs

Attachment B of the August 28, 2003, letter discussed fire modeling to provide insight into the potential for a particular fire scenario to damage a target. With regard to the fire modeling analysis provide the following:

15. From Attachment B of the exemption request, it appears that the CFAST and HEATING computer fire models were used to determine that in the event of a fire in Fire Area 72, cables, cable trays and motor terminal boxes for service water pumps would not be damaged. The staff requests the detailed analysis, including assumptions and results of the fire modeling for further staff review. In addition, provide the input files used in the CFAST and HEATING fire simulation for all fire scenarios for Fire Area 72.

16. The page B-8 and B-9, paragraph under, “Initial Assessment,” states “The critical failure mode was found to be damage to the power and control cables, either in the cable trays or in the motor terminal boxes.”

    For each fire scenario, provide a description of the target parameters (e.g., thermoplastic or thermoset cables including wiring that is part of the motor terminal boxes), target damage threshold values, and bases for the damage threshold values. For the fire damage threshold values related to cables, what is the impact when reassessing the scenarios based on a damage threshold of 400°F for thermoplastic cables or 625°F for thermoset cables, as applicable. (See RAI #2 above for reference.) Provide cable construction information (i.e., insulation and jacket material, such as XLPE/PVC) for cables installed in cable trays, motor terminal boxes or exposed (such as air drops) in Fire Area 72, including vendor and/or manufacturer.

    The fifth bullet on page B-9 states that “...all cables in the SWIS are IEEE-383.” Please clarify that this statement is applicable to all cables (e.g., power, control, signal) of interest in the SWIS including any cable that could act as an intervening combustible, and cable/wiring that are part of the motor terminal boxes.

17. Where applicable in Fire Area 72, discuss how thermoset and thermoplastic cables located in the same electrical raceway were addressed with respect to fire propagation and fire damage to target cables.

18. The page B-12, “Fire Modeling Input Assumptions,” eighth bullet states that doors and openings were assumed shut. Provide a summary basis for this assumption and demonstrate that it bounds the open door assumption (i.e., open door when fire brigade or operators enter the SWIS or its zones). Describe how mechanical ventilation systems were addressed in the fire modeling and provide any ventilation flow rates provided by mechanical equipment.

19. The page B-11, “Fire Modeling Input Assumptions,” ninth bullet states that the transient combustibles consisted of an equal mix of cellulose based and plastic based material. Provide the basis for why flammable and/or combustible liquids or flammable gases were not assumed in light of the fact that at power maintenance activities have occurred in the SWIS (see RAI #16 above). Provide the basis for the transient fire 332 Btu/s heat release rate for 600 seconds as done in scenario 1.

20. The page B-12 and B-13 “Fire Modeling Results” for Unit 1 and Unit 2 Service Water pump fire scenario discusses service water pump oil fires and loss of pumps by a single pump fire. Page B-13 states that, “CFAST was used to determine the smoke temperature in the SWIS and HEATING was used to calculate the surface temperature of the Service Water pump motor junction box targets and the east wall cable tray target.” Describe the results for a pump fire with loss of pumps <u>and</u> effects of fire to nearby ‘A’ or ‘B’ train cables as applicable (e.g., Unit 2, Train A Service Water pump fire and effects to adjacent service water pumps and to nearby Train B cable trays). Include in the description any effects from intervening combustibles, ceiling height, fire plume, hot gas layer, radiant heat flux from flames and from hot gas layer, and beam pockets.

21. The page B-13, scenario 3 assumes a fire involving 8 gallons of lube oil versus scenario 2 which assumes 22.5 gallons of lube oil. Provide the basis for the lower value of lube oil in scenario 3.

22. Drawing D-171331 shows beam pockets formed by concrete beams. Describe how the fire modeling accounted for these pockets (i.e., potential hot gas layer forming initially in a single beam pocket, etc.), where applicable.

<u>Additional RAIs</u>

23. Regulatory Guide 1.174, Section 2 discusses five key principles that a change is expected to meet. Describe how each key principle is met with the proposed change.

24. Pages B-16 and B-17 discuss Defense-in-Depth/Safety Margins. There are seven bullets listed on Page B-16 that are taken from Regulatory Guide 1.174 to determine if consistency with defense-in-depth is maintained. The bottom half of Page B-16 provides a very brief discussion on each of the seven bullets. However, the discussion for each item is very brief or does not provide any basis for a reviewer to assess whether each item has been fulfilled. For each item, provide a qualitative or traditional

engineering argument or by using PRA results contained in the accident sequences or cutsets, as appropriate.

25. Pages B-16 and B-17 discuss Defense-in-Depth/Safety Margins. There are two bullets listed on Page B-17 that are taken from Regulatory Guide 1.174 to determine if sufficient safety margins are maintained. Regulatory Guide 1.174, section 2.2.1.2 provides a set of acceptable guidelines to ensure sufficient safety margins:

> "Codes and standards or their alternatives approved for use by the NRC are met.
>
> Safety analysis acceptance criteria in the LB (e.g., FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty."

The second bullet on page B-17 addresses the safety margins guidance for the proposed change or revision by accounting for analysis and data uncertainty. The licensee compares MEFS and LFS to account for analysis and data uncertainty on Pages B-12 through B-14.

Scenario 1 appears to report the safety margin in terms of heat release rate (HRR) (and time) and mass of combustibles. Scenarios 2 and 3 appear to report safety margin in terms of mass (or volume) of combustibles only. Please clarify why it is appropriate to consider only the mass (or volume) of combustibles for safety margins in Scenarios 2 and 3 where the lube oil is pooled within a bermed or confined area (include the basis for why HRR was not considered).

For each of the three scenarios, discuss acceptability of the safety margins and their values, taking into account uncertainties with fire modeling and with fire input data (parameters). Provide a discussion on the acceptability of the margins between the MEFS and LFS considering the potential effect and severity on the plant if a LFS occurred.

26. The page B-18, Conclusions 2 and 3 are based on a deterministic re-analysis that demonstrates fire damage cannot result in spurious operation of the strainer pit valves (service water header strainer inlet MOVs and swing pump discharge MOVs). (See Page B-7 also). Provide the assumptions and the basis for this conclusion.

Joseph M. Farley Nuclear Plant

cc:

Mr. Don E. Grissette
General Manager -
Southern Nuclear Operating Company
Post Office Box 470
Ashford, Alabama 36312

Mr. B. D. McKinney, Licensing Manager
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201-1295

Mr. M. Stanford Blanton
Balch and Bingham Law Firm
Post Office Box 306
1710 Sixth Avenue North
Birmingham, Alabama 35201

Mr. J. B. Beasley, Jr.
Executive Vice President
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, Alabama 36130-1701

Chairman
Houston County Commission
Post Office Box 6406
Dothan, Alabama 36302

Resident Inspector
U.S. Nuclear Regulatory Commission
7388 N. State Highway 95
Columbia, Alabama 36319

William D. Oldfield
SAER Supervisor
Southern Nuclear Operating Company
P. O. Box 470
Ashford, Alabama 36312